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

March 6, 2008

The Honorable Dale E. Klein
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Klein:

SUBJECT: SUMMARY REPORT – 549th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, FEBRUARY 7-9, 2008, AND OTHER RELATED ACTIVITIES OF THE COMMITTEE

During its 549th meeting, February 7-9, 2008, the Advisory Committee on Reactor Safeguards (ACRS) discussed several matters and completed the following reports and letters.

REPORTS

Reports to Dale E. Klein, Chairman, NRC, from William J. Shack, Chairman, ACRS:

- Review and Evaluation of the NRC Safety Research Program, dated March 6, 2008.
- State-of-the-Art Reactor Consequence Analyses (SOARCA) Project, dated February 25, 2008.

LETTERS

Letter to David J. O'Brien, Commissioner, Department of Public Service, State of Vermont, from William J. Shack, Chairman, ACRS:

- Final ACRS Review of the Vermont Yankee License Renewal Application, dated February 19, 2008.

Letters to Luis A. Reyes, Executive Director for Operations, NRC, from William J. Shack, Chairman, ACRS:

- Draft Final Revision 1 to Regulatory Guide 1.45, "Guidance on Monitoring and Responding to Reactor Coolant System Leakage," dated February 22, 2008.
- Cable Response To Live Fire (CAROLFIRE) Testing and Fire Model Improvement Program, dated February 28, 2008.

HIGHLIGHTS OF KEY ISSUES

1. License Renewal Application for the Vermont Yankee Nuclear Power Station

The Committee met with the representatives of Entergy Nuclear Operations, Inc., (the applicant) and the NRC staff to discuss the license renewal application for the Vermont Yankee Nuclear Power Station (VYNPS) and the associated Safety Evaluation Report (SER). The operating license for VYNPS expires on March 21, 2012. The applicant has requested approval for continued operation for a period of 20 years beyond the current license expiration date.

In the SER, with the exception of an issue related to environmentally assisted fatigue (EAF) of reactor coolant pressure boundary components, the staff documented its review of the license renewal application and other information submitted by Entergy and obtained during the audits and inspections conducted at the plant site. The staff reviewed: the completeness of the applicant's identification of structures, systems, and components that are within the scope of license renewal; the integrated plant assessment process; the applicant's identification of the plausible aging mechanisms associated with passive, long-lived components; the adequacy of the applicant's Aging Management Programs; and the identification and assessment of time-limited aging analyses requiring review.

For the remaining EAF issue, the applicant has submitted additional confirmatory analysis information that is currently being reviewed by the staff. The staff currently plans to complete the final SER, including resolution of the EAF issue, such that the ACRS will be able to complete its review of the VYNPS license renewal application at its March 2008 meeting.

Committee Action

The Committee plans to continue its discussion of the VYNPS License Renewal Application and the associated final SER, especially the resolution of the EAF issue, during its March 2008 meeting.

2. Draft Final Revision 1 to Regulatory Guide 1.45, "Guidance on Monitoring and Responding to Reactor Coolant System Leakage"

The Committee met with representatives of the NRC staff regarding the proposed Revision 1 to Regulatory Guide 1.45. Regulatory Guide 1.45 was first issued in 1973 to provide guidance on leak detection in containment. It recommended that three separate methods of measurement be employed to detect leaks of one gallon per minute or less from unidentified sources. Following the Davis-Besse reactor vessel head event, one of the areas identified for examination was the need for additional guidance in the area of leak detection from the reactor coolant system. An examination of operating experience showed that over half of reported leaks were too small to be detected by measurement methods and were found by visual inspection. Large leaks were detected by the installed measurement systems. The Revised Regulatory Guide recommends the use of local detection methods in potentially critical areas such as those where small leaks could expose low-alloy steel to borated water. Regulatory Guide 1.45, Revision 1 also recommends inclusion of monitoring and trending procedures in the plant technical specifications. Regulatory Guide 1.45, Revision 1 will be applied only to new reactors.

Committee Action

The Committee issued a letter to the Executive Director for Operations on this matter, dated February 22, 2008, recommending that Regulatory Guide 1.45, Revision 1 be issued.

3. Proposed Licensing Strategy for the Next Generation Nuclear Plant (NGNP)

The Committee met with the representatives of the Department of Energy (DOE) and the NRC staff to discuss the development of the draft licensing strategy report prepared by a DOE and NRC joint working group in response to the Energy Policy Act of 2005 (EPAct). The EPAct directed DOE and the NRC to describe the ways in which the current light water reactor licensing requirements could be adapted for the prototype NGNP, the analytical tools that would be needed by the NRC to independently verify the NGNP safety performance, research and development (R&D) activities the NRC will need to conduct to review the NGNP license application, and a budget estimate associated with the licensing strategy. The licensing strategy development report needs to be submitted to Congress by August 7, 2008. The EPAct also mandated that the NGNP provide process heat for hydrogen generation.

The DOE and NRC staff had undertaken jointly a "phenomena identification and ranking table (PIRT) process" to assess the knowledge base for key phenomena, the adequacy and developmental needs for the analytical tools, and the R&D needs. The DOE staff described the technical challenges and experience associated with the high-temperature gas-cooled reactor technology and the associated use of process heat for hydrogen generation. DOE representatives also described the operating conditions for a pre-conceptual design, the needed technology development areas, ongoing and future test programs, and R&D needs. The NRC staff discussed the options for the licensing approach, highlights of the PIRT findings, needs for tools and data to perform confirmatory safety analyses, and other infrastructure needs.

The ACRS members discussed their comments and questions with the staff. The interface between the NGNP reactor and the hydrogen generation plant was one area of ACRS interest.

Committee Action

The Committee plans to continue its discussion of the NGNP issues during its April 2008 meeting.

4. Cable Response to Live Fire (CAROLFIRE) Testing and Fire Model Improvement Program

The Committee met with representatives of the NRC staff, Sandia National Laboratories, and the National Institute of Standards and Technology (NIST) to discuss results of the Cable Response to Live Fire (CAROLFIRE) Testing and Fire Model Improvement Program. This Program was based on Regulatory Issue Summary (RIS) 2004-03 Rev. 1, which had explicitly described a set of cable/circuit configurations in need of more research to determine failure characteristics.

The purpose of the CAROLFIRE Project was to experimentally investigate the various failure modes of electrical cables when exposed to fires, in configurations described in the RIS as needing more research. During the meeting, NRC and NIST staff representatives described a series of experiments in which cables were subjected to a fire environment in both a small-scale, highly controlled facility, and in a larger, more realistic room-sized facility, while observing the times and various modes of failure. A calculational model for estimating the internal temperature of a cable as a function of time had also been developed and compared to the data. The results of the program will be published in a NUREG/CR report. The Members provided some suggestions for improving the presentation of the results, with the aim of making these results more useful to the users.

Committee Action

The Committee issued a letter to the Executive Director for Operations on this matter, dated February 28, 2008, recommending that NUREG/CR-6931, "Cable Response to Live Fire (CAROLFIRE)," including the electronic data sets, be published. The Committee also recommended that the staff continue to analyze the CAROLFIRE data and develop additional guidance regarding the use of the results.

5. Boiling Water Reactor Owners Group's (BWROG) Proposed Containment Overpressure Credit Methodology

The Committee was briefed by representatives of the NRC staff and the Boiling Water Reactor Owners Group (BWROG) regarding a proposed containment overpressure methodology which is documented in the Topical Report, NEDO-3337P, "Containment Overpressure Credit for Net Positive Suction Head (NPSH)," Revision 0. This methodology was developed to address some of the comments made by the ACRS during its review of the extended power uprate (EPU) applications. The Committee commented on the acceptability of relying on containment overpressure credit in meeting the required NPSH and the increases in both the credit and the duration needed for EPU operation. The Committee also commented on the lack of consistency in the licensees' approaches in determining the containment overpressure credit, pointing out the need for a well-defined risk assessment for some of the event scenarios.

The BWROG briefed the Committee on the proposed guidance process and the newly developed statistical methodology for calculating the containment response and the overpressure credit needed. This methodology will reduce some of the conservatisms currently employed in the deterministic containment analyses methodology.

The NRC staff presented the regulatory history and positions on crediting containment overpressure in meeting the required NPSH. In addition, the NRC staff discussed its positions for accepting containment overpressure credit. The staff stated that if there is no practical alternative, containment overpressure credit is accepted, provided that the containment overpressure is calculated in a conservative manner that minimizes the available containment pressure response.

The ACRS members provided feedback on issues that may need to be addressed in more detail before the approval of the proposed methodology. The members commented that the Topical Report should address in more detail the sampling and the uncertainty distribution method, including the manner in which interdependent and correlated variables are defined. Members also commented that in developing the variations on key parameters, the operator actions should also be factored in. The containment response calculations should also account for the accuracy of the code models in addition to the uncertainty range of the key input parameters.

Committee Action

This was an information briefing. No Committee action was necessary. The Committee plans to review the staff's evaluation of the proposed methodology described in Topical Report, NEDO-33347P, "Containment Overpressure Credit for Net Positive Suction Head (NPSH)," Revision 0.

6. 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 February 2008 meeting, the Committee completed its biennial review and evaluation of the Reactor Safety Research Program sponsored by the NRC.

Committee Action

The Committee issued a report to the Commission, dated March 5, 2008, transmitting an advance copy of its 2008 biennial report on, "Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program." The final report will be published as NUREG-1635, Vol. 8.

RECONCILIATION OF ACRS COMMENTS AND RECOMMENDATIONS/EDO COMMITMENTS

- The Committee considered the EDO's response of February 1, 2008, to comments and recommendations included in the November 20, 2007, ACRS letter concerning Chapters 2, 5, 8, 11, 12, and 17 of the NRC staff's SER with Open Items related to the certification of the ESBWR [Economic Simplified Boiling Water Reactor] design. The Committee decided that it was satisfied with the EDO's response. **The EDO stated that the staff has sent a request for additional information to General Electric-Hitachi Nuclear Energy (GEH) to obtain the necessary information for developing the source term of radioactive materials released into the reactor coolant system. The EDO committed to provide this information to ACRS.**

- The Committee considered the EDO's response of December 6, 2007, to comments and recommendations in the October 19, 2007, ACRS letter concerning the draft final Generic Letter 2007-02, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems." The Committee decided that it was satisfied with the EDO's response. **The EDO indicated that the staff will provide the ACRS an opportunity to review proposed interim measures or topical reports developed as a result of this Generic Letter.**

- The Committee considered the EDO's response of January 30, 2008, to comments and recommendations included in the December 20, 2007, ACRS letter concerning Draft Final NUREG-1829, "Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process," and Draft NUREG-XXXX, "Seismic Considerations for the Transition Break Size." The Committee decided it was satisfied with the EDO's response.

- The Committee considered the EDO's response of January 30, 2008, to comments and recommendations included in the December 27, 2007, ACRS letter concerning the AREVA Detect and Suppress Stability Solution and Methodology. The Committee decided that it was satisfied with the EDO's response.

- The Committee considered the EDO's response of December 27, 2007, to comments and recommendations included in the November 19, 2007, ACRS letter on the staff's implementation of Lessons Learned from Reviews of Early Site Permit (ESP) Applications. The Committee decided that it was satisfied with the EDO's response.

- The Committee considered the EDO's response of December 28, 2007, to comments and recommendations included in the November 20, 2007, ACRS letter on the Southern Nuclear Operating Company (SNC) Application for the Vogtle Early Site Permit and the associated NRC Safety Evaluation Report (SER) with Open Items. The Committee decided that it was satisfied with the EDO's response.

OTHER RELATED ACTIVITIES OF THE COMMITTEE

During the period from December 9, 2007, through February 6, 2008, the following Subcommittee meetings were held:

- Safety Research Program – December 18, 2007

The Subcommittee discussed the scope of long-term research the agency needs to consider. At this meeting, the Subcommittee had the benefit of presentations by John Ahearn, former NRC Chairman, Alex Marion, Executive Director of Nuclear Operations and Engineering at the Nuclear Energy Institute (NEI), Tom Miller of U.S. Department of Energy (DOE), and Robert Hill from Argonne National Laboratory representing the DOE's Global Nuclear Energy Partnership (GNEP). During this meeting, the Subcommittee also had presentations from Brian Sheron, Director, Office of Nuclear Regulatory Research, and Gary Holohan, Deputy Director, Office of New Reactors.

- Reliability & Probabilistic Risk Assessment – December 19, 2007

The Subcommittee discussed Draft NUREG-1855, "Guidance on the Treatment of Uncertainties in Risk-Informed Decisionmaking."

- ESBWR – January 16 and 17, 2008

The Subcommittee discussed Chapters 4, 6, 15, and 21 of the SER with Open Items associated with the ESBWR design certification application.

- Thermal-Hydraulic Phenomena, and Reliability and Probabilistic Risk Assessment – January 18, 2008

The Subcommittees discussed results of the Cable Response to Live Fire (CAROLFIRE) Testing and Fire Model Improvement Program and related matters.

- Safety Research Program –February 5, 2008

The Subcommittee met with Jacques Repussard and Michel Schwarz representing France's Institut de Radioprotection et de Sûreté Nucléaire (IRSN); Carlo Vitanza representing the Nuclear Energy Agency (NEA) of the Organization of Economic Cooperation and Development (OECD); and Christer Viktorsson representing the Nuclear Installation Safety Division of the International Atomic Energy Agency (IAEA). This meeting was held to obtain international perspectives on long-term reactor safety research.

- Future Plant Designs – February 6, 2008

The Subcommittee discussed the proposed licensing strategy for the Next Generation Nuclear Plant and related matters.

- Planning and Procedures – February 6, 2008

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 review the Vermont Yankee Nuclear Power Station license renewal application and the associated final SER, specifically the resolution of the environmentally assisted fatigue issue, during its March 2008 meeting.

- The Committee plans to review Chapters 9, 10, 13, and 16 of the SER with Open Items associated with the ESBWR design certification application during its March 2008 meeting.

- The Committee plans to continue its review of the proposed licensing strategy for NGNP during its April 2008 meeting.

- The Committee plans to review the staff's evaluation of the BWROG containment overpressure credit methodology described in the Topical Report, NEDO-33347P, "Containment Overpressure Credit for Net Positive Suction Head (NPSH)," Revision 0.

- The Committee would like to be kept informed of the staff's progress in analyzing its CAROLFIRE test data and developing guidance for future use of these data.

- The Committee plans to have further interaction with the staff to discuss the progress made in the SOARCA project.

PROPOSED SCHEDULE FOR THE 550th ACRS MEETING

The Committee agreed to consider the following topics during the 550th ACRS meeting, to be held on March 6-8, 2008:

- License Renewal Application and the final SER for the James A. FitzPatrick Nuclear Power Plant

- License Renewal Application and the final SER for the Vermont Yankee Nuclear Power Station

- Selected Chapters of the SER Associated with the ESBWR Design Certification Application

- Meeting with Commissioner Lyons regarding items of mutual interest.

- Anticipated Future Committee Schedule and Workload

Sincerely,

/RA/

William J. Shack
Chairman

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/RA/

William J. Shack
Chairman

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

April 9, 2008

MEMORANDUM TO: Carol A. Brown, Technical Secretary
Advisory Committee on Reactor Safeguards

FROM: Cayetano Santos, Chief
Reactor Safety Branch
Advisory Committee on Reactor Safeguards

SUBJECT: MINUTES OF THE 549th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS), February 7 - 9, 2008

I certify that, to the best of my knowledge and belief, the minutes of the subject meeting are an accurate record of the proceedings for that meeting.

ADAMS Accession: ML080990354

	ACRS	SUNSI	
NAME	CSantos	JFlack	
DATE	04/09/08	04/09/08	

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

MEMORANDUM TO: Carol A. Brown, Technical Secretary
Advisory Committee on Reactor Safeguards

FROM: Cayetano Santos, Chief Cayetano Santos
Reactor Safety Branch
Advisory Committee on Reactor Safeguards

SUBJECT: MINUTES OF THE 549th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS), February 7 - 9, 2008

I certify that, to the best of my knowledge and belief, the minutes of the subject meeting are an accurate record of the proceedings for that meeting.

ADAMS Accession: ML080990354

	ACRS	SUNSI		
NAME	CSantos CS	JFlack		
DATE	4/9/08	4/9/2008		

CERTIFIED

Date Issued:
Date Certified:

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February 7 - 9, 2008

APPENDICES

REPORTS

Reports to Dale E. Klein, Chairman, NRC, from William J. Shack, Chairman, ACRS:

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MINUTES OF THE 549th MEETING OF THE
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
February 7 - 9, 2008
ROCKVILLE, MARYLAND

The **549th** meeting of the Advisory Committee on Reactor Safeguards (ACRS) was held in Conference Room 2B3, Two White Flint North Building, Rockville, Maryland, on **February 7 - 9, 2008**. Notice of this meeting was published in the *Federal Register* on **January 24, 2008** (73 FR 4287) (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.

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: Dr. William J. Shack (Chairman), Dr. Mario V. Bonaca (Vice-Chairman), Dr. Dennis Bley,Dr. Said Abdel-Khalik (Member-at-Large), Dr. George E. Apostolakis, Dr. Sam Armijo, Dr. Michael Corradini, Mr. Otto L. Maynard, Dr. Dana A. Powers, Mr. Jack Sieber, and Mr. John Stetkar. For a list of other attendees, see Appendix III.

I. Chairman's Report (Open)

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

Dr. William J. Shack, Committee Chairman, convened the meeting at 8:30 A.M. He announced in his opening remarks that the meeting was being conducted in accordance with the provisions of the Federal Advisory Committee Act. In addition, he reviewed the agenda for the meeting and noted that no written comments or requests for time to make oral statements from members of the public had been received. Dr. Shack also noted that a transcript of the open portions of the meeting was being kept and speakers were requested to identify themselves and speak with clarity and volume. He discussed the items of current interest and administrative details for consideration by the full Committee.

II. License Renewal Application for the Vermont Yankee Nuclear Power Station

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

The Committee met with the representatives of Entergy Nuclear Operations, Inc., (the applicant) and the NRC staff to discuss the license renewal application for the Vermont Yankee Nuclear Power Station (VYNPS) and the associated Safety Evaluation Report (SER). The operating license for VYNPS expires on March 21, 2012. The applicant has requested approval for continued operation for a period of 20 years beyond the current license expiration date.

In the SER, with the exception of an issue related to environmentally assisted fatigue (EAF) of reactor coolant pressure boundary components, the staff documented its review of the license renewal application and other information submitted by Entergy and obtained during the audits and inspections conducted at the plant site. The staff reviewed: the completeness of the applicant's identification of structures, systems, and components that are within the scope of license renewal; the integrated plant assessment process; the applicant's identification of the plausible aging mechanisms associated with passive, long-lived components; the adequacy of the applicant's Aging Management Programs; and the identification and assessment of time-limited aging analyses requiring review.

For the remaining EAF issue, the applicant has submitted additional confirmatory analysis information that is currently being reviewed by the staff. The staff currently plans to complete the final SER, including resolution of the EAF issue, such that the ACRS will be able to complete its review of the VYNPS license renewal application at its March 2008 meeting.

Committee Action

The Committee plans to continue its discussion of the VYNPS License Renewal Application and the associated final SER, especially the resolution of the EAF issue, during its March 2008 meeting.

III. Draft Final Revision 1 to Regulatory Guide 1.45, "Guidance on Monitoring and Responding to Reactor Coolant System Leakage"

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

The Committee met with representatives of the NRC staff regarding the proposed Revision 1 to Regulatory Guide 1.45. Regulatory Guide 1.45 was first issued in 1973 to provide guidance on leak detection in containment. It recommended that three separate methods of measurement be employed to detect leaks of one gallon per minute or less from unidentified sources. Following the Davis-Besse reactor vessel head event, one of the areas identified for examination was the need for additional guidance in the area of leak detection from the reactor coolant system. An examination of operating experience showed that over half of reported leaks were too small to be detected by measurement methods and were found by visual inspection. Large leaks were detected by the installed measurement systems. The Revised Regulatory Guide recommends the use of local detection methods in potentially critical areas such as those where small leaks could expose low-alloy steel to borated water. Regulatory Guide 1.45, Revision 1 also recommends inclusion of monitoring and trending procedures in the plant technical specifications. Regulatory Guide 1.45, Revision 1 will be applied only to new reactors.

Committee Action

The Committee issued a letter to the Executive Director for Operations on this matter, dated February 22, 2008, recommending that Regulatory Guide 1.45, Revision 1 be issued.

IV. Proposed Licensing Strategy for the Next Generation Nuclear Plant (NGNP)

[Note: Ms. Maitri Banerjee was the Designated Federal Official for this portion of the meeting.]

The Committee met with the representatives of the Department of Energy (DOE) and the NRC staff to discuss the development of the draft licensing strategy report prepared by a DOE and NRC joint working group in response to the Energy Policy Act of 2005 (EPAct). The EPAct directed DOE and the NRC to describe the ways in which the current light water reactor licensing requirements could be adapted for the prototype NGNP, the analytical tools that would be needed by the NRC to independently verify the NGNP safety performance, research and development (R&D) activities the NRC will need to conduct to review the NGNP license application, and a budget estimate associated with the licensing strategy. The licensing strategy development report needs to be submitted to Congress by August 7, 2008. The EPAct also mandated that the NGNP provide process heat for hydrogen generation.

The DOE and NRC staff had undertaken jointly a "phenomena identification and ranking table (PIRT) process" to assess the knowledge base for key phenomena, the adequacy and developmental needs for the analytical tools, and the R&D needs. The DOE staff described the technical challenges and experience associated with the high-temperature gas-cooled reactor technology and the associated use of process heat for hydrogen generation. DOE representatives also described the operating conditions for a pre-conceptual design, the needed technology development areas, ongoing and future test programs, and R&D needs. The NRC staff discussed the options for the licensing approach, highlights of the PIRT findings, needs for tools and data to perform confirmatory safety analyses, and other infrastructure needs.

The ACRS members discussed their comments and questions with the staff. The interface between the NGNP reactor and the hydrogen generation plant was one area of ACRS interest.

Committee Action

The Committee plans to continue its discussion of the NGNP issues during its April 2008 meeting.

V. Cable Response to Live Fire (CAROLFIRE) Testing and Fire Model Improvement Program

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

The Committee met with representatives of the NRC staff, Sandia National Laboratories, and the National Institute of Standards and Technology (NIST) to discuss results of the Cable Response to Live Fire (CAROLFIRE) Testing and Fire Model Improvement Program. This Program was based on Regulatory Issue Summary (RIS) 2004-03 Rev. 1, which had explicitly described a set of cable/circuit configurations in need of more research to determine failure characteristics.

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Committee Action

The Committee issued a letter to the Executive Director for Operations on this matter, dated February 28, 2008, recommending that NUREG/CR-6931, "Cable Response to Live Fire (CAROLFIRE)," including the electronic data sets, be published. The Committee also recommended that the staff continue to analyze the CAROLFIRE data and develop additional guidance regarding the use of the results.

VI. Boiling Water Reactor Owners Group's (BWROG) Proposed Containment Overpressure Credit Methodology

[Note: Ms. Zena Abdullahi was the Designated Federal Official for this portion of the meeting.]

The Committee was briefed by representatives of the NRC staff and the Boiling Water Reactor Owners Group (BWROG) regarding a proposed containment overpressure methodology which is documented in the Topical Report, NEDO-3337P, "Containment Overpressure Credit for Net Positive Suction Head (NPSH)," Revision 0. This methodology was developed to address some of the comments made by the ACRS during its review of the extended power uprate (EPU) applications. The Committee commented on the acceptability of relying on containment overpressure credit in meeting the required NPSH and the increases in both the credit and the duration needed for EPU operation. The Committee also commented on the lack of consistency in the licensees' approaches in determining the containment overpressure credit, pointing out the need for a well-defined risk assessment for some of the event scenarios.

The BWROG briefed the Committee on the proposed guidance process and the newly developed statistical methodology for calculating the containment response and the overpressure credit needed. This methodology will reduce some of the conservatisms currently employed in the deterministic containment analyses methodology.

The NRC staff presented the regulatory history and positions on crediting containment overpressure in meeting the required NPSH. In addition, the NRC staff discussed its positions for accepting containment overpressure credit. The staff stated that if there is no practical alternative, containment overpressure credit is accepted, provided that the containment overpressure is calculated in a conservative manner that minimizes the available containment pressure response.

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Committee Action

This was an information briefing. No Committee action was necessary. The Committee plans to review the staff's evaluation of the proposed methodology described in Topical Report, NEDO-33347P, "Containment Overpressure Credit for Net Positive Suction Head (NPSH)," Revision 0.

VII. ACRS Report on the NRC Safety Research Program

[Note: Mr. 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 February 2008 meeting, the Committee completed its biennial review and evaluation of the Reactor Safety Research Program sponsored by the NRC.

Committee Action

The Committee issued a report to the Commission, dated March 5, 2008, transmitting an advance copy of its 2008 biennial report on, "Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program." The final report will be published as NUREG-1635, Vol. 8.

VIII. Executive Session

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

A. RECONCILIATION OF ACRS COMMENTS AND RECOMMENDATIONS/EDO COMMITMENTS

- The Committee considered the EDO's response of February 1, 2008, to comments and recommendations included in the November 20, 2007, ACRS letter concerning Chapters 2, 5, 8, 11, 12, and 17 of the NRC staff's SER with Open Items related to the certification of the ESBWR [Economic Simplified Boiling Water Reactor] design. The Committee decided that it was satisfied with the EDO's response. **The EDO stated that the staff has sent a request for additional information to General Electric-Hitachi Nuclear Energy (GEH) to obtain the necessary information for developing the source term of radioactive materials released into the reactor coolant system. The EDO committed to provide this information to ACRS.**

- The Committee considered the EDO's response of December 6, 2007, to comments and recommendations in the October 19, 2007, ACRS letter concerning the draft final Generic Letter 2007-02, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems." The Committee decided that it was satisfied with the EDO's response. **The EDO indicated that the staff will provide the ACRS an opportunity to review proposed interim measures or topical reports developed as a result of this Generic Letter.**

- The Committee considered the EDO's response of January 30, 2008, to comments and recommendations included in the December 20, 2007, ACRS letter concerning Draft Final NUREG-1829, "Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process," and Draft NUREG-XXXX, "Seismic Considerations for the Transition Break Size." The Committee decided it was satisfied with the EDO's response.

- The Committee considered the EDO's response of January 30, 2008, to comments and recommendations included in the December 27, 2007, ACRS letter concerning the AREVA Detect and Suppress Stability Solution and Methodology. The Committee decided that it was satisfied with the EDO's response.

- The Committee considered the EDO's response of December 27, 2007, to comments and recommendations included in the November 19, 2007, ACRS letter on the staff's implementation of Lessons Learned from Reviews of Early Site Permit (ESP) Applications. The Committee decided that it was satisfied with the EDO's response.

- The Committee considered the EDO's response of December 28, 2007, to comments and recommendations included in the November 20, 2007, ACRS letter on the Southern Nuclear Operating Company (SNC) Application for the Vogtle Early Site Permit and the associated NRC Safety Evaluation Report (SER) with Open Items. The Committee decided that it was satisfied with the EDO's response.

OTHER RELATED ACTIVITIES OF THE COMMITTEE

During the period from December 9, 2007, through February 6, 2008, the following Subcommittee meetings were held:

- Safety Research Program – December 18, 2007

The Subcommittee discussed the scope of long-term research the agency needs to consider. At this meeting, the Subcommittee had the benefit of presentations by John Ahearn, former NRC Chairman, Alex Marion, Executive Director of Nuclear Operations and Engineering at the Nuclear Energy Institute (NEI), Tom Miller of U.S. Department of Energy (DOE), and Robert Hill from Argonne National Laboratory representing the DOE's Global Nuclear Energy Partnership (GNEP). During this meeting, the Subcommittee also had presentations from Brian Sheron, Director, Office of Nuclear Regulatory Research, and Gary Holohan, Deputy Director, Office of New Reactors.

- Reliability & Probabilistic Risk Assessment – December 19, 2007

The Subcommittee discussed Draft NUREG-1855, "Guidance on the Treatment of Uncertainties in Risk-Informed Decisionmaking."

- ESBWR – January 16 and 17, 2008

The Subcommittee discussed Chapters 4, 6, 15, and 21 of the SER with Open Items associated with the ESBWR design certification application.

- Thermal-Hydraulic Phenomena, and Reliability and Probabilistic Risk Assessment – January 18, 2008

The Subcommittees discussed results of the Cable Response to Live Fire (CAROLFIRE) Testing and Fire Model Improvement Program and related matters.

- Safety Research Program –February 5, 2008

The Subcommittee met with Jacques Repussard and Michel Schwarz representing France's Institut de Radioprotection et de Sûreté Nucléaire (IRSN); Carlo Vitanza representing the Nuclear Energy Agency (NEA) of the Organization of Economic Cooperation and Development (OECD); and Christer Viktorsson representing the Nuclear Installation Safety Division of the International Atomic Energy Agency (IAEA). This meeting was held to obtain international perspectives on long-term reactor safety research.

- Future Plant Designs – February 6, 2008

The Subcommittee discussed the proposed licensing strategy for the Next Generation Nuclear Plant and related matters.

- Planning and Procedures – February 6, 2008

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 review the Vermont Yankee Nuclear Power Station license renewal application and the associated final SER, specifically the resolution of the environmentally assisted fatigue issue, during its March 2008 meeting.

- The Committee plans to review Chapters 9, 10, 13, and 16 of the SER with Open Items associated with the ESBWR design certification application during its March 2008 meeting.

- The Committee plans to continue its review of the proposed licensing strategy for NGNP during its April 2008 meeting.

- The Committee plans to review the staff's evaluation of the BWROG containment overpressure credit methodology described in the Topical Report, NEDO-33347P, "Containment Overpressure Credit for Net Positive Suction Head (NPSH)," Revision 0.

- The Committee would like to be kept informed of the staff's progress in analyzing its CAROLFIRE test data and developing guidance for future use of these data.

- The Committee plans to have further interaction with the staff to discuss the progress made in the SOARCA project.

B. Report on the Meeting of the Planning and Procedures Subcommittee Held on February 6, 2008

<u>Review of the Member Assignments and Priorities for ACRS Reports and Letters for the February ACRS Meeting</u>

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

<u>Anticipated Workload for ACRS Members</u>

The anticipated workload for ACRS members through April 2008 was discussed. 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.

<u>Office of the Inspector General's Audit of the NRC License Renewal Program</u>

The Office of the Inspector General (OIG) performed an audit of the NRC license renewal program to determine the effectiveness of the NRC's review of the license renewal applications. The report documenting the results of the OIG audit was sent to all members by Gary Hammer in early January 2008.

The OIG concluded that overall, NRC had developed a comprehensive license renewal process to evaluate applications for extended periods of operations. However, OIG identified areas where improvements would enhance program operations.

OIG recommends that the Executive Director for Operations:

- Establish report-writing standards in the Project Team Guidance for describing the license renewal review methodology and providing support for conclusions in the licensee renewal reports.

- Revise the report quality assurance process for license renewal report review to include:

  —Establishing management controls for NRR and Division of License Renewal management to gauge the effectiveness of team leader and peer group report reviews, and

  —Implementing procedures that would specify additional report quality assurance steps to be taken in the event that the team leader and peer group report reviews fail to ensure report quality to management's expectations.

- Clarify guidance and adjust procedures for auditors' and inspectors' removal of licensee-provided documents from license renewal sites.

- Establish requirements and management controls to standardize the conduct and depth of license renewal operating experience reviews.

- Expedite completion of the details for a revised Inspection Procedure 71003.

- Communicate the details of revised Inspection Procedures 71003 to all applicable staff and stakeholders.

- Establish a review process to determine whether or not Interim Staff Guidance meets the provisions of 10 CFR 54.37(b), and document accordingly.

In addition, OIG recommends that the Commission:

- Affirm or modify the 1995 Commission's Statement of Considerations Position regarding the applicability of the backfit rule to license renewal applicants.

The staff and OIG disagree with regard to the applicability of the backfit rule to license renewal. The Commission is in the process of resolving this issue.

<u>Petition by Nine Intervener Groups to Suspend License Renewal Reviews for Four Plants</u>

On January 3, 2008, nine Intervener Groups filed a petition, requesting the Commission to suspend license renewal proceedings for the Oyster Creek, Indian Point, Pilgrim, and Vermont Yankee nuclear plants including NRC staff technical reviews and/or adjudicatory hearings, and conduct a comprehensive overhaul of the manner in which reviews of license renewal applications are carried out. Among several things they requested the Commission to perform an:

> Independent verification of whether the newly conducted NRC staff safety reviews for Oyster Creek, Indian Point, Pilgrim, and Vermont Yankee provide sufficient basis for the safety findings required by the Atomic Energy Act. If they do not, the Commission should establish a process for the reviews to be supplemented.

They state that the independent review mentioned above could either be carried out directly by the Commission, or could be delegated to the NRC's Atomic Safety and

Licensing Board, the Office of the Inspector General, or the ACRS. If the reviews are delegated, ultimate responsibility for their results should rest with the Commission.

Please be reminded that the Committee has completed its review of the license renewal applications for Oyster Creek and Pilgrim. It is scheduled to complete its review of the Vermont Yankee license renewal application in March 2008.

Since the Commission has not yet ruled on this petition, the ACRS should not discuss this matter and express its views.

Annual Visit to a Plant and Meeting with the Regional Administrator

Each year, the members visit a plant and hold a meeting with the Regional Administrator. In 2007, the members visited San Onofre and met with the Region IV Administrator. This year, the members need to visit a plant in Region III and meet with the Region III Administrator. Mr. Sieber, the Chairman of the Plant Operations and Fire Protection Subcommittee, recommends that the members visit either LaSalle, Dresden, or Quad Cities.

Meeting with the Commission

The ACRS meeting with the Commission, previously scheduled for May 9, 2008, has been moved to June 5, 2008, between 1:30 and 3:30 p.m., because of the unavailability of some Commissioners. The ACRS staff will propose a list of topics for this meeting for Committee approval during the March ACRS meeting.

Regulatory Information Conference

The 2008 Regulatory Conference is scheduled to be held on March 11-13, 2008, at the Bethesda North Marriott Hotel. This Conference brings a diverse group of stakeholders together to discuss significant and timely regulatory activities. The Conference will focus on various technical areas related to operating reactors, new and advanced reactors, as well as reactor research. A proposed schedule for this conference is attached.

Interview of a Candidate for Potential Membership on the ACRS

The members are scheduled to interview a candidate with operating experience during lunchtime on Friday, February 8, 2008. Subsequent to the interview, the ACRS Chairman needs to provide feedback on this candidate to the Chairman of the ACRS Member Candidate Screening Panel.

NRC Budget for FY2008

The Agency is no longer operating under a continuing resolution, since President Bush signed the appropriations bill which provides the Agency with $9.6 million above its budget request, for a total of $926.1 million. This adds $2.2 million to the NRC budget for international activities to support enhancing foreign regulators' programs to increase security over radioactive sources, and $15 million to support nuclear education, including scholarships and graduate fellowships. The appropriation bill reduces funding from the Nuclear Waste Fund for the Agency's high-level waste activities by $8.2 million.

Commission Meeting on New Reactor Issues

The Commission is scheduled to hold a meeting on new reactor issues on February 20, 2008. A proposed schedule for this meeting is attached. Dr. Corradini has been invited to attend this meeting to provide presentation on the following topics:

- NAS Review of DOE's Nuclear Energy Research Program with respect to Next Generation Nuclear Plant (NGNP).
- ACRS Review of Future Plant Designs and NGNP.

The Committee is scheduled to prepare a report on the proposed licensing strategy for NGNP during its March 2008 meeting. Therefore, Dr. Corradini will not have the Committee's views prior to the February 20, 2008 Commission meeting. Dr. Corradini may want to provide his presentation slides to the Committee at the February meeting and obtain feedback.

If Dr. Corradini wants to present additional views, he should make it clear to the Commission that those are his personal views and do not necessarily reflect those of the ACRS.

Quadripartite Working Group Meeting

France's Groupe Permanent Réacteurs (GPR) will host the second Quadripartite WG meeting in France on the general topic of "EPR" on October 9-10, 2008. Dr. Powers, Dr. Bonaca, and Mr. Stetkar will be attending this meeting.

Dr. Powers, Chairman of the EPR Subcommittee, proposed the following topics:

- PRA
- Digital I&C
- Fire Risk
- Quality Assurance

GPR has the following questions:

1) Does ACRS want to discuss how the PRA results are used at the design stage?
2) For thermal-hydraulics, can we understand the studies perform for design accidents studies or is included the severe accidents studies?

3) The topic "Quality Assurance" is too general and needs to be refined. Is it the quality of the realization?

Further, GPR has proposed the following topics:

1) the severe accidents (low pressure core melt, core catcher design, early releases sequences preclusion)
2) Break preclusion
3) External hazards (external flooding)

GPR's concern is that if all ACRS and GPR proposed topics are selected, two days may not be sufficient to have a detailed discussion on these topics. They suggest selecting the topics based on the number of presentations, so that the more the number of presentations (at least one from each Country) on a topic, then that topic would be a likely candidate for the agenda. GPR is planning a visit to "Flamanville 3" (a nuclear power plant using EPR technology) on October 8^{th}. They would like to know the number of visitors as soon as possible.

<u>State of Vermont's Request to Postpone ACRS Review of the Vermont Yankee License Renewal Application</u>

In a letter to the ACRS Chairman dated January 25, 2008, Mr. David O'Brien, Commissioner, Vermont Department of Public Service, requests that the ACRS postpone its final review of the Vermont Yankee Licensee Renewal Application, which is now scheduled for the February ACRS meeting, to the March ACRS meeting. This request stems from the fact that there are still RAIs awaiting answers from Entergy. Subsequent to receiving answers to the RAIs, the staff will have to analyze and reach conclusions. This would mean that the final SER will not be available to the ACRS and the public until close to the date of the February meeting. Vermont wants ACRS to have ample time to review the SER prior to performing the final review of the Vermont Yankee License Renewal Application.

The staff previously stated that it would submit information to the ACRS on its evaluation of the response submitted by Entergy to RAIs related to the TLAA on environmentally assisted fatigue during the week of January 20, subject to receiving necessary information from Entergy in a timely manner. On January 30, 2008, the staff has received information from Entergy. The staff is in the process of evaluating the information submitted by Entergy.

Since the final (complete) SER will not be available to the ACRS prior to the February meeting, the Committee should consider completing its report to the Commission at the March meeting. The staff previously told the cognizant ACRS staff that even if the ACRS issues its report in March, it will not impact the staff's schedule for approving the Vermont Yankee License Renewal Application.

<u>Proposed Merger of ACRS and ACNW&M</u>

In a Staff Requirements Memorandum, dated February 5, 2008, the Commission states the following:

- The Commission has approved the merger of ACNW&M [as a Subcommittee] back into the ACRS.

- The Executive Director of the ACRS/ACNW&M should complete all necessary administrative actions to facilitate this merger in an orderly fashion.

- The transition plan should address disposition of topics currently in the ACNW&M action plan, particularly for issues under active consideration, and whether they should continue under the new Subcommittee.

- Prior to the merger of the two Committees, the ACNW&M will continue to meet under the direction of Dr. Ryan to complete the activities as outlined in the transition plan.

Member Issue

- Travel Request

NRC and DOE are co-sponsoring a Workshop on U.S. Nuclear Power Plant Life Extension Research and Development to gain a better understanding from stakeholders and the scientific community on needed research to support continued operation of current LWRs beyond 60 years. This Workshop is scheduled to be held on February 19-21, 2008, at Hyatt Regency, Bethesda.

Drs. Armijo, Bonaca, and Shack request Committee approval and support to attend this Workshop.

PROPOSED SCHEDULE FOR THE 550th ACRS MEETING

The Committee agreed to consider the following topics during the 550th ACRS meeting, to be held on March 6-8, 2008:

- License Renewal Application and the final SER for the James A. FitzPatrick Nuclear Power Plant

- License Renewal Application and the final SER for the Vermont Yankee Nuclear Power Station

- Selected Chapters of the SER Associated with the ESBWR Design Certification Application

- Meeting with Commissioner Lyons regarding items of mutual interest.

- Anticipated Future Committee Schedule and Workload

- The Committee plans to have further interaction with the staff to discuss the progress made in the SOARCA project.

C. PROPOSED SCHEDULE FOR THE 550th ACRS MEETING

The Committee agreed to consider the following topics during the 550th ACRS meeting, to be held on March 6-8, 2008:

- License Renewal Application and the final SER for the James A. FitzPatrick Nuclear Power Plant
- License Renewal Application and the final SER for the Vermont Yankee Nuclear Power Station
- Selected Chapters of the SER Associated with the ESBWR Design Certification Application
- Meeting with Commissioner Lyons regarding items of mutual interest.
- Anticipated Future Committee Schedule and Workload

Regulations that states NCUA will provide notice in the **Federal Register** when funds in the program are available.

By the National Credit Union Administration Board on January 17, 2008.

Mary F. Rupp,

Secretary, NCUA Board.

[FR Doc. E8–1147 Filed 1–23–08; 8:45 am]

BILLING CODE 7535–01–P

NUCLEAR REGULATORY COMMISSION

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

The ACRS Subcommittee on Planning and Procedures will hold a meeting on February 6, 2008, Room T–2B1, 11545 Rockville Pike, Rockville, Maryland.

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

The agenda for the subject meeting shall be as follows:

Wednesday, February 6, 2008, 8:30 a.m. Until 10 a.m.

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

Members of the public desiring to provide oral statements and/or written comments should notify the Designated Federal Officer, Mr. Sam Duraiswamy (telephone: 301–415–7364) between 7:30 a.m. and 4 p.m. (ET) five days prior to the meeting, if possible, so that appropriate arrangements can be made. Electronic recordings will be permitted only during those portions of the meeting that are open to the public. Detailed procedures for the conduct of and participation in ACRS meetings were published in the **Federal Register** on September 26, 2007 (72 FR 54695).

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

Dated: January 15, 2008.

Charles G. Hammer,

Acting Chief, Reactor Safety Branch.

[FR Doc. E8–1071 Filed 1–23–08; 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 February 7–9, 2008, 11545 Rockville Pike, Rockville, Maryland. The date of this meeting was previously published in the **Federal Register** on Monday, October 22, 2007 (72 FR 59574).

Thursday, February 7, 2008, Conference Room T–2B3, Two White Flint North, Rockville, Maryland

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

8:35 a.m.–10:30 a.m.: Final Review of the License Renewal Application for the Vermont Yankee Nuclear Power Station (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff and Entergy Nuclear Operations regarding the License Renewal Application for the Vermont Yankee Nuclear Power Station and the associated NRC staff's Final Safety Evaluation Report.

10:45 a.m.–12 p.m.: Draft Final Revision to Regulatory Guide 1.45 (DG–1173), "Guidance on Monitoring and Responding to Reactor Coolant System Leakage" (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding draft final Revision 1 to Regulatory Guide 1.45 (DG–1173) and the staff's resolution of public comments.

1 p.m.–3 p.m.: Proposed Licensing Strategy for the Next Generation Nuclear Plant (NGNP) (Open/Closed)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff and Department of Energy regarding the proposed licensing strategy for the Next Generation Nuclear Plant.

[Note: A portion of this session may be closed to prevent disclosure of information the premature disclosure of which would be likely to significantly frustrate implementation of a proposed agency action pursuant to 5 U.S.C. 552b(c)(9)(B).]

3:15 p.m.–5 p.m.: Cable Response to Live Fire (CAROLFIRE) Testing and Fire Model Improvement Program (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff and its contractors regarding the results of the CAROLFIRE Testing and Fire Model Improvement Program, including staff's resolution of public comments.

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, as well as a proposed report on State-of-the-Art Reactor Consequence Analysis (SOARCA) program.

Friday, February 8, 2008, Conference Room T–2B3, Two White Flint North, Rockville, Maryland

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

8:35 a.m.–10:30 a.m.: Proposed BWR Owners Group (BWROG) Topical Report on Methodology for Calculating Available Net Positive Suction Head (NPSH) for ECCS Pumps (Open/Closed)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff and the BWR Owners Group regarding the proposed topical report on Methodology for Calculating the Available NPSH for ECCS Pumps, including NRC staff's position on this topical report.

[Note: A portion of this session may be closed to discuss and protect information that is proprietary to BWROG and their contractors pursuant to 5 U.S.C. 552b(c)(4).]

10:45 a.m.–11:30 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.

11:30 a.m.–11:45 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.

11:45 a.m.–12 p.m.: Subcommittee Report (Open)—The Committee will hear a report by the Chairman of the ACRS Subcommittee on Reliability and Probabilistic Risk Assessment (PRA) regarding Draft NUREG–1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking," that was discussed during the meeting on December 19, 2007.

1 p.m.–3 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.

3:15 p.m.–7 p.m.: Preparation of ACRS Reports (Open)—The Committee will discuss proposed ACRS reports.

Saturday, February 9, 2008, Conference Room T–2B3, Two White Flint North, Rockville, Maryland

7:30 a.m.–9:30 a.m.: Draft ACRS Report on the NRC Safety Research Program (Open)—The Committee will continue its discussion of the draft ACRS report on the NRC Safety Research Program.

9:45 a.m.–1 p.m.: Preparation of ACRS Reports (Open)—The Committee will continue its discussion of proposed ACRS reports.

1 p.m.–1:30 p.m.: Miscellaneous (Open)—The Committee will discuss matters related to the conduct of Committee activities and matters and specific issues that were not completed during previous meetings, as time and availability of information permit.

Procedures for the conduct of and participation in ACRS meetings were published in the **Federal Register** on September 26, 2007 (72 FR 54695). In accordance with those procedures, oral or written views may be presented by members of the public, including representatives of the nuclear industry. Electronic recordings will be permitted only during the open portions of the meeting. Persons desiring to make oral statements should notify the Cognizant ACRS staff named below five days before the meeting, if possible, so that appropriate arrangements can be made to allow necessary time during the meeting for such statements. Use of still, motion picture, and television cameras during the meeting may be limited to selected portions of the meeting as determined by the Chairman. Information regarding the time to be set aside for this purpose may be obtained by contacting the Cognizant ACRS staff prior to the meeting. In view of the possibility that the schedule for ACRS meetings may be adjusted by the Chairman as necessary to facilitate the conduct of the meeting, persons planning to attend should check with the Cognizant ACRS staff if such rescheduling would result in major inconvenience.

In accordance with Subsection 10(d) (Pub.L. 92–463), I have determined that it may be necessary to close portions of this meeting noted above to discuss and protect information classified as proprietary to BWROG, and their contractors pursuant to 5 U.S.C. 552b(c)(4), and information the premature disclosure of which would be likely to significantly frustrate implementation of a proposed agency action pursuant to 5 U.S.C. 552b(c)(9)(B).

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. Girija S. Shukla, Cognizant ACRS staff (301–415–6855), between 7:30 a.m. and 4 p.m., (ET). ACRS meeting agenda, meeting transcripts, and letter reports are available through the NRC Public Document Room at *pdr@nrc.gov*, or by calling the PDR at 1–800–397–4209, or from the Publicly Available Records System (PARS) component of NRC's document system (ADAMS) which is accessible from the NRC Web site at *http://www.nrc.gov/reading-rm/adams.html* or *http://www.nrc.gov/reading-rm/doc-collections/* (ACRS & ACNW Mtg schedules/agendas).

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

Individuals or organizations requesting this service will be responsible for telephone line charges and for providing the equipment and facilities that they use to establish the video teleconferencing link. The availability of video teleconferencing services is not guaranteed.

Dated: January 17, 2008.

Annette Vietti-Cook,

Secretary of the Commission.

[FR Doc. E8–1189 Filed 1–23–08; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards; Subcommittee Meeting on Safety Research Program; Notice of Meeting

The ACRS Subcommittee on Safety Research Program will hold a meeting on February 5, 2008, Room T–2B1, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance.

The agenda for the subject meeting shall be as follows:

Tuesday, February 5, 2008—9:30 a.m. Until the Conclusion of Business

The Subcommittee will discuss the scope of long-term research the agency needs to consider. The purpose of this meeting is to gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Members of the public desiring to provide oral statements and/or written comments should notify the Designated Federal Official, Dr. Hossein P. Nourbakhsh (Telephone: 301–415–5622) five days prior to the meeting, if possible, so that appropriate arrangements can be made. Electronic recordings will be permitted. Detailed procedures for the conduct of and participation in ACRS meetings were published in the **Federal Register** on September 26, 2007 (72 FR 54695).

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

Dated: January 15, 2008.

Charles G. Hammer,

Acting Chief, Reactor Safety Branch.

[FR Doc. E8–1073 Filed 1–23–08; 8:45 am]

BILLING CODE 7590–01–P

OFFICE OF THE UNITED STATES TRADE REPRESENTATIVE

[Docket No. WTO/DS–291]

WTO Dispute Settlement Proceedings Regarding Measures of the European Communities Affecting the Approval and Marketing of Biotech Products

AGENCY: Office of the United States Trade Representative.

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

February 14, 2008

SCHEDULE AND OUTLINE FOR DISCUSSION
550th ACRS MEETING
MARCH 6-8, 2008

THURSDAY, MARCH 6, 2008, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

2) 8:35 - ~~10:30~~ 10:20 A.M. Final Review of the License Renewal Application for the James A. FitzPatrick Nuclear Power Plant (Open) (MVB/MB)
   2.1) Remarks by the Subcommittee Chairman
   2.2) Briefing by and discussions with representatives of the NRC staff and Entergy Nuclear Operations, Inc. regarding the License Renewal Application for the James A. FitzPatrick Nuclear Power Plant and the associated NRC staff's Final Safety Evaluation Report (SER).

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

~~10:30~~ 10:20 - 10:45 A.M. *****BREAK*****

3) 10:45 - 12:15 P.M. Final Review of the License Renewal Application for the Vermont Yankee Nuclear Power Station (Open) (MVB/CGH/CLB)
   3.1) Remarks by the Subcommittee Chairman
   3.2) Briefing by and discussions with representatives of the NRC staff and Entergy Nuclear Operations, Inc. regarding the License Renewal Application for the Vermont Yankee Nuclear Power Station and the associated NRC staff's Final SER, specifically, resolution of the environmentally assisted fatigue issue, and other related matters.

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

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

4) 1:15 - 3:15 P.M. Selected Chapters of the SER Associated with the ESBWR Design Certification Application (Open/Closed) (MLC/CGH)
   4.1) Remarks by the Subcommittee Chairman
   4.2) Briefing by and discussions with representatives of the NRC staff and General Electric – Hitachi Nuclear Energy (GEH) regarding selected Chapters of the SER With Open

Items associated with the ESBWR design certification application.

[Note: A portion of this session may be closed to protect information that is proprietary to GEH and its contractors pursuant to 5 U.S.C. 552b (c) (4).]

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

3:15 - 3:30 P.M. ***BREAK***

5) 3:30 - 3:45 P.M. Subcommittee Report (Open) (JDS/MB)
Report by and discussions with the Chairman of the ACRS Subcommittee on Plant License Renewal regarding interim Review of the License Renewal Application for the Wolf Creek Generating Station discussed during the Subcommittee meeting on March 5, 2008.

6) 3:45 - 7:00 P.M. Preparation of ACRS Reports (Open)
Discussion of proposed ACRS reports on:
- 6.1) License Renewal Application for the James A. FitzPatrick Nuclear Power Plant (MVB/MB)
- 6.2) License Renewal Application for the Vermont Yankee Nuclear Power Station (MVB/CGH/CLB)
- 6.3) Selected Chapters of the SER Associated with the ESBWR Design Certification Application (MLC/CGH)

FRIDAY, MARCH 7, 2008, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

8) 8:35 - 9:30 A.M. Meeting with Commissioner Lyons (Open) (WJS/GSS)
- 8.1) Remarks by the ACRS Chairman
- 8.2) Discussions with Commissioner Lyons regarding items of mutual interest.

9) 9:30 -10:15 A.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (WJS/FPG/SD)
- 9.1) Discussion of the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee during future ACRS meetings.
- 9.2) Report of the Planning and Procedures Subcommittee on matters related to the conduct of ACRS business, including anticipated workload and member assignments.

10:15 – 10:30 A.M. ***BREAK***

10) 10:30 – 11:30 A.M. Reconciliation of ACRS Comments and Recommendations (Open) (WJS, et al. /CS, et al.)
Discussion of the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.

11) 11:30 – 12:30 P.M. Preparation of ACRS Reports (Open)
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11.1) License Renewal Application for the James A. FitzPatrick Nuclear Power Plant (MVB/MB)
11.2) License Renewal Application for the Vermont Yankee Nuclear Power Station (MVB/CGH/CLB)
11.3) Selected Chapters of the SER Associated with the ESBWR Design Certification Application (MLC/CGH)

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

12) 1:30 - 7:00 P.M. Preparation of ACRS Reports (Open)
Continue discussion of proposed ACRS reports listed under Item 11.

SATURDAY, MARCH 8, 2008, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

13) 8:30 A.M. - 1:00 P.M. Anticipated Future Committee Schedule and Workload (Open) (WJS/FPG)
Discussion of anticipated future ACRS schedule and workload.

(10:30 - 10:45 A.M. BREAK)

14) 1:00 - 1:30 P.M. Miscellaneous (Open) (WJS/FPG)
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.

One (1) electronic copy and thirty-five (35) hard copies of the presentation materials should be provided to the ACRS.

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
549TH FULL COMMITTEE MEETING

February 7-9, 2008

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TODAY'S DATE: February 7, 2008

	NAME	NRC ORGANIZATION
1	Thomas Kenyon	NRC/NRO/DNRL/NRGA
2	Sud Basu	NRC/RES
3	Yuri Orechwa	NRC/DSS
4	SHAH MALIK	NRC/RES
5	FAROUK ELTAWILA	NRC/RES
6	DONALD HELTON	NRC/RES
7	Rob Versluis	US DOE
8	N. P. KADAMBI	NRC/RES
9	Mike Waterman	RES/DE
10	Don Carlson	RES/DSA
11	Allen Howe	NRC/NRO
12	Amy Hull	NRC/RES
13	Autumn Szabo	NRC/RES
14	David Stroup	NRC/RES
15	John Monninger	NRC/RES
16	Aixa Belen-Ojeda	NRC/RES
17	Mark Henry Salley	NRC/RES
18	Daniel Frumkin	NRC/NRR/DRA/AFPB
19	Sara Bernal	NRO/DCIP
20	Gabriel Taylor	RES/FRB
21	Kevin McGrattan	NIST
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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
549TH FULL COMMITTEE MEETING

February 7-9, 2008

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TODAY'S DATE: February 7, 2008

	NAME	NRC ORGANIZATION
1	Ken Chang	NRR/DLR
2	Jonathan Rowley	NRR/DLR
3	JAMES MEDOFF	NRR/DLR
4	PETER WEN	NRR/DLR
5	Qi Gan	NRR/DLR
6	John Fair	NRR/DE
7	P T Kuo	NRR/DLR
8	SAMSON LEE	NRR/DLR
9	DON DUBE	NRO/DSRA
10	On Yee	NRR/DLR
11	Robert Sun	NRR/DLR
12	Bill Rogers	NRR/DLR
13	Stacie Sakai	NRR/DLR
14	Tommy Le	NRR/DLR
15	RANI FRANOVICH	NRR/DLR
16	Raj Auluck	NRR/DLR
17	Samuel Hernandez	NRR/DLR
18	John Daily	NRR/DLR
19	Farideh Saba	NRR/DORL
20	Steven Jones	NRR/DSS
21	Drew Stuyvenberg	NRR/DLR
22	ED SMITH	NRR/DSS
23	Chris Sydnor	NRR/DCI
24	Yeon-Ki Chung	NRR/DLR
25	Cheng-Yang Li	NRO/DSRA
26	John Ridgely	NRC/RES
27	ANDREA VALENTIN	NRC/RES
28	Michael Boggi	NRC/RES

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
549TH FULL COMMITTEE MEETING

February 7-9, 2008

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TODAY'S DATE: February 7, 2008

	NAME	AFFILIATION
1	William Corwin	ORNL
2	Charles Brinkman	Westinghouse
3	~~DONALD HELTON~~	
4	Marvin Shaw	DOE
5	Jim Riccio	Greenpeace
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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
549TH FULL COMMITTEE MEETING

February 7-9, 2008

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TODAY'S DATE: February 7, 2008

	NAME	AFFILIATION
1	DAVID MANNAI	ENTERGY
2	Beth Bristol	Entergy
3	Richard Schaller	STARS P/I Aging Mgt COB
4	Garry G. Young	Entergy
5	David J. Lach	Entergy
6	Norm Rademacher	Entergy
7	John Dreyfuss	Entergy
8	A. Michael Metell	Entergy
9	JIM FITZPATRICK	Entergy
10	Paul Johnson	Entergy
11	Rick Plasse	ENTERGY
12	Gary Stevens	Structural Integrity Assoc.
13	Scott Goodwin	Entergy
14	Larry Lukens	Entergy
15	JEFF JEFFRIES	FENOC
16	ALAN COX	ENTERGY
17	Joe Hopenfeld	NEC
18	Chalmer Myer	SNC
19	Jeffrey Weik	PPL SUSQUEHANNA
20	Sarah Hofmann	Vermont Dept. of Public Service
21	Martin O'Neill	Morgan, Lewis & Bockius LLP
22	~~St~~	
23	Jim Nicholaus	PNNL
24	Steven Dolley	Platts/Inside NRC
25	TREVON COOK	U.S. DOE
26	~~Michael Boggi~~	
27	THOMAS J O'CONNOR	USDOE
28	SYD BALL	ORNL

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
549TH FULL COMMITTEE MEETING

February 7-9, 2008

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TODAY'S DATE: February 8, 2008

NAME	AFFILIATION
FRED EMERSON	GE HITACHI
Vincent Andersen	ERIN
Alan Wojchowski	NMC
PAUL HECK	TVA/BROWNS FERRY
Tim Abney	GE Hitachi
FRAN BOLGER	GE Hitachi
GUANGJUN LI	GE HITACHI
Randy Bunt	BWROG/SNC
John Freeman	Exelon Nuclear
JD WOLCOTT	TVA/BWROG
SPYROS TRAIFOROS	LINK

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
549TH FULL COMMITTEE MEETING

February 7-9, 2008

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TODAY'S DATE: February 8, 2008

NAME	NRC ORGANIZATION
Bruce Heida	NRR/DSS/SCVB
Robert Dennig	NRR/DSS/SCVB
AHSAN SALLMAN	NRR/DSS/SCVB
TOM BOYCE	NRR/DORL
JOHN JOLICOEUR	RES/DSA/NARB
Jocelyn Mitchell	RES/DSA
DON HELTON	NRC/RES/DSA

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

February 14, 2008

SCHEDULE AND OUTLINE FOR DISCUSSION
550th ACRS MEETING
MARCH 6-8, 2008

THURSDAY, MARCH 6, 2008, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

2) 8:35 - 10:30 A.M. Final Review of the License Renewal Application for the James A. FitzPatrick Nuclear Power Plant (Open) (MVB/MB)
   2.1) Remarks by the Subcommittee Chairman
   2.2) Briefing by and discussions with representatives of the NRC staff and Entergy Nuclear Operations, Inc. regarding the License Renewal Application for the James A. FitzPatrick Nuclear Power Plant and the associated NRC staff's Final Safety Evaluation Report (SER).

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

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

3) 10:45 - 12:15 P.M. Final Review of the License Renewal Application for the Vermont Yankee Nuclear Power Station (Open) (MVB/CGH/CLB)
   3.1) Remarks by the Subcommittee Chairman
   3.2) Briefing by and discussions with representatives of the NRC staff and Entergy Nuclear Operations, Inc. regarding the License Renewal Application for the Vermont Yankee Nuclear Power Station and the associated NRC staff's Final SER, specifically, resolution of the environmentally assisted fatigue issue, and other related matters.

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

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

4) 1:15 - 3:15 P.M. Selected Chapters of the SER Associated with the ESBWR Design Certification Application (Open/Closed) (MLC/CGH)
   4.1) Remarks by the Subcommittee Chairman
   4.2) Briefing by and discussions with representatives of the NRC staff and General Electric – Hitachi Nuclear Energy (GEH) regarding selected Chapters of the SER With Open Items associated with the ESBWR design certification application.

[Note: A portion of this session may be closed to protect information that is proprietary to GEH and its contractors pursuant to 5 U.S.C. 552b (c) (4).]

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

3:15 - 3:30 P.M. *****BREAK*****

5) 3:30 - 3:45 P.M. Subcommittee Report (Open) (JDS/MB)
Report by and discussions with the Chairman of the ACRS Subcommittee on Plant License Renewal regarding interim Review of the License Renewal Application for the Wolf Creek Generating Station discussed during the Subcommittee meeting on March 5, 2008.

6) 3:45 - 7:00 P.M. Preparation of ACRS Reports (Open)
Discussion of proposed ACRS reports on:
- 6.1) License Renewal Application for the James A. FitzPatrick Nuclear Power Plant (MVB/MB)
- 6.2) License Renewal Application for the Vermont Yankee Nuclear Power Station (MVB/CGH/CLB)
- 6.3) Selected Chapters of the SER Associated with the ESBWR Design Certification Application (MLC/CGH)

FRIDAY, MARCH 7, 2008, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

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- 9.1) Discussion of the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee during future ACRS meetings.
- 9.2) Report of the Planning and Procedures Subcommittee on matters related to the conduct of ACRS business, including anticipated workload and member assignments.

10:15 – 10:30 A.M. *****BREAK*****

10) 10:30 – 11:30 A.M. Reconciliation of ACRS Comments and Recommendations (Open) (WJS, et al. /CS, et al.)
Discussion of the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.

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11.3) Selected Chapters of the SER Associated with the ESBWR Design Certification Application (MLC/CGH)

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

12) 1:30 - 7:00 P.M. Preparation of ACRS Reports (Open)
Continue discussion of proposed ACRS reports listed under Item 11.

SATURDAY, MARCH 8, 2008, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

13) 8:30 A.M. - 1:00 P.M. Anticipated Future Committee Schedule and Workload (Open) (WJS/FPG)
Discussion of anticipated future ACRS schedule and workload.

(10:30 - 10:45 A.M. BREAK)

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APPENDIX V

LIST OF DOCUMENTS PROVIDED TO THE COMMITTEE
549th ACRS MEETING
February 7-9, 2008

MEETING HANDOUTS

AGENDA ITEM #	DOCUMENTS/HANDOUTS LISTED IN ORDER
1.	Opening Remarks by the ACRS Chairman
2.	Final Review of License Renewal Application for the Vermont Yankee Nuclear Power Station 1. Slides of the same name from Entergy 2. Chemistry Effects on EAS, Entergy (slides) 3. VYNPS Safety Evaluation Report, Slides from NRC/NRR/Rowley
3.	Draft Final Revision 1 to Regulatory Guide 1.45 (DG-1173), "Guidance on Monitoring and Responding to Reactor Coolant System Leakage" 4. Slides of the same name from NRC/NRO, RES and NRR
4.	Proposed Licensing Strategy for the Next Generation Nuclear Plant (NGNP) 5. Draft Agenda for the session 6. NGNP Licensing Strategy, Slides from NRC/RES and NRO 7. NGNP Design and Technology Development Status, Slides from Trevor Cook (DOE) and David Petti (INL) 8. Letter to Honorable Lando W. Zech, Jr. (former NRC Chairman); submitted to ACRS by J. Riccio, Green Peace on 2/7/08
6.	Cable Response to Live Fire (CAROLFIRE) Testing and Fire Model Improvement Program 9. Slides from NRC/RES, Mark Henry
7.	Opening Remarks by ACRS Chairman
8.	Proposed BWR Owners Group (BWROBG) Topical Report on Methodology for Calculating Available Net Positive Suction Head (NPSH) for ECCS Pumps 10. Slides from NRC/NRR, Richard Loebel 11. Slides from BWROG, Alan Wojchouski
9.	Future ACRS Activities/Report of the Planning and Procedures Subcommittee
10.	Reconciliation of ACRS Comments and Recommendations 12. Handout of the same name
11.	Subcommittee Report
12.	Draft ACRS Report on the NRC Safety Research Program
13.	Preparation of ACRS Reports

**Copies of most of the handouts can be found posted on the ACRS portion of the NRC Public Website.
[Note: Some documents listed herein may have been provided or prepared for the Committee use only. These documents must be reviewed prior to release to the public.]

Presentation Outline

1. Background
2. Safety Significance
3. Elements of Leakage Monitoring Program
4. Regulatory Positions
5. Disposition of Public Comments

RG 1.45, Rev 1. "Guidance on Monitoring and Responding to Reactor Coolant System Leakage"

RG 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems"

Chang-Yang Li, Office of New Reactors (301-415-2830, cyl1@nrc.gov)
Makuteswara Srinivasan, Office of Nuclear Regulatory Research (301-415-6356, mxs5@nrc.gov)
Kenneth Karwoski, Office of Nuclear Reactor Regulation (301-415-2752, kjk1@nrc.gov)
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Presented to the Advisory Committee on Reactor Safeguards (ACRS)
549th ACRS Meeting, February 7, 2008.

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Questions

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Section 4 Conclusion

- Review of the confirmatory EAF analysis is ongoing
  - VY provided additional information addressing effect of nozzle configuration difference on recirculation nozzle CUF, and
  - Additional information regarding water chemistry impact on F_{en}
- The staff's review of Section 4 is incomplete

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License Conditions

- The first license condition requires the applicant to include the UFSAR supplement required by 10 CFR 54.21(d) in the next UFSAR update, as required by 10 CFR 50.71(e), following the issuance of the renewed license.
- The second license condition requires future activities identified in the UFSAR supplement to be completed prior to the period of extended operation.
- The third license condition requires that all capsules in the reactor vessel that are removed and tested meet the requirements of American Society for Testing and Materials (ASTM) E 185-82 to the extent practicable for the configuration of the specimens in the capsule. Any changes to the capsule withdrawal schedule, including spare capsules, must be approved by the staff prior to implementation. All capsules placed in storage must be maintained for future insertion. Any changes to storage requirements must be approved by the staff, as required by 10 CFR Part 50, Appendix H.

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Metal Fatigue Reanalysis (continued)

Plant-specific confirmatory EAF analysis (cont.):

- Plant-specific benchmarking calculations on the feedwater nozzle bound the results for the Core Spray and Recirculation outlet nozzles because:
  - More transients
  - More cycles for transients
  - More severe transients
  - Much higher cumulative usage factor (CUF) from previous calculations

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Metal Fatigue Reanalysis (continued)

Preliminary conclusions of EAF analysis:

- The CUFs calculated by existing EAF analysis for the VY feedwater nozzle are conservative
- Calculated CUF for VY feedwater, recirculation outlet, and core spray nozzles are well within code allowable of 1.0 for analyzed transients and cycles
- Fatigue Monitoring Program will ensure that the actual transient cycles remain within the analyzed cycles

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Metal Fatigue Reanalysis

- NRC onsite audit of reanalysis calculations on October 9 and 10, 2007
- Six audit questions added to Question and Answer (Q&A) database
  - Formal response on November 14, 2007
- RAI sent on November 27, 2007
- Response to RAI received on December 11, 2007
- Conference call on December 18, 2007
- Public meeting on January 8, 2008
  - Agreed to submit plant-specific confirmatory environmentally assisted fatigue (EAF) analysis

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Metal Fatigue Reanalysis (continued)

- Plant-specific confirmatory EAF analysis:
  - Performed benchmarking calculations on the VYNPS feedwater using:
    - ✓ Previous axisymmetric finite element model (FEM)
    - ✓ ASME NB-3200 methodology
    - ✓ Previous analyzed transient definitions and cycles
    - ✓ All six stress components (3 direct + 3 shear)
    - ✓ ANSYS computer code
    - ✓ ASME elastic-plastic correction factor applied
    - ✓ Same water chemistry input
    - ✓ Environmental fatigue life correction factor (F_{en}) bounding for each transient pair
    - ✓ Stress intensities corrected for modulus of elasticity (E) values

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Section 3 Conclusion

- Based on its review of the AMRs and AMPs, the staff concludes that the applicant has demonstrated that the effects of aging will be adequately managed such that the SSCs will serve their intended function during the period of extended operation

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Section 4: Time-Limited Aging Analyses

- For TLAA evaluation, applicant must comply with either 10 CFR 54.21(c)(1)(i), (ii), or (iii)
- VY revised LRA to comply with 54.21(c)(1)(iii) in September 17, 2007 letter
  - Using Fatigue Monitoring Program AMP
    - ✓ Consistent with GALL Report X.M1, "Metal Fatigue of reactor Coolant Pressure Boundary"
    - ✓ Corrective Actions element of AMP allows for reanalysis of components to demonstrate limits will not be exceeded during extended period of operation
  - Transmitted results of its reanalysis

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Section 2 Conclusion

- The applicant's scoping and screening methodology consistent with the requirements of 10 CFR 54.4 and 54.21(a)(1)
- SSCs within the scope of license renewal and subject to AMR are consistent with the requirements of 10 CFR 54.4 and 54.21(a)(1)

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Section 3: Aging Management Review Results

Aging Management Programs (AMPs)

- 39 AMPs
  - 10 are NEW programs
    - ✓7 in the original LRA
    - ✓3 added during review
  - 29 are EXISTING programs
  - 21 programs with exceptions and/or enhancements

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Section 2.3 – Scoping and Screening

Turbine Building Scoping

- Regional Inspection findings
  - ➢ Scoping of segments of the service water and diesel fuel oil systems were not in accordance with guidance
- Resolution
  - ➢ VY placed fluid system components within the Turbine Building within scope
    - ✓LRA revised to add new "Summary of Aging Management Evaluation" Tables
    - ✓LRA revised to add to or delete from existing evaluation Tables
    - ✓LRA revised to add new "Components Subject to AMR" Tables
    - ✓LRA revised to add to or delete from existing AMR Tables

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Section 2.3 – Scoping and Screening

Cooling Tower Scoping

- Operational Event
  - ➢ August 21, 2007 partial collapse of cooling tower No. 2, cell No. 4 (CT 2-4)
- August 29, 2007 issued an RAI asking applicant to verify whether affected cells should be in-scope and whether scoping had been appropriately done
- Resolution
  - ➢ CT 2-1, CT 2-2, and CT 2 deep basin meet criteria of 10 CFR 54.4(a)
  - ➢ CT 2-3 through CT 2-11 do not meet criteria of 10 CFR 54.4(a)

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Section 2.3 – Scoping and Screening

Mechanical Systems (continued)

- Confirmatory Item 2.3.3.2a-1
  - Verify the location of the license renewal scope boundary for pipe section 2"-SW-566C (which is included in the nonsafety-related portion of the Service Water System).
- Resolution
  - Located in reactor building
  - In-scope for potential spatial interaction with safety-related systems

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Section 2.3 – Scoping and Screening

Mechanical Systems (continued)

- Confirmatory Item 2.3.3.2a-2
  - Verify that portions of the nonsafety-related piping, which is attached to safety-related piping, are included up to the first seismic or equivalent anchor of the Service Water System.
- Resolution
  - All nonsafety-related portions of Service Water System attached to safety-related systems are included up to first seismic or equivalent anchor and in-scope
  - Additional components added to LRA due to spatial impact in turbine building

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Section 2.3 – Scoping and Screening

Mechanical Systems

- Confirmatory Item 2.3.3.13e-1
  - Verify if all components subject to an AMR for the Circulating Water (CW) System were included in the LRA
- Resolution
  - Any nonsafety-related portion of CW system in a building containing safety-related components is in-scope
  - Additional components added to LRA due to spatial impact in turbine building

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Section 2.3 – Scoping and Screening

Mechanical Systems (continued)

- Confirmatory Item 2.3.3.13m-1
  - Verify if all components subject to an AMR for the Reactor Water Cleanup System were included in the LRA
- Resolution
  - Any nonsafety-related portion of Reactor Water Cleanup System in a building containing safety-related components is in-scope
  - No additional components added to LRA

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Regional Inspection

- Inspection noted Weaknesses
  - Turbine Building scoping analysis missed nonsafety affects safety components
  - Containment Management had an inconsistent monitoring program
  - Fire Water System lacked corrosion monitoring and biofouling management

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Regional Inspection

- CONCLUSION
  - The inspection team concluded the screening and scoping of non-safety related systems, structures, and components, was implemented as required by the rule and the aging management portions of the license renewal activities were conducted as described in the application.

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License Renewal Inspections

Michael Modes

Region I Inspection Team Leader

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Regional Inspection

- Two Weeks on Site
  - 10 CFR 54.2(a) One inspector week
  - 19 Aging Management Programs 12 inspector weeks
- One Week at Beginning of Outage
  - Confirmatory Inspection of internal base sill seal
  - Confirmatory Inspection of drywell condition
  - Follow Up on Torus Ultrasonic Testing

6

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

Overview

Recap of June 2007 sub-committee meeting

- 386 Audit Questions
- 85 RAIs Issued
- Safety Evaluation Report with Confirmatory Items (SER) was issued March 30, 2007
  - Zero (0) Open Items
  - Six (6) Confirmatory Items
- Three (3) License Conditions

3

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

Overview (continued)

Subsequent to sub-committee meeting

- Resolution of Confirmatory Items
- 6 additional Audit Questions
  - 392 total
- 3 additional RAIs issued
  - 87 total
- One unresolved item
  - Adequacy of environmental fatigue calculations

4

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

Advisory Committee on Reactor Safeguards (ACRS) License Renewal Full Committee

Vermont Yankee Nuclear Power Station
Safety Evaluation Report

February 7, 2008

Jonathan Rowley, Project Manager
Office of Nuclear Reactor Regulation

1

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

Introduction

- Overview
- License Renewal Inspections
- Section 2: Scoping and Screening Review
- Section 3: Aging Management Review Results
- Section 4: Time-Limited Aging Analyses (TLAAs)

2

Chemistry Effects on EAF

- Fen factors per NUREG/CR-6583 and NUREG/CR-5704
- Inputs selected to maximize contribution
  - Sulfur content maximum of 0.015 used
  - Strain rate less than 0.001%/sec for all transients
  - Temperature – 550°F used for all locations
  - Dissolved oxygen (DO) based on Plant measured data including excursions
  - DO inputs represent mean plus 1 standard deviation of measured data

Entergy

VY Plant Specific EAF Calculations

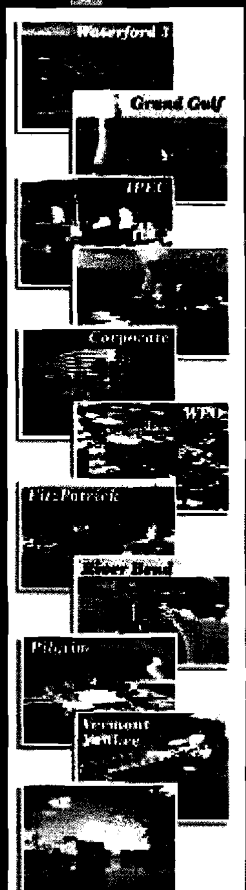
- Locations identified in NUREG/CR-6260
- EAF relationships (Fen) from NUREG/CR-6583 and NUREG/CR-5704
- Design transients used
- Cycles projected to 60 years
- Existing analysis RPV Shell and Lower Head & RR Inlets Nozzle per NB3200
- New Analysis for FDW, RR Outlet , & CS nozzle using VY analysis approach
- New Class I Piping Analyses for RR/RHR Piping & Feedwater Piping per NB 3600

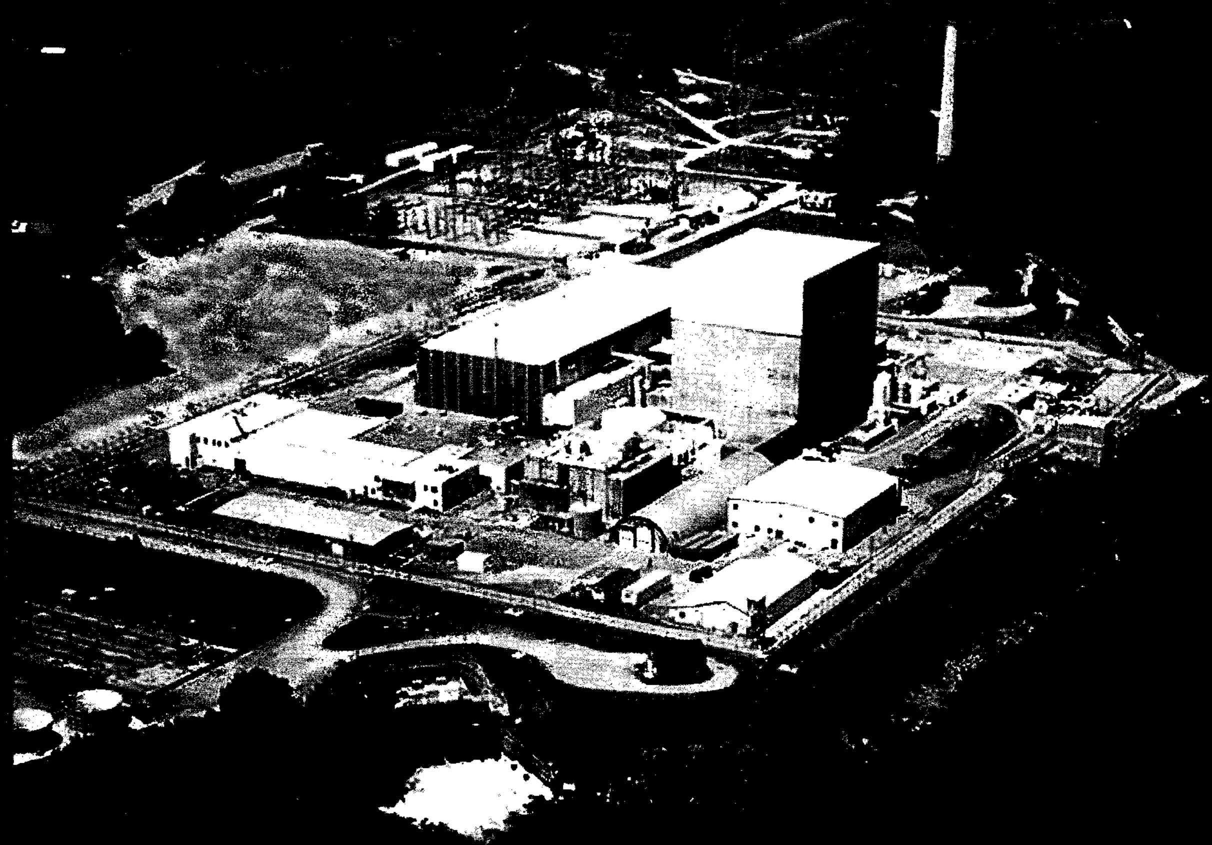
Entergy

Confirmatory Calculation

- Feedwater nozzle analyzed
- Controlling nozzle most severe design transients and fatigue usage
- Same FEM model, transients, cycles and water chemistry used
- All 6 stress components combined per NB-3216.2
- ASME fatigue analysis per NB-3222.4(e) same as existing
- Fen factors bounding for each transient pair
- Results confirm adequacy of existing VY analysis approach

Entergy


Waterford 3
Grand Gulf
IPEC
Corporate
WPO
FitzPatrick
River Bend
Pilgrim
Vermont Yankee



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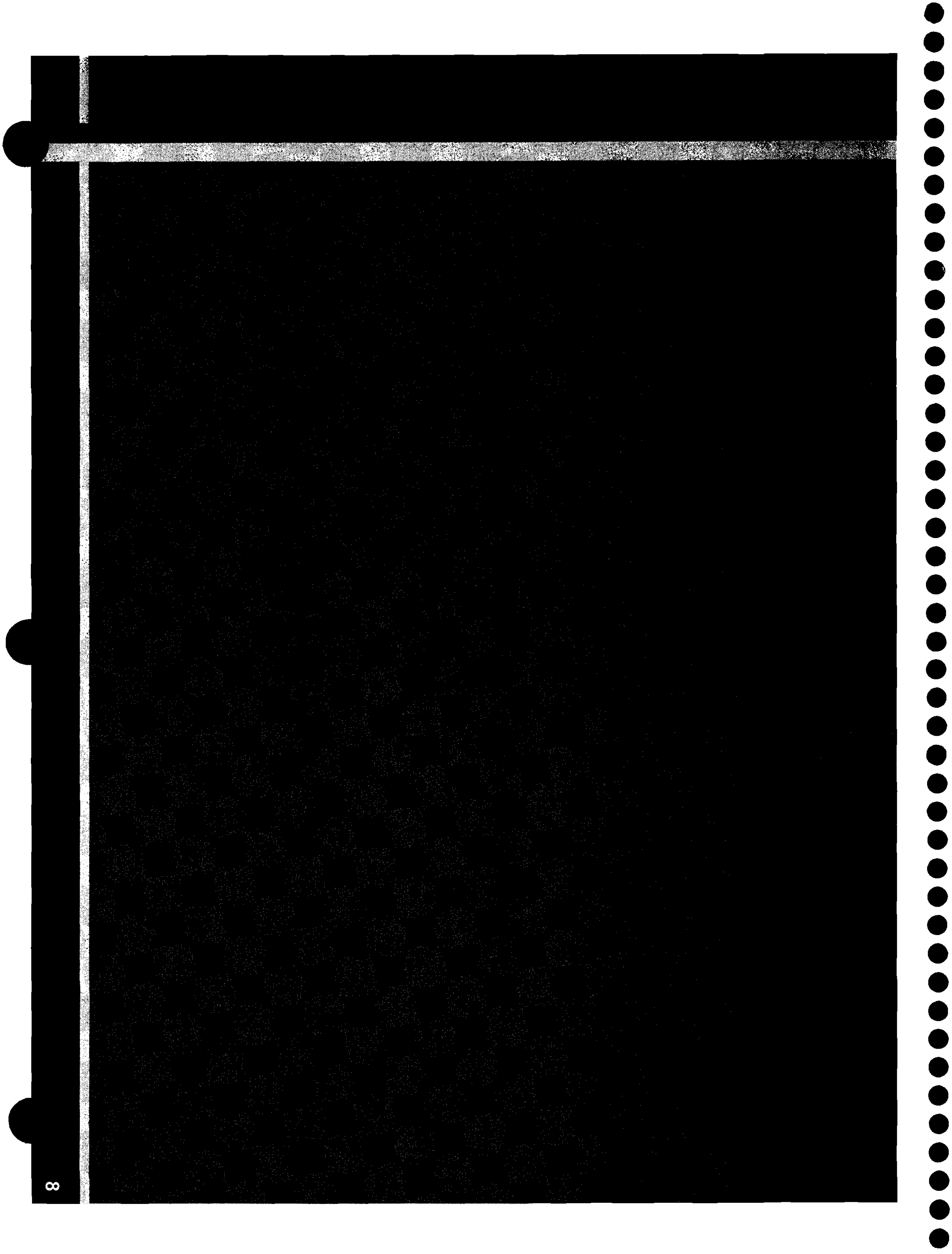

8

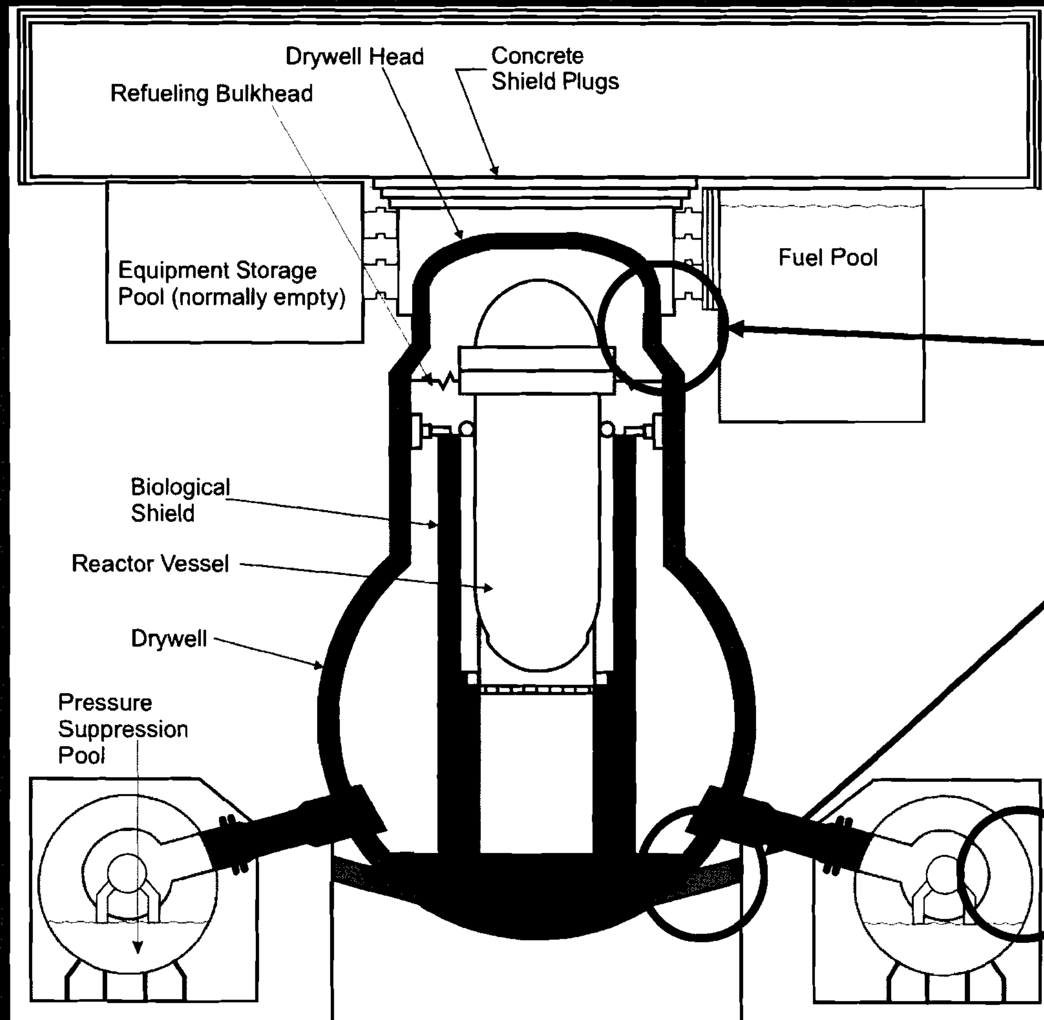
6

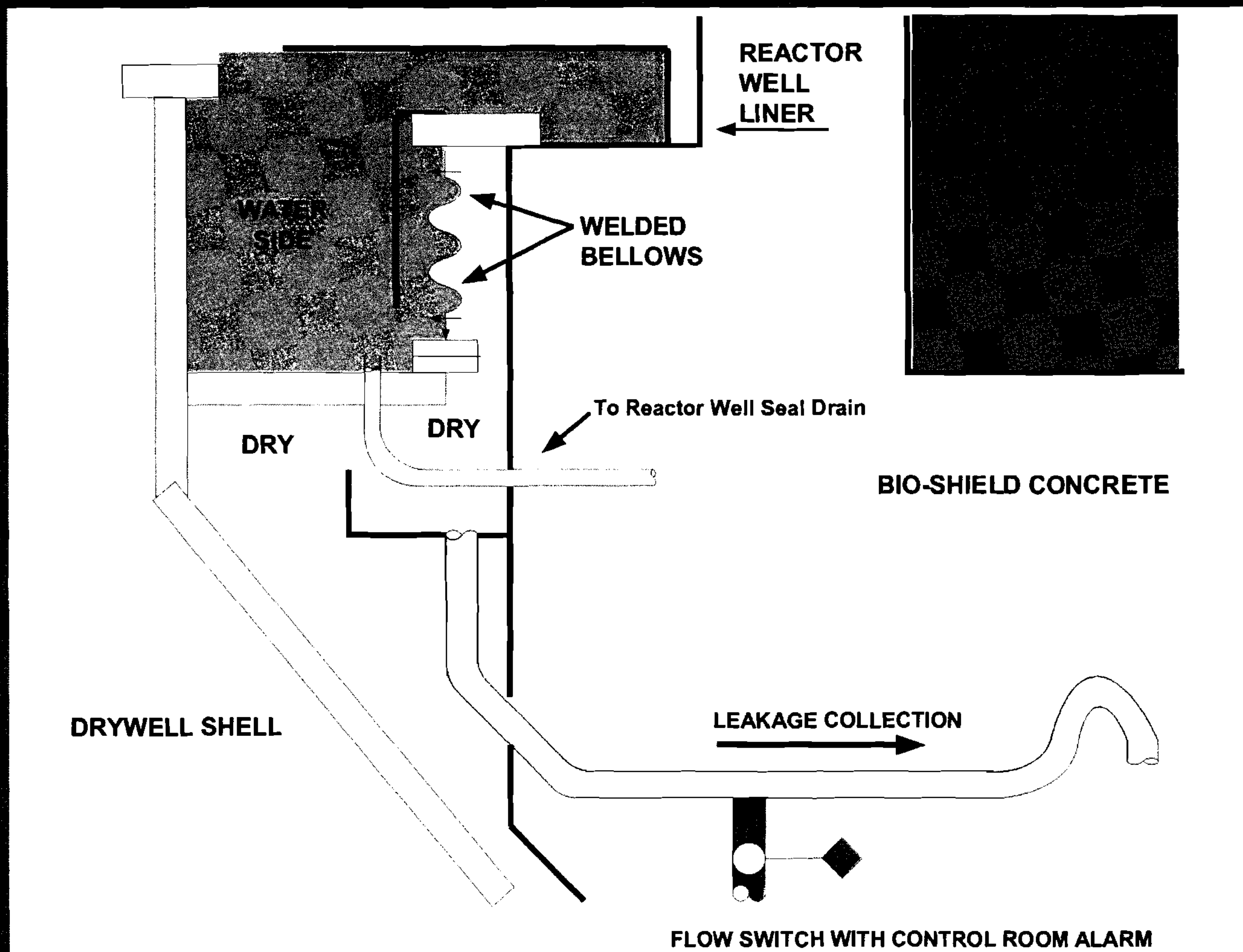
11

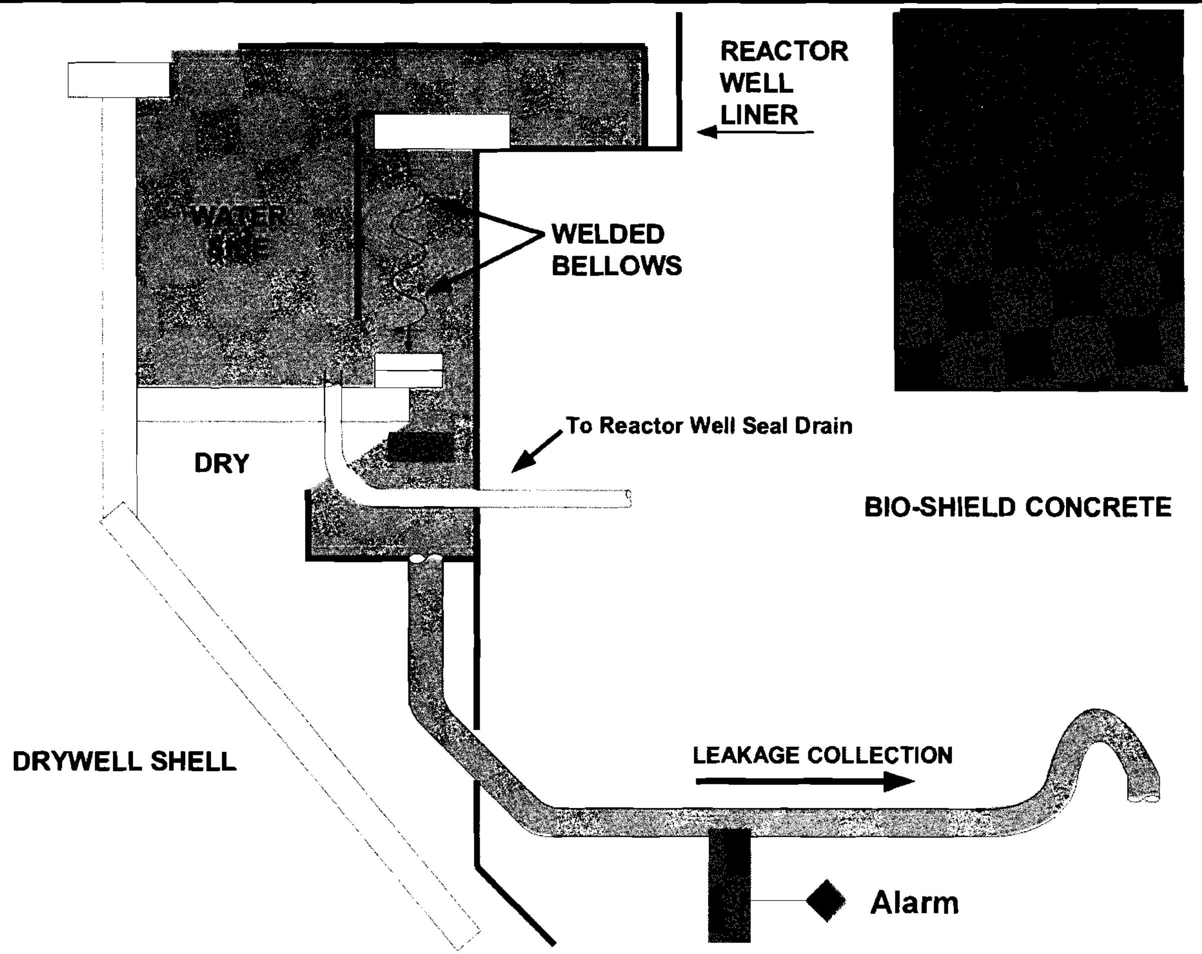
15

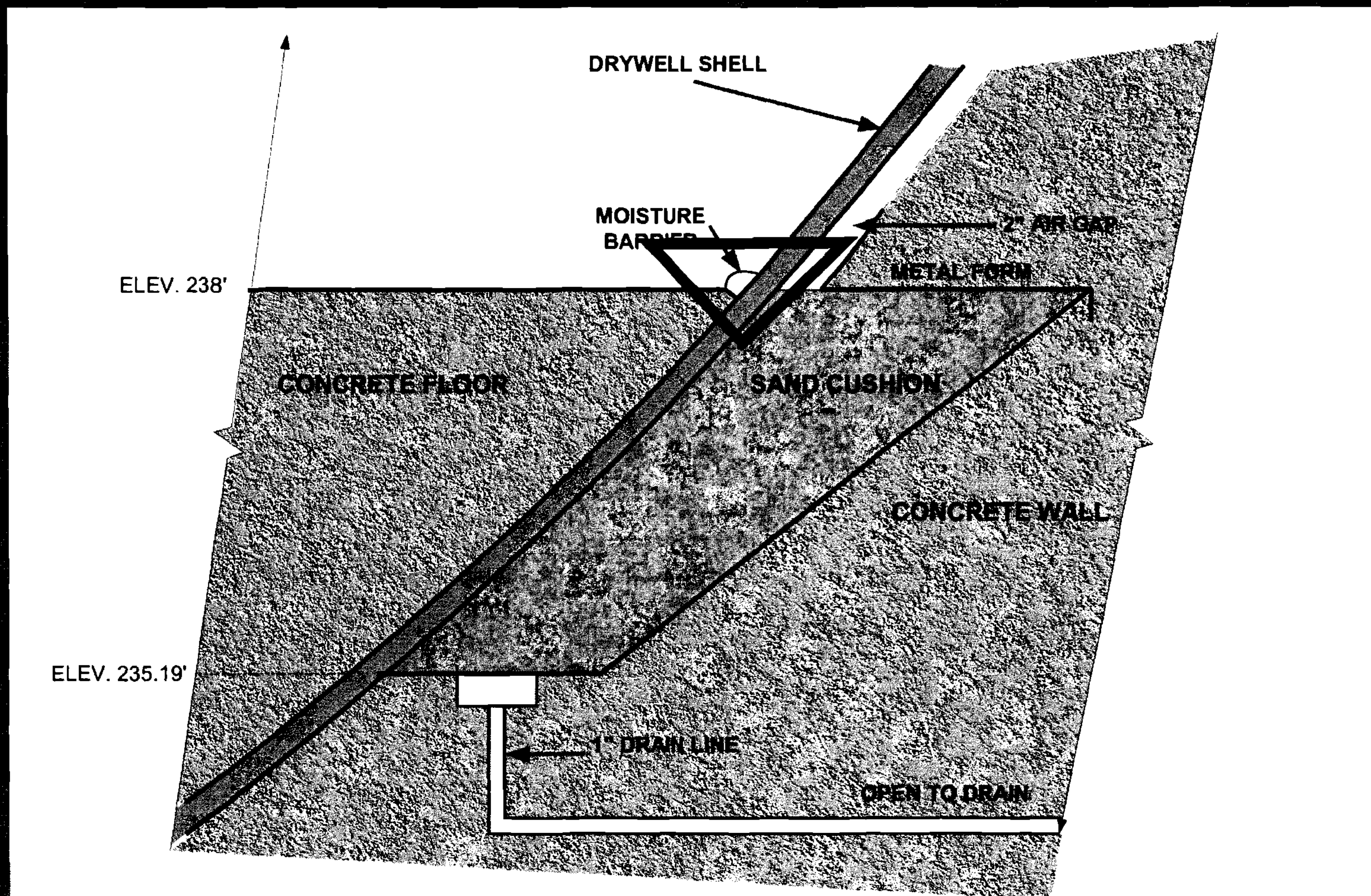
17

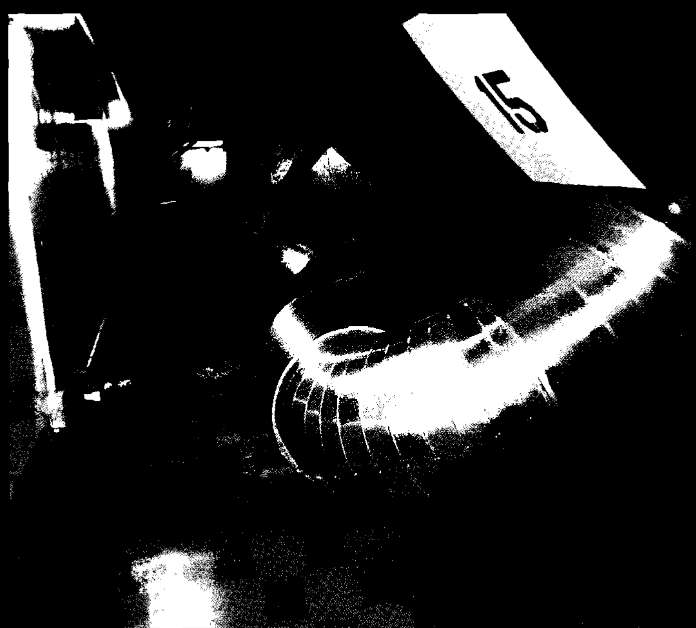
Location	Analysis	EAF CUF / Allowable
Safe End	EAF Analysis	**0.2560 / 1.0000**
	Confirmatory Analysis	**0.0994 / 1.0000**
Nozzle Corner (Blend Radius)	EAF Analysis	**0.6392 / 1.0000**
	Confirmatory Analysis	**0.3531 / 1.0000**


Drywell Head
Concrete Shield Plugs
Refueling Bulkhead
Equipment Storage Pool (normally empty)
Fuel Pool
Biological Shield
Reactor Vessel
Drywell
Pressure Suppression Pool


REACTOR
WELL
LINER
WATER
SIDE
WELDED
BELLOWS
To Reactor Well Seal Drain
DRY
DRY
BIO-SHIELD CONCRETE
DRYWELL SHELL
LEAKAGE COLLECTION
FLOW SWITCH WITH CONTROL ROOM ALARM


REACTOR
WELL
LINER
WELDED
BELLOWS
To Reactor Well Seal Drain
DRY
BIO-SHIELD CONCRETE
DRYWELL SHELL
LEAKAGE COLLECTION
Alarm


DRYWELL SHELL
MOISTURE BARRIER
2" AIR GAP
METAL FORM
ELEV. 238'
CONCRETE FLOOR
SAND CUSHION
CONCRETE WALL
ELEV. 235.19'
1" DRAIN LINE
OPEN TO DRAIN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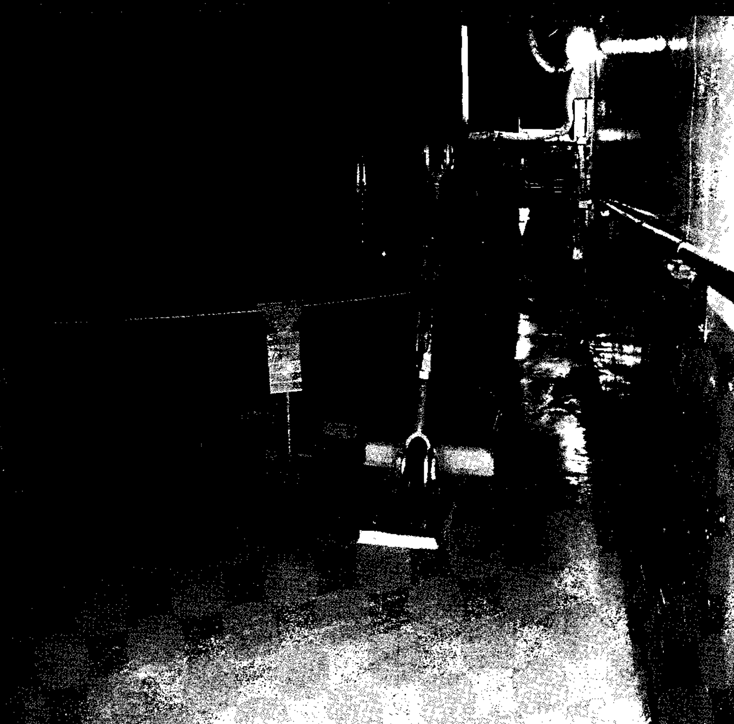
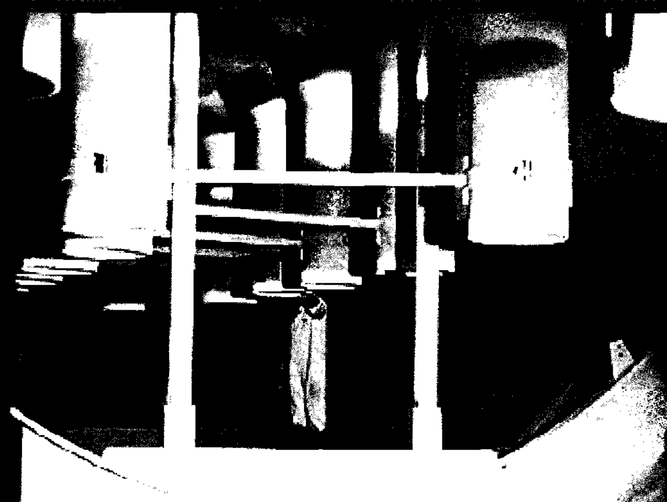
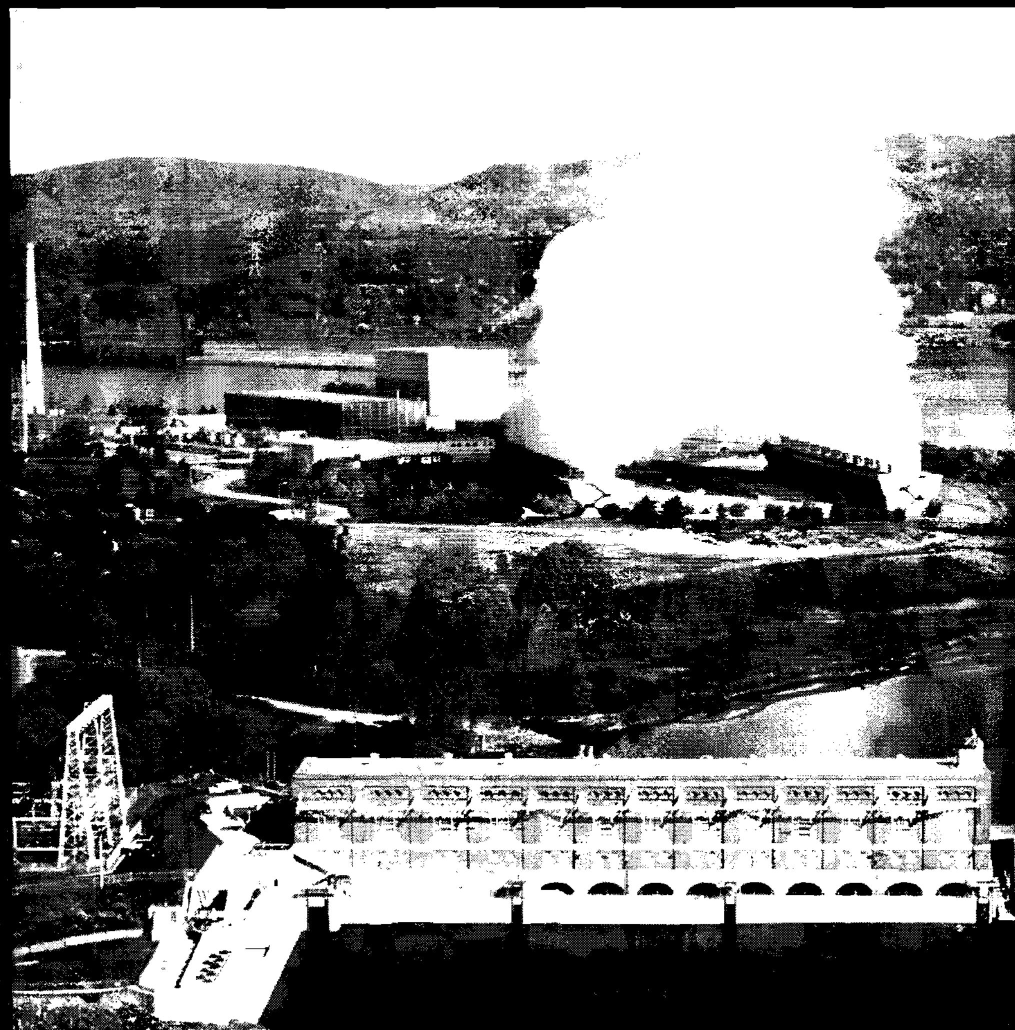
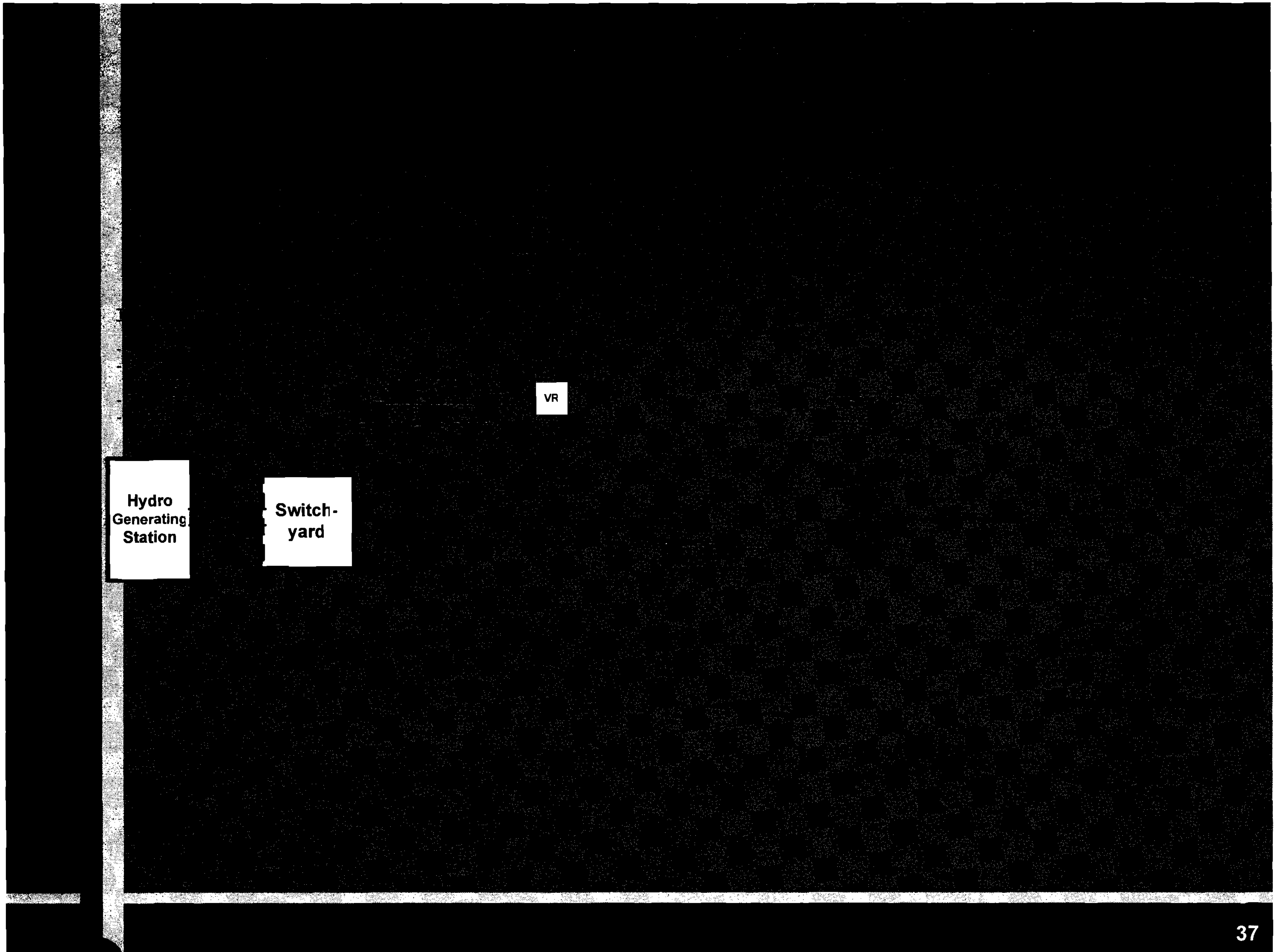
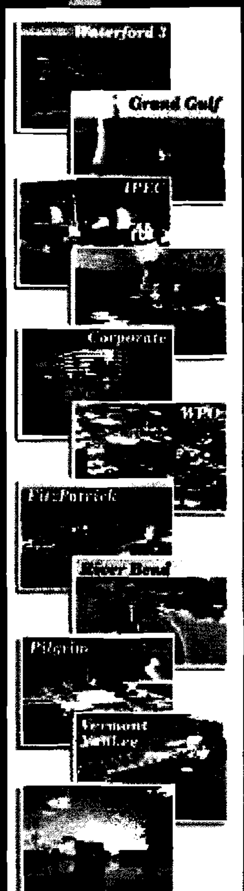
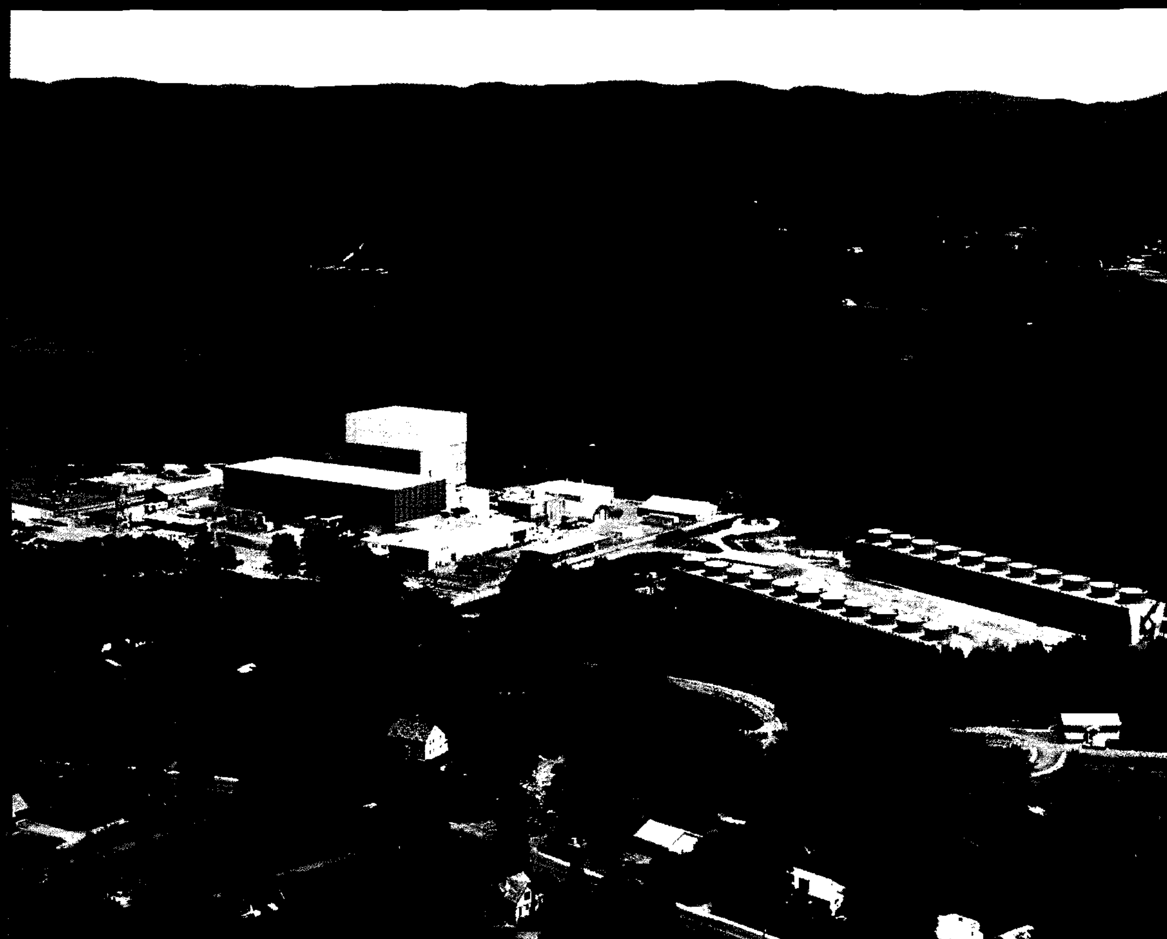







Hydro Generating Station
Switch-yard
VR


Waterford 3
Grand Gulf
IPEC
Corporate
WPO
FitzPatrick
River Bend
Pilgrim
Vermont Yankee



Background

- The vessel head corrosion incident at Davis Besse (2002) caused NRC to re-visit requirements pertaining to leakage detection
  - Evaluate changes needed in regulatory positions
  - Examine addition of new regulatory positions
- RES issued NUREG/CR-6861, "Barrier Integrity Research Program: Final Report" (ML043580207)
  - Leakage detection could be improved.
  - Low levels of leakage at localized areas could be detected by modern techniques; such monitoring may provide the opportunity for corrective actions to be taken early thus avoiding boric acid corrosion.
  - Leakage limits will not ensure structural integrity of all components in the reactor cooling system; leakage rates less than the technical specification limit can result in high corrosion rates depending on the actual conditions associated with the leak (temperature of metal, leakage rate, resultant temperature of the boric acid solution, and the availability of oxygen).
  - Lowering the technical specification leakage limits may increase the number of plant shutdowns, inspections, and personal exposure.
  - Reductions in the coolant activity over the years has limited the usefulness of gaseous reactivity monitoring systems.

Staff Review and Analysis

- With the exception of Davis-Besse, the corrosion of the vessel head at other plants, if any, has not been significant.
- NRC order (EA-03-009) was issued in response to Davis-Besse to minimize the likelihood of developing structurally significant cracks in the vessel head penetration; consequently the likelihood of vessel head corrosion is also minimized.
- Effectiveness of existing inspection and monitoring programs provide adequate protection; substantial increase in safety will not result from a change in leakage detection capability limits or leakage detection systems.

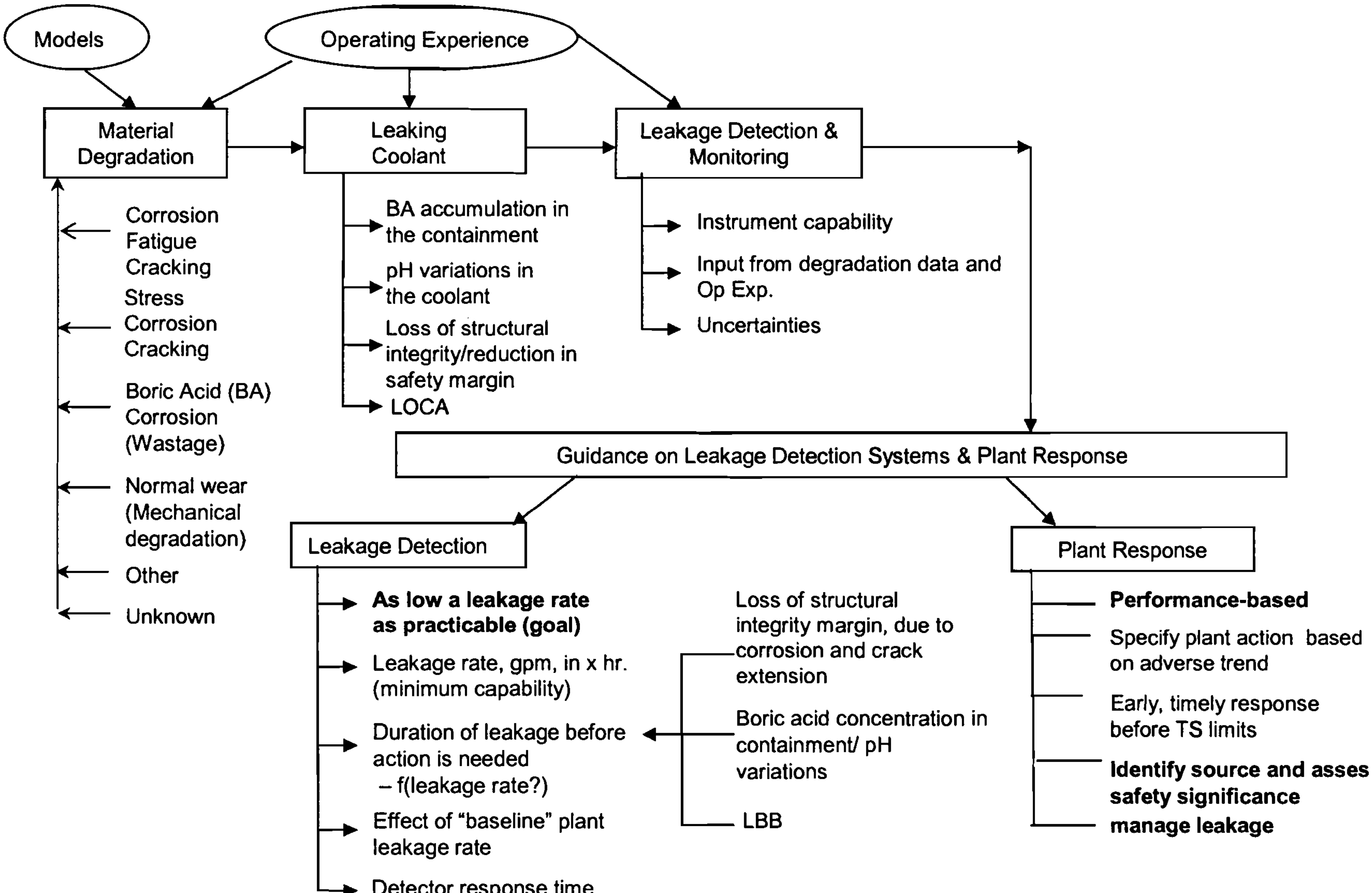
→ Issue a revision to leakage monitoring Regulatory Guide 1.45.

(Preferably, a performance-based and not prescriptive approach)

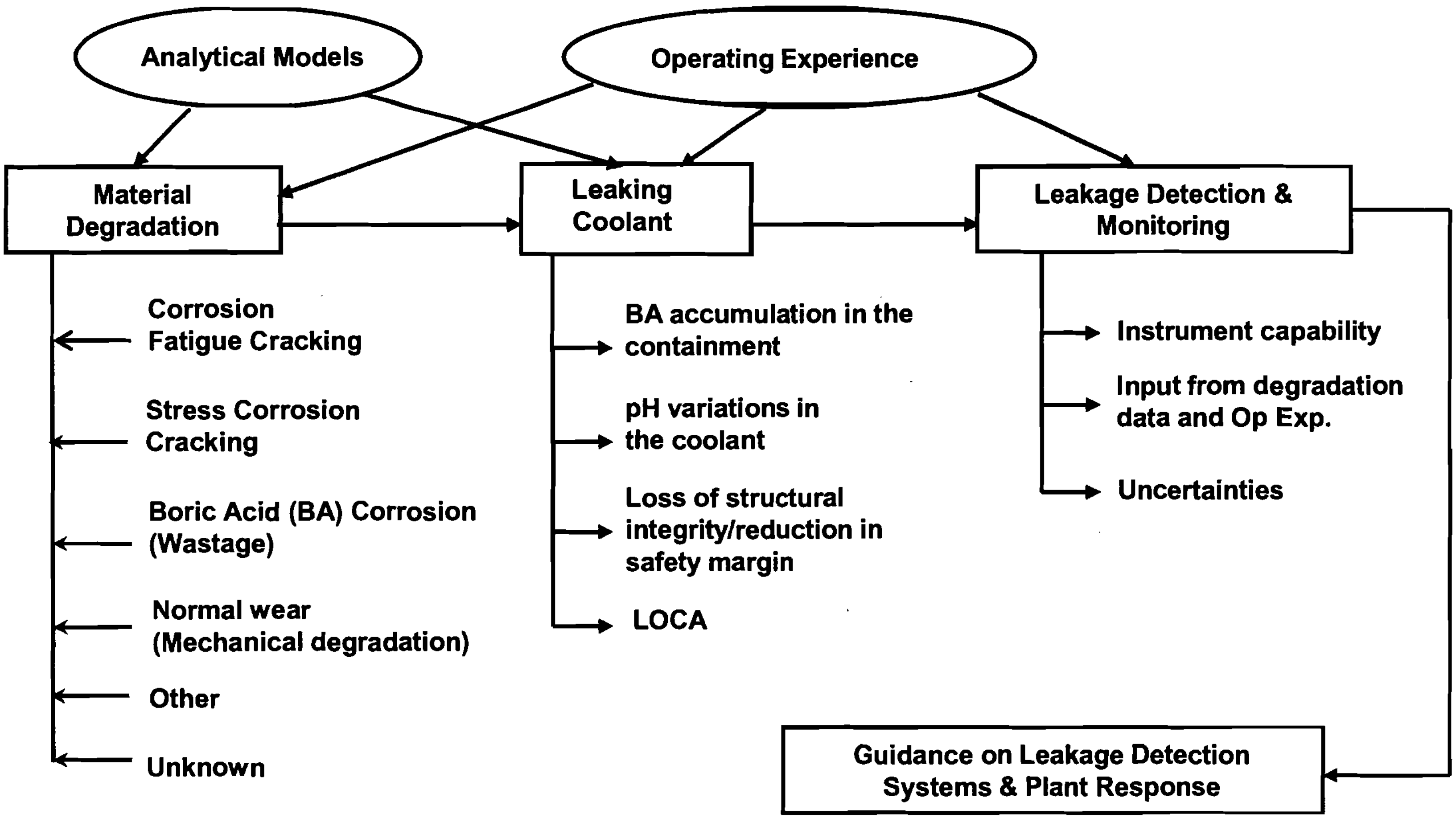
Safety Concerns/Issues

1. **Low-level leakage for a long period of time is a potential safety concern.**
   - Stress corrosion cracking may result in a loss of structural integrity at low leak rates.
   - Leakage can affect the integrity of nearby components by promoting corrosion.
   - Leakage can affect the sensitivity of other instruments or mask other leaks (high background leakage may mask a smaller, more significant leak).
   - Leakage can result in accumulation of chemical compounds (e.g., boric acid) which may affect other systems (e.g., accumulation of boric acid in containment could challenge the ability to maintain the pH of the ECCS sump following a LOCA).
2. **Leakage monitoring is necessary for the application of LBB.**
3. **Risk-informed Emergency Core Cooling System (ECCS) rulemaking (i.e., 10 CFR 50.46) considers the effect of leakage monitoring.**

Elements of performance-based guidance on reactor coolant leakage monitoring and response in LWRs



Elements of performance-based guidance on reactor coolant leakage monitoring and response in LWRs



Elements of performance-based guidance on reactor coolant leakage monitoring and response in LWRs

Guidance on Leakage Detection Systems & Plant Response

Leakage Detection

- As low a leakage rate as practicable (goal)
- Leakage rate, gpm, in x hr. (minimum capability)
- Duration of leakage before action is needed – f(leakage rate?)
  - Loss of structural integrity margin, due to corrosion and crack extension
  - Boric acid concentration in containment/ pH variations
  - LBB
- Effect of "baseline" plant leakage rate
- Detector response time

Plant Response

- Performance-based
- Specify plant action based on adverse trend
- Early, timely response before TS limits
- Identify source and assess safety significance
- Manage leakage

Leakage Monitoring Addressing Safety Concerns

1. Industry Developing Standard Guidelines for Response to Low-Level Detected Leakage (9/29/05 meeting with NRC, ML052760006)
2. Existing PWR Leakage Detection Systems Capable of Detecting Leaks Below 0.1 gpm
3. Key Issue is Duration of Leakage; Technical Specifications Allow Indefinite Period of Unidentified Leakage Below 1.0 gpm.
4. Revised RG 1.45 Provides Requirements on Monitoring and Plant Response to Leakage:
   - timely identification of the source of leakage,
   - trending plant leakage rate data, and
   - specifying plant action to manage leakage, following the confirmation of any adverse trend in leakage rate.

RG 1.45, Rev 1.

- Title changed to "Guidance on Monitoring and Responding to Reactor Coolant System Leakage"
- Regulatory Position – Newly Categorized as:
  - General Positions – Five (5)
  - Leakage Monitoring-Related Positions – Six (6)
  - Operations-Related Positions – Four (4)
  - Technical Specification Position – One (1)

RG 1.45, Rev 1.
General Positions

(1) The source and location of reactor coolant leakage should be identifiable to the extent practical, and the plant should measure the leakage rate.

(2) Plant should collect or otherwise isolate leakage to the primary reactor containment from identified sources so that the following criteria are fulfilled:

   (a) flow rates from identified sources are monitored separately from the flow rates from unidentified sources.

   (b) plant can establish and monitor flow rate

(3) Plant should monitor critical components of the RCPB for leaks.

(4) Plant should monitor intersystem leakage for systems connected to the RCPB.

(5) The capabilities of the leakage monitoring systems should be known. In addition, the capabilities should ensure effective management of leakage.

RG 1.45, Rev 1.
Leakage Monitoring-Related Positions

(6) Plant procedures should include collection of leakage to the primary reactor containment from unidentified sources so that the total flow rate can be detected, monitored, and quantified for flow rates greater than or equal to 0.05 gal/min (0.19 L/min).

(7) Plant should use leakage detection systems with a response time (not including the transport delay time) of 1 hour or better for a leakage rate of 1 gal/min (3.8 L/min).

RG 1.45, Rev 1.
Leakage Monitoring-Related Positions

(8) Plant technical specifications should identify at least two independent and diverse instruments and/or methods that have the detection and monitoring capabilities detailed above. The methods to consider for incorporation in the technical specifications include, but are not limited to, the following:

(a) monitoring sump level or flow
(b) monitoring airborne particulate radioactivity
(c) monitoring condensate flow rate from air coolers

In addition to the monitoring systems detailed in the technical specifications, plant should use other systems to detect and monitor for leakage, even if they do not have the capabilities specified in regulatory position 7. These supplemental instruments/methods may include, but are not limited to, the following:

(a) monitoring airborne gaseous radioactivity
(b) monitoring humidity of the containment
(c) monitoring temperature of the containment
(d) monitoring pressure of the containment
(e) monitoring acoustic emission
(f) conducting video surveillance

RG 1.45, Rev 1.
Leakage Monitoring-Related Positions

(9) At least one of the leakage monitoring systems required by the plant technical specifications (as described in Regulatory Position 8 above) should be capable of performing its function(s) following any seismic event that does not require plant shutdown.

(10) The leakage monitoring systems, including those with location detection capability, should have provisions to permit calibration and testing during plant operation to ensure functionality or operability, as appropriate.

(11) Plant should periodically analyze the trend in the unidentified and identified leakage rates. When the leakage rate increases noticeably from the baseline leakage rate, the plant should evaluate the safety significance of the leak. The plant should determine the rate of increase in the leakage to verify that plant actions can be taken before the plant exceeds technical specification limits.

RG 1.45, Rev 1. Operations-Related Positions

(12) Plant should establish procedures for responding to leakage. These procedures should address the following considerations and should ensure that no adverse safety consequences result from the leakage:

(a) Plant procedures should specify operator actions in response to leakage rates less than the limits set forth in the plant technical specifications. The procedures should include actions for confirming the existence of a leak, identifying its source, increasing the frequency of monitoring and verifying the leakage rate (through a water inventory balance), responding to trends in the leakage rate, performing a walkdown outside containment, planning a containment entry, adjusting alarm setpoints, limiting the amount of time that operation is permitted when the sources of the leakage are unknown, and determining the safety significance of the leakage.

(b) Plant procedures should specify the amount of time the leakage detection and monitoring instruments (other than those required by technical specifications) may be out of service to ensure that the leakage rate is effectively monitored during all phases of plant operation (i.e., hot shutdown, hot standby, startup, transients, and power operation).

RG 1.45, Rev 1.
Operations-Related Positions

(13) **Plant should provide should provide output and alarms from leakage monitoring systems in the main control room . Procedures for converting the instrument output to a leakage rate should be readily available to the operators. (Alternatively, these procedures could be part of a computer program so that the operators have a real-time indication of the leakage rate as determined from the output of these monitors.) Periodic calibration and testing of leakage monitoring systems should take place. The alarm should provide operators an early warning signal so that they can take corrective actions, as discussed in Regulatory Position 12 above.**

(14) **During maintenance and refueling outages, plant should take actions to identify the source of any unidentified leakage that was detected during plant operation. In addition, corrective action should take place to eliminate the condition resulting in the leakage.**

RG 1.45, Rev 1. Technical Specification Position

(15) Plant technical specifications should include the limiting conditions for identified, unidentified, RCPB, and intersystem leakage, and they should address the availability of various types of instruments to ensure adequate coverage during all phases of plant operation (not including cold shutdown and refueling modes of operation).

PUBLIC COMMENTS ON DG-1173

AND THEIR DISPOSITION IN RG 1.45 REV. 1

DG-1173 Public Comments

Two (2) from Nuclear Energy Institute (NEI), three (3) from AREVA NP, and five (5) from Strategic Teaming and Resources Sharing (STARS)

NEI Comments

Comment	Disposition
Use of indirect methods to monitor leakage from critical components.	The staff agreed that indirect methods of leakage detection to monitor critical components may be used as long as risk-significance can be assessed.
Consideration of Inspection Manual Chapter (IMC) 2515, Attachment 1 for RG revision	IMC 2515 was considered during the revision of RG 1.45. It was decided not to incorporate this reference into the RG because: (1) the guidance in IMC 2515 may not always be conservative, (2) the guidance in IMC 2515 may be too restrictive in some instances, and (3) the IMC may change more frequently than the RG.

DG-1173 Public Comments

AREVA NP Comments

Comment	Disposition
Regulatory Position 8, 1st paragraph, sub-step (a) should be revised to state "monitoring sump level or flow" to be consistent with the specification of "two" in the first paragraph. It is assumed that Regulatory Position 8, 1st paragraph, sub-step (b) is the second required method.	The staff agreed with this comment and made the revision in the RG. The staff also included monitoring condensate flow as item (c). The staff notes that sub-steps (a), (b) and (c) in Regulatory Position 8 are not intended to limit the licensee to these specific methods. These are examples of methods commonly chosen by many licensees previously.
Draft RP 9 contains two separate RPs. Recommend that these be two separate RPs.	The staff agreed with this comment. The RG has been revised to retain the first sentence as a regulatory position. The second sentence has been deleted (see disposition of the next AREVA comment below).
Draft RP 9: With respect to leakage monitoring capability for leak-before-break (LBB) monitoring: Recommend that the capability guidance for the LBB detection system be revised to be clear that it does not necessarily have to be able to detect the leakage determined from the LBB analysis within 1 hour. Rather, AREVA NP believes that the detection capability should be addressed in plant procedures and would be based on the type of detection system and its location.	The staff has withdrawn the proposed staff position 9, second sentence in DG-1173. When a LBB analysis is submitted for the plant, the staff evaluates the LBB analysis procedures of the licensee or the applicant as per the guidance provided in Standard Review Plan (SRP) 3.6.3 to ensure that such analysis incorporates the provisions of leakage monitoring as per this regulatory guidance. Thus, there is no need for a staff position on leakage monitoring, specific to LBB.

DG-1173 Public Comments

STARS Comments

Comment	Disposition
On leakage into containment: Steam leakage to containment atmosphere in pressurized water reactors can be predominately secondary steam leakage. In current designs, leakage collected in the containment sump cannot be directly correlated to primary "unidentified leakage" without sampling.	The staff agreed with the comment, and added the following sentences to the Regulatory Guide: "It is important to note that there may be leakage into the containment from systems other than the RCS (e.g., secondary side steam leakage in a pressurized water reactor). This non-RCS leakage may increase the unidentified leakage rate. Chemical analysis of samples of the unidentified leakage may provide an indication of whether the unidentified leakage is from the RCS or from other sources."
On RP 6: RCS inventory balance is the current method used to calculate RCS leak rate however, the current equipment installed in some plants may not be sensitive enough to accurately measure an RCS leak rate of 0.05 gpm. While RCS leakage is collected in the containment sumps, the sumps would not be sensitive to an inflow of 0.05 gpm, especially in the early stages of a small RCS leak when most of the hot coolant (steam) would be present in the containment atmosphere.	Although implementation of this guide may provide a safety benefit for current operating plants, it was not intended to be applicable to currently operating plants (since evaluations in response to the lessons learned from the Davis-Besse vessel head degradation indicated that such changes could not be justified). However, for plants licensed after the issuance of this revision to the guide, it is the staff's position that the leakage monitoring system would be capable of detecting a 0.05 gpm leak given the potential safety significance of low levels of leakage. Such monitoring capability should be achievable using current instrumentation and monitoring methods.

DG-1173 Public Comments

STARS Comments

Comment	Disposition
On leakage into containment: The draft RG stated that methods that monitor air temperature and pressure may also be used to infer leakage of the coolant to the containment. STARS commented that such methods are applicable to large leaks only.	The staff agreed with the comment, and revised the text to clarify that these methods can only detect large leaks.
The draft regulatory positions 14 and 15 leads the reader to believe that the NRC expects licensees to monitor RCS leakage during refueling outages. RCS operational leakage requirements in MODE 5 and 6 are currently not required because the reactor coolant pressure is far lower, resulting in lower stresses and a reduced potential for leakage. Regulatory positions 14 and 15 either need further clarification and justification or they should be deleted. An explanation of acceptable leakage monitoring methods during refueling outages needs to be included if justification can be made for refueling outage monitoring.	The staff agreed that the RCS operational leakage requirements in MODE 5 and 6 are not required. Positions 14 and 15 were appropriately clarified.

DG-1173 Public Comments

STARS Comments

Comment	Disposition
The concluding paragraph of the Regulatory Analysis Section of the Draft Guide implied that current licensees will automatically adopt the latest revision of the regulatory guide. In order to adopt the guide without exception, licensees would need to upgrade their equipment. Therefore, for many licensees adopting the revised regulatory guide would not be practical.	RG 1.45 Rev 1 will be referenced in the Standard Review Plan and will be applicable only to new reactors (per the requirements of 10 CFR 50.34(h)). No backfitting is intended or approved in connection with the issuance of RG 1.45, Rev 1.

4

Advisory Committee on Reactor Safeguards
Meeting On Next Generation Nuclear Plant
Licensing Strategy

February 7, 2008
Rockville, MD

- DRAFT AGENDA-

FULL COMMITTEE MEETING – FEBRUARY 7, 2008

Topics	Presenters	Time
Opening Remarks	M. Corradini, ACRS	1:00 pm - 1:05 pm
Staff Introduction	J. Jolicoeur, RES	1:05 pm - 1:10 pm
NGNP Design and Technology	T. Cook, DOE	1:10 pm - 1:30 am
NGNP Licensing Strategy & NRC Needs for Analytical Tools and R&D	S. Basu, RES T. Kenyon, NRO	1:30 pm - 2:15 pm
Subcommittee Discussion	M. Corradini, ACRS	2:15 pm - 2:45 pm
Closing Remarks	M. Corradini, ACRS	2:45 pm – 3:00 pm

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.

Sidhamay
Basu
Jolicuer
Tsh
IRES

5

NGNP Licensing Strategy

Briefing for ACRS

Sud Basu, RES

Tom Kenyon, NRO

February 7, 2008

NGNP – A Congressional Mandate

- Energy Policy Act 2005 (P.L. 109-58, Subtitle C)
  - Sec. 641(a)
    - The Secretary (of Energy) shall establish a project to be known as the *"Next Generation Nuclear Plant Project"*
  - Sec. 644(a)
    - The NRC shall have licensing and regulatory authority for any reactor authorized under this subtitle
  - Sec. 645(c)
    - Not later than *September 30, 2021*, the Secretary shall complete construction and begin operations of the *prototype* nuclear reactor (NGNP) ...

NGNP Licensing Strategy - Mandate

- Energy Policy Act 2005 (P.L. 109-58, Subtitle C)
  - Sec. 644(b)
    - Not later than 3 years after the enactment of the Act, the Secretary (of Energy) and the Chairman (of NRC) shall jointly submit to the Congress a licensing strategy for the prototype nuclear reactor (NGNP)
  - Licensing Strategy to include
    - Ways in which current licensing requirements for LWRs need to be adapted for a prototype NGNP
    - Description of analytical tools NRC will need
    - Other R&D activities for development of licensing review infrastructure
    - Estimate of resource requirements associated with the licensing strategy

NGNP – Product Description

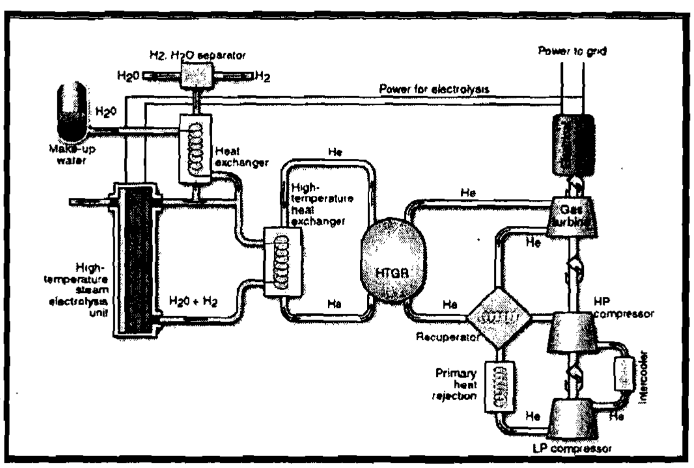
- NGNP Licensing Strategy
  - Licensing approach
  - NRC needs for analytical tools and supporting technical basis
  - Other NRC R&D needs (for licensing review)
  - Resource needs

- Deliverable
  - Licensing Strategy Report to Congress August 7, 2008

NGNP – The Machine

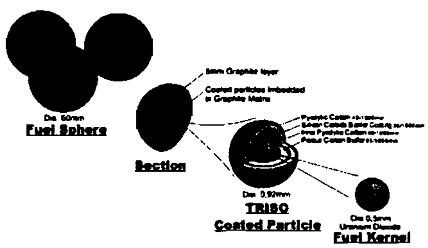
- An advanced reactor concept for nuclear electricity production and hydrogen cogeneration
  - Very high temperature gas-cooled reactor (VHTR)
    - Reactor outlet temperature 900°C and above
  - TRISO coated particle fuel
  - Helium cooled and graphite moderated
  - Coupled hydrogen plant
    - Hydrogen plant power 10% of reactor power
    - Hybrid thermo-chemical or high temperature electrolysis process

NGNP – The Machine



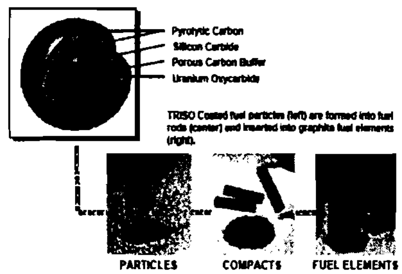
VHTR Fuel Forms

Pebble Bed Reactor



Fuel Sphere

Prismatic Block Reactor



Hexagonal Block w/ Compacts

Licensing Approach

- Licensing options
  - Statutory requirements
  - Process options (Part 50, Part 52)
  - Technical requirements options
    - Deterministic approach
    - Partially risk-informed approach
    - Fully risk-informed approach
    - New body of risk-informed performance-based regulations

Key Technical Needs

- Fuel performance
- High temperature materials and graphite performance
- Core thermal-fluid and neutronics
- Fission products transport and source term
- Evaluation model development and assessment

Potential Policy Issues

- Defense-in-Depth (DiD)
- Use of PRA in the Licensing Process
- Source Term
- Containment Functional Performance

(Many issues identified previously and some deliberated on by the Commission)

NRC Needs for Analytical Tools PIRT Process

- Phenomena Identification and Ranking Table (PIRT) process completed for phenomena relevant to NGNP safety
  - PIRT topical areas
    - Thermal-fluids and accident analysis
    - High temperature materials including graphite
    - Process heat and hydrogen co-generation
    - Fission products transport and consequence
    - TRISO-coated fuels
- Assessment of knowledge base for important phenomena
- Assessment of data gaps and adequacy of analytical tools
- Development needs for analytical tools

Highlights of PIRT Findings

- In thermal-fluids, few phenomena are design-specific and many are generic to HTGRs (VHTRs)
  - Knowledge and data required for development of models and tools for confirmatory analysis
- In high temperature materials and graphite areas, many phenomena are manufacturing/fabrication related; vendors' R&D programs in place or planned
- Very few generic phenomena in process heat area, most are design-specific
- Some issues require longer-term R&D effort (e.g., fuels, fission products transport, codes and methods)

Needs for Analytical Tools

- Confirmatory analysis tools in thermal-fluids (accident analysis), fuel behavior, and fission products transport areas
- Confirmatory tools in materials and structural analysis areas
- Safety analysis tools in process heat applications
- Strategy to modify/adapt existing tools for NGNP applications; supplement with special purpose tools as necessary
- Strategy to utilize tools and data from domestic and international programs to the maximum extent feasible while maintaining independence in analysis

Needs in Other Technical Areas

- Structural failure modeling of concrete at high temperatures
- Instrumentation and control systems for high temperature environment
- High temperature sensor technology
- Human factor issues
- PRA tools – scope, quality, guidance

Other Infrastructure Needs

- Technical basis infrastructure
  - Development of codes and standards
    - Ongoing DOE/ASME/ANS activities
  - Technical basis to support development of tech spec requirements
- Licensing review infrastructure
  - Regulatory guidance
  - Staff training and skill development

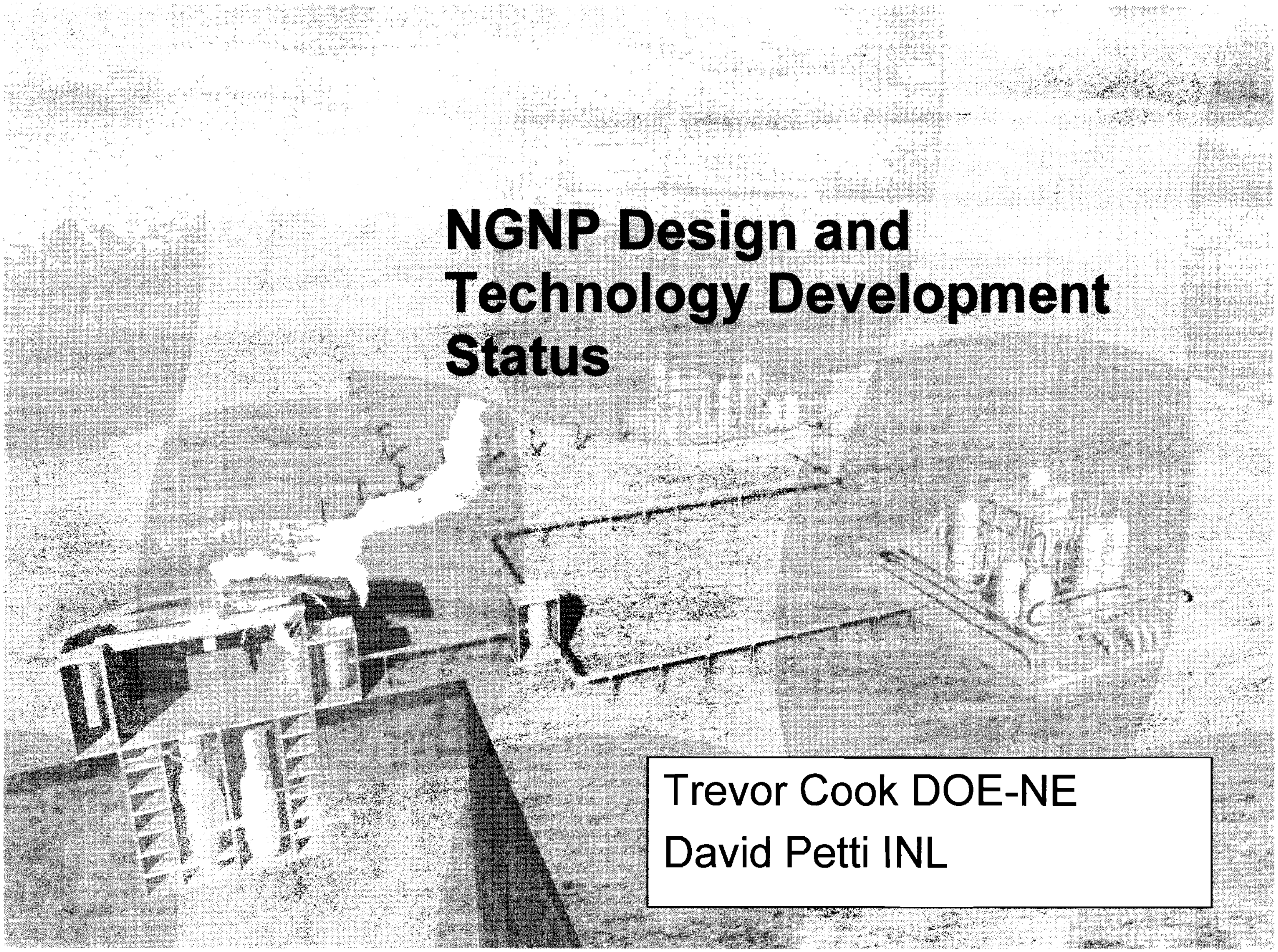
Documentation Status

- Licensing Strategy Report to Congress – due August 7, 2008
- Licensing Strategy Technical Basis Report (NUREG-1902) – work in progress
- PIRT reports (NUREG/CR-6944) in publication
- PIRT report (NUREG/CR-6844) on HTGR fuel -- published July 2004

Next Step

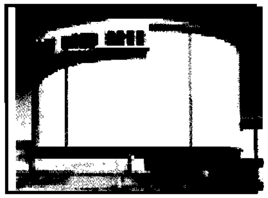
- Final draft of the Licensing Strategy Technical Basis Report (NUREG-1902) – March 2008
- Draft Report to Congress – March 2008
- ACRS Full Committee meeting planned – April 2008

6

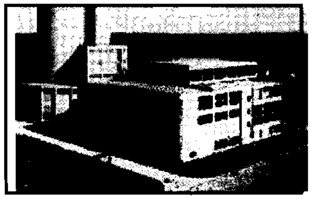

NGNP Design and Technology Development Status
Trevor Cook DOE-NE
David Petti INL

High Temperature, Gas-Cooled Reactor Experience

HTGR PROTOTYPE PLANTS | **DEMONSTRATION PLANTS**



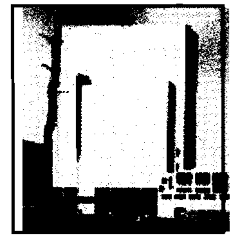
DRAGON
(U.K.)
1963 - 76



AVR
(FRG)
1967 - 1988

PEACH BOTTOM 1
(U.S.A.)
1967 - 1974

FORT ST. VRAIN
(U.S.A.)
1976 - 1989

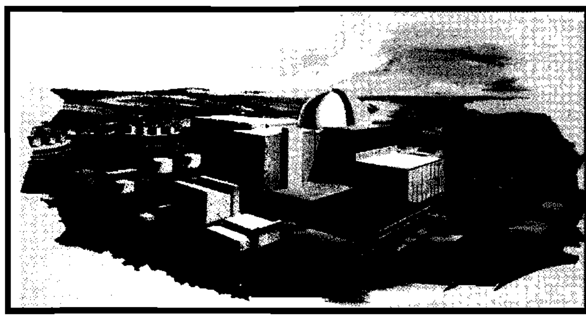


THTR
(FRG)
1986 - 1989

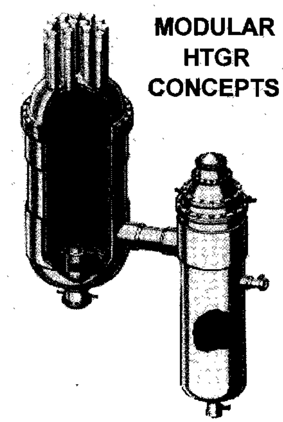
HTGR TECHNOLOGY PROGRAM

- MATERIALS
- COMPONENTS
- FUEL
- CORE
- PLANT TECHNOLOGY

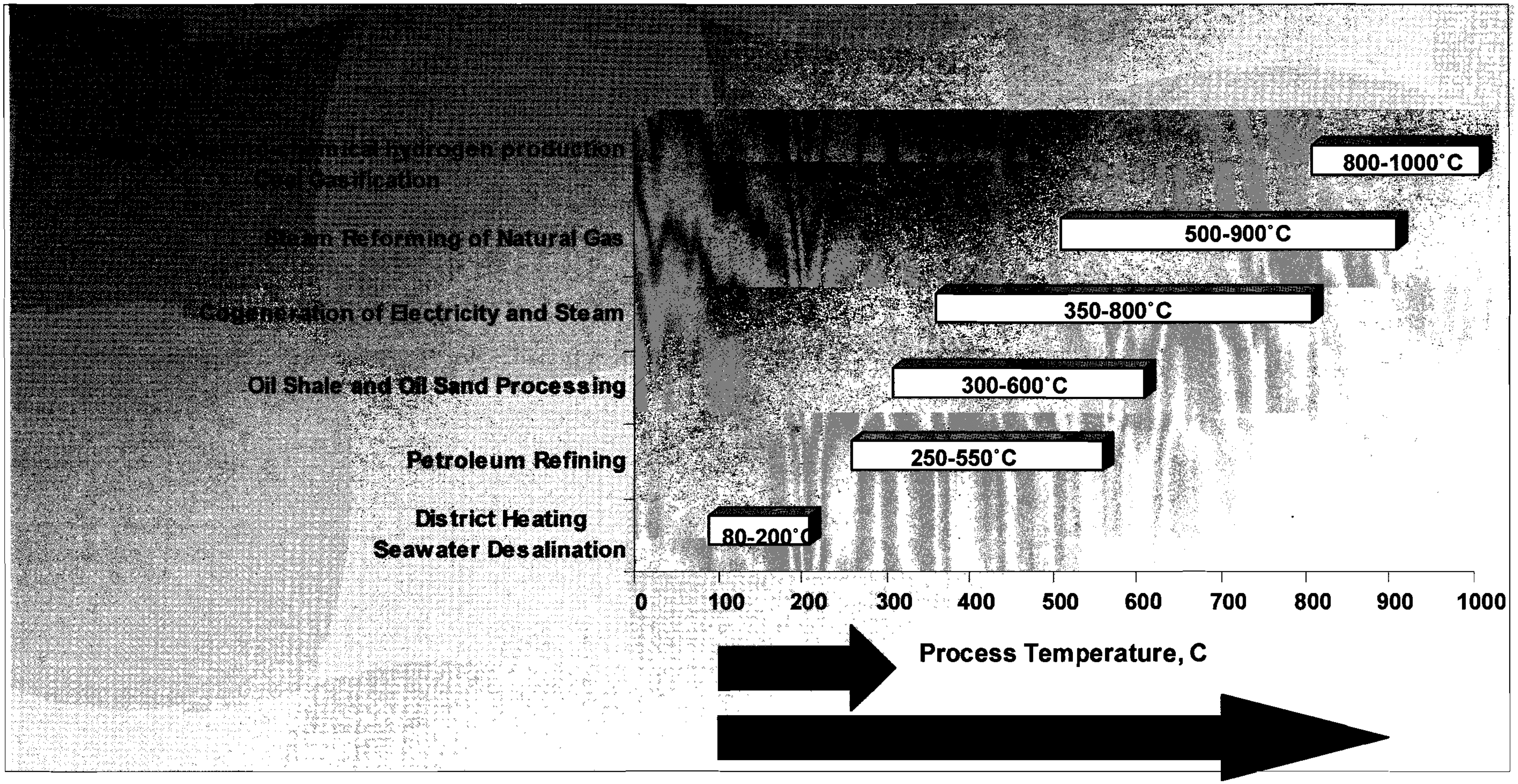
LARGE HTGR PLANTS



MODULAR HTGR CONCEPTS



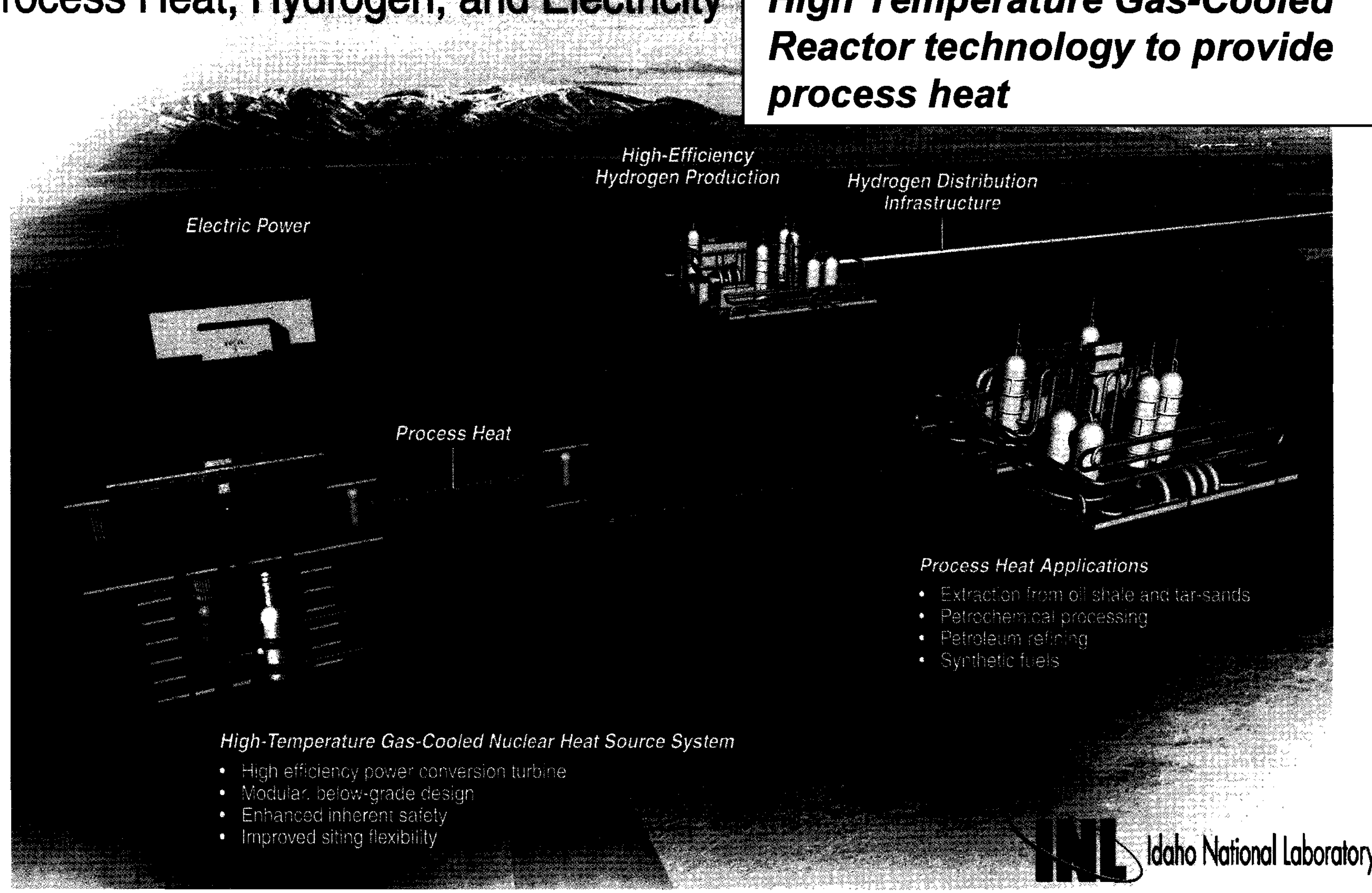
Application Temperature Requirements



Next Generation Nuclear Plant

Process Heat, Hydrogen, and Electricity

Enables commercialization of High Temperature Gas-Cooled Reactor technology to provide process heat



Pre-Conceptual Design Results

The table below presents a set of preliminary selections for the NGNP design that are based Pre-Conceptual Design studies. These preliminary selections serve as the point of departure for the NGNP conceptual design effort.

Property	Design Selection
Reactor type	**Prismatic block or Pebble Bed**
Reactor power	**~500 MW(t) to 600 MW(t)**
Power conversion cycle	**Indirect / TBD**
Number of loops	**TBD**
Primary coolant	**Helium**
Core inlet helium temperature	**350°C - 500°C**
Core outlet helium temperature	**850°C - 950°C**
Secondary loop working fluid	**Helium**
Hydrogen production process	**SI, HyS, HTE**

Pre-Conceptual Design Summary

Recommended Plant Operating Conditions

Item	Westinghouse Team	AREVA Team	General Atomics Team
	Functional & Operational Requirements		
Power Level, Mwt	**500 Mwt**	**565 Mwt**	**550 – 600 Mwt**
Outlet Temperature, °C	**950 °C**	**950 °C**	**Up to 950 °C**
Inlet Temperature, °C	**400 °C**	**500 °C**	**490 °C**
Cycle Configuration	**Indirect – Series hydrogen process and power conversion**	**Indirect – Parallel hydrogen process and power conversion**	**Direct PCS** **Parallel indirect hydrogen process**
Secondary Fluid	**He**	**He-Nitrogen**	**He**
Power Conversion Configuration	**Indirect – Rankine**	**Indirect – Combined Cycle**	**Direct – Gas Turbine** **Direct / Indirect – Combined Cycle option**
Power Conversion Power	**100 % of reactor power**	**100 % of reactor power**	**100 % of Reactor Power**
Hydrogen Plant Power	**10% of reactor power**	**10% of reactor power**	**5 Mwt – HTE** **60 Mwt – S-I**

Pre-Conceptual Design Summary

Recommended Plant Configurations

Item	Westinghouse Team	AREVA Team	General Atomics Team
	Functional & Operational Requirements		
Reactor Core Design	**Pebble Bed**	**Prismatic**	**Prismatic**
Fuel	**TRISO UO2**	**TRISO UCO**	**TRISO** **Variable**
Reactor Pressure Vessel Design	**Cooled by primary coolant**	**Not cooled; potentially insulated**	**Not cooled**
RPV Material	**508/533**	**9Cr1Mo**	**2-1/4 Cr – 1Mo** **9 Cr – 1 Mo**
Intermediate Heat Exchanger	**Printed Circuit Heat Exchanger (PCHE), In-617 material**	**Power – Helical Coil Shell & Tube, In-617** **Process – PCHE or Fin-Plate, In-617**	**Process – printed circuit heat exchanger**
Hydrogen Plant	**Initial – High Temperature Electrolysis (HTE)** **Longer Term - Hybrid thermo-chemical plus electrolysis**	**Initial – High Temperature Electrolysis (HTE)** **Longer Term – Sulfur Iodine**	**Initial – High Temperature Electrolysis** **Longer Term – Sulfur-Iodine**
Power Conversion	**Rankine; standard fossil power turbine generator set**	**Combined cycle using commercial turbine generator equipment**	**Direct gas turbine** **Option -- Direct Combined Cycle**

Key VHTR Technology Development Areas

- Fuel Development and Qualification
- Source Term Qualification
- Graphite Materials Qualification
  - Structural (non-fuel) graphite
  - Ceramic composites (C_f/C and SiC_f/SiC)
  - Structural ceramics (Fused silica, SiC, alumina)
- High Temperature Material Qualification
  - Intermediate heat exchanger (IHX)
  - Hot Duct and hot piping materials
  - Reactor Pressure Vessel (RPV)
  - Core structural metals (core barrel, control rods)
- Design and Safety Methods and Validation

Overview of AGR Program Activities

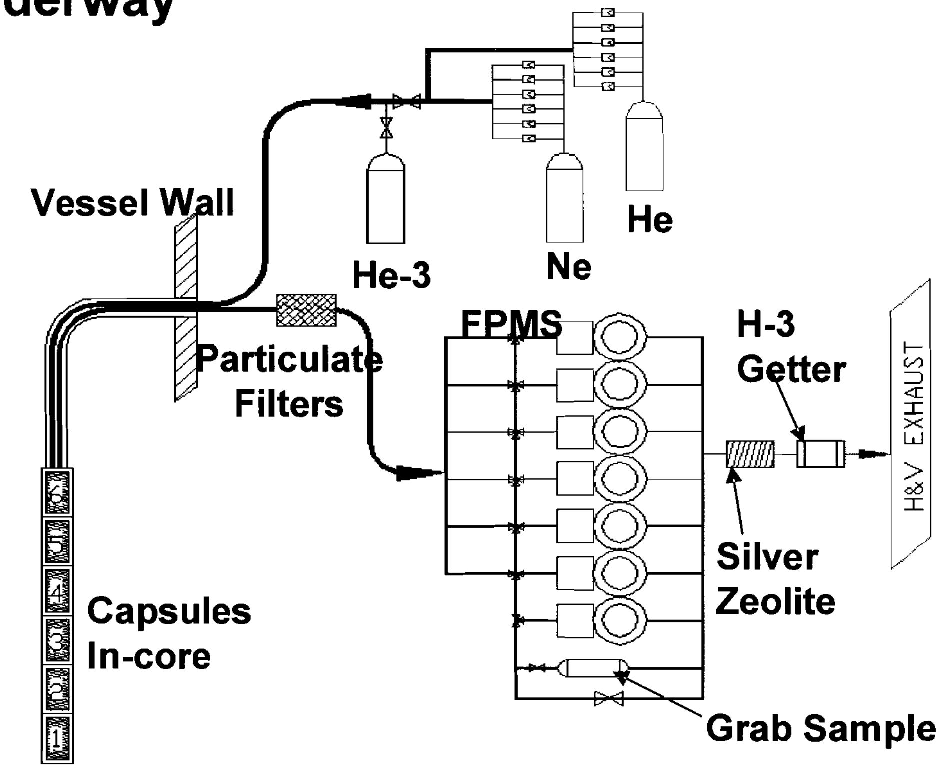
Purpose	Irradiation	Safety Tests &PIE	Models
Early lab scale fuel Capsule shakedown Coating variants German type coatings	[redacted]	[redacted]	[redacted]
Large scale fuel Performance Demonstration	[redacted]	[redacted]	
Failed fuel to determine retention behavior	[redacted]	[redacted]	[redacted]
Fuel Qualification Proof Tests	[redacted]	[redacted]	
Fuel and Fission Product Validation	[redacted]	[redacted]	
Fission Product Transport/Retention	[redacted]	[redacted]	

feedback

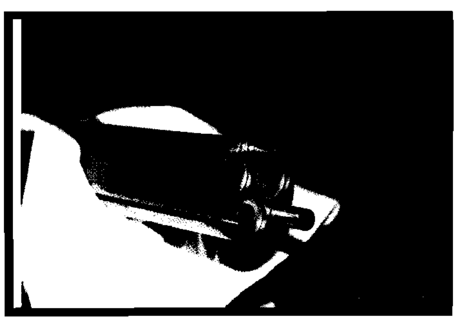
feedback

NGNP Fuel Irradiation Capsule is Underway

- 2.25 year irradiation expected
  - goal burnup ~ 15% FIMA
  - T_{max} < 1250°C, T_{avg} ~ 1150°C
  - fast fluence < 5 x 10^{25} n/m^2
- Irradiation began in December 2006
- 230 full power days of irradiation with no particle failures
- 370 more full power days required to meet irradiation goal

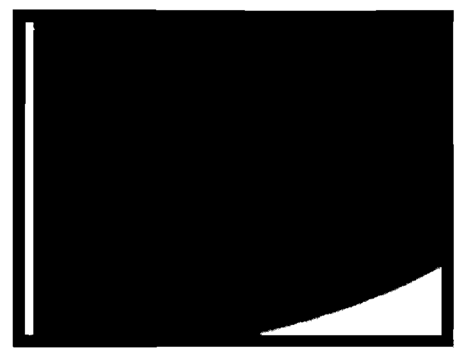


Individual capsule assembly with fuel compacts



Completed Test Train

Insertion into INL ATR

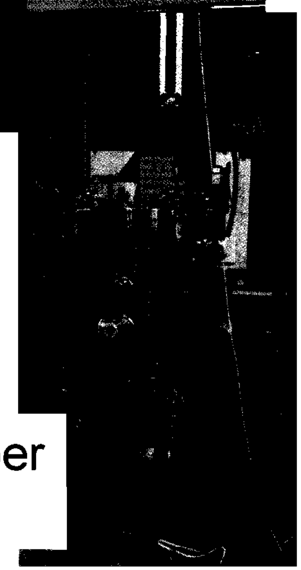


INL Fuel Annealing Furnace: Getting ready for safety testing of fuel

Key Features:

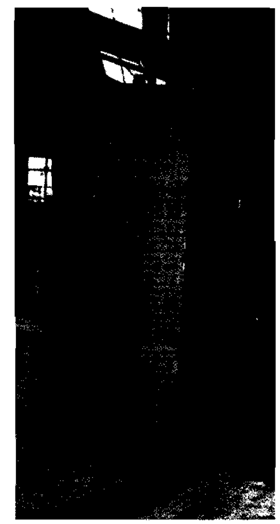
- Helium internal atmosphere
- Tantalum heating element (2000ºC max)
- Tantalum hot zone materials
- Liquid cooling for cold finger & furnace chamber
- Fully integrated, computer controlled system operation
- Automatic cold plate transfer during annealing experiment
- Hot zone capacity for up to ~6 cm diameter sphere
- Chamber and heat shields mechanically lifted to facilitate fuel sample loading/unloading.

May 2007

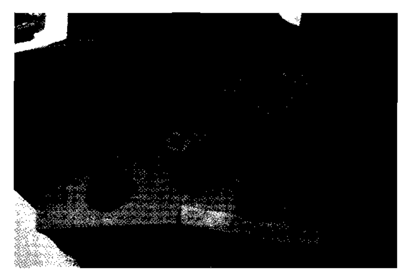


December 2006

Objectives of Graphite Program



- Qualify new grades of graphite anticipated for future VHTRs (PBMR, NGNP) to demonstrate in-reactor behavior at least as good as that used in former German and US gas reactors. (NGNP is focusing on prismatic PCEA and pebble NGB-18)
  - Establish statistical unirradiated thermo-mechanical and thermo-physical properties
    - Characterize lot to lot and billet to billet variations
  - Establish irradiated thermo-mechanical and thermo-physical properties
  - Develop understanding of life limiting phenomena at high dose and temperature (e.g. irradiation induced creep)
  - Develop appropriate constitutive relations
  - Establish reliable predictive thermo-mechanical FEM model
  - Establish relevant ASTM standards and ASME design rules
- Evaluate processing route and raw material constituents influences on graphite

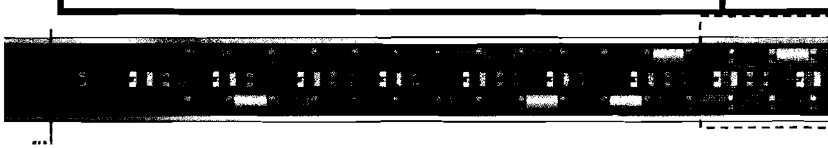


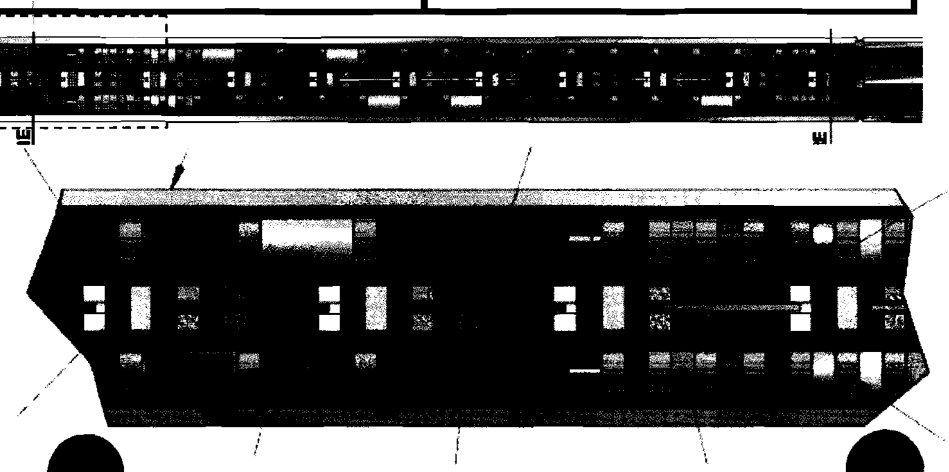
NGNP Graphite Materials Qualification: AGC-1 Activities

- Characterizing unirradiated properties of samples
- Testing fabrication, operation and assembly mockups of key aspects of the irradiation capsule AGC-1 to ensure success when actual capsule undergoes irradiation.
- Anticipated irradiation date is March 2009.
- Graphite grades for irradiation creep:
  - H-451, IG-110, & IG-430 = Reference grades
  - PCEA, NBG-18, & NBG-17 = New grades
- Graphite types for piggy-back specimens

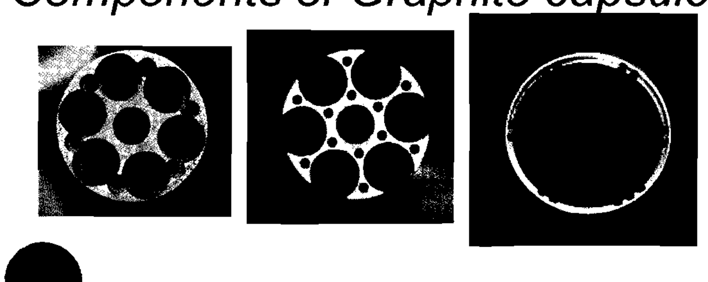
Control System Mockup

Selected and reference	Perspective types	Additional types
H-451, IG-110, IG-430, PCEA, NBG-18, and NBG-17	NBG-25, PCIB, PPEA, NBG-10, BAN	HLM, PGX, S2020, HOPG, and A3 matrix





Components of Graphite capsule

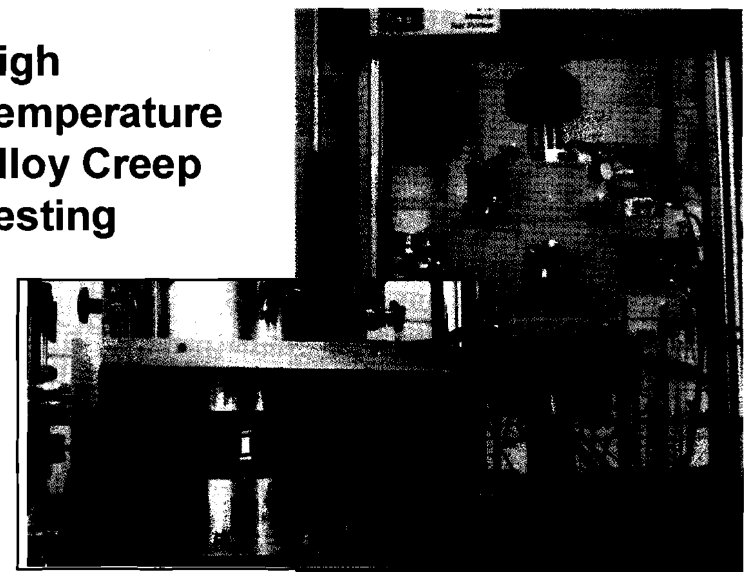


NGNP High Temperature Materials Status

- Technology development is required to qualify a material for the IHX that can be used as a heat transfer and structural material at 850-950°C
  - Inconel 617 and Haynes 230 are candidate Ni based alloys
  - Key issues are:
    - Creep and creep/fatigue life
    - Effects of impurities in He on alloy microstructure and performance
    - Development of database necessary for ASTM/ASME Code Qualification
  - Currently performing creep, creep/fatigue and environmental effects testing to determine differences in alloys for ultimate use in materials selection

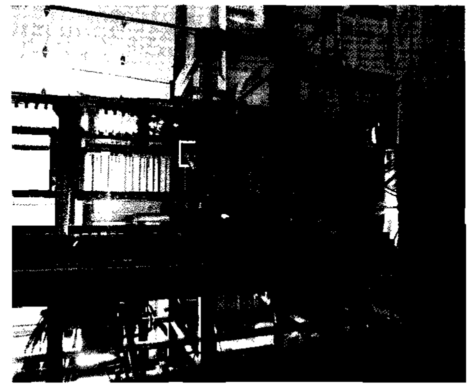
High Temperature Alloy Low Velocity Environmental Effects Testing

High Temperature Alloy Creep Testing

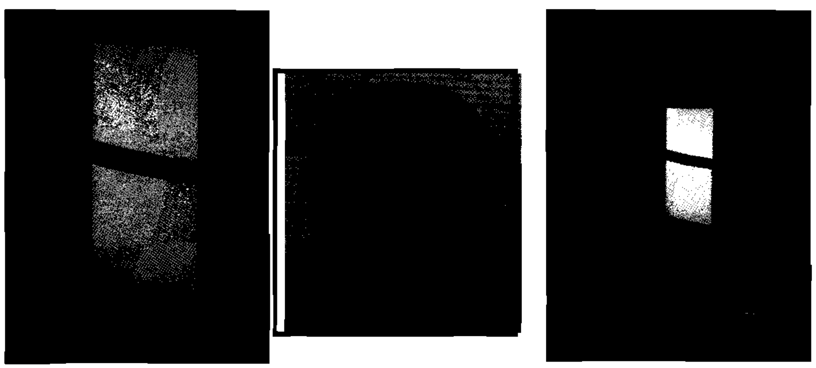


NGNP Design & Safety Methods R&D

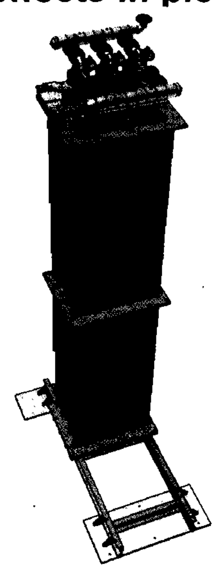
- Developing state of the art neutronic model for pebble bed and prismatic reactors
- Developing improved CFD models for flow in upper and lower plena of VHTR
- Developing improved air ingress models (collaboration with Korea)
- Planning for integrated scaled testing of RCCS



INL's matched index of refraction (MIR) facility to study 3-D flow effects in plena



Graphite/air reaction rate testing



ANL facility to validate VHTR reactor cavity cooling system behavior

FY-08 Planned Activities

- Fuels:
  - Continue AGR-1 irradiation
  - Continue pilot scale coating and compacting development for UCO and UO2 leading to AGR-2
- Graphite
  - Complete AGC-1 final design
  - Continue non-irradiated characterization of graphite
- High Temperature Materials
  - Development of acquisition strategy and technology development plan
  - Continue environmental testing, creep and creep fatigue testing of candidate alloys
- Methods
  - Continue benchmarking and validation
  - Develop test plan for RCCS validation tests

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Submitted by J Riccio, Green Peace at ACRS Full Committee meeting on Next Generation Nuclear Plant on 2/7/2008, Rockville, MD.

D880720

The Honorable Lando W. Zech, Jr.
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Zech:

SUBJECT: REPORT ON KEY LICENSING ISSUES ASSOCIATED WITH DOE SPONSORED REACTOR DESIGNS

During the 339th meeting of the Advisory Committee on Reactor Safeguards, July 14-16, 1988, we met with members of the NRC Staff and the Department of Energy (DOE) Staff and reviewed a draft Commission Paper on "Key Licensing Issues Associated with DOE Sponsored Reactor Designs," dated February 9, 1988. This subject was also considered during our 334th, 335th, 336th, and 337th meetings on February 11-13, 1988; March 10-12, 1988; April 7-9, 1988; and May 5-7, 1988, respectively. Our Subcommittee on Advanced Reactor Designs met on January 6, 1988 to discuss this matter. We also had the benefit of the documents referenced to this letter.

The Commission, in a letter dated July 9, 1987, instructed the staff to develop such a key-issues paper in advance of projected safety evaluation reports on each of the three conceptual designs being proposed by DOE and its contractors. The Committee believes this was a wise decision; it is appropriate to confront and attempt to resolve the most important safety and licensing issues in a general and direct way, rather than only by reacting to design proposals. In doing this, the NRC Staff has undertaken an important and difficult task. It can be viewed as an attempt to create, from the top down, a comprehensive rationale for licensing requirements. This would be very different from the existing body of regulations for light water reactors (LWRs), which has grown an element at a time in a more reactive and pragmatic fashion.

The nation has more than thirty years of experience in the development and realization of practical nuclear power. The DOE sponsored designers have made use of this experience and of associated research and analytical development to create three conceptual designs which they believe offer significant advantages over existing LWR plants.

Similarly, the NRC should take advantage of experience in the regulation and safety analysis of plants to create an improved approach to the specification of safety requirements. In doing this, care must be taken that regulatory requirements do not unnecessarily frustrate the development of advanced reactors. The regulations should permit the application of innovative reactor concepts while protecting the health and safety of the public. We believe this can be done, but additional effort on the part of the Commissioners and the NRC Staff will be required. False urgency should be avoided; it is more important to do the job right than to do it soon.

The staff effort so far has been thoughtful and productive, and provides appropriate preliminary guidance. They have identified four key issues as a basis for review of the design proposals:

~ Accident selection

~ Siting source term selection and use

~ Adequacy of containment systems

~ Adequacy of off-site emergency planning.

We believe these are important issues, but they do not adequately encompass the full set of concerns. We comment below on these issues and then discuss several additional issues that we believe are also important and deserve further development. We suggest that the staff's key-issues paper be regarded as preliminary guidance and that a continuing program of development and dialogue is necessary before criteria are considered final.

ACCIDENT SELECTION

The staff has proposed four event categories for selection of design basis events based on estimates of the probability of events that might challenge a given system and on past practice and engineering judgment.

For the second of these event categories (EC-II), the staff would require that there be tolerance for single failures, that only safety-grade systems should be credited in meeting the event challenge, and that reactor plant systems should continue to operate normally in response to the challenge. We believe this general approach is sound, but requires two caveats:

~ Credit for performance of nonsafety grade equipment in this class of events should be permitted when this can be justified. Designation of a component or system as safety grade is intended to ensure it has certain specific attributes. Among these are the ability to resist certain seismic events, ability to function within certain harsh environments, and a high level of reliability (supposedly guaranteed by a quality assurance program). Not all postulated initiating events are challenges to all of these attributes. Selectivity should be permitted when sufficient information is available about the nature of the design basis event.

~ We agree there should not be complete dependence on probabilistic arguments. Although estimates of probability are a proper first-cut approach to the definition of event categories, uncertainty in these estimates is large. Judgments are needed about whether and how to include as design criteria the capability to accommodate phenomena and sequences that are not specifically indicated to be necessary by probabilistic estimates.

CONTAINMENT SYSTEMS

Containment structures clearly are intended to restrict release to the environment of radioactive materials resulting from a severe accident. For LWRs, although the design bases for containments have included a source term related to severe accidents, the design pressures and temperatures have been those related to a large-break LOCA rather than those resulting from an accident involving severe core damage. Whether this seemingly inconsistent but pragmatic approach has served the nuclear power enterprise well can be debated. On the one hand, some of the severe accident issues facing the NRC and the industry today are a legacy of that approach. On the other hand, such a containment performed very well in the TMI-2 accident. Research over the past few years indicates that most existing containments would be reasonably effective in reducing the consequences of severe accidents.

The staff proposal for severe accident and containment requirements for advanced reactors seems to be taking a different, but not necessarily better approach, than that used for LWRs. Their contention is that, if the early lines of defense, namely:

- prevention of challenges to protection systems, and

- prevention of core damage by protection systems

are effective enough, then the next two lines of defense, namely:

- a conventional containment structure, and

- an emergency plan for the area around the site,

are not necessary.

The so-called prevention and protection attributes of the three designs being proposed by DOE and its contractors are indeed impressive. The modular high temperature gas cooled reactor (MHTGR) has no conventional containment structure, but relies instead on the capacity of its unique fuel particles to retain fission products, even at abnormally high temperatures, with high reliability. The two liquid metal reactor (LMR) designs have containers around the reactor vessels, but these have low volume and pressure capacity. It is unclear how they would accommodate a challenge greater than minor leakage of sodium coolant.

Accidents can be postulated that would challenge the defense-in-depth concepts being advanced. For the LMRs, a contemporaneous failure of the guard vessel and the reactor vessel, coupled with a sodium fire, would seem to lead to severe consequences. For the MHTGR, a fire in the graphite moderator, perhaps permitted by massive failures of the reactor vessel and core support, might also have severe consequences. Whether these or other accidents could be effectively mitigated by a containment enclosure, or a filtered vent, has not been determined.

We note that in all three designs, absence of containment helps to make feasible one of the major safety advantages, passive systems for removing decay heat. In each case, the reactor vessel surroundings are designed so that air from outside the plant will flow by natural buoyancy through the reactor vessel cavity and thereby remove decay heat. This seems to be a highly effective heat transfer means if the reactor vessel and core are intact. If they are not, this ready supply of oxygen and access to the environment might be a problem. This seems to be a major safety trade-off.

We are not prepared at the present time to accept these approaches to defense in depth as being completely adequate. Further, we are not prepared at this time to accept the arguments that increased prevention of core melt or increased retention capacity of the fuel provide adequate defense in depth to justify the elimination of the need for conventional containment structures. This is not to say that we could not decide otherwise in the future, in response to an unusually persuasive argument.

EMERGENCY PLANNING

We agree with the present approach of the staff's proposal. However, we believe that emergency planning should be reexamined in an effort to describe an approach that would be applicable to all types of reactors.

ADDITIONAL ISSUES

How safe should these plants be?

We believe the debate about how safe is safe enough is concluded. The safety goal policy is in place. That should stand as the definition of how safe these advanced reactors, as well as future LWRs, should be. There are, of course, matters of interpretation and implementation with regard to safety goal policy. These need to be dealt with for all types of reactor plant designs. The focus of licensing and regulation for advanced reactors should be consistent with the safety goal policy; no more, no less, no enhancements, no compromises.

The Advanced Reactor Policy states that advanced reactors must be at least as safe as the current generation of LWRs. The staff interprets this to mean the "evolutionary" generation of LWRs now being reviewed by the NRC for preliminary design certification.

We believe the Advanced Reactor Policy requires no more than, and should require no more than, the level of safety called for in the safety goal policy. Reactor developers, i.e., DOE and the industry, may seek a design that is safer than the safety goal would suggest as necessary, or whose safety is more readily apparent to the public. Those are not unreasonable goals for a developer in seeking public acceptance or more economic operation. However, it seems to us inappropriate for the NRC to ratchet on the standard of safety it has established as necessary and sufficient.

To what extent should regulatory requirements accommodate public perception?

The draft paper states that the staff has incorporated only technical considerations in the development of its proposed positions. In particular, they have not attempted to accommodate external factors, such as public perception. We applaud this restraint. And we counsel the Commission to keep safety regulations unambiguously related to protection of the public health and safety.

Extra capacity in decay heat removal and scram systems

The three DOE designs provide much more capacity in decay heat removal and scram systems than are provided in present LWRs. While these important systems in LWRs must be tolerant of single failures, the advanced reactors go well beyond that. The reason for this is the intent to build more robustness into the first two layers of defense in depth and thus permit less in the last two layers, containment and emergency planning.

Two independent scram systems are provided in two of the three proposed designs. Each system is somewhat diverse in design and tolerant, within itself, of single failure. All three design proposals have multiple systems for decay heat removal. In addition to being diverse and resistant to single failure, the extra systems have inherent passive attributes. They apparently will function effectively without motive power or operator intervention.

However, a caution is necessary. Experience in operation and analysis has indicated that redundancy, i.e., extra systems or components, is not as powerful in improving reliability as might be expected. Too often the nature of initiating challenges, or of the complex sequence of events in accidents, seems to cause the extra parts of a system to be faulted along with the main system. The diverse and passive nature of the three designs being considered might ameliorate such unwanted interdependency, but further study is warranted. In addition, while the three proposed designs have these positive features, it is not clear that the NRC's proposed requirements would provide assurance that these desirable diverse and passive attributes would be guaranteed.

Need for prototyping

The staff proposes only modest requirements for prototype testing of the advanced reactor designs. Although, they have recently added a proposed requirement that any designs not incorporating a containment must be tested in prototype at a remote site, we question whether this is enough to carry the process to a point at which the NRC would be willing to license an unlimited number of new power plants. For example, the metallic LMR cores are claimed to have very favorable, inherently stable characteristics in responding to possible transients. These characteristics were not well understood a decade ago.

An excellent experimental and analytical program by ANL with the EBR-II reactor at INEL has effectively demonstrated that the EBR-II system does exhibit such inherently stable and predictable behavior. However, it is not yet clear that such characteristics can be assured for the larger and different LMRs to be used in commercial electric power production. We believe that a more and extensive series of prototype tests will be necessary before design certification could be granted.

Use of cost-benefit analysis

The staff paper proposes that prospective licensees should be required to demonstrate through cost-benefit analysis that design features alternative to those being proposed are not warranted. Presumably, the NRC staff would review such analyses and perhaps suggest alternatives. We believe this is an unworkable and unnecessary strategy. The NRC should concentrate its efforts on specifying design requirements that will result in plants that are in conformance with the safety goal. Consideration of alternatives and costs is properly a function of the designer and owner of a plant. The NRC should have enough confidence in its safety goal that it does not feel the need for the proposed approach.

Design for resistance to sabotage

It is often stated that significant protection against sabotage can be inexpensively incorporated into a plant if it is done early in the design process. Unfortunately, this has not been done consistently because the NRC has developed no guidance or requirements specific for plant design features, and there seems to have been no systematic attempt by the industry to fill the resulting vacuum. We believe the NRC can and should develop some guidance for designers of advanced reactors. It is probably unwise and counterproductive to specify highly detailed requirements, as those for present physical security systems, but an attempt should be made to develop some general guidance.

Operation and staffing

Little is said in the staff paper about requirements for operation and staffing of advanced reactors. We find this to be a serious oversight. Experience with LWRs has shown that issues of operation and staffing are probably more important in protecting public health and safety than are issues of design and construction. The designers of the three reactor proposals seem to be claiming that the designs are so inherently stable and error-resistant that the questions of operation and staffing, so important for LWRs, are unimportant for the advanced reactors. And that, in fact, the advanced plants can be operated with only a very small staff. We believe these claims are unproven and that more evidence is required before they can be accepted.

The two major accidents that have been experienced in nuclear power, those at TMI-2 and Chernobyl 4, were caused, in large measure, by human error. These were not simple "operator errors" but instead were caused by deliberate, but wrong, actions. There are some indications that the advanced reactor designs being considered have certain characteristics tending to make them less vulnerable to such mal-operation. But, this has not been demonstrated in any systematic way. The traditional methods of PRA are not capable of such analyses; but, we believe a systematic evaluation should be made. There seems little merit in making claims for the improved safety of new reactor designs if they have not been evaluated against the actual causes of the most important reactor accidents in our experience.

Will regulatory criteria evolve?

The Staff proposal provides for a future milestone in the ongoing design-review-licensing process at which the NRC will step back and make sure that the agreements reached early in the process are still valid, given possible new information and understandings. We believe this is wise and necessary, although it does place a potential licensee at some risk. It should be recognized that this milestone activity might have to include the possibility of changes in the actual requirements, as well as interpretations of requirements.

Focus on the most important residual uncertainties

Although the staff paper discusses uncertainties relative to the development of requirements and designs, it should provide a clearer statement of what the staff believes to be the most important of these. This would assist policymakers in making judgments about the designs and requirements and, perhaps, about whether certain avenues of research should be further pursued before or in parallel with licensing.

Additional comments by ACRS Member Carlyle Michelson are presented below.

Sincerely,

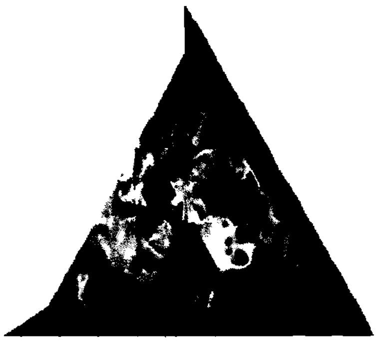
Willliam Kerr
Chairman

Additional Comments by ACRS Member Carlyle Michelson

It is not clear to me that the safety goal in its present form was intended to apply to advanced reactors which do not have conventional containment systems. The guidelines for regulatory implementation might have been different if the Commission had considered that the defense-in-depth approach might not include a containment system on future plants.

It would be unfortunate if the frequency of large release criterion suggested in the present guidelines is used as a basis for justifying the omission of a containment system for an advanced reactor plant at a time when advanced LWRs which might be able to meet the same criterion are required to have containments.

References:
1. Draft Commission Paper from Victor Stello, Jr., for the Commissioners, Subject: Key licensing issues associated with DOE sponsored advanced reactor designs, dated February 9, 1988
2. U.S. Nuclear Regulatory Commission, NUREG-1226, "Development and Utilization of the NRC Policy Statement on the Regulation of Advanced Nuclear Power Plants," published June 1988



ACRS Full Committee
549th Meeting
February 7, 2008

Office of Nuclear Regulatory Research

Mark Henry Salley, P.E.
Chief, Fire Research Branch
MXS3@nrc.gov
301-415-2840

Agenda

- Cable Response to Live Fire (CAROLFIRE) Project is complete
  - Request a letter from ACRS

CAROLFIRE

- RIS 2004-03
- Three Volumes:
  - Volume 1 Circuit Interaction
  - Volume 2 Thermal Data
  - Volume 3 Fire Modeling Improvements
- Extensive Review:
  - Peer-reviewed
  - Public Comment
  - ACRS Quality Review
  - ACRS Subcommittee Review
  - Asking for ACRS Letter

Principle Presenters

- Mr. Gabe Taylor
  - NRC/RES
- Dr. Kevin McGrattan
  - National Institute Standards and Technology

Advisory Committee on Reactor Safeguards
549th Meeting February 7, 2008

CAROLFIRE

Cable Response to Live Fire

Presented by:
Gabe Taylor
Office of Nuclear Regulatory Research
Fire Research Branch

Office of Nuclear Regulatory Research

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CAROLFIRE Objectives

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- Resolution of 'Bin 2' circuit configuration
  - Regulatory Issue Summary (RIS) 2004-03, Rev. 1, - "Risk-informed Approach For Post-Fire Safe-Shutdown Circuit Inspection"
  - Document places cable/circuit configurations in one of three bins:
    - Bin 1 : Circuit configurations that are most likely to fail
    - **Bin 2 : Circuit configurations that need more research to determine failure characteristics**
    - Bin 3 : Circuit configurations that are unlikely or least likely to fail
- Fire Model Improvement
  - To reduce uncertainty associated with predictions of fire-induced cable damage

Summary & CAROLFIRE Results of RIS 2004-03 'Bin 2' Items

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- Item A – Inter-cable shorting for Thermoset Cable
  - Plausible, but less likely than intra-cable failure mode
- Item B – Inter-cable shorting between Thermoplastic and Thermoset Cable
  - Plausible, but less likely than intra-cable failure mode
- Item C – Configurations requiring failures of three or more cables
  - Plausible
    - i.e., How many failures should be considered?
    - No a priori limit; dependent on scenario; risk significance

Summary & CAROLFIRE Results of RIS 2004-03 'Bin 2' Items

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- Item D – Multiple spurious operations in control circuits with "properly sized" CPTs
  - Inconclusive, results do not coincide with NEI/EPRI results
- Item E – Fire-Induced hot shorts lasting longer than 20 minutes
  - Unlikely
- Item F – Spurious actuations for cold shutdown circuits (Item F was not investigated by CAROLFIRE)

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CAROLFIRE was a Collaborative Effort

- Office of Nuclear Reactor Regulation
- Office of Nuclear Regulatory Research
- Sandia National Laboratories
- National Institute of Standards and Technology
- University of Maryland

Peer Review

U.S.NRC
United States Nuclear Regulatory Commission
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- CAROLFIRE Test Plan was developed by SNL and went through the RES peer review process
- All Collaborative partners participated in Peer Review
  - Nathan Siu (RES)
  - Dan Frumkin and Naeem Iqbal (NRR)
  - Anthony Hamins (NIST)
  - Mohammad Modarres (UMd)
  - Vern Nicolette (SNL)
- External expert and author of the EPRI report on the NEI/EPRI circuit tests of 2001
  - Dan Funk (EDAN Engineering)

CAROLFIRE Testing Approach

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- Two Scales of testing were pursued
  - Small-scale radiant heating experiments
  - Intermediate-scale open burn tests

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Small Scale Tests

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- *Penlight* heats target cables via grey-body radiation from a heated shroud
- Well controlled, well instrumented tests
- Allows for many experiments in a short time
- Single cables and small cable bundles (up to six cables)
- Cable trays, air drops, conduits

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Typical Penlight Setup for CAROLFIRE

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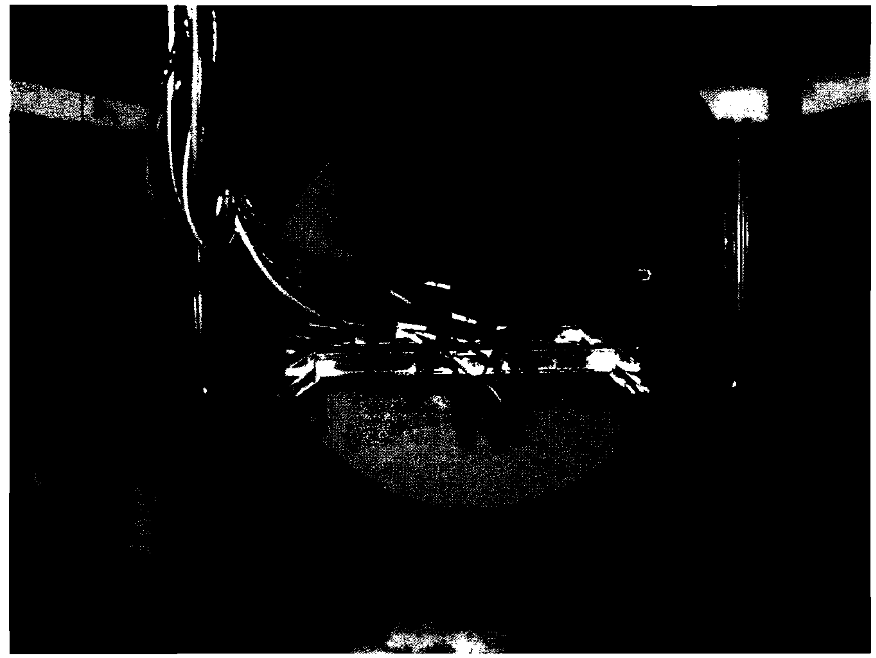
Open Tray

Closed Tray



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Typical Penlight Setup for CAROLFIRE

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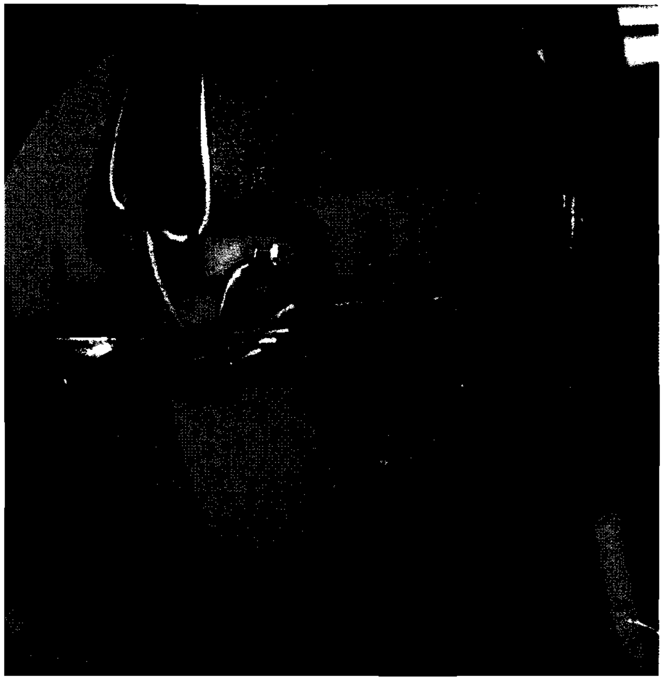
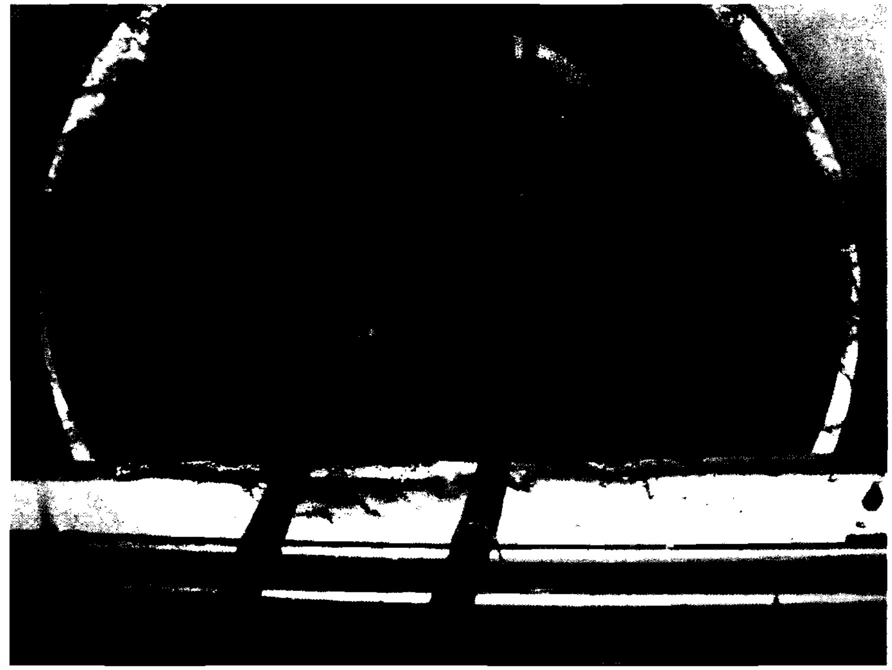
Conduit



Air Drop



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TS vs. TP Physical Failure Characteristics

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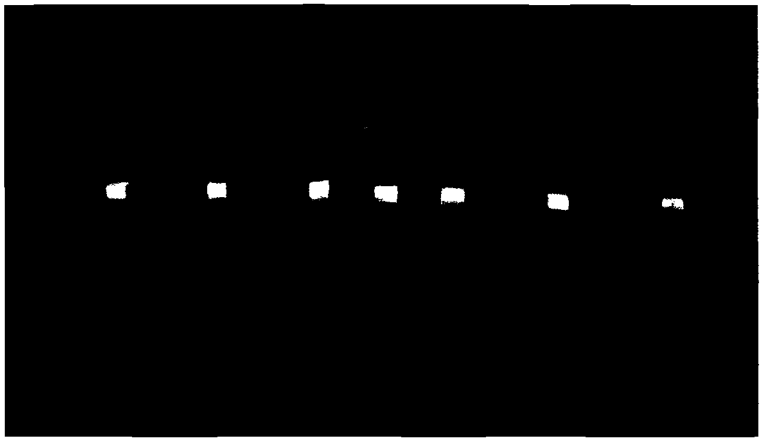
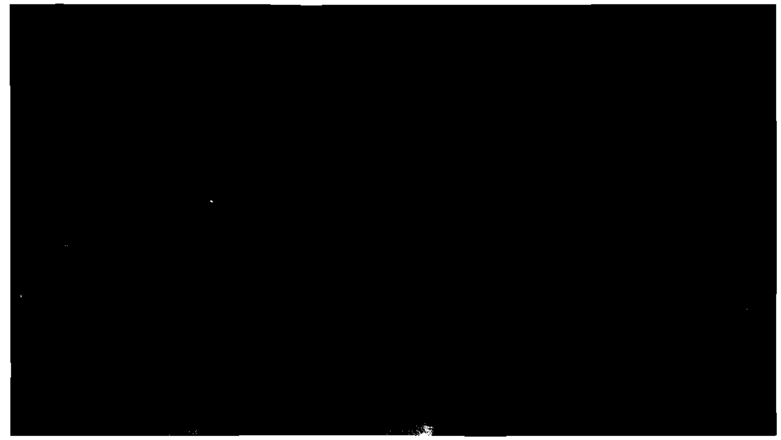
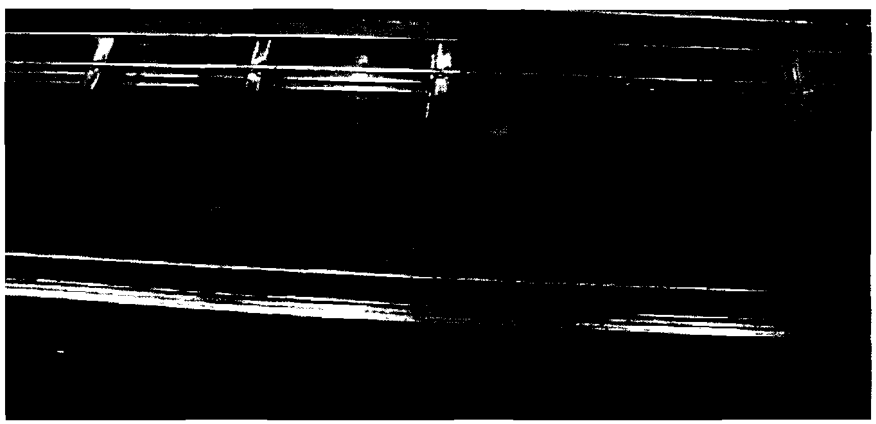




Thermoplastic

Thermoset



Penlight did allow cables to burn and burning was common

Intermediate-Scale Tests

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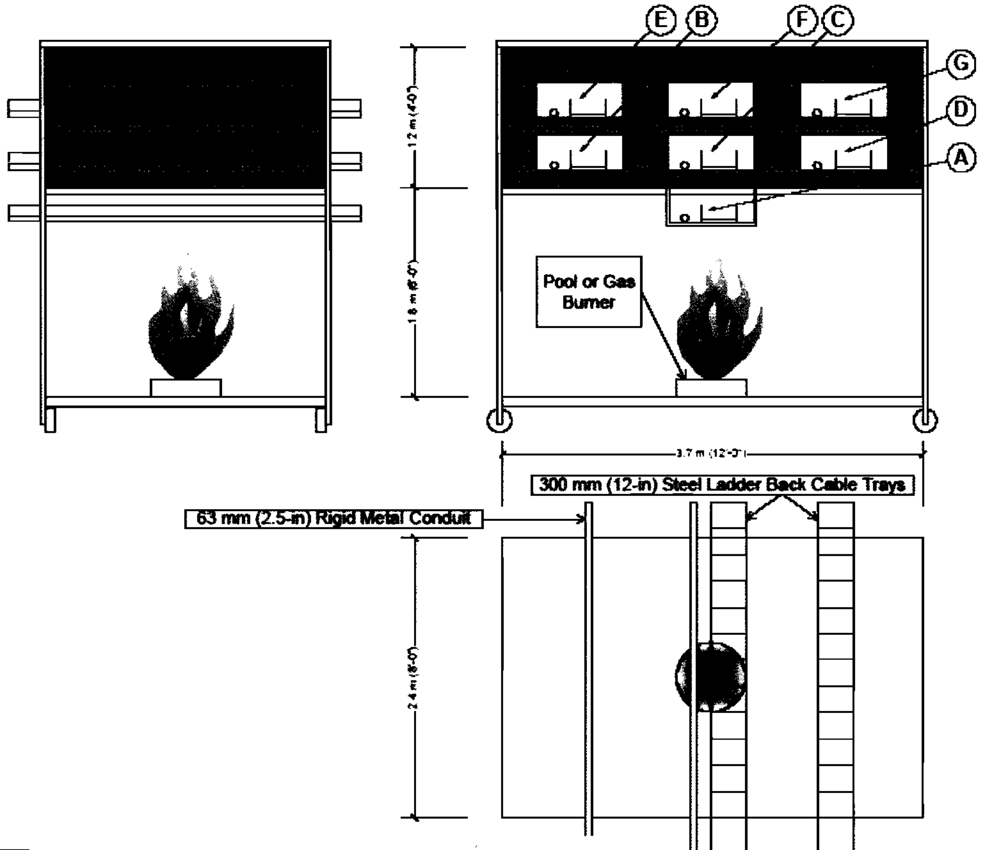
Layout of the intermediate-scale test structure.
Structure was located within a larger test facility.



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Intermediate-Scale Tests

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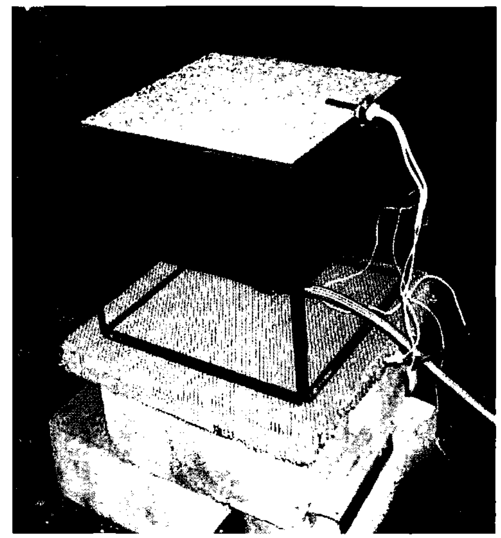
- Less controlled, but a more realistic testing scale
- Located in larger test facility
- Propene (Propylene) gas diffusion burner fire source (200 kW typical)
- Cables in trays, conduits and air drop



Typical Setups

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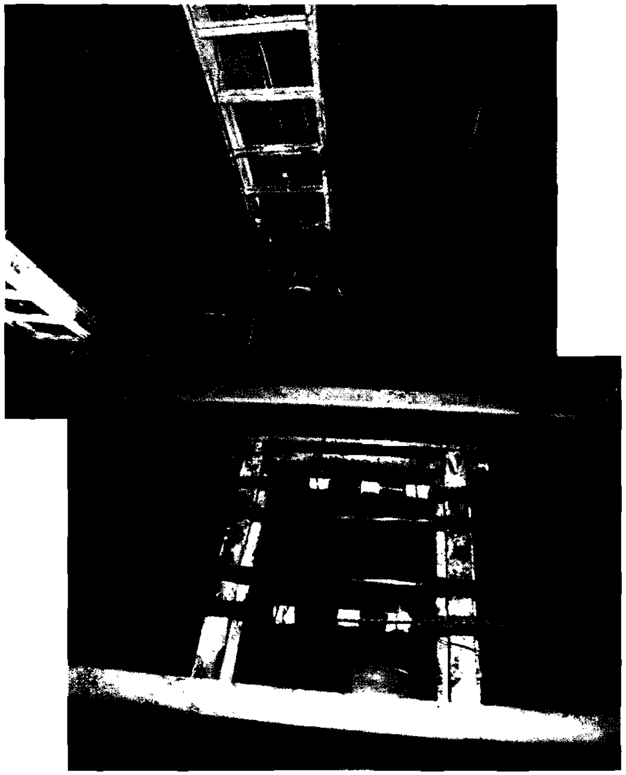
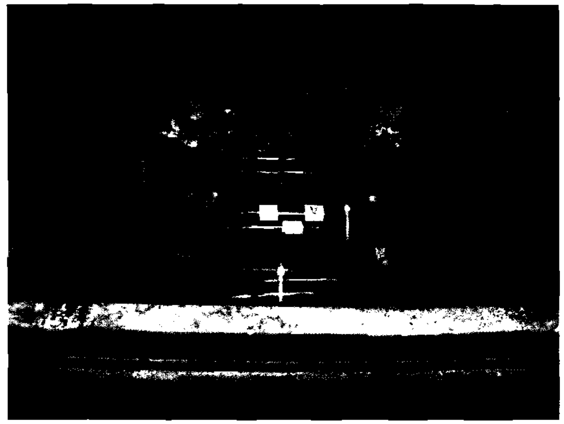
Single cables

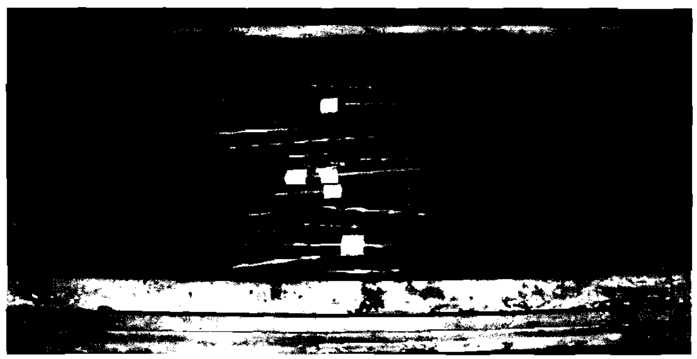




Bundles

Airdrops

Random fill trays



Cable Selection

- Testing a broad range of cable products
  - 15 cable products tested
    - 9 Control (8 were 12 AWG – 7/C)
    - 4 Instrument (16 or 18 AWG, 2/C or 12/C)
    - 2 Power (8 AWG, 3/C)
  - CAROLFIRE excluded armored cables
    - Duke armored cable tests

Photo of Tested Cables

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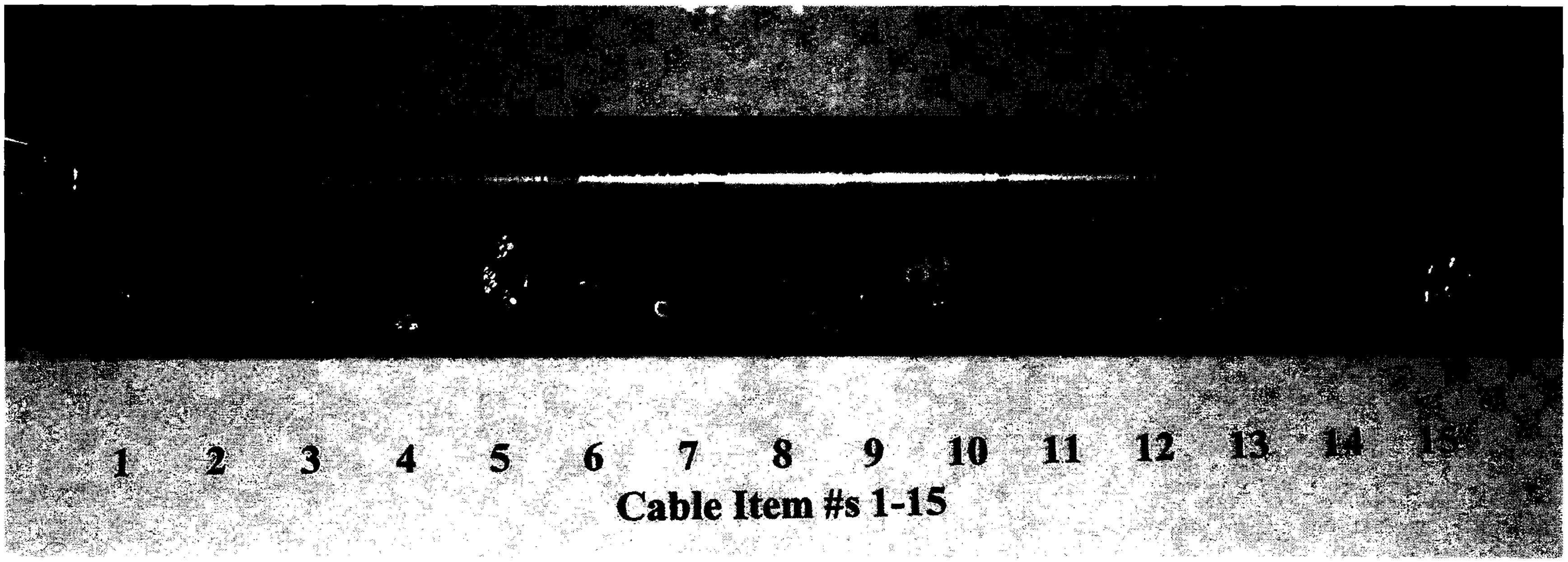

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Electrical Instrumentation

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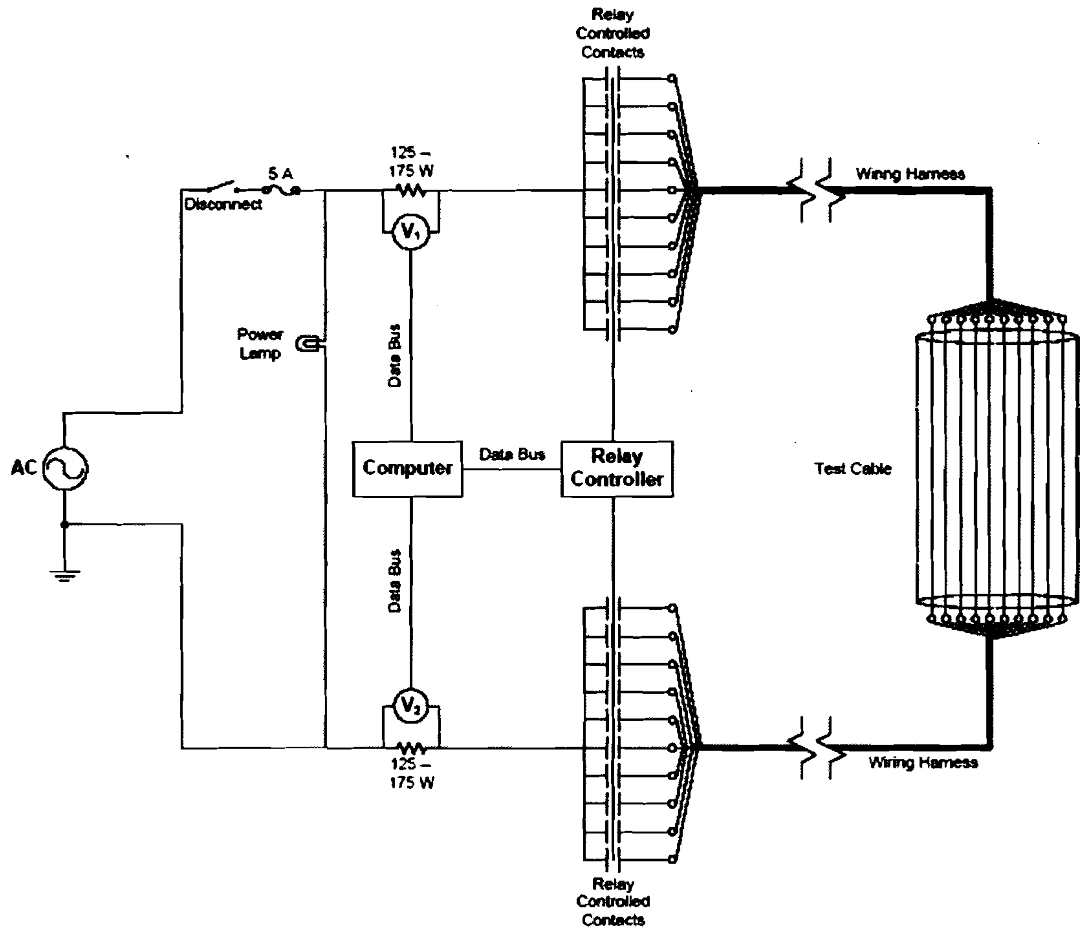
Insulation Resistance Monitoring System

- All tests – SNL Insulation Resistance Measurement System (IRMS)
- Continuous measurement of cable degradation and functionality
- Very detailed look at conductor interactions
- Patented system developed and deployed originally during the NEI/EPRI tests (NUREG/CR-6776)



IRMS Results

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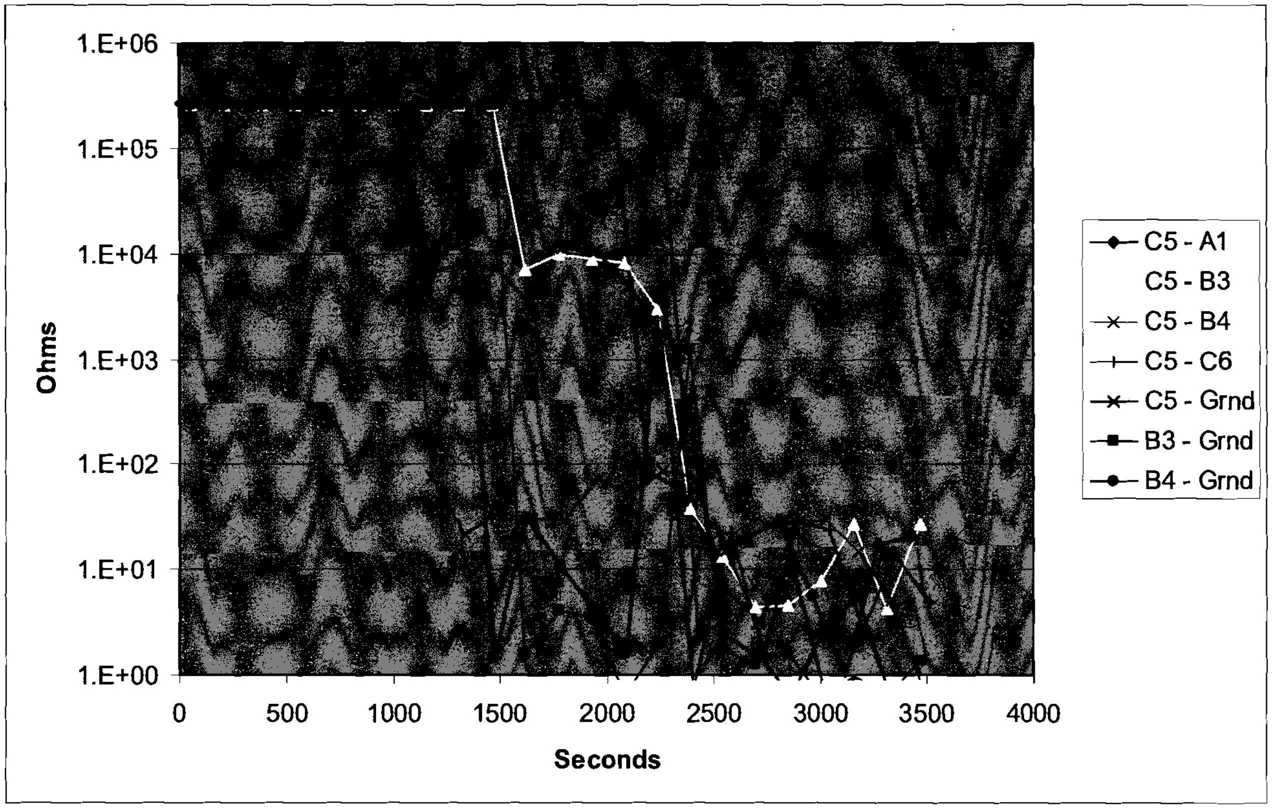

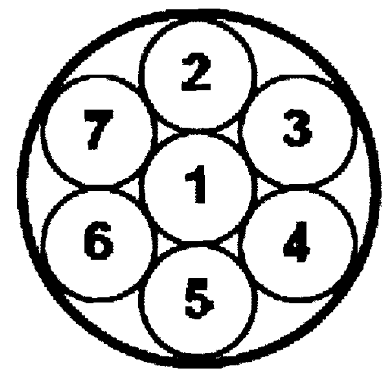

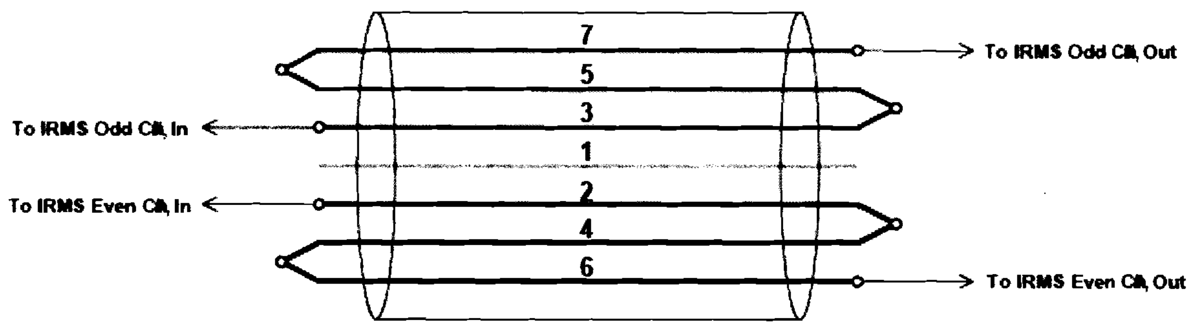

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Electrical Instrumentation

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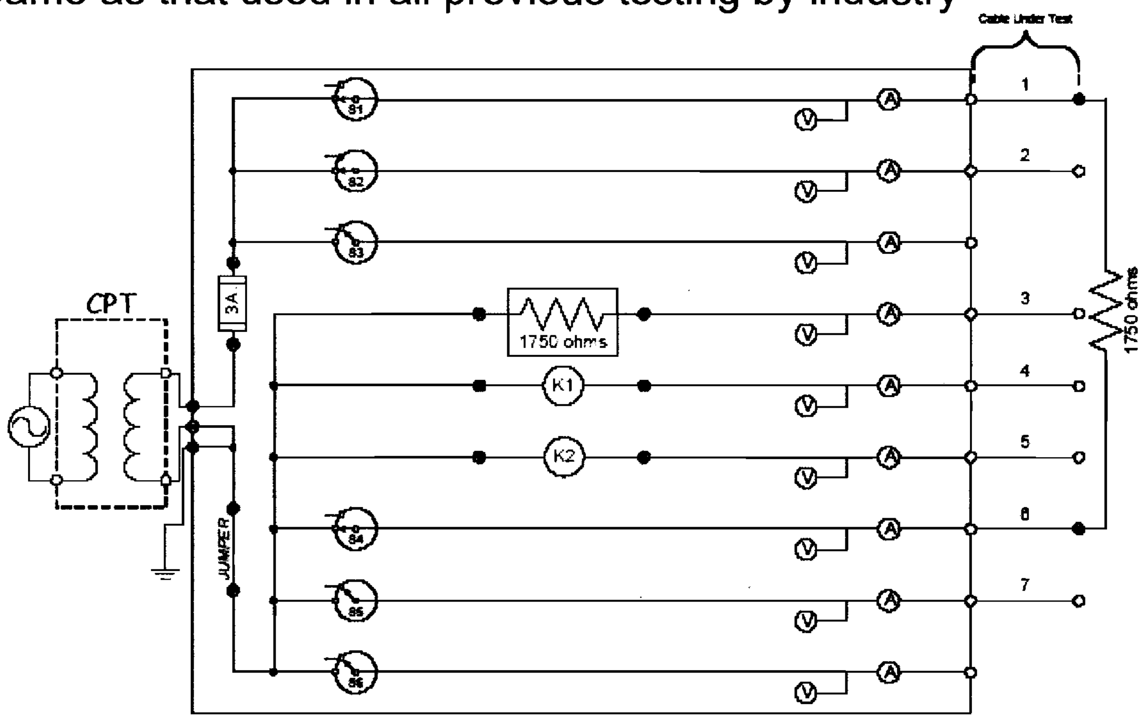
- Intermediate-scale only: control circuit simulators allow for testing of various circuit configurations
- Base configuration is the typical MOV control circuit
  - Same as that used in all previous testing by industry



Thermal Instrumentation

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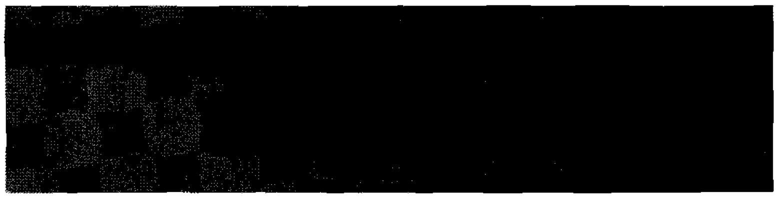
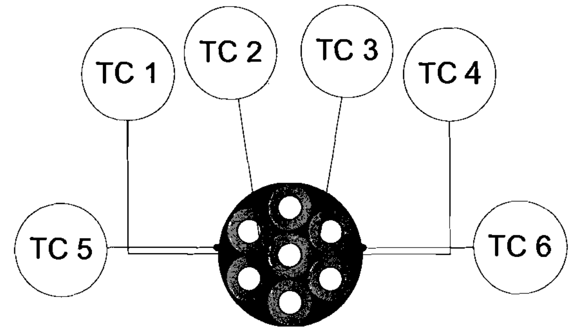
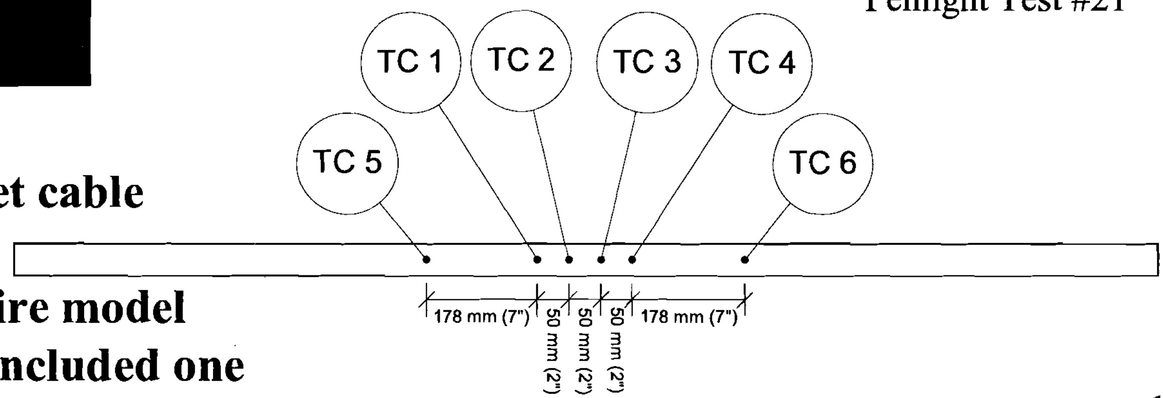
Sub-jacket placement



Sub-jacket TC bead location



Penlight Test #21



Measurements made of sub-jacket cable temperatures are one of the key measurements of interest to the fire model improvement efforts. Every test included one or more such measurements.

Raceway Temperatures

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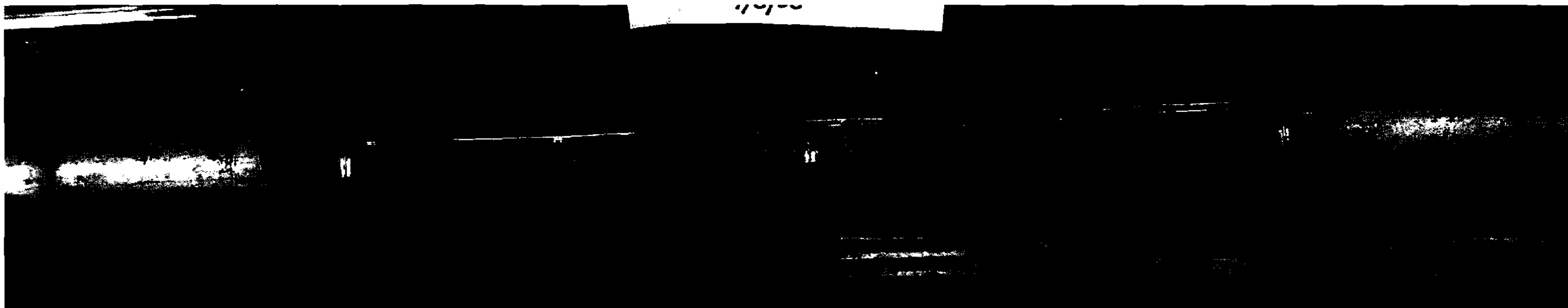

Conduit and cable tray surface temperatures are also important to fire modeling efforts.

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Electrical & Thermal Data

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- All tests were extensively documented in excel spreadsheets that includes:
  - Shorting Summary
  - Thermocouple Map
  - Plots of various electrical failure characteristics and temperatures
  - Processed and Raw Data
- All test data will be placed onto a CD and issued with the NUREG/CR
- Pictures and other related documents will also be included on a CD

NEI Test Compartment

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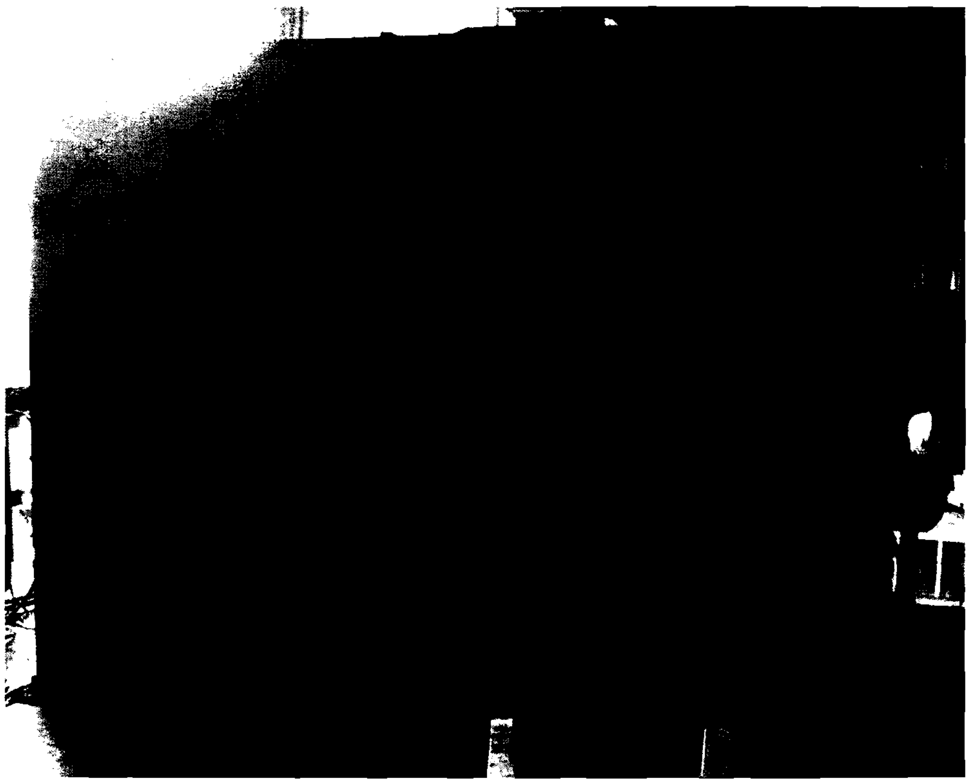
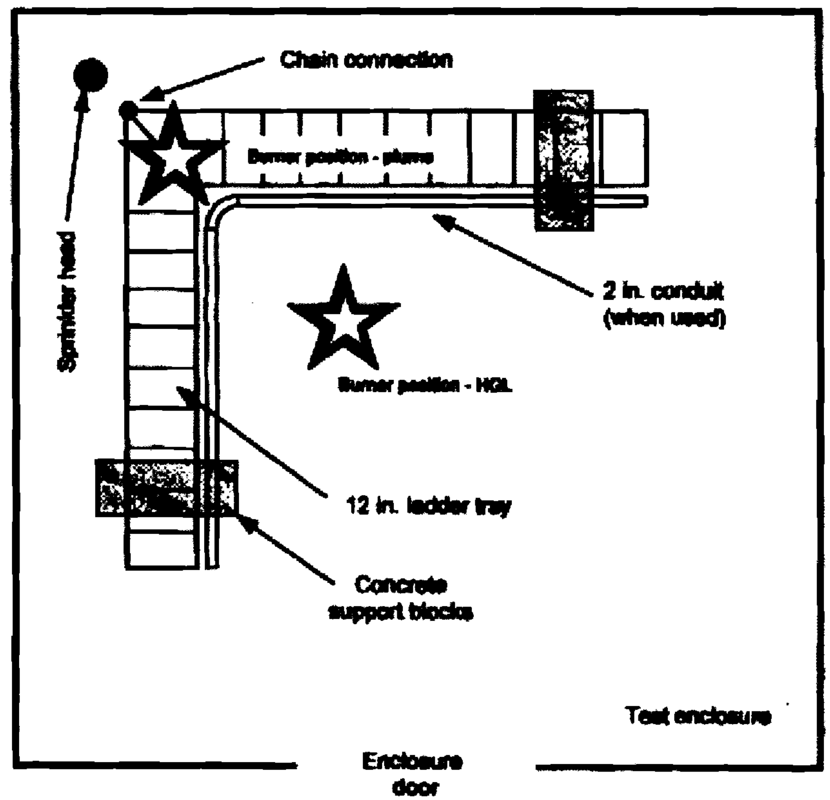


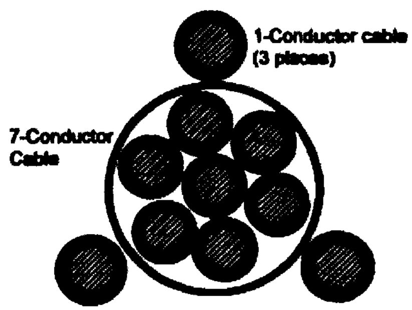


CAROLFIRE to NEI/EPRI Comparison

- 18 tests
- EPRI Report 1003326
- 10'x10'x8'
- Varied several parameters
- Long times to failure for HGL
- MOV test Circuit
- SNL IRMS was used and results are reported in NUREG/CR-6776

Parameter

Raceway loading
Raceway configuration
Exposure Conditions
Cables
Bundling Arrangements
Cable Combinations
Cable Thermal Response
CPT Size



Review of CAROLFIRE Research As It Relates to Bin 2 Items

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- Item A – Thermoset-to-Thermoset
  - Plausible
    - one solid case of TS-to-TS shorting as primary failure
    - Several cases of secondary or tertiary failure mode
- Item B – Themoset-to-Themoplastic
  - Plausible
    - One case of hot short from a TS-to-TP cable

Conclusions on Bin 2 Items

U.S.NRC
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- Item C – Concurrent for three or more cable failures
    - i.e., How many failures should be considered?
  - Plausible
    - No a priori limit; dependent on scenario; risk significance
    - Every test program conducted to date has seen as many as four out of four simulated control circuits spuriously actuate, including CAROLFIRE
- Item D – Concurrent spurious actuations given properly sized CPT
  - Inconclusive
    - Larger than intended CPT versus actuation device ratings were tested (What is meant by "properly sized")
    - No apparent affect on spurious actuations

Conclusions on Bin 2 Items

U.S.NRC
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- Item E – Hot shorts lasting more than 20 minutes
  - Unlikely

  Longest Hot Short
    - CAROLFIRE ~ 7.6 minutes
    - NEI/EPRI ~ 11.3 minutes
    - Duke armored cable tests showed similar results
  - All data appear to indicate that once cable degradation begins, it will cascade through all modes within a relatively short time

Public Comment Process

U.S.NRC
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Protecting People and the Environment

- Two sources of public comments:
  - Industry comments collected and submitted through NEI
  - ACRS comments
- Additional NRC staff comments

Key Public Comments

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- The "cable physical characteristics" table was expanded to include quantitative copper/plastic ratios
- Thermal (heat transfer) properties - Unfortunately, are not available for the materials and could not be provided
- Added a summary table for Penlight results
- New plots overlaying cable thermal and electrical response
- New plots illustrating the temperature at failure

Examples of New Plots

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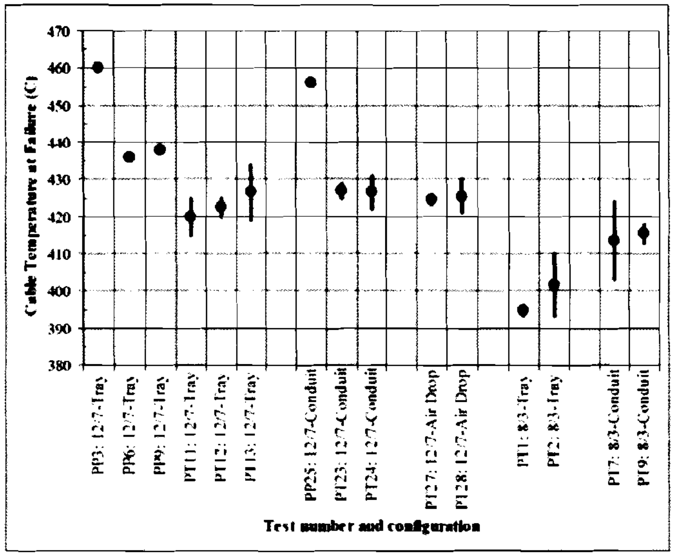


Figure 5.31: Compilation of test results for the XLPE-insulated cables.

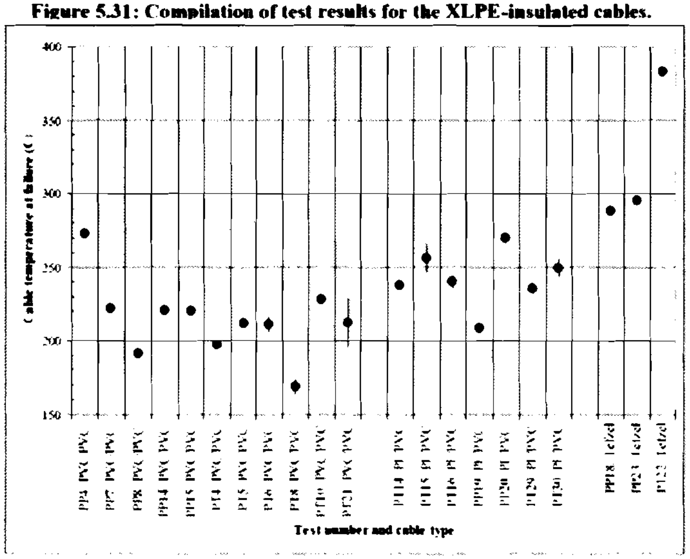
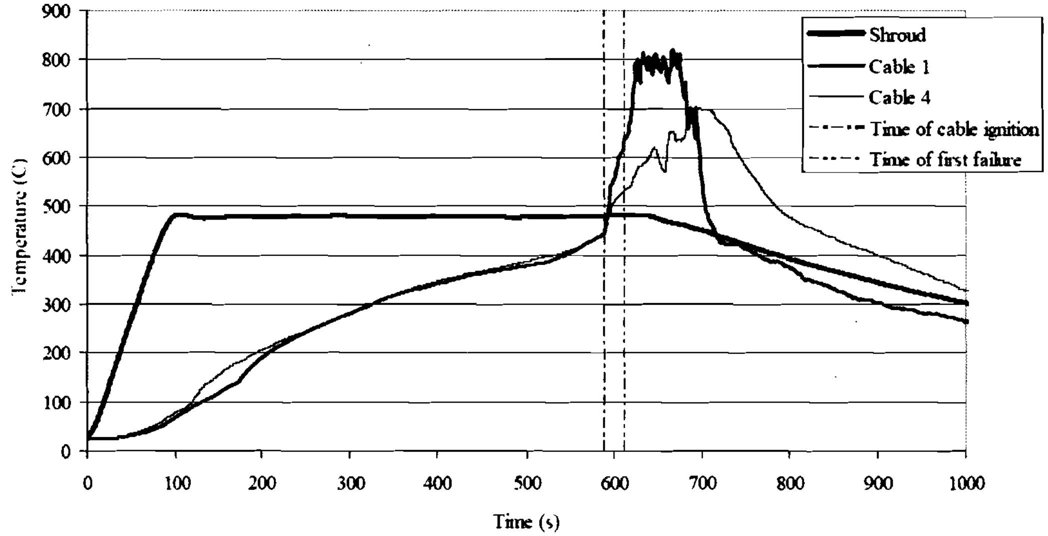


Figure 5.32: Compilation of the test results for the TP cable types.



Summary

- CAROLFIRE has contributed to two critical need areas
  - Data for resolution of RIS 2004-03
  - Improving the fire modeling of cable response and failure
- CAROLFIRE represents a valuable source of information that the fire protection community world-wide will likely be using for many years to come

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Protecting People and the Environment

Fire Branch Research

Thermally-Induced Electrical Failure (THIEF) Model

Kevin McGrattan
National Institute of Standards and Technology

Office of Nuclear Regulatory Research

NIST
National Institute of Standards and Technology
Technology Administration, U.S. Department of Commerce

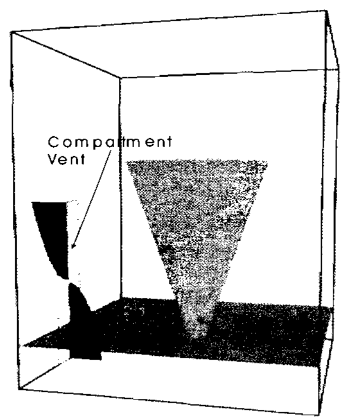
Three Classes of Fire Models

Hand Calculations

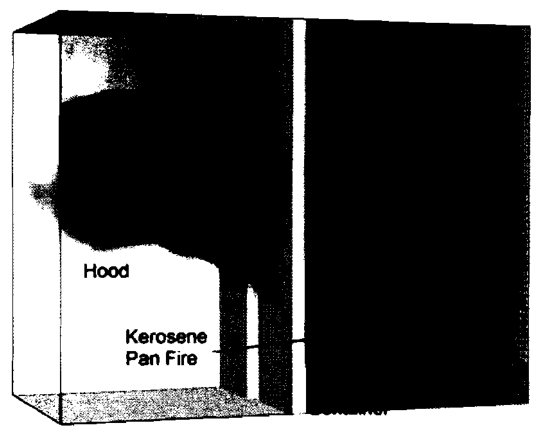
$$T_g - T_\infty = 6.85\left(\frac{\dot{Q}^2}{A_0\sqrt{H_0}\,h_k A_T}\right)^{1/3}$$

McCaffrey, Quintiere, Harkleroad (MQH)

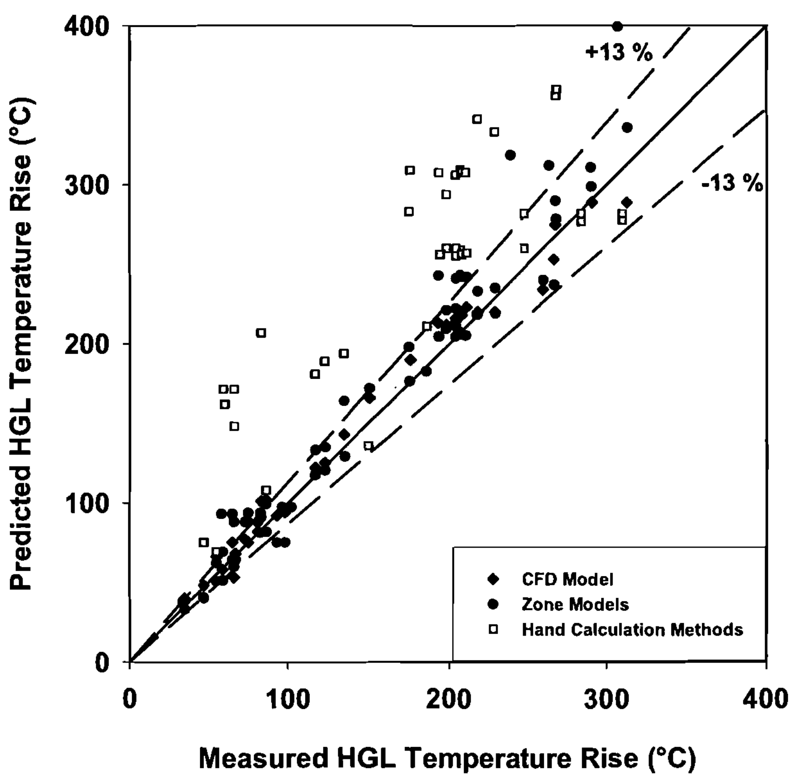
Two-Zone Models

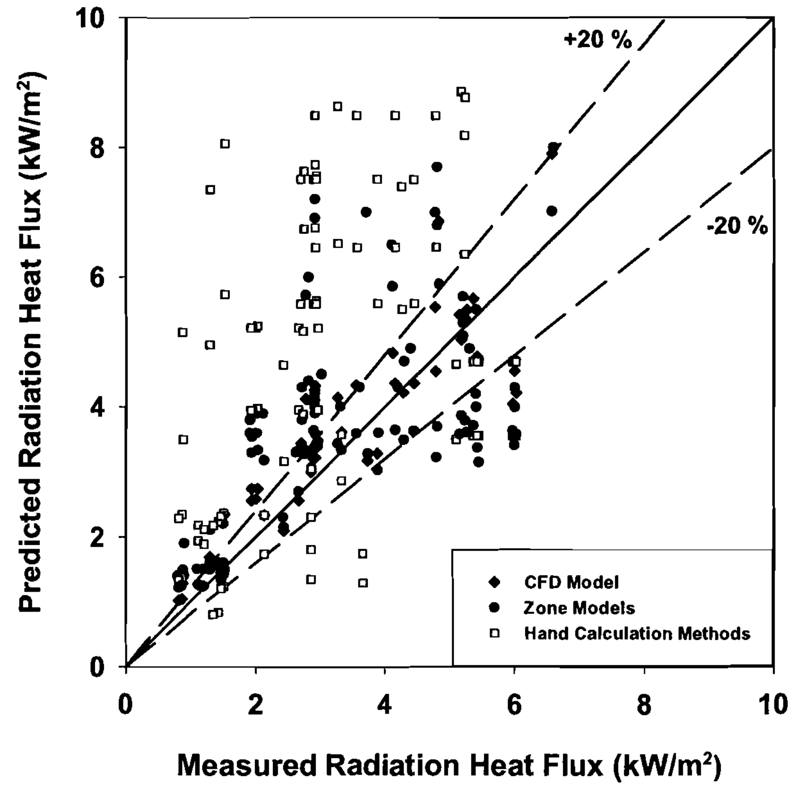


CFD



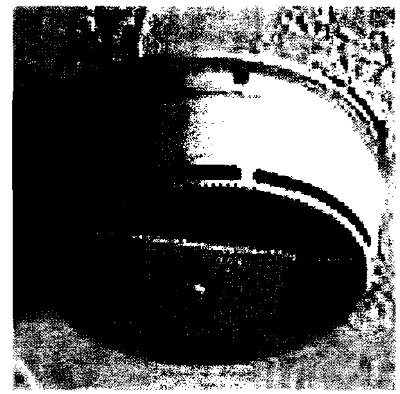
Results of NRC V&V (NUREG 1824)





From *Fire Protection Engineering*, Spring 2007

Simple Response Models in Fire



$$\frac{dT_l}{dt} = \frac{\sqrt{|\mathbf{u}|}}{\mathrm{RTI}}(T_g - T_l)$$

$$\frac{dY_c}{dt} = \frac{Y_e(t) - Y_c(t)}{L/\mathbf{u}}$$

Solve for link temperature using velocity **u** and gas temperature from Fire Model. The RTI (Response Time Index) is unique to each sprinkler.
Source: Gunnar Heskestad, Factory Mutual

Solve for smoke chamber concentration using external smoke concentration and velocity **u** from Fire Model. L is a length scale unique to each detector.

THIEF Model

1.5 kJ/kg/K

0.2 W/m/K

$$\rho c \frac{\partial T}{\partial t} = \frac{1}{r} \frac{\partial}{\partial r} k r \frac{\partial T}{\partial r}$$

Mass per unit length/Area

$$k \frac{\partial T}{\partial r}(R,t) = \dot{q}''$$

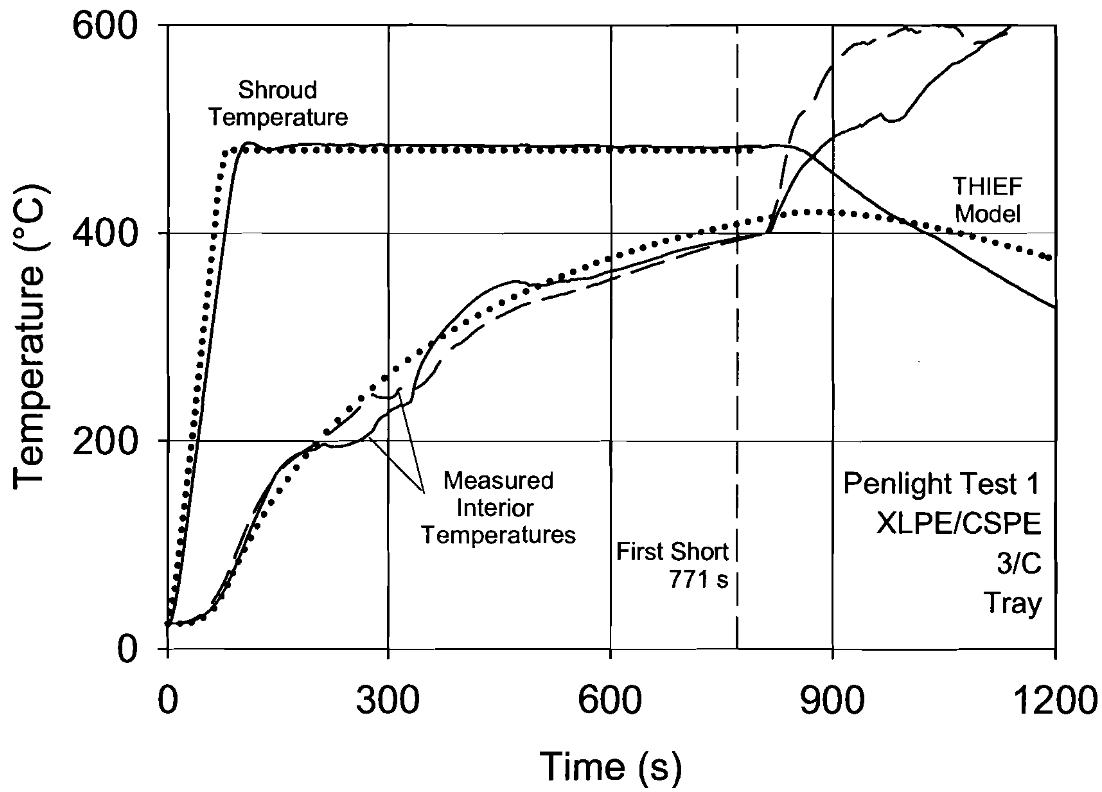
Predicted by Fire Model

- 1-D heat conduction for cable *T*(r,t)
- Homogenous cylinder, i.e. no layers
- Constant thermal conductivity (k)
- Constant specific heat (c)
- Bulk density (ρ) determined from mass and diam.
- Failure temperature obtained experimentally

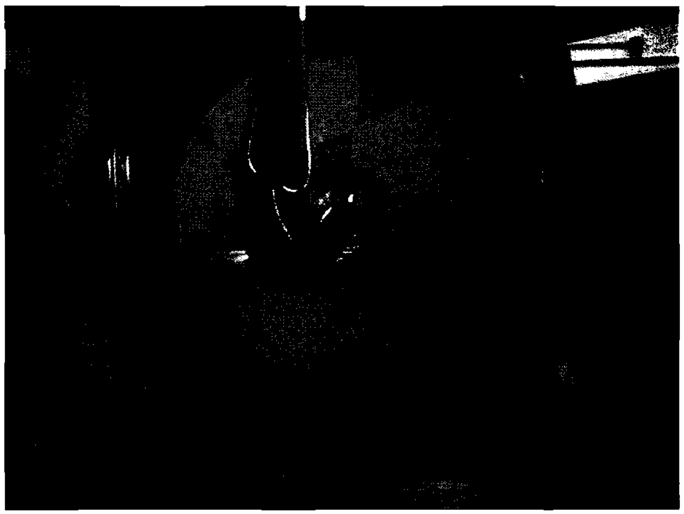
Source: Andersson and Van Hees, SP Fire, Sweden.

Penlight Results

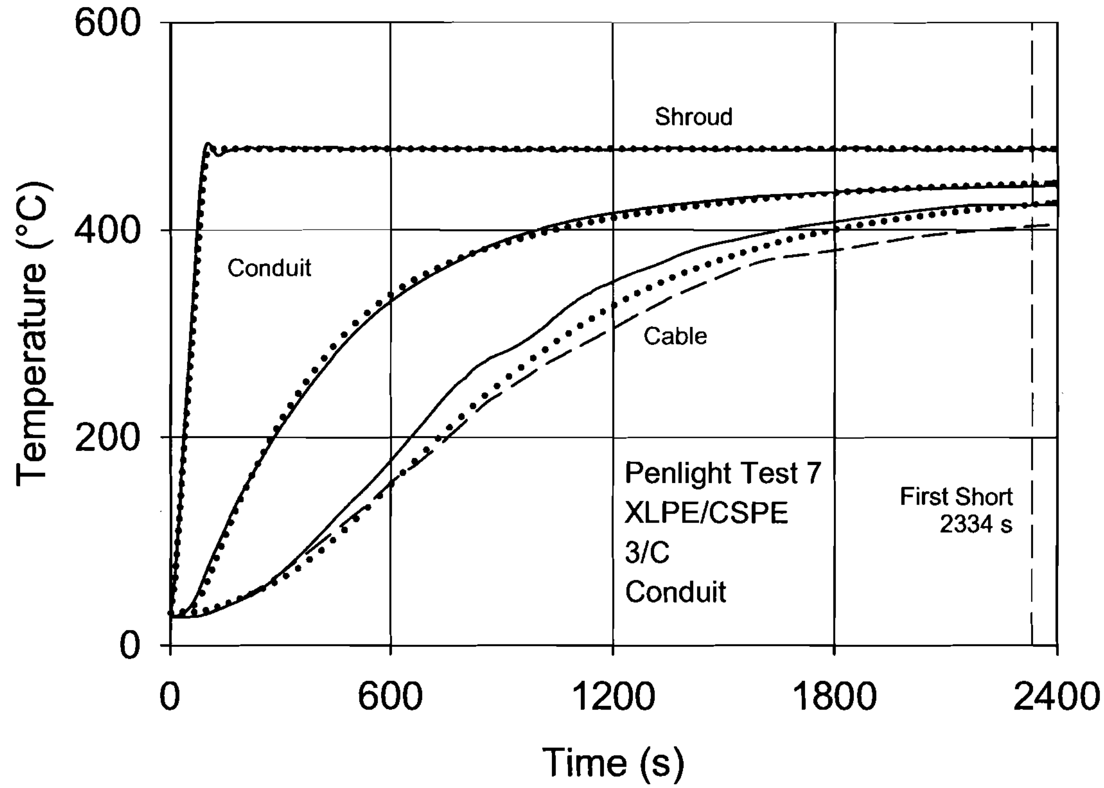
Courtesy S Nowlen and F Wyant
Sandia National Laboratory



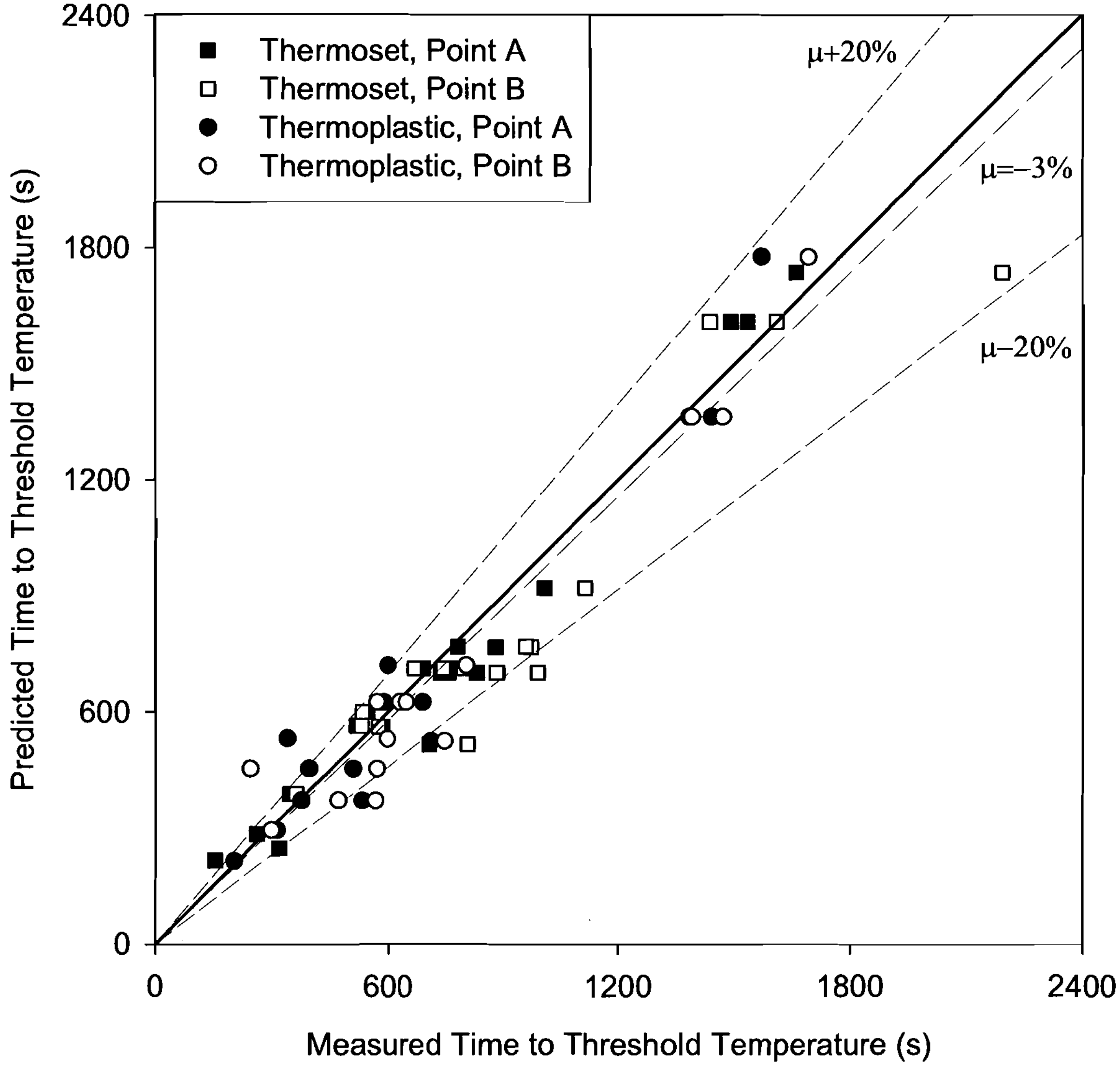
Conduit in Penlight



Courtesy S Nowlen and F Wyant
Sandia National Laboratory



Summary of Penlight Results

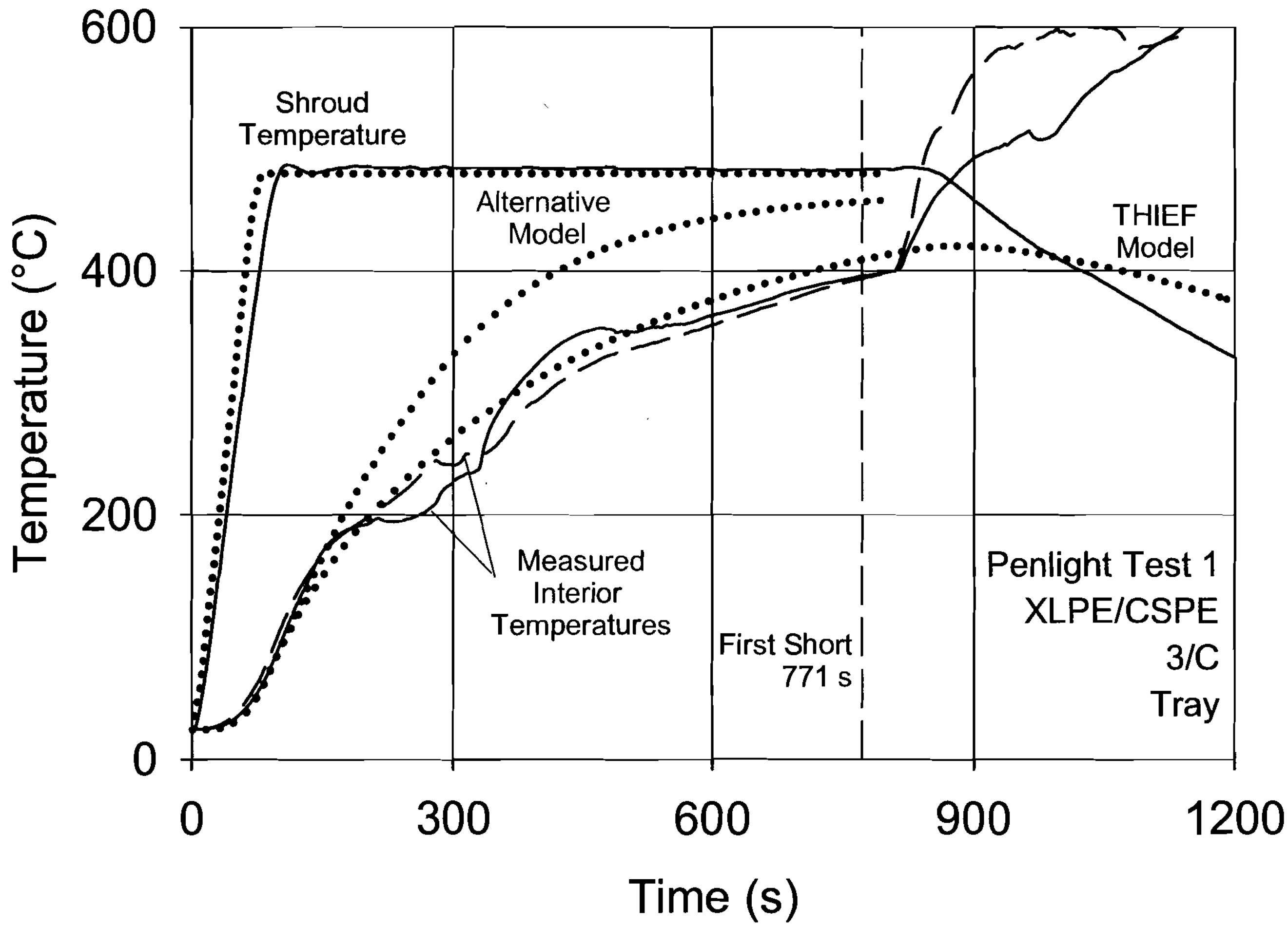


Why does THIEF work?

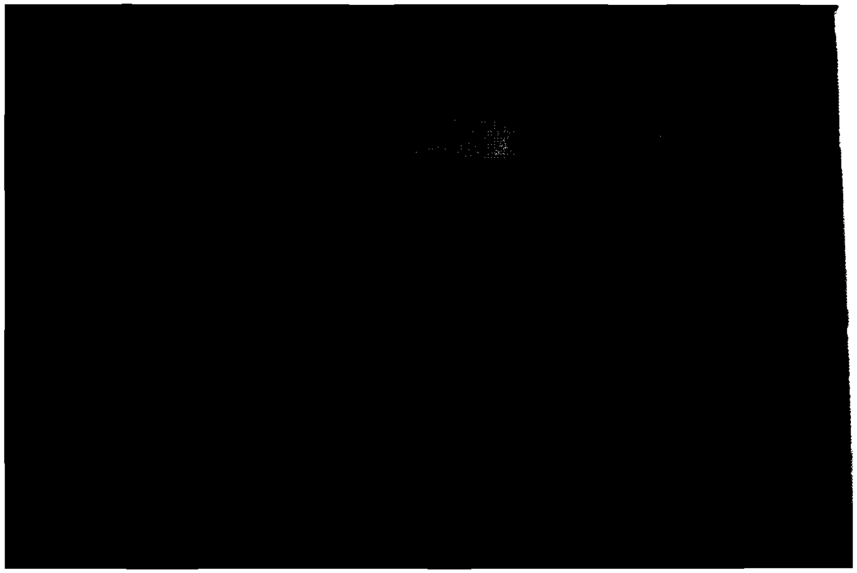
Specific Heat
Copper: 0.4 kJ/kg/K
Polymer: 1.5 kJ/kg/K

Density
Copper: 8960 kg/m³
Polymer: 1380 kg/m³

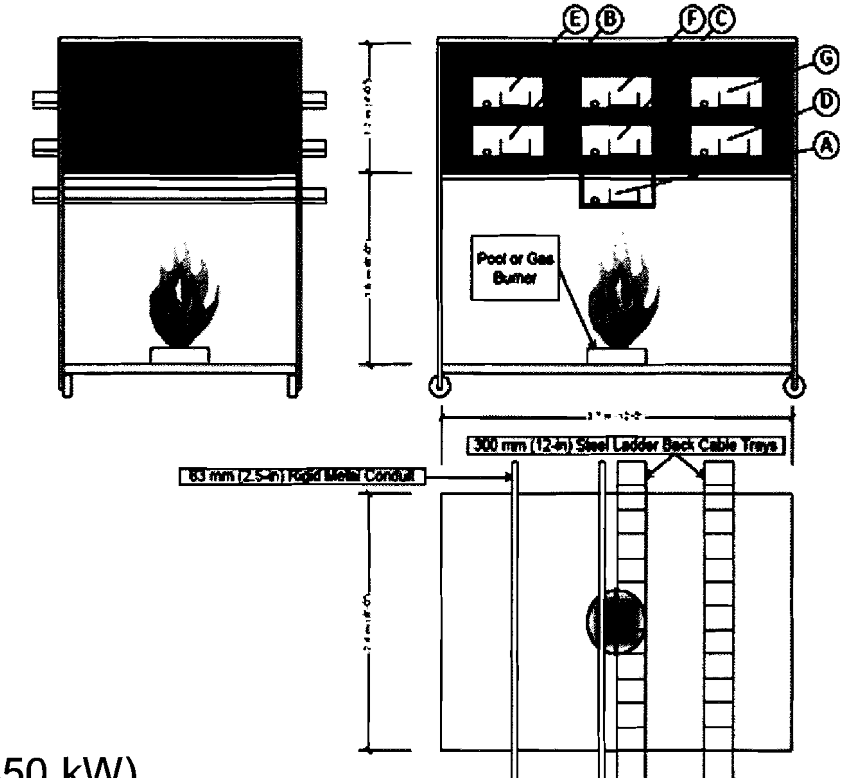
Alternative Model
Two layers: Polymer jacket around a polymer/copper mixture



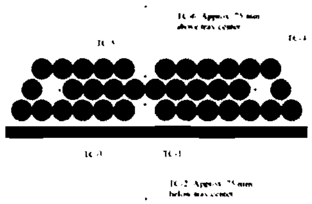
Intermediate-Scale Tests

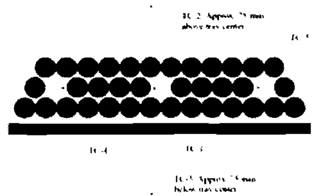


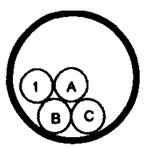
Courtesy Steve Nowlen and Frank Wyant, Sandia National Labs

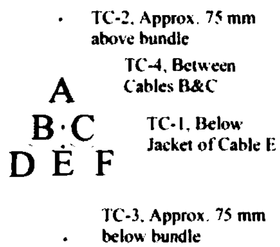


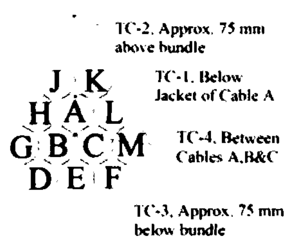
- Less controlled, but a more realistic scale
- Hood is roughly the size of a typical ASTM E 603 type room fire test facility
- Propene (Propylene) burner fire (200 kW to 350 kW)
- Cables in trays, conduits and air drop

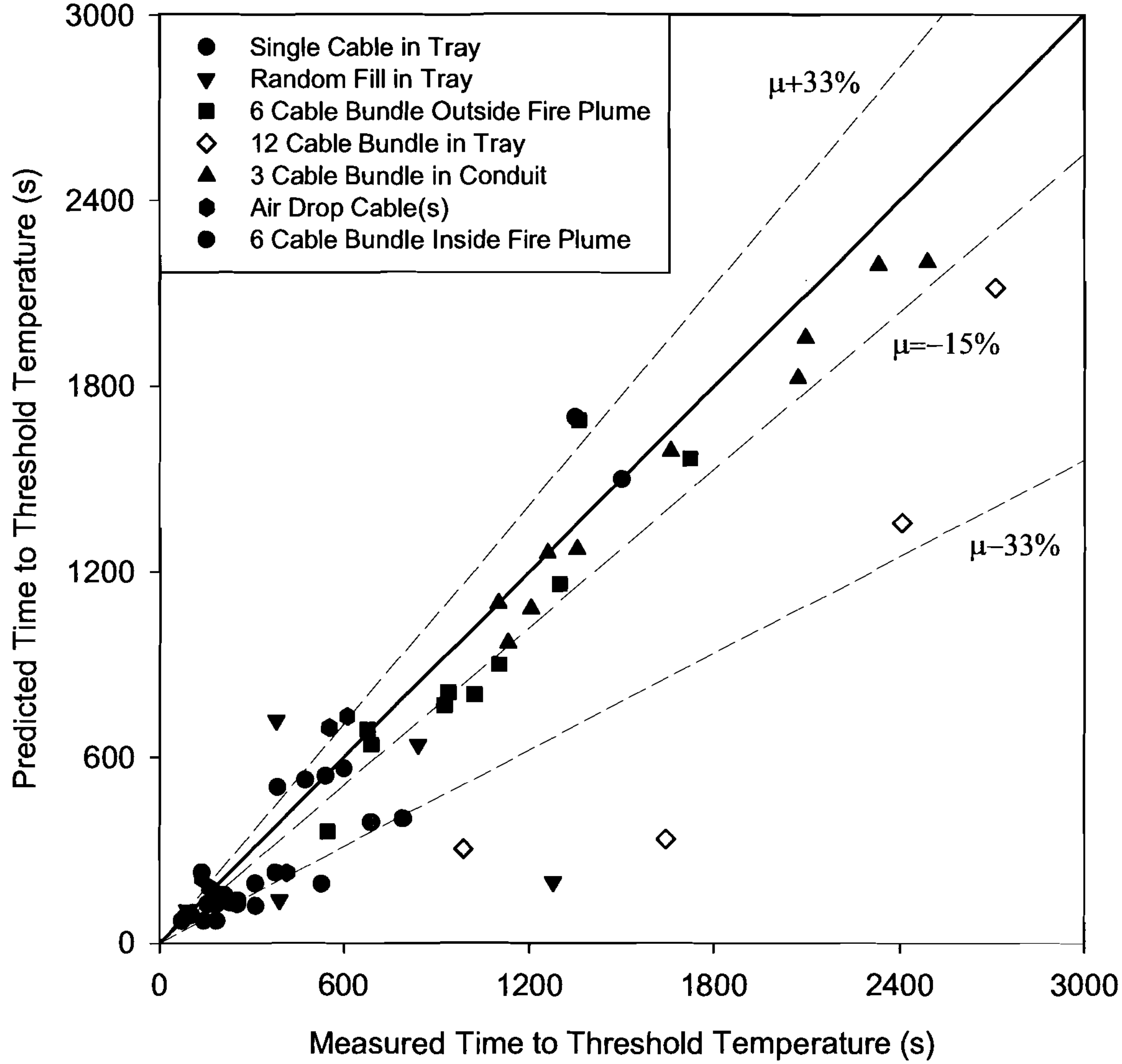





1
A
B
C


TC-2, Approx. 75 mm above bundle
TC-4, Between Cables B&C
A
B C
D E F
TC-1, Below Jacket of Cable E
TC-3, Approx. 75 mm below bundle


TC-2, Approx. 75 mm above bundle
J K
H A L
G B C M
D E F
TC-1, Below Jacket of Cable A
TC-4, Between Cables A,B&C
TC-3, Approx. 75 mm below bundle


Single Cable in Tray
Random Fill in Tray
6 Cable Bundle Outside Fire Plume
12 Cable Bundle in Tray
3 Cable Bundle in Conduit
Air Drop Cable(s)
6 Cable Bundle Inside Fire Plume
μ+33%
μ=−15%
μ−33%
Predicted Time to Threshold Temperature (s)
Measured Time to Threshold Temperature (s)
0
600
1200
1800
2400
3000

Summary

- The THIEF (Thermally-Induced Electrical Failure) model is simple because of limited thermophysical cable properties and limited accuracy in fire model calculations

- The THIEF model is currently being implemented in the FDTs (NRC spreadsheet-based fire calculations), CFAST (NIST zone model), and FDS (NIST CFD model).

CREDIT FOR CONTAINMENT ACCIDENT PRESSURE IN DETERMINING AVAILBLE NPSH

Richard Lobel
Office of Nuclear Reactor Regulation
February 8, 2008

CREDIT FOR CONTAINMENT ACCIDENT PRESSURE

- <u>PURPOSE</u>

- Brief review of history and applicable NRC regulations and guidance related to the use of containment accident pressure in determining the available NPSH of ECCS and containment heat removal pumps

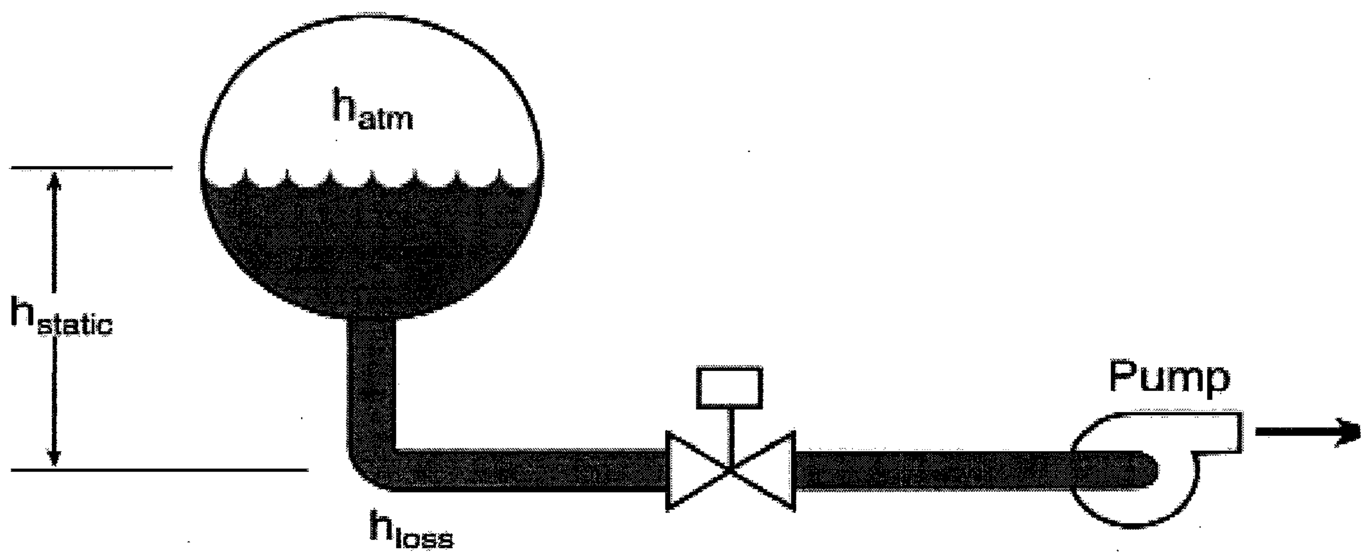
CREDIT FOR CONTAINMENT ACCIDENT PRESSURE

- Introduction

- Draft RG 1.82 Revision 4: An acceptable approach would quantify the uncertainty in NPSH calculations
- Discussions with BWROG
- NRC staff briefed on proposed BWROG method at October 2007 meeting.

CREDIT FOR CONTAINMENT ACCIDENT PRESSURE

Available NPSH = $h_{atm} + h_{static} - h_{loss} - h_{vp}$



CREDIT FOR CONTAINMENT ACCIDENT PRESSURE

- BACKGROUND-1

- Some early reactors licensed crediting containment accident pressure for NPSH
- Regulatory Guide 1.1: (1970)No credit for increase in containment accident pressure
- Regulatory Guide (RG)1.82 Revision 0: (1974) 50% blockage
- USI A-43: RG 1.82 Rev. 1: (1985) LOCA – debris blockage, air entrainment, sump design

CREDIT FOR CONTAINMENT ACCIDENT PRESSURE

- <u>BACKGROUND-2</u>

- Bulletin 96-03: RG 1.82 Rev. 2 (1996) BWR strainer guidance
- Generic Letter (GL) 97-04 (1997) Requested information on crediting containment accident pressure. Resulted in revisions to NPSH analyses for some plants.
- Bulletin 2001-03 (GSI 191): RG 1.82 Rev. 3 (2003)
  - No credit for containment accident pressure
  - Acceptable for certain operating reactors when design "cannot be practicably altered"

CREDIT FOR CONTAINMENT ACCIDENT PRESSURE

- Staff Position:
  - Credit for containment accident pressure in determining available NPSH is allowed when:
  - (1) analysis has conservatively demonstrated that sufficient pressure is available for design basis accidents, and
  - (2) for beyond design basis accidents, an acceptable level of safety is maintained

CREDIT FOR CONTAINMENT ACCIDENT PRESSURE

- <u>STATUS</u>

- Plants crediting containment accident pressure:
  - 18 BWRs (Mark I containments)
  - 10 PWRs (5 Subatmospheric containments)*

- Standard Review Plan Section 6.2.2 allows credit for containment accident pressure during the LOCA injection phase

CREDIT FOR CONTAINMENT ACCIDENT PRESSURE

- CREDIT IN OTHER REGULATIONS
  - 10 CFR 50.46 Containment pressure must be conservatively minimized
  - Dose calculations assume leakage at La (< 1percent mass/24 hours)
  - ATWS, Station Blackout and Appendix R (Fire) acceptance criteria require demonstration of containment integrity by satisfying containment pressure and temperature design limits

CREDIT FOR CONTAINMENT ACCIDENT PRESSURE

- ACCEPTABILITY OF CREDIT FOR CONTAINMENT ACCIDENT PRESSURE BASED ON:
  - High confidence in containment integrity
  - Conservative calculations
  - Design of emergency pumps
  - No significant impact on emergency operating procedures
  - Minimal impact on plant risk

Containment Overpressure Credit for NPSH

ACRS/NRC/BWROG Meeting

NRC Headquarters

Rockville, MD

February 8, 2008

Alan Wojchouski (NMC)
BWROG COP Committee Chairman

Purpose of Presentation

- Present background, objectives and work scope
- Provide overview of the Licensing Topical Report
- Describe how the LTR address ACRS concerns with granting containment overpressure credit

Background

- In late 2005, NRC requested BWROG to provide information that could be used by staff to address ACRS issues with approval of containment overpressure credit for NPSH

- Committee was approved by BWROG Executives in May 2006

Background

- BWROG Objective
  - Develop guidance for NRC approval of credit for containment overpressure where practical alternative approaches do not exist
    - Define conservatisms in methodology
    - Assess safety implications
    - Define reasonable and consistent requirements and methods

Results

- For DBA LOCA and Special Events, both the change in CDF and the change in LERF fall within the RG 1.174 "very small" risk increase region

- Deterministic (current licensing basis) approach gives a conservative assessment of NPSHa

- Statistical (realistic) approach demonstrates margin inherent in deterministic approach

- Low pressure ECCS performance not dependent on containment integrity

- Pumps have been shown to survive periods of operation when the NPSHa was below NPSHr

Work Scope

- Identify example plant – Monticello
- Review containment analysis inputs and methods for conservatisms
- Perform sensitivity study to assess impact of input parameters on containment response
- Identify input parameters in the example plant NPSH analysis that can be changed to minimize containment overpressure credit (COP)

Work Scope - Continued

- Perform containment analyses for example plant
  - Develop methodology
    - Licensing basis inputs – deterministic
    - Realistic inputs – statistical
    - Compare results

- Perform risk assessment using results of realistic analysis

- Assess effect of credit for containment overpressure on special events (i.e., Appendix R, SBO, ATWS)

Overview of Methodology

- Calculate NPSHa without COP (deterministically)
  - Conservative assumptions, for DBA LOCA and special events
  - Determine wetwell pressure so NPSHa = NPSHr

- If NPSHa without COP is lower than NPSHr,
  - Ensure deterministic NPSHa with COP is higher than NPSHr
  - Evaluate statistically (Monte Carlo)
    - This provides realistic evaluation of the event in support of COP request based on the deterministic calculations

NPSH Overview

- Available NPSH can be expressed as

$$NPSHa = [(Pww - Pv) \times 144/\rho_w] + [Hpool - Hpump - Hloss]$$
$$= \quad Hww \quad + \quad Hpl$$

Where:

NPSHa	Available NPSH for pump (ft)
Pww	Wetwell airspace pressure (psia)
Pv	Saturation vapor pressure at suppression pool temperature (psia)
ρ_w	Density of suppression pool water (lbm/ft3)
Hpool	Elevation of suppression pool surface (ft)
Hpump	Elevation of pump suction (ft)
Hloss	Suction strainer and suction line losses from suppression pool to pump (ft)

Deterministic Approach

- Current licensing basis accident scenarios with applicable limiting single failures are used in the NPSHa determination
  - Bounding values for containment initial conditions
  - Resulting pool temperature response is maximized and the available wetwell pressure is minimized
- This approach will give a conservative assessment of NPSHa

Statistical Approach

- Takes credit for variabilities in the analysis input values
- The order statistics method is employed
  - Input variabilities are defined statistically and combined through a Monte Carlo process
  - 59 random draws are made from the corresponding probability distributions to achieve 95/95. Containment pressure and temperature time-histories are calculated for the 59 cases
- Allows for calculating more realistic NPSHa values, which can be used to quantify the conservatism in the deterministic analysis

DBA LOCA Approaches

- Deterministic approach: Uses either the maximum or the minimum value for each input parameter
  - Depends upon which direction is conservative

- Statistical approach: All the input parameters will not be at their extreme (maximum or minimum) values at the same time
  - For the statistical approach with realistic assumptions, input parameters that can be statistically defined are selected

Statistical Approach

- The following input parameters were statistically varied:
  - Initial reactor power
  - Decay heat value after reactor SCRAM
  - Initial suppression pool temperature
  - Service water (ultimate heat sink) temperature
  - RHR heat exchanger heat removal capability
  - Initial suppression pool volume
  - Initial drywell temperature
  - Initial drywell pressure
  - Initial wetwell pressure
  - Initial containment leakage rate

Statistical Approach

- Value of Hww is calculated as a function of time for each of the multiple 59 trials (calculations), based on outputs of
  - Pool temperature
  - Pool volume (height)
  - Wetwell airspace pressure

- From the set of 59 time-histories, the minimum values of Hww are obtained as a function of time, and the resulting minimum values are used as 95/95 values

Effects of Reduced NPSH

- The effects of reduced NPSHa below the NPSHr will cause increased cavitation and reduction in the total dynamic head of the pump.
  - The effects will be flow surging, increased noise and vibration levels at the pump.
  - As the NPSHa is further reduced, a condition called head collapse will be entered
    - This condition is where the percentage of liquid that is in vapor phase is so great that pump flow ceases
- Pump tests were performed for extended periods where the NPSHa was substantially below NPSHr
  - Pumps were shown to recover after NPSHa was restored
  - No visible damage was noted after running for extended periods and after head collapse

Risk Assessment

- The risk analysis assesses the impact on plant risk if containment accident pressure is assumed not present (e.g., postulated pre-existing primary containment failure) during the postulated accident scenarios such that inadequate LP ECCS pumps NPSH occurs

- The DBA-LOCA risk analyses presented are sufficiently generic and conservative such that the results are applicable to the BWR fleet. Non-LOCA events are also considered in this analysis in a simplified fashion to bound the BWR fleet.

Risk Assessment Conclusions

- The risk impact results for the example BWR plant for COP credit for DBA-LOCAs are
  - Δ CDF = 9.0E-9 /year
  - Δ LERF = 9.0E-9 /year
- Both the change in CDF and the change in LERF fall within the RG 1.174 "very small" risk increase region
- Even with inclusion of Special Events and External Events, the risk impact is still "very small"

Example Plant Analysis

- Monticello plant-specific data was provided to GE for NPSH analysis
  - Five years of plant data for eight input parameters and probability distribution for each parameter

- Plant specific Containment DBA-LOCA NPSH analysis completed
  - Three scenarios analyzed
    - Short term < 600 Seconds (using limiting single failure)
    - Long Term > 600 seconds (using limiting single failure)
    - Containment overpressure failure
  - Each in two ways
    - Deterministic approach (standard licensing basis analysis)
    - Statistical approach (Monte Carlo analysis)

Figure A-1 Comparison of Suppression Pool Temperature for Short-term DBA-LOCA (with Loop Selection Logic Failure) between Deterministic Analysis and Statistical Analysis

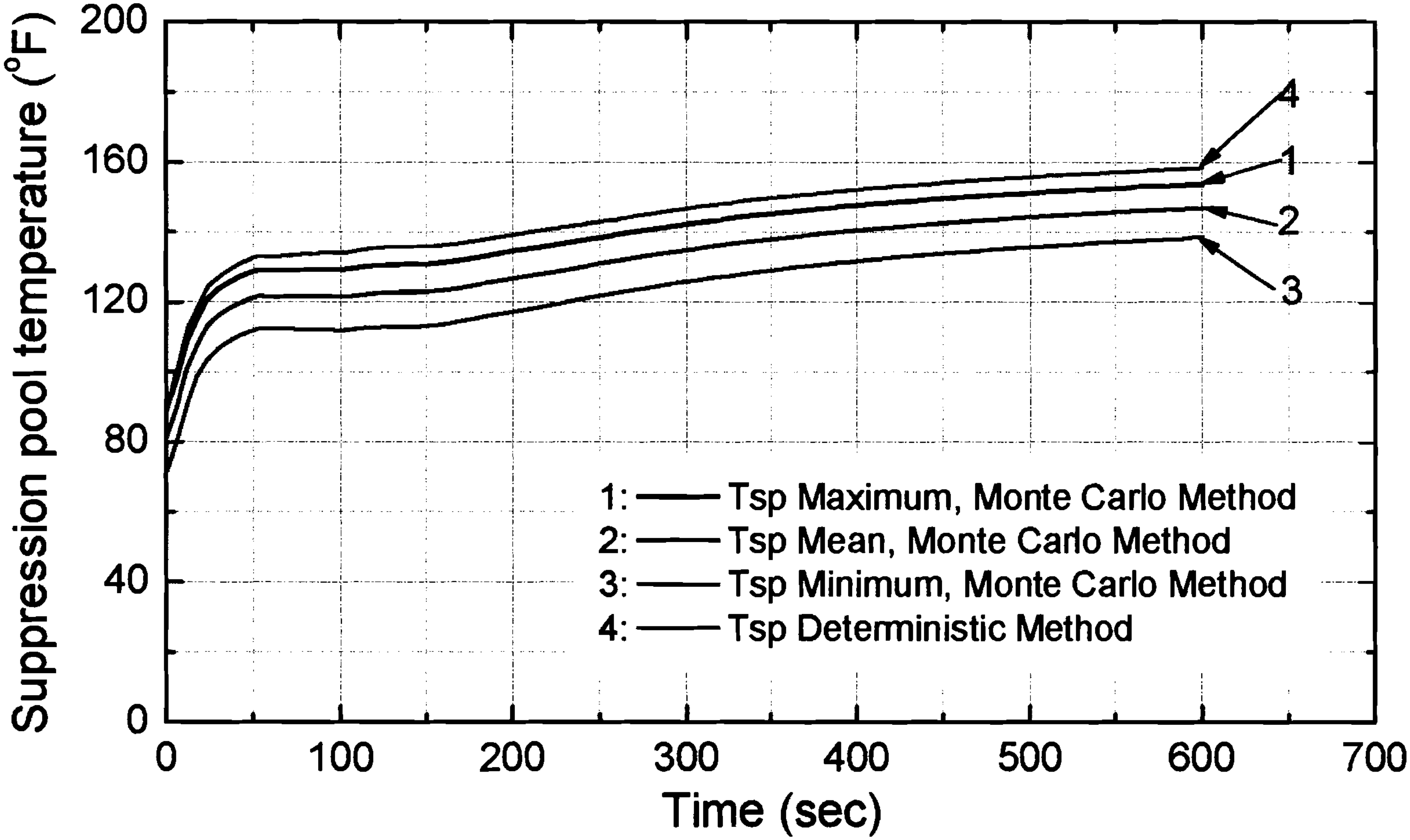
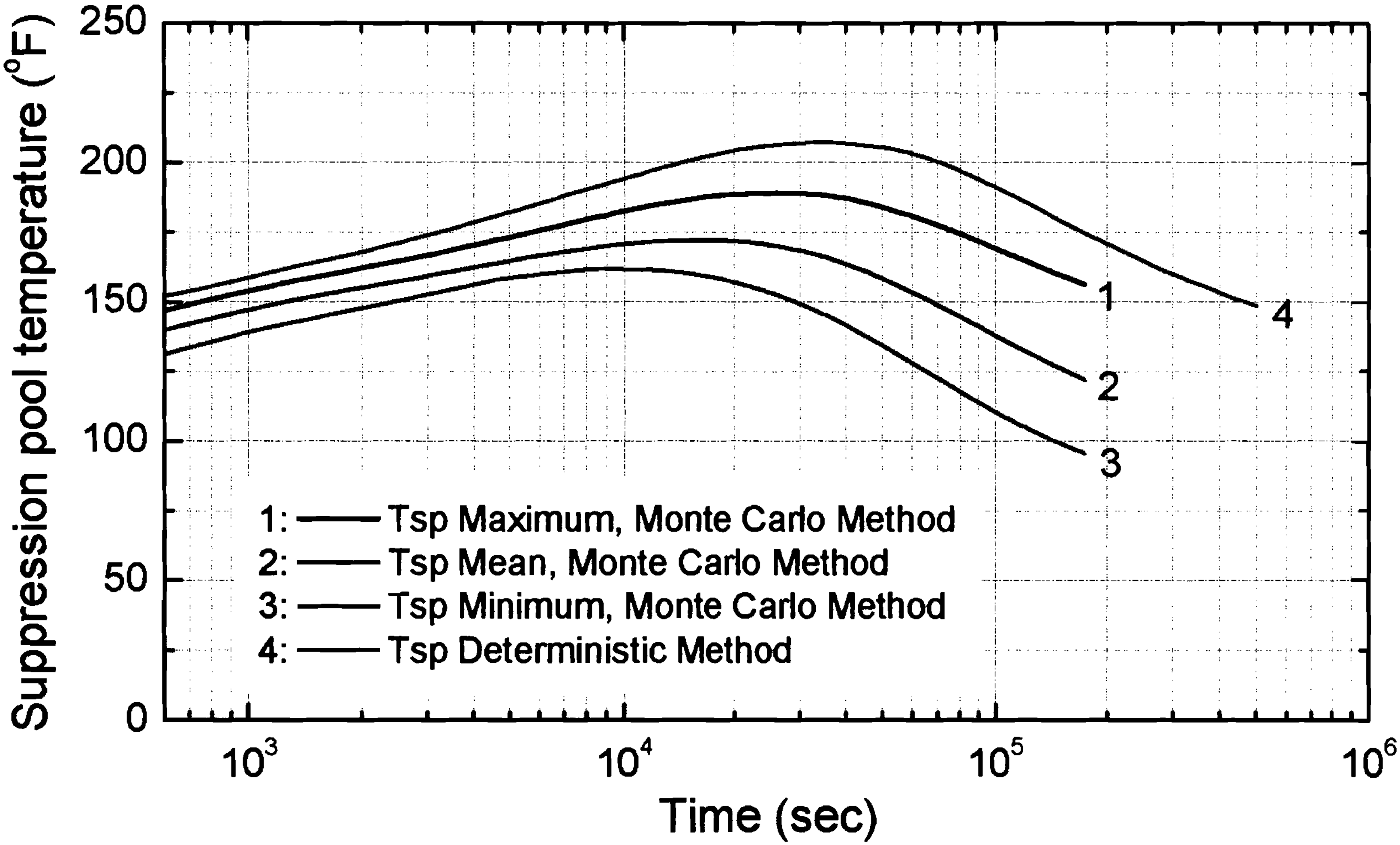
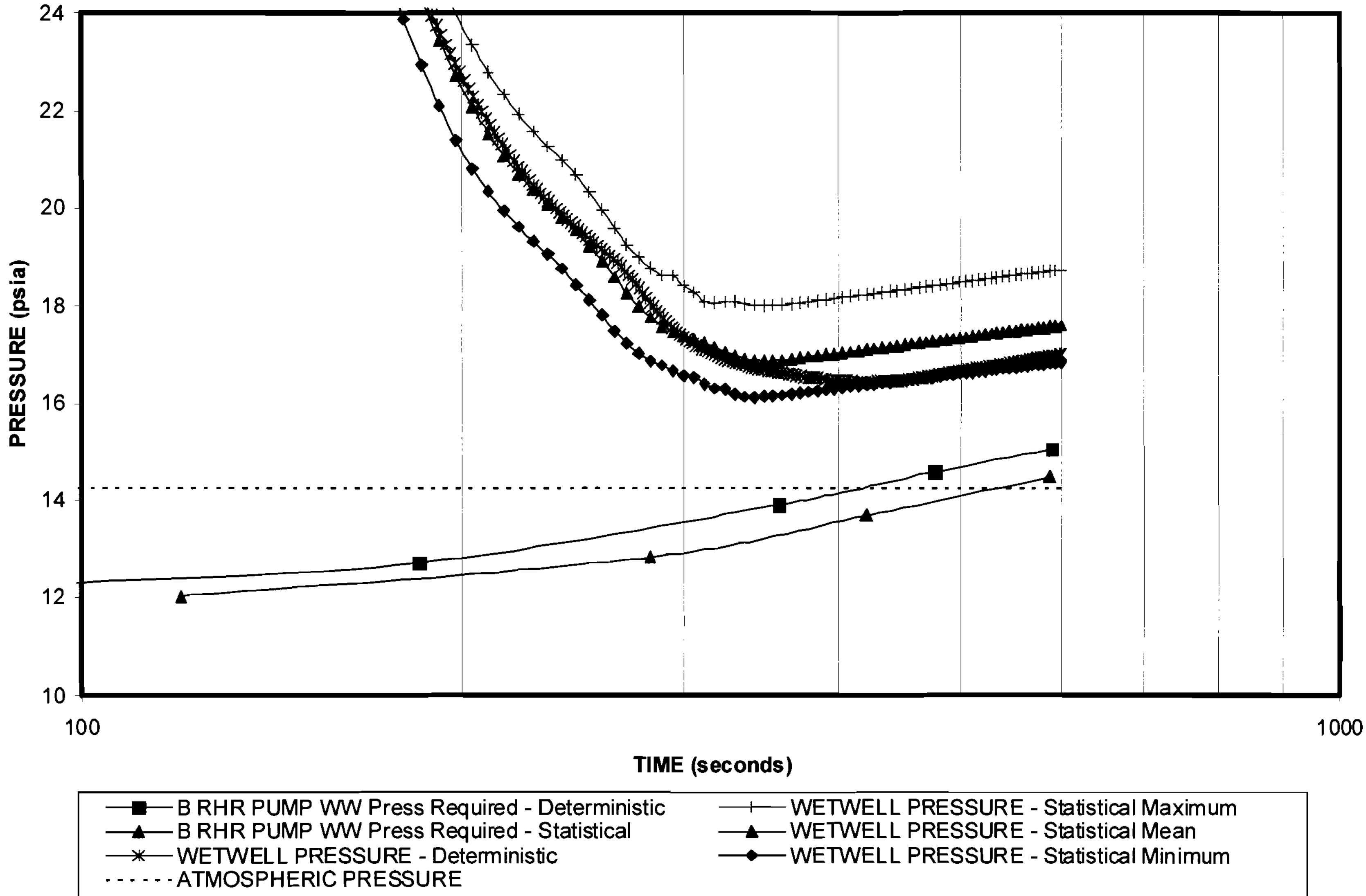


Figure A-2 Comparison of Suppression Pool Temperature for Long-term DBA-LOCA (with Diesel Generator Failure) between Deterministic Analysis and Statistical Analysis



RHR CONTAINMENT PRESSURE REQUIRED FOR ADEQUATE NPSH DURING THE SHORT TERM PHASE OF DBA LOCA (LPCI LOOP SELECTION FAILURE, OFFSITE POWER AVAILABLE AND DEBRIS LOADING ON SUCTION STRAINERS)



RHR CONTAINMENT PRESSURE REQUIRED FOR ADEQUATE NPSH DURING THE LONG TERM PHASE OF DBA LOCA
(11 DG FAILURE, LOOP AND DEBRIS LOADING ON SUCTION STRAINERS)

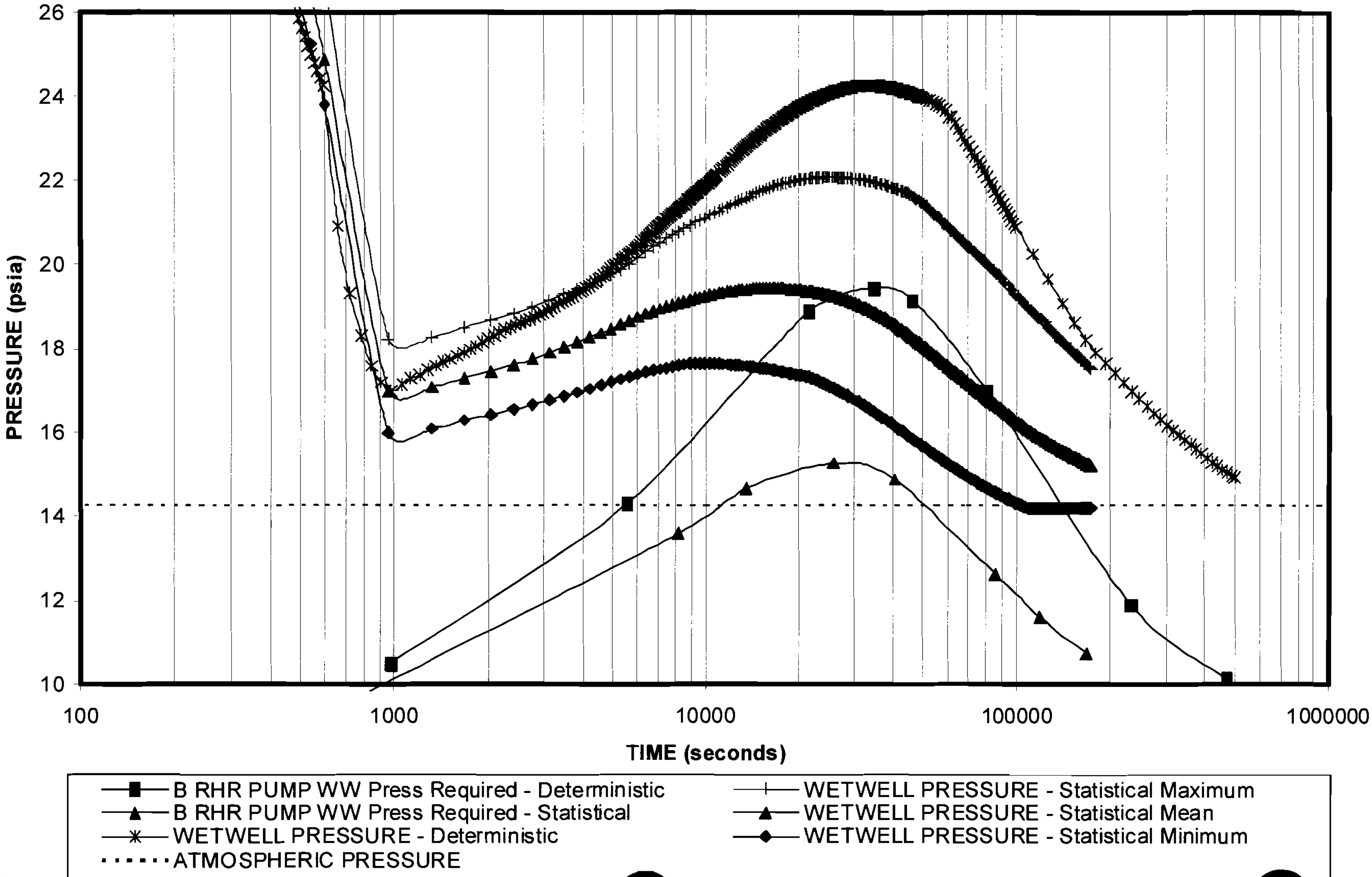
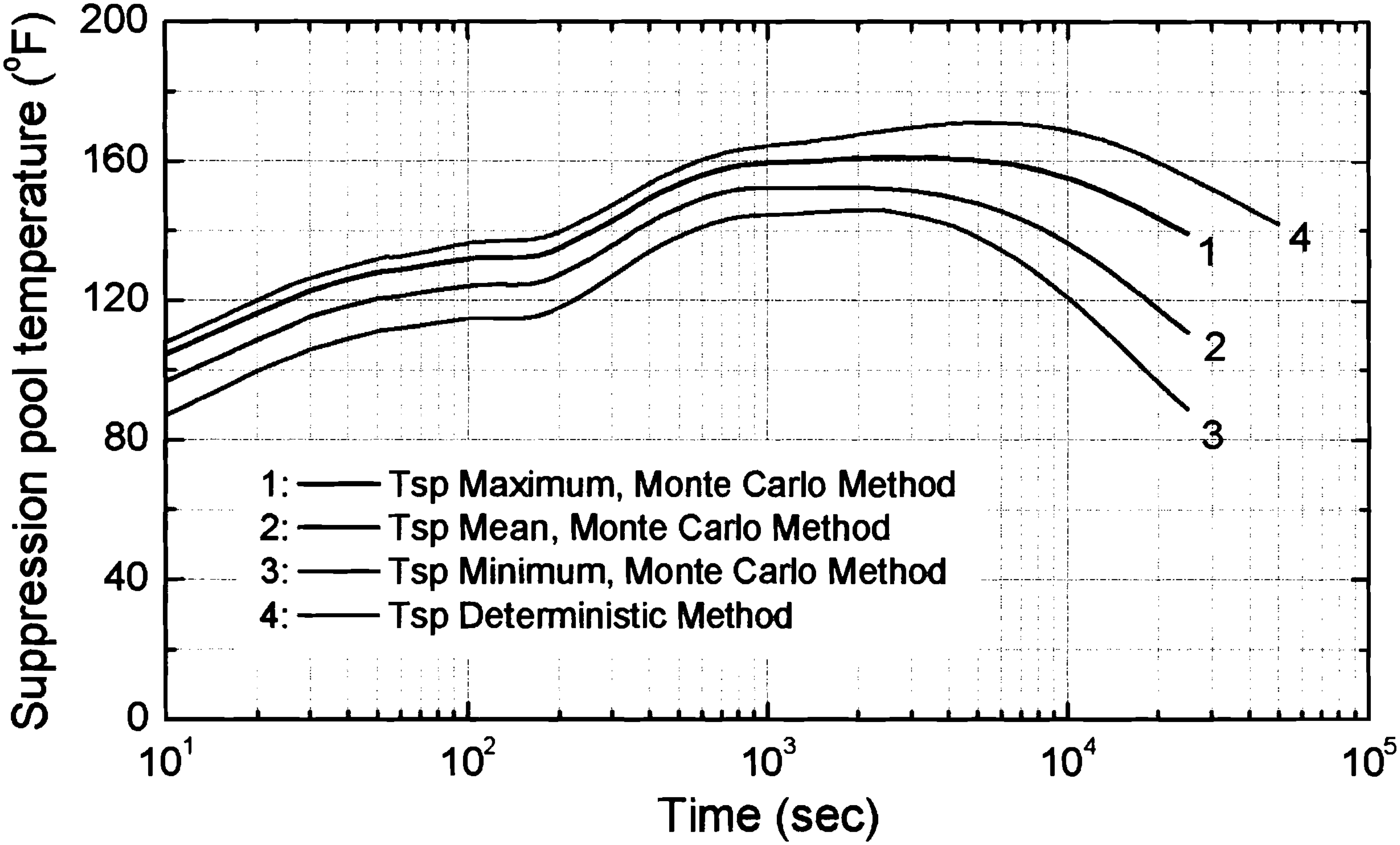


Figure B-1 Suppression Pool Temperature Response to DBA-LOCA with All Safety Systems Available for Case of No Containment Overpressure



Special Events

- NPSH methodology for special events (ATWS, SBO, Appendix R) is presented in the LTR
  - Brief descriptions of each of the special events
  - Similarities and contrasts to the DBA-LOCA NPSH analyses
  - Identified conservatisms in Special Event NPSH evaluations

Special Events

- The NPSHa determinations will be completed on a plant-specific basis
  - Expected that the deterministic approach utilizing nominal input values will be used to calculate NPSHa for special events
  - Should this approach show that NPSHa < NPSHr, then the statistical approach utilizing the mean output values will be used to show the expected realistic response to the event

Conclusions

- The change to CDF and LERF due to crediting COP is "very small"

- If containment integrity is not available, the ECCS can realistically perform its intended safety function

Thank you for your attention

12

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ACRS MEETING HANDOUT

Meeting No.	Agenda Item	Handout No.:
549th	10	10.1

Title **RECONCILIATION OF ACRS COMMENTS AND RECOMMENDATIONS**

Authors

SAM DURAISWAMY

List of Documents Attached

See attached list

10

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

From Staff Person
SAM DURAISWAMY

SUBJECT	ANALYSIS	EDO LTR.	ACRS LTR.
Interim Letter: Southern Nuclear Operating Company Application For the Vogtle Early Site Permit And The Associated NRC Safety Evaluation Report With Open Items (DAP/DCF)	01/02/08 (pp. 1-2)	12/28/07 (p. 3)	11/20/07 (pp. 4-7)
Staff's Implementation Of Lessons Learned From Reviews of Early Site Permit Applications (DAP/DCF)	01/02/08 (p. 8-9)	12/27/07 (pp. 10-11)	11/19/07 (pp. 12-15)
Draft Final Generic Letter 2007-02, "Managing Gas Accumulation in Emergency Core Cooling Decay Heat Removal, and Containment Spray Systems" (SAK/DEB)	01/25/08 (pp. 16-17)	12/06/07 (p. 18)	10/19/07 (pp. 19-20)
Draft Final NUREG-1829, "Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process," and Draft NUREG-XXXX, "Seismic Considerations for the Transition Break Size" (GEA/GSS)	02/06/08 (pp. 21-22)	01/30/08 (pp. 23-24)	12/20/07 (pp. 25-29)
Chapter 2, 5, 8, 11, 12, and 17 of the NRC Staff's Safety Evaluation Report with Open Items Related to the Certification of the ESBWR Design (MLC/CGH)	02/07/08 (pp. 30-32)	02/01/08 (pp. 33-36)	11/20/07 (pp. 37-41)
AREVA Detect and Suppress Stability Solution and Methodology (SAK/ZA)	02/08/08 (pp. 42-43)	01/30/08 (pp. 44-46)	12/27/07 (pp. 47-51)

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

January 2, 2008

MEMORANDUM TO: Dana Powers, Chair
Early Site Permits Subcommittee

FROM: David C. Fischer, Senior Staff Engineer David C. Fischer

SUBJECT: ANALYSIS OF EDO RESPONSE TO ACRS INTERIM LETTER: SOUTHERN NUCLEAR OPERATING COMPANY APPLICATION FOR THE VOGTLE EARLY SITE PERMIT AND THE ASSOCIATED NRC SAFETY EVALUATION REPORT WITH OPEN ITEMS

Attached is a copy of the EDO's December 28, 2007, letter of response to the ACRS's November 20, 2007, interim letter on Southern Nuclear Operating Company's (Southern Nuclear's) application for the Vogtle early site permit and the associated NRC safety evaluation report (SER) with open items. A copy of the Committee's letter is also attached.

Committee Letter

In its letter, the Committee concluded:

1. The staff has undertaken a thorough review and, where appropriate, independent analysis of the Vogtle early site permit application.

2. The staff has requested that the applicant further assess the post-construction hydrology of the site, the seismic hazard at the site, and weather extremes at the site. We support these requests for additional assessment.

3. The decision by the applicant to propose a specific nuclear power plant design in conjunction with the early site permit application has probably resulted in fewer permit conditions in the SER on the application.

EDO Response

The EDO's response stated that the staff is currently working to resolve several open items in the areas of meteorology, hydrology, geology, seismology, and emergency planning. The staff will prepare an SER with no open items and will provide this report to the ACRS. Following the ACRS meeting on the SER with no open items (tentatively scheduled for June 2008), the staff will address any potential issues raised by the Committee prior to issuing the SER. The staff indicated that the SER with no open items would include the staff's review of the applicant's limited work authorization (LWA-2) request which was submitted by SNC on August 15, 2007.

Analysis

The EDO response is satisfactory.

Attachments: As stated

cc: ACRS Members
C. Santos
S. Duraiswamy

December 28, 2007

Dr. William J. Shack, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

SUBJECT: INTERIM LETTER: SOUTHERN NUCLEAR OPERATING COMPANY APPLICATION FOR THE VOGTLE EARLY SITE PERMIT AND THE ASSOCIATED NRC SAFETY EVALUATION REPORT WITH OPEN ITEMS

Dear Dr. Shack:

Thank you for your letter dated November 20, 2007, regarding the safety evaluation report (SER) with open items on Southern Nuclear Operating Companys (SNC) early site permit (ESP) application for the Vogtle site. As discussed during the 547th meeting of the Advisory Committee on Reactor Safeguards (ACRS) on November 1, 2007, the staff is currently working to resolve several open items in the areas of meteorology, hydrology, geology, seismology, and emergency planning.

The staff will prepare an SER with no open items and will provide this report to the ACRS. Following the ACRS meeting on the SER, the staff will address any potential issues resulting from this meeting prior to issuance of the SER.

The staff would like to remind the ACRS that a limited work authorization (LWA) request was submitted by SNC on August 15, 2007, and is being reviewed in conjunction with the ESP application. The staff intends the SER with no open items to include staff's review of the LWA supplemental request.

The staff appreciates the ACRS' feedback on the SER with open items and looks forward to the next meeting in June 2008.

Sincerely,

/RA Martin J. Virgilio for/

Luis A. Reyes
Executive Director
for Operations

cc: Chairman Klein
Commissioner Jaczko
Commissioner Lyons
SECY

ACRSR-2275

November 20, 2007

Mr. Luis A. Reyes
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

SUBJECT: INTERIM LETTER: SOUTHERN NUCLEAR OPERATING COMPANY APPLICATION FOR THE VOGTLE EARLY SITE PERMIT AND THE ASSOCIATED NRC SAFETY EVALUATION REPORT WITH OPEN ITEMS

Dear Mr. Reyes:

During the 547th meeting of the Advisory Committee on Reactor Safeguards (ACRS), November 1-3, 2007, we began our review of the Vogtle[1] early site permit application and the associated safety evaluation report (SER) with open items prepared by the NRC staff. This matter was also reviewed by our Subcommittee on Early Site Permits on October 24, 2007. During these reviews, we had the benefit of discussions with representatives of the NRC staff and Southern Nuclear Operating Company (Southern Nuclear or "applicant"). We also had the benefit of the documents referenced. We review early site permit applications to fulfill the requirement of 10 CFR 52.23 that the ACRS report on those portions of an early site permit application that concern safety.

CONCLUSIONS

1. The staff has undertaken a thorough review and, where appropriate, independent analysis of the Vogtle early site permit application.

2. The staff has requested that the applicant further assess the post-construction hydrology of the site, the seismic hazard at the site, and weather extremes at the site. We support these requests for additional assessment.

3. The decision by the applicant to propose a specific nuclear power plant design in conjunction with the early site permit application has probably resulted in fewer permit conditions in the SER on the application.

DISCUSSION

The site currently occupied by Units 1 and 2 of the Vogtle Electric Generating Plant was approved originally for four units, but only two were built. The units now present at the site are 3,565 MWt Westinghouse pressurized water reactors. Also on the site is Plant Wilson which is a six-unit, oil-fueled combustion turbine facility.

---

[1] Vogtle is named for Alvin Ward Vogtle whose exploits in World War II were the inspiration for the character played by Steve McQueen in the movie The Great Escape.

Southern Nuclear has proposed to locate two Westinghouse AP1000 advanced nuclear power plants on the site. The AP1000 has a thermal power of 3,400 MWt. These power plants, designated Vogtle Units 3 and 4, will be located adjacent to and west of the existing Vogtle units. The early site permit application is unusual in that the applicant has selected a specific nuclear power plant design rather than relying on a plant parameter envelope as has been the case in previous applications for an early site permit. The applicant has also provided a complete and integrated emergency plan rather than providing only the major features of an emergency plan, as has been the case in previous early site permit applications.

Population in the Vicinity of the Site

The Vogtle site is located in rural Georgia approximately 15 miles east-northeast of Waynesboro, Georgia (population 5,813), and 26 miles southeast of Augusta, Georgia (population 195,182). Augusta, Georgia, is the population center nearest the site. Numerous small towns are located within 50 miles of the site. Only the town of Girard (population 227) is within 10 miles of the Vogtle site. The site is across the Savannah River from the Department of Energy's Savannah River Site, which has several thousand employees. There are several shutdown production reactors and active facilities for processing tritium and defense wastes at the Savannah River Site. The Department of Energy is proposing to construct the Mixed Oxide (MOX) Fuel Fabrication Facility on the Savannah River Site.

Based on 2000 census data, the combined resident and transient populations within 5 miles and within 10 miles of the site (aside from those working at the Savannah River Site) are 687 and 3,560, respectively. The population within 50 miles of the site is expected to approximately quadruple over the next 60 years but will not exceed an average of 500 people per square mile within 10 miles of the site.

Industrial Hazards in the Site Vicinity

With the exception of activities at the Department of Energy's Savannah River Site, there are no industrial activities of substance near the site. Hazardous material transport by rail and highway pose little threat to the site. The Savannah River is not used as a commercial transportation route at this time. Though there is a large military reservation in the vicinity of the site, projected activities do not pose significant threats to the nuclear power plant site.

Aircraft Hazard

A commercial airline route passes within 2 miles of the proposed site. Projected increases in traffic along this route are not sufficient to raise site hazards to the point of regulatory concern.

Meteorology

Weather at the Vogtle site is mild. Extreme cold and heavy winter precipitation are not common. Summers are hot with periods of stable ambient atmosphere. The applicant has based estimates of temperature extremes on a database covering a period of 30 years. In light of the duration of an early site permit (20 years) and the design life of any modern nuclear power plant constructed on the site (60 years), this appears to be an inadequate base of data for estimating temperature extremes. Moreover, the well known 50-year weather cycles along the east coast of the United States make the adequacy of the applicant's database even more dubious. The staff has asked the applicant to reassess the bases for estimates of weather extremes at the site.

Geology and Seismicity of the Site

The Vogtle site is located on the coastal plain below the Appalachian Piedmont. The ground is largely uncompacted sediments above the Blue Bluff Marl and compacted sands below the Blue Bluff Marl. Bedrock is at a depth of over 1000 feet. The Charleston seismic center poses the greatest threat to the site. The applicant has gone to great lengths to demonstrate that the Pen Branch Fault underlying the site is not a capable fault and does not contribute to the seismic threat to nuclear facilities on the site. The Eastern Tennessee Seismic Zone is about 200 miles from the site and poses only a modest threat to the facility.

The applicant has proposed to excavate to the Blue Bluff Marl and replace the natural materials with an engineered fill for the entire power block of each of the two proposed nuclear power plants. This is much as was done for Vogtle Units 1 and 2. The excavation and engineered fill relieve a number of erosion and seismic concerns. The applicant has relied to a large extent on the characterization of the Blue Bluff Marl done for Units 1 and 2 to characterize the basement material for Units 3 and 4. The staff has asked for more characterization of the Blue Bluff Marl immediately below the proposed locations for the new units.

The applicant has used the Electric Power Research Institute seismic hazard methodology. The applicant has updated the seismic hazard posed by the Charleston seismic zone including a significant increase in the frequency of large earthquakes to once every 500 years. Unfortunately, the Charleston seismic zone is not associated with a specific geological feature and consequently its precise location is not well known. The applicant has used a weighted average of possible regions for the seismic zone. The staff has identified data that suggest the seismic zone might be closer to the Vogtle site than considered by the applicant. Consideration of this data may move the centroid of seismic activity closer to the site and increase the seismic risk at the site. The staff has asked the applicant to provide additional information to support its conclusion that large earthquakes most likely do not occur further inland, closer to the Vogtle site.

The applicant did not update the characterization of the Eastern Tennessee Seismic Zone in the assessment of the seismic threat to the site. The staff has identified data that suggest an update of the Eastern Tennessee Seismic Zone should be done.

The estimate of local seismicity, aside from that caused by the Charleston seismic center, has been based on averaging several expert opinions. The staff questions the inclusion of one of the expert opinions in the analysis.

Hydrology

Failures of dams on the Savannah River could produce floods in the vicinity of the Vogtle site. Analyses performed by the applicant and reviewed by the staff show that conservative estimates of the maximum floods do not threaten the site.

Ground-water motion on the site will be affected by the construction of nuclear power plants on the site. The ground-water motion could affect transport of radionuclides. The applicant has analyzed the ground-water motion. The staff has, however, identified an alternative pathway for water flow and has asked the applicant to consider this alternative.

Emergency Plan

The applicant has developed an integrated emergency plan and provided revised evacuation time estimates. The staff has asked the applicant to ensure that local agencies review these time estimates since they may affect the actions of the agencies in the event of an emergency.

We conclude that the staff is preparing a quality SER on the Vogtle early site permit application and we look forward to reviewing the final application and SER.

ACRS member Professor Said Abdel-Khalik did not participate in the Committee's deliberations regarding this matter.

Sincerely,

/RA/

William J. Shack
Chairman

References:

1. U.S. Nuclear Regulatory Commission, Safety Evaluation Report With Open Items, "Safety Evaluation Report for the Vogtle Early Site Permit Application," August 30, 2007.

2. Southern Nuclear Operating Company, "Vogtle Early Site Permit Application," Revision 2, NRC Docket No. 52-00011, April 2007.

3. Report dated October 12, 2007, from William J. Hinze, Advisory Committee on Nuclear Waste and Materials, to Dana Powers, ACRS, "Review of Vogtle Early Site Permit Application and NRC's Safety Evaluation Report for the Vogtle Application."

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

January 2, 2008

MEMORANDUM TO: Dana Powers, Chair
Early Site Permits Subcommittee

FROM: David C. Fischer, Senior Staff Engineer David C. Fischer

SUBJECT: ANALYSIS OF EDO RESPONSE TO ACRS LETTER ON STAFF'S IMPLEMENTATION OF LESSONS LEARNED FROM REVIEWS OF EARLY SITE PERMIT APPLICATIONS

Attached is a copy of the EDO's December 27, 2007, letter of response to the ACRS's November 19, 2007, letter on the staff's implementation of lessons learned from reviews of early site permit (ESP) applications. A copy of the Committee's letter is also attached.

Committee Letter

In its letter, the Committee stated that the NRC staff has moved effectively to address within the regulatory process many of the lessons learned from the reviews of early site permit applications. In addition, the Committee said that the staff still needs to provide guidance to applicants on adequate measures to ensure the quality, integrity, and retrievability of data obtained from the Internet.

EDO Response

The staff expressed its appreciation of the ACRS' acknowledgment that it has "moved effectively to address within the regulatory process many of the lessons learned from the reviews of early site permit applications." The EDO response indicated that the staff will continue to communicate its expectations for early site permit applications during the Design-Centered Working Group meetings, public workshops, and other means, to ensure continued progress.

The EDO response stated that the staff conducted inspections to verify that the quality assurance programs governing early site permit applications met the applicable requirements of Appendix B to 10 CFR Part 50. These inspections also verified that effective controls were in place to provide reasonable assurance of the completeness and accuracy of data used in the applications consistent with 10 CFR 50.9, "Completeness and Accuracy of Information." However, the NRC staff agreed that additional clarification is warranted in existing regulatory guidance to clearly convey regulatory requirements relative to the completeness and accuracy of early site permit and combined operating license applications. In addition, the EDO response indicated that the staff will review its inspection procedures and review guidance to ensure that the quality, integrity, completeness, and accuracy of data obtained from internet sources are appropriately addressed.

In conclusion, the EDO's response states that the staff appreciates the insight the ACRS has provided and recognizes it as a valuable contribution to the NRC staff's continued success in reviewing new reactor applications.

Analysis

The EDO response is satisfactory.

Attachments: As stated

cc: ACRS Members
C. Santos
S. Duraiswamy

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

December 27, 2007

Dr. William J. Shack, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

SUBJECT: STAFF'S IMPLEMENTATION OF LESSONS LEARNED FROM REVIEWS OF EARLY SITE PERMIT APPLICATIONS

Dear Dr. Shack:

Thank you for your letter dated November 19, 2007, to Chairman Klein regarding the staff's implementation of lessons learned from reviews of early site permit applications during the 547th meeting of the Advisory Committee on Reactor Safeguards (ACRS). The U.S. Nuclear Regulatory Commission (NRC) staff expresses its appreciation of the ACRS' acknowledgment that it has "moved effectively to address within the regulatory process many of the lessons learned from the reviews of early site permit applications." These successes are a direct result of the common understanding developed with the applicants. The NRC staff will continue to communicate its expectations for early site permit applications during the Design-Centered Working Group meetings, public workshops, and other means, to ensure continued progress.

During a meeting with the NRC staff, the ACRS raised a concern regarding a previous recommendation for the NRC staff to develop guidance to ensure the quality, integrity, and retrievability of data obtained from the internet by Title 10 of the *Code of Federal Regulations* (CFR) Part 52 applicants. The NRC staff conducted inspections to verify that the quality assurance programs governing early site permit applications met the applicable requirements of Appendix B to 10 CFR Part 50. These inspections also verified that effective controls were in place to provide reasonable assurance of the completeness and accuracy of data used in the applications consistent with 10 CFR 50.9, "Completeness and Accuracy of Information."

To date, the NRC staff has not identified any issues related to the completeness and accuracy of data obtained from the internet and referenced in these applications. However, the NRC staff agrees that additional clarification is warranted in existing regulatory guidance to clearly convey regulatory requirements relative to the completeness and accuracy of early site permit and combined operating license applications. In addition, the NRC staff will review its inspection procedures and review guidance to ensure that the quality, integrity, completeness, and accuracy of data obtained from internet sources are appropriately addressed.

The NRC staff appreciates the insight the ACRS has provided and recognizes it as a valuable contribution to the NRC staff's continued success in reviewing new reactor applications.

Sincerely,

/RA Martin J. Virgilio for/

Luis A. Reyes
Executive Director
for Operations

cc: Chairman Klein
Commissioner Jaczko
Commissioner Lyons
SECY

ACRSR-2273

November 19, 2007

The Honorable Dale E. Klein
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

SUBJECT: STAFF'S IMPLEMENTATION OF LESSONS LEARNED FROM REVIEWS OF EARLY SITE PERMIT APPLICATIONS

Dear Chairman Klein:

At the conclusion of our review of the North Anna, Grand Gulf, and Clinton early site permit applications, we met with the NRC staff and representatives of some applicants to discuss lessons that had been learned during the review process and that might be applicable to the review of future early site permit applications and combined license (COL) applications. We reported to the Executive Director for Operations on this meeting in a letter dated September 22, 2006.

In a November 8, 2006 Staff Requirements Memorandum, resulting from the meeting with the ACRS, the Commission requested that as licensing under 10 CFR Part 52 continues, the Committee advise the Commission on effectiveness and efficiency of staff's implementation of lessons learned in areas it has reviewed, for example, the development of guidance documents for early site permit applications. During the 547th meeting of the Advisory Committee on Reactor Safeguards, November 1-3, 2007, we met with the NRC staff to review progress on implementation of the lessons learned in the regulatory process as well as the effectiveness and efficiency of such implementation. This matter was also discussed with the NRC staff at a meeting of our Subcommittee on Early Site Permits held on October 24, 2007. We are pleased to report to you the progress the staff has made on implementation of the lessons learned.

CONCLUSION AND RECOMMENDATION

The NRC staff has moved effectively to address within the regulatory process many of the lessons learned from the reviews of early site permit applications.

The staff still needs to provide guidance to applicants on adequate measures to ensure the quality, integrity, and retrievability of data obtained from the Internet.

DISCUSSION

The staff has made more progress than we would have expected in the implementation of the lessons learned from the review of early site permit applications. The lessons and synoptic accounts of staff actions are provided below.

Develop common understanding between the staff and applicants concerning expectations.

The staff has completed pertinent updates to NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants;" issued Regulatory Guide 1.206, "Combined License Applications for Nuclear Power Plants;" and has developed Office Instruction NRO-REG-100, "Acceptance Review Process for Design Certifications and Combined License Applications." Furthermore, the staff has been interacting with the nuclear industry and potential applicants through the Design-Centered Working Groups.

The staff has done much to facilitate the development of common understandings. This is a most important undertaking and will continue to need attention. An incomplete understanding of staff expectations by the applicant resulted in many requests for additional information and open items in the staff's Safety Evaluation Report (SER) for the ongoing Vogtle early site permit application.

Clarify the applicability of 10 CFR Part 21, "Reporting of Defects and Noncompliance," requirements for early site permit applications.

10 CFR Part 52 makes it clear that 10 CFR Part 21 is applicable to early site permit applicants.

Clarify the applicability of 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants," requirements for early site permit applications.

Again, 10 CFR Part 52 makes it clear that the Appendix B quality assurance requirements are applicable to early site permit applicants.

Develop improved guidance on electronic submission of applications.

The staff has improved and clarified the process for electronic submission of applications. This has included documentation and even video clips of the process. However, additional progress can still be made in this area.

Incorporate into staff guidance definitions of terms such as "License Conditions" and "COL action items."

The staff has incorporated these definitions into the Standard Review Plan and has trained reviewers regarding the definitions.

Develop guidance for the review of the performance-based methodology for assessing seismic hazards.

The staff has issued Regulatory Guide 1.208, "A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion."

Review the development and study of long-term weather cycles for periods of up to 100 years.

The staff has made appropriate modifications to the Standard Review Plan to recognize that there are cycles in the weather. Such cycles are especially well known for the east coast of the United States. The staff has made contact with knowledgeable technical societies, will be attending pertinent scientific conferences, and is proposing research studies of trends in the frequencies and intensities of hurricanes.

Update guidance for the review of site hydrology.

The staff has updated the Standard Review Plan. It is updating its regulatory guide on analysis of flooding. The staff is also investigating possible threats to coastal nuclear power plants posed by tsunamis including tsunamis that might come from submarine landslides in the Cape Verde islands.

Develop guidance for the treatment of the high frequency component of seismic ground motion.

The staff has provided guidance in both the Standard Review Plan and in Regulatory Guide 1.208.

Develop guidance on the use of Internet data.

The staff has not taken action on our recommendation that they develop guidance to ensure that data obtained from the Internet are valid now and retrievable in the future. At many points in the early site permit applications data derived from the Internet are used. We expect increased reliance on Internet databases in the future. Data obtained from the Internet do not have the immutable quality of the printed page. Such data can be altered by intent, through misadventure or through malice. Therefore, the NRC needs to provide applicants with guidance to ensure that data they obtain from the Internet are valid in the sense that they reflect the intent of the developer of the database. The data may be needed long after an early site permit has been approved and after many revisions of the electronic site from which the data were originally obtained. Consequently, guidance on ensuring the retrievability of the data is also needed. Furthermore, based on our recent review of the Vogtle early site permit application, it may be necessary for the NRC to interact with other government agencies to assist applicants in obtaining the validation that the staff feels is necessary for the data provided by these agencies via the Internet.

Sincerely,

/RA/

William J. Shack
Chairman

References:

1. Memorandum dated November 8, 2006, from Annette L. Vietti-Cook, Secretary of the Commission, NRC, to John T. Larkins, Executive Director, ACRS ; Subject: Staff Requirements — Meeting with Advisory Committee on Reactor Safeguards, 2:30 P.M., Friday, October 20, 2006, Commissioners' Conference Room, One White Flint North, Rockville, Maryland (Open to Public Attendance).

2. Letter dated September 22, 2006, from G. B. Wallis, Chairman, ACRS, to L. A. Reyes, Executive Director for Operations, NRC, Subject: "Lessons Learned From the Review of Early Site Permit Applications."

3. Draft United States Geological Survey Report, revision dated September 30, 2007, "The Current State of Knowledge Regarding Potential Tsunami Sources Affecting U.S. Atlantic and Gulf Coasts."

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

January 25, 2008

MEMORANDUM TO: Said Abdel-Khalik, Issue Chair

FROM: David Bessette, Senior Staff Engineer

SUBJECT: ANALYSIS OF EDO RESPONSE TO ACRS LETTER: DRAFT FINAL GENERIC LETTER 2007-02, "MANAGING GAS ACCUMULATION IN EMERGENCY CORE COOLING, DECAY HEAT REMOVAL, AND CONTAINMENT SPRAY SYSTEMS"

Attached is a copy of the EDO's December 6, 2007 response to the ACRS letter of October 19, 2007, regarding the subject generic letter on gas intrusion. A copy of the Committee's letter is also attached.

Committee Letter

In its letter, the Committee concluded:

1 Draft Generic Letter 2007-XX should be issued.

2. ACRS concurs with the Requested Actions and Information specified in the Draft Generic Letter.

The Committee stated that the frequent occurrence of gas intrusion events and lack of detailed documentation of surveillance results point to weaknesses in technical specifications in as least some plants, and that these weaknesses need to be addressed.

The Committee also indicated that it would like the opportunity to review any proposed interim measures or topical reports developed as a result of this Generic Letter.

Finally, the Committee agreed that it is important to share the information to be developed as a result of this Generic Letter with the Office of New Reactors and the industry's New Reactors Working Group.

EDO Response

The Staff issued the final Generic Letter (2008-01) on January 11, 2008 (ML072910759).

The EDO indicated that NRC staff have met with the industry informing them that changes to Technical Specifications will be pursued utilizing the information being developed as a result of Generic Letter 2008-01.

The EDO stated that the Staff will provide the ACRS the opportunity to review proposed interim measures or topical reports developed as a result of Generic Letter 2008-01.

Finally, the NRC staff will also continue to share information developed as a result of this generic letter with the Office of New Reactors and the industry's New Reactors Working Group.

Analysis

The EDO response is satisfactory. There are no points of disagreement.

Attachments: As stated

cc: ACRS Members
C. Santos
S. Duraiswamy

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

December 6, 2007 RECEIVED

DEC - 7 2007

Dr. William J. Shack, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

SUBJECT: DRAFT FINAL GENERIC LETTER 2007-02, "MANAGING GAS ACCUMULATION IN EMERGENCY CORE COOLING, DECAY HEAT REMOVAL, AND CONTAINMENT SPRAY SYSTEMS"

Dear Dr. Shack:

I am responding to your October 19, 2007, letter regarding the draft final generic letter titled, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems." The Advisory Committee on Reactor Safeguards (ACRS or the Committee) recommended that the proposed generic letter be issued.

Regarding the Committee's comment that there are technical specification (TSs) weaknesses that need to be addressed, the Nuclear Regulatory Commission (NRC) staff had previously met with the industry informing them that changes to TSs will be pursued utilizing the information being developed as a result of this generic letter.

The NRC staff will provide the ACRS the opportunity to review proposed interim measures or topical reports developed as a result of this generic letter. The NRC staff will also continue to share information developed as a result of this generic letter with the Office of New Reactors and the industry's New Reactors Working Group.

Sincerely,

Luis A. Reyes
Executive Director
for Operations

cc: Chairman Klein
Commissioner Jaczko
Commissioner Lyons
SECY

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

ACRSR-2271

October 19, 2007

Mr. Luis A. Reyes
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: DRAFT FINAL GENERIC LETTER 2007-XX, "MANAGING GAS INTRUSION IN EMERGENCY CORE COOLING, DECAY HEAT REMOVAL, AND CONTAINMENT SPRAY SYSTEMS"

Dear Mr. Reyes:

During the 546th meeting of the Advisory Committee on Reactor Safeguards, October 4-5, 2007, we reviewed the draft final Generic Letter 2007-XX, "Managing Gas Intrusion in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems." During our review, we had the benefit of discussions with representatives of the NRC staff and the Nuclear Energy Institute. We also had the benefit of the documents referenced.

RECOMMENDATION

Generic Letter 2007-XX, "Managing Gas Intrusion in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems," should be issued as final.

BACKGROUND

Gas intrusion into the emergency core cooling, decay heat removal, and containment spray systems ("subject systems") can lead to loss of operability or degradation of performance. It may also lead to piping damage due to water hammer effects. Over the past 20 years, the NRC staff has published 20 Information Notices, two Generic Letters, and a NUREG, and also interacted with the nuclear industry many times regarding the gas intrusion issue. An event in 1997 at Oconee Unit 3 damaged two of the plant's three high-pressure injection pumps and rendered them nonfunctional. Following that event, an industry-wide initiative was undertaken to address the gas intrusion issue. Based on the industry's actions, the NRC staff concluded that no generic action was necessary at that time. However, despite the design and operational measures taken to prevent gas intrusion and accumulation in the subject systems, and the high level of awareness of their potential impact on system performance, significant gas intrusion events have continued to occur, prompting the issuance of this Generic Letter.

DISCUSSION

Emergency core cooling, decay heat removal, and containment spray systems must be sufficiently full of water in order to successfully fulfill their intended functions when called upon during an accident. The number of gas intrusion problems that have been identified at some facilities raises concerns as to whether similar problems exist at other facilities.

Technical Specifications (TS) require periodic surveillance of the subject systems to confirm operability. The frequent occurrence of gas intrusion events and lack of detailed documentation of surveillance results point to TS weaknesses. We believe these weaknesses need to be addressed.

The amount of gas that can be ingested without significant impact on pump operability and reliability is not well established. NUREG/CR-2792 provides some guidance (based on expert opinions) on the amount of gas ingestion that can be tolerated without significant degradation of pump performance. The industry plans to perform work to develop additional criteria to assess operability. Studies will also be performed to evaluate gas detection techniques and the associated accuracies. We would like the opportunity to review any proposed interim measures or topical reports developed as a result of this Generic Letter.

The staff's resolution of the public comments provided during the process of preparing this Generic Letter is appropriate. We agree with the staff and the industry that it is important to share the information to be developed as a result of this Generic Letter with the Office of New Reactors and the industry's New Reactors Working Group.

Sincerely,

/RA/

William J. Shack
Chairman

REFERENCES:

1. Memorandum dated October 1, 2007, from James T. Wiggins, Deputy Director, Office of Nuclear Reactor Regulation, to Frank P. Gillespie, Executive Director, Advisory Committee on Reactor Safeguards, transmitting:

   - Proposed Generic Letter 2007-XX, "Managing Gas Intrusion in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems" (ML053460427).

   - Staff Resolution of Public Comments Received on the Proposed Generic Letter (ML072410212).

   - Redline/Strikeout Version of Proposed GL Showing Changes Due to Public Comments (ML072410253).

2. U.S. Nuclear Regulatory Commission/Creare Inc., P.S. Kamath, T.J. Tantillo, W.L Swift, NUREG/CR-2792, "An Assessment of Residual Heat Removal and Containment Spray Pump Performance Under Air and Debris Ingesting Conditions," September 1982.

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

February 6, 2007

MEMORANDUM TO: George E. Apostolakis, Chair
Reliability and PRA Subcommittee

FROM: Girija S. Shukla, Senior Program Manager /RA/
Reactor Safety Branch, ACRS

SUBJECT: ANALYSIS OF EDO RESPONSE TO ACRS LETTER ON DRAFT FINAL NUREG-1829, "ESTIMATING LOSS-OF-COOLANT ACCIDENT (LOCA) FREQUENCIES THROUGH THE ELICITATION PROCESS," AND DRAFT NUREG-XXXX, "SEISMIC CONSIDERATIONS FOR THE TRANSITION BREAK SIZE"

Attached is a copy of the January 30, 2008 EDO letter of response to the December 20, 2007 ACRS letter on the subject draft NUREG reports related to loss-of-coolant accident (LOCA) frequencies. A copy of the Committee's letter is also attached.

Committee Letter

In its December 20, 2007 letter the ACRS recommended that:

- NUREG-1829 on estimating LOCA frequencies through the expert elicitation process, and the NUREG report on seismic considerations for the transition break size (TBS) should be published.
- Regulatory decisions should be based on the totality of the results from the sensitivity studies rather than the results from individual methods of expert judgment aggregation.
- A set of consistent guidelines should be established for the elicitation and aggregation of expert judgments including the performance of sensitivity studies. These guidelines should be used throughout the agency.

EDO Response

The EDO response stated that the staff agrees with the Committee's recommendations, as follows, and that both reports are expected to be publicly available in February 2008.

The staff selected the proposed TBS in the draft rule by considering typical reactor coolant pressure boundary piping sizes to ensure an acceptably low break frequency after accounting for uncertainties in the NUREG-1829 LOCA frequency estimates. Risk contributions associated with factors not considered in the NUREG-1829 study were also addressed to ensure that the failure propensity beyond the TBS remains low.

The staff also agrees that it may be beneficial to establish guidance for conducting elicitations and aggregating expert judgments. Any additional effort will build on relevant existing guidance. RES will coordinate with other program offices to determine the need for further guidance.

In addition, on August 10, 2007, Commission provided the staff with additional guidance for developing the risk-informed revision to the ECCS rule. In addition to addressing the Commission Guidance as part of this revision, the staff will also address many of the recommendations from the Committee's letter dated November 20, 2006. The staff will also brief the Committee on this revised rule before releasing it for public comment. A revised schedule for this rulemaking is currently scheduled to be sent to the Commission in March 2008.

Analysis

The EDO's response is satisfactory.

Attachments: As stated

cc: ACRS Members F. Gillespie S. Duraiswamy C. Santos

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

January 30, 2008

Dr. William J. Shack, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

SUBJECT: DRAFT FINAL NUREG-1829, "ESTIMATING LOSS-OF-COOLANT ACCIDENT FREQUENCIES THROUGH THE ELICITATION PROCESS," AND DRAFT NUREG-XXXX, "SEISMIC CONSIDERATIONS FOR THE TRANSITION BREAK SIZE"

Dear Dr. Shack:

I am responding to your letter of December 20, 2007, concerning your review of the subject draft NUREG-series reports (Agency-wide Documents Access and Management System (ADAMS) Accession No. ML073440143). I appreciate the time and effort the Advisory Committee on Reactor Safeguards (the Committee) has devoted to reviewing these reports.

The NUREG-1829 report describes efforts by the staff of the U.S. Nuclear Regulatory Commission to develop loss-of-coolant accident (LOCA) frequencies using an expert elicitation process. The NUREG-XXXX report addresses the potential seismic effects on the failure propensity of flawed and unflawed piping, as well as indirect failures of other components and component supports that could lead to piping failure. The staff developed these reports to support a voluntary risk-informed revision of the regulatory requirements for the emergency core cooling system (ECCS), as set forth in Title 10, Section 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," of the *Code of Federal Regulations*. In particular, the subject NUREG reports support the development of a transition break size (TBS) which is smaller than the existing double-ended guillotine break that is considered in the design of the ECCS.

In your letter, the Committee provided the following three recommendations:

1. NUREG-1829 on estimating LOCA frequencies through the expert elicitation process and the NUREG report on seismic considerations for the TBS should be published.
2. Regulatory decisions should be based on the totality of the results from the sensitivity studies rather than the results from individual methods of expert judgment aggregation.
3. A set of consistent guidelines should be established for the elicitation and aggregation of expert judgments including the performance of sensitivity studies. These guidelines should be used throughout the agency.

With respect to the first recommendation, the staff is actively finalizing both NUREG reports for publication. Both reports are expected to be publicly available in February 2008.

The staff also generally agrees with the second recommendation. In particular, the staff selected the proposed TBS in the draft rule by considering typical reactor coolant pressure boundary piping sizes to ensure an acceptably low break frequency after accounting for uncertainties in the NUREG-1829 LOCA frequency estimates. Risk contributions associated with factors not considered in the NUREG-1829 study (e.g., seismic loading, heavy load drop, rare water hammer loading) were also addressed to ensure that the failure propensity beyond the TBS remains low. In particular, NUREG-XXXX addresses the failure of piping greater than the TBS under seismic loading.

The staff also agrees that it may be beneficial to establish guidance for conducting elicitations and aggregating expert judgments. Any additional effort will build on relevant existing guidance such as NUREG-1563, "Branch Technical Position on the Use of Expert Elicitation in the High-Level Radioactive Waste Program," and NUREG/CR-6372, "Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and Use of Experts." RES will coordinate with other program offices to determine the need for further guidance. Any plans for completion would be contingent on the availability of resources identified through the Planning, Budgeting, and Performance Management Process.

In addition, please note that on August 10, 2007, Commission provided additional guidance for developing the risk-informed revision to the ECCS rule in the staff requirements memorandum (SRM) for SECY-07-0082, "Rulemaking to Make Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements," (ADAMS Accession No. ML072220595). In addition to addressing the SRM as part of this revision, the staff will also address many of the recommendations from the Committee's letter dated November 20, 2006 (ADAMS Accession No. ML063190465). The staff will brief the Committee on this revised rule before releasing it for public comment. A revised schedule for this rulemaking is currently scheduled to be sent to the Commission in March 2008.

Sincerely,

WFKane for

Luis A. Reyes
Executive Director
for Operations

cc: Chairman Klein
Commissioner Jaczko
Commissioner Lyons
SECY

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

December 20, 2007

The Honorable Dale E. Klein
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: DRAFT FINAL NUREG-1829, "ESTIMATING LOSS-OF-COOLANT ACCIDENT (LOCA) FREQUENCIES THROUGH THE ELICITATION PROCESS," AND DRAFT NUREG-XXXX, "SEISMIC CONSIDERATIONS FOR THE TRANSITION BREAK SIZE"

Dear Chairman Klein:

During the 548th meeting of the Advisory Committee on Reactor Safeguards, December 6-8, 2007, we reviewed the draft final NUREG-1829, "Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process," and draft NUREG-XXXX, "Seismic Considerations for the Transition Break Size." Our Reliability and Probabilistic Risk Assessment Subcommittee reviewed this matter during a meeting on November 27, 2007. During these reviews, we had the benefit of discussions with representatives of the NRC staff. We also had the benefit of the documents referenced.

RECOMMENDATIONS

1. NUREG-1829 on estimating LOCA frequencies through the expert elicitation process, and the NUREG report on seismic considerations for the transition break size (TBS) should be published.

2. Regulatory decisions should be based on the totality of the results from the sensitivity studies rather than the results from individual methods of expert judgment aggregation.

3. A set of consistent guidelines should be established for the elicitation and aggregation of expert judgments including the performance of sensitivity studies. These guidelines should be used throughout the agency.

DISCUSSION

The Transition Break Size

An essential element of the proposed risk-informed alternative to the existing 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear reactors," is the concept of "transition break size." In a Staff Requirements Memorandum dated July 1, 2004, the Commission directed the staff to define the TBS as that break size that has a frequency of occurrence of about 10^{-5} per reactor year. Loss-of-coolant accidents due to breaks smaller than the TBS are expected to have frequencies of occurrence greater than 10^{-5} per reactor year and would remain design-basis accidents (DBAs). They would be analyzed using the methods, assumptions, and criteria currently prescribed in 10 CFR 50.46. Accidents due to breaks larger than the TBS are expected to have lower frequencies of occurrence and would become beyond design-basis accidents. Consequently, they would be analyzed without the additional conservatisms associated with DBAs.

The size of the transition break cannot be determined from operating experience or mechanistic calculations alone. We must rely on expert judgment supported by the available evidence and analyses. The resulting uncertainty is managed by selecting a conservative TBS and by ensuring that breaks greater than the TBS can be mitigated, i.e., by invoking a structuralist defense-in-depth principle for this range of break sizes.

The staff has produced two reports, NUREG-1829 and NUREG-XXXX, which help to provide the basis for selecting a conservative TBS. NUREG-1829 presents the results of a formal expert evaluation of the state of the art and NUREG-XXXX focuses on the impact of seismic events on TBS.

The authors of NUREG-1829 acknowledge the limitations of expert opinion elicitation processes as well as the fact that one could use several ways to aggregate these opinions. The study provides the results of a series of sensitivity studies that help decisionmakers understand the magnitude of the uncertainties in the TBS. As expected, many public comments addressed issues associated with individual aggregation methods. Although the authors of NUREG-1829 have provided reasonable answers to these comments, it is the totality of results from the sensitivity studies that shapes our state of knowledge rather than the results from individual methods.

NUREG-XXXX provides additional insights by investigating seismically induced failures in unflawed piping, flawed piping, and indirect piping failures caused by the failure of other components and supports. The results of the study indicate that, for Pressurized-Water Reactors (PWRs) east of the Rocky Mountains, the likelihood of seismically induced failures in unflawed piping of size greater than the TBS is very low for earthquakes with 10^{-5} and 10^{-6} annual probabilities of exceedance. Even for pipes with long surface flaws, the depths of these flaws must be greater than 30-40% of the wall thickness for a high likelihood of failure during such earthquakes. Inspection programs, leak detection systems, and other measures taken to eliminate failure mechanisms such as stress corrosion cracking should make the likelihood of such cracks very low.

Both of these NUREG reports provide results and insights that can form the basis for the selection of the TBS. They should be published.

Expert Judgment

Using expert judgments to evaluate the state-of-the-art in issues that cannot be resolved by statistical or mechanistic methods is an approach that has been pioneered by the NRC. These issues usually involve rare events and divergence of opinions among knowledgeable investigators and practitioners.

The Senior Seismic Hazard Analysis Committee (SSHAC) investigated the paralyzing differences in probabilistic seismic hazards between the NRC and the Electric Power Research Institute (EPRI) (NUREG/CR-6372). SSHAC stated: "The Committee's most important conclusion is that differences in PSHA [Probabilistic Seismic Hazard Analysis] results are due to procedural rather than technical differences. Thus, in addition to providing a detailed documentation on state-of-the-art elements of a PSHA, this report provides a series of procedural recommendations." These recommendations dealt with the use of expert judgments. It is worth pointing out that the SSHAC work was sponsored by the NRC, DOE, and EPRI. It was reviewed by a National Research Council Panel, which stated: "The panel believes that the SSHAC report makes a solid contribution to the methodology of hazard analysis, especially in the use of expert opinion."

The goal of the SSHAC guidance is to develop a probability distribution representing the state of knowledge of the informed technical community. To achieve this, the SSHAC guidance recommends that the appropriate method for aggregating expert estimates is one that encourages complete sharing of information and full consideration and discussion of the evidence supporting each expert's judgment. The approach asks the experts to state their own opinions first and then defend their positions, based on all the evidence at their disposal. This sharing of evidence puts the experts on equal footing and ensures that they understand the bases for the judgments of others. The approach then asks each expert to take on a new role, that of evaluator.

Under this reframing of the problem, the experts, acting as evaluators, propose probability distributions reflecting the state of knowledge of the informed technical community. This is done after significant interaction has taken place among them. Ideally, the experts agree upon a consensus distribution. The SSHAC report recommends that the results of any mechanistic aggregation of opinions be scrutinized and modified if they are inconsistent with the overall judgment of the experts and the study integrators. The National Research Council Panel agrees and states: "Do not accept the results of a mechanical combination rule unless they are consistent with judgment."

We note that this elicitation process gives considerable attention to the extreme values of the distribution, challenging each evaluator to consider all factors that could drive the results higher or lower. We acknowledge that this approach requires very effective control of bias and the interaction among experts, but that is true of all elicitation efforts.

For their baseline methodology, the authors of NUREG-1829 take the geometric average of each set (lower, median, and upper bound) of the expert supplied percentiles. This averaging is performed after the experts have exchanged views and their opinions have been adjusted for possible bias by the study integrators. The authors subscribe to the view that a group estimate should be defined as a value near the center of the group opinion; i.e., their approach focuses on getting the center value of the estimate right. In this study, the geometric mean does produce a value near the center of the group estimates[1].

The method called "Mixture Distribution Aggregation" in NUREG-1829 is the mechanistic aggregation approach recommended by SSHAC and was used by the team that developed NUREG-1150. In this method, the composite probability distribution of the frequency of a break of a certain size is the arithmetic average of the panelists' probability distributions (not of the percentiles).

[1] It is important to recognize that the geometric average of percentiles can be controlled by a very low outlier. Similarly, the arithmetic average of percentiles can be controlled by a high outlier. In the current study, there are no extreme low outliers for the final evaluations; therefore, the geometric mean gives a fair estimate of the *center* of the distributions.

In response to comments provided during the ACRS Subcommittee meeting, the authors of NUREG-1829 also produced results using the Mixture Distribution Aggregation method. The panelists went through a significant exchange of views. They were not asked, however, to act as evaluators, i.e., to produce distributions that reflect the views of the informed technical community; their distributions represented their own uncertainties. The authors of NUREG-1829 state: "The mixture distribution approach does not attempt to develop aggregated estimates that represent the central group opinion as does the baseline methodology, but rather attempts to exhibit the full range of variability among the panelist responses." We believe that employing a method that "exhibits the full range of variability among the panelist responses" is important and useful for a study whose results will form the basis of regulations. In these cases, understanding the breadth of informed opinion is more important than central estimates.

There is no compelling mathematical reason supporting a particular aggregation method[2]. Each requires assumptions that may or may not be justified. We find the attempt to develop a consensus distribution that represents the technical community's views intellectually appealing. To help the experts develop consensus, sensitivity studies need to be conducted including possible adjustment for bias and various aggregation schemes.

The elicitation of expert judgments is a process that the NRC will continue to use to inform regulatory decisionmaking involving important matters. The method employed to process these judgments cannot be left up to the discretion of the team performing each new study. The Office of Nuclear Regulatory Research should investigate the existing methods and propose a set of consistent guidelines to be used throughout the agency.

Sincerely,

William J. Shack
Chairman

REFERENCES

1. U.S. Nuclear Regulatory Commission, NUREG-1829, "Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process," and associated Appendixes A through M, 2005.

2. U.S. Nuclear Regulatory Commission, NUREG-XXXX, "Seismic Considerations for the Transition Break Size," 2005.

3. U.S. Nuclear Regulatory Commission, NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," 1990.

4. U.S. Nuclear Regulatory Commission, NUREG/CR-6372, "Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and Use of Experts," [Prepared by Senior Seismic Hazard Analysis Committee (SSHAC)], 1997.

[2] The theoretically correct method for combining expert judgments is to treat them as evidence in a Bayesian framework. To date, this approach is impractical. Development of a consensus distribution reflecting the breadth of concerns of the technical community is an excellent way to select an informed prior distribution for later Bayesian analysis.

5. Staff Requirements Memorandum from Annette L. Vietti-Cook, Secretary, U.S. Nuclear Regulatory Commission, to Luis A. Reyes, Executive Director for Operations, U.S. Nuclear Regulatory Commission, "Staff Requirements -SECY-04-0037 - Issues Related to Proposed Rulemaking to Risk-Inform Requirements Related to Large Break Loss-of-Coolant Accident (LOCA) Break Size and Plans for Rulemaking on LOCA with Coincident Loss-of-Offsite Power," dated July 1, 2004.

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

February 7, 2008

MEMORANDUM TO: Michael L. Corradini, Chair
ESBWR Subcommittee

FROM: Charles G. Hammer, Senior Staff Engineer *Charles G. Hammer*

SUBJECT: ANALYSIS OF EDO RESPONSE TO ACRS LETTER ON CHAPTERS 2, 5, 8, 11, 12, AND 17 OF THE NRC STAFF'S SAFETY EVALUATION REPORT WITH OPEN ITEMS RELATED TO THE CERTIFICATION OF THE ESBWR DESIGN

Attached is a copy of the EDO's February 1, 2008 letter of response to the ACRS' November 20, 2007 letter on Chapters 2, 5, 8, 11, 12, and 17 of the NRC staff's safety evaluation report with open items related to the certification of the ESBWR design. A copy of the Committee's letter is also attached.

Committee Letter

In its November 20, 2007 letter the ACRS provided three detailed comments on Chapters 5 and 12 as follows:

1. The staff should further investigate the adequacy of controls on post-weld grinding. GE-Hitachi Nuclear Energy Americas, LLC, (GEH) has placed controls on the use of grinding wheels and wire brushes in the fabrication of the ESBWR components and structures to prevent potentially degrading materials from entering the system. However, post-weld grinding can degrade the resistance of austenitic stainless steels and nickel-based alloys to various stress-corrosion cracking (SCC) mechanisms when exposed to the reactor coolant. The controls on welding practice should be revised to eliminate such practices to the extent possible and to mitigate their consequences in those instances in which grinding is unavoidable.

2. Although the materials chosen for the pressure boundary are resistant to SCC under normal boiling-water reactor water chemistry, experience indicates that core internals will be susceptible to irradiation-assisted stress-corrosion cracking (IASCC) unless more controls are placed on water chemistry. ACRS would like the opportunity to review ESBWR reactor coolant system chemistry controls in future meetings.

3. Although the basis for the estimated source term for radioactive materials released from fuel into the RCS seems reasonable, the Committee would like to review the data and the analysis procedure used to develop the source team.

EDO Response

The EDO response is summarized as below for each of the three detailed ACRS comments:

1. The staff recognizes that excessive cold working of austenitic stainless steels and nickel-based alloys makes them more susceptible to SCC even when using materials (i.e., low-carbon stainless steel and niobium-modified Alloy 600) that are considered to be resistant to SCC. However, the staff states that post-weld grinding of austenitic stainless steel and nickel-based alloy welds during the fabrication of reactor coolant pressure boundary components is unavoidable in many instances, such as, during the removal of temporary attachments, surface contouring of welds to facilitate nondestructive examinations, and removal of welding defects. The staff notes that welding defects discovered during the fabrication process by the various examination methods that are in excess of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) acceptance criteria must be repaired. The staff makes use of review guidance in the standard review plan and design and inspection criteria in the ASME Code to provide an adequate basis to ensure the long-term integrity of structures, systems, and components (SSCs) important to safety. Revision 4 of the ESBWR design control document (DCD) partly addresses this issue for austenitic stainless steels used for reactor vessel internals and the reactor coolant pressure boundary. That is, during the fabrication, cold working will be controlled by applying limits in hardness, bend radii, and surface finish on ground surfaces. Revision 4 of the ESBWR DCD is silent, however, on the control of cold working of nickel-based alloys. The staff has been discussing additional controls on grinding with GEH, which the staff will consider if such controls are proposed by either GEH or a combined operating license (COL) applicant.

2. The staff recognizes the potential benefits of controls on water chemistry. The staff notes that the applicable Standard Review Plan and design and inspection criteria in the ASME Code have evolved over time, but specific requirements to address IASCC through water chemistry controls have not been developed as part of the current regulatory requirements. The staff has discussed such controls with GEH and will consider them if they are proposed by either GEH or a COL applicant. Although there are no regulatory or ASME Code requirements for a design certification applicant, like GEH, to require the use of a hydrogen water chemistry system, the staff still considers the reactor internals less susceptible to IASCC for the following reasons:

   - Only low-carbon stainless steel and nickel alloys modified for high SCC resistance will be specified for reactor internals.
   - Strict controls on the fabrication and installation processes for the reactor internals will be used.
   - Application of surface finishing techniques will be used to remove surface cold work in the weld heat-affected zones of the major structural welds in the large internals.

3. The staff has sent a request for additional information to GEH to obtain the necessary information for developing the source term of radioactive materials released into the RCS and will provide this information to ACRS once received.

Analysis

Regarding the ACRS comment no.1 above, the staff recognizes the Committee's concerns regarding eliminating, to the extent possible, post-weld grinding to reduce IGSCC of austentic stainless steels and nickel-based alloys. The staff notes that, in Revision 4 of the DCD, GEH has partly addressed the issue of cold working for austenitic stainless steels by applying limits on hardness, bend radii, and surface finish, but that GEH has not placed similar controls for nickel-based alloys. The staff is engaging GEH regarding additional controls on grinding. It is not clear at this point in time whether GEH will eventually have in place the practice of eliminating post-weld grinding, to the extent possible, that the Committee has recommended. However, since the Committee will have an opportunity to revisit this issue when the final SER is reviewed and given that the staff is currently engaging GEH regarding this issue, the EDO's response to this ACRS comment appears to be satisfactory at this time.

Regarding the ACRS comment no. 2 above, the staff recognizes the Committee's concerns regarding the need for more controls on water chemistry to reduce IASCC of core internals. The staff notes that there are no regulatory or ASME Code requirements to place greater controls on water chemistry, but notes that the specified reactor internals materials are less susceptible to IASCC. The staff has discussed the need for greater controls on water chemistry with GEH, but it is not clear if GEH will eventually have in place the controls that the Committee has recommended. However, since the Committee will have an opportunity to revisit this issue when the final SER is reviewed and given that the staff is currently engaging GEH regarding this issue, the EDO's response to this ACRS comment appears to be satisfactory at this time.

Regarding the ACRS recommendation no. 3 above, the staff has requested GEH to provide the necessary data and analysis procedure used to develop the source term for radioactive materials released from fuel into the RCS. **The staff stated they will forward this to the ACRS once received.** The EDO's response to this ACRS recommendation is satisfactory.

Attachments: As stated

cc: ACRS Members F. Gillespie S. Duraiswamy C. Santos

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

February 1, 2008

Dr. William J. Shack, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

SUBJECT: INTERIM LETTER: CHAPTERS 2, 5, 8, 11, 12, AND 17 OF THE U.S. NUCLEAR REGULATORY COMMISSION STAFF'S SAFETY EVALUATION REPORT WITH OPEN ITEMS RELATED TO THE CERTIFICATION OF THE ESBWR DESIGN

Dear Dr. Shack:

This is in response to the Advisory Committee on Reactor Safeguards' (ACRS or the Committee) November 20, 2007, letter regarding the review of the General Electric-Hitachi Nuclear Energy Americas, LLC, (GEH) application for certification of the economic simplified boiling-water reactor (ESBWR) plant design. During the ACRS meeting on November 2, 2007, the staff discussed its safety evaluation reports (SERs) with open items (OIs) for Chapters 2, 5, 8, 11, 12, and 17 of the ESBWR design certification application with the full committee. These discussions included the status of OIs identified in the SERs as well as the technical concerns associated with them. The ACRS raised specific concerns on Chapter 5 associated with minimizing the potential for stress-corrosion cracking of austenitic stainless steels and nickel-based alloys and measures to minimize and mitigate post-welding processes that could contribute to this type of corrosion. In addition, the Committee raised concerns associated with the use of water chemistry controls as a measure to minimize irradiation-assisted stress-corrosion cracking. The enclosure to this letter discusses the staff's responses to these specific ACRS concerns. The staff continues to work with GEH to obtain satisfactory resolution to the OIs presented in the SERs and looks forward to presenting the resolutions to these OIs to the ACRS during future presentations on the final safety analysis report for the ESBWR design certification application.

The ACRS also stated that, although the basis for the estimated source term for radioactive materials released from fuel into the RCS seems reasonable, the Committee would like to review the data and the analysis procedure used to develop the source team. The staff has sent a request for additional information to GEH to obtain this material and will provide this information to ACRS once received.

Thank you for your comments. I appreciate the willingness of the ACRS to engage with the staff on a chapter-by-chapter review process for the SERs with OIs and believe this process has greatly facilitated the staff's review. My staff looks forward to continued interactions with the Committee on the SERs with OIs for the remaining chapters of the ESBWR design certification application.

Sincerely,

Luis A. Reyes
Executive Director
for Operations

Enclosure:
Staff Response to ACRS Comments

cc: Chairman Klein
Commissioner Jaczko
Commissioner Lyons
SECY

The U.S. Nuclear Regulatory Staff Response to the Advisory Committee on Reactor Safeguards Interim Letter Dated November 20, 2007, Regarding Safety Evaluation Reports with Open Items on the ESBWR Design Certification Application

The staff prepared responses to comments from the Advisory Committee on Reactor Safeguards (ACRS) on the staff's safety evaluation report (SER) with open items for Chapter 5, "Reactor Coolant System and Connected Systems," of the economic simplified boiling-water reactor (ESBWR) design certification application. The staff responded to these concerns during the ACRS ESBWR subcommittee meeting on January 16 and 17, 2008, and continues to work with the applicant to develop satisfactory resolution to these concerns and to revise the ESBWR design control document accordingly. The staff plans to discuss final resolution of these concerns during the ACRS full committee meeting on the final SER for the ESBWR design certification application.

ACRS Comment: The staff should further investigate the adequacy of controls on post-weld grinding. GE-Hitachi Nuclear Energy Americas, LLC, (GEH) has placed controls on the use of grinding wheels and wire brushes in the fabrication of the ESBWR components and structures to prevent potentially degrading materials from entering the system. However, post-weld grinding can degrade the resistance of austenitic stainless steels and nickel-based alloys to various stress-corrosion cracking (SCC) mechanisms when exposed to the reactor coolant. The controls on welding practice should be revised to eliminate such practices to the extent possible and to mitigate their consequences in those instances in which grinding is unavoidable.

Staff Response: The staff recognizes that excessive cold working of austenitic stainless steels and nickel-based alloys makes them more susceptible to SCC even when using materials (i.e., low-carbon stainless steel and niobium-modified Alloy 600) that are considered to be resistant to SCC. However, post-weld grinding of austenitic stainless steel and nickel-based alloy welds during the fabrication of reactor coolant pressure boundary components is unavoidable in many instances, such as, during the removal of temporary attachments, surface contouring of welds to facilitate nondestructive examinations, and removal of welding defects. The staff notes that welding defects discovered during the fabrication process by the various examination methods that are in excess of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) acceptance criteria must be repaired. The staff makes use of review guidance in the standard review plan and design and inspection criteria in the ASME Code to provide an adequate basis to ensure the long-term integrity of structures, systems, and components (SSCs) important to safety. Revision 4 of the ESBWR design control document (DCD) partly addresses this issue for austenitic stainless steels used for reactor vessel internals and the reactor coolant pressure boundary. That is, during the fabrication, cold working will be controlled by applying limits in hardness, bend radii, and surface finish on ground surfaces. Revision 4 of the ESBWR DCD is silent, however, on the control of cold working of nickel-based alloys. The staff has been discussing additional controls on grinding with GEH, which the staff will consider if they are proposed by either GEH or a combined operating license (COL) applicant.

Enclosure

ACRS Comment: Although the materials chosen for the pressure boundary are resistant to SCC under normal boiling-water reactor water chemistry, experience indicates that core internals will be susceptible to irradiation-assisted stress-corrosion cracking (IASCC) unless more controls are placed on water chemistry. ACRS would like the opportunity to review ESBWR reactor coolant system chemistry controls in future meetings.

Staff Response: The staff recognizes the potential benefits of controls on water chemistry. The staff makes use of review guidance in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," and design and inspection criteria in the ASME Code to provide an adequate basis to ensure the long-term integrity of SSCs important to safety. This review guidance and the design and inspection codes have evolved over time, recognizing the benefits of tighter controls on water chemistry. However, specific requirements to address IASCC through water chemistry controls have not been developed as part of the U.S. Nuclear Regulatory Commission's regulatory requirements. Such controls have been a subject of discussion with GEH and will be considered by the staff if they are proposed by either GEH or a COL applicant. Although there are no regulatory or ASME Code requirements for a design certification applicant, like GEH, to require the use of a hydrogen water chemistry system, the staff still considers the reactor internals less susceptible to IASCC for several reasons, which are summarized here and described in more detail in the staff's safety evaluation report for Chapter 4 of the ESBWR DCD. Only low-carbon stainless steel and nickel alloys modified for high SCC resistance will be specified for reactor internals. Strict controls of the fabrication and installation processes for the reactor internals will be used. Application of surface finishing techniques will be used to remove surface cold work in the weld heat-affected zones of the major structural welds in the large internals.

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

ACRSR-2274

November 20, 2007

Mr. Luis A. Reyes
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: INTERIM LETTER: CHAPTERS 2, 5, 8, 11, 12, AND 17 OF THE NRC STAFF'S SAFETY EVALUATION REPORT WITH OPEN ITEMS RELATED TO THE CERTIFICATION OF THE ESBWR DESIGN

Dear Mr. Reyes:

During the 547th meeting of the Advisory Committee on Reactor Safeguards, November 1-3, 2007, we met with representatives of the NRC staff and General Electric – Hitachi Nuclear Energy Americas, LLC, (GEH) to discuss six Chapters from the Safety Evaluation Report (SER) related to the Economic Simplified Boiling Water Reactor (ESBWR) design certification application. Our ESBWR Subcommittee held meetings on October 2-3 and October 25, 2007, to discuss the technical aspects of the ESBWR design as well as the staff's SER, remaining open items, and the combined license (COL) action items for each of these SER Chapters. We had the benefit of the documents referenced.

RECOMMENDATIONS

1. We plan to review the staff's resolution of open items in SER Chapters 2, 5, 8, 11, 12, and 17 during future meetings.
2. The controls on welding practice should be revised to eliminate, to the extent possible, post-weld grinding of materials susceptible to stress corrosion cracking and to mitigate its consequences in those instances when grinding is unavoidable.
3. Many of the ESBWR systems described in these Chapters may interact with systems discussed in other SER Chapters that have not been reviewed. We will consider and comment on safety implications of any system interactions in future interim letters and in our final report

BACKGROUND

The ESBWR utilizes a direct-cycle power conversion system with natural circulation in the reactor vessel under normal operation and passive emergency core cooling system (ECCS) operation without the need of emergency alternating current power systems for core cooling within the first 72 hours following a reactor transient or accident. It also uses passive containment cooling to ensure heat transport to the ultimate heat sink for all accident scenarios. To cope with a severe reactor accident, the ESBWR design incorporates a lower drywell core retention device and allows passive drywell flooding to provide long-term debris cooling.

GEH submitted the ESBWR design certification application on August 24, 2005. Subsequently, based on staff requests, GEH submitted additional material and the staff formally accepted the complete application in December 2005. The staff issued Requests for Additional Information (RAIs) and based on the original application and GEH responses to the RAIs, the staff is preparing an SER with open items as well as COL action items. At the request of the staff, we agreed to review the staff's SER on a chapter-by-chapter basis to help timely completion of the review of the ESBWR design certification application, as well as effective resolution of our concerns prior to issuing the final SER. Accordingly, the staff has provided SER Chapters 2, 5, 8, 11, 12, and 17 with open items and COL action items for our review.

DISCUSSION

Based on the information presented to us to date, we have the comments provided below.

Chapter 2: Site Characteristics
Site characteristics include potential hazards in proximity of the plant, meteorology, hydrology, geology, seismology, and geotechnical parameters. An applicant for a COL that references the ESBWR design control document (DCD) will establish the site characteristics when it applies for a COL, or it will reference an early site permit (ESP) that reflects these characteristics. In either case, the COL applicant must show that the site parameters considered in the ESBWR DCD bound the actual site characteristics. Should the ESBWR design parameters not encompass the actual site characteristics, the COL applicant will need to demonstrate by other means, that the proposed reactor plant design is acceptable at the proposed site.

The staff identified several open items and COL action items in this Chapter. The open items seek to clarify inconsistencies in the documentation, to require additional information, and to verify that certain site meteorological assumptions are bounding. The Standard Review Plan specifies that the plant site parameters in the design certification be representative of a reasonable number of sites. The staff has found that this provision has been met.

Chapter 5: Reactor Coolant System and Connected Systems
The reactor coolant system (RCS) includes those systems and components that contain or transport fluids coming from or going into the reactor core. These systems form the major portion of the RCS pressure boundary. The SER Chapter 5 documents the staff's evaluation of the RCS pressure boundary and associated systems (e.g., pressure vessels, piping, pumps, and valves) out to and including the outboard isolation valves.

The staff identified several open items and COL action items in this Chapter. In the SER, the staff identified the need for additional information on materials specification (e.g., materials for specific classes of valves, specific steel alloy contents, filler-weld material), materials processing and qualification, and inservice inspection procedures for a range of systems and components.

The staff should further investigate the adequacy of controls on post-weld grinding. GEH has placed controls on the use of grinding wheels and wire brushes in the fabrication of the ESBWR components and structures to prevent potentially degrading materials entering the system. However, post-weld grinding can degrade the resistance of austenitic stainless steels and nickel-based alloys to various stress corrosion cracking mechanisms when exposed to the reactor coolant. The controls on welding practice should be revised to eliminate such practices to the extent possible and to mitigate their consequences in those instances when grinding is unavoidable.

Although the materials chosen for the pressure boundary are resistant to stress corrosion cracking under normal boiling water reactor water chemistry, experience indicates that core internals will be susceptible to irradiation assisted stress corrosion cracking unless more controls are placed on water chemistry. We would like the opportunity to review ESBWR RCS chemistry controls in future meetings.

One of the key subsystems in the RCS pressure boundary is the isolation condenser, which provides a redundant path to passively remove heat under a range of transient and accident conditions. This system performs an important safety function that will be evaluated in subsequent SER Chapters. The current open items relate to materials qualification and inservice inspection issues. Resolution of these open items could allow the staff to finalize its conclusions on the RCS. Comments and questions about system interactions may arise later with regard to specific safety issues and accident sequences.

Chapter 8: Electric Power

The on-site and off-site electric power systems include those systems that supply power to safety and non-safety related equipment. The ESBWR design does not require Class IE alternating current electrical power to accomplish the plant's safety related functions. The isolation condenser, a passive safety system for the RCS, and the passive containment cooling system require only Class IE direct current power to perform their functions during the initial 72 hours following all accident sequences.

The staff identified an open item in this Chapter, e. g., GEH should provide a loading profile for the safety related batteries to verify that they are properly sized to meet the design requirement for the initial 72 hour time period. The staff's review of the safety related electric power systems identified a need to consider system interactions. For example, confirmation is needed that the Class IE uninterruptible power supplies are not compromised by the lack of active room cooling during an extended accident sequence. This type of system interaction will need to be considered.

Chapter 11: Radioactive Waste Management

The radioactive waste management system for the ESBWR controls the handling and treatment of gaseous, liquid, and solid radioactive wastes. The release of radioactivity to the reactor coolant is part of the design basis for the radioactive waste system. This system is designed and operated to limit the dose to plant workers and members of the public to within regulatory limits and to ensure that doses are as low as reasonably achievable. The staff's review of the radioactive waste management system identified three open items that require better design definition of the skid-mounted 'mobile' radioactive waste systems as well as a number of COL action items and confirmatory items. We concur with these open items and action items.

GEH has used an assumed "source term" for radioactive materials released from the fuel into the RCS. The source term was estimated based on operational experience from the current fleet of boiling water reactors. The staff has accepted this source term as conservative for the ESBWR. Although this approach seems reasonable, we would like to review the data and the analysis procedure used to develop the source term.

Chapter 12: Radiation Protection

This Chapter describes the types and quantities of radioactive materials expected to be produced during the operation of the ESBWR, as well as the means for controlling or limiting radiation exposures within the requirements of 10 CFR Part 20. The measures are intended to ensure that radiation exposures to plant personnel, contractors, and the general public, resulting from plant operation and anticipated operational occurrences are within regulatory limits and are as low as reasonably achievable. The SER identified several open items in this Chapter that need to be addressed.

Chapter 17: Quality Assurance

The quality assurance program (QAP) for the ESBWR is based on the standard GEH QAP documented in GE topical report NEDO-11209-04A . The staff inspected the implementation of the GEH QAP for the ESBWR activities as part of the review of this Chapter. Based on the review, the staff identified an open item whereby the applicant will provide the list of risk-significant systems, structures, and components that are within the scope of the design reliability assurance program.

We plan to review the resolution of the open items identified on the above Chapters during future meetings.

Sincerely,

/RA/

William J. Shack
Chairman

References:

1. Memorandum from David B. Matthews, Director, Division of New Reactor Licensing (DNRL), Office of New Reactors (NRO), to Frank P. Gillespie, Executive Director, Advisory Committee on Reactor Safeguards and Advisory Committee on Nuclear Waste and Materials (ACRS/ACNW&M), dated August 31, 2007, transmitting SER with open items for Chapter 2, "Site Characteristics" (ML072270679 and ML072270468).

2. Memorandum from David B. Matthews, Director, DNRL, NRO, to Frank P. Gillespie, Executive Director, ACRS/ACNW&M, dated August 31, 2007, transmitting SER with open items for Chapter 5, "Reactor Coolant System and Connected Systems" (ML070780172 and ML072290103).

3. Memorandum from David B. Matthews, Director, DNRL, NRO, to Frank P. Gillespie, Executive Director, ACRS/ACNW&M, dated August 31, 2007, transmitting SER with open items for Chapter 8, "Electric Power" (ML072120282 and ML072120144).

4. Memorandum from David B. Matthews, Director, DNRL, NRO, to Frank P. Gillespie, Executive Director, ACRS/ACNW&M, dated September 24, 2007, transmitting SER with open items for Chapter 11, "Radioactive Waste Management" (ML072340212 and ML072340198).

5. Memorandum from David B. Matthews, Director, DNRL, NRO, to Frank P. Gillespie, Executive Director, ACRS/ACNW&M, dated September 24, 2007, transmitting SER with open items for Chapter 12, "Radiation Protection" (ML071730022 and ML072340020).

6. Memorandum from David B. Matthews, Director, DNRL, NRO, to Frank P. Gillespie, Executive Director, ACRS/ACNW&M, dated August 27, 2007, transmitting SER with open items for Chapter 17, "Quality Assurance" (ML072140668 and ML072140652).

7. Letter from James C. Kinsey, Project Manager, ESBWR Licensing, GEH, to NRC, dated February 22, 2007, transmitting ESBWR Design Control Document, Revision 3 (ML070660561).
8. General Electric Company, NEDO-11209-04A, Revision 8, "GE Nuclear Energy Quality Assurance Program Description," March 1989.
9. 10 CFR Part 20, "Standards for Protection Against Radiation."

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

February 8, 2008

MEMORANDUM TO: Dr. Said Abdel-Khalik

FROM: Z. Abdullahi, Senior Staff Engineer

SUBJECT: ANALYSIS OF EDO RESPONSE TO ACRS LETTER CONCERNING AREVA DETECT AND SUPPRESS STABILITY SOLUTION AND METHODOLOGY

Attached for your information is a copy of the EDO's January 30, 2007 response to the December 27, 2007, ACRS letter related to the proposed AREVA detect and suppress stability solution and methods. A copy of the Committee's letter is also attached.

Committee Letter

In its letter, the Committee concluded that the Enhanced Option III methodology, subject to limitations and conditions, is an acceptable methodology to detect and suppress oscillations in expanded flow window operating domains. The Committee also recommended that:

Recommendation 3

The errors in the neutron monitoring systems due to bypass voiding be documented and preferably be reviewed and approved on generic basis;

Recommendation 4

The five percent hot channel oscillation magnitude (HCOM) adjustment be justified further and that that the staff evaluate the additional supporting justifications and document the basis for its acceptability;

Recommendation 5

The validation of the RAMONA5-FA steady-state dryout correlations for application to unstable oscillatory conditions be documented and submitted for the staff's review and approval

Recommendation 6

The final safety analysis report document the evaluation of the adequacy of the 10 percent penalty applied to the DIVOM slopes calculations, using RAMONA5-FA.

EDO Response

The EDO response accepted all of the recommendations and conclusions. The EDO response describes the solution path forward in implementing the Committee's recommendations, including the staff's plan to obtain additional data; request and review the additional supporting technical justification and document the evaluations. The EDO response notes that in implementing Recommendation 6 the 10 percent penalty on the DIVOM slope would translate to a 0.03 penalty in the OLMCPR for a given OPRM scram. It also states that the staff plans to perform extensive follow-up review of the RAMONA5-FA code.

Analysis

The EDO's response is satisfactory.

cc: ACRS members
C. Santos
S. Duraiswamy
F. Gillespie

January 30, 2008

Dr. William J. Shack, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

SUBJECT: AREVA DETECT AND SUPPRESS STABILITY SOLUTION AND METHODOLOGY

Dear Dr. Shack:

On behalf of the U.S. Nuclear Regulatory Commission, I would like to thank you for your December 27, 2007, letter which provided the Advisory Committee on Reactor Safeguards' (ACRS or the Committee) views on the staff's draft safety evaluation of AREVA Topical Reports ANP-10262P, Rev. 0, "Enhanced Option III Long Term Stability Solution," and BAW-10255P, Rev. 2, "Cycle-Specific DIVOM Methodology Using the RAMONA5-FA Code." Your letter was in response to discussions with the staff and AREVA during the 548th meeting of the ACRS, December 6-8, 2007, and provided six recommendations. The staff's responses to the ACRS recommendations are provided below.

Recommendation 1:

The Enhanced Option III (EO-III) methodology, subject to the limitations and conditions imposed in the staff's draft safety evaluation and recommendations 3 and 4 below, is an acceptable methodology to detect and suppress oscillations in expanded flow-window operating domains.

Staff Response:

We appreciate your support for the staff's recommendation to accept EO-III methodology to detect and suppress oscillations in expanded flow-window operating domains subject to the limitations and conditions imposed in the staff's draft safety evaluation. Our responses regarding recommendations 3 and 4 are discussed below.

Recommendation 2:

The methods and procedures documented in BAW-10255P, Rev. 2, subject to the limitations and conditions imposed in the staff's draft safety evaluation and recommendations 3, 5, and 6 below, represent an acceptable methodology to calculate delta critical power ratio (CPR) over initial CPR versus oscillation magnitude (DIVOM) slope values.

Staff Response:

We appreciate your support for the staff's recommendation to accept the methods and procedures documented in BAW-10255P, Rev. 2, to calculate values of DIVOM slope subject to the limitations and conditions imposed in the staff's draft safety evaluation. Our responses regarding recommendations 3, 5, and 6 are discussed below.

Recommendation 3:

The applicant's methodology for evaluating the impact of average power range monitor (APRM) and oscillation power range monitor (OPRM) errors caused by bypass voiding should be documented. It would be preferable if such methodology were reviewed and approved on a generic rather than a plant-specific basis.

Staff Response:

We accept your recommendation to evaluate, on a generic basis to facilitate follow-on reviews, the impact of bypass voiding on calibration errors associated with the LPRMs that can affect the APRMs and OPRMs. The staff is in the process of reviewing with the fuel vendor (AREVA) a methodology to propagate errors induced by the presence of bypass voiding in LPRM, APRM, and OPRM channel calibrations to determine the appropriate setpoints. Also, the staff has requested that AREVA document the methodology used to propagate errors induced by the presence of bypass voiding in the OPRM channel calibrations. Additionally, the staff has requested that AREVA provide a generic-basis methodology. In the meantime, the staff will continue to evaluate the impact of OPRM errors on a plant-specific basis.

Recommendation 4:

Additional justification is needed for the adequacy of the proposed 5-percent hot channel oscillation magnitude (HCOM) adjustment to account for the increased oscillation growth ratios expected for operation in expanded flow-window operating domains. The staff should review such justification and document the basis for its acceptability.

Staff Response:

We accept your recommendation for the staff to obtain additional justification and review such justification and document the basis for the acceptability of the proposed 5-percent HCOM adjustment to account for the increased oscillation growth ratios expected at the time of scram when operating in the expanded flow-window operating domains. The staff review will also include the effect that higher decay ratios (DR) could have on the delta CPR, due to the delay of the reactor shutdown after scram initiation, and will describe how the result of a biasing factor of 1.3 in the HCOM DR probability distribution translates into a 5-percent penalty.

Recommendation 5:

Validation of the RAMONA5-FA steady-state dryout correlations for use under unstable oscillation conditions should be documented and submitted for the staff's review and approval.

Staff Response:

We accept your recommendation to review and approve the validation of the RAMONA5-FA steady-state dry-out correlation for use under unstable oscillation conditions. The staff will review additional data provided by AREVA related to the oscillatory flow dry-out measurements and additional details about the oscillatory dry-out benchmarks will be included in the safety evaluation report (SER).

Recommendation 6:

In the final safety evaluation, the staff should justify the adequacy of its proposed 10-percent penalty on the DIVOM slopes calculated by RAMONA5-FA for expanded flow-window operating domains.

Staff Response:

We accept your recommendation. The staff will review the justification proposed by the fuel vendor regarding the adequacy of the proposed 10-percent penalty on the DIVOM slopes calculated by RAMONA5-FA for expanded flow-window operating domains in the final SER. The staff will provide additional discussion in the final SER to demonstrate that a 10-percent DIVOM penalty adequately bounds any RAMONA5-FA uncertainties. The staff estimates that a 10-percent penalty on the DIVOM slope would translate to approximately a 0.03 penalty in the OLMCPR for a given OPRM scram setpoint, which is a significant penalty. Additionally, as a follow-on activity, a more extensive review of the code and its application will be conducted to determine if the penalty can be reduced.

The fuel vendor has acknowledged that they will provide the additional information needed for staff review.

The staff appreciates the Committee's continued interest and collaborative efforts with the staff on the AREVA detect and suppress stability solution and methodology.

Sincerely,

/RA Martin J. Virgilio for/

Luis A. Reyes
Executive Director
for Operations

cc: Chairman Klein
Commissioner Jaczko
Commissioner Lyons
SECY

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

ACRSR-2278

December 27, 2007

Mr. Luis A. Reyes
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: AREVA DETECT AND SUPPRESS STABILITY SOLUTION AND METHODOLOGY

Dear Mr. Reyes:

During the 548th meeting of the Advisory Committee on Reactor Safeguards (ACRS), December 6–8, 2007, we reviewed the staff's draft safety evaluations of AREVA Licensing Topical Reports ANP-10262P, Revision 0, "Enhanced Option III Long Term Stability Solution," and BAW-10255P, Revision 2, "Cycle-Specific DIVOM Methodology Using the RAMONA5-FA Code." The ACRS Thermal-Hydraulic Phenomena Subcommittee also reviewed this matter on November 14, 2007. During these reviews, we had the benefit of presentations by and discussions with representatives of the staff and AREVA. We also had the benefit of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

1. The Enhanced Option III (EO-III) methodology, subject to the limitations and conditions imposed in the staff's draft safety evaluation and recommendations 3 and 4 below, is an acceptable methodology to detect and suppress oscillations in expanded flow window operating domains.

2. The methods and procedures documented in BAW-10255P, Revision 2, subject to the limitations and conditions imposed in the staff's draft safety evaluation and recommendations 3, 5, and 6 below, represent an acceptable methodology to calculate delta critical power ratio (CPR) over initial CPR versus oscillation magnitude (DIVOM) slope values.

3. The applicant's methodology for evaluating the impact of average power range monitor (APRM) and oscillation power range monitor (OPRM) errors caused by bypass voiding should be documented. It would be preferable if such methodology were reviewed and approved on a generic rather than a plant-specific basis.

4. Additional justification is needed for the adequacy of the proposed 5 percent hot channel oscillation magnitude (HCOM) adjustment to account for the increased oscillation growth ratios expected for operation in expanded flow window operating domains. The staff should review such justification and document the basis for its acceptability.

5. Validation of the RAMONA5-FA steady-state dryout correlations for use under unstable oscillation conditions should be documented and submitted for the staff's review and approval.

6. In the final safety evaluation, the staff should justify the adequacy of its proposed 10 percent penalty on the DIVOM slopes calculated by RAMONA5-FA for expanded flow window operating domains.

DISCUSSION

During the past decade, the Boiling Water Reactor Owners Group (BWROG) has developed and the staff has approved three different long-term stability options. Among these is the Option III long-term stability solution, which is a detect and suppress system that relies on signals from the local power range monitors (LPRMs). Small numbers of closely spaced LPRMs are grouped into OPRM cells. The OPRM signals are analyzed on-line; if instability is detected and confirmed, automatic action is taken to suppress the oscillations before compromising the safety margins.

DIVOM correlates the fractional decrease in CPR to the hot channel oscillation magnitude. The DIVOM correlation is used to define the OPRM amplitude scram setpoint. Evaluations by General Electric in 2001 identified a non-conservative deficiency in the generic DIVOM curve developed by the BWROG. For high radial peaking and high peak bundle power-to-flow ratios, the regional mode DIVOM slopes were found to be significantly higher than the licensed generic curve. A high DIVOM slope requires lowering the OPRM scram setpoint, which may result in an increase of false oscillation identifications. The generic DIVOM curve was subsequently eliminated and substituted with a cycle-specific DIVOM analysis.

Since implementation of the long-term stability solutions, two instability events have occurred, one at Nine Mile Point 2 in July 2003 and another at Perry in December 2004. Both events occurred in Option III plants. The Nine Mile Point 2 event was attributed to deficiencies in Option III related to the adjustable parameters for the period-based detection algorithm (PBDA) used to confirm the presence of an instability. The parameters have since been reset to more sensitive settings.

BWRs are licensed to operate within specific power and core-flow conditions referred to as "operating domains" in power-flow maps. In recent years, the industry has been moving toward expanded operating domains with increasing power densities and power-to-flow ratios. This trend is detrimental to the stability characteristics of the reactor, inasmuch as it increases the probability of instability events and increases the severity of such events, if they were to occur. EO-III, documented in AREVA Licensing Topical Report ANP-10262P, is an evolutionary extension of the current Option III detect and suppress solution for use in expanded flow domains up to the maximum extended load line limit analysis-plus (MELLLA+).

The key feature of the EO-III methodology is the recognition that ill-conditioned DIVOM curves are the result of multiple (superimposed) instability mode excitations. In essence, the relationship between the detected parameter (oscillation magnitude) and the fractional change in the limiting parameter (delta CPR over initial CPR) (i.e., the DIVOM relationship) breaks down when multiple instability modes coexist. Multiple instability modes are more likely to occur under expanded flow domain operations. The limiting case corresponds to single (or a few) hydraulic channel oscillations superimposed on the regional mode oscillation.

EO-III resolves the ill-conditioned DIVOM problem by defining an exclusion region enforced by an automatic scram, referred to as the stability protection trip (SPT) region. Single channel hydraulic mode excitations do not occur outside the SPT region. All detect and suppress functions of the current Option III are maintained outside the SPT exclusion region, where the

DIVOM curve should be well behaved. Cycle-specific DIVOM curves based on regional instabilities are calculated for reactor states with hydraulically stable channels. The proposed methodology to define the boundary of the exclusion region using the previously approved STAIF code is acceptable.

The high-growth ratios expected in expanded flow domain operations may not allow sufficiently rapid suppression of the instability to avoid violation of the safety limit minimum critical power ratio (SLMCPR) as the oscillation quickly grows during the scram delay. To address this issue, the applicant imposes a 5 percent penalty on the HCOM to conservatively account for the anticipated increase in the oscillation growth ratios for operation in expanded flow domains up to MELLLA+. AREVA performed sensitivity analyses by scaling the probability distributions of the growth ratio used in the licensing-basis methodology for the Option III detect and suppress solution. It is not clear that the parameter ranges used in these sensitivity analyses cover all expected conditions for expanded flow domain operations. Hence, further analyses to support the adequacy of the 5 percent HCOM penalty are necessary.

Bypass voiding at high-power/low-flow conditions can result in calibration errors for both OPRM cells and APRM signals. Increased voiding reduces the sensitivity of the LPRM detectors, particularly in the upper elevations. The LPRM errors propagate to the OPRM and APRM channels when signals from the LPRM detectors at different levels are combined. OPRM uncertainties will result in a reduction of the OPRM PBDA setpoint, while APRM uncertainties will affect the SPT exclusion region boundary. The EO-III topical report does not address the effects of bypass voiding. The staff proposes that plant-specific EO-III applications should include an evaluation of the uncertainty induced by bypass voiding on the OPRM and APRM readings. The applicant's methodology for evaluating the APRM and OPRM calibration errors and accounting for the effects of such errors on the SPT region boundary and the PBDA setpoint should be documented. To ensure uniformity of application, it would be preferable if such methodology were submitted for review and approval on a generic rather than a plant-specific basis.

Plant-specific EO-III applications will need to address issues related to hardware and software implementation, including provision for backup stability protection if the EO-III primary solution is declared inoperable. We agree with the staff's conclusion that plant-specific applications should include the specifications of the backup stability protection.

Topical Report BAW-10255(P), Revision 2, presents a methodology to evaluate the cycle-specific DIVOM curve using the transient system code RAMONA5-FA. The code is based on RAMONA3, originally developed by Brookhaven National Laboratory and later modified by Studsvik-Scandpower to become RAMONA5 V2.4. Several enhancements have been made in the transition from RAMONA5 V2.4 to RAMONA5-FA. RAMONA5-FA predictions have been compared against reactor event data, as well as data from the Karlstein Thermal Hydraulics (KATHY) stability tests and oscillatory dryout-rewetting tests.

To develop the DIVOM curve, the code needs to correctly model the loss of CPR margin caused by the power-flow oscillation. Comparisons with the KATHY hydraulic loop data and reactor benchmarks show that RAMONA5-FA can adequately predict the frequencies and growth rates of the oscillations. Comparisons between the KATHY oscillatory dryout-rewetting test data and CPR predictions obtained using the RAMONA5-FA steady-state CPR correlations show that the code can predict the dryout times reasonably well. However, the limited data included in topical report BAW-10255(P) suggest a nonconservative bias in the predicted CPR values at the onset of dryout. To ensure adequacy of the safety limit, a quantitative comparison between predictions of the steady-state dryout correlations and the test data for unstable oscillation conditions, including a statistical evaluation of the errors, should be submitted to the staff for review.

While the AREVA DIVOM methodology described in topical report BAW-10255(P) is consistent with the previously approved BWROG methodology for calculating generic DIVOM slope values, the RAMONA5-FA code has not been fully reviewed by the staff. The staff plans to perform a full review of the RAMONA5-FA code, including constitutive relations, numerics, neutronic methods, and benchmarks. In the interim, the staff proposes the addition of a 10 percent penalty to the DIVOM slopes calculated by RAMONA5-FA for expanded flow domain operations. The adequacy of this penalty needs to be demonstrated.

We look forward to further interactions with the staff on these issues.

Sincerely,

/RA/

William J. Shack
Chairman

REFERENCES

1. US NRC, Draft Safety Evaluation for AREVA NP, Inc., Topical Report ANP-10262(P), Revision 0, "Enhanced Option III Long Term Stability Solution," Revised Version, December 3, 2007 (ML073331093)

2. AREVA Topical Report ANP-10262(P), Revision 0, "Enhanced Option III Long Term Stability Solution," January 31, 2006 (ML060330647)

3. AREVA Topical Report, BAW-10255(P), Revision 2, "Cycle-Specific DIVOM Methodology Using the RAMONA5-FA Code," January 30, 2006 (ML060330502)

4. Letter to NRC from R. L. Gardner, AREVA, ANP-10262Q1(P), Revision 1, "Response to Request for Additional Information–ANP-10262(P)," October 2007 (ML073610384)

5. Oak Ridge National Laboratory Technical Evaluation Report, "Evaluation of Licensing Topical Report ANP-10262(P), 'Enhanced Option III Long Term Stability Solution,'" October 2007 (ML073020619)

6. General Electric Company-Nuclear Energy Topical Report, NEDO-31960-A, "BWR Owners Group Long-Term Stability Solutions Licensing Methodology," May 1991

7. General Electric Company-Nuclear Energy Topical Report, NEDO-31960-A, Supplement 1, "BWR Owners Group Long-Term Stability Solutions Licensing Methodology," November 1995

8. General Electric Company-Nuclear Energy Topical Report, NEDO-32465-A, "BWR Owners Group Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications," August 1996

9. Letter to NRC from R. L. Gardner, AREVA, BAW-10255Q1(P), Revision 1, "Response to Request for Additional Information–BAW-10255(P)," October 2007 (ML073610355)

10. Oak Ridge National Laboratory Technical Evaluation Report, Review of AREVA BAW-10255(P), Revision 2, "Cycle-Specific DIVOM Methodology Using the RAMONA5-FA Code," October 2007 (ML073120285)

11. Siemens Power Corporation Topical Report, EMF-CC-074(P)(A), Volumes 1 through 4, "STAIF A Computer Program for BWR Stability Analysis in the Frequency Domain," Siemens Power Corporation, 1993 through 2000 (Volume 1 through Volume 4)

12. US NRC NUREG/CR-3664, W. Wulff, H.S. Cheng, D.J. Diamond, and M. Khatib-Rahbar, "A Description and Assessment of RAMONA-3B MOD. 0 CYCLE 4: A Computer Code with Three-Dimensional Neutron Kinetics for BWR System Transients," 1984 (ADAMS Legacy Library Number 8405210615)

13. Letter to NRC from J. S. Post, General Electric, "Stability Reload Licensing Calculations Using Generic DIVOM Curve," August 31, 2001 (ML012490522)

14. Presentation to NRC from Michael May, Exelon Corp., "Stability Option III DIVOM Part 21 Closure Plan," August 15, 2003