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

November 6, 2002

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Meserve:

SUBJECT: SUMMARY REPORT - 496th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, OCTOBER 10-12, 2002, AND OTHER RELATED ACTIVITIES OF THE COMMITTEE

During its 496th meeting, October 10-12, 2002, the Advisory Committee on Reactor Safeguards (ACRS) discussed several matters and completed the following reports:

REPORTS

The following reports were issued to Richard A. Meserve, Chairman, NRC, from George E. Apostolakis, Chairman, ACRS:

- Confirmatory Research Program on High-Burnup Fuel, dated October 17, 2002
- Draft Report "Guidance for Performance-Based Regulation," dated October 17, 2002

HIGHLIGHTS OF KEY ISSUES

1. Confirmatory Research Program on High-Burnup Fuel

The Committee heard presentations by and held discussions with representatives of the NRC staff regarding the NRC's confirmatory research program on high-burnup fuel, including research on creep of high-burnup fuel cladding. The Committee also heard a presentation by representatives of the Electric Power Research Institute (EPRI) regarding their topical report on reactivity insertion accidents.

The NRC Office of Nuclear Regulatory Research (RES) has undertaken a research program to confirm that the current limit on fuel burnup (62 Gwd/t) is adequate. A research program of experimental and analytic research involving the collaboration of NRC, EPRI, and numerous foreign partners has been organized. That program

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addresses high-burnup fuel behavior under conditions of design-basis loss-of-coolant accidents (LOCAs) and boiling water reactor (BWR) anticipated transients without scram (ATWS), as well as reactivity insertion events in pressurized water reactors (PWRs). RES has also completed phenomena identification and ranking table studies for high-burnup fuel under a variety of conditions. RES currently supports a single criterion for fuel failure, whereas EPRI has proposed the continued use of two separate criteria.

Committee Action

The Committee issued a report to NRC Chairman Meserve on this matter dated October 17, 2002, concluding that RES has a well-organized and leveraged program of confirmatory research on the behavior of high-burnup fuel. However, the Committee remains concerned that the time-temperature conditions used in the study of high-burnup fuel during design-basis LOCAs may not reveal phenomena unique to high-burnup fuel.

2. Overview of European Simplified Boiling Water Reactor (ESBWR), SWR 1000 (Boiling Water Reactor), Advanced CANDU Reactor (ACR 700) Pre-Application Review

The Committee heard presentations by and held discussions with representatives of the NRC staff and industry concerning the NRC staff's schedule for the review of new reactors and the design features of the ESBWR (GE Nuclear Energy), SWR 1000 (Framatome ANP), and ACR 700 (AECL Technologies). These designs are evolutionary water reactor designs with near-term deployment expectations. The ESBWR and SWR 1000 are large boiling water reactor designs. The ACR 700 is a light-water cooled evolution of the CANDU 6 reactor design, utilizing slightly enriched uranium fuel.

Committee Action

This briefing was for the Committee's information. No Committee action was taken at this time.

3. Catawba and McGuire License Renewal Application

The Committee received a briefing by the Chairman of the ACRS Subcommittee on License Renewal on the staff's Safety Evaluation Report (SER), with open items, regarding the Duke Energy Company (Duke) license renewal application (LRA) for the Catawba and McGuire Nuclear Stations. The Committee found the staff's SER to be well written, thorough, and complete. However, the Committee was concerned that (i) open items existed that were very similar to items that had been resolved for previous LRAs (e.g., scoping of fire protection); and (ii) the staff did not have a program to verify implementation of licensee commitments prior to entering the license renewal period.

Committee Action

The Committee decided not to write an interim letter at this time. The Committee plans to review the final SER in February 2003 and provide a report to the Commission.

4. Policy Issues Related to Advanced Reactor Licensing

The Committee heard presentations by and held discussions with representatives of RES regarding issues with potential policy implications resulting from technical considerations related to licensing of future non-light water reactor designs. The staff discussed the schedule and proposed options for resolution of seven policy issues. These policy issues were identified as: expectation for safety; defense-in-depth; use of international codes and standards; event selection; source term; containment vs. confinement; and emergency preparedness.

Currently, no decision has been made regarding the need for a generic licensing approach for future plants as the number and type of future non-LWR plant applications is uncertain. This is supported by Exelon's recent decision to phase out the Pebble Bed Modular Reactor (PBMR) pre-application review. Nevertheless, the NRC staff believes that the establishment of guidance in key areas will benefit all stakeholders by improving the effectiveness, efficiency, and predictability of the review process.

Committee Action

This briefing was for the Committee's information. The Committee plans to discuss this matter during future meetings. The ACRS Subcommittee on Future Plant Designs has scheduled a meeting on November 21, 2002 to continue the discussion on this matter.

5. Program Plan for Low-Power Shutdown (LPSD) Standardized Plant Analysis Risk (SPAR) Model Development and Cancellation of Revision 4i of SPAR Models

The Committee heard presentations by and held discussions with representatives of RES regarding: program plan for LPSD SPAR model development; reasons for cancellation of Revision 4i of SPAR models; insights from onsite review of LPSD SPAR model for Surry Nuclear Power Plant Units 1 and 2; and LPSD SPAR model template for PWRs. There are two LPSD models available for use by the staff based on shutdown PRAs performed at Surry and Grand Gulf. Recently, the staff has begun work on two tasks: (a) use the LPSD model templates to develop LPSD models for the lead plants in the various plant cases; and (b) update and enhance the SPAR model human reliability analysis methodology. The Committee discussed with the staff its reasons for canceling Revision 4i of the SPAR models specifically to develop a response to Commissioner McGaffigan's question with regard to ACRS views on this matter. The staff indicated that since RES had intended to begin Revision 4i of the SPAR models in FY2005, it has not yet defined in detail the scope of this revision. RES has decided to cancel Revision 4i of the SPAR models and focus its resources on the development of models for external events.

Committee Action

The Committee did not issue a letter or report on this matter. With regard to the appropriateness of canceling Revision 4i of the SPAR models, even though details of the planned Revision 4i were not made available, the Committee considered the staff's decision to be appropriate.

6. Guidance for Performance-Based Regulation

The Committee heard presentations by and held discussions with representatives of RES regarding the draft NUREG/BR "Guidance for Performance-Based Regulation." The staff informed the Committee that the research and development effort on performance-based approaches is over. The implementation and execution of the guidance document is meant to be applied to all three arenas of agency activity: reactors, materials, and waste. It was concluded that the performance-based initiative will become part of the overall risk-informed and performance-based activity.

Committee Action

The Committee issued a report to NRC Chairman Meserve on this matter dated October 17, 2002, stating that the Committee agreed with the staff's proposal to publish the guidance document as a NUREG/BR report. Prior to its publication, the staff should provide more discussion of safety margins and performance parameters. The Committee will continue to meet with the staff to discuss further progress on implementation of performance-based regulatory activities.

7. Discussion of Steam Dryer Failure Event at the Quad Cities Nuclear Power Station Unit 2

The Committee discussed issues associated with the steam dryer damage observed at Unit 2 of the Quad Cities Nuclear Power Station in July 2002. Specifically, the unit was forced to shut down subsequent to initiating an Extended Power Uprate (EPU) pursuant to the GE Nuclear Energy EPU Program. Damage to the dryer was believed to be caused by high-cycle fatigue resulting from the increased steam flow at EPU conditions.

Committee Action

The Committee intends to discuss issues resulting from this event with BWR licensees during its future reviews of extended power uprates.

8. Recent Operating Events

The Committee, in its efforts to continue awareness of recent operating events, discussed a pump motor fire at Farley, a hydrogen dryer fire at McGuire, emergency preparedness issues at Cooper, and a potential auxiliary feedwater system problem at Point Beach.

RECONCILIATION OF ACRS COMMENTS AND RECOMMENDATIONS

- The Committee considered the EDO's response of September 3, 2002, to comments and recommendations included in the ACRS report dated July 18, 2002, concerning "Risk Metrics and Criteria for Reevaluating the Technical Basis of the Pressurized Thermal Shock Rule."

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

OTHER RELATED ACTIVITIES OF THE COMMITTEE

During the period from September 12, 2002 through October 9, 2002, the following Subcommittee meetings were held:

- License Renewal - October 8, 2002

The staff briefed the Subcommittee on the safety evaluation report, with open items, for the Duke Energy Company license renewal application for the Catawba and McGuire Nuclear Stations. The applicant also participated in the discussions.

- Reactor Fuels - October 9, 2002

The Subcommittee reviewed RES confirmatory research program on high-burnup fuels and the EPRI topical report on reactivity insertion accidents.

- Planning and Procedures - October 9, 2002

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

- ACRS review of the proposed resolution of GSI-185, "Control of Reactivity Following Small-Break Loss-of-Coolant Accidents in Pressurized Water Reactors" has been postponed, pending receipt of the RES draft final report on its resolution approach and the outcome of subsequent review by the ACRS Subcommittee on Thermal Hydraulic Phenomena.

- The Committee plans to review the final SER related to the Duke Energy Company license renewal application for the Catawba and McGuire Nuclear Stations in February 2003.

- The Committee plans to discuss during a future meeting RES plans to explore the risk consequences of taking fuel to higher levels of burnup.

- The Committee plans to continue to meet with the staff to discuss further progress on the performance-based regulatory activities.

- The Committee plans to discuss policy issues related to advanced reactor licensing during future meetings. The ACRS Subcommittee on Future Plant Designs is scheduled to discuss this matter during a meeting on November 21, 2002.

PROPOSED SCHEDULE FOR THE 497th ACRS MEETING

The Committee agreed to consider the following topics during the 497th ACRS meeting, November 7-9, 2002:

- Proposed Resolution of Generic Safety Issue (GSI)-189, "Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident"
- Early Site Permit Process
- Peach Bottom License Renewal Application—Subcommittee Report
- Westinghouse AP1000 Design
- Risk-Informed Improvements to Standard Technical Specifications
- Organizational and Personnel Matters (Closed)
- Safeguards and Security Activities (Closed)

ACRS Meeting Dates for CY2003

The Committee has approved the following dates for the ACRS meetings in CY2003:

—	January 2003 - No meeting
499	February 6-8, 2003
500	March 6-8, 2003
501	April 10-12, 2003
502	May 8-10, 2003
503	June 11-13, 2003
504	July 9-11, 2003
—	August 2003 - No meeting
505	September 11-13, 2003
506	October 2-4, 2003
507	November 6-8, 2003
508	December 4-6, 2003

Sincerely,

George E. Apostolakis

George E. Apostolakis
Chairman

Date Issued: 11/19/2002
Date Certified: 12/2/2002

TABLE OF CONTENTS
MINUTES OF THE 496th ACRS MEETING

OCTOBER 10-12, 2002

REPORTS

The following reports were issued to Richard A. Meserve, Chairman, NRC, from George E. Apostolakis, Chairman, ACRS:

- Confirmatory Research Program on High-Burnup Fuel, dated October 17, 2002

- Draft Report "Guidance for Performance-Based Regulation," dated October 17, 2002

APPENDICES

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

496th ACRS Meeting
October 10-12, 2002

CERTIFIED

MINUTES OF THE 496th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS OCTOBER 10-12, 2002 ROCKVILLE, MARYLAND

The 496th meeting of the Advisory Committee on Reactor Safeguards (ACRS) was held in Conference Room 2B3, Two White Flint North Building, Rockville, Maryland, on October 10-12, 2002. Notice of this meeting was published in the *Federal Register* on September 26, 2002 (65 FR 60703) (Appendix I). The purpose of this meeting was to discuss and take appropriate action on the items listed in the meeting schedule and outline (Appendix II). The meeting was open to public attendance. There were no written statements or requests for time to make oral statements from members of the public regarding the meeting.

A transcript of selected portions of the meeting is available in the NRC Public Document Room at the One White Flint North Building, Mail Stop 1F-15, Rockville, MD, 20852-2738. [Copies of the transcript are available for purchase from Neal R. Gross and Co., Inc., 1323 Rhode Island Avenue, NW, Washington, DC 20005-3701, and on the ACRS/ACNW Web page at (www.NRC.gov/ACRS/ACNW).]

ATTENDEES

ACRS Members: ACRS Members: Dr. George Apostolakis (Chairman), Dr. Mario V. Bonaca (Vice Chairman), Dr. F. Peter Ford, Dr. Thomas S. Kress, Mr. Graham M. Leitch, Dr. Dana A. Powers, Dr. Victor H. Ransom, Mr. Stephen L. Rosen, Dr. William J. Shack, Mr. John D. Sieber, and Dr. Graham B. Wallis. For a list of other attendees, see Appendix III.

I. Chairman's Report (Open)

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

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

II. Confirmatory Research Program on High-Burnup Fuel (Open)

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

Dr. Dana A. Powers, Chairman of the ACRS Reactor Fuels Subcommittee, stated that the purpose of this meeting was to hear presentations by representatives of the NRC staff regarding the NRC's confirmatory research program on high-burnup fuel, including research on creep of high-burnup fuel cladding to support safety regulation of dry cask fuel storage. The Committee also heard a presentation by representatives of the Electric Power Research Institute (EPRI) regarding A topical report on reactivity initiated accidents.

Ms. Undine Shoop, Office of Nuclear Reactor Regulation (NRR), stated that in a July 6, 1998, memorandum to the Commission from L. Joseph Callan, former Executive Director for Operations, regarding "Agency Program Plan for High Burnup Fuel," the NRC staff prepared a program plan for high burnup fuel. This program plan addressed (a) a range of issues that were previously discussed with the Commission, and (b) provided a licensing and research strategy for confirming the safety of currently approved burnup levels, for considering further burnup extensions that the industry is expected to request. The staff indicated that for all issues, a basis is given for concluding that there is no immediate safety concern at the current burnup limit of 62 Gwd/t. However, confirmatory research is under way for other issues where the basis involved large data uncertainties and analyses. The industry will have to perform the research necessary to develop the data base to support extended burnup ranges higher than 62 Gwd/t.

NRR prepared a review plan to focus resources and to provide a detailed review and identify all the elements needed to complete the review. The elements of the review plan could include data verification, theory and model, fuel rod failure threshold, and core coolability limit. NRR expects to provide the final review plan by December 31, 2002.

Dr. Ralph Meyer, NRC's Office of Nuclear Regulatory Research (RES), stated that RES has undertaken a research program to confirm that the current limit on fuel burnup (62 Gwd/t) is adequate. A research program of experimental and analytic research involving the collaboration of NRC, EPRI, and numerous foreign partners has been organized. Such a program addresses high-burnup fuel behavior under conditions of design-basis loss-of-coolant accidents (LOCAs) and boiling water reactor (BWR) anticipated transients without scram (ATWS), as well as reactivity insertion events in pressurized water reactors (PWRs). RES has also completed phenomena identification and ranking table studies for high-burnup fuel under a variety of conditions. RES currently supports a single criterion for fuel failure.

Ms. Rosa Yang, EPRI, briefed the Committee regarding EPRI's topical report. The purpose of the topical report is to describe the technical bases supporting a set of revised acceptance criteria for use in the safety analysis of the hot-zero power and hot-full power reactivity initiated accidents (RIA) in pressurized water reactors (PWRs) and BWRs. The primary RIA events considered in EPRI's topical report are the postulated control rod ejection accident (REA) for PWRs and the postulated control rod drop accident (RDA) for BWRs. The revised RIA acceptance criteria has been developed as part of the on-going industry efforts to extend fuel rod average burnup levels beyond the current limit of 62 Gwd/t.

EPRI's revised acceptance criteria is defined in terms of the radial average peak fuel enthalpy and as a function of rod average burnup. Two separate criteria have been developed to ensure long-term cooling of the reactor core; and account for radiological release to the environment following cladding failure. The two separate criteria approach is consistent with the NRC Regulatory Guide 1.77 which contains a limit on the maximum radial average fuel enthalpy to satisfy the requirements of 10 CFR Part 50, Appendix A.

Committee Action

The Committee issued a report to NRC Chairman Meserve on this matter dated October 17, 2002, concluding that RES has a well-organized and leveraged program of confirmatory research on the behavior of high-burnup fuel under the conditions of reactivity insertion events in PWRs, design-basis LOCAs, and ATWS in BWRs. However, the Committee remain concerned that the time-temperature conditions used in the study of high-burnup fuel during design-basis LOCAs may not reveal phenomena unique to high-burnup fuel.

III. Overview of European Simplified Boiling Water Reactor (ESBWR) SWR 1000 (Boiling Water Reactor), Advanced CANDU Reactor (ACR 700) Pre-Application Review (Open)

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

Dr. Thomas Kress, Chairman of the Future Reactor Designs Subcommittee, stated that this briefing was intended to provide the ACRS members with an opportunity to learn more about the features of the ESBWR (GE Nuclear Energy), SWR-1000 (Framatome), and the ACR-700 (AECL Technologies) designs. No Committee action was expected at this time. The NRC staff is progressing in its review of the AP-1000 and was working with GE Nuclear Energy (GENE) to develop a schedule for the pre-application review of the ESBWR. The NRR staff is also engaged in discussions with the organizations responsible for the SWR-1000, ACR-700, PMBR, GT-MHR, and IRIS designs.

Industry Presentations

A. S. Rao, General Electric (GE), described the features of the ESBWR design. This design is rated at 1390 MWe, has a number of passive safety systems, and has enhanced natural circulation capabilities as compared to standard BWRs. No pumps are needed to provide coolant flow to the core for normal operation. Experience gained with previous GENE BWR designs were used to develop the simpler ESBWR design. The ESBWR does not have recirculation pumps, safety system pumps, or a safety diesel generator, and has a much smaller safety building volume than previous GENE BWR plants. The taller reactor pressure vessel has a larger water inventory and gravity-driven systems for inventory makeup. The larger vessel water inventory results in improved LOCA performance. The NRC staff is currently engaged in its pre-application review of the ESBWR.

Roger Stoudt described the features of the SWR-1000 design. The design is rated at 1253 MWe, and has evolved from Framatome's BWR product line. All active systems have passive safety-related backup systems to perform nuclear safety functions. The SWR design incorporates safety-related passive systems that are designed to meet all nuclear safety criteria without reliance on active systems. These passive features include an emergency condenser, a passive containment cooling condenser, a passive outflow reducer, and a passive safety system actuation device. The design also has features for core-melt retention in the reactor pressure vessel. Large water inventories are maintained inside the reactor pressure vessel and inside the containment for heat storage and flooding. The NRC staff expects a pre-application submittal for the SWR-1000 in mid FY 2004.

Victor Snell, AECL, described the features of the ACR-700 design. The design is a 700 MWe evolutionary extension of the CANDU 6 reactor design. The ACR-700 is designed to meet customer requirements of $1000/KWe construction cost, a construction schedule of 36 months, a plant operating life of 60 years, and a capacity factor of more than 90%. The ACR-700 will use light water for core cooling, heavy water for the moderator, and slightly enriched uranium fuel. The design includes two fully independent shutdown systems and a steel-lined dry pressure containment. Elevated reserve water tanks provide gravity-driven water to the reactor coolant system and steam generators. The NRC staff expects a design certification application submittal for the ACR-700 in late CY 2004.

Committee Action:

This briefing was for Committee information. No action was taken at this time.

IV. Catawba and McGuire License Renewal Application (Open)

[Note: Mr. Timothy J. Kobetz was the Designated Federal Official for this portion of the meeting.]

The Committee received a briefing by the License Renewal Subcommittee Chairman regarding highlights the staff's safety evaluation report (SER), with open items, regarding the Duke Energy Company (Duke) license renewal application (LRA) for Catawba and McGuire Nuclear Stations. The Committee found the staff's SER to be well written, thorough, and complete. However, the Committee was concerned that open items existed that were very similar in substance to items that had been resolved for other LRAs (e.g., scoping of fire protection). In addition, the Committee was concerned that the staff does not appear to have a program in place to ensure confirmatory items are inspected to verify implementation prior to entering the license renewal period.

Committee Action

The Committee expects to be briefed by the staff once all of the open items have been resolved and write a letter report summarizing the Committee's views on this matter.

V. Policy Issues Related to Advanced Reactor Licensing (Open)

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

Dr. Thomas Kress, Chairman of the ACRS Future Plant Designs Subcommittee, heard presentations by representatives of RES regarding issues with potential policy implications resulting from technical considerations related to licensing of future non-light water reactor designs.

Mr. Thomas King (RES) stated that for licensing of new reactor designs is substantially different than the current generation of light water reactors, the Commission has encouraged interactions between NRC and designers at the pre-application stage to identify early in the process key safety and licensing issues. Recently, the staff began the AP1000 design review and has interacted with Exelon and the Department of Energy (DOE) to identify key issues related to the pebble bed modular reactor (PBMR) and an approach for resolution. In addition, General Atomic has expressed interest in conducting pre-application activities on its gas turbine modular helium reactor (GT-MHR), a 600 Mwt high temperature gas-cooled reactor (HTGR), and that DOE is considering licensing issues in its Generation IV reactor development program.

On June 6, 2002, the NRC staff presented a draft Commission paper to the Committee and provided a status report on issues with potential policy implications resulting from technical considerations associated with advanced reactor licensing. In the draft paper the staff identified five areas:

- Event selection and safety classification
- Fuel performance and qualification
- Source term
- Containment vs. Confinement
- Emergency evacuation

Related to each of the above five areas, the staff considered two overarching policy issues:

- How to implement the Commission's expectation that advanced reactors will provide enhanced margins of safety and/or utilize simplified, inherent, passive, or other innovative means to accomplish their safety functions?
- What should be the relationship of NRC safety requirements to international safety requirements?

The Committee, in its report of June 17, 2002, made the following conclusions and recommendations regarding the draft Commission paper:

1. The RES staff has identified appropriate policy issues and posed questions that must be addressed to resolve them.
2. The existing agency positions on some of these policy issues should be reevaluated because of new perspectives on risk-informed regulation and defense-in-depth, as well as the new reactor designs that may be proposed.
3. The need for greater specificity in the application of defense-in-depth should be made a separate overarching issue.

The EDO responded to the ACRS letter and stated that the staff agreed with recommendations (1) and (3) above, and has revised the policy issue paper to include the defense-in-depth as a separate overarching issue. However, on recommendation (2) above, the staff plans to continue interactions with the ACRS to solicit more views.

Consequently, the staff has revised the draft Commission paper. The revised paper (SECY-02-0139) provides a status report on issues with potential policy implications related to licensing non-light water reactor designs. In this revised version, the staff identified seven policy issues of a more specific technical nature. These are:

- How to implement the Commission's expectations for enhanced safety (as expressed in the Commission's Policy Statements on Advanced Reactors and Severe Accidents).
- How to specify Defense-in-Depth for non-light water reactors.

- How should NRC requirements for non-light water reactors relate to international safety standards and requirements.
- To what extent should a probabilistic approach be used to establish the plant licensing basis.
- Under what conditions, if any, should scenario-specific accident source terms be used for licensing decisions regarding containment and site suitability.
- Under what conditions, if any, can a plant be licensed without a pressure-retaining containment building.
- Under what conditions, if any, can emergency planning zones be reduced, including a reduction to the site exclusion area boundary.

Currently, the RES staff is considering certain options for resolution of the above issues. However, there is no decision that has been made regarding the need for a generic licensing approach for future plants and that the number and type of future non-LWR plant applications is uncertain, especially following Exelon's recent decision to phase out the PBMR pre-application reviews. The NRC staff believes that the establishment of guidance in key areas will benefit all stakeholders by improving the effectiveness, efficiency and predictability of the review process. RES plans to provide the Commission with the proposed options by December 30, 2002.

Committee Action

This briefing was for information only. The Committee will continue to follow this matter during future meetings, including a Subcommittee scheduled for November 21, 2002.

VI. Program Plan for Low-Power Shutdown (LPSD) Standardized Plant Analysis Risk (SPAR) Model Development and Cancellation of Revision 4i of SPAR Models (Open)

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

Dr. Powers, cognizant ACRS Member, provided a preamble, stating that the purpose of the session is to discuss the following with representatives of the Office of Nuclear Regulatory Research (RES):

- Program plan for LPSD SPAR model development
- Insights from onsite review of the LPSD SPAR model for Surry Nuclear Power Plant Units 1 and 2
- Scope of the proposed Revision 4i of SPAR models and the reasons for canceling this Revision
- LPSD SPAR model template for PWRs.

Dr. Powers stated that during the review of the RES budget, Commissioner McGaffigan asked about ACRS views on the appropriateness of canceling proposed Revision 4i of SPAR models. The ACRS had not received any documentation that defines the scope of the proposed Revision 4i and he suggested that the RES staff explain to the Committee the scope of Revision 4i and the reasons for canceling this revision.

Program Plan for LPSD SPAR Model Development

Mr. O'Reilly, RES, briefly described the program plan for LPSD SPAR model development. Key points made by Mr. O'Reilly include the following:

- There are two LPSD models available for use by the staff. One is a pressurized water reactor (PWR) LPSD model based on the detailed Surry shutdown PRA that is documented in NUREG/CR-6144, "Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Surry, Unit 1" and the other a boiling water reactor (BWR) LPSD model based on the detailed Grand Gulf Shutdown PRA documented in NUREG/CR-6143, "Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Grand Gulf, Unit 1."

- Prototype templates and associated guidelines have been developed for: all PWRs; BWR 5/6s; and BWR 4s.

- RES has identified lead plants (Millstone 3, Byron 1 and 2, Oconee 1, 2, and 3, Millstone 2, Palo Verde 1, 2, and 3, Peach Bottom 1 and 2, Surry 1 and 2, and Grand Gulf) for which LPSD models will be developed by using the existing LPSD SPAR model template.

- RES has updated Human Reliability Analysis (HRA) methodology and documentation for SPAR models to: ensure that methodology and documentation comply with proposed ASME Standard on PRA; add uncertainty analysis capability; and to provide referenced documents on SPAR HRA methodology.

Stating that the authors of the Surry shutdown PRA describes the study as a scoping study contrary to the staff's claim that it is a detailed study, Dr. Powers asked whether it was a difference in perspectives. Mr. O'Reilly responded that it may be a difference in perspective, but he believes that the authors downplayed what they did in the study.

In response to a question from Mr. Leitch, Mr. O'Reilly stated that initially the SPAR models covered only full power operation and later expanded to meet the needs of the staff analysts to analyze other areas such as low-power and shutdown operations and external events.

In response to a question from Mr. Rosen regarding the results of benchmarking the NRC SPAR models with the licensees' PRAs, Mr. O'Reilly stated that in some plants they were able to reproduce the results exactly. In some cases, there were many differences. He added that using the licensees' PRA models would be difficult because they have not undergone a thorough review and also they vary plant to plant.

Onsite Review of the LPSD SPAR Model for Surry Nuclear Plant

Mr. O'Reilly discussed briefly the insights gained from the onsite review of the LPSD SPAR model for the Surry Nuclear Power Plant Units 1 and 2. Key points made by Mr. O'Reilly include the following:

- In general, there was good agreement between LPSD SPAR model and the Surry LPSD PRA.
- Items to be considered in the development of future SPAR models include:
  - The potential for containment sump plugging during LPSD operation appears to have a much higher likelihood compared to that during full power operation because of the increased level of personnel activity in the plant during LPSD operation.
  - Some plants operate in mid-loop with RCS closed.
  - Reflux cooling is only possible when RCS is closed and can be modeled as a passive phenomenon.

In response to a question from Dr. Kress whether the staff has the plant-specific database associated with plant shutdowns, Mr. O'Reilly said that they do not have such information at this time. However, they use the information obtained for performing the Surry shutdown PRA. During the onsite review, they try to get such information updated by the licensee.

Dr. Kress asked whether the staff plans to compare the LPSD SPAR models with the LPSD operations standard being prepared by the American Nuclear Society (ANS). Mr. Baranowsky responded that all of the SPAR model development work is being done in light of the ASME PRA standard. However, they still need to improve on the documentation.

In response to a question from Mr. Rosen whether the staff has considered using a peer review team from the industry to review the SPAR models, Mr. Baranowsky stated that the staff has not done so. However, he believes that onsite review of the LPSD SPAR models with plant-specific LPSD PRAs serves the same purpose. Mr. Rosen suggested that the staff consider using a industry peer review group to review the SPAR models, which he believes would provide insights on the weakness of the models.

<u>Revision 4i of the SPAR Models and the Reasons for Canceling this Revision</u>

Mr. O'Reilly discussed the scope of the proposed Revision 4i of SPAR models and the reasons for canceling this Revision. He said that Revision 3 of SPAR models was developed to:

- Add more initiating events (e.g., medium and large LOCAs, secondary system initiating events)
- Model other support systems (service water system, component cooling water system, etc.) besides ac power systems
- Enhance treatment of CCFs
- Add uncertainty analysis capability for equipment performance
- Add new human reliability analysis methodology.

Mr. O'Reilly stated that Revision 3 of SPAR models captures about 80-85% of internal events. The estimated total cost of Revision 3 SPAR models development is $3.8 million. The proposed Revision 4i is intended to enhance and improve Revision 3 for modeling of more systems not covered by Revision 3. The cost-benefit of developing Revision 4i to capture 15-20% of the systems not captured by Revision 3 is difficult to justify. They could modify any of Revision 3 SPAR models for a modest cost to capture other systems not captured by Revision 3.

Dr. Powers stated that since the staff did not specify the details of Revision 4i, it would be difficult to assess the appropriateness of canceling Revision 4i. He asked whether the staff could provide more details on the scope of Revision 4i. Mr. Baranowsky responded that since Revision 4i was intended to begin in FY 2005, the staff did not define in detail the scope of Revision 4i. After further discussion, the staff stated that they decided to cancel the proposed Revision 4i based on cost-benefit considerations. They plan to focus its resources on the development of SPAR models for external events.

Mr. O'Reilly briefly discussed the LPSD operation SPAR model template for PWRs developed by INEEL. The template is a starting point for developing plant-specific LPSD model that includes the core damage risk resulting from loss of RHR events, loss of offsite power events, and loss of inventory events.

Committee Action

The Committee decided that since details of the Revision 4i of SPAR models were not available, it would be difficult to assess the appropriateness of canceling this revision. However, since the staff intends to focus its resources on the development of SPAR models for external events, the Committee considered the staff's decision to cancel the planned Revision 4i of SPAR models appropriate.

VII. Guidance for Performance-Based Regulation (Open)

[Note: Mr. Howard J. Larson was the Designated Federal Official and Mr. August W. Cronenberg was the cognizant staff engineer for this portion of the meeting.]

Dr. George Apostolakis, Chairman of the Reliability and PRA Subcommittee, stated that the Committee would hear from Mr. Prasad Kadambi of the RES staff to discuss results of their efforts to formulate high-level guidance for performance-based regulatory activities. Mr. Kadambi noted the presence of Mr. John Flack (RES) and Mr. Christopher Grimes (NRR) from the staff.

Mr. Flack stated that the discussions today would center on high-level guidelines for incorporating performance-based concepts into the regulatory framework, such as rulemaking activities, regulatory guidance, and technical specifications. Mr. Flack went on to note that in November the Committee would hear from the staff on how such performance-based concepts were to be incorporated into agency 'coherence activities.' He stated that the staff requested a letter of endorsement from the Committee for publication of the draft guidance ("Guidance for Performance-Based Regulation") as a NUREG/BR-report.

Mr. Kadambi stated that the developmental phase of the NRC's performance-based regulatory initiative was now complete with the availability of the subject guidance document. He also noted that based on RES efforts, the staff should be able to move forward with plans to incorporate performance-based principles within the scope of regulatory coherence activities. He noted that the report was a first step in satisfying the goals stated in the Commission's White Paper for an integrated process for risk-informed/performance-based regulation. Mr. Kadambi then went on to provide a historical perspective of the impetus of development of the guidance, and noted several SECY documents, specifically SECY-99-281 and SECY-00-91 ("High Level Guidelines for Performance-Based Activities") and NUREG/CR-5392 ("Elements of an Approach to Performance-Based Regulatory Oversight").

Mr. Kadambi noted feedback from the agency's performance-based regulation working group, indicating that if the guidelines were at too high a level, it would not be useful. Past comments indicated that any performance-based should articulate attributes for regulation, but not the direction for implementation. He noted that the staff's first attempt at developing implementation guidance resulted in a overly formal and general approach using decision theory.

Mr. Kadambi provided several examples of recent regulatory activities which incorporated performance-based attributes, including the proposed 10CFR50.44 rulemaking for monitoring hydrogen buildup in LWR plants. Mr. Graham Leitch, ACRS Member, asked if 10 CFR 50.44 implies just performance-based monitoring of hydrogen buildup. Mr. Kadambi responded no, rather 10CFR50.44 is both risk-informed and performance-based, with no particular emphasis on performance attributes. He stated that 10CFR50.44 provides one recent example of how performance measures can be used in the regulatory process.

Mr. Kadambi continued by illustrating the essential features of the approach suggested for performance-based guidance, as documented in the draft NUREG/BR-report. He outlined a four-step process as follows: Step 1: Define Regulatory Issue and Its Context, which addresses the four NRC performance goals. Step 2: Identify The Safety Function, where, for example, a primary safety function for a fuel storage facility is stability against vibratory motion. Step 3: Identify Appropriate Safety Margins, such as design limits. Step 4: Select Performance Parameters/Criteria, where the level at which performance will be evaluated is considered as well as cost effective implementation. Dr. Apostolakis stated that some definition of "Safety Function" is critical to this effort, but not well defined in step-2 of the draft document. He also stated that step-3 does not adequately define what is meant by margin.... "margin to CDF, margin to a design parameter, margin to critical timing for operator action," which are several examples. He noted that all are margins, but each is a different margin. Mr. Kadambi noted that in the report it is stated that 'Safety Function' is defined as the safety function or functions that can impact the regulatory issue at hand. He noted that step-2 goes onto ask questions of the staff developing the performance-based guidance, what equipment, systems, or procedures are necessary to satisfy the 'safety function' and what level of safety is required to meet regulations. He said the correct use of these high level guides should provide a signature (identify) of the important safety function to be met. Dr. Wallis, ACRS Member, stated that these criteria or guidelines for identifying safety function(s) were too vague. Dr. Bonaca, ACRS Member, stated that the same was true for step-3 (Identify Appropriate Safety Margins), that the guidance in the draft document were at too high a level. In response Mr. Kadambi agreed that the staff would revisit the issue.

Mr. Kadambi concluded by stating that the staff was requesting a letter that provides ACRS views on the approaches outlined in the draft guidance document. He noted that such publication would end the developmental phase of the performance-based regulatory initiative. He said that endorsement of this approach would enable more rapid progress toward increasing the use of performance-based approaches in a broader range of regulatory activities. Mr. Kadambi further noted that plans call for the 'Risk Management Team,' to provide policy direction for implementation of any performance-based regulation activities. Finally, he stated that the RES staff plans to develop another NUREG document in FY-2003, which will provide more detail on formal decision methods in support of performance-based regulatory activities.

Committee Action

The Committee issued a report to NRC Chairman Meserve on this matter dated October 17, 2002, stating that the Committee agreed with the staff's proposal to publish the guidance document as a NUREG/BR report. Prior to its publication, the staff should provide more discussion of safety margins and performance parameters. The Committee will continue to meet with the staff to discuss further progress on implementation of performance-based regulatory activities.

X. Executive Session (Open)

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

A. Reconciliation of ACRS Comments and Recommendations

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

- The Committee considered the EDO's response of September 3, 2002, to comments and recommendations included in the ACRS report dated July 18, 2002, concerning "Risk Metrics and Criteria for Reevaluating the Technical Basis of the Pressurized Thermal Shock Rule."

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

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

The Committee heard a report from the ACRS Chairman and the Executive Director, ACRS, regarding the Planning and Procedures Subcommittee meeting held on October 9, 2002. The following items were discussed:

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

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

Anticipated Workload for ACRS Members

The anticipated workload for ACRS members through December 2002 was discussed. The objectives were:

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

During this session, the Subcommittee also discussed and developed recommendations included in the Future Activities List.

Foreign Travel Update

The final travel arrangements for the October foreign travel have been made and a detailed itinerary has been put together by Tanya Winfrey. For the Germany portion of the trip (Quadripartite Meeting), the cost of the sleeping rooms and an ancillary fee have been prepaid by the Government in the form of a registration fee for both the members and staff. Therefore, members should not have to pay any rooming charge for their stay in Germany during the week of the Quadripartite Meeting.

The technical papers for the Quadripartite Meeting have been sent to the RSK and the Commissioners. The slides for presenting those papers are done. Sherry Meador is coordinating the translation so that both English and German versions of the presentations and overheads are available at the meeting in Germany.

Celebration of the 500th ACRS Meeting

As agreed to by the members, invitations were sent to the NRC Commissioners to participate at the 500th ACRS meeting ceremony, which is scheduled for March 4-5, 2003. (This is also coincidental with the Committee's 50th Anniversary.) NRC Chairman Meserve, all Commissioners, and Bill Travers, EDO, have agreed to participate. Drs. Hal Lewis, Robert Seale, Bill Stratton, J. Ernest Wilkins, Stephen Hanauer, and Mr. Dave Ward, have agreed to serve on the panels. Dr. Remick has agreed to be a lunch time speaker. Mr. Ralph Beedle, NEI, and Mr. Bert Wolf, GE, have agreed to participate in the celebration.

A letter has been forwarded to all of the Panel participants and speakers thanking them for their willingness to participate in the 500th meeting ceremony. Additionally, accommodations are being arranged for all of the guests to stay in the area along with developing transportation from the hotel to the White Flint building.

Role and Use of PRA in the Regulatory Decisionmaking Process

Mr. Karl Fleming of Technology Insights, has started to plan for the Committee's "white paper" addressing the role and use of PRA in the regulatory decisionmaking process. He will conduct interviews with NRC staff on October 10-11, 2002, on what needs to be done to enhance PRA submittals. Mr. Fleming will present a draft plan for researching and compiling the information during the Saturday session of the October full Committee meeting. To the extent time permits, he will meet with members individually during the October meeting. Dr. Hossien Nourbakhsh has been designated as the Project Manager for this activity and will work closely with the contractor to guide this effort.

Meeting with the EDO

The Planning and Procedures Subcommittee met with the EDO and the Deputy EDOs on Friday, October 12, 2002, to discuss several issues, including:

- ACRS/NRC staff coordination
- Adequacy of the NRC staff's review of power uprate applications
- License Renewal Issues
- High Burnup Fuel Issues
- Revision 1 to Reg. Guide 1.174

- Significant issues that the NRC staff expects to submit to the ACRS for review in the next two years.
- DPV regarding proposed 10 CFR 50.69

As suggested by the Committee during its September meeting, a formal meeting between the EDO/DEDOs/Office Directors and ACRS will be scheduled during a future ACRS meeting.

Quad Cities Unit 2 - Damage to Steam Generator Dryer Resulting from Power Uprate Operation

On July 11, 2002, Quad Cities Unit 2 was shut down to investigate the irregularities in the steam flow, reactor pressure and level, and moisture carryover in the main steamlines. The results of the investigation revealed that a cover plate of the steam dryer was missing. Subsequent investigation led to the identification of pieces of the cover plate. A large piece was found in the Separator and there were indications that pieces had been transported into "A" main steamline and the vessel. Most of the pieces have been retrieved.

The cause of the damage is believed to be due to high-cycle fatigue induced by the cover plate natural frequency, nozzle chamber standing wave acoustic frequency, and vortex shedding frequency -- all coinciding at 180 Hz.

During the September 2002 ACRS meeting, the Committee asked Drs. Ford and Ransom to review this matter and propose a course of action, noting any generic implications.

ACRS Senior Fellow

A contract for the remaining ACRS Senior Fellow position was awarded to Link Technologies, Inc. The company will provide the equivalent of one full time employee (FTE) over the course of next year. The FTE is equivalent to 2087 hours of manpower. The company has proposed the use of eight individuals who offer a wide array of expertise. The individuals are Dr. John Austin, Ali Tabatabai, Dr. Spyros Traiforos. Dr. Bernard Snyder, Phillip McKee, Jeff Woody, Charles Haughney, and Peter Kiang. In order to effectively utilize this contract, a work plan is being developed. The work plan will coincide with high priority topics under review by the Committee. Since the contract has been executed effective September 30, a plan needs to be put in place as soon as possible preferably by October 30, 2002. Dr. Savio, in consultation with Dr. Bahadur, has been designated as the Project Manager. Suggested topics for the work plan should be provided to Dr. Savio.

Proposed Tasks for Dr. Nourbakhsh, ACRS Senior Fellow

During the September meeting, the Subcommittee proposed the following tasks for Dr. Nourbakhsh and recommended that members propose other tasks:

- Review NUREG-1150 to see if parameter and model uncertainties can be extracted from the overall uncertainty in order to have an estimate of just the model uncertainty contribution. This estimate could then be used in regulatory decisions involving PRA results that include only parameter uncertainty. [May need to use current PRAs with parameter uncertainty quantified for the NUREG-1150 reference plants.]

- The ACRS has proposed that frequency-consequence (F-C) curves could replace or supplement CDF and LERF as a risk-acceptance metric that would capture the full range of potential radioactivity releases and be made consistent with the safety goals. A White paper is needed to "flesh out" this proposal:

  What Consequence to Use? Would it be TEDE?
  What are the F-C values produced by PRAs?
  How are these related to CDF and LERF?
  What would be a reasonable 3-region set of curves to use as risk-acceptance values?

On September 29, 2002, Dr. Wallis suggested that Dr. Nourbakhsh look at the history of the "momentum equation" in RELAP and other codes and advise the Committee about what should be done.

Dr. Nourbakhsh has provided his initial thoughts on the feasibility of extracting parameter and model uncertainties from NUREG-1150 overall uncertainty.

ACRS Retreat for 2003

Due to budget limitations, the ACRS retreat is scheduled to be held in Rockville on January 23-25, 2003. The topics below have been proposed by the Planning and Procedures Subcommittee and Dr. Powers.

Topics proposed by the Planning and Procedures Subcommittee include

- Dr. Fleming's draft "White Paper" on PRA.
- Mr. Rosen's report on Davis-Besse
- Certification process for advanced reactors
- Member issues (process, ACRS practice)

Subsequently, Dr. Powers suggested that the following issues associated with ACRS process and procedures be discussed at the retreat:

- ACRS effectiveness and self-assessment
- Proliferation of Subcommittee meetings
- Compensation process
- ACRS strategy for reviewing technical issues -- proactive vs. reactive
- Member assignments and Subcommittee responsibilities
- Discussion and resolution of differing technical views among ACRS members in areas such as risk-informed regulatory process at the meetings instead of via electronic medium.

As requested by the ACRS Chairman, Dr. Bonaca, ACRS Vice Chairman, discussed the issues raised by Dr. Powers during the Subcommittee meeting.

ACRS Meeting Dates for CY 2003

Proposed ACRS meeting dates for CY 2003 were provided to the members during the September ACRS meeting. [Note: In response to members' comments the June meeting has been moved from June 4-6 to June 11-13, and the September meeting from 3-5 to 11-13.]

Tour of Vender Facilities

Members and Consultants of the Thermal-Hydraulic Phenomena Subcommittee toured the GE facilities in San Jose, CA, on September 23-24 and the Framatome ANP-Richland facilities in Richland, WA on September 25-26, 2002, to obtain information on the details of fuel and reactor core design methodology for use by the Committee in reviewing core power uprate applications. Dr. Apostolakis suggested that Dr. Wallis provide a report to the Committee on this matter.

Tour of Global Nuclear Fuel Cycle Facility

The members of the joint Subcommittee of ACRS/ACNW (Kress, Garrick, and Levenson) are scheduled to tour the Global Nuclear Fuel Cycle Facility in Wilmington, NC, on November 5, 2002. The ACRS members of the joint Subcommittee, Drs. Kress and Powers, decided not to attend this tour. Since only two ACNW members plan to attend this tour, the Subcommittee needs to decide whether it is worthwhile having this tour.

Staff Response to Dr. Kenneth D. Bergeron Regarding TVA's License Amendment Request to Produce Tritium at the Watts Bar Nuclear Power Plant

In a letter of September 13, 2001, Dr. Kenneth Bergeron, formerly associated with the Sandia National Laboratories, and now a member of the public, expressed concern about the NRC staff's review of TVA's license amendment request to irradiate tritium-producing burnable absorber rods at the Watts Bar Nuclear Power Plant. As suggested by the Committee during the October 2001 meeting, Dr. Larkins sent a memorandum to the EDO on October 18, 2001 transmitting Dr. Bergeron's letter and requesting that the EDO keep the ACRS informed of the staff's disposition of Dr. Bergeron's concerns.

The staff sent a letter to Dr. Bergeron on September 6, 2002 responding to his concerns. The staff also sent a memorandum to Dr. Larkins informing him of the staff's response to Dr. Bergeron.

Meeting with Laurence Williams, NII

As requested previously by the Planning and Procedures Subcommittee, Dr. Larkins has contacted the NII Chief Inspector's office in the UK concerning a meeting with Laurence Williams, Chief Inspector for NII, and the ACRS. We have been notified by the NII Chief Inspector's Technical Support Staff that Mr. Williams will be in the U.S. in December and more specifically in the Washington area on the 5th of December, and it may be possible to spend some time with some of the ACRS members. Dr. Larkins will e-mail the Chief Inspector's office and set up a time for Mr. Williams to meet the ACRS members during the 498th meeting, December 5-7, 2002. The discussions will center around the certification of the AP1000 design. As we understand it, the UK has considered building two advanced reactor plants and is currently focusing on the Westinghouse AP1000. As such, they are interested in the certification process required by the NRC, including ACRS involvement.

MEMBER ISSUES

Implications of Boric Acid

Dr. Kress sent an e-mail stating that the Davis-Besse event raises broader concerns than the safety culture issue and he believes that it is a "boric acid" issue. He states that boric acid is used simply for convenience and flexibility in controlling the reactivity as burnup proceeds during a fuel cycle. He believes it is not really needed. BWRs do very well without boric acid and by using burnable poisons (gadolinium), and PWRs could do just as well. He suggests that the Committee urge the staff to take a broader look at the implications of boric acid other than just what happened at Davis-Besse. Mr. Sieber does not agree with Dr. Kress' idea about eliminating the use of boric acid in PWR coolant systems.

C. Future Meeting Agenda

Appendix IV summarizes the proposed items endorsed by the Committee for the 497th ACRS Meeting, November 7-9, 2002.

The 496th ACRS meeting was adjourned at 11:30 a.m. on October 12, 2002.

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

December 2, 2002

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

FROM: George E. Apostolakis, Chairman
Advisory Committee on Reactor Safeguards

SUBJECT: CERTIFIED MINUTES OF THE 496th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS), OCTOBER 10-12, 2002

I certify that based on my review of the minutes from the 496th ACRS full Committee meeting, and to the best of my knowledge and belief, I have observed no substantive errors or omissions in the record of this proceeding subject to the comments noted below.

Sincerely,

George E. Apostolakis
Chairman

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

November 19, 2002

MEMORANDUM TO: ACRS Members

FROM: Sherry Meador Sherry Meador
Technical Secretary

SUBJECT: PROPOSED MINUTES OF THE 496th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS - OCTOBER 10-12, 2002

Enclosed are the proposed minutes of the 496th meeting of the ACRS. This draft is being provided to give you an opportunity to review the record of this meeting and provide comments. Your comments will be incorporated into the final certified set of minutes as appropriate.

Attachment:
As stated

3:45–4 p.m. Break
4–5 p.m. *Issues 2: Defense-in-Depth*
—Options for Resolution:
- Case-by-case determination
- Develop a description of defense-on-depth attributes
- Develop a process for determining defense-in-depth measures

—Key Considerations
- IAEA description

October 23, 2002

9–10 a.m. Issue 3: Use of International Codes and Standards
—Options for Resolution:
- Review and endorse international codes and standards only as needed to review an application
- Review and endorse existing international codes and standards, whenever practical to fill gaps in the NRC infrastructure
- Participate in the development of and endorse international codes and standards, whenever practical to fill gaps in the NRC infrastructure
- Harmonize, as much as possible, NRC licensing requirements with other regulatory bodies

—Key Considerations:
- Cost and schedule

10–10:15 a.m. Break

10:15–11:15 a.m. Issue 4: Event Selection and Safety Classification

—Options for Resolution:
- Deterministic selection of events to be considered in the design and safety classification
- Probabilistic selection of events to be considered in the design and safety classification.
- Combination of deterministic and probabilistic

—Key Considerations:
- Probabilistic risk assessment quality and completeness

11:15–12:30 p.m. Lunch

12:30–1:30 p.m. Issue 5: Licensing Source Term

—Options for Resolution:
- Use a bounding source term representative of core damage accident (LWR practice)
- Use a range of source terms corresponding to the design basis accident scenerios
- Key Considerations:
- Ensuring plant and fuel performance over the life of the plant

1:30–2:30 p.m. Issue 6: Containment vs. Confinement

—Options for Resolution:
- Require non-LWRs to have a pressure retaining containment building
- Allow a design without a pressure retaining containment provided certain performance criteria are met

—Key Considerations:
- Confidence in accident selection and source term

2:30–2:45 p.m. Break
2:45–3:45 p.m. Issue 7: Emergency Preparedness
—Options for Resolution:
- Retain current requirements and emergency planning zone size
- Allow a reduction in emergency planning zone size

—Key Considerations:
- Confidence in accident selection and source term

3:45–5 p.m. General Discussion and Wrapup

Dated at Rockville, Maryland, this 20th day of September 2002.

For the Nuclear Regulatory Commission.

Farouk Eltawila,

Director, Division of Systems Analsis and Regulatory Effectiveness, Office of Nuclear Regulatory Research.

[FR Doc. 02–24439 Filed 9–25–02; 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 October 10–12, 2002, in Conference Room T–2B3, 11545 Rockville Pike, Rockville, Maryland. The date of this meeting was previously published in the **Federal Register** on Monday, November 26, 2001 (66 FR 59034).

Thursday, October 10, 2002

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

8:35 a.m.–10 a.m.: Confirmatory Research Program on High-Burnup Fuel (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC Staff and the Electric Power Research Institute (EPRI) regarding the confirmatory research program on high-burnup fuel and the EPRI topical report on reactivity insertion accidents.

10:15 a.m.–12:15 p.m.: Overview of European Simplified Boiling Water Reactor (ESBWR), SWR 1000 (Boiling Water Reactor), Advanced CANDU Reactor (ACR 700) Pre-Application Review (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC Staff and industry on the ESBWR (General Electric 1380 MWe), SWR 1000 (Framatome ANP-Siemens 1000 Mwe) and ACR 700 (Advanced CANDU Reactor 700 Mwe) advanced reactor designs.

1:15 p.m.–2:45 p.m.: Catawba and McGuire License Renewal Application (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff and Duke Energy Company regarding the license renewal application and draft Safety Evaluation Report for the Catawba Nuclear Station Units 1 and 2 and McGuire Nuclear Station Units 1 and 2.

2:45 p.m.–4 p.m.: Policy Issues Related to Advanced Reactor Licensing (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding changes to policy issues related to the licensing of advanced reactors resulting from the resolution of ACRS comments and recommendations included in its June 17, 2002 report.

4:15 p.m.–7 p.m.: Proposed ACRS Reports (Open)—The Committee will discuss proposed ACRS reports on matters considered during this meeting.

Friday, October 11, 2002

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

8:35 a.m.–10 a.m.: Program Plan for Low-Power Shutdown (LPSD) Standardized Plant Analysis Risk (SPAR) Model Development and Cancellation of Revision 4i of SPAR Models (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the program plan for LPSD SPAR Model Development, the reasons for canceling plans for the development of revision 4i of the SPAR Models, and insights from the onsite review of the LPSD SPAR model for Surry Nuclear Plant Units 1 and 2.

10:15 a.m.–11:30 a.m.: Guidance for Performance-Based Regulation (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the draft NUREG/BR "Guidance for Performance-Based Regulation."

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 (EDO) to comments and recommendations included in recent ACRS reports and letters. The EDO responses are expected to be made available to the Committee prior to the meeting.

1:15 p.m.–2:15 p.m.: Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open)—The

Committee will discuss the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee during future meetings. Also, it will hear a report of the Planning and Procedures Subcommittee on matters related to the conduct of ACRS business, and organizational and personnel matters relating to the ACRS.

2:15 p.m.–3:15 p.m.: Report Regarding Recent Operating Events (Open)—Report by the cognizant ACRS member regarding recent operating events of interest.

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

Saturday, October 12, 2002

8:30 a.m.–1 p.m.: Proposed ACRS Reports (Open)—The Committee will discuss 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. The Committee will also discuss its plans for preparing a "white paper" on the use of PRA in the regulatory decisionmaking process.

Procedures for the conduct of and participation in ACRS meetings were published in the **Federal Register** on October 3, 2001 (66 FR 50462). 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 Associate Director for Technical Support 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 Associate Director 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 Associate Director if such rescheduling would result in major inconvenience.

Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements, and the time allotted therefor can be obtained by contacting Dr. Sher Bahadur, Associate Director for Technical Support (301–415–0138), between 7:30 a.m. and 4:15 p.m., EDT.

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

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

Dated: September 20, 2002.

Annette Vietti-Cook,

Secretary of the Commission.

[FR Doc. 02–24437 Filed 9–25–02; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Draft Decommissioning Questions and Answers Regarding Clarification of License Termination Guidance of the Nuclear Regulatory Commission's Office Nuclear Material Safety and Safeguards; Notice of Availability

AGENCY: Nuclear Regulatory Commission.

ACTION: Notice of availability and request for public comment.

SUMMARY: The Nuclear Regulatory Commission's (NRC) Office of Nuclear Material Safety and Safeguards (NMSS) is announcing the availability of draft decommissioning questions and answers regarding clarification of license termination guidance, for public comment.

The Nuclear Energy Institute (NEI) and NRC staff identified an approach to clarify existing guidance associated with the License Termination Rule (10 CFR part 20, subpart E), in concert with NMSS" decommissioning guidance consolidation project. Under this approach, NEI's License Termination Task Force (Task Force) generated questions (Qs) associated with decommissioning issues that are common to the industry. The Task Force also proposed answers (As) to the questions and submitted the Q&As to NRC staff for review. NRC staff reviewed the Q&As and the supporting technical bases and provided comments to NEI on September 28, 2001. An open meeting was held between NRC, NEI, and industry representatives on December 4, 2001, to discuss each Q&A and the technical issues to ensure that the questions were properly asked and answered and were supported by a defensible technical basis. NRC staff and NEI further developed the Q&As so that they adequately reflect NRC regulations and guidance and include a sound technical basis.

As a result of this cooperation, eight Q&As have been found acceptable by NRC staff. Seven of the Q&As were to be incorporated into the draft document "Consolidated NMSS Decommissioning Guidance: Characterization, Survey, and Determination of Radiological Criteria" (NUREG–1757, Volume 2) to solicit public comment on them. However, two Q&As were inadvertently omitted. Therefore, five Q&As are included in Volume 2 of NUREG–1757, and three Q&As are included in the "supplementary information" section of this notice. Volume 2 of NUREG–1757 is being published for public comment on or close to the date of this notice. NRC is seeking public comment on the Q&As and Volume 2 of NUREG–1757 in order to receive feedback from the widest range of interested parties and to ensure that all information relevant to developing the document is available to the NRC staff. These draft documents are being issued for comment only and are not intended for interim use. The NRC will review public comments received on the draft documents. Suggested changes will be incorporated, where appropriate, in response to those comments, and a final document will be issued for use. The final Q&As will be included in the text of the final document of Volume 2 of NUREG–1757.

DATES: Comments on this draft document should be submitted by December 26, 2002. Comments received after that date will be considered to the extent practicable.

APPENDIX II

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

September 19, 2002

SCHEDULE AND OUTLINE FOR DISCUSSION
496th ACRS MEETING
OCTOBER 10-12, 2002

<u>THURSDAY, OCTOBER 10, 2002, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND</u>

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

2) 8:35 - ~~10:00~~ 10:10 A.M. <u>Confirmatory Research Program on High-Burnup Fuel</u> (Open) (DAP/TSK/MME)
- 2.1) Remarks by the Subcommittee Chairman
- 2.2) Briefing by and discussions with representatives of the NRC staff and the Electric Power Research Institute (EPRI) regarding the confirmatory research program on high-burnup fuel and the EPRI topical report on reactivity insertion accidents.

10:10 - 10:25
~~10:00~~ - ~~10:15~~ A.M. *****BREAK*****

3) 10:25 - 12:30
~~10:15~~ - ~~12:15~~ P.M. <u>Overview of European Simplified Boiling Water Reactor (ESBWR), SWR 1000 (Boiling Water Reactor), Advanced CANDU Reactor (ACR 700) Pre-Application Review</u> (Open) (TSK/RPS)
- 3.1) Remarks by the Subcommittee Chairman
- 3.2) Briefing by and discussions with representatives of the NRC staff and industry on the ESBWR (General Electric 1380 Mwe), SWR 1000 (Framatome ANP-Siemens 1000 Mwe) and ACR 700 (Advanced CANDU Reactor 700 Mwe) advanced reactor designs.

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

4) 1:30 - 1:55
~~1:15~~ - ~~2:45~~ P.M. <u>Catawba and McGuire License Renewal Application</u> (Open) (MVB/GML/TJK/SD)
- 4.1) Remarks by the Subcommittee Chairman
- 4.2) Briefing by and discussions with representatives of the NRC staff and Duke Energy Company, as needed, regarding the license renewal application and draft Safety Evaluation Report for the Catawba Nuclear Station Units 1 and 2 and McGuire Nuclear Station Units 1 and 2.

5) 1:55-2:40 Report Regarding Recent Operating Events (GML/MWW)

~~5)~~ 6) 2:45 - ~~4:00~~ 5:10 P.M. Policy Issues Related to Advanced Reactor Licensing (Open) (TSK/MME)

5.1) Remarks by the Subcommittee Chairman
5.2) Briefing by and discussions with representatives of the NRC staff regarding changes to policy issues related to the licensing of advanced reactors resulting from the resolution of ACRS comments and recommendations included in its June 17, 2002 report.

~~4:00 - 4:15~~ 5:10 - 5:25 P.M. *****BREAK*****

~~6)~~ 7) ~~4:15 - 7:00~~ 5:25 - 6:15 P.M. Proposed ACRS Reports (Open)

Discussion of proposed ACRS reports on:

6.1) Confirmatory Research Program on High-Burnup Fuel (DAP/TSK/MME)
6.2) ~~Catawba and McGuire License Renewal Application (tentative) (MVB/GML/TJK/SD)~~

FRIDAY, OCTOBER 11, 2002, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

~~8)~~ 9) 8:35 - 10:00 A.M. Program Plan for Low-Power Shutdown (LPSD) Standardized Plant Analysis Risk (SPAR) Model Development and Cancellation of Revision 4i of SPAR Models (Open) (DAP/SD)

8.1) Remarks by the Subcommittee Chairman
8.2) Briefing by and discussions with representatives of the NRC staff regarding the program plan for LPSD SPAR Model Development, the reasons for canceling plans for the development of revision 4i of the SPAR Models, and insights from the onsite review of the LPSD SPAR Model for Surry Nuclear Plant Units 1 and 2.

10:00 - ~~10:15~~ 10:25 A.M. *****BREAK*****

~~9)~~ 10) ~~10:15 - 11:30~~ 10:25 - 11:35 A.M. Guidance for Performance-Based Regulation (Open)(GEA/HJL)

9.1) Remarks by the Subcommittee Chairman
9.2) Briefing by and discussions with representatives of the NRC staff regarding the draft NUREG/BR "Guidance for Performance-Based Regulation."

~~10)~~ 11) ~~11:30 - 11:45~~ 11:35 - 11:37 A.M. Reconciliation of ACRS Comments and Recommendations (Open) (GEA, et al./SD, et al.)

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

11:37 - 1:30
~~11:45~~ - ~~1:15~~ **P.M.** ***LUNCH***

1:30-3:00
~~11~~ (1,2) ~~1:15~~ - ~~2:15~~ P.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (GEA/JTL/SD)

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

11.2) Report of the Planning and Procedures Subcommittee on matters related to the conduct of ACRS business, and organizational and personnel matters relating to the ACRS.

3:00-3:15 BREAK

~~12) 2:15 - 3:15 P.M. Report Regarding Recent Operating Events (Open) (GML/MWW) Report by the cognizant ACRS member regarding recent operating events of interest.~~

13) 3:15 - ~~7:00~~ 7:15 P.M. Preparation of ACRS Reports (Open)
(4:00-4:15 P.M. BREAK) Discussion of proposed ACRS reports on:

3:35-5:50 13.1) Confirmatory Research Program on High-Burnup Fuel (DAP/TSK/MME)

~~13.2) Catawba and McGuire License Renewal Application (tentative) (MVB/GML/TJK/SD)~~

~~13.3) Program Plan for LPSD SPAR Model Development (DAP/SD)~~

5:55-6:45 13.4) Guidance for Performance-Based Regulation (GEA/HJL)

SATURDAY, OCTOBER 12, 2002, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

14) 8:30 - ~~1:00~~ 11:30 P.M. Preparation of ACRS Reports (Open
(10:30-10:45 A.M. BREAK) Continue discussion of the proposed ACRS reports listed under Item 13.

8:30-8:45 Guidance for Performance-Based Regulation - Final

15) ~~1:00 - 1:30~~ P.M. Miscellaneous (Open) (GEA/JTL)
8:45-11:30
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. The Committee will also discuss its plans for preparing a "white paper" on the use of PRA in the regulatory decisionmaking process.

NOTE:

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

- **Thirty-Five (35) copies of the presentation materials should be provided to the ACRS.**

APPENDIX III: MEETING ATTENDEES

496th ACRS MEETING
OCTOBER 10-12, 2002

<u>NRC STAFF</u> (October 10, 2002)

D. Terao, NRR
R. Tripathi, NRR
L. Fields, NRR
P. Sekerak, NRR
S. Newberry, RES
L. Burkhart, NRR
A. Cubbage, NRR
M. Rubin, RES
J. Lyons, NRR
M. Scott, NRR
R. Franovich, NRR
J. Fair, NRR
P. Balingin, NRR
J. Rivera-Ortiz, NRR
J. Ortega-Luciano, NRR
B. Rogers, NRR
A. Rivera-Varona, NRR
C. Grimes, NRR
J. Kramer, RES
S. Browde, RES
A. Rubin, RES
M. Razzazun, NRR
R. Meyer, RES
B. Pascarelli, NRR
J. Voglewede, RES
R. Caruso, NRR
F. Eltawila, RES
S. Basu, RES
A. Levin, OCM/RAM
G. Bagchi, NRR
A. Drozo, NRR
J. Sebrosky, NRR
G. Thron, NRR
J. Wilson, NRR
R. Lee, RES
A. Behbahani, RES
J. Kelly, RES
F. Odorr, RES
D. Carlson, RES
U. Shoop, NRR
J. Flack, RES
M. Mayfield, RES
J. Isom, NRR

<u>ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC</u>

W. Slagle, Westinghouse
V. Langman, AECL Technologies, Inc.
C. Reid, Bechtel
V. Snell, AECL
J. Weil, McGraw-Hill
J. Meyer, ISL
J. Mallay, Framatome ANP
J. Riccio, Green Peace
C. Ahini, French Embassy
G. George, PA Consulting Group
B. Gitnick, ISL, Inc.
A. Rao, GE
J. Lehner, BNL
A. Meyer, NEI

<u>NRC STAFF</u> (October 11, 2002)
P. Baranowksy, RES
S. Newberry, RES
M. Cheok, RES
V. Hodge, NRR
C. Grimes, NRR
G. DeMoss, RES
M. Shah, NMSS

<u>ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC</u>
B. Youngblood, ISL
C. Brinkman, Westinghouse

APPENDIX IV

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

October 22, 2002

SCHEDULE AND OUTLINE FOR DISCUSSION
497th ACRS MEETING
NOVEMBER 7-9, 2002

THURSDAY, NOVEMBER 7, 2002, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

2) 8:35 - 10:00 A.M. Proposed Resolution of Generic Safety Issue (GSI)-189, "Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident" (Open) (TSK/MWW/MRS)
   - 2.1) Remarks by the Subcommittee Chairman
   - 2.2) Briefing by and discussions with representatives of the NRC staff on the results of their additional analyses and proposed recommendations for resolving GSI-189.

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

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

3) 10:15 - 11:45 A.M. Early Site Permit Process (Open) (TSK/MME)
   - 3.1) Remarks by the Subcommittee Chairman
   - 3.2) Briefing by and discussions with representatives of the NRC staff regarding Early Site Permit Process.

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

4) 11:45 - 12:15 P.M. Peach Bottom License Renewal Application (Open) (GML/RRA/TJK)
   Report by the Subcommittee Chairman regarding the October 30, 2002 Plant License Renewal Subcommittee meeting on the license renewal application for the Peach Bottom Nuclear Plant, Units 2 and 3.

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

5) 1:15 - 3:15 P.M. Westinghouse AP1000 Design (Open) (TSK/MME/PAB)
   - 5.1) Remarks by the Subcommittee Chairman
   - 5.2) Briefing by and discussions with representatives of Westinghouse regarding the design features of and test information on, the AP1000 design. The NRC staff will provide a status report regarding its review schedule.

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

6) 3:30 - 5:00 P.M Risk-Informed Improvements to Standard Technical Specifications (Open) (SLR/MWW)
- 6.1) Remarks by the Subcommittee Chairman
- 6.2) Briefing by and discussions with representatives of the NRC staff regarding staff's progress on risk-informed improvements to Standard Technical Specifications and related matters.

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

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

7) 5:15 - 6:00 P.M. Report Regarding Recent Operating Events (Open) (GML/MWW) Report by the Cognizant ACRS member regarding recent operating events of interest.

8) 6:00 - 7:00 P.M. Preparation of ACRS Reports (Open) Discussion of proposed ACRS reports on:
- 8.1) Proposed Resolution of GSI-189 (TSK/MWW/MRS)
- 8.2) Early Site Permit Process (TSK/MME)
- 8.3) Risk-Informed Improvements to Standard Technical Specifications (SLR/MWW)

FRIDAY, NOVEMBER 8, 2002, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

10) 8:35 - 12:00 Noon Organizational and Personnel Matters (Closed) (GEA/JTL) The Committee will discuss organizational and personnel matters as well as the potential improvements to internal ACRS policies and procedures.

[NOTE: This session will be closed pursuant to 5 U.S.C. 552b(c) (2) and (6) to discuss organizational and personnel matters that relate solely to internal personnel rules and practices of ACRS, and information the release of which would constitute a clearly unwarranted invasion of personal privacy.]

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

11) 1:00 - 4:00 P.M. Safeguards and Security Activities (Closed) (GEA/RPS)
[THIS SESSION WILL BE HELD IN ROOM T-8E8]
- 11.1) Report by the Subcommittee Chairman regarding matters discussed at the October 31, 2002 meeting of the ACRS Subcommittee on Safeguards and Security.
- 11.2) Discussion of the content of a proposed report to the Commission on Safeguards and Security matters.

[**NOTE:** This session will be closed pursuant to 5 U.S.C. 552b(c)(1) to protect national security information.]

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

12) 4:15 - 5:00 P.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (GEA/JTL/SD)

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

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

13) 5:00 - 5:15 P.M. Reconciliation of ACRS Comments and Recommendations (Open) (GEA, et al./SD, et al.)
Discussion of the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.

5:15 - 5:30 P.M. ***BREAK***

14) 5:30 - 7:00 P.M. Preparation of ACRS Reports (Open)
Discussion of proposed ACRS reports on:

14.1) Proposed Resolution of GSI-189 (TSK/MWW/MRS)

14.2) Early Site Permit Process (TSK/MME)

14.3) Risk-Informed Improvements to Standard Technical Specifications (SLR/MWW)

SATURDAY, NOVEMBER 9, 2002, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

15) 8:30 - 10:00 A.M. Preparation of ACRS Reports (Open)
Continue discussion of the proposed ACRS reports listed under Item 14.

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

16) 10:15 - 12:15 P.M. Annual ACRS Report on the NRC Safety Research Program (Open) (FPF/RPS)

16.1) Remarks by the Subcommittee Chairman regarding matters discussed at the November 6, 2002 Subcommittee meeting.

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

12:15 - 12:30 P.M. ***BREAK***

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

NOTE:

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

APPENDIX V
LIST OF DOCUMENTS PROVIDED TO THE COMMITTEE
496TH ACRS MEETING
OCTOBER 10-12, 2002

[Note: Some documents listed below may have been provided or prepared for Committee use only. These documents must be reviewed prior to release to the public.]

MEETING HANDOUTS

AGENDA ITEM NO.	DOCUMENTS

1 Opening Remarks by the ACRS Chairman
1. Items of Interest, dated October 10-12, 2002

2 Confirmatory Research Program on High-Burnup Fuel
2. Update on Issues in 1998 Agency Program Plan for High-Burnup Fuel presentation by R. Meyer, RES [Viewgraphs]
3. EPRI Topical Report on Reactivity Initiated Accidents presentation by U.Shoop, NRR [Viewgraphs]

3 Overview of European Simplified Boiling Water Reactor (ESBWR), SWR 1000, (Boiling Water Reactor), Advanced CANDU Reactor (ACR 700) Pre-Application Review
4. New Reactor Licensing presentation by J. Lyons, NRR [Viewgraphs]
5. ESBWR Design and Technology Overview (1390 Mwe natural circulation with passive safety systems) presentation by A. S. Rao, GE [Viewgraphs]
6. SWR 1000 Design Overview presentation by R. Stroudt, Framatome [Viewgraphs]
7. Advanced CANDU Reactor (ACR) presentation by Dr. V. G. Snell, AECL Technologies, Inc. [Viewgraphs]

5 Policy Issues Related to Advanced Reactor Licensing
8. Technical Related Policy Issues for Future Non-Light Water Reactors presentation by T. King, RES [Viewgraphs]

8 Program Plan for Low-Power Shutdown (LPSD) Standardized Plant Analysis Risk (SPAR) Model Development and Cancellation of Revision 4i of SPAR Models
9. Low Power/Shutdown SPAR Model Development presentation by RES [Viewgraphs]

9 Guidance for Performance-Based Regulation
10. Guidance for Performance-Based Regulation presentation by RES

[Viewgraphs]

10 <u>Reconciliation of ACRS Comments and Recommendations</u>

11 . Reconciliation of ACRS Comments and Recommendations [Handout #10.1]

11 <u>Report of the Planning and Procedures Subcommittee/Future ACRS Activities</u>

12. Final Draft Minutes of Planning and Procedures and Future ACRS Activities Subcommittee Meeting - October 9, 2002 [Handout # 11.1]
13. Tour of GE and Framatome Facilities, September 23-26, 2002

MEETING NOTEBOOK CONTENTS

TAB	DOCUMENTS
2	Confirmatory Research Program on High Burnup Fuel
	1. Table of Contents
	2. Proposed Schedule
	3. Status Report, dated
	4. Agency Program Plan for High Burnup Fuel dated July 6, 1998
	5. Memorandum from S. Collins to A. Thadani dated January 31, 2002
	6. ACRS letter dated March 14, 2002
	7. EDO Response dated June 11, 2002
3	Overview of European Simplified Boiling Water Reactor (ESBWR), SWR 1000, and ACR-700 Reactor Designs
	8. Table of Contents
	9. Proposed Schedule
	10. Project Status Report
	11. April 18, 2002 letter from GE Nuclear Energy to Samuel Collins and material describing features of the ESBWR design
	12. May 31, 2002 letter from Framatome ANP and material describing features of the SWR-1000 design
	13. Material describing features of the ACR-700 design
4	McGuire Nuclear Station Units 1 and 2, Catawba Nuclear Station Units 1 and 2, License Renewal Application
	14. Table of Contents
	15. Status Report
	16. Duke response to a staff request for additional information regarding reactor vessel neutron embrittlement
5	Policy Issues Related to Advanced Reactor Licensing
	17. Table of Contents
	18. Proposed Schedule
	19. Status Report
	20. Draft Commission Paper
	21. ACRS Report dated June 17, 2002
	22. EDO's response dated July 23, 2002
	23. SECY-02-0139 dated July 22, 2002
8	Program Plant for Low-Power and Shutdown (LPSD) Operations SPAR Model

Development and Cancellation of Proposed Revision 4i of the SPAR Models

24. Proposed Schedule
25. Status Report
26. Note dated July 17, 2002, to Dana Powers, ACRS, from Pat Baranowsky, RES, transmitting the Program Plan for Low-Power and Shutdown SPAR Model Development
27. RES response to Commissioner McGaffigan's question regarding cancellation of Revision 4i of SPAR Models
28. Summary of Comments and Insights from Onsite Review of Low-Power and Shutdown SPAR Model for Surry 1 and 2
29. Excerpt from INEEL Report on Low-Power and Shutdown Operation Standardized Plant Analysis Risk Model Template for PWRs, August 2002 [Sensitive Homeland Security Information not for Public Disclosure]

9 Guidance on Performance-Based Regulation

30. Table of Contents
31. Proposed Schedule
32. Status Report

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
496th FULL COMMITTEE MEETING

OCTOBER 10-12, 2002

OCTOBER 10, 2002
Today's Date

NRC STAFF PLEASE SIGN IN FOR ACRS MEETING

PLEASE PRINT

NAME	NRC ORGANIZATION
David Terao	NRC/NRR/DE
Raji Tripathi	NRC/RES/DSARE
Leslie Fields	NRR/ADIP/NRLPO
PATRICK SEKERAK	NRR/DE/EMEB
Scott Newberry	NRC RES DRAA.
LARRY BURKHART	NRC/NRR/NRLPO
Amy Cubbage	NRC/NRR/NRLPO
MICHAEL B. RUBIN	RES/DSARE/SMSAB
Jim Lyons	NRR/NRLPO
Michael Scott	NRC/NRR/NRLPO
Rani Franovich	NRC/DRIP/RLEP
John Fair	NRC/DE/EMEB
Peter Balmain	NRC/DIPM
Joel E. Rivera-Ortiz	NRR/DRIP
Jonathan A. Ortega-Luciano	NRR/DLPM/LPD2
Bill Rogers	NRR/DIPM/IEHB
AIDA RIVERA-VARONA	NRR/DIPM/IEHB
Chris Grimes	NRR/DRIP/RPRP
Jack Kramer	RES/DSARE/REAHFB
Shana Browde	RES/DSARE/REAHFB
ALAN RUBIN	RES/DRAA

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
496th FULL COMMITTEE MEETING

OCTOBER 10-12, 2002

OCTOBER 10, 2002
Today's Date

NRC STAFF PLEASE SIGN IN FOR ACRS MEETING

PLEASE PRINT

NAME	NRC ORGANIZATION
Ralph Meyer	RES
Bob Pascarelli	NRR/NRLPO
[redacted]	
JOHN VOGLEWEDE	RES/DSARE
RALPH CARUSO	NRR/SRXB
FAROUK ELTAWILA	RES/DSARE
SUD BASU	RES/DSARE
ALAN LEVIN	OCM/RAM
Goutam Bagchi	NRR/DE
ANDRE DROZD	NRR/DSSA
Joel Sebrosky	NRR/NRLPO
Edward D Throm	NRR/DSSA
JERRY Wilson	NRR/NRLPO
RICHARD LEE	RES/DSARE/SMSAB
A. Behbahani	RES/DSARE/SMSAB
JOE KELLY	RES/DSARE/SMSAB
Frank Odar	RES/DSARE/SMSAB
Donald E. Carlson	RES/DSARE/REAHFB
UNDINE SHOOP	NRR/SRXB
JOHN FLACK	RES/DSARE/REAHFB

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
496th FULL COMMITTEE MEETING

OCTOBER 10-12, 2002

OCTOBER ~~12,~~ 10 2002
Today's Date

NRC STAFF PLEASE SIGN IN FOR ACRS MEETING

PLEASE PRINT

NAME	NRC ORGANIZATION
Michael Mayfield	RES/DET
James A. Isom	NRR

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
496th FULL COMMITTEE MEETING

OCTOBER 10-12, 2002

OCTOBER 10, 2002
Today's Date

ATTENDEES PLEASE SIGN IN FOR ACRS MEETING

PLEASE PRINT

NAME	AFFILIATION
William H. Slagle	Westinghouse
Vince Langman	AECL Technologies Inc.
CAL REID	BECHTEL
V. G. SNELL	AECL
Jenny Weil	McGraw-Hill
Jim Meyer	ISL
JAMES F. MALLAY	FRAMATOME ANP
JIM RICCIO	GREENPEACE
CHARPENTIER Alexis	French Embassy
Muhammad Razzaque	SRXB/NRR
Glenn R. George	PA Consulting Group
BEN GITNICK	ISL, INC.
Atam Rao	GE
~~Joel F.~~	
John Lehner	BNL
~~[illegible]~~	
Adrian Heymer	NEI

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
496th FULL COMMITTEE MEETING

OCTOBER 10-12, 2002

OCTOBER 11, 2002
Today's Date

NRC STAFF PLEASE SIGN IN FOR ACRS MEETING

PLEASE PRINT

NAME	NRC ORGANIZATION
Patrick Baranowsky	RES/DRAA/OERAB
Scott Newberry	RES DRAA.
Mike Cheok	RES/DRAA/OERAB
Vern Hodge	NRR/DRIP/RORP/OES
Chris Grimes	NRR/DRIP/RPRP
Gary DeMoss	RES/DRAA/OERAB
Mahendra J. Shah	NMSS/SFPO/TRD/TRB

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
496th FULL COMMITTEE MEETING

OCTOBER 10-12, 2002

OCTOBER 11, 2002
Today's Date

ATTENDEES PLEASE SIGN IN FOR ACRS MEETING

PLEASE PRINT

NAME	AFFILIATION
Bob Youngblood	ISL
Charles Brinkman	Westinghouse

ITEMS OF INTEREST
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
496TH MEETING
OCTOBER 10-12, 2002

Page

SPEECHES

REGULATORY ACTIVITIES

OFFICE OF PUBLIC AFFAIRS

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Washington, DC 20555-001 **E-mail: opa@nrc.gov**
Web Site: Public Affairs Web Site

No. S-02-024

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ONE YEAR AFTER - REFLECTIONS ON NUCLEAR SECURITY

Dr. Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission

at the
INFOCAST Conference
"The Nuclear Renaissance: Maximizing the Value of Nuclear Assets"

Washington, DC
September 11, 2002

Introduction

We meet today on the anniversary of a singular moment in our Nation's recent history. I am sure that each of us has personal memories of the horrifying September terrorist attacks. An image that remains clear in my mind today is the view from the 17th floor of NRC Headquarters of the smoke following the attack on the Pentagon. To say that the world was changed forever by these events is both a cliche and an understatement. For those who lost family members, friends, and colleagues on that day, the world will surely never be the same again. For those who survived the attacks or were near enough to be witnesses, the images will be forever seared in memory. For the rest of us, merely seeing the video images again during this period of remembrance brings back all the emotions we experienced on that terrible day.

Given the riveting effect that the attacks have had on the Nation, it is appropriate on this anniversary to spend a few moments to reflect on the implications of those horrendous events.

In the aftermath of the attacks, the security surrounding the Nation's critical infrastructure, including the Nation's commercial nuclear power plants, has become a central concern. I have been particularly gratified to have played a part in the collaborative work that has occurred among the Federal government, State and local officials, and licensees and industry working groups to ensure that the already robust protection of these facilities was further enhanced. All levels of government, as well as NRC licensees, have made significant contributions to bolster defenses against the increased threat of a terrorist attack. I would like to express my heartfelt appreciation to them.

The events of September 11 have also had a broader impact. We all look at the world in a different way. Society is wary of potential terrorist activities and is concerned about the facilities that they might be interested in attacking. This heightened alertness is manifested in the increased reporting by members of the public of possible suspicious activities in the vicinity of nuclear plants. It is also clear that the threat of terrorism will be an abiding issue for the long term. As a result, there is a demand for action by government to preserve the security of its citizens.

The NRC has fully accepted that responsibility. But this is not a task that can be completed overnight. Although the NRC has taken many significant actions, some major challenges remain. Let me take a moment to reflect upon the state of security at NRC-licensed facilities and then to focus on the future.

There are three fundamental points that I would like to emphasize at the outset. First, the physical protection at nuclear power plants was strong before September 11. I am aware of no other industry that has had to satisfy the tough security requirements that the NRC has had in place for a quarter of a century. And these requirements have been significantly augmented over the past year. The plants are surrounded by multiple fences with continuously monitored perimeter detection and surveillance systems. They are guarded by well trained and well armed security forces. Nuclear power plants are constructed to withstand hurricanes, tornadoes, and earthquakes, making them among the most formidable structures in existence. The plants also benefit from redundant and diverse safety equipment so that if any active component becomes unavailable, another component or system will satisfy its function. Operators are trained to respond to unusual events, and carefully designed emergency plans are in place. In short, the security at power plants is very strong and the plants have an inherent capacity to withstand severe events of all types, including those that might be initiated by terrorists.

Second, there have been no specific credible threats of a terrorist attack on nuclear power plants since September 11. The NRC has worked closely with intelligence and law enforcement personnel to assess the threats that may be directed at nuclear facilities. Although it is difficult to predict when and where terrorists may strike next, the robust security at nuclear plants should serve as a significant deterrent. Nonetheless, it is prudent to presume that al Qaeda may consider nuclear facilities as potential targets. As a result, NRC has put in place a five-level threat advisory and protective measures system that requires licensees to take specific actions in response to changes in the threat conditions.

Third, in light of the events of September 11, the NRC has recognized the need to reexamine past security strategies to ensure that we have the right protections in place for the long term. Shortly after the attacks, we began a comprehensive review of our requirements for physical protection and security. We are undertaking a reexamination of the assumptions that underlie the current regulatory framework and we are making any changes that are necessary. We have already taken actions as a result of this review, and more will be taken in the coming months.

NRC's Response following the September 11 Terrorist Attacks

- Following the attacks, the NRC issued over 30 safeguards and threat advisories to the major licensed facilities, placing them on the highest security level. Security across the nuclear industry was enhanced as a result of these actions, and many of the strengthened security measures are now requirements as a result of subsequently issued NRC Orders. The security enhancements include increased security patrols, augmented security forces, additional security posts, increased vehicle standoff distances, and enhanced coordination with the law enforcement and intelligence communities.

- The Commission has also enhanced access control at nuclear power plants. This may be one of the most effective means of preventing a successful attack, because an insider could provide significant assistance to an attacking force. NRC regulations require that individuals having unescorted access to nuclear power plants undergo a background investigation which includes credit checks, employment history, reference examination, psychological testing, and a criminal history check conducted by the FBI. Further restrictions include prohibitions on the use of temporary unescorted access in sensitive areas.

- Improvements in communications have been a central feature of our activities. Not only have we had frequent interactions with licensees concerning the security of their facilities, but also we have improved linkages with other parts of government. For example, we are in close and continuous contact with the intelligence and law enforcement communities and we have advised licensees to enhance protocols for involving governmental entities in the defense of their facilities.

- The Commission has also completed an initial assessment of power reactor vulnerabilities to the intentional malevolent use of commercial aircraft in suicidal attacks and has initiated a broad-ranging research program to understand the vulnerabilities of various classes of facilities to a wide spectrum of attacks. We are developing measures to mitigate vulnerabilities that are identified. Although our work in this area is ongoing, the Commission has directed nuclear power plant licensees to develop specific plans and strategies to respond to an event that could result in damage to large areas of their plants from impacts, explosions, or fire. In addition, licensees must provide assurance that their emergency planning resources are sufficient to respond to such an event.

- The Commission is working closely with other Federal agencies to revise the design basis threat that provides the foundation for the security programs of power plant licensees. Significant changes are likely. The Commission's Orders effectively provide enhanced security in the interim while this work in underway.
- Inspection of security capability is necessary to provide confidence in the adequacy of defensive measures. The Commission has decided that full security performance reviews, including force-on-force exercises, will be carried out in the future at each nuclear power plant on a three-year cycle, instead of the eight-year cycle that had been applied in the past. These reviews have commenced with table-top exercises that for the first time involve a wide array of Federal, State, and local law enforcement and emergency planning officials.
- The NRC has developed a new Threat Advisory and Protective Measures System in response to Homeland Security Presidential Directive-3. When a new threat condition is declared, the NRC will promptly notify affected licensees of the condition and refer them to the predefined protective measures that we have developed for each threat level. The new system has been formally communicated to licensees, Governors, State Homeland Security Advisors, Federal agency administrators, and other appropriate officials. We had the opportunity to exercise this system yesterday afternoon when the Attorney General announced that the threat condition had moved to the Orange (high) level.
- The Commission is actively involved in efforts to defend against possible terrorist use of radiological dispersal devices. Following the terrorist attacks of last September, NRC alerted licensees, suppliers, and shippers of the need to enhance security against the threat of theft of radioactive material. The NRC is conducting a comprehensive evaluation of controls to protect those radioactive materials that constitute the greatest hazard to public health and safety. For example, we are evaluating approaches for "cradle-to-grave" control of radioactive sources which might be used in a radiological dispersal device and are reexamining the import and export licensing for these isotopes. We are also working with the Office of Homeland Security and other agencies to ensure that the Federal Government is prepared to respond to an event involving a radiological dispersal device.
- In April, we established the Office of Nuclear Security and Incident Response (NSIR) to improve communications and coordination both within and external to the NRC on security and safeguards issues. This office is responsible for developing overall safeguards and security policies and is the central point of contact with the Office of Homeland Security. It contains our Incident Response organization; coordinates with Federal response and law enforcement agencies; and directs our counter-intelligence, information security, and secure communications activities.
- In short, the NRC has taken a wide variety of steps over the past year in response to the changing environment in which we find ourselves.

Looking to the Future

Nonetheless, there are issues that remain before us and the Nation. Let me mention a few:

- First, there are limits to the defensive capabilities that should be expected of nuclear plant operators. For example, the defense against aircraft attacks should certainly be the responsibility of governmental authorities, as should the defense against attackers with significant military capabilities. As a result, there must be an allocation of responsibility between the licensee's security organization and the government. Establishing the boundary that defines the responsibilities that should be borne by the private sector and those that should be assumed by the government has proven difficult for all types of civilian infrastructure. There is no quick answer that can be developed by the NRC in isolation from the other parts of government.
- Let me note in this connection that, given the current threat environment, an abundance of

governmental response forces -- local, State, and Federal -- would be dispatched to engage any attackers at a nuclear facility and to lend assistance, regardless of the scope and nature of the attack. The real issue is not whether governmental entities will provide assistance, but rather when such resources will arrive and how they will be used to defend the facility. This has practical implications because the security framework should reflect the joint security capability of both the licensee and the government.

- Second, there needs to be an integrated national strategy to protect critical infrastructure of all types. The defense of nuclear facilities should not be viewed in isolation, but should be part of an overall national defensive scheme. The effort to develop such a strategy is underway. In some respects the nuclear industry is the pathfinder because of the extensive security capabilities that it had in place before September 11. Establishing and implementing an integrated national strategy will be an important task for the new Department of Homeland Security.

- Third, we need to ensure coordination with Agreement and non-Agreement States in implementing security measures for radioactive materials. Agreement States have responsibility for roughly three quarters of the radioactive sources in the United States. Thus, any action the NRC might take to prevent a terrorist from using a radiological dispersal device will impact Agreement State licensees. In enhancing the security of nuclear materials, we must preserve NRC's ultimate responsibility for protection of common defense and security, while maintaining the integrity of the Agreement State programs. Moreover, States must be heavily involved with securing hazardous unlicensed sources and in establishing holding or disposal areas for materials.

- Fourth, there is a difficult challenge in maintaining the appropriate public access to information. The NRC has strived to ensure public confidence by being one of the most open agencies in the U.S. government; we recognize the reality that suspicions are nurtured if our activities are not fully accessible to the concerned public. But some information must be withheld because it could help a terrorist. We thus have the dilemma of trying to balance the public's right to know against the need for secrecy in certain areas.

- Fifth, we must confront the reality that the concern for nuclear matters arises from an abiding public fear that devastating consequences will necessarily result from an attack on a nuclear power plant or from the detonation of a radiological dispersal device. These fears are certainly greatly exaggerated. But putting nuclear events in context has proven extraordinarily difficult because of ingrained public attitudes. This may have the unfortunate consequence that too little attention is provided to the defense of other types of infrastructure for which the consequences of a successful terrorist attack could be far greater.

- Finally, although security must be an abiding concern, we cannot allow it to displace or to diminish the obligation to protect public health and safety from accidents. This has been a particular challenge in the United States because, for reasons wholly apart from security, we are in a period of dramatic change. Our nuclear plant licensees continue to seek to extend their operating licenses beyond the original 40-year term and to increase the power output of their facilities. There continues to be interest in the possibility of new construction. And after decades of technical studies and political debate, we confront the need for decisions associated with the establishment of a possible national disposal facility for spent fuel and high-level waste. September 11 has added another important task at a time of intense activity in the nuclear arena.

In conclusion, let me note again that our nuclear facilities are the strongest and most well protected civilian facilities in our country. But we recognize the need to enhance those protections. The NRC is dedicated to meeting the obligation to protect the public health and safety and the common defense and security from threats of all kinds. We have accomplished much over the last year, but we have more to do and we are on track to do it.

Thank you.

OFFICE OF PUBLIC AFFAIRS

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REMARKS TO STAFF

by

Dr. Richard A. Meserve
Chairman, U.S. Nuclear Regulatory Commission

September 11, 2002
8:40 a.m.

Good Morning. This is Chairman Meserve.

At 8:45 a.m. on this day, one year ago, the first of four hijacked aircraft crashed into the North Tower of the World Trade Center, turning an ordinary day into a cataclysm. Many of us at the NRC were glued to television monitors in an effort to understand the significance of the crash when the second hijacked aircraft struck the South Tower at 9:03 a.m. Many of us were stunned by the terrible images. Minutes later, at 9:38 a.m., we could see the smoke rising from the Pentagon crash from the upper floors of the NRC Headquarters building. Subsequently, at about 10:03 a.m., the last of the four hijacked aircraft crashed in an empty field in rural Pennsylvania, the result of the determined efforts of the passengers to prevent terror from succeeding a fourth time. In the space of one hour and 18 minutes, our vulnerability was fully exposed.

To say that the world was changed forever by these events is both a cliche and an understatement. For those who lost family members, friends, and colleagues on that day, the world will surely never be the same again. For those who survived the attacks or were near enough to be witnesses, the images will be forever seared in memory. For the rest of us, merely seeing the video images again during this period of remembrance brings back all the emotions we experienced on that terrible day.

But the events of September 11 triggered actions as well as memories. They set in motion a coordinated Federal response designed to protect our Nation from such attacks. You have played a critical role in these activities. The NRC has taken measures to enhance security at NRC-licensed facilities; undertaken a comprehensive review of our safeguards and security programs; established a continuing dialogue with other Federal agencies, including the Office of Homeland Security; taken steps to cope with the threat of a radiological dispersal devices; established the Office of Nuclear Security and Incident Response; developed a new Threat Advisory and Protective Measures System; and improved security at our own buildings, among other actions. My fellow Commissioners and I take great pride in the part the staff has taken in the government-wide response to the terrorist attacks. We are grateful for the support the NRC staff has provided.

We have more that we need to accomplish together, and I am confident that we will be successful. As our efforts proceed, I can think of no greater honor we can pay to those who gave their lives on September 11, 2001, than to dedicate our continuing activities to their memory. I ask you, therefore, to join me now in a moment of silence to honor the victims of the terrorist attacks of one year ago and to renew our commitment to the work that lies ahead.

(Silence: one minute)

Again, thank you for your support.

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A YEAR LATER: MAINTAINING FOCUS

by

The Honorable Greta Joy Dicus
Commissioner
U.S. Nuclear Regulatory Commission

Nuclear Non-Operating Owners Group
Fall 2002 Conference

Williamsburg, Virginia
September 18, 2002

INTRODUCTION

Good afternoon everyone.

I am delighted to be here, with you, today.

Let me begin by thanking and congratulating the organizers and sponsors of this conference. I think you have done a wonderful job providing an informative and interesting agenda. This is the first time I have had the pleasure of meeting many of you and I am grateful for the opportunity.

I see that some of you had the opportunity to tour Newport News Shipbuilding Facility yesterday. It is an impressive complex. My technical assistant spent 13 long months working at Newport News shipyard during the overhaul of the USS Nimitz and informs me that, as impressive as it may be, about 8 hours is enough time.

In preparing for this discussion, I learned that approximately 20% of the US nuclear operating capacity is owned by non-operating owners and that over 46% of US nuclear plants have shared ownership capacity. We typically do not delve into the business side, but I was impressed by the important role that you can play in helping to ensure a safe and appropriate national energy mix. Thank you.

About a year ago, I delivered a speech to a structural mechanics international meeting in Washington. I quoted Yogi Berra throughout my talk and used one of Yogi's quotes as the title for my speech: "The Future Ain't What It Used To Be." The theme of my discussion that day related to NRC's improvements and challenges and the role that NRC would likely be playing in an almost certain nuclear power resurgence. That was before September 11^{th}. That was before the emergent safety issue at the Davis Besse nuclear plant.

A year later, the future still ain't what it used to be - - but it never is. In my view, the pace of any nuclear power resurgence has been slowed, to allow us to reflect on these events and seek further improvements. I still believe the future of nuclear power is bright - - as long as we continue to learn and focus on safe and secure operations.

THE IMPACT OF September 11TH

The challenges in a post-September 11th environment for the NRC and the nuclear industry have been daunting. The nuclear industry appears to be the relentless focus of political and public scrutiny. When one reflects on this, it is completely understandable, although perhaps not completely justified. But one thing is a certainty, the focus will always be there.

Nuclear power plants are among the most hardened potential targets of terrorist attacks. As you know, each nuclear plant has a well-trained and well-armed security force, a robust security plan, and design features which would make a successful terrorist attack unlikely. Nonetheless, in the light of September 11th, the NRC and the industry realized that vulnerabilities need be further reduced.

As a result, individual plant operators have taken actions which they deemed prudent and the NRC has imposed, by Order, additional requirements to further improve the security of these facilities. The specific actions are sensitive, but generally include requirements for increased patrols, augmentation of the number and capabilities of security guards, additional security posts, installation of additional barriers, enhanced coordination with law enforcement and military authorities, and restrictive site access controls for personnel.

The NRC has also re-organized to better meet the needs in a post-September 11th world. We recently established a new Office of Nuclear Security and Incident Response to provide a single focused organization for security, safeguards, and emergency response. The new Office also provides a central interface between the NRC and the Office of Homeland Security, other Executive Branch agencies, and Congress.

Just last month, NRC implemented a new Threat Advisory and Protective Measures System. The system corresponds to the color-coded national Homeland Security Advisory System and provides the NRC with the flexibility to advise protective measures for each threat level. Our first experience with the new system came just last week when the national threat level was raised from "Yellow" to "Orange."

One of the next, important steps, in considering appropriate NRC and industry actions is to consider revisions to the Design Basis Threat - - the threat to which nuclear power plants are required to defend against. The staff is currently working with other government agencies, the intelligence community and the industry to consider appropriate revisions to the Design Basis Threat for commercial power reactors. It is certainly possible that, in order to defend against a revised Design Basis Threat, additional security enhancements would be required. Revising the Design Basis Threat is an ongoing process, with the next interation scheduled to be completed in the near future.

REACTOR OVERSIGHT PROCESS

I will not dwell on the seriousness of the reactor head degradation that was discovered at Davis Besse. This is an incredibly important issue that has caused NRC to look inward and outward into the process and events that lead up to this discovery. On a recent agenda, I noted two presentations related to Davis Besse. So I suspect you will be fully briefed on the technical details.

If the event at Davis Besse tells me anything, it affirms for me that defense-in-depth must always remain the foundation of safety. The event at Davis Besse also confirms for me that our regulatory process is sufficiently robust to handle emergent safety issues quickly and effectively. But, we must learn from Davis Besse and continue to look at ways to make our processes better.

Our revised reactor oversight process has dramatically improved our oversight of commercial nuclear power plants. As one of our Regional Administrators put it: the revised reactor oversight process is "relentless". The process helps focus resources on those areas that are most important to safety and then keeps the pressure on to ensure that there is demonstrated and sustained improvement in deficient areas. Recently, Indian Point Unit 2 moved down into the "Degraded Cornerstone" column of the Action Matrix. Oconee Unit 1 and Cooper Nuclear Station has moved up into the "Multiple or Repetitive Degraded Cornerstone" column of the Action Matrix - - requiring a higher level of Agency oversight.

THE FUTURE OF NUCLEAR POWER

The future of nuclear power depends on maintaining safety. We must never compromise safety as we continue to demonstrate creativity, openness, resolve and resilience in meeting each and every new challenge. The NRC and the industry will play a key role. The NRC's role is to provide stable and predicable processes, provide independent and vigorous oversight, and thus ensure that the public remains confident that we are a strong and effective regulator. The industry' role is to operate safely by setting and maintaining high standards, even above those required by regulation.

Chairman Meserve has indicated that viability of the nuclear option is absolutely dependent on the maintenance of safe operations, the NRC's -- and the industry's -- highest priority must be the protection of public health and safety. If we fail in ensuring safety, the emerging optimism about nuclear energy will quickly disappear. I agree.

Licensing of a new plant, whether under 10 CFR Part 50 or Part 52, will be a significant challenge to the NRC. While we currently do not anticipate a return to the feverish pace of licensing for new plants that occurred in the mid-1970's, we are taking prudent steps to ensure that NRC is prepared to meet a potential new plant licensing submittal.

Both the Office of Nuclear Reactor Regulation and the Office of Regulatory Research have re-organized to support increased interaction with the industry and stakeholders, establish a new plant licensing infrastructure, support timely identification and resolution of technical and policy issues, and prepare for an effective transfer of technology.

Recently, with the renewed interest in future plant licensing, the staff began the AP-1000 design certification review and has interacted with Exelon and the Department of Energy (DOE) to identify key issues related to the pebble bed modular reactor (PBMR) and an approach for their resolution. In addition, General Atomics (GA) has expressed interest in conducting pre-application activities on their gas turbine modular helium reactor (GT-MHR), a 600 Mwt high-temperature gas-cooled reactor (HTGR), and DOE is considering licensing issues in their Generation IV reactor development program. DOE's 2010 initiative foresees a possible application for a combined license as early as 2005.

The US industry support for pebble-bed technology has stepped back some in the recent months. Changes in industry leadership and some difficult technical issues are resulting in a slower than anticipated pace of activity on the pebble-bed. But, overall the pace of interest in future reactors designs has increased. The staff has recently met with representatives from Atomic Energy of Canada, Limited to discuss pre-application review activities for the ACR-700 design and also met with representatives from Framatome to discuss the SWR-1000 design.

The companion element of building new nuclear power plants is the siting process - - finding a place. Much effort is underway to "exercise" our early site permit process, work out some of the issues, and within the next few years, possibly have an approved site for construction of a new reactor.

The sites that are the primary focus of these reviews are existing reactor sites that can accommodate an additional facility. Dominion's North Anna site, Entergy's Grand Gulf site and as-yet-unspecified Exelon site are in the mix of possible early site permit review candidates.

Of course, associated with some of the newer designs will likely be a host of technical and policy challenges. Some of these challenges include high-temperature materials performance, qualification of accident analysis codes and methods, qualification of coated particle fuel, and the need for "containment or confinement". To meet these challenges, we must continue to have a strong nuclear research program. I am, and I believe that the Commission is, committed to strengthening our research program.

LICENSE RENEWAL

A year ago, I characterized our experience with the license renewal process as - "our initial experience."

Today, I think we are experienced veterans. Our process remains stable and efficient. We have completed the reviews for Calvert Cliffs Units 1 and 2; Oconee Units 1, 2 and 3; and Arkansas Nuclear One Unit 1; Hatch Units 1 and 2; Turkey Point Units 3 and 4. License renewal reviews for fourteen other units are underway.

The license renewal reviews completed to date have emphasized safety and been completed ahead of schedule. We believe that this is a noteworthy accomplishment and recognize that potential challenges lie ahead with the simultaneous review of many renewal applications. We continue to work to improve the effectiveness and efficiency of our license renewal process.

A year ago, circumferential cracking around control rod drive penetrations found at Oconee and Arkansas Nuclear One nuclear plants was prominent. This year, the reactor head degradation found at Davis Besse is at the forefront of technical and regulatory issues. These examples should serve to remind us that age-related degradation is an issue that can affect all operating reactors. It should also help emphasize the importance and strength of our current processes to deal effectively with emergent safety concerns. Ongoing efforts to further our understanding of age-related degradation are important and we should continue to vigorously explore new techniques that help us better detect, characterize, and assess the impact of these degradations. Analytical tools for assessing the risk significance of degradation help ensure the actions we take are appropriate, coherent, and timely.

HUMAN CAPITAL INITIATIVES

Whether there is resurgence of nuclear power or not, the changing nuclear workforce provides enormous management challenges that must be addressed today. The current inflow of new talent does not equal the outflow of experienced workers. Even when we are able to attract talented young men and women, the lack of upward mobility or lack of variety in career paths may result in segments of the workforce moving outside the nuclear area. Maintaining and cultivating core competencies in nuclear-related areas is a key concern for the industry and the NRC.

Two years ago, at the NRC, the ratio of NRC employees who are over 60 years of age to those under 30 was between 5 and 6 to1. The same ratio at NASA, for comparison, was approximately 2:1. Moreover, approximately fifteen percent of NRC's engineers are already eligible for retirement and another four percent of the current workforce of engineers will become eligible for retirement each year for the next few years.

Today, a focus on entry-level hiring and our two-year Nuclear Safety Intern Program have served to shift the age ratio of the workforce from 6:1 to 2:1, with a total of 121 intern program participants. Twenty-three percent of the employees in the Office of Nuclear Regulatory Research and 21 percent of the employees in the Office of Nuclear Reactor Regulation are eligible for retirement today. While the number of employees eligible to retire remains proportionally high, the percent of those employees who decide to retire is down slightly. In 2001, actual retirements at NRC were 15 percent of those eligible.

In addition, we appear to be stemming the adverse trend of engineering capability loss by exercising a number of human capital strategies to recruit, hire, develop, and retain a talented, diverse workforce. NRC uses targeted recruitment, "signing bonuses" for applicants with critical skills, student loan repayment benefits, fellowships, technical training, and leadership development programs. The downturn in other segments of the economy and the excitement about the future of nuclear power appears to contribute to an improved outlook at NRC. But, the human capital crisis is not over. Demand still outnumbers supply.

Should the resurgence of new nuclear power plant flourish, I think the Agency will be faced with at least two competing forces that will affect NRC resources. One force will be good for the agency and would involve establishing new positions, reviewing cutting-edge technology, and increasing upward mobility. The other force would be from outside the agency resulting from government and industry competing, under different rules, for the same resources.

It is clear that both the NRC and the industry must be pro-active and aggressive in seeking out talent early, training them and planning smartly for what the future may bring. We need to be able to respond to emerging technology, deal with emerging issues, and deal effectively in the international environment. Our credibility as an effective, competent regulator and the industry's credibility as effective and competent operators hinges on maintaining a strong technical expertise.

YUCCA MOUNTAIN

A year ago, the Department of Energy had not made its recommendation regarding the location for a high-level waste repository. Today, Yucca Mountain has been approved by the President, withstood the Governor of the State of Nevada's "veto" and is the designated site for disposal of the nation's spent fuel and high-level waste from civilian reactors. DOE has indicated that it intends to submit an application to NRC to construct the Yucca Mountain facility in December of 2004. The law then gives NRC up to four years to decide whether to grant the license, including the completion of the administrative proceeding.

As the Chairman has stated, it is not an exaggeration to say that no single NRC decision or set of decisions since the response to Three Mile Island is likely to be scrutinized as closely, from a technical, legal, and public confidence standpoint, as those concerning this one-of-a-kind facility at Yucca Mountain.

The NRC has for several years been making preparations for the eventuality of an application for a high-level waste repository. Although our regulations that will govern the review of the high-level waste repository are risk-informed and performance-based, major challenges exist in demonstrating compliance with the requirements.

The system contains both natural and engineered barriers and the system of barriers must function effectively for 10,000 years -- longer than recorded human history. As you can understand, this is unlike any licensing proceeding the agency has faced in the past. Probably the most complex aspects of the review will be the post-closure period of performance, because it involves estimations of repository performance over thousands of years.

CONCLUSION

In summary, a year ago, our focus shifted to security of nuclear facilities and materials. Major changes occurred at the NRC and within the nuclear industry and some ongoing initiatives slowed to support the surge in effort toward security. More security-related changes will likely be necessary and our focus remains high in the security area - - as it should.

However, our safety focus never changed. It cannot. A successful terrorist attack or a reactor accident carry similar devastating effects on public confidence and potential public health and safety issues. In the aftermath of September 11th, we continued to move forward to improve our regulatory processes and focus resources on safety.

The trade-off between safety and security is not a zero-sum game. We cannot rob Peter to pay Paul.

Again, thank you and I would be pleased to answer any questions that you may have.

PNO-IV-02-052A - Entergy Operations, Inc.

October 4, 2002

PRELIMINARY NOTIFICATION OF EVENT OR UNUSUAL OCCURRENCE -- PNO-IV-02-052A

This preliminary notification constitutes EARLY notice of events of POSSIBLE safety or public interest significance. The information is as initially received without verification or evaluation, and is basically all that is known by the Headquarters staff on this date.

Facility	__ **Licensee Emergency Classification**
Entergy Operations, Inc.	__ Notification of Unusual Event
Waterford 3 and River Bend Station	__ Alert
River Bend Station, St. Francisville, Louisiana	__ Site Area Emergency
Docket: 50-382 (W3), 50-458 (RBS), 50-416 (GG)	__ General Emergency
License Nos.: NPF-38 (W3), NPF-47 (RBS), NPF-29 (GG)	X Not Applicable

Subject: ***Update of Response to Hurricane/severe Weather on Gulf Coast***

Description:

At 10 p.m. on October 3, 2002, Region IV deactivated its Incident Response Center and returned to a normal oversight posture. The Incident Response Center had been activated at 4 p.m. on October 2, 2002, to track Hurricane Lili.

Hurricane Lili made landfall near New Iberia, Louisiana, at approximately 8 a.m., October 3. Three sites, Waterford-3, River Bend Station, and Grand Gulf, were potentially effected, but Lili tracked west of all three sites. The approximate maximum windspeed experienced at Waterford-3 was 50 mph, at River Bend was 40 mph, and at Grand Gulf was 35 mph. No damage was reported at any site. The only concerns were with respect to the adequacy of the emergency response infrastructures at the sites. The only impediments that have been identified relate to the operability of sirens in the Grand Gulf area. River Bend reduced reactor power as a precautionary action.

As part of the Region's response, the Region IV Government Liaison Officer co-located with FEMA Region VI in Denton, Texas. Coordination with FEMA Region VI was maintained throughout the response.

Region IV inspectors will be on site throughout the day to monitor licensee site activities. The states of Louisiana and Mississippi have been informed.

Region IV has informed the EDO, NRR, and NSIR.

CONTACTS: David N. Graves (817) 860-8141
Vincent G. Gaddy (817) 860-8114
Charles J. Paulk (817) 860-8236

[Preliminary Notification Index | News and Information | Main NRC Home Page | E-mail]

PNO-I-02-017 - Susquehanna Steam Electric Station

DCS No.: 050387388021003
Date: October 3, 2002

PRELIMINARY NOTIFICATION OF EVENT OR UNUSUAL OCCURRENCE- PNO-I-02-017

This preliminary notification constitutes EARLY notice of events of POSSIBLE safety or public interest significance. The information is as initially received without verification or evaluation, and is basically all that is known by the Region I staff on this date.

Facility	**Licensee Emergency Classification**
Susquehanna Steam Electric Station	Notification of Unusual Event
Units 1 and 2	Alert
PPL Susquehanna, LLC	Site Area Emergency
Berwick, PA 18603-0035	General Emergency
	X Not Applicable

Docket No.: 050-387
050-388
License No.: NPF-14
NPF-22

SUBJECT: ***Unusual Event Declared Due to Onsite Fire/Explosion Within the Protected Area***

On October 3, 2002, a transformer failure resulted in a loss of one of the two offsite power sources to the Susquehanna Steam Electric Station. At 3:15 a.m. plant operators declared an Unusual Event due to a fire and explosive failure of Startup Transformer (T-20). T-20 is located outside and adjacent to the Unit-2 Turbine Building and within the protected area of the plant. The event occurred at approximately 2:30 a.m. The fire was extinguished in about 10 minutes by the automatic response of the fire deluge system. PPL terminated the Unusual Event at 5:52 a.m. The damage from the fire appears to be limited to the startup transformer and the immediate surrounding area. PPL's initial investigation of the cause of the failure was able to rule out any malicious act and is now focusing on internal failures or solar magnetic effects.

The Unit-1 reactor was at 100% power and Unit-2 reactor was critical at low power in the process of starting up at the time of the loss of T-20. Following the loss of T-20, the essential loads on T-20 automatically swapped to the T-10 Startup Transformer. Unit-1 continued to operate at 100% power and was minimally affected by the loss of T-20. The operators manually shutdown Unit-2 because both reactor recirculation pumps tripped as a result of the loss of T-20.

As of 2 p.m. on October 3, Unit-1 was at 100% power with one of the offsite power sources to Unit-1 unavailable because of the loss of the T-20. PPL is evaluating repair options to restore the second offsite power source. The T-10 Startup Transformer is available as the offsite power source for both Units-1 and 2. Unit-2 is proceeding to cold shutdown.

The resident inspectors responded to the event and observed control room activities. PPL made initial notifications to the state, local officials and NRC.

The Region I Public Affairs is prepared to respond to media inquiries.

The Commonwealth of Pennsylvania has been informed.

The information presented herein has been discussed with PPL and is current as of 1:00 p.m. on October 3, 2002.

Contact:	S. Hansell	D. Florek	M. Shanbaky
	570-542-2134	(610) 337-5185	(610) 337-5209

[Preliminary Notification Index | News and Information | Main NRC Home Page | E-mail]

PNO-IV-02-052 - Entergy Operations, Inc.

[Preliminary Notification Index | News and Information | Main NRC Home Page | E-mail]

October 2, 2002

PRELIMINARY NOTIFICATION OF EVENT OR UNUSUAL OCCURRENCE -- PNO-IV-02-052

This preliminary notification constitutes EARLY notice of events of POSSIBLE safety or public interest significance. The information is as initially received without verification or evaluation, and is basically all that is known by the Headquarters staff on this date.

Facility	__ **Licensee Emergency Classification**
Entergy Operations, Inc.	__ Notification of Unusual Event
Waterford 3 and River Bend Station	__ Alert
River Bend Station, St. Francisville, Louisiana	__ Site Area Emergency
Docket: 50-382 (W3) and 50-458 (RBS)	__ General Emergency
License Nos.: NPF-38 (W3) and NPF-47 (RBS)	X Not Applicable

Subject: ***Region IV Response to Hurricane/severe Weather on Gulf Coast***

Description:

As of 1 p.m. (CDT) on October 2, Hurricane Lili was located approximately 325 miles south of New Orleans, Louisiana. Projected landfall is along the extreme upper Texas or western Louisiana Gulf Coast midday on October 3. Waterford 3 Steam Electric Station in Taft, Louisiana; River Bend Station in St. Francisville, Louisiana; and South Texas Project near Bay City, Texas, are in or near the projected storm path. Grand Gulf nuclear power plant on the Mississippi River, near Port Gibson, Mississippi, is not considered a primary hurricane target since it is a substantial distance inland.

Region IV has implemented its hurricane response procedure. The Region IV Incident Response Center will be activated for monitoring of the hurricane and plant conditions beginning at approximately 4 p.m. (CDT) today and will continue until the hurricane passes and no longer threatens plant operations. Inspectors from the RIV office in Arlington, Texas are currently on site reviewing licensee preparations for the impending severe weather and to enable the resident inspectors to take care of personal preparations for the storm. They will remain on site to monitor plant conditions until the storm is no longer a threat. An NRC liaison will be dispatched to the Federal Emergency Management Agency's Region 6 office in Denton, Texas, and will maintain frequent communications with the states of Louisiana and Texas as needed.

The states of Louisiana and Texas have been informed.

Region IV plans to issue a press release. Region IV has informed the OEDO, NRR, and PA.

This information is current as of 1:30 p.m. (CDT) on October 2, 2002.

CONTACTS: David N. Graves (817) 860-8141
Vincent G. Gaddy (817) 860-8114
Charles J. Paulk (817) 860-8236

[Preliminary Notification Index | News and Information | Main NRC Home Page | E-mail]

October 3, 2002

EA-02-124

Mr. John L. Skolds, President
Exelon Nuclear
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: CONFIRMATORY ORDER (EFFECTIVE IMMEDIATELY)
(OFFICE OF INVESTIGATIONS REPORT NO. 3-2001-005)

Dear Mr. Skolds:

The enclosed Confirmatory Order is being issued to Exelon Generation Company, LLC, (Exelon) and AmerGen Energy Company, LLC (AmerGen) (Licensees) in order to confirm certain commitments, as set forth in Section V of the Order, to assure the Licensees' compliance with the Commission's employee protection regulations, 10 CFR 50.7. In view of the Confirmatory Order and consent by the Licensees thereto, dated September 27, 2002, the NRC is exercising its enforcement discretion pursuant to Section VII.B. of the NRC Enforcement Policy, and will not issue Notices of Violation or propose a civil penalty in this matter. As indicated on September 27, 2002, the Licensees agreed to the terms of this Order, agreed that this Order shall be effective immediately, and waived all rights to a hearing on all or any part of this Order.

Pursuant to Section 223 of the Atomic Energy Act of 1954, as amended, any person who willfully violates, attempts to violate, or conspires to violate, any provision of this Order shall be subject to criminal prosecution as set forth in that section. Violation of this Order may also subject the person to civil monetary penalties.

Questions concerning this Order should be addressed to James Luehman, Deputy Director, Office of Enforcement, who can be reached at telephone number 301-415-2741.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and the enclosed Order will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS) accessible from the NRC Web site at www.nrc.gov/reading-rm/adams.html.

Sincerely,

/RA/

Frank J. Congel
Director, Office of Enforcement

Dockets No.
50-456; 50-457 (Braidwood 1 & 2)
50-454; 50-455 (Byron 1 & 2)
50-461 (Clinton)
50-010; 50-237; 50-249 (Dresden 1, 2 & 3)

50-373; 50-374 (LaSalle 1 & 2)
50-352; 50-353 (Limerick 1 & 2)
50-219 (Oyster Creek)
50-171; 50-277; 50-278 (Peach Bottom 1, 2 & 3)
50-254; 50-265 (Quad Cities 1 & 2)
50-289 (Three Mile Island 1)
50-295; 50-304 (Zion 1 & 2)

Licenses No.
NPF-72; NPF-77 (Braidwood 1 & 2)
NPF-37; NPF-66 (Byron 1 & 2)
NPF-62 (Clinton)
DPR-2; DPR-19; DPR-25 (Dresden 1, 2 & 3)
NPF-11; NPF-18 (LaSalle 1 & 2)
NPF-39; NPF-85 (Limerick 1 & 2)
DPR-16 (Oyster Creek)
DPR-12; DPR-44; DPR-56 (Peach Bottom 1, 2 & 3)
DPR-29; DPR-30 (Quad Cities)
DPR-50 (Three Mile Island 1)
DPR-39; DPR-48 (Zion)

Enclosure: Confirmatory Order (Effective Immediately)

cc w/encl:
Site Vice President - Braidwood Station
Site Vice President - Byron Station
Site Vice President - Clinton Station
Site Vice President - Dresden Nuclear Power Station
Site Vice President - LaSalle County Station
Site Vice President - Limerick Generating Station
Site Vice President - Oyster Creek Nuclear Power Station
Site Vice President - Peach Bottom Atomic Station
Site Vice President - Quad Cities Nuclear Power Station
Site Vice President - Three Mile Island Nuclear Power StationL
Dresden Nuclear Power Station Decommissioning Plant Manager
Zion Nuclear Power Station Decommissioning Plant Manager
Senior Vice President - Nuclear Services
Senior Vice President - Mid-West Regional Operating Group
Vice President - Mid-West Operations Support
Vice President - Licensing and Regulatory Affairs
A. F. Kirby, III External Operations - Delmarva Power & Light Co.
H. C. Kresge, Manager, External Operations, Connectiv
R. McLean, Power Plant Siting, Nuclear Evaluations
R. Ochs, Maryland Safe Energy Coalition
J. H. Walter, Chief Engineer, Public Safety Commission of Maryland
Mr. & Mrs. Dennis Hiebert, Peach Bottom Alliance
Mr. & Mrs. Kip Adams
Document Control Desk - Licensing (Braidwood, Byron, Clinton, Dresden, LaSalle, Quad Cities, and Zion)
Illinois Department of Nuclear Safety
State Liaison Officer, State of Illinois

State Liaison Officer, State of Wisconsin
State Liaison Officer, State of Iowa
State of Maryland
State of New Jersey
Commonwealth of Pennsylvania
Chairman, Illinois Commerce Commission
A. C. Settles, Illinois Department of Nuclear Safety
Vice President - Law and Regulatory Affairs MidAmerican Energy Company
W. Leach, Manager of Nuclear MidAmerican Energy Company
K. Nollenberger, County Administrator
M. Aguilar, Assistant Attorney General
Mayor, City of Zion
TMI-Alert (TMIA)
D. Allard, PADER
M. Schoppman, Framatome
R. Shadis, New England Coalition Staff
N. Cohen, Coordinator - Unplug Salem Campaign
E. Gbur, Coordinator - Jersey Shore Nuclear Watch
E. Zobian, Coordinator - Jersey Shore Anti Nuclear Alliance
Chief - Division of Nuclear Safety
Secretary, Nuclear Committee of the Board

United States of America

U.S. Nuclear Regulatory Commission

20

In the Matter of	)	EA-02-124
Exelon Generation Company, LLC	)	
and	)	
AmerGen Energy Company, LLC	)	
	)	
	)	
Braidwood Station, Units 1 & 2	)	Dockets No. 50-456; 50-457
Byron Station, Units 1 & 2	)	50-454; 50-455
Clinton Power Station	)	50-461
Dresden Nuclear Power Station, Units 1, 2 & 3	)	50-10; 50-237; 50-249
LaSalle County Station, Units 1 & 2	)	50-373; 50-374
Limerick Generating Station, Units 1 & 2	)	50-352; 50-353
Oyster Creek Nuclear Generating Station	)	50-219
Peach Bottom Atomic Power Station, Units 1, 2 & 3	)	50-171: 50-277; 50-278
Quad Cities Nuclear Power Station, Units 1 & 2	)	50-254; 50-265
Three Mile Island Nuclear Station, Unit 1	)	50-289
Zion Nuclear Power Station, Units 1 & 2	)	50-295; 50-304
	)	
	)	
	)	Licenses No. NPF-72; NPF-77
	)	NPF-37; NPF-66
	)	NPF-62
	)	DPR- 2; DPR-19; DPR-25
	)	NPF-11; NPF-18
	)	NPF-39; NPF-85
	)	DPR-16
	)	DPR-12; DPR-44; DPR-56
	)	DPR-29; DPR-30
	)	DPR-50
	)	DPR-39; DPR-48

CONFIRMATORY ORDER MODIFYING LICENSES (EFFECTIVE IMMEDIATELY)

I

Exelon Generation Company, LLC (Exelon) and AmerGen Energy Company, LLC (AmerGen) (Licensees) are the holders of twenty-one NRC Facility Operating Licenses issued by the Nuclear Regulatory Commission (NRC or Commission) pursuant to 10 CFR Part 50, which authorizes the operation of the specifically named facilities in accordance with the conditions specified in each license. Licenses No. NPF-72 and NPF-77 were issued on July 2, 1987, and May 20, 1988, to operate the Braidwood Station, Units 1 and 2. Licenses No. NPF-37 and NPF-66 were issued on February 14, 1985, and January 30, 1987, to operate Byron Station, Units 1 and 2. License No. NPF-62 was issued on April 17, 1987 to operate the Clinton Power Station. Licenses No. DPR-2 and DPR-25 were issued on September 28, 1959, and January 12, 1971, to operate Dresden Nuclear Power Station, Units 1 and 3 (Dresden Station Unit 1 is currently in decommissioning). License No. DPR-19 was extended on February 20, 1991, for Dresden Nuclear Power Station, Unit 2. Licenses No. NPF-11 and NPF-18 were issued on April 17, 1982, and February 16, 1983, to operate LaSalle County Station, Units 1 and 2. Licenses No. NPF-39 and NPF-85 were issued on August 8, 1985, and August 25, 1989, to operate the Limerick Generating Station, Units 1 and 2. License No. DPR-16 was extended on July 2, 1991, for the Oyster Creek Nuclear Generating Station. License No. DPR-12 was issued on January 24, 1966, to operate Peach Bottom Atomic Power Station, Unit 1, which was shut down on October 31, 1974, and is in safe storage. Licenses No. DPR-44 and DPR-56 were issued on October 25, 1973, and July 2, 1974, to operate Peach Bottom Atomic Power Station, Units 2 & 3. Licenses No. DPR-29 and DPR-30 were issued on December 14, 1972, for the operation of both units at the Quad Cities Nuclear Power Station, Units 1 and 2. License No. DPR-50 was issued on April 19, 1974, to operate the Three Mile Island Nuclear Power Station, Unit 1. Licenses No. DPR-39 and DPR-48 were issued on October 19, 1973, and November 14, 1973, for operation of the Zion Nuclear Power Station, Units 1 and 2 (the Zion Station is currently in decommissioning).

II

On January 29, 2001, the NRC Office of Investigations (OI) initiated an investigation to determine if a former Exelon employee performing work at the Byron Station had been discriminated against for raising safety concerns. In its Report No. 3-2001-005, issued March 26, 2002, OI concluded that an Exelon corporate manager deliberately discriminated against the former employee on August 25, 2000, in violation of the NRC regulations prohibiting employment discrimination, 10 CFR 50.7, "Employee Protection," by not selecting the employee for a new position. On June 17, 2002, the NRC staff contacted Exelon management to schedule a predecisional enforcement conference. To expedite resolution of this matter, Exelon requested the opportunity to present a settlement proposal to the NRC prior to a predecisional enforcement conference. The NRC staff agreed to this request.

III

Representatives of Exelon met with the NRC staff on July 2, July 18, July 30, September 9 and September 11, 2002, to discuss the terms of the Exelon settlement proposal. In an August 5, 2002 letter, Exelon described the proposed settlement and on September 27, 2002, the Licensees committed to a number of corrective actions with respect to employee protection, agreed to have the corrective actions confirmed by Order, and admitted that a violation of 10 CFR 50.7 had occurred. The corrective actions include, but are not limited to, counseling management personnel involved in the violation of 10 CFR 50.7, and training all vice-presidents and plant managers throughout the Licensees' organization (at every nuclear station and at corporate headquarters) on the provisions of the employee protection regulation. These individuals, in turn, will train their subordinate managers. The Licensees will also modify management training programs as appropriate regarding the provisions of 10 CFR 50.7.

IV

On September 27, 2002, the Licensees consented to issuance of this Order with the commitments described in Section V below, waived any right to a hearing on this Order, and agreed to all terms of this Order, including that it shall be effective immediately.

I find that the Licensees' commitments as set forth in Section V, below, are acceptable and necessary, and conclude that since Exelon admitted the violation of 10 CFR 50.7 and since the Licensees committed to taking comprehensive corrective actions by implementing this Confirmatory Order, the NRC staff's concern regarding employee protection can be resolved through confirmation of the Licensees' commitments by this Order. I further find that the Licensees' approach to resolving this matter is salutary and efficient, and that this resolution is in the public interest. Accordingly, the NRC staff exercises its enforcement discretion pursuant to Section VII.B.6 of the NRC Enforcement Policy and will not issue Notices of Violation or a civil penalty in this case.

V

Accordingly, pursuant to sections 103, 104b, 161b, 161i, 161o, 182 and 186 of the Atomic Energy Act of 1954, as amended, and the Commission's regulations in 10 CFR 2.202 and 10 CFR Part 50, IT IS HEREBY ORDERED, EFFECTIVE IMMEDIATELY, THAT LICENSE NOS. NPF-72, NPF-77, NPF-37, NPF-66, NPF-62, DPR-2, DPR-19, DPR-25, NPF-11, NPF-18, NPF-39, NPF-85, DPR-16, DPR-12, DPR-44, DPR-56, DPR-29, DPR-30, DPR-50, DPR-39, AND DPR-48 ARE MODIFIED AS FOLLOWS:

1. Exelon will counsel and coach personnel involved in the violation of 10 CFR 50.7, which occurred on August 25, 2000, to emphasize the importance of a safety conscious work environment and provisions of 10 CFR 50.7. The counseling will be conducted by a corporate Exelon executive not involved in the violation described herein and who shall be senior to those counseled.

2. An Exelon corporate executive will train and coach every executive-level employee (defined to include plant managers and all vice-president level personnel) throughout the licensed organizations, including every nuclear station and headquarters, on the employee protection provisions of 10 CFR 50.7. The sessions will be conducted by an Exelon executive knowledgeable about the issues involved in the August 25, 2000, violation and will be held in small groups to assure focus and interactive involvement of every executive. The sessions will include a case study of the selection decision that caused this enforcement action and a discussion of the lessons learned.

3. Each executive trained pursuant to Paragraph 2 above will be provided a communications package for use in training the managers in that executive's chain-of-command regarding these issues and the Licensees' expectations for handling employee interactions.

4. The Licensees will enhance training on the prevention of employment discrimination beyond that in its existing management training programs. Lesson plans and other materials used in management training programs on the prevention of employment discrimination will be reviewed and revised as appropriate to address maintaining a safety conscious work environment and the employee protection provisions of 10 CFR 50.7. The on-going training will be conducted at a frequency consistent with the Licensees' existing policies, practices and procedures.

5. The Licensees will review the internal candidate selection process to ensure that the process incorporates the principles of employee protection under 10 CFR 50.7.

6. A communication will be distributed to all employees of the Licensees' organizations that strongly reaffirms management's commitment to fostering a safety-conscious work

environment in all organizations at all sites and in its headquarters organization. The Licensees will also reaffirm to all employees the Licensees' commitments to a strong and viable Employee Concerns Program and will reiterate the various means that all employees may employ to raise issues that may be of concern to them.

7. Exelon will review all work environment surveys conducted since September 2000 at the Byron Station (where the former employee previously worked) to assure that management responses to any findings were implemented to assure that no residual effect exists in the safety-conscious work environment at the station as a result of the selection decision. Exelon will provide to the Regional Administrator, NRC Region III, Lisle, Illinois, a written description of the results of this review and any actions taken or planned to be taken to assure that a safety conscious work environment exists at the Byron Station.

8. The Licensees will accomplish these actions within six months of the date of this Order and will furnish a written report of the results achieved to the Director, Office of Enforcement, within 30 days following completion.

The Director, Office of Enforcement may relax or rescind, in writing, any of the above conditions upon a showing by the Licensees of good cause.

VI

Any person adversely affected by this Confirmatory Order, other than the Licensees, may request a hearing within 20 days of its issuance. Where good cause is shown, consideration will be given to extending the time to request a hearing. A request for extension of time in which to submit a request for a hearing must be made in writing to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and include a statement of good cause for the extension. Any request for a hearing shall be submitted to the Secretary, Office of the Secretary of the Commission, U.S. Nuclear Regulatory Commission, ATTN: Chief, Rulemaking and Adjudications Staff, Washington, DC 20555. Copies of the hearing request shall also be sent to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555; to the Assistant General Counsel for Materials Litigation and Enforcement at the same address; to the Regional Administrator, NRC Region III, 801 Warrenville Road, Lisle, IL 60532-4351; to the Regional Administrator, NRC Region I, 475 Allendale Road, King of Prussia, PA 19406-1415; and to the Licensees. Because of continuing disruptions in delivery of mail to United States Government offices, it is requested that requests for hearing be transmitted to the Secretary of the Commission either by means of facsimile transmission to 301-415-1101 or by e-mail to hearingdocket@nrc.gov and also to the Office of the General Counsel either by means of facsimile transmission to 301-415-3725 or by e-mail to OGCMailCenter@nrc.gov. If such a person requests a hearing, that person shall set forth with particularity the manner in which his interest is adversely affected by this Order and shall address the criteria set forth in 10 C.F.R. § 2.714(d).[(1)]

If a hearing is requested by a person whose interest is adversely affected, the Commission will issue an Order designating the time and place of any hearing. If a hearing is held, the issue to be considered at such hearing shall be whether this Order should be sustained.

In the absence of any request for hearing, or written approval of an extension of time in which to request a hearing, the provisions specified in Section V above shall be final twenty (20) days from the date of this Order without further order or proceedings. If an extension of time for requesting a hearing has been approved, the provisions specified in Section V shall be final when the extension expires if a hearing request has not been received. A REQUEST FOR HEARING SHALL NOT STAY THE IMMEDIATE EFFECTIVENESS OF THIS ORDER.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

/RA/

Frank J. Congel
Director, Office of Enforcement

Dated at Rockville, Maryland
this 3rd Day of October 2002

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1. The most recent version of Title 10 of the Code of Federal Regulations, published January 1, 2002, inadvertently omitted the last sentence of 10 C.F.R. 2.714(d) and subparagraphs (d)(1) and (2), regarding petitions to intervene and contentions. Those provisions are extant and still applicable to petitions to intervene. Those provisions are as follows: "In all other circumstances, such ruling body or officer shall, in ruling on-(1) A petition for leave to intervene or a request for hearing, consider the following factors, among other things: (i) The nature of the petitioner's right under the Act to be made a party to the proceeding. (ii) The nature and extent of the petitioner's property, financial, or other interest in the proceeding. (iii) The possible effect of any order that may be entered in the proceeding on the petitioner's interest. (2) The admissibility of a contention, refuse to admit a contention if: (i) The contention and supporting material fail to satisfy the requirements of paragraph (b)(2) of this section; or (ii) The contention, if proven, would be of no consequence in the proceeding because it would not entitle petitioner to relief.

September 9, 2002

EA-02-159

Mr. Peter E. Katz
Vice President - Calvert Cliffs Nuclear Power Plant
Constellation Generation Group
Calvert Cliffs Nuclear Power Plant, Inc.
1650 Calvert Cliffs Parkway
Lusby, MD 20657-4702

SUBJECT: FINAL SIGNIFICANCE DETERMINATION FOR A WHITE FINDING AND NOTICE OF VIOLATION AT CALVERT CLIFFS NUCLEAR POWER PLANT (NRC Inspection Report 50-317/02-010; 50-318/02-010)

Dear Mr. Katz:

The purpose of this letter is to provide you with the final results of our significance determination for the preliminary White finding identified during an inspection completed on July 19, 2002, at the Calvert Cliffs Nuclear Power Plant, Units 1 & 2. The results of the inspection were discussed with you and other members of your staff at an exit meeting on July 19, 2002. The inspection finding was assessed using the significance determination process and was preliminarily characterized as White, a finding with low to moderate importance to safety, which may require additional NRC inspections. The basis for this preliminary White finding was explained in our August 12, 2002 letter that transmitted the subject inspection report.

This preliminary White finding involved 49 sirens located in Calvert County, Maryland, which are part of your alert and notification system (ANS), that were not capable of being activated in a timely manner between August 14, 2001 and November 5, 2001 (84 days) due to the removal of a computer icon used for activating the sirens at the 911 Center. This preliminary White finding was also associated with an apparent violation of 10 CFR Part 50, Appendix E.

In a letter dated August 12, 2002, the NRC transmitted the inspection report and provided you an opportunity to either request a regulatory conference to discuss this finding, or explain your position in a written response. In a telephone conversation with Mr. David Silk of NRC, Region I, on August 14, 2002, Mr. Mark Geckle of your staff indicated that the Constellation Generation Group did not contest the characterization of the risk significance of this finding and declined the opportunity to discuss this issue in a Regulatory Conference or provide a written response.

Based on the information developed during the inspection, the NRC has concluded that the inspection finding is appropriately characterized as White. You have 10 business days from the date of this letter to appeal the staff's determination of significance for the identified White finding. Such appeals will be considered to have merit only if they meet the criteria given in NRC Inspection Manual Chapter 0609, Attachment 2. In addition, the White finding was associated with a violation of 10 CFR Part 50, Appendix E, which requires that a capability exist to complete the initial notification of the public within the plume exposure pathway Emergency Planning Zone (EPZ) within about 15 minutes from the time that State and local officials are notified that a situation exists requiring urgent action. The violation is cited in the enclosed Notice of Violation and the circumstances surrounding the violation were described in detail in the subject inspection report. In accordance with the NRC Enforcement Policy, NUREG-1600, this Notice of Violation is considered escalated enforcement action because it is associated with a White finding.

You are required to respond to this letter and should follow the instructions specified in the enclosed Notice of Violation when preparing your response. Because plant performance for this issue has been determined to be in the regulatory response band, we will use the NRC Action Matrix to determine the most appropriate NRC response for this event. We will notify you, by separate correspondence, of that determination.

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

Sincerely,

/RA/

Hubert J. Miller
Regional Administrator

Docket Nos.: 50-317, 50-318
License Nos.: DPR-53, DPR-69

Enclosure: Notice of Violation

cc w/encl:
M. Geckle, Director, Nuclear Regulatory Matters (CCNPPI)
R. McLean, Administrator, Nuclear Evaluations
K. Burger, Esquire, Maryland People's Counsel
R. Ochs, Maryland Safe Energy Coalition
J. Petro, Constellation Power Source
State of Maryland (2)

NOTICE OF VIOLATION

Constellation Generation Group
Calvert Cliffs Nuclear Power Plant

Docket Nos. 50-317; 50-318
License Nos. DPR-53; DPR-69
EA-02-159

During an NRC inspection conducted between July 15 - July 19, 2002, the results of which were discussed at an exit meeting on July 19, 2002, a violation of NRC requirements was identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," NUREG-1600, the violation is listed below:

> 10 CFR 50.54(q) requires in part, that a licensee authorized to possess and operate a nuclear power reactor shall follow and maintain in effect emergency plans which meet the standards in 10 CFR 50.47(b) and the requirements of Appendix E of this part.

10 CFR 50.47(b)(5) requires in part, that procedures have been established for notification, by the licensee, of State and local response organizations; and means to provide early notification to the populace within the plume exposure pathway Emergency Planning Zone (EPZ) have been established.

10 CFR 50, Appendix E, Section IV.D.3, states in part, that a design objective of the prompt public notification system shall be to have the capability to essentially complete the initial notification of the public with the plume exposure pathway EPZ within about 15 minutes after declaring an emergency. The licensee shall demonstrate that State and local officials have the capability to make a public notification decision promptly on being informed by the licensee of an emergency condition.

Contrary to the above, between August 14, 2001 and November 5, 2002, the licensee was not capable of completing the initial notification of the public with the plume exposure pathway EPZ within about 15 minutes in Calvert County, Maryland. Specifically, the computer icon in the 911 Center for Calvert County used to activate the sirens had been inadvertently removed, which disabled the system and prevented it from being activated in a timely manner.

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

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

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

Because your response will be made available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS), to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the public without redaction. ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room). If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request withholding of such material, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.790(b) to support a request for withholding confidential commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21.

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

Dated this 9th day of September 2002

(2)

United States Nuclear Regulatory Commission

UPDATE ON ISSUES IN 1998 AGENCY PROGRAM PLAN FOR HIGH-BURNUP FUEL

Ralph Meyer

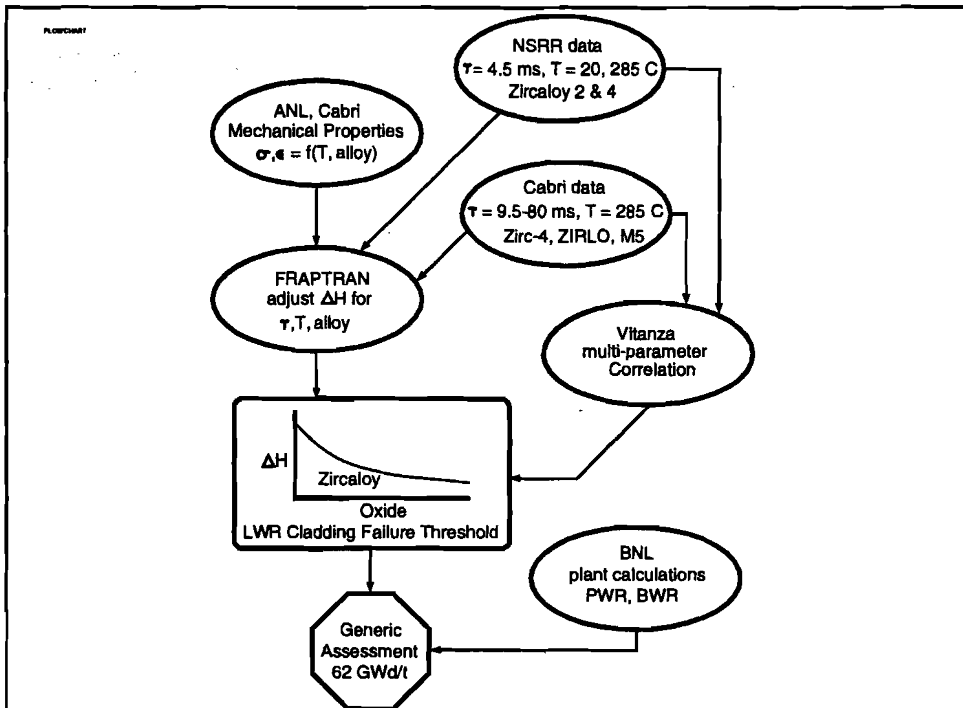
Office of Nuclear Regulatory Research

ACRS

October 10, 2002

ORIGINAL LIST OF ISSUES

1	Cladding Integrity and Fuel Design Limits	Resolved in original plan (no further discussion)
2	Control Rod Insertion Problems	Resolved in original plan (no further discussion)
3	Criteria and Analysis for Reactivity Accidents	NRC confirmatory assessment at 62 GWd/t, early 2005. Revision of Reg. Guide 1.77, TBD.
4	Criteria and Analysis for Loss-of-Coolant Accidents	Zircaloy criteria and models at 62 GWd/t, 2004. New performance-based criteria possible.
5	Criteria and Analysis for BWR Power Oscillations (ATWS)	Schedule to be determined
6	Fuel Rod and Neutronic Computer Codes for Analysis	Resolved
7	Source Term and Core Melt Progression	Technical issues essentially resolved. Revision of Reg. Guide 1.183, TBD.
8	Transportation and Dry Storage	Research Information Letter, 2004
9	High Enrichments (>5%)	No activity needed now (no further discussion)



Correlation for the RIA Failure Threshold (Vitanza 2001)

$$H_F = \left[200 \cdot \frac{25 + 10D}{Bu} + 0.3\Delta\tau\right]\left(1 - \frac{0.85OX}{W}\right)^2$$

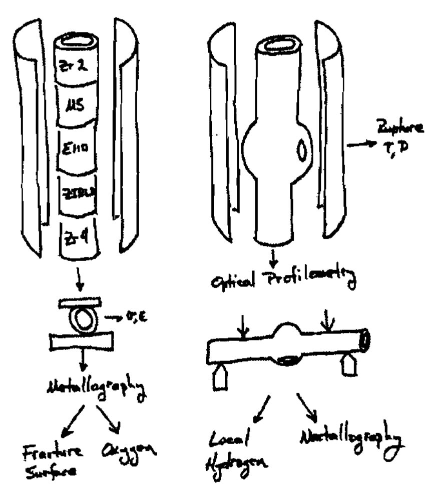
H_F = Fuel Enthalpy Failure Limit (maximum of 200 cal/g)
Bu = Burnup in MWd/kg
D = 0% (brittle) to 1% (ductile) cladding hoop strain limit
$\Delta\tau$ = Pulse Width (maximum of 75 msec)
OX = Oxide thickness in (um)
W = Cladding wall thickness (um)

CRITERIA AND ANALYSIS FOR LOSS-OF-COOLANT ACCIDENTS

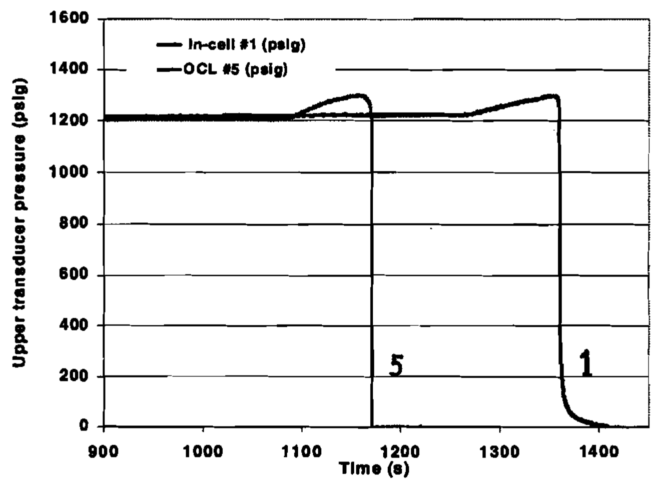
ISSUE: Embrittlement criteria in 10 CFR 50.46 and related evaluation models are probably affected by burnup and alloy. Check and revise if necessary.

METHOD: (see following slides)

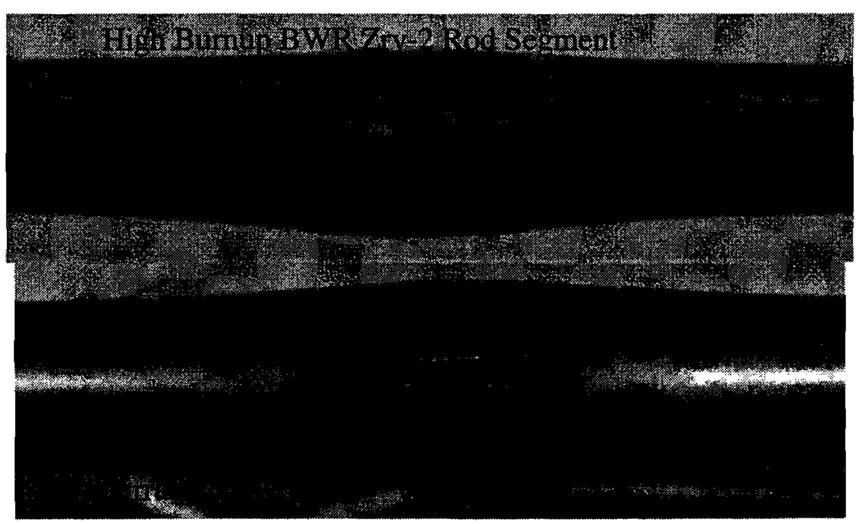
SCHEDULE: Zircaloy criteria and models at 62 GWd/t in 2004



PRESSURE HISTORIES FOR IN-CELL TEST #1 AND OUT-OF-CELL TEST #5



BURST OPENING COMPARISON

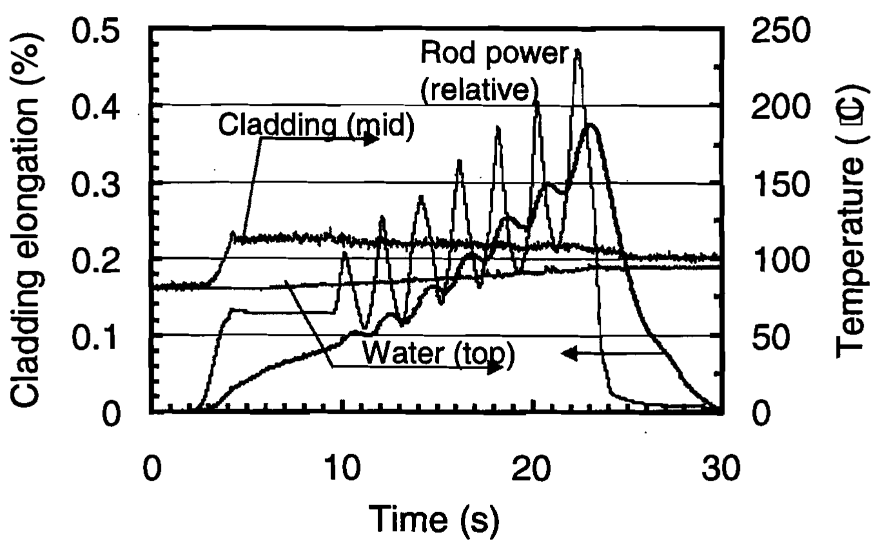


CRITERIA AND ANALYSIS FOR BWR POWER OSCILLATIONS (ATWS)

ISSUE: 280 cal/g limit currently used may not be adequate to ensure benign result in PRA for "successfully" terminated oscillations

METHOD: Analytical + some experimental separate effects

SCHEDULE: TBD



SOURCE TERM AND CORE MELT PROGRESSION

ISSUE: Applicability of NUREG-1465 source terms to high-burnup fuel

METHOD: Expert elicitation, more data

SCHEDULE: Expert elicitation completed in June 2002
VERCORS, PHEBUS, VEGA data as available
Revision of Reg. Guide 1.183 TBD

TRANSPORTATION AND DRY STORAGE

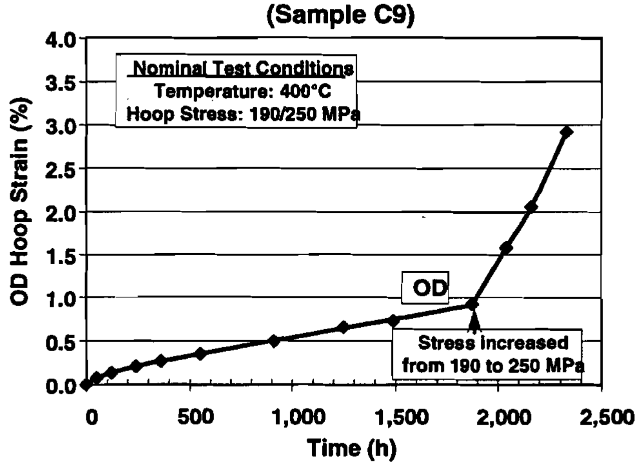
ISSUE: What is the effect of burnup on fission product inventory (shielding, heat source, activity) and cladding degradation (removal from storage)?

METHOD: Direct tests and measurements

SCHEDULE: ANL tests on Zircaloy in 2003
Research Information Letter in 2004

Summary Test Results

Effect of Increased Stress
(Sample C9)



Preliminary HBR Creep Matrix
(07/12/02 Version)

H-content wppm	Temp. °C	Stress MPa	Time h	Predicted Strain, %
650±50	400	220	TBD	TBC
650±50	400	190	TBD	TBC
650±50	400	160	TBD	TBC
650±50	420	160	TBD	TBC
650±50	380	220	TBD	TBC
650±50	380	190	TBD	TBC
650±50	380	160	TBD	TBC
650±50	360	220	TBD	TBC
650±50	360	190	TBD	TBC

Proposed Test Matrix/Schedule Cabri Project

- CIP-0 series: Two tests in the Na-loop in 2002
- CIP-Q :Qualification test for the water loop in 2005
- CIP-1 : Tests in water loop, comparison tests of CIP-0 tests, 2006+
- CIP-2: High burnup UO2 fuel, >80 GWD/T
- CIP-3: Mechanistic understanding on effects of pulse width, fuel microstructure, etc
- CIP-4 Study of high burnup MOX fuel, > 60 GWD/T
- CIP-5 To be defined

CIP0 Tests Will Determine Future Scope Of RIA

- RIA criteria proposed was based on Zircaloy clad
- Two additional RIA tests in CABRI Na-loop in 2002
  - CIP0-2
    - M5 rod (~ 20μm, ~73 GWd/T)
    - Test will be performed in 10/02
    - 30 ms, with enthalpy of ~95 cal/g (based on calculations)
  - CIP0-1
    - ZIRLO rod (~ 100μm, ~73 GWd/T)
    - Test will be performed in 11/02
    - 30 ms, with enthalpy of ~90 cal/g (based on calculations)
- New parameters involved
  - Higher burnup, 63 GWD/T 73 GWD/T
  - New alloys, M5 and Zirlo

EPRI Topical Report on Reactivity Initiated Accidents

Undine Shoop
Office of Nuclear Reactor Regulation
October 10, 2002

3

RIA Criteria History

- Agency Program Plan for High Burnup Fuel – July 6, 1998
  - Industry will have to provide the Criteria, Data base, and Models for Burnup > 62 GWD/MTU
  - Industry will have to perform the research necessary to develop the data base to support extended burnup ranges > 62 GWD/MTU
  - RES will confirm criteria for burnup < 62 GWD/MTU

NRC Preliminary Review Plan Purpose

- To focus resources appropriately to provide a detailed review and identify all the elements needed to complete the review

NRC Preliminary Review Plan Elements

- Data Verification
  - Correct application in the methodology
  - Correct application in a manner consistent with the methods used to generate it
  - Statistically sound combination of the data sets
- SED/CSED Theory and Model
  - Investigation and verification of the equivalence of SED/CSED model to Rice's J/Jc formulation
  - FRAPTRAN independent verification
- Fuel Rod Failure Threshold
  - Validation of this application
  - Review of applicability to current and future proposed fuel types
- Core Coolability Limit
  - Application verification

NRC Preliminary Review Plan Elements – Cont.

- FALCON Code
  - Review of the code
- Fuel Dispersal
  - Review data for applicability of the phenomena to the proposed safety limit
- Uncertainty and Conservatism
  - Data uncertainty verification
  - Conservatism confirmation
- Limitations of the Criteria
  - Review data for limits of applicability which would create limitations of the methodology application
- Safety Evaluation Conditions of Acceptance
- Revision of associated RG and SRPs

Future Activities

- Final Review Plan – December 31, 2002

New Reactor Licensing
Presentation to the ACRS

October 10, 2002

James Lyons, Director
New Reactor Licensing Project Office
Office of Nuclear Reactor Regulation

New Reactor Licensing Schedule

ID	Task Name	2002–2006 (Qtr 1 2002 – Qtr 3 2006)
1	**Early Site Permits**	
2	Exelon	
3	Entergy	
4	Dominion	
5	**Design Certifications**	
6	**AP1000 Certification**	
7	Application submitted	03/28
8	Request for Additional Information	09/30
9	Draft Safety Evaluation Report	06/16
10	Final Safety Evaluation Report	July 2004
11	Final Design Approval	October 2005
12	Rulemaking completed	December 2005
13	**ESBWR pre-application**	
14	kickoff meeting	6/20&21/02
15	Phase 1 completed	August 2002
16	Request for Additional Information	June 2003
17	Draft Safety Evaluation Report	September 2003
18	Phase 2 complete	December 2003
19	ESBWR Design Certification Applic.	DC application - Early CY 2004 (projected)
20	**ACR-700 pre-application review**	DC application late CY2004 (projected)
21	**SWR-1000 pre-application review**	Preapp submittal mid 2004 - design cert app late 2005 (projected)
22	**GT-MHR pre-application review**	DC application late CY2004 (projected)
23	**IRIS pre-application review**	DC application late CY2004 (projected)
24	**PBMR pre-application review**	DC early CY2006 (proj)

GE Nuclear Energy

ESBWR Design and Technology Overview

(1390 MWe natural circulation with passive safety systems)

A.S. Rao

October 10, 2002
ACRS Meeting
Rockville, Maryland

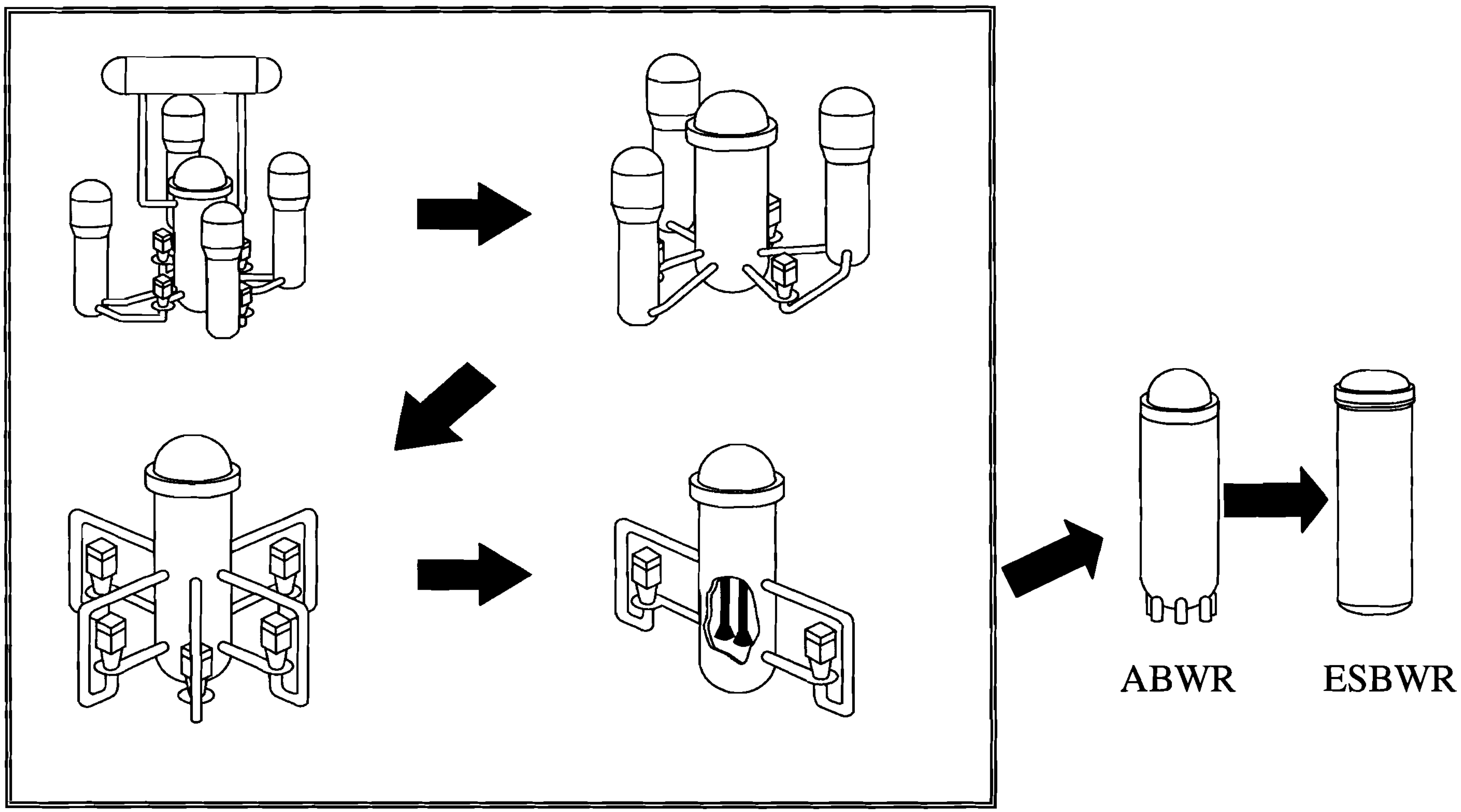
5

Outline

- ESBWR evolution
- Design summary
  - Design philosophy
  - Vessel and passive safety systems
  - Containment and buildings
- Features that improve plant performance
- Technology programs and methodology
- Summary and Conclusion

Pre-application review is a 12 month plan to close technology issues

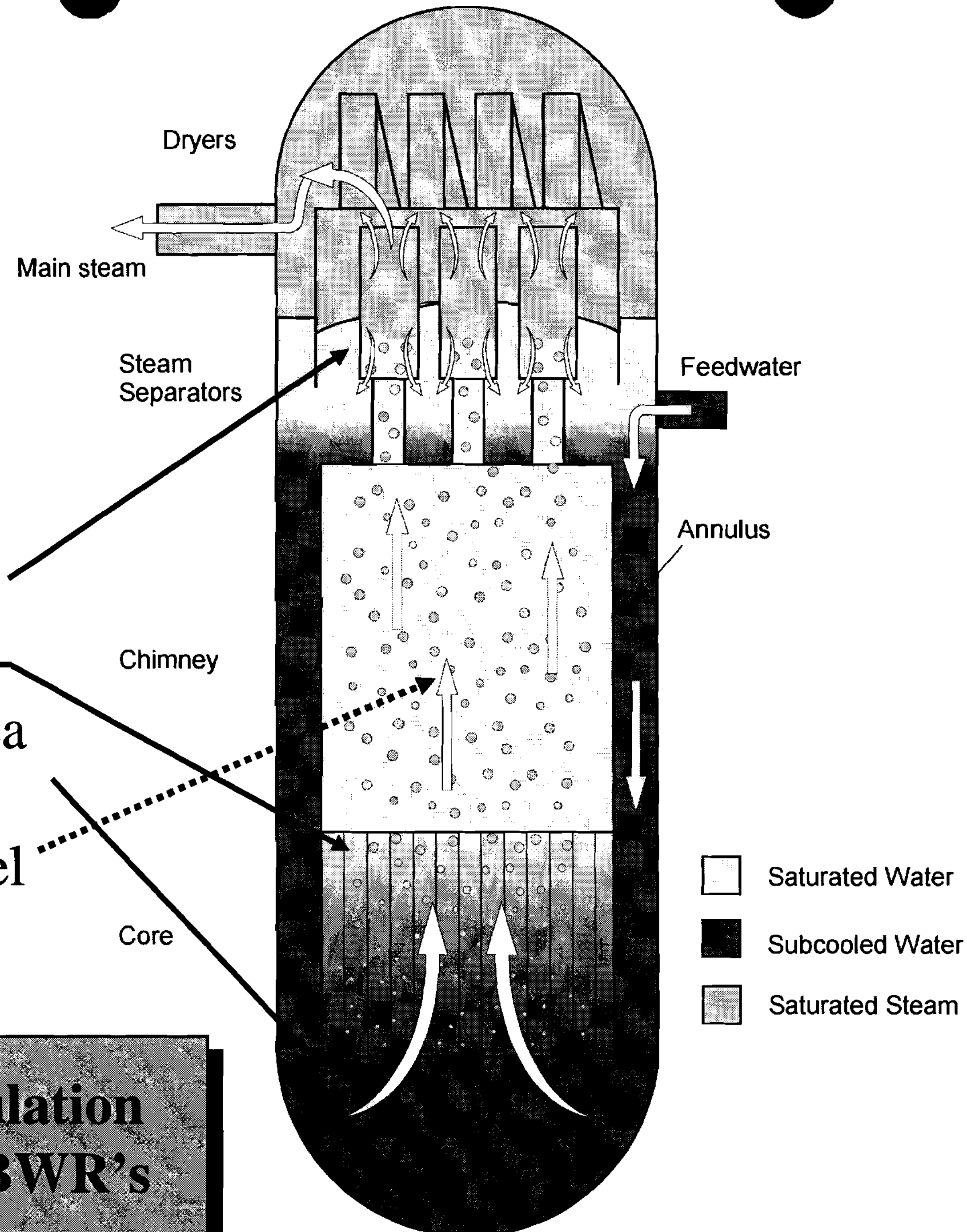
Evolution of the ESBWR Reactor Design



Evolution and Innovation Towards Simplicity

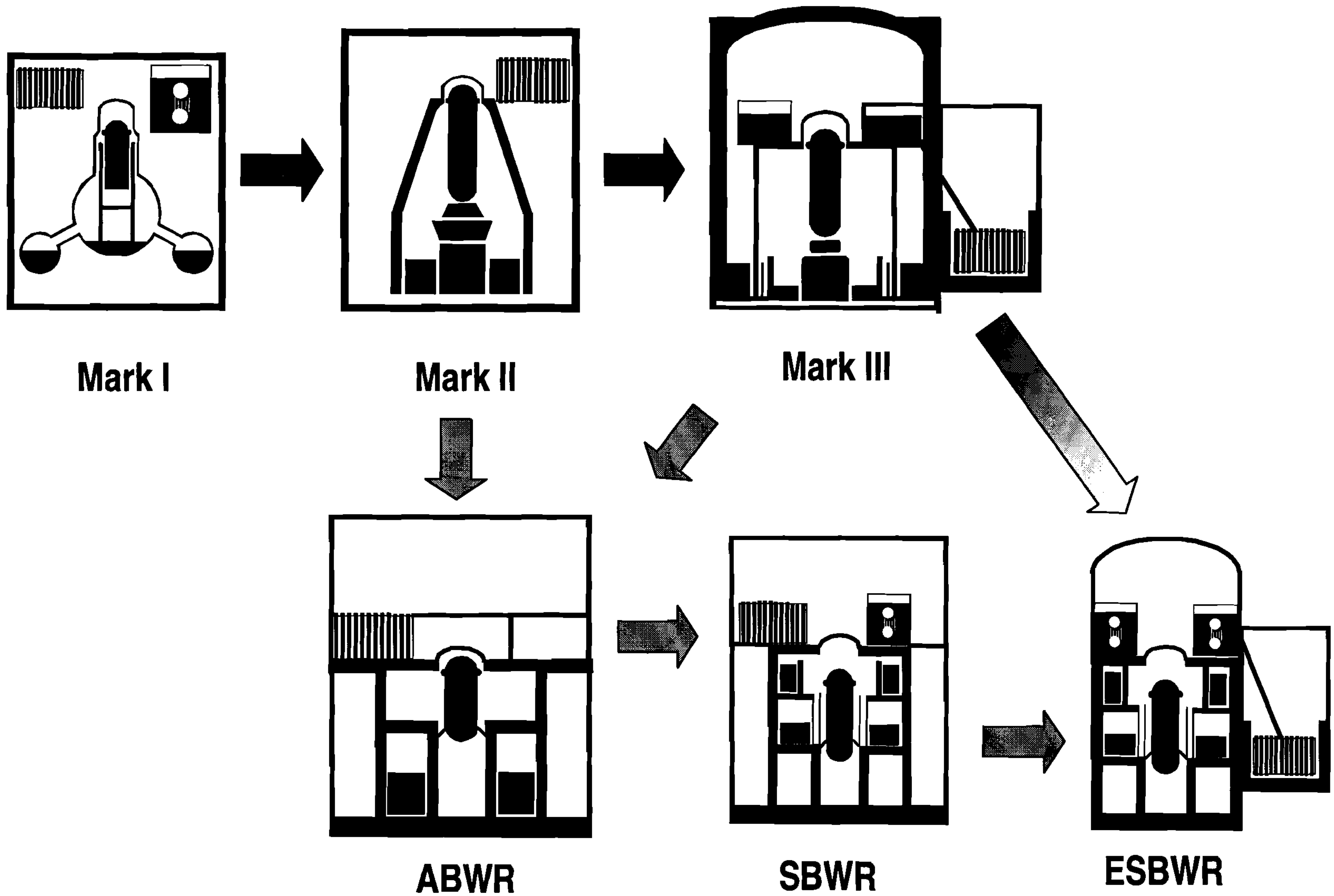
No pumps needed for normal operation

- Reduced flow restrictions
  - improved separators
  - shorter core
  - increase downcomer area
- Higher driving head
  - chimney and taller vessel



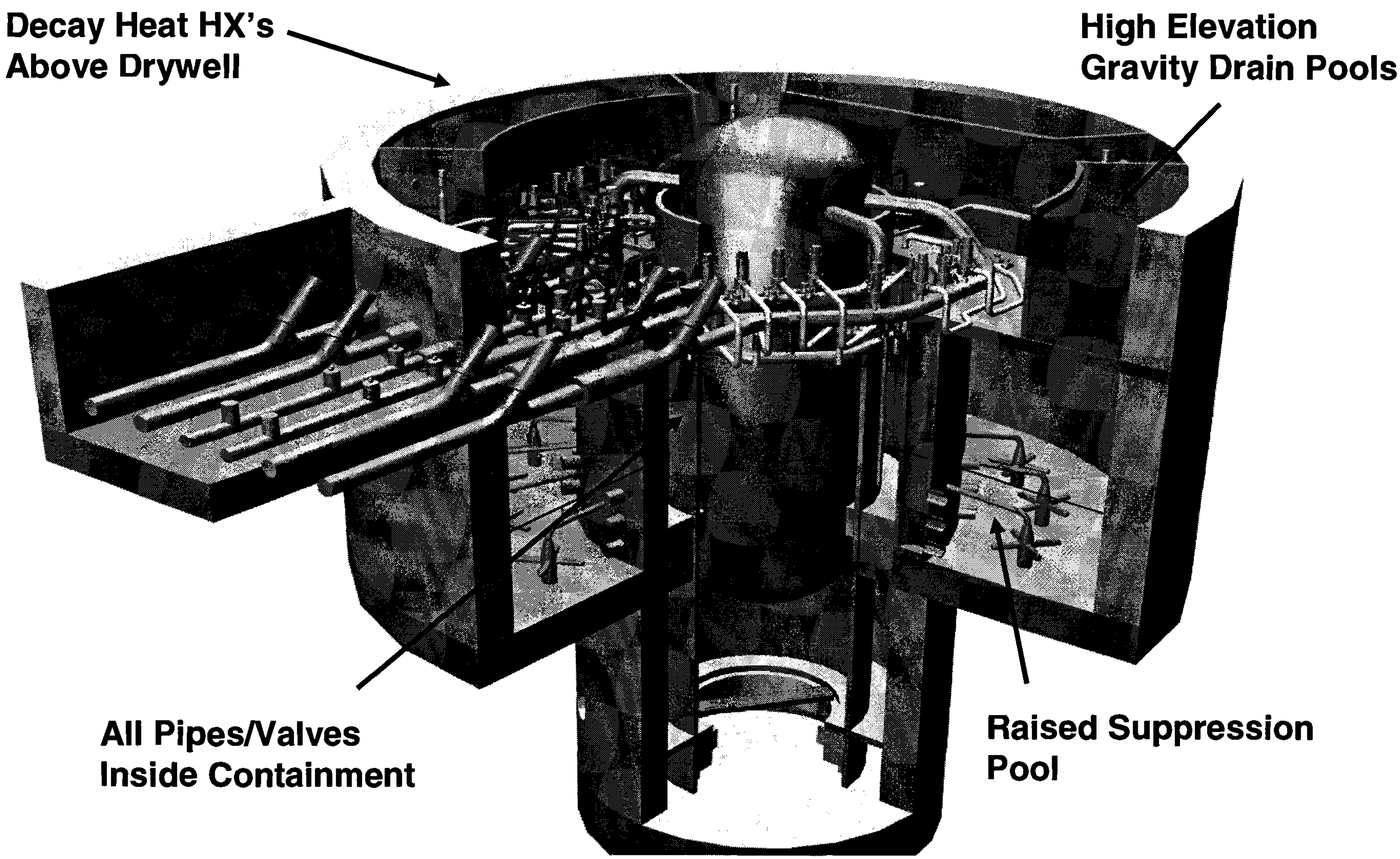
Enhanced Natural Circulation Compared to Standard BWR's

Evolution of BWR Containments



Evolution and Innovation Towards Simplicity

Passive Safety Systems Within Containment Envelope



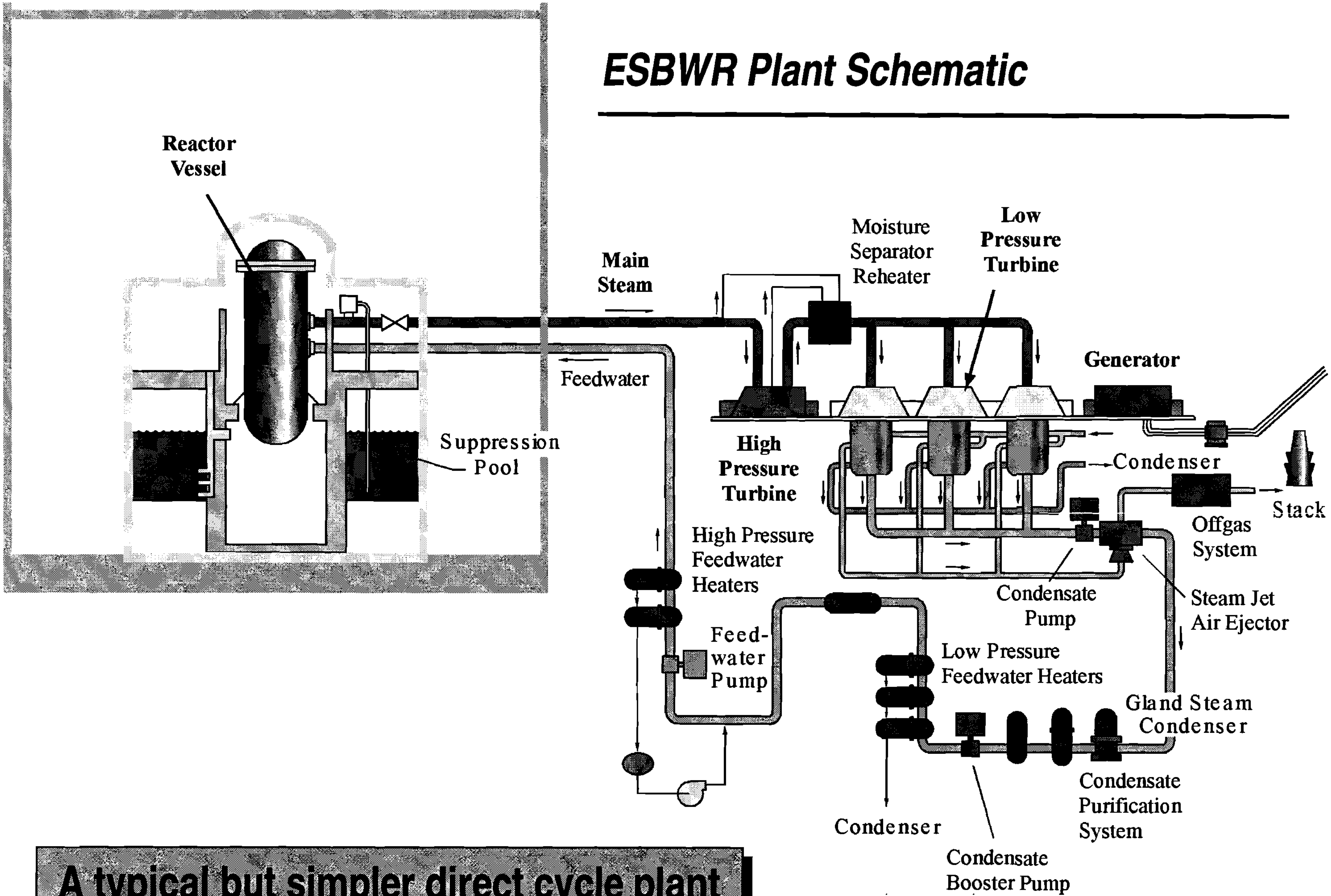
ar02-6

Comparison of key ESBWR parameters to operating BWRs

Parameter	BWR/4-Mk I (Browns Ferry 3)	BWR/6-Mk III (Grand Gulf)	ABWR	ESBWR
Power (MWt/MWe)	3293/1098	3900/1360	3926/1350	4000/1390
Vessel height/dia (m)	21.9/6.4	21.8/6.4	21.1/7.1	27.7/7.1
Fuel Bundles (number)	764	800	872	1020
Active Fuel Height (m)	3.7	3.7	3.7	3.0
Power density (kw/l)	50	54.2	51	54
Recirculation pumps	2(large)	2(large)	10	zero
Number of CRDs/type	185/LP	193/LP	205/FM	121/FM
Safety system pumps	9	9	18	zero
Safety diesel generator	2	3	3	zero
Vessel pressure, Mpa	7.1	7.1	7.1	7.1
Safety Bldg Vol. (m^3/MWe)	115	150	160	70

Evolution within a small range minimizes operational risks

ESBWR Plant Schematic



A typical but simpler direct cycle plant

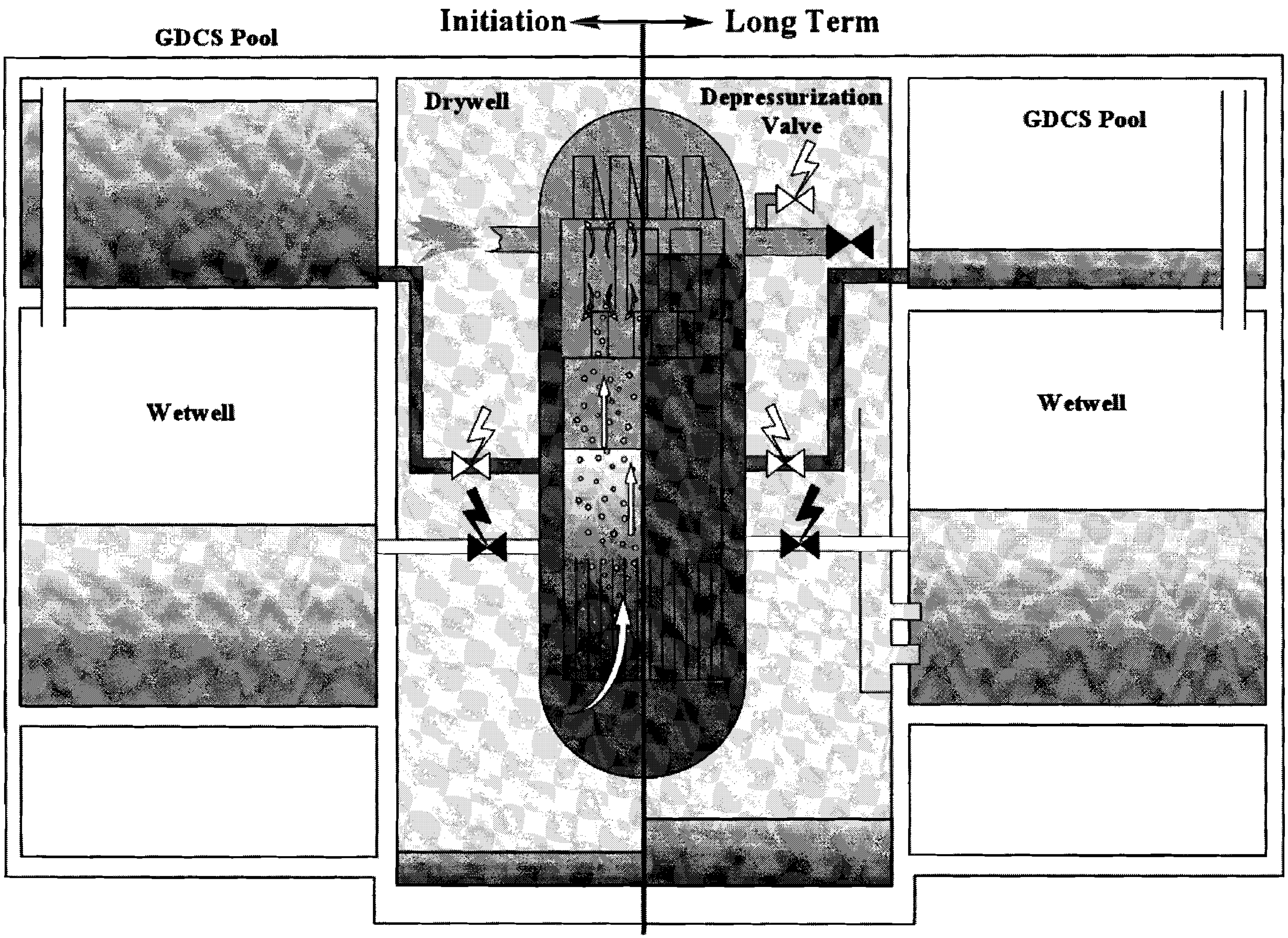
AR0103- 8

Design philosophy for core cooling

- ***Increase inventory in the vessel***
  - **Use taller vessel - NEW**
  - **Increase amount of subcooled water - NEW**
- ***Minimize inventory loss from the vessel***
  - **Eliminate large pipes below the core and minimize other pipes - NEW**
- ***Keep core covered after initial blowdown***
  - **Shorter core lower in the vessel - - NEW**
- ***Provide inventory makeup – low head using gravity***
  - **Provide diverse depressurization system for high reliability - NEW**
  - **Required makeup rate is very low**
    - Multiple tanks rely on gravity
  - **No high capacity systems needed**
  - **Fewer systems interactions**
- ***Utilize improved BWR analyses tools* - NEW**

Design features improved the plant response

Gravity Driven Cooling System (GDCS) - Main Steam Line Break



ar02-10

Design Philosophy for decay heat removal

- ***Remove Decay Heat From Vessel***
  - **Main Condenser**
  - **Normal shutdown cooling system – a full pressure system - NEW**
  - **Isolation condensers - NEW**
  - **Remove vessel heat through relief valve opening**
- ***If Needed, Remove Heat From Drywell***
  - **Passive containment cooling (PCC) Hx (safety-grade) - NEW**
    - Always available and drywell/wetwell pressure difference drives the flow through the heat exchangers
    - Condensed steam returns to drywell/vessel, non-condensables collect in the wetwell airspace
    - No operator action needed for 72 hours
  - **Suppression pool cooling (non-safety)**

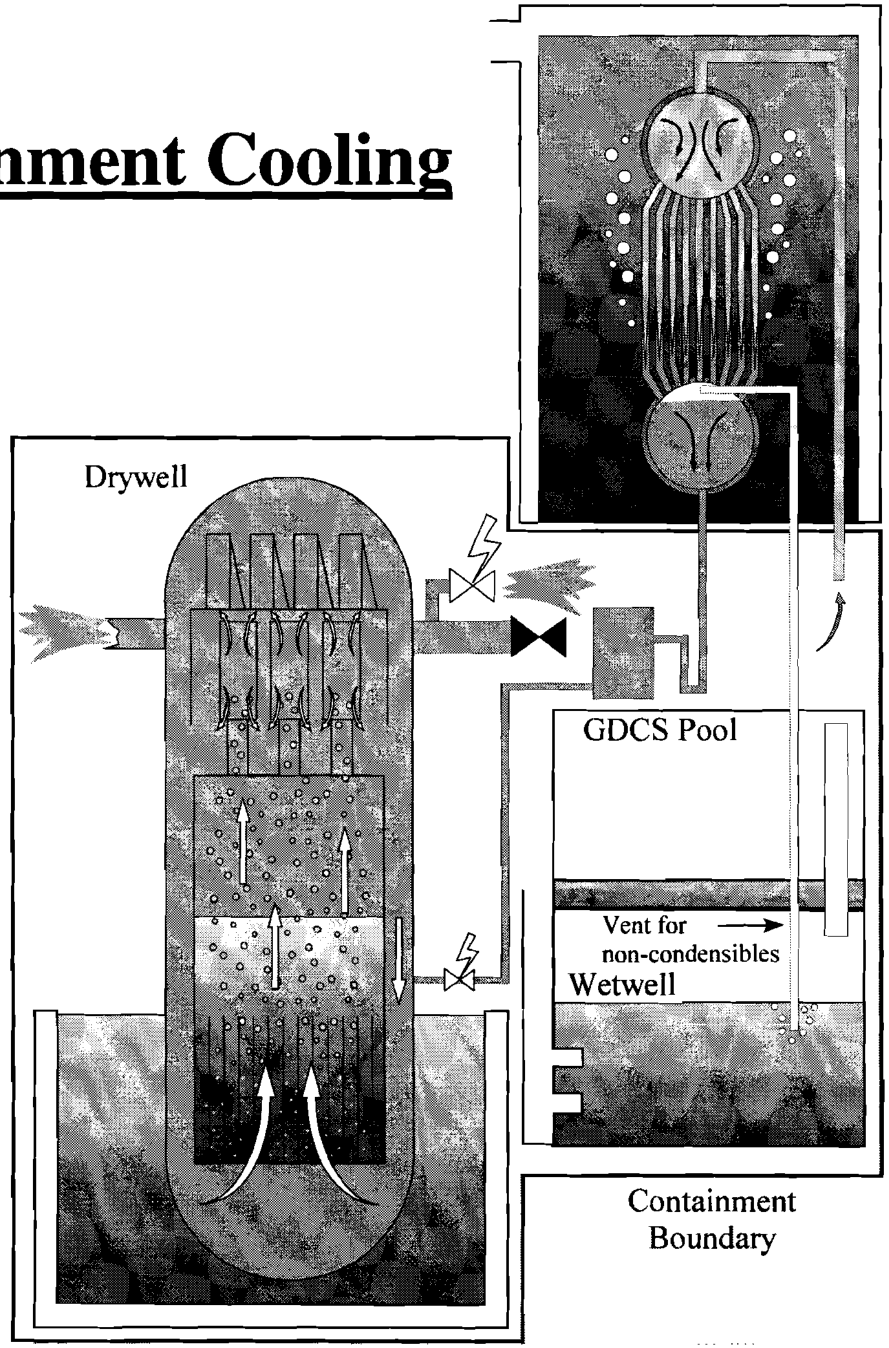
Several Diverse Means of Decay Heat Removal

Decay Heat Removal from Containment - How it works

- ***Initially steam (blowdown energy) flows to large heat sink in containment (suppression pool) and through heat exchangers***
- ***Longer term (decay heat) steam flows to heat exchanger (based on pressure differences) and heat is transferred outside containment***
  - ***Vertical tube heat exchangers in a pool of water***
- ***Containment pressure determined by non-condensables in wetwell airspace and vapor pressure***

Concept is simple, reliable - extensive testing and analysis provide high confidence in the design margin

Passive Containment Cooling



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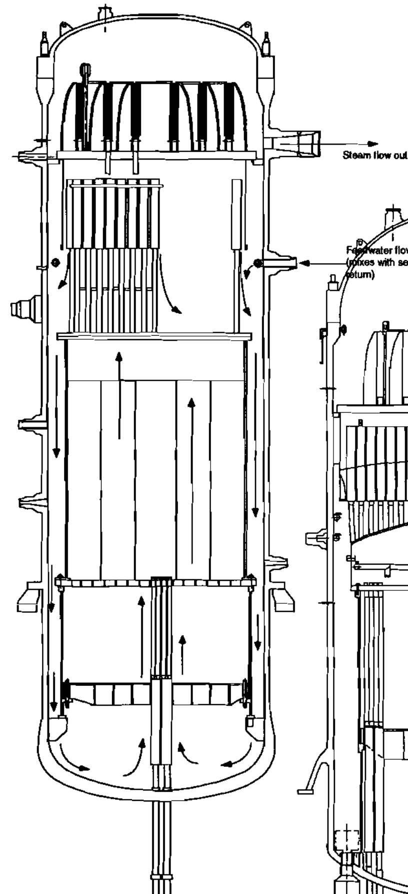
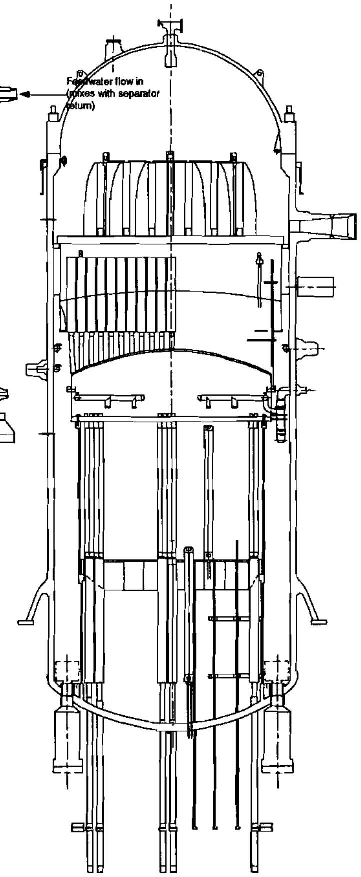
Design Features Affecting LOCA Response

	ESBWR	ABWR	BWR5	BWR4
Large pipes below core	No	No	Yes	Yes
Core height, m	3.05	3.66	~3.66	~3.66
TAF above RPV bottom	~ 1/4	~ 1/2	~1/2	~1/2
Separator standpipes	Long	Short	Short	Short
Vessel height, m	27.7	21.1	~21.9	~21.8
Water volume outside shroud (above TAF), m^3	**222**	**88**	**94**	**92**



ESBWR

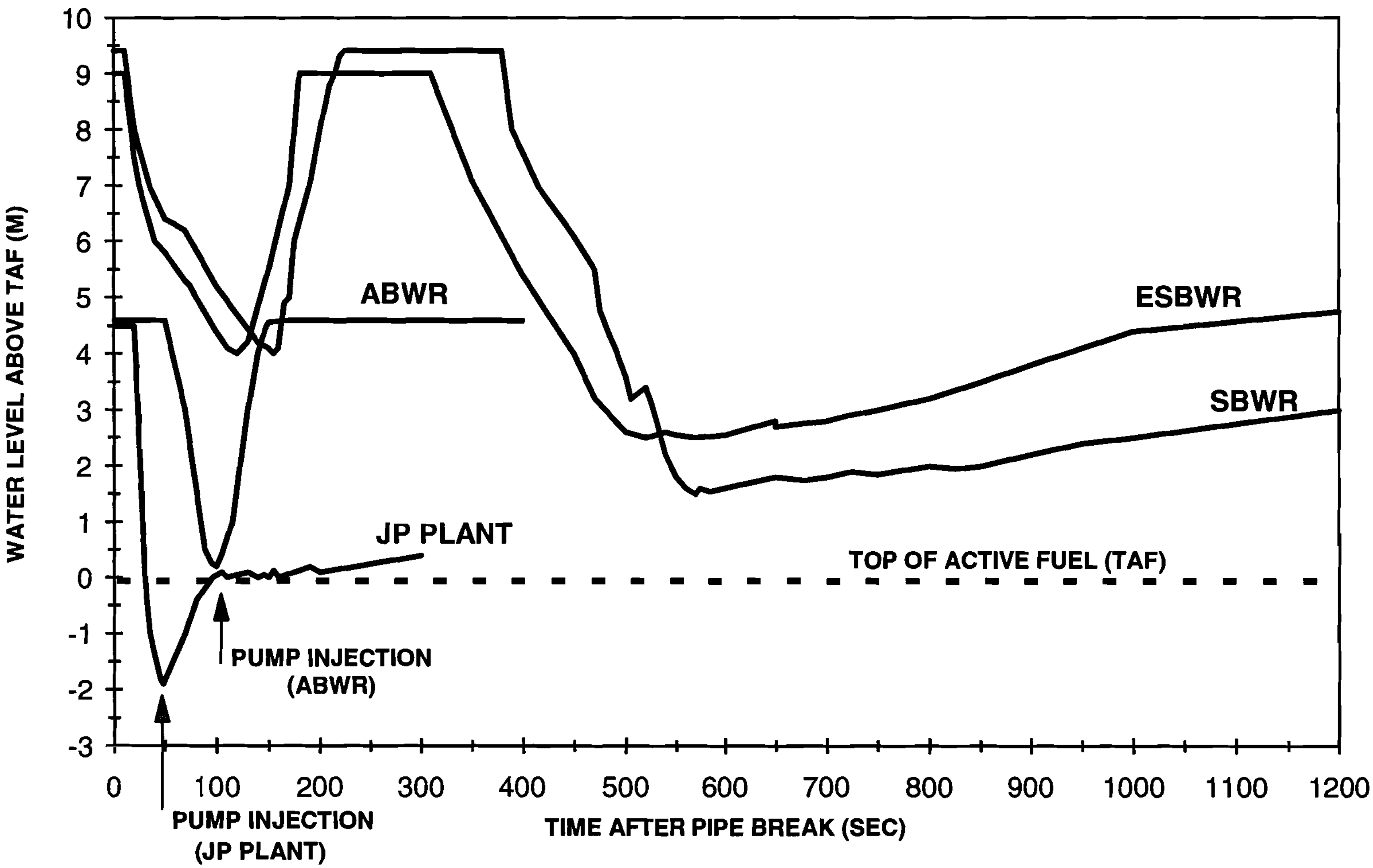


ABWR

ESBWR's greater water inventory results in improved plant LOCA performance

Water Level in Shroud Following a Typical Break

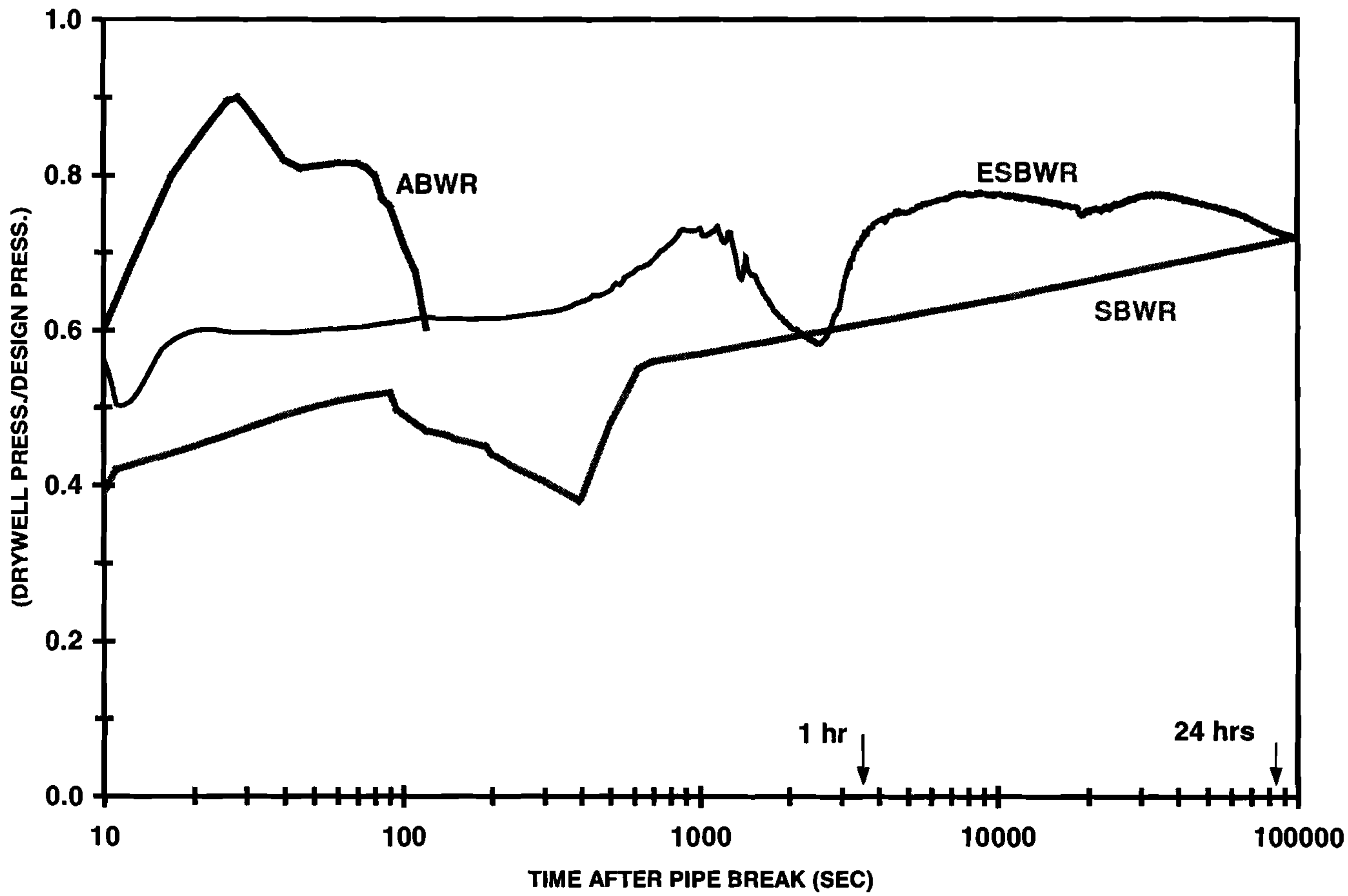
(values are intended to show typical trends for limiting breaks)



Margin to core uncovery – 3m ± 0.3m

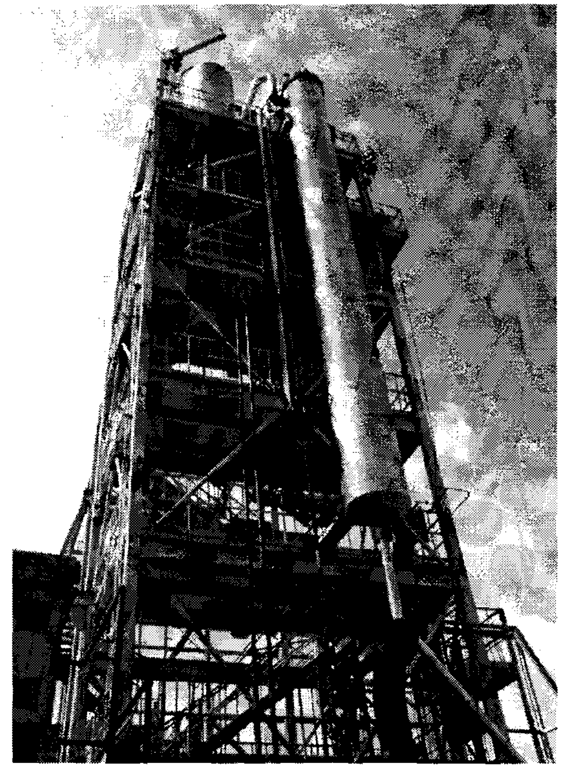
Containment Pressure Following a Pipe Break

(values are intended to show typical trends for limiting breaks – ESBWR has lower design pressure than SBWR)

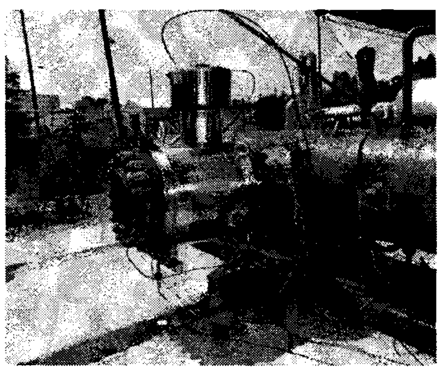


Large margin to design pressure

Technology Program for Features New to SBWR/ESBWR



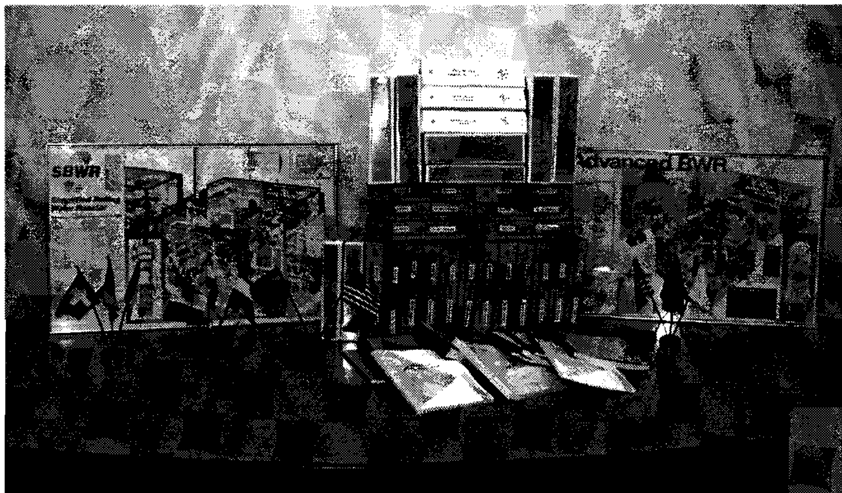
- ***Component tests***
  - Full scale components tests – DPV valves and vacuum breaker
  - Full scale isolation condensers & PCCS heat exchangers,
- ***Integral tests***
  - Integral tests at different scales – 1/400 to 1/25
  - System interaction tests
  - Large hydrogen releases
- ***Testing used to qualify computer codes***
- ***Extensive international cooperation***
- ***Extensive review and participation by NRC staff***
  - Test matrix
  - Running of actual tests
- ***Decay Heat Removal – additional ESBWR tests***
  - 8 Integrated system tests run in PANDA



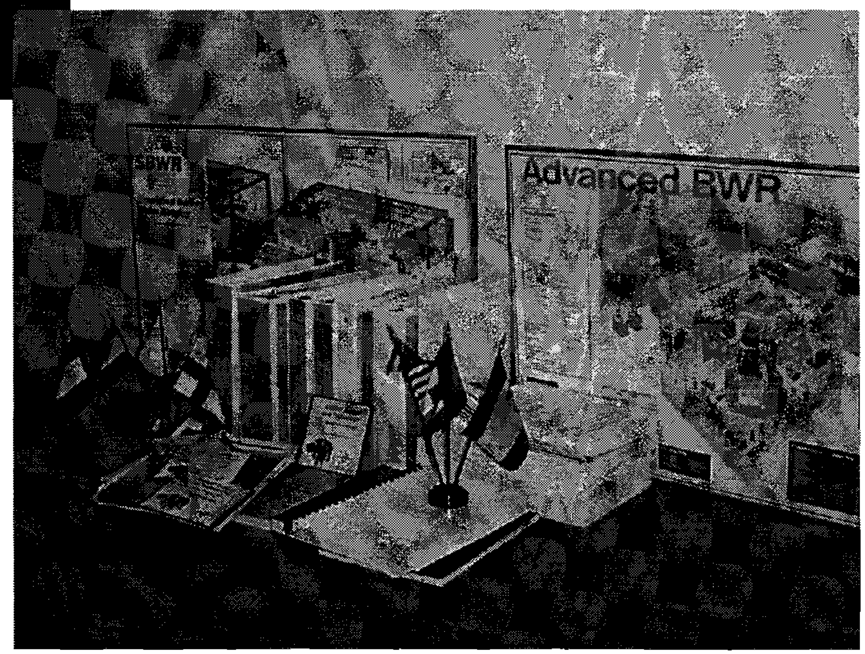
Reactor Depressurization Valve in the Test Facility

A complete, multi-year technology program supports the design

ESBWR Design/Technology based on SBWR and ABWR

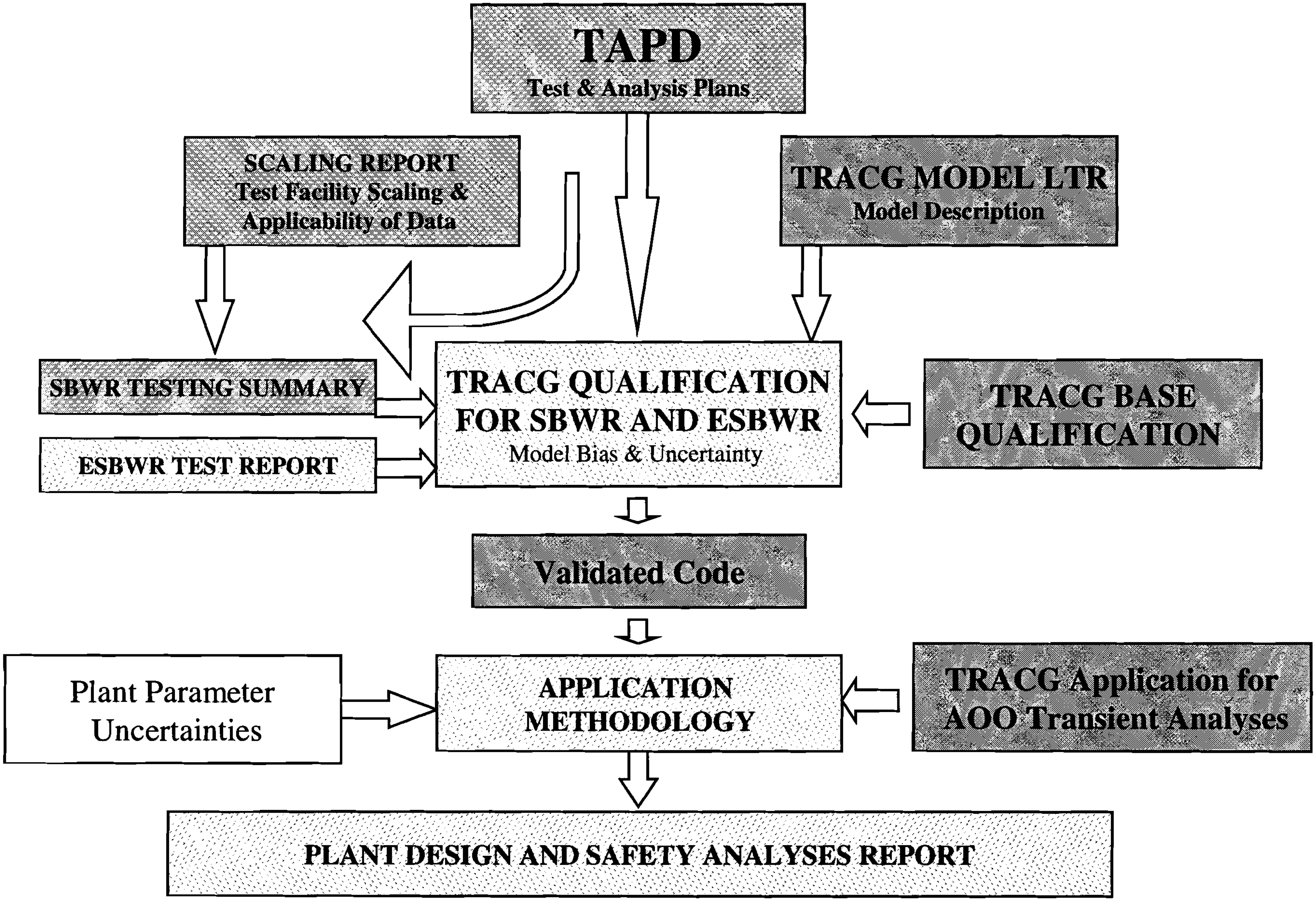


Extensive new submittals



Extensive SBWR submittals and reviews, new test data and reports, coupled with design changes to add margin

ESBWR Technology Program Elements



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Summary and Conclusions

- ***Passive safety systems have simplified the plant design***
- ***Plant evaluations are simpler***
  - Less complex analyses
  - Low parameter uncertainty - $\pm$ 0.5C for PCT!
- ***Substantial margins exist in the design***
  - Improved mechanistic codes show better performance
  - Defense in depth systems provide additional back-up
- ***Extensive qualification of TRACG***
- ***Technology issues extensively studied***
  - Independent studies provide confidence in technical bases

Performance improved by design features
Improved performance measured by qualified methods

FRAMATOME ANP

6

SWR 1000 Design Overview

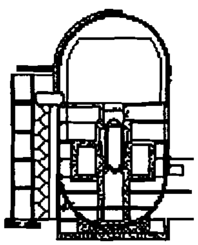
Roger Stoudt

October 10, 2002
Advisory Committee on Reactor Safeguards
Rockville, MD

FRAMATOME ANP

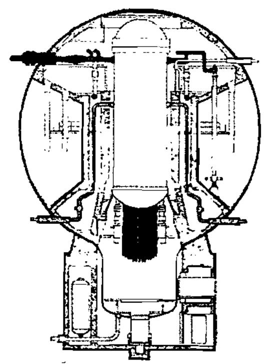
Evolution of Framatome ANP's BWR Technology

- Kahl
- Gundremmingen A
- Lingen (1st Fine Motion CRD - 1968)



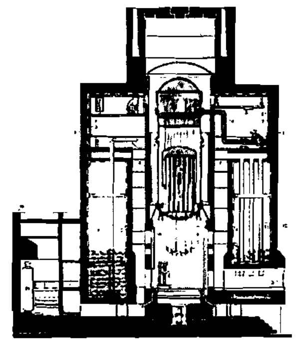
Full pressure containment - 61

- Würgassen
- Brunsbüttel (1st Internal recirc pump - 1977)
- Philippsburg 1
- Isar 1
- Tullnerfeld
- Krümmel

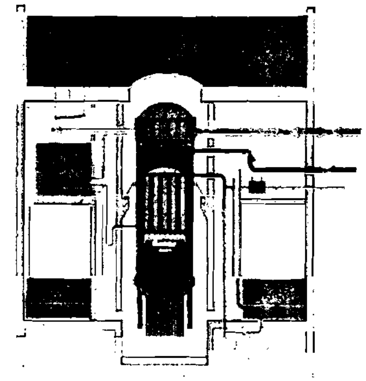


Product Line 69

- Gundremmingen B/C (3 train RHR & prestressed concrete containment - 1984/85)



Product Line 72



SWR 1000

FRAMATOME ANP

SWR 1000 Plant Parameters

> Thermal Power	3370 MW
> Electric Net Power	1253 MW
> Number of 12x12 fuel elements	664
> Inner Diameter of RPV	7.12 m (23.4 ft)
> Fuel Element Active Length	3.0 m (9.84 ft)
> Number of control rods	157
> Number of main recirculation pumps	8
> RPV pressure	75 bar (1088 psia)
> Number of Safety Relief Valves	8
> Emergency Condenser (EC) Capacity	4 x 66 MW
> Containment Cooling Condenser	4 X 4.8 MW
> Number of Passive Flooding Systems	4
> Containment Diameter	32.0 m (105 ft)
> Maximum Containment Pressure	7.9 bar (115 psia)

FRAMATOME ANP

Safety Approach

> All active systems have passive safety-related backup to perform nuclear safety functions

> SWR 1000 defense-in-depth design incorporates safety-related passive systems that are designed to meet all nuclear safety criteria without reliance on active systems

FRAMATOME ANP

SWR 1000 Passive Safety Concept

Containment cooling condensers for containment heat removal (4)

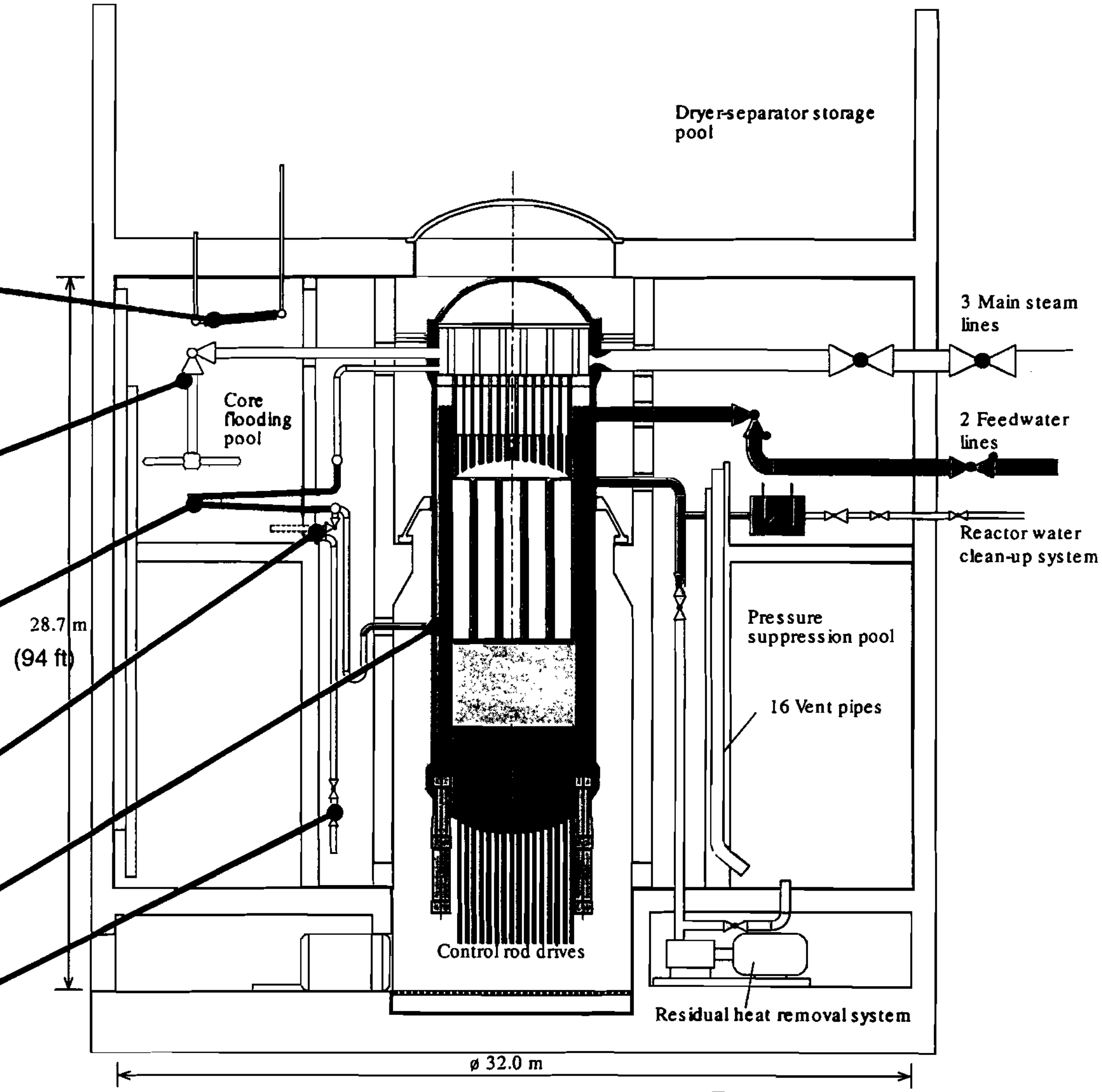
SRV for reactor over-pressure protection and reactor depressurization (8)

Emergency condensers for heat removal from the RPV (4)

Flooding lines for passive core flooding in the event of LOCA (4)

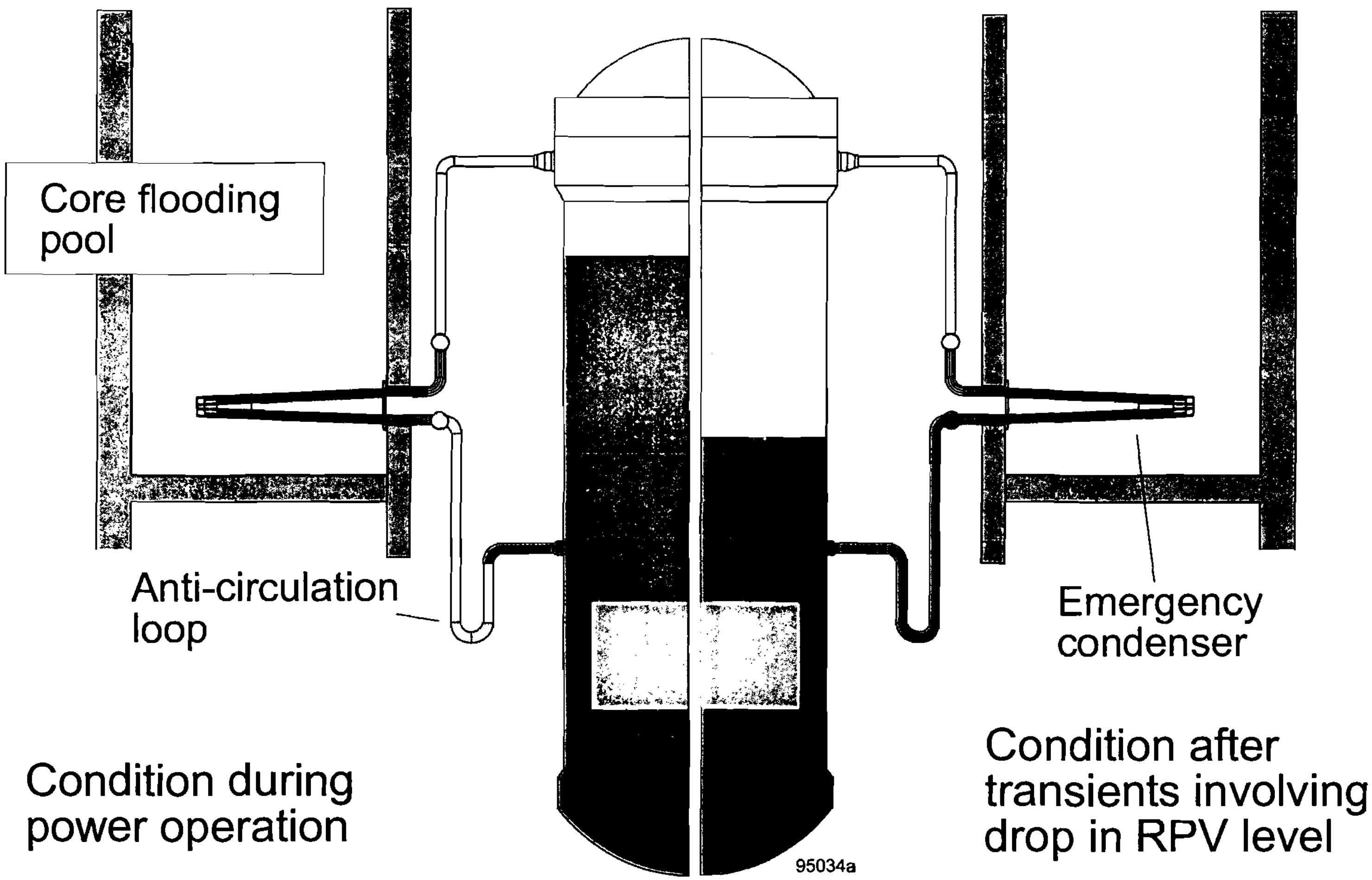
Passive Outflow Reducer (4)

Drywell flooding line (event of a core melt accident)



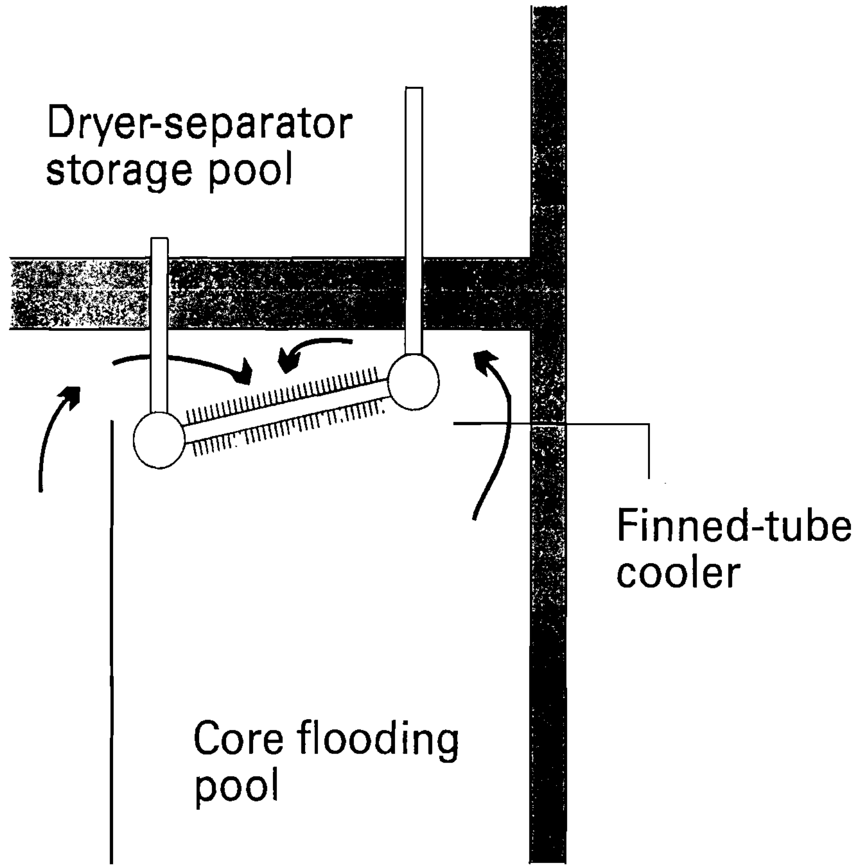
FRAMATOME ANP

Passive Safety Systems: Emergency Condenser



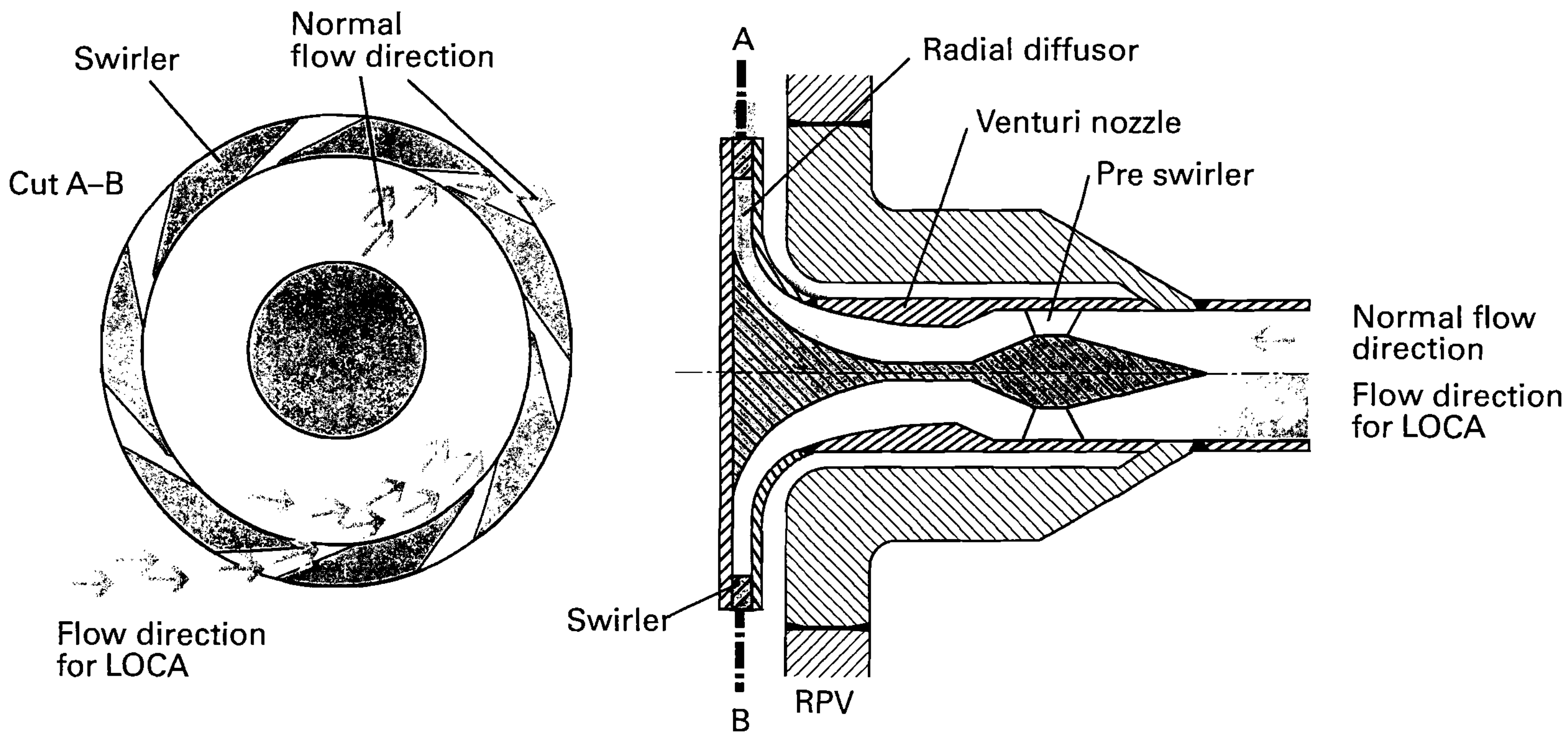
FRAMATOME ANP

Passive Safety Systems: Containment Cooling Condenser



Containment Cooling Condenser

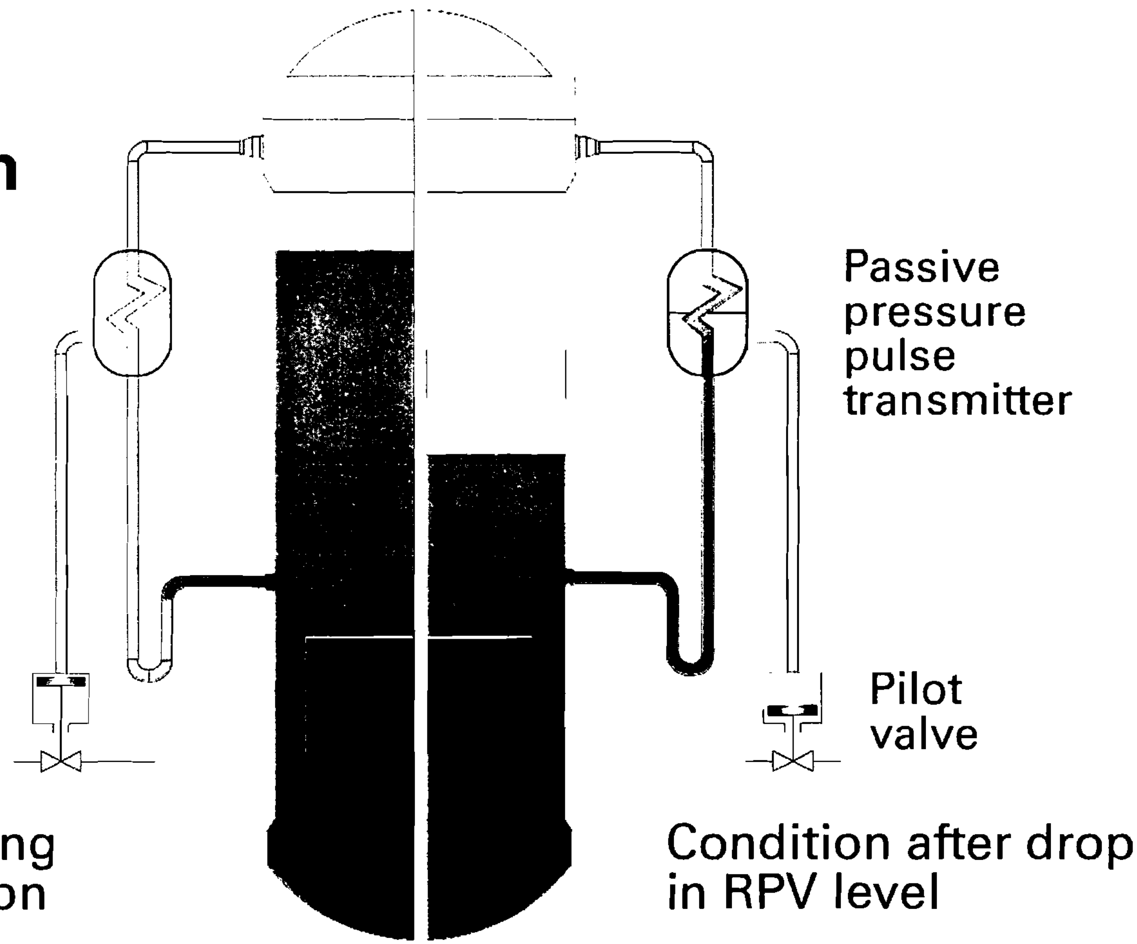
Passive Outflow Reducer



FRAMATOME ANP

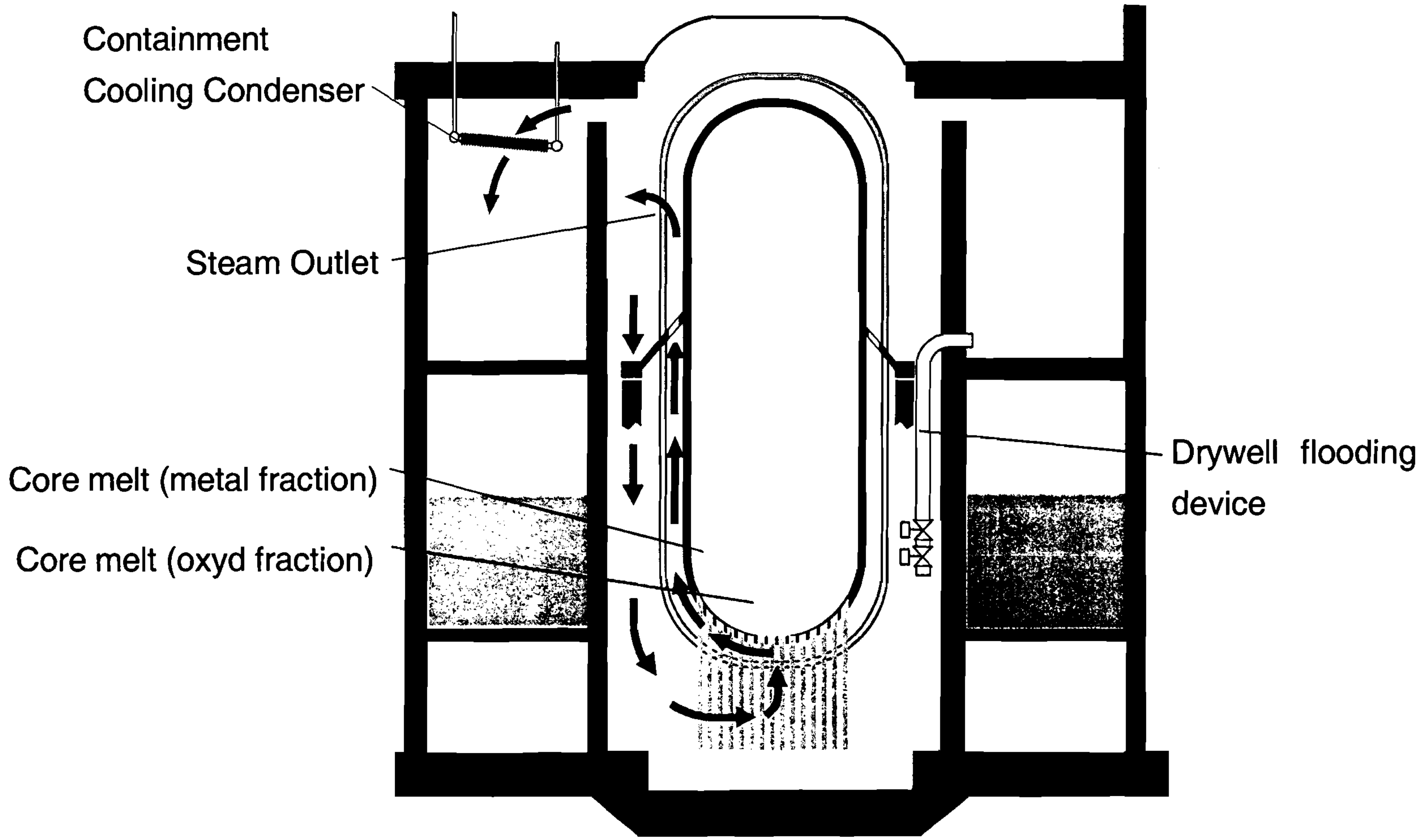
Passive Pressure Pulse Transmitter: Passive Safety System Actuation Device (Patented)

PPPT actuates
- Reactor Scram
- MSIV
- Depressurization



FRAMATOME ANP

Severe Accident Control
Core Melt Retention in the RPV



FRAMATOME ANP

Testing

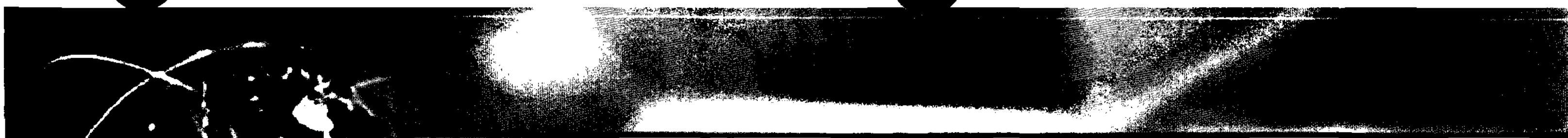
> Tests Performed
- Emergency Condenser (EC)
- Containment Cooling Condenser (CCC)
- Passive Pressure Pulse Transmitter (PPPT)
- Passive Outflow Reducer (POR)
- RPV Flooding Line
- Reactor Pressure Vessel Exterior Cooling
- CONGA - CCC heat transfer in presence of aerosols
- SCRAM Tank

> Future Tests
- Fast Acting Boron Injection System
- Spring Support Check Valve (RPV Flooding Line)
- Vent Pipes and Quenchers
- Control Rod Drives

Summary

> Important SWR 1000 Features
- Large water inventory inside the RPV
- Large water inventories inside the containment for heat storage and flooding
- Nitrogen-inerted containment atmosphere
- Passive equipment for heat removal from the RPV and containment
- Passive actuation of key safety functions
- Passive, external cooling of the RPV and melt retention within the RPV in the case of severe accidents

> In the event of transients or LOCAs and utilizing only passive systems, stable conditions can be established without outside intervention of personnel for several days.



Advanced CANDU Reactor (ACR™)

Presentation to the Advisory Committee on Reactor Safeguards October 2002

Dr. V.G. Snell
Director, Safety & Licensing
ACR

AECL
TECHNOLOGIES INC.

2

Advanced CANDU Reactor (ACR™)

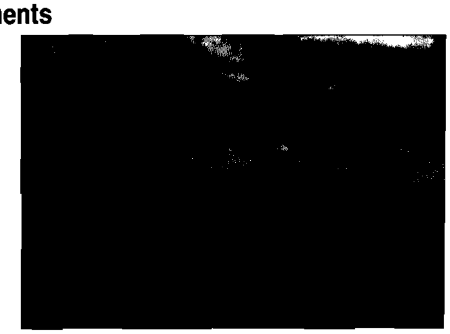
Presentation to the
Advisory Committee on Reactor Safeguards
October 2002

Dr. V.G. Snell
Director, Safety & Licensing
ACR

AECL TECHNOLOGIES INC.

Outline

1. What is ACR?
2. Meeting Customer Requirements
3. Technical Summary
4. Safety Improvements
5. Technology Base
6. Status
7. Licensing Opportunities
8. Conclusions



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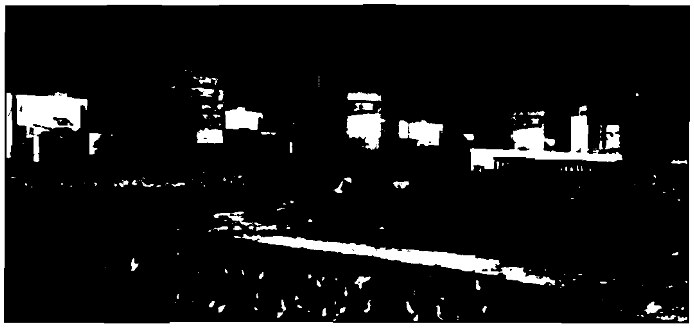
1. What is ACR?

- The ACR is an evolutionary extension of the proven CANDU 6, which has eight units in operation on four continents, two units currently under construction, and one which went critical in September 2002



2. Meeting Customer Requirements

Specific overnight capital cost:	$1,000/kWe
Construction schedule:	36 months
LUEC:	$30/MWh
Capacity factor:	>90%
Plant Operating Life:	60 years

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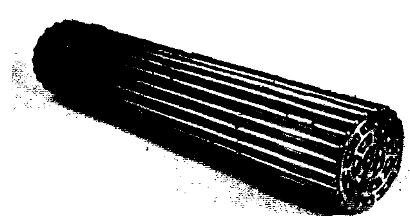
Achieving Low Capital Cost

- Current operating CANDUs:
  - Natural uranium fuel
  - Heavy water (D_2O) coolant
  - Heavy water (D_2O) moderator
- ACR - relax constraint of Natural Uranium Fuel and --
  - Use light water coolant
  - Reduce core size & reduce amount of heavy water moderator
  - Increase pressure tube thickness
    - Increase reactor coolant system (RCS) pressure
    - Increase thermal efficiency
- Retain intrinsic proven CANDU features

3. Technical Summary - Fuel

- 0.5m (1.6 foot) long CANFLEX fuel bundle
- On-power refueling
- 43 fuel rods
  - 2.0 wt% ^{235}U SEU in 42 rods
  - NU + 4% dysprosium in central rod
- Fuel burn-up 20,500 MWd/MT (U)
  - higher than NU CANDU average
  - modest vs. LWRs
- Higher bundle power, lower rod rating



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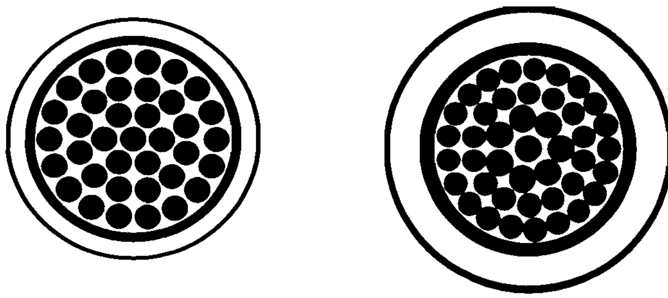
Fuel Channel

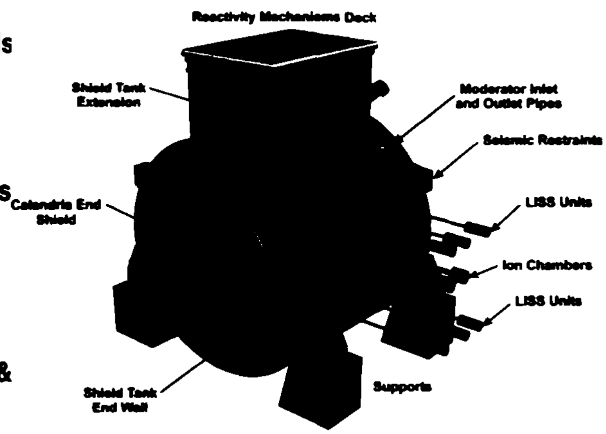


- 37-rod NU fuel
- Zr – 2.5% Nb pressure tube
- Zr-2 calandria tube
- Insulating gap between pressure tube and calandria tube

- 43-rod SEU fuel
- Thicker Zr – 2.5% Nb pressure tube
- Stronger Zr-4 calandria tube
- Larger gap between pressure tube and calandria tube

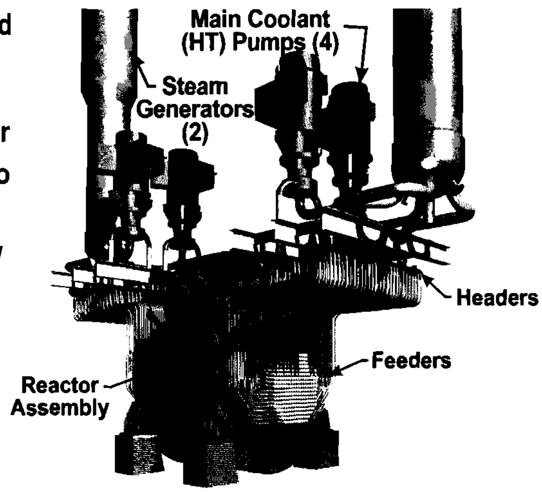
Reactor

- Horizontal fuel channels surrounded by low temperature, low pressure moderator
- Steel calandria contains moderator & supports fuel channels
- Shield tank surrounds calandria and contains light water for thermal & biological shielding
- All reactivity devices in moderator



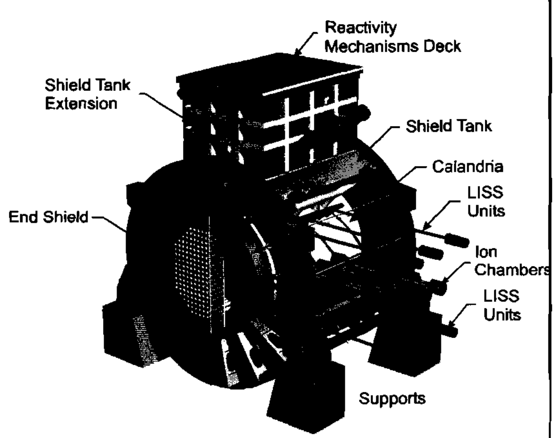
Reactor Coolant System

- Each channel is connected at its inlet and outlet by small (feeder) pipes to headers, above the reactor
- Above headers – similar to PWRs
- No large pipes at or below core level
- Tolerates pump seizure
- Natural circulation, even with some void



Safety Systems

- Two fully independent shutdown systems
  - SDS1 – rods drop in moderator
  - SDS2 – liquid absorber injected into reflector
- Two stage Emergency Core Cooling System
  - Initial injection from pressurized tanks
  - Long term pumped recovery
- Steel-lined dry pressure containment



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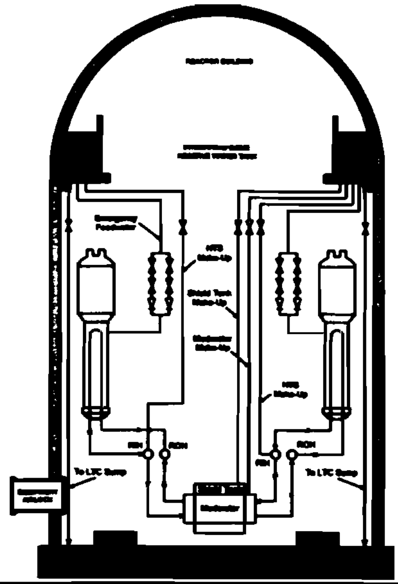
Severe Accident Resistance

Elevated Reserve Water Tank can add water by gravity to:

- Reactor coolant system
- Steam generators
- Moderator
- Shield tank

Moderator can remove decay heat from fuel channels without UO_2 melting

Shield tank can slow down or arrest graceful severe core damage progression



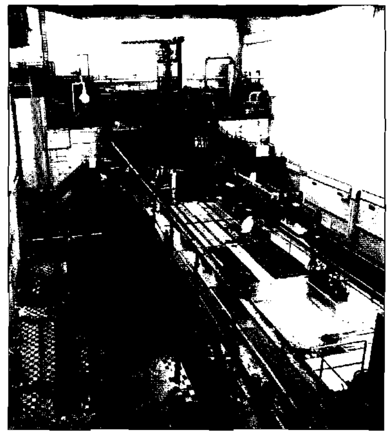
4. Safety Improvements

- Small negative void coefficient
- More negative power coefficient over operating range
- Larger thermal margins due to CANFLEX fuel
- Pressure-tube failure contained within calandria tube
- Improved heat sink reliability
- Inter-unit ties enhance reliability of safety support systems
- Inherent shutdown on single channel failure
- Steel-lined dry containment
- Extended seismic qualification
- Severe accident prevention & mitigation
- Design insights from generic CANDU PRA; ACR design-assist PRA

5. Technology Base

- ACR is an evolutionary improvement of operating CANDUs
- CANDU technology base developed & maintained by AECL & CANDU utilities
  - 2000 people at Chalk River Laboratories
- ACR R&D is anticipatory
  - Modest extension of databases to ACR conditions
  - Confirm code validity
  - Confirm performance of modified components

NRU Reactor – Fuel, Materials & Safety Tests



Anticipatory R&D for ACR

- Fuel
- Fuel channel
- Fuel handling
- Components
- Safety code qualification

Moderator test facility

ZED-2 Reactor

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6. Status

- ACR-700 concept complete (reference design); ACR-1000 being considered
- Non-site-specific engineering complete 2005
- Hitachi investing in BOP optimization and plant-wide modularization
- Construction strategy and schedule defined
- Working with Canadian, US and UK utilities to bring ACR to commercialization

Status – Licensing

- Pre-application review started with USNRC; expect 2 years
  - Application for Standard Design Certification and/or COL
- Pre-licensing review also started in Canada to confirm licensability under Canadian regulations
- Possibility of pre-licensing review in UK

7. Licensing Opportunities

- Use of extensive Canadian regulatory, R&D & operating experience in NRC review
  - Acceptance of equivalence in meeting safety requirements
- Flexibility of NRC requirements to accommodate a technology both similar to and different from LWRs
- Co-operation with parallel regulatory reviews in Canada and possibly UK
  - Extent of common ground & consistency

8. Conclusions

- ACR is an evolutionary design building on proven CANDU 6 design and operation
- ACR meets the market economic, schedule and risk requirements
- SEU CANFLEX fuel contributes to improvements in both economics and safety
- R&D is anticipatory; modest extension of conditions and components
- NRC review requirements and processes could take advantage of prior CANDU licensing experience, and parallel reviews in Canada (& possibly UK).

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UNITED STATES NUCLEAR REGULATORY COMMISSION

Technical Related Policy Issues for Future Non-Light Water Reactors

Presentation to ACRS-Full Committee
October 10, 2002
T. L. King, NRC/RES
301-415-6345

Objective of Briefing

- To discuss the schedule and options for resolution of the seven policy issues for future non-LWRs described in SECY-02-0139:
  - Expectations for safety
  - Defense-in-depth
  - Use of international codes and standards
  - Event selection
  - Source term
  - Containment vs. confinement
  - Emergency preparedness
- To solicit verbal feedback from the Committee regarding the options, including advantages and disadvantages, and to discuss plans for future meetings with ACRS

Background

- Current regulations are a combination of generic and LWR oriented requirements

- Previous licensing of non-LWR designs was based upon:
  - A review of the design against the regulations current at that time
  - A case-by-case determination regarding the applicability of the regulations
  - The need for additional requirements to address any unique aspects of the design/technology

- Pre-application reviews are an opportunity for early review and guidance on licensing/safety issues

Background

Continued

- **Useful to get Commission guidance early:**
  - To support case-by-case reviews
  - To support development of a generic approach (framework)
- **Pre-application work to date on PBMR and GT-MHR has identified technical issues with potential policy implication for non-LWRs**
- **Some of these issues had been raised in previous pre-application reviews (e.g., MHTGR)**
- **Scope of issues**
  - Reactor design
  - Reactor operation

Schedule

- Public Workshop
  - October 22-23, 2002
  - Doubletree Hotel, Rockville
- ACRS
  - November/December
  - Subcommittee/Full Committee
- Paper due to Commission
  - December 30, 2002

Expectations for Enhanced Safety

- **Issue**: How to implement the Commission's expectations for enhanced safety (as expressed in the Commission's Policy Statements on Advanced Reactors and Severe Accidents)
- **Options**:
  - Require current level of safety
    - With expectation that applicants will provide enhanced safety
  - Require enhanced level of safety
    - e.g., more stringent CDF
  - Require enhanced level of confidence
    - e.g., additional testing, additional oversight
  - Encourage industry to implement enhanced safety

Expectations for Enhanced Safety

Continued

- **Key Considerations:**
  - Additional reactors
    - Per site
    - Nationwide
  - Safety Goal Policy
    - Risk to individuals around a plant vs. site?
  - Performance Goal
    - Maintain safety - impact of more plants nationwide on performance measures?
  - Role of enhanced accident prevention in compensating for larger uncertainties in severe accident area?
  - Implications for future LWRs?

Defense-in-Depth (DID)

- **Issue: How to specify DID for non-LWRs**
  - Mentioned in Commission policies, but no articulation as to the elements of DID
  - Commission definition provided of DID in 1999 RIPB regulation white paper
  - IAEA and INSAG have description of DID

- **Options:**
  - Case-by-case determination, depending upon:
    - Plant design
    - Uncertainties
  - Develop description or policy statement articulating the elements of DID
  - Develop description or policy statement articulating DID as programmatic process

Defense-in-Depth (DID)

Continued

- **Key Considerations:**
  - Scope of DID?
    - Programmatic vs. physical elements
    - Reactor design vs. Other factors
  - RROP Cornerstones?
  - Foundation for future licensing framework?
  - Guidance for areas other than licensing e.g.:
    - Reg Analysis Guidelines?
  - Implications for future LWRs?
  - Coordination with non-reactor activities?

International Codes and Standards

- **Issue:** How should NRC requirements for non-LWRs relate to international safety standards and requirements?

- **Options:**
  - No specific initiative
    - Review on an as necessary basis as part of an applicant's licensing submittal
  - Review and endorse existing codes and standards, whenever practical
  - Participate in the development of codes and standards and endorse, whenever practical
  - Attempt to harmonize requirements with other regulatory bodies

International Codes and Standards

Continued

- **Key Considerations:**
  - NRC Management Directive 6.5
    - Public Law 104- 113
    - Office of Management & Budget Circular A-119
  - International nature of future design efforts and marketing
  - Usefulness in compensating for areas where there are gaps in NRC expertise or infrastructure?

Event Selection

- **Issue:** **To what extent can a probabilistic approach be used to establish the licensing basis:**
  - Event selection?
  - Safety classification?
  - Replace single failure criterion?

- **Options:**
  - Use a deterministic approach, supplemented by PRA
  - Use a probabilistic approach
  - Use a probabilistic approach, supplemented by engineering judgement

Event Selection

Continued

- **Key Considerations**:
  - Previous Commission guidance
    - SRM of July 30, 1993
  - Probabilistic criteria for event categories?
  - Probabilistic criteria for safety classification?
  - Probabilistic approach to replace the SFC?
  - PRA quality, completeness, document control?
  - Level of confidence?

Source Term

- **Issue:** Under what conditions should scenario specific accident source terms be used for licensing decisions?

- **Options:**
  - Develop a deterministic bounding ST
  - Allow the use of scenario specific ST

Source Term

Continued

- **Key Considerations:**
  - Previous Commission guidance
    - SRM of July 30, 1993
  - Scenario specific approach may depart from practice where ST is based upon core melt
  - Role of robust ST in DID?
  - Scenario specific approach puts more burden on understanding plant, fuel and fission product behavior over the life of the plant
  - Level of confidence?

Containment vs. Confinement

- **Issue**: Under what conditions can a plant be licensed without a pressure retaining containment building?

- **Options**:
  - Require a pressure retaining building
  - Allow a design without a pressure retaining building

Containment vs. Confinement

Continued

- **Key Considerations:**
  - Previous Commission guidance
    - SRM dated July 30, 1993
  - Related to resolution of event selection and ST issue
  - Should a pressure retaining building be a fundamental element of DID?
  - Impact on safety?
  - What criteria should be met to allow a design without a pressure retaining building?

Emergency Preparedness

- **Issue:** Under what conditions can the EPZ be reduced, including a reduction to the EAB?

- **Options:**
  - No reduction from current requirements
  - Allow a reduction in the EPZ
  - Allow a graded approach within the EPZ

Emergency Preparedness

Continued

- **Key Considerations:**
  - Previous Commission guidance
    - SRM of July 30, 1993
  - Related to defense-in-depth
    - last line of DID
  - Related to resolution of event selection, ST and containment issue
  - What criteria would be used to reduce the EPZ?
  - Credit for long response time?

LOW POWER/SHUTDOWN SPAR MODEL DEVELOPMENT

UNITED STATES NUCLEAR REGULATORY COMMISSION

PATRICK D. O'REILLY (301-415-7570)
OPERATING EXPERIENCE RISK ANALYSIS BRANCH
DIVISION OF RISK ANALYSIS AND APPLICATIONS
OFFICE OF NUCLEAR REGULATORY RESEARCH

PRESENTATION TO ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

OCTOBER 11, 2002

6

OUTLINE OF PRESENTATION

- **Low Power/Shutdown (LP/SD) SPAR Model Development Program Plan.**

- **Onsite QA Review of LP/SD SPAR Model for Surry 1 & 2.**

- **Cancellation of Revision 4i SPAR Model Development.**

LOW POWER/SHUTDOWN (SDP/SD) SPAR MODEL DEVELOPMENT PROGRAM PLAN

BACKGROUND

<u>FY 1996:</u>

- Produced PWR (Surry) LP/SD SPAR Model:
    - Based on Detailed Surry Shutdown PRA Developed by NRC/BNL (NUREG/CR-6144).
    - Developed for use with DOS version of SAPHIRE.
    - Not user-friendly.
    - Not peer reviewed.
    - Adapted Human Reliability Analysis (HRA) methodology from full power (Revision 3) SPAR model development effort for use in LP/SD SPAR models.

BACKGROUND (Continued)

<u>FY 2001:</u>

- Produced BWR (Grand Gulf) LP/SD SPAR Model
  - Based on Detailed Shutdown PRA Developed by NRC/SNL (NUREG/CR-6143).
  - User-friendly.
  - Compatible with Windows-based SAPHIRE/GEM.
  - Internal peer review of model.

- Developed LP/SD SPAR Model Specification, Prototype Templates, and Associated Guidelines for Developing Other LP/SD Models
  - Received technical guidance from interoffice SPAR Model Users' Group (SMUG).
  - Determined usefulness of current LP/SD models originally developed for ASP Program for current applications.

BACKGROUND (Continued)

FY 2001:

- Reviewed LP/SD events analyzed in ASP Program to determine if model content was sufficient to address these event types; identified necessary changes.

- Met with SMUG and key model users to identify users needs and desired model characteristics.

- Developed and Demonstrated Prototype Templates to SMUG:

  - All PWRs.
  - BWR 5/6s.
  - BWR 4s.

LP/SD SPAR MODEL TEMPLATE FOR PWRs

- Starting Point for Developing a Plant-Specific LP/SD Risk Model That Includes Core Damage Risk Resulting from:
  - Loss of RHR events.
  - Loss of offsite power events.
  - Loss of inventory events.

- Essentially a Working LP/SD Model with No Plant-Specific Fault Tree Logic
  - Event trees generally applicable to all PWRs.
  - Some fault trees also generally applicable to all PWRs.
  - Remaining fault trees include undeveloped events in place of the logic required to model system failures at any particular plant.
  - To expand the model to represent a particular plant - expand undeveloped events into appropriate fault tree logic.

EXPANSION OF TEMPLATES INTO LP/SD SPAR MODELS
Lead Plants

- Identified Lead Plants in Eight Plant Classes (Classification Consistent with Revision 3i SPAR Models):
  - Millstone 3
  - Byron 1 & 2
  - Oconee 1, 2, & 3
  - Millstone 2
  - Palo Verde 1, 2, & 3
  - Peach Bottom 1 & 2
  - Surry 1 & 2
  - Grand Gulf
- Start with Existing LP/SD SPAR Model Template
  - PWR
  - BWR 5/6
  - BWR 4

EXPANSION OF TEMPLATES INTO LP/SD SPAR MODELS
Lead Plants (Continued)

- Add All System Fault Tree Logic from the Corresponding Revision 3i SPAR Model.
- Add All Basic Event Information from the Revision 3 SPAR Model.
- Revise LOOP and EDG Recovery Probabilities to Reflect Longer Recovery Times during LP/SD.
- Modify System Logic so that System Configuration is Properly Represented in Each Plant Operating State Group (POSG).
- Review System Success Criteria.
- Add New Test and Maintenance Events and Modify the Values to Reflect LP/SD Conditions.

EXPANSION OF TEMPLATES INTO LP/SD SPAR MODELS
Lead Plants (Continued)

- Revise the Recovery Rules as Necessary to Consider New Technical Specification-Disallowed Maintenance Combinations in Effect during LP/SD.

- Modify Human Error Probabilities (HEPs) to Reflect Longer Action/Recovery Times Available during LP/SD Operation.

- To Develop a LP/SD SPAR Model for Another Plant in the Same Class:

    - Follow same steps as those identified above for the lead plant.

    - Document development process and incorporate in Users Manual - include assumptions.

INTERNAL QA REVIEW OF DRAFT LEAD PLANT MODEL

MODEL AND DOCUMENTATION REVIEW

- Review:
    - Event trees.
    - Fault trees.
    - Basic event data.
    - Common cause failure modeling.
    - Graphical Evaluation Module (GEM) and GEMDATA.
    - Human Reliability and Recovery.
    - Revision log.
- Model Testing:
    - Perform appropriate (PWR or BWR) suite of tests.
    - Document results of model testing in prescribed format.

ONSITE QA REVIEW OF DRAFT LP/SD SPAR MODEL AGAINST LICENSEE'S LP/SD PRA MODEL

- QA Procedure Developed from Procedure Used for Onsite Review of Rev. 3 SPAR Models.
- Areas Covered by Review:
  - Event Tree Structure.
  - Success Criteria.
  - Dependencies.
  - Plant Operating States (POSs).
  - Plant Operating State Groups (POSGs).
  - Time Windows (TWs).
- Documentation of Onsite Review
  - Reported in separate appendix to revised Users' Manual.

SPAR HRA METHODOLOGY

- First Developed for NRC by INEEL in 1994 for Use in Accident Sequence Precursor (ASP) Program.
- Revised in 1999 to Incorporate Desirable Aspects of Other HRA Methods and Sources and Tailored to SPAR Model Usage.
- Uses a Three-Page Worksheet to Rate a Series of Performance Shaping Factors (PSFs) and Dependency Factors to Arrive at a Screening Level Human Error Probability (HEP) for a Given Task.

UPDATED HRA METHODOLOGY AND DOCUMENTATION FOR SPAR MODELS

- Purpose of Improvements:
  - Ensure that methodology and documentation comply with proposed ASME Standard on PRA.
  - Provide a referenceable document on SPAR HRA methodology.
- Add Uncertainty Analysis Capability
- Review Existing Full Power PSFs ; Identify Needed Changes.
- Add Specific Application to Analysis of LP/SD Events/Conditions.
  - Review insights regarding PSFs during LP/SD operation obtained from other LP/SD work.
- Document Improved Methodology in a NUREG/CR Report.

ONSITE QA REVIEW OF LP/SD SPAR MODEL FOR SURRY

ONSITE REVIEW OF LP/SD SPAR MODEL FOR SURRY

- Conducted August 15, 2002.
- Held in Conjunction with NRR's Review of LP/SD SDP Analysis Tool.
- Participants: NRC HDQ Staff, Region II SRA, INEEL Staff, BNL Staff, Licnsee's PRA Staff.
- Scope of Review of LP/SD SPAR Model:
  - Event tree structure.
  - Success criteria.
  - Dependencies.
  - Plant Operating States (POSs).
  - Plant Operating State Groups (POSGs).
  - Time Windows.

ONSITE REVIEW OF LP/SD SPAR MODEL FOR SURRY
(Continued)

PLANT-SPECIFIC REVIEW INSIGHTS:

- In General, Found Good Agreement between LP/SD SPAR Model and the Surry LP/SD PRA.
    - Both based on NUREG/CR-6144.
    - Surry LP/SD PRA uses IE frequencies taken from NUREG/CR-6144.
    - LP/SD SPAR model also uses NUREG/CR-6144 IE frequencies.

- LP/SD SPAR Model for Surry Separates Out Loss of RHR Caused by Loss of Level Control from Loss of RHR Initiating Event Group.
    - Differs from treatment in Licensee's LP/SD PRA model.
    - Based on implications of recovering RHR - from NRR review of LP/SD-related inspection findings.
    - Consistent with NRR's LP/SD SDP Analysis Tool.

ONSITE REVIEW OF LP/SD SPAR MODEL FOR SURRY
(Continued)

<u>GENERIC REVIEW INSIGHTS</u> (Consider in Future Model Development)

- Potential for containment sump plugging during LP/SD operations appears to have a higher likelihood compared to that at full power.
  - Due to increased level of personnel activity during LP/SD.
- Some plants operate in mid-loop with the RCS closed.
- Reflux cooling is only possible when RCS is closed, and can be modeled as a passive phenomenon.
- If the RCS is depressurized, some losses of inventory are self-terminating.
  - Any losses of inventory caused by over-draining will only drain to the bottom of the hot leg.

ONSITE REVIEW OF LP/SD SPAR MODEL FOR SURRY
(Continued)

- At some plants, preferred method of RCS makeup (given a loss of inventory during LP/SD) is gravity feed from the RWST.

- When considering the possibility of gravity feeding the RCS from the RWST, the analyst should consider the need to make up to the RWST.

- The analyst should consider the possibility of crediting the accumulators for makeup to the RCS.

    - Might increase available time for recovery.

<u>EVALUATION OF REVIEW RESULTS:</u>

- Inconclusive Relative to SPAR Model QA Acceptance Criteria.

- Further Discussion with Licensee Planned.

CANCELLATION OF PLANS FOR REVISION 4i SPAR MODEL DEVELOPMENT

CANCELLATION OF PLANS FOR REV. 4i SPAR MODELS

- Current and Future Plans for SPAR Model Development and Associated Budget Specify Plans for Developing Level 1 Models for Full Power and Low Power/Shutdown Operations, Level 2/LERF Models, and External Events (e.g., fires, flooding, seismic, etc.) Analysis Capability.

- Revision 3 SPAR Models Developed by Improving Revision 2QA Models To:

    - Add more initiating events (e.g., med. & large LOCAs, sec. system IEs).
    - Model other support systems (SWS,CCW, etc.) besides emergency ac power.
    - Enhance treatment of CCFs.
    - Add uncertainty analysis capability (for equipment performance).
    - Add new HRA methodology (currently being enhanced to add uncertainty analysis capability).

- Revision 3 SPAR Models Capture ~80-85% of Internal Events CDF.

CANCELLATION OF PLANS FOR REV. 4i SPAR MODELS
(Continued)

- Est. Total Cost of Revision 3 SPAR Model Development = $3.8 million.
  - Produce/conduct onsite QA reviews of 72 models.
  - Project on schedule.

- Consequences of Canceling Development of Set of Rev. 4i SPAR Models.
  - No extensive effort to revise Rev. 3 SPAR models.
  - Line items in future SPAR Model Development Program budget:
    - Maintain and improve existing SPAR models.

      User-Friendly front-end Interface for SDP
      Staff/contractor monitor technical issues - model revisions

    - Provide technical support to model users.

Briefing for ACRS

on

Guidance for Performance-Based Regulation

N. Prasad Kadambi, NRC/RES/REAHFB

October 11, 2002

10

OUTLINE

- Summary
- Historical Background
- Why "Guidance?"
- General characteristics of process.
- Illustration of process through example.
- Revised high-level guidelines.
- Conclusions.

Summary

- The developmental phase of the NRC's performance-based regulatory initiative is now complete with the availability of a suitable guidance document.

- The research work on the principles of performance-based regulation, and the applications on specific projects, have given us confidence that a broader range of activities should be encompassed by this work.

- The staff's plans to incorporate performance-based regulation within the scope of "regulatory coherence activities" will enable wider application of the concepts, including exploration of areas of research that may benefit from formal decision methods.

- If ACRS supports the staff's approaches and actions so far, the agency will have come closer to realizing the goals of the Commission's White Paper on risk-informed and performance-based regulation which is to have an integrated regulatory process.

Historical Background

- DSI-12, Commission White Paper on "Risk-Informed and Performance-Based Regulation", and Strategic Plan

- SECY-99-281, "The Vision of the RES Role":

"To achieve the agency's goals to maintain safety while reducing unnecessary burden through realistic assessments, RES will: ...
coordinate agency efforts to become more risk-informed and performance-based;"

- SECY-00-191, "High-Level Guidelines for Performance-Based Activities", and NUREG/CR-5392, "Elements of an Approach to Performance-Based Regulatory Oversight" were published after Advisory Committee reviews

- SECY-01-0205, "Status Report on Performance-Based Approaches to Regulation"

- Actions and milestones:

    - An integrated process in accordance with White Paper (on-going)
    - Pilot projects (individual milestones)
    - User friendly guidance document -- FY 2002
    - Communication Plan -- Mid-FY 2002

Why "Guidance"

- Feedback from Performance-Based Regulation Working Group [PBRWG] indicated that high-level guidelines are at too high a level. They articulate attributes, but do not provide direction on implementation.

- Staff's first attempt at developing implementation guidance resulted in a highly formal and overly general presentation of decision theory. Hence, the staff has adopted a two-step process in which the simplified guidance is expected to be sufficient in most cases, and a more formal approach pursued if necessary.

- Although the White Paper on "Risk-Informed and Performance-Based Regulation" provided definitions for all important terms, including "Performance-Based Approach", a consistent application for "performance-based regulation" (PBR) is not being realized (eg. see "Rulemaking Activities Plan"). Feedback indicates the need for user friendly guidance.

- Instead of a Management Directive, staff informed Commission in SECY-01-205 that a user friendly guidance document would be developed as a companion to NUREG/BR-0058 "Regulatory Analysis Guidelines".

General Characteristics of Process

- Guidance document completes the developmental phase of the staff's PBR efforts.
- It represents an internally self-consistent approach to regulation originating from the "White Paper" and applicable to the three arenas of regulatory responsibility.
- Process aspects of "Guidance" bear strong resemblance to formal decision theory with the flexibility for varying degrees of formality and quantification.
- Guidance naturally integrates "risk-informed" with "performance-based" regulation.
- It fulfills expectation expressed in SECY-01-205, and substantially responds to commitment made to ACRS:
    - "Eventually, an integrated process is expected that, in accordance with the Commissions's White Paper, combines the "risk-informed" and "performance-based" elements to regulatory decision-making."
    - Uses terminology employed by the published literature in the area of formal methods for decision-making.
- Expected to meet the needs for including PBR alternatives in majority of regulatory issues.

Illustration of "Guidance" Process

- Illustration of steps in the guidance process will be based on recent performance-based actions:

  - The Reactor Oversight Process (ROP) is demonstrably risk-informed and performance-based.

  - The proposed rulemaking on 10 CFR Part 50.44 incorporates a performance-based approach to hydrogen monitoring.

  - The proposed rulemaking on 10 CFR Part 72 relative to ISFSI and MRS facilities incorporates a performance-based approach to cost-beneficial geological and seismological analysis for the regulatory analysis.

- Pilot projects show that finding performance-based elements in a regulatory action requires, not a formulaic approach, but a systematic search for less prescriptive measures.

- The formalism provided by the high-level guidelines and the guidance steps helps maintain consistency and coherence.

Illustration of "Guidance" Process (continued)

- *Step 1: Define regulatory issue and its context:*
  - Arena is generally clear, but sub-arena may require internal discussion
  - Potentially addresses all four NRC performance goals.
  - Expected outcome is to provide appropriate regulatory requirements and supporting framework.
- *Step 2: Identify safety functions:*
  - ROP structure provides benefit not available for the other examples.
  - The rulemaking on 10 CFR 50.44 identified hydrogen monitoring as the safety function for application of a performance-based approach.
  - For the ISFSI, example safety functions were identified as stability against soil liquifaction during vibratory motion, and cask sliding and resulting displacements during an earthquake event.

Illustration of "Guidance" Process (continued)

- *Step 3: Identify safety margins:*
  - Safety margins in ROP are expressed as DCDF from inspections or PIs.
  - Performance targets for hydrogen monitoring function are based on reliability, availability and capability. Comparison with observed performance through servicing, testing and calibration provides a measure of safety margin.
  - Safety margins for ISFSI are substantial because casks are designed for challenges of handling and transportation.
- *Step 4: Select performance parameters and criteria:*
  - The level at which performance will be evaluated is considered here.
  - ROP may institute time at risk-significant configuration as a performance parameter at shutdown. This is an example of setting a high-level parameter. The criterion would have more considerations than risk model computations.
  - The regulatory analysis application for ISFSIs is an example of a performance-based approach to cost effective implementation of a regulation.

Illustration of "Guidance" Process (continued)

- *Step 5: Formulate a performance-based alternative:*

    - The considerations inherent in the staff's responses in Steps 1-4 would have decided the viability of a performance-based approach. If it is viable, the information developed includes candidate performance parameters.

    - The context of the regulatory issue (including consideration of defense-in-depth) determines which parameters are selected and how they are used in a regulatory action. Eg: Level of detail for analysis supporting siting of ISFSI.

    - The resolution of the regulatory issue should consider optimization within the regulatory framework, using prescriptive elements as needed. Eg: Regulatory guidance incorporating hydrogen monitoring into the maintenance rule program.

    - Any flexibility provided by a regulatory action may include consideration of appropriate licensee incentives to perform in a superior manner. Eg: ROP approach to risk significant shutdown configurations.

Revised High-Level Guidelines

- Three groups of guidelines maintain substantial similarity to those discussed in public and stakeholder interaction:
  - Viability guidelines *(Can a performance-based approach be developed?)*
  - Assessment guidelines *(Is it worthwhile to develop a performance-based change?)*
  - Guidelines for consistency with regulatory principles *(Are we being consistent with basic regulatory principles?)*
- Viability guidelines are same as "White Paper" definition with rearrangement to put margin first and include qualitative measures.
- Assessment guidelines include consideration of NRC's performance goals, assessment of net benefit, and optimal use of regulatory framework.
- Regulatory principles include defense-in-depth considerations, Option 3 framework, and RG 1.174 philosophy.
- Formal treatment of defense-in-depth will be incorporated into later document.

Conclusion

- Staff requests a letter that provides ACRS views on the approach taken by the guidance document, and on the completion of the developmental phase of the performance-based regulatory initiative with the guidance document (subject to finalization).

- Staff plans to incorporate its PBR efforts into "regulatory coherence activities". This will enable more rapid progress toward increasing the use of performance-based approaches in a broader range of activities.

- An inter-office group, the Risk Management Team, will coordinate and provide policy direction to implementation of PBR activities.

- RES will develop a NUREG document in FY-2003 that provides more detail on formal decision methods as applied in support of performance-based approaches as well as other applications of such methods.

12

ACRS MEETING HANDOUT

Meeting No.	Agenda Item	Handout No.:
496	11	11.1

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

Authors
JOHN T. LARKINS

List of Documents Attached

11

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

INTERNAL USE ONLY

G:PlanPro(ACRS):ppmins.496
October 10, 2002

SCHEDULE AND OUTLINE FOR DISCUSSION
ACRS PLANNING AND PROCEDURES MEETING
OCTOBER 9, 2002

The ACRS Subcommittee on Planning and Procedures held a meeting on October 9, 2002, in Room T 2 B3, Two White Flint North Building, Rockville, Maryland. The purpose of the meeting was to discuss matters related to the conduct of ACRS business. The meeting was convened at 1:30 p.m. and adjourned at 4:10 p.m.

ATTENDEES

MEMBERS

M. Bonaca
T. Kress

ACRS STAFF

J. T. Larkins
S. Bahadur
H. Larson
S. Duraiswamy
R. P. Savio
J. Gallo
S. Meador

NRC STAFF

I. Schoenfeld, OEDO

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

Member assignments and priorities for ACRS reports and letters for the October ACRS meeting are attached (pp. 10-13). Reports and letters that would benefit from additional consideration at a future ACRS meeting were discussed.

RECOMMENDATION

The Subcommittee recommends that the assignments and priorities for the October 2002 ACRS meeting be as shown in the attachment (pp. 10-13).

2) Anticipated Workload for ACRS Members

The anticipated workload for ACRS members through December 2002 is attached (pp. 10-13). 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 the item included in Section II of the Future Activities List (p. 14).

RECOMMENDATION

The Subcommittee recommends that the members provide comments on the anticipated workload. Changes will be made, as appropriate. The Committee should decide on the Subcommittee's recommendations on the item in Section II of the Future Activities List.

3) Foreign Travel Update

Since last ACRS meeting, the final travel arrangements for the October foreign travel have been made and a detailed itinerary has been put together by Tanya Winfrey. For the Germany leg of the trip (Quadripartite Meeting), the cost of the sleeping rooms and an ancillary fee have been prepaid by the Government in the form of a registration fee for both the members and staff. Therefore, members should not have to pay any rooming charge for their stay in Germany during the week of the Quadripartite Meeting as it has been prepaid.

The technical papers for the Quadripartite Meeting have been sent to the RSK and the Commissioners. The slides for presenting those papers are done. Sherry Meador is coordinating the translation so that both English and German versions of the presentations and overheads are available at the meeting in Germany.

RECOMMENDATION

The Subcommittee recommends that each member review a copy of the detailed travel itinerary. Comments or changes should be provided to Tanya Winfrey immediately.

The slides for the Quadripartite Meeting have been completed and forwarded to be translated. The PowerPoint presentation will be forwarded to the RSK with a backup copy on CD to be available.

4) Celebration of the 500th ACRS Meeting

As agreed to by the members, invitations were sent to the NRC Commissioners to participate at the 500th ACRS meeting ceremony, which is scheduled for March 4-5, 2003. (This is also coincidental with the Committee's 50th Anniversary.) NRC Chairman Meserve and all Commissioners as well as Bill Travers, EDO, have agreed to participate. Invitations were also sent to those who are expected to serve as panel members.

Drs. Hal Lewis, Robert Seale, Bill Stratton, J.Ernest Wilkins, Stephen Hanauer, and Mr. Dave Ward, have agreed to serve on the panels. Because of his health problems, Dr. Okrent has decided not to be a lunch time speaker. Dr. Remick has agreed to be a lunch time speaker.

As decided by the Committee at the June 2002 meeting, invitations were sent to Mr. Ralph Beedle, NEI, and Mr. Bert Wolf, GE, to participate in the celebration. They agreed to participate as Panel members.

A letter has been forwarded to all of the Panel participants and speakers thanking them for their willingness to participate in the 500th meeting ceremony. Additionally, accommodations are being arranged for all of the guests to stay in the area along with developing transportation from the hotel to the White Flint building.

RECOMMENDATION

The Subcommittee recommends that the Executive Director keep the Committee informed of further developments, including the status of obtaining funding from the agency. The possibility of the Committee sponsoring a reception at the end of the first day was discussed and it was recommended that the Full Committee consider this matter after the staff collects information on cost.

5) Role and Use of PRA in the Regulatory Decisionmaking Process

Mr. Karl Fleming of Technology Insights has started to plan for the Committee's "white paper" addressing the role and use of PRA in the regulatory decisionmaking process. He is in Rockville, Maryland, and plans to conduct interviews with NRC staff on October 10-11, 2002, hearing their views on what needs to be done to enhance PRA submittals. Mr. Fleming will present a draft plan for researching and compiling the information during the Saturday session of the October full Committee meeting. To the extent time permits, he will meet with members individually during the October meeting. Dr. Hossien Nourbakhsh has been designated as the Project Manager for this activity and will work closely with the contractor to guide this effort.

RECOMMENDATION

The Subcommittee recommends that members who would like to provide individual input into this project contact Dr. Nourbakhsh to coordinate a conference call with the contractor.

During Mr. Fleming's briefing at the October ACRS meeting regarding his plans for developing a draft "White Paper," the members should provide feedback to Mr. Fleming on his plans.

6) Meeting with the EDO

The Planning and Procedures Subcommittee plans to meet with the EDO and the Deputy EDOs (DEDOs) on Friday, October 12, 2002, between 12:00 and 1:00 p.m. to discuss several issues, including the following:

- ACRS/NRC staff coordination
- Adequacy of the NRC staff's review of power uprate applications
- License Renewal Issues
- High Burnup Fuel Issues
- Revision 1 to Reg. Guide 1.174
- Significant issues that the NRC staff expects to submit to the ACRS for review in the next two years.
- DPV regarding proposed 10 CFR 50.69 (pp. 15-16)

As suggested by the Committee during its September meeting, a formal meeting between the EDO/DEDOs/Office Directors and ACRS will be scheduled for a future ACRS meeting.

RECOMMENDATION

The Subcommittee recommends that the Chairman report to the Committee on the results of the above meeting.

7) Quad Cities Unit 2 - Damage to Steam Generator Dryer Resulting from Power Uprate Operation (Open)

On July 11, 2002, Quad Cities Unit 2 was shut down to investigate the irregularities in the steam flow, reactor pressure and level, and moisture carryover in the main steamlines. The results of the investigation revealed that a cover plate of the steam dryer was missing. Subsequent investigation led to the identification of pieces of the cover plate. A large piece was found in the Separator and there were indications that pieces had been transported into "A" main steamline and the vessel. Most of the pieces have been retrieved.

The cause of the damage is believed to be due to high-cycle fatigue induced by the cover plate natural frequency, nozzle chamber standing wave acoustic frequency, and vortex shedding frequency -- all coinciding at 180 Hz.

During the September 2002 ACRS meeting, the Committee asked Drs. Ford and Ransom to review this matter and propose a course of action, noting any generic implications (pp. 17-19).

RECOMMENDATION

The Subcommittee recommends that Drs. Ford and Ransom provide a progress report on their assignment at the October meeting. They should develop a course of action for consideration by the Subcommittee and the full Committee during their November 2002 meetings.

8) ACRS Senior Fellow

A contract for one of the ACRS Senior Fellow positions has been awarded to Link Technologies, Inc. The company will provide the equivalent of one full time employee (FTE) over the course of next year. The FTE is equivalent to 2087 hours of manpower. The company has proposed the use of eight individuals who offer a wide array of expertise. The individuals are Dr. John Austin, Ali Tabatabai, Dr. Spyros Traiforos. Dr. Bernard Snyder, Phillip McKee, Jeff Woody, Charles Haughney, and Peter Kiang. In order to effectively utilize this contract, a work plan is being developed. The work plan will coincide with high priority topics under review by the Committee. Since the contract has been executed effective September 30, a plan needs to be put in place as soon as possible preferably by October 30, 2002. Dr. Savio, in consultation with Dr. Bahadur, has been designated as the Project Manager. Suggested topics for the work plan should be provided to Dr. Savio.

RECOMMENDATION

The Subcommittee recommends that members identify topics that they would like to have included in the work plan. The Executive Director in consultation with the Planning and Procedures Subcommittee will prioritize the Tasks for Link Technologies, Inc. and provide periodic status reports to the Full Committee.

9) Proposed Tasks for Dr. Nourbakhsh, ACRS Senior Fellow

During the September meeting, the Subcommittee proposed the following tasks for Dr. Nourbakhsh and recommended that members propose other tasks:

- Review NUREG-1150 to see if parameter and model uncertainties can be extracted from the overall uncertainty in order to have an estimate of just the model uncertainty contribution. This estimate could then be used in regulatory decisions involving PRA results that include only parameter uncertainty. [May need to use current PRAs with parameter uncertainty quantified for the NUREG-1150 reference plants.]

- The ACRS has proposed that frequency-consequence (F-C) curves could replace or supplement CDF and LERF as a risk-acceptance metric that would capture the full range of potential radioactivity releases and be made consistent with the safety goals. A White paper is needed to "flesh out" this proposal:

> What Consequence to Use? Would it be TEDE?
> What are the F-C values produced by PRAs?
> How are these related to CDF and LERF?
> What would be a reasonable 3-region set of curves to use as risk-acceptance values?

On September 29, 2002, Dr. Wallis suggested that Dr. Nourbakhsh look at the history of the "momentum equation" in RELAP and other codes and advise the Committee about what should be done.

Dr. Nourbakhsh has provided his initial thoughts on the feasibility of extracting parameter and model uncertainties from NUREG-1150 overall uncertainty (pp. 20-28).

Also attached are e-mails from Drs. Kress and Apostolakis regarding Hossein Nourbakhsh's work (pp. 29-30)

<u>RECOMMENDATION</u>

The Subcommittee agrees with the conclusion reached by Dr. Nourbakhsh that separating out parameter and model uncertainties, while possible in principle, is somewhat an overwhelming task and the direct application of the results to any specific plant would be highly questionable. The Subcommittee recommends the following:

- Dr. Nourbakhsh need not have to pursue this task.
- Pursue the alternative tasks proposed in the Feasibility Memorandum by Dr. Nourbakhsh.
- Pursue the task on F-C curves.

The priority recommended by the Subcommittee for Dr. Nourbakhsh's tasks is as follows:

- AP1000 PRA
- F-C Curves
- Alternative tasks proposed by Dr. Nourbakhsh
- History of the "momentum equation" in RELAP and other codes
-

10) ACRS Retreat for 2003

Due to budget limitations, the ACRS retreat is scheduled to be held in Rockville on January 23-25, 2003. The topics below have been proposed by the Planning and Procedures Subcommittee and Dr. Powers.

Topics proposed by the Planning and Procedures Subcommittee include

- Dr. Fleming's draft "White Paper" on PRA.
- Mr. Rosen's report on Davis-Besse
- Certification process for advanced reactors
- Member issues (process, ACRS practice)

Subsequently, Dr. Powers suggested that the following issues associated with ACRS process and procedures be discussed at the retreat (pp. 31-42):

- ACRS effectiveness and self-assessment
- Proliferation of Subcommittee meetings
- Compensation process
- ACRS strategy for reviewing technical issues -- proactive vs. reactive
- Member assignments and Subcommittee responsibilities
- Discussion and resolution of differing technical views among ACRS members in areas such as risk-informed regulatory process at the meetings instead of via electronic medium.

As requested by the ACRS Chairman, Dr. Bonaca, ACRS Vice Chairman, discussed the issues raised by Dr. Powers during the Subcommittee meeting.

RECOMMENDATION

The Subcommittee recommends the following:

- Since the agency is currently operating under "continuing Resolution" and is expected to operate under this condition until at least the first quarter of next year, the Committee should defer its retreat until 2004.

- The Committee should start the February 2003 meeting a day earlier to discuss the process issues raised by Dr. Powers, other member issues, as well as the ACRS self-assessment.

11) ACRS Meeting Dates for CY 2003

Proposed ACRS meeting dates for CY 2003 were provided to the members during the September ACRS meeting. Changes proposed by the members (pp. 43-47) are included in the attached calendar (pp. 48-59). The Committee needs to decide on these dates during the October ACRS meeting. [Note: In response to members' comments the June meeting has been moved from June 4-6 to June 11-13, and the September meeting from 3-5 to 11-13.]

RECOMMENDATION

The Subcommittee recommends that the Committee discuss the changes proposed by the members and approve the dates for CY 2003 ACRS meetings.

12) Tour of Vender Facilities

Members and Consultants of the Thermal-Hydraulic Phenomena Subcommittee toured the GE facilities in San Jose, CA, on September 23-24 and the Framatome ANP-Richland facilities in Richland, WA on September 25-26, 2002, to obtain information on the details of fuel and reactor core design methodology for use by the Committee in reviewing core power uprate applications. Dr. Apostolakis suggested that Dr. Wallis provide a report to the Committee on this matter.

RECOMMENDATION

The Subcommittee recommends that, as suggested by Dr. Apostolakis, Dr. Wallis provide a brief report to the full Committee at the October 2002 meeting regarding tour of the GE and Framatome facilities.

13) Tour of Global Nuclear Fuel Cycle Facility

The members of the joint Subcommittee of ACRS/ACNW (Kress, Garrick, and Levenson) are scheduled to tour the Global Nuclear Fuel Cycle Facility in Wilmington, NC, on November 5, 2002. The ACRS members of the joint Subcommittee Drs. Kress and Powers decided not to attend this tour. Since only two ACNW members plan to attend this tour, the Subcommittee needs to decide whether it is worthwhile having this tour.

RECOMMENDATION

Because of budget constraints due to the agency being under a continuing resolution, the Executive Director has recommended that this tour be deferred to some future time when funds become available. The Subcommittee agreed with the recommendation of the Executive Director.

14) Staff Response to Dr. Kenneth D. Bergeron Regarding TVA's License Amendment Request to Produce Tritium at the Watts Bar Nuclear Power Plant

In a letter of September 13, 2001, Dr. Kenneth Bergeron, formerly associated with the Sandia National Laboratories, and now a member of the public, expressed concern about the NRC staff's review of TVA's license amendment request to irradiate tritium-producing burnable absorber rods at the Watts Bar Nuclear Power Plant. As suggested by the Committee at the October 2001 meeting, Dr. Larkins sent a memorandum to the EDO on October 18, 2001 (pp. 60-65) transmitting Dr. Bergeron's letter and requesting that the EDO keep the ACRS informed of the staff's disposition of Dr. Bergeron's concerns.

The staff sent a letter to Dr. Bergeron on September 6, 2002 (pp. 66-91) responding to his concerns. The staff also sent a memorandum to Dr. Larkins informing him of the staff's response to Dr. Bergeron.

15) Meeting with Laurence Williams, NII

Kress
Bonaca
Wallis
Powers

As requested previously by the Planning and Procedures Subcommittee, Dr. Larkins has contacted the NII Chief Inspector's office in the UK concerning a meeting with Laurence Williams, Chief Inspector for NII, and the ACRS (p. 92). We have been notified by the NII Chief Inspector's Technical Support Staff that Mr. Williams will be in the U.S. in December and more specifically in the Washington area on the 5th of December, and it may be possible to spend some time with some of the ACRS members. Dr. Larkins will e-mail the Chief Inspector's office and try to set up a time for a meeting with some of the ACRS members during the 498th meeting, December 5-7, 2002. These discussions will center around the certification of the AP1000 design. As we understand it, the UK has considered building two advanced reactor plants and is currently focusing on the Westinghouse AP1000. As such, they are interested in the certification process required by the NRC, including ACRS involvement.

RECOMMENDATION

The Subcommittee recommends that Dr. Larkins keep the members informed as to the scheduling of this matter.

16) Member Issues

Implications of Boric Acid

Dr. Kress sent an e-mail (p. 93) stating that the Davis-Besse event raises broader concerns than the safety culture issue and he believes that is "boric acid" issue. He states that boric acid is used simply for convenience and flexibility in controlling the reactivity as burnup proceeds during a fuel cycle. He believes it is not really needed. BWRs do very well without boric acid and by using burnable poisons (gadolinium), and PWRs could do just as well. He suggests that the Committee urge the staff to take a much broader look at the implications of boric acid than just what happened at Davis-Besse.

Mr. Sieber does not agree with Dr. Kress' idea about eliminating the use of boric acid in PWR coolant systems (p. 94). His views are included in the attached e-mail (pp. 93-94).

ANTICIPATED WORKLOAD
October 10-12, 2002

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Apostolakis	--	Larson	Guidance for Performance-Based Regulation	A	To meet the CTM schedule	--
Bonaca	Leitch	Kobetz/Duraiswamy	License Renewal Application for Catawba and McGuire Units 1 and 2	A **[Interim letter as needed]**	To meet the CTM schedule	--
Kress	--	Savio	Overview of ESBWR, SWR 1000, and ACR 700 Pre-application review	--	--	--
		El-Zeftawy	Policy issues related to advanced reactor licensing	--	--	--
Leitch	--	Weston	Significant recent operating events	--		
Powers	--	Duraiswamy	LPSD SPAR Model Development Plan and Cancellation of revision 4i of SPAR Model	A	Response to a Commissioner's request	--
	Kress	El-Zeftawy	Confirmatory Research Program on High-Burnup Fuel	B	To provide ACRS views	--

10

ANTICIPATED WORKLOAD
November 7-9, 2002

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Apostolakis	--	Savio	Safeguards and Security activities **[Discussion of Draft Report]**	Report to be complet-ed in December	--	--
Ford	--	Savio	Status of annual ACRS report to the Commission on the NRC Safety Research Program	--	--	--
Kress	--	Weston/ Snodderly	Proposed Resolution of GSI-189, "Susceptibility of the Ice Condenser and Mark III Containments to Early Failure From Hydrogen Combustion During a Severe Accident" **[May be deferred to December]**	A	To meet the CTM schedule	--
		El-Zeftawy	Early Site Permit Process	B	To provide early feedback	--
		Savio	Advanced Reactor Research Plan	--	--	--
		El-Zeftawy/Boehnert	Westinghouse AP1000 Design	--	--	--
Leitch	--	Weston	Significant recent operating events	--	--	--
	Bonaca	Assa/Kobetz	Peach Bottom License Renewal application	-- **Interim letter as needed]**	--	--

ANTICIPATED WORKLOAD
NOVEMBER 7-9, 2002 (CONTINUED)

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Rosen	--	Weston	Risk-Informed Improvements to Standard Technical Specifications	A	To meet the CTM schedule	–

ANTICIPATED WORKLOAD
DECEMBER 5-7, 2002

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Apostolakis	--	Snodderly/ Duraiswamy	Draft Final Regulatory Guide DG-1122, "Determining the Technical Adequacy of PRA Results for Risk-Informed Activities"	A	To meet the CTM schedule	--
		Savio	Safeguards and Security Activities **[Discussion of the Report]**	A	To provide early advice to the Commission	--
Ford	Sieber	Weston	Vessel head penetration cracking and Vessel head degradation	A	To meet the CTM schedule	--
		Savio	Draft annual ACRS report to the Commission on the NRC Safety Research Program	--	--	--
Kress	--	El-Zeftawy	Policy issues related to advanced reactor licensing	--	--	--
Leitch	Bonaca	Kobetz/Duraiswamy	North Anna and Surry License Renewal Application **[Final Review]**	A	To meet the CTM schedule	--
		Weston	Significant recent operating events	--	--	--
Powers	--	Assa/Larson	Draft Final ANS External Events Methodology Standard	A	To meet the CTM schedule	--
Ransom	Wallis	Boehnert	Resolution of GSI-185, "Control of Recriticality Following Small Break LOCAs in PWRs"	A	To meet the CTM schedule	--

ANTICIPATED WORKLOAD
DECEMBER 5-7, 2002 (CONTINUED)

Wallis	--	Boehnert	Framatome ANP Richland, Inc. S-RELAP5 Realistic Large-Break LOCA Code	A	To meet the CTM schedule	--

II. ITEM REQUIRING COMMITTEE ACTION

1. Draft Final ANS External Events Methodology Standard (Open) (DAP/RA/HJL)
ESTIMATED TIME: 2 hours

Purpose: Determine a Course of Action

Review schedule specified in CTM [B. Budnitz, ANS/N. Chokshi, RES]. The NRC staff previously requested the American Society of Mechanical Engineers (ASME) and American Nuclear Society (ANS) to develop Standards for use by industry in standardizing and upgrading their PRAs to facilitate risk-informed decisionmaking. ASME developed a Standard on internal events which the Committee reviewed and provided comments in letters dated March 25, 1999, and July 20, 2000.

The Committee reviewed the draft ANS Standard on external events PRA and provided comments in a letter dated February 9, 2001. The public comment period for the draft ANS External Events Methodology Standard closed in April 2001. Significant comments have been received during the public comment period and ANS is in the process of resolving these comments. The revised Standard, which reflects incorporation of pubic comments, will be resubmitted to the ANS Committee for approval. ANS has reconciled all issues resulting from the resolution of public comments and requested to brief the full Committee in November 2002.

Dr. Powers recommends that this item be scheduled for the December ACRS meeting, subject to resolution of public comments.

October 3, 2002

MEMORANDUM TO: John Larkins

FROM: August W. Cronenberg

SUBJECT: Synopsis Concerning Differing Professional View (DPV) Regarding Proposed 10CFR50.69 Rulemaking ("Risk Informed Categorization and Treatment of os Systems, Structures, and Components for Nuclear Power Plants")

Synopsis: On Sept. 26, 2002 three (3) NRC/NRR staff engineers filed DPVs concerning the proposed rulemaking on 10CFR50.69, sometimes referred to as Option-2. The three parties are Mr. David C. Fischer, Mr. Thomas Scarbrough, and Mr. John R. Fair, all senior engineers with the Mechanical & Civil Engineering Branch, Div. of Engineering, within NRR. All three participated in the development of Option 2 and thus quit familiar with the issues involved.

The central focus and commonality of their concerns relate to the treatment requirements for components classified in the RISC-3 class (Safety-Related/Low Safety Significance). Although each provide somewhat different arguments and supporting documentation for their case, commonality is evident in each of the DPV's. Each essentially asserts that the treatment requirements for components classified as RISC-3 under the current language of 10CFR50.69 are at such a high level that they are vague, therefore not sufficient to reasonably assure the functionality of RISC-3 classified components. Each DPV also asserts that the original language of the rule was altered in the "Concurrence Process", so that staff input to assure RISC-3 component functionality was largely eliminated.

In a one-page memo attached ("Changes to 10CFR50.69"), staff engineer Cronenberg alerted members of the Reliability and PRA Subcommittee as to redline/strikeout changes to the proposed 10CFR50.69 rule resulting from the Concurrence Process. That memo indicated that the intent of these changes was to relax requirements to meet consensus standards and control measures in the draft rule. A prior E-mail of 8-28 also indicated that the ACRS might have a problem with such changes. A copy of that Memo is attached.

cc: S. Duraiswamy
S. Bahadur
P&P Members...Apostolakis, Bonaca, Kress

Item 6
15

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

Aug. 29, 2002
(Overnight FEDX)

MEMORANDUM TO: Reliability and PRA Subcommittee, Apostolakis, Bonaca, Powers, Rosen, Ford, Shack, Kress

FROM: August W. Cronenberg, Senior Staff Engineer

SUBJECT: **Changes to 10CFR50.69**

The purpose of this memorandum is to forward to PRA subcommittee members recent redline/strikeout changes to the proposed 10CFR50.69 rulemaking on Risk-Informed Categorization of SSCs. Such changes result from the Concurrence Process, per input from the LT (Leadership Team) and ET (Executive Team). The intent of these changes is to relax somewhat requirements to meet consensus standards and control measures in the draft rule. Per my prior E-mail of 8-28 , the ACRS might have a problem with such changes.

Attached are the following:

a) Rulemaking Issue Notation Vote (note: strikeout and vertical lines indicating additions)
b) Federal Register Notice of Rulemaking for 10CFR50 (only pages with strikeout are attached)
c) Note From Chris Grimes on Rule Changes (also sent as attachment in prior e-mail of 8-28)

cc: Bahadur, Duraiswamy, Larkins, Larson

16

Received: from igate.nrc.gov
by nrcgwia.nrc.gov; Sun, 06 Oct 2002 13:57:20 -0400
Received: from nrc.gov
by smtp-gateway ESMTPœ id NAA15750;
Sun, 6 Oct 2002 13:55:03 -0400 (EDT)
From: FPCTFord@aol.com
Received: from FPCTFord@aol.com
by imo-m07.mx.aol.com (mail_out_v34.13.) id 6.f2.23026960 (4446);
Sun, 6 Oct 2002 13:58:11 -0400 (EDT)
Message-ID: <f2.23026960.2ad1d3b3@aol.com>
Date: Sun, 6 Oct 2002 13:58:11 EDT
Subject: Quad Cities Steam Dryer
To: apostola@MIT.edu, mvbonaca@snet.net, PAB2@nrc.gov, TSKress@aol.com, JTL@nrc.gov, dapower@sandia.gov, ransom@ecn.purdue.edu, historyart@computron.net, GMLeitch@aol.com, wjshack@anl.gov, JDSIEBER@aol.com, Graham.B.Wallis@Dartmouth.edu
MIME-Version: 1.0
Content-Type: text/plain; charset="ISO-8859-1"
Content-Transfer-Encoding: quoted-printable
X-Mailer: AOL 5.0 for Windows sub 138

Subject; Quad Cities Unit 2 Steam Dryer Failure

At the September P & P meeting I was asked to follow up on the flow induced vibration (FIV) failure of the Quad Cities Unit 2 steam dryer which occurred soon after the plant had initiated power uprate operations.

I requested that this particular FIV topic be added to the agenda of a meeting that was held in San Jose, September 22-23. This meeting was hosted by GE Nuclear Energy for our thermal-hydraulics subcommittee in order to discuss various thermal-hydraulic codes relevant to power uprate applications. Consequently there was not time for a detailed discussion of the steam dryer design features, etc. but, nevertheless, some useful information was given. The salient points are given below.

1. The steam dryer is regarded as a non-safety related component. This status was determined in the industry report VIP-06 "Safety Assessment of BW
R
Internals" which was approved by the NRC in an SER in September 1998. In tha
t
report it was determined that even if various sub-assemblies in the dryer cracked (e.g. the support and hold down brackets) structural integrity would

be maintained even if there was a pressure transient due to a main steam lin
e
break. It was assumed that cracking would be discovered during inspection when the dryer was removed during refueling outages and that this frequency was sufficient to minimize the danger of component failure during operation.

Even if a loose part was created during operation the NRC-approved VIP-06 report stated there was no danger to reactor safety; this latter assumption was challenged at our meeting (for instance the consequence of loose part damage to the MSIV or the jet and recirculation pumps), with no definitive resolution..

2. Cracking has been observed in steam dryer assemblies "numerous times";

17

these have included the dryer hood, drain line, tie bar, support bracket, dryer cover plate. The cracking mechanisms have been a mixture of intergranular stress corrosion cracking and transgranular fatigue. Where loose parts have been involved, their disposition has been in agreement with

GE's prediction and they have given no safety concern. In no case have thes
e
instances led to plant shutdown , until the incident at Quad Cities Unit 2.

3. In many cases the utilities affected by the cracking have asked GE to instrument the plant subassemblies so that mechanical responses may be measured during subsequent operation; these tests have included pressure, strain and acceleration measurements. Thus there is a reasonable measuremen
t
data base on the actual dryer to supplement the "paper"analyses that have been conducted, and these have led to recommendations to the BWR owners from

GE regarding various operational procedures; e.g. attention to reassembly procedures. It was further pointed out that many of the steam dryer designs were modified based on these in-reactor tests; however, such changes were no
t
made to the steam dryer at Quad Cities.

4. There were no changes recommended to the dryer design specifically becaus
e
of power uprate (PU) conditions. This decision was based not on vibration tests on model assemblies (since GE does not have such a facility), but on the basis of analyses and data from operating plant mentioned above. Reasons

for not changing the dryer design with PU included the following;

- No dome pressure change with PU.
- Even though the stress on the component will increase with flow ra
te
(squared) it was decided that there was not a significant change in IGSCC potential
- It was recognized that the potential for fatigue cracking would be

increased based, for instance, on strain measurements conducted on the Susquehanna support lugs, but it was thought that this was manageable given the inspection frequency and the fact that the dryer was not a safety-relate
d
component.

5. Obviously some of the conclusions above relating to changes in steam dryer design associated with PU can be challenged in the light of Quad Cities, but the actions being taken at GE (admittedly reactively) include th
e
evaluation of the cracking probability for all steam dryers in the BWR fleet

and the consequence of dryer failure, should it occur. The former will rely

primarily on vibrational analyses, calibrated by the field test measurements

(in item 3), and the latter will focus primarily on a loose parts analysis. The vibrational analyses are being backed up by a limited mechanical testin
g

program on a 1/16 scale model dryer Utilities will then be advised as to whether modifications to their steam dryers should be made based on the resultant frequency/consequence assessment.

6. When challenged as to the likelihood of FIV incidences similar to Quad Cities occurring again, GE admitted that it could not be dismissed, but deemed that the risk was minimized by the actions being taken in item 5 and the fact that the consequence was relatively low.

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

October 2, 2002

MEMORANDUM TO: George Apostolakis, Thomas Kress
ACRS Members

FROM: H.N.
Hossein Nourbakhsh, Senior Fellow

SUBJECT: EXTRACTING PARAMETER AND MODEL UNCERTAINTIES FROM THE NUREG 1150 OVERALL UNCERTAINTY

Attached, for your comments, are my initial thoughts on feasibility of extracting parameter and model uncertainties from the NUREG-1150 overall uncertainty.

CC: John Larkins
Sher Bahadur

Item 9
20

Feasibility of Using the NUREG-1150 Uncertainty Analyses in Regulatory Decisions Involving PRA Results

by
Hossein P. Nourbakhsh
ACRS Senior Fellow
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

1. Introduction

Probabilistic Risk Assessment (PRA) is being used increasingly as an important element in regulatory decision making. A concern associated with the results of PRAs stems from their susceptibility to model uncertainties. These uncertainties are often of such a magnitude that they make the decision making process difficult.

There are two classes of epistemic uncertainty that impact the results of PRAs: parametric uncertainty and model uncertainty. Parameter uncertainties are those associated with the values of the fundamental parameters of the PRA model, such as initiating event frequencies and equipment failure rates that are used in quantifying the accident sequence frequencies.

Model uncertainties are those associated with the use of models for specific events or phenomena used in the development of the PRA model. Examples include approaches to modeling reactor coolant pump seal behavior and containment pressurization at reactor vessel breach.

The NUREG-1150 study [Ref. 1] was a major effort to put into a risk perspective the insights into system behavior and phenomenological aspects of severe accidents. An important characteristic of this study was the inclusion of the uncertainties in the calculations of core damage frequency and risk that exist because of incomplete understanding of reactor systems and severe accident phenomena.

Five specific commercial nuclear power plants were analyzed in NUREG-1150: Surry Power Stations, a 3-loop Westinghouse PWR with a subatmospheric containment; Zion, a 4-Loop Westinghouse PWR with large dry containment; Sequoyah ,a 4-loop Westinghouse PWR with ice-condenser containment; Peach Bottom, a BWR-4 reactor with a Mark I containment; and Grand Gulf, a BWR-6 reactor with a Mark III containment.

Assessing modeling uncertainties associated with the results of a plant-specific PRA is very resource intensive. The purpose of this paper is to review NUREG-1150 to see if parameter and model uncertainties can be extracted from the overall uncertainty in order to have an estimate of just the model uncertainty contribution. This estimate could then be used in regulatory decisions involving PRA results that include only parameter uncertainty.

The NUREG-1150 constituent analyses are briefly summarized first in order to provide a framework for discussion presented later on feasibility of utilizing the NUREG-1150 information on model uncertainty associated with each of the key elements of risk analysis.

21

2. Overview of the NUREG-1150 Study

Figure 2.1 displays schematically the components of the NUREG-1150 analytical process which consists of the following key elements:

(1) Systems analysis and models of plant response to various initiating events, quantification of accident sequences leading to core damage;

(2) Analysis of the accident progression and containment performance to determine various possible ways the accident could evolve given core damage;

(3) Source term analysis, the release of radioactive material to the environment for various outcomes of the accident progression;

(4) Consequence analysis, the health impacts of each of the source terms.

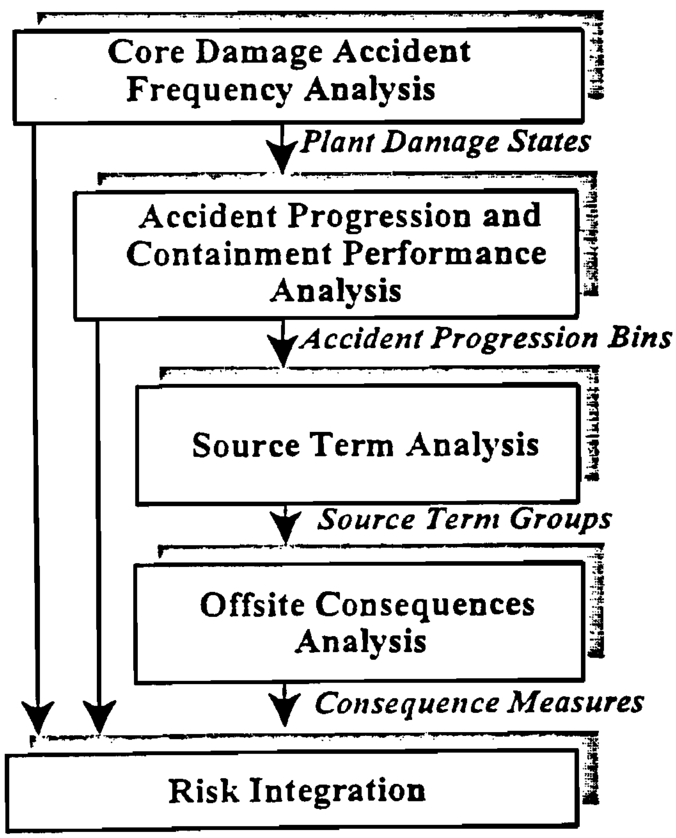

Figure2.1 Elements of The NUREG-1150 Risk analysis Process

22

Integrated risk was obtained by combining the frequency of core damage, the conditional probability of the release paths, and the value of the consequences of each source terms conditional on the release into a single risk measure. By repeating the calculation several times with different input values (over specific ranges) of key parameters, a distribution of offsite risk estimates was obtained from which the uncertainties in the risk were estimated.

2.1 Core-Damage Accident Frequency Analysis

The core damage frequency analysis considered accidents initiated by events occurring during normal full power operation of the plants (internal events). The analysis of accident frequencies for the Surry and Peach Bottom also included the consideration of accidents initiated by external events (e.g. earthquake, floods, fires). The analysis consists of fault trees and event trees delineating the accident sequences leading to core damage.

The calculations of core damage frequency and risk included the quantitative analysis of uncertainties. This analysis was performed using the Latin hypercube sampling technique. Probability distributions for many parameters for which the uncertainties were estimated to be large and important to risk were developed. For example 48 variables were sampled in accident frequency analysis for the Surry plant. Probability distributions for many of the most important accident frequency variables were generated using statistical analyses of plant data or data from other published sources (Ref. 1) For certain key issues in the uncertainty analysis, the elicitation of expert judgment was used to develop the needed probability distributions. An example of the accident analysis issues evaluated by the expert panels was the frequency and size of reactor coolant pump (RCP) seal failures before the onset of core damage in PWRs.

The outcome of the frequency analysis was a group of accident sequences leading to core damage and their associated frequencies. The accidents were then grouped into plant damage states (PDSs), based on similarity of plant conditions, to define the entry points for the subsequent accident progression analysis.

2.2 Accident Progression and Containment Performance Analysis

For each general type of accident, defined by the plant damage states, an analysis was performed to develop Phenomenological conditions and containment response for each accident progression path which determine the timing and failure mode of containment and influence the transport and release of radionuclides.

Accident progression was analyzed using a single accident progression event tree (APET) developed for each plant which was evaluated with the EVNTRE code [Ref. 2]. The specification of each PDS defines the entry conditions to the APET. The accident progression event trees developed for this study made extensive use of the available severe accident computer code calculations and experimental observations.

The elicitation of expert judgment was used to develop probability distributions for fourteen accident progression, containment loading, and structural response issues. Probability distributions for many other issues believed to be of less importance to risk were also developed by analysts on the project staff or by phenomenologists from national laboratories using techniques like those employed with the expert panels.

The APET developed for Surry and Zion in NUREG-1150 had over 70 event questions and many of the questions had several (more than two) outcomes; there are thus far too many paths through the tree to allow consideration of each individual path in terms of the subsequent source term and consequence analysis. The outcomes of the paths were then grouped into accident progression bins (APBs) which have similar characteristics and define the entry conditions for the source term analysis.

2.3 Source Term Analysis

The magnitude and composition of radioactive materials released to the environment with associated energy content, time, initial elevation and duration of release together are termed the "source term".

The source term analysis tracks the release and transport of radioactive materials from the core, through the reactor coolant system, then to the containment and other buildings, and finally into the environment. The removal and retention of radioactive materials by natural processes, such as deposition on surfaces, and by engineered safety systems, such as sprays, are accounted for in each location.

For the NUREG-1150 risk analysis, the source terms were calculated using simplified parametric algorithm. The parametric equations describe the source terms as the product of release fractions and transmission factors at successive stages in the accident progression for a variety of release pathways, a variety of accident progressions, and nine classes of radionuclides. This approach led to development of separate computer codes for each plant, i.e., the XSOR codes [Ref. 3]. The parametric models used in the XSOR codes are not time dependent. These codes generate source terms only in terms of early and delayed releases. The timing of release is particularly important for the prediction of early health effects.

None of the basic parameters used in the XSOR codes are internally calculated. The values for the parameters must be specified by the user or chosen from a distribution of values by a sampling algorithm. The input data on the more important parameters were constructed in the forms of probability distributions. Such distributions were developed using the elicitation of expert judgement. For a few parameters that were judged of lesser importance or not considered as uncertain, single-valued estimates were used in XSOR models.

The source term analysis resulted in characterizing thousands of source terms (20,000 for Surry and 75,000 for Grand Gulf) associated with tens of plant damage states, hundreds of accident progression bins, and the variation in source term phenomenological issues which were included in the propagation of uncertainties.

For the risk analysis, radioactive releases were grouped using the PARTITION program [Ref. 4] according to their potential to cause early or latent cancer fatalities and warning time. Through this "partitioning" process, the large number of radioactive releases calculated with the XSOR codes were collected into a small set of source term groups (30 to 60 in number for each plant). This set of groups was then used in the offsite consequence calculations.

2.3 Offsite Consequence Analysis

The severe accident radiological releases are of concern because of their potential for impacts on the surrounding environment and population. The impact of such releases to the atmosphere can manifest themselves in a variety of early and delayed health effects, loss of habitability of areas close to the plant site, and economic losses [Ref. 1]. In NUREG-1150 study, the consequence measures, early fatalities, population dose (person-rem), and latent cancer fatalities, were calculated for each source term group by the MAACS code [Ref. 5]. The output of MACCS for each source term group is a distribution of the consequences, conditional on occurrence of the source term, which incorporates the uncertainty (variability) due to weather as well as the uncertainty in the underlying health (dose-response) models.

3. Feasibility of Utilizing the NUREG-1150 Information on Model Uncertainty

As stated in the introduction, an important characteristic of NUREG -1150 study was the inclusion of quantitative estimates of the uncertainties in the calculations of core damage frequency and risk. Both types of epistemic uncertainties (parametric uncertainty and model uncertainty) were included in the NUREG-1150 study, and no effort was made to differentiate between the effects of the two types of uncertainties.

Important source of uncertainties exist in all four stages of risk analysis shown in Figure 2.1. In order for uncertainties in accident phenomena to be included in the probabilistic risk analyses conducted for the NUREG-1150 study, they had to be expressed in terms of uncertainties in the "high level" or summary parameters that were used in the study.

The NUREG-1150 analytical procedure for risk analysis is a cumbersome process which involves numerous computer codes and data transfer. Therefore, an effort to extract parameter and model uncertainties from overall uncertainty may become very involved and resource intensive. In addition, any such evaluation of model uncertainty should reflect the more recent technical knowledge and understanding of severe accident phenomena. For example, since the completion of NUREG-1150 study, advances have been made in the ability to predict the early containment failure phenomena of direct containment heating (DCH) and liner meltthrough that should be reflected in the model uncertainty. It should be noted that, as a part of a study to assess the risk significance of containment and related engineered safety features (ESF) system performance requirements [Ref. 6], the accident progression event trees (APETs) for Zion and Peach Bottom that had been used for NUREG-1150 were modified to reflect the current knowledge of early containment failure phenomena (DCH and Liner meltthrough).

The direct applicability of overall model uncertainty calculated for NUREG-1150 plants to other plants of similar NSSS (nuclear steam supply system) and containment design may be questionable. There are plant specific features and operational practices that may influence the likelihood and the severity of specific events or phenomena during the progression of severe accidents. For example, the results of individual plant examinations (IPEs)[Ref. 7] indicate that, specific containment features lead to unique and significant failure modes. For instance, the large probability values of early containment failures found in the IPEs for both Palisades and Davis Besse, do not result from the high pressure loads associated with DCH. Instead, the values are attributed to the special features of the particular designs of the plants. The location of the (ESF) sump in Palisades causes the flow of molten core debris from the reactor cavity into this sump and subsequently into the ESF recirculation piping. In the IPE analysis, the debris is assumed to eventually melt through the pipe wall and enter the auxiliary building, resulting in a large containment failure area. For Davis Besse, one of the few PWR plants that have large dry containments of steel construction, the largest fraction of early containment failure is associated with the potential failure of the containment wall via direct contact with core debris.

In spite of plant specific nature of NUREG -1150 quantitative results (e.g., core damage frequency, containment performance, risk), this study provides valuable insights on severe accident phenomenological issues and associated state- of -knowledge uncertainties which are very useful to the study of plants of similar NSSS and containment design.

While a formal propagation of the uncertainty is the best way to account for model uncertainties, under certain circumstances, it can be demonstrated that the model uncertainties associated with many phenomenological issues are not important to the overall risk. For example, it can be demonstrated that the bulk of risk significant contributing scenarios do not involve highly uncertain phenomenological issues. For instance, in a PWR plant with a high frequency of containment bypass sequences (i.e., Event V and SGTR) and with a high probability of depressurization of reactor coolant system (RCS), the model uncertainties associated with the thermally induced failure of steam generator tubes and direct containment heating are not important to the overall uncertainty in the early fatality risk.

NUREG-1150 and the results of the IPE Insights Program [Ref. 7] provide important sources of information that may be used to develop a simplified systematic methodology for assessing the plant specific importance of individual phenomenological issues and their model uncertainties to the overall uncertainty. The feasibility and options for developing such assessment methodology should be further explored. Development of such assessment methodology can greatly reduce the effort necessary for a formal propagation of the uncertainty to account for model uncertainty.

It may also be desirable to assign some ranking of "risk importance" among the various phenomenological issues that are considered in a plant PSA model. For example, a risk importance measure of "Risk Significance Worth", somewhat similar to Risk Achievement Worth (RAW) used for risk importance ranking of various plant components, can be defined as:

$$\text{Risk Significance Worth} = R\ (\text{issue } 1)^{+}/R_0$$

Where R (issue $1)^{+}$ is the calculated risk, using the bounding (conservative) assumptions in quantifying the phenomenological issue -1, and R_0 is the base-case risk. Examples of phenomenological issues include containment pressurization due to DCH, containment failure pressure, and probability of temperature-induced steam generator tube rupture. Other risk importance measures somewhat similar to Fussel-Vessly (F-V) or Risk Reduction Worth (RRW) can also be defined. It should be noted that various risk metrics (e.g. LERF) could be used for defining these risk importance measures. Development of importance measures for phenomenological issues should also be further explored. Such risk importance measures for phenomenological issues can be useful for directing any needed sensitivity analysis (in the absence of any formal model uncertainty analysis), as well as for developing research priorities to reduce the overall model uncertainty.

4. Summary and Conclusions

Assessing modeling uncertainties associated with the results of a plant-specific PRA is very resource intensive. NUREG-1150 constituent analyses was reviewed to see if parameter and model uncertainties could be extracted from the overall uncertainty in order to have an estimate of just the model uncertainty contribution. This estimate could then be used in regulatory decisions involving PRA results that include only parameter uncertainty.

The NUREG-1150 analytical procedure for risk analysis was found to be a cumbersome process which involves numerous computer codes and data transfer. Therefore, an effort to extract parameter and model uncertainties from overall uncertainty may become very involved and resource intensive. In addition, any such evaluation of model uncertainty should reflect the more recent technical knowledge and understanding of severe accident phenomena.

The direct applicability of overall model uncertainty calculated for NUREG-1150 plants to other plants of similar NSSS (nuclear steam supply system) and containment design may be questionable. There are plant specific features and operational practices that may influence the likelihood and the severity of specific events or phenomena during the progression of severe accidents. In spite of plant specific nature of NUREG -1150 quantitative results, this study provides valuable insights on severe accident phenomenological issues and associated state- of -knowledge uncertainties which are very useful to the study of plants of similar NSSS and containment design.

While a formal propagation of the uncertainty is the best way to account for model uncertainties, under certain circumstances, it can be demonstrated that the model uncertainties associated with many phenomenological issues are not important to the overall risk. NUREG-1150 and the results of the IPE Insights Program [Ref. 7] provide important sources of information that may be used to develop a simplified systematic methodology for assessing the plant specific importance of individual phenomenological issues and their model uncertainties to the overall uncertainty. The feasibility and options for developing such assessment methodology should be further explored. Development of such assessment methodology can greatly reduce the effort necessary for a formal propagation of the uncertainty to account for model uncertainty.

It may also be desirable to assign some ranking of "risk importance" among the various phenomenological issues that are considered in a plant PSA model. This paper has presented examples for such risk importance measures. Development of importance measures for phenomenological issues should also be further explored. Such risk importance measures for phenomenological issues can be useful for directing any needed sensitivity analysis (in the absence of any formal model uncertainty analysis), as well as for developing research priorities to reduce the overall model uncertainty.

4. References

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

2. J. M. Griesmeyer and L. N. Smith, "A Reference Manual for the Event Progression Analysis Code (EVNTRE)," Sandia National Laboratories, NUREG/CR-5174, SAND88-1607, September 1989.

3. H. N. Jow, W. B. Murfin and J. D. Johnson, "XSOR Codes User's Manual," Sandia National Laboratories, NUREG/CR-5360, December 1993.

4. R. L. Iman et al., "PARTITION: A Program for Defining the Source Term/Consequence Analysis Interface in the NUREG-1150 Probabilistic Risk Assessments," Sandia National Laboratories, NUREG/CR-5253, SAND88-2940, May 1990.

5. D. I. Chanin et al., "MELCOR Accident Consequence Code System (MACCS)," Sandia National Laboratories, NUREG/CR-4691, Vols. 1-3, SAND86-1562, February 1990.

6. H. P. Nourbakhsh, A. L. Hanson, and W. T. Pratt, "Risk Importance of Containment and Related ESF System Performance Requirements," Brookhaven National Laboratory, NUREG/CR-6418, BNL-NUREG-52489, November 1998.

7. U.S. Nuclear Regulatory Commission, "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance", NUREG-1560, November 1996.

From: George Apostolakis <apostola@mit.edu>
To: <jtl@nrc.gov>
Date: 10/6/02 10:54AM
Subject: P&P MEETING

John:

Item 9: I strongly support the recommendation that Hossein work on the F-C curves. This is a forward-looking subject that could be of great importance to the licensing of future plants.

Hossein's conclusions regarding NUREG-1150 do not surprise me. It is indeed very difficult to produce an overall estimate of model uncertainty. I don't think it's worth pursuing this issue further at this time. Let's wait until Fleming gives us his input.

Item 10: I disagree that we need a facilitator. The ACRS chairman has moderated all the retreats (except the one in Boston). I have not heard any complaints that efficiency has suffered. Even in Boston, a man of Neil Todreas's stature was uncomfortable moderating our sessions. So-called "professional" moderators are very annoying because they don't understand the issues under discussion and all they care about is keeping time.

The Subcommittee cannot just recommend that "adequate time" be provided. The Subcommittee should actually specify the time. I still think that four hours would be sufficient, although some flexibility would be required. I agree that starting the retreat with this item is a good idea. I do believe that Mario's wishes on this matter should be respected, even though the election has not taken place yet.

George

CC: <sxd1@nrc.gov>

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George

From: "Powers, Dana A" <dapower@sandia.gov>
To: "APOSTOLAKIS, George" <apostola@mit.edu>, "BONACA,Mario" <mvbonaca@snet.net>, "FORD, F. Peter" <FPCTFord@aol.com>, "KRESS, T.S." <TSKress@aol.com>, "LEITCH, Graham" <gmleitch@aol.com>, "ROSEN, Steve" <historyart@computron.net>, "SHACK, Bill" <wjshack@anl.gov>, "sieber, jACK" <jdsieber@aol.com>, "WALLIS, Graham B." <Graham.B.Wallis@Dartmouth.edu>
Date: 9/17/02 10:55AM
Subject: Members' Issues for the ACRS Retreat

The current plan for the ACRS retreat relegates issues of ACRS process and procedures to a four hour period - undoubtably at the end of the ordeal. The rest of the time is to be spent on technical issues that really seem to be more appropriate for regular committee meetings. Perhaps it is the case that there are no real members' issues to discuss. But, it does seem worthwhile to discuss this and see if there are issues that might be worthy of more than the cursory treatment allowed in the current agenda for the retreat. Perhaps we should compile a list of issues of process and procedure that could be the subject of discussion at the retreat to see if the list will fit comfortably within the allotted 4 hours. Some issues that I can well imagine include the following:

- ACRS effectiveness and self-assessment

There seems to be a general degradation in the efficiency of the ACRS. We continue to have meetings of prolonged duration that we had when we were generating 6 to 7 letters per meeting. Now we generate about 2-3. When we were producing a larger number of letters we even had an hour for members to work on letters that has now disappeared to support the lower productivity. This seems strange. We hail self-assessment by licensees as a valuable method for improving quality, but we are avoiding application of this technique to ourselves as a full committee or to our subcommittees.

- Proliferation of subcommittee meetings

We seem to have entered an era involving a lot of subcommittee meetings - some of which are only marginally longer than a presentation to the full committee. This comes at a time when the engineering support for subcommittee meetings by experienced ACRS staff engineers is limited because of the transfer of Markely and Dudley. Boehnart keeps threatening to retire which would further detract from the ability of the staff to support multiple subcommittee meetings. Can we continue to operate as we have been?

- Compensation process

A new compensation process has been inflicted on the members that is completely opague. We have no idea what numbers to bill for what work and cannot ascertain what charges have been made.

- Limited evidence of ACRS strategy

There is limited evidence of any strategy by the ACRS. We are back on what Prof. Wallis has so poignantly called the reactive treadmill. We detected weakness of a severe nature in the way NRC staff is doing power uprate reviews, but we did not follow up and, in fact, simply let the staff get away with inadequate reviews for fear of penalizing licensees. We see the staff ignoring technical evidence of high burnup fuel vulnerability, but we

Item 10
31

let them go ahead with this. We are not holding the staff to high technical standards. There is little evidence that the ACRS is searching for issues that others within the agency are overlooking.

- Confusion in Issue Assignments and Subcommittee responsibilities

Issues that based on title seem to belong to one subcommittee are showing up in the domain of other subcommittees. Members who cannot attend subcommittee meetings are being found assigned responsibility for draft letters.

- Differing opinions within the ACRS are not getting aired

There are some areas within risk-informed regulatory plans that there appear to be splits within the ACRS, but these issues get debated electronically rather than within the context of meetings.

I am sure others issues can be identified and it would be of interest to see what these issues are.

Dana

CC: "'sxd1@nrc.gov'" <sxd1@nrc.gov>

Item 10
32

From: "Mario V. Bonaca" <mvbonaca@snet.net>
To: <RANSOM@ECN.PURDUE#062#EDU>, "Powers, Dana A" <dapower@sandia.gov>, "FORD, F. Peter" <FPCTFord@aol.com>, "KRESS, T.S." <TSKress@aol.com>, "LEITCH, Graham" <gmleitch@aol.com>, "ROSEN, Steve" <historyart@computron.net>, "SHACK, Bill" <wjshack@anl.gov>, "sieber, jACK" <jdsieber@aol.com>, "WALLIS, Graham B." <Graham.B.Wallis@Dartmouth.edu>, "George Apostolakis" <apostola@MIT.EDU>
Date: 9/20/02 12:47PM
Subject: Re: Members' Issues for the ACRS Retreat: Mario's Charge

George,

Since we haven't yet elected a new Chaiman, I assume that you are asking me to take care of these proposals because I am currently Vice-Chairman, and as such, I will be glad to do it.

I believe that the issues Dana raises are significant and central to the functioning of the committee. I think that this planning session should have priority on all other business, if any, so it should be held at the beginning of the retreat and last for as long as it takes.
On the other hand, we don't want it to become just a complaining and venting session. So we need to plan an agenda that should start with a strategy session. I propose that at the end of the next P&P, you, John, Tom and I get together in executive session and work on this agenda, to identify topics and assign adequate time to each topic. We will then bring this agenda to the committee and finalize it. Once decided how much time we need for this session, we will fit into the retreat other topics as time allows.

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Mario

----- Original Message -----
From: "George Apostolakis" <apostola@MIT.EDU>
To: "Powers, Dana A" <dapower@sandia.gov>; "BONACA,Mario" <mvbonaca@snet.net>; "FORD, F. Peter" <FPCTFord@aol.com>; "KRESS, T.S." <TSKress@aol.com>; "LEITCH, Graham" <gmleitch@aol.com>; "ROSEN, Steve" <historyart@computron.net>; "SHACK, Bill" <wjshack@anl.gov>; "sieber, jACK" <jdsieber@aol.com>; "WALLIS, Graham B." <Graham.B.Wallis@Dartmouth.edu>
Cc: <sxd1@nrc.gov>
Sent: Tuesday, September 17, 2002 2:16 PM
Subject: Re: Members' Issues for the ACRS Retreat: Mario's Charge

> Mario:
>

Item 10
33

> ! defer to you on what to do with these proposals.
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> George
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> At 08:56 AM 9/17/2002 -0600, Powers, Dana A wrote:
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CC: <sxd1@nrc.gov>, "John Larkins" <JTL@nrc.gov>

Item 10
34.

From: George Apostolakis <apostola@MIT.EDU>
To: "Mario V. Bonaca" <mvbonaca@snet.net>, <RANSOM@ECN.PURDUE#062#EDU>, "Powers, Dana A" <dapower@sandia.gov>, "FORD, F. Peter" <FPCTFord@aol.com>, "KRESS, T.S." <TSKress@aol.com>, "LEITCH, Graham" <gmleitch@aol.com>, "ROSEN, Steve" <historyart@computron.net>, "SHACK, Bill" <wjshack@anl.gov>, "sieber, jACK" <jdsieber@aol.com>, "WALLIS, Graham B." <Graham.B.Wallis@Dartmouth.edu>
Date: 9/20/02 2:16PM
Subject: Re: Members' Issues for the ACRS Retreat: Mario's Charge

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Regarding the location of the retreat: Apparently, some members feel that the ACRS members should not enjoy themselves after hours (surely they don't mean that we will neglect our business and go to the gambling tables). Yucca mountain is not within our charter, yet it is of such major importance to the nuclear enterprise that I don't see how it would harm the committee getting a first-hand look. Besides, you never know what we will be asked to do in the future (see the joint ACNW/ACRS subcommittee).

The Government gets its money's worth from us. The rate of pay is laughable and we definitely put in more than eight hours per day when we are in Rockville. I, for one, am offended when someone hints that I am trying to exploit the government's resources. Finally, I believe that some people underestimate the value of the camaraderie that is created when we all (including spouses) get together in a relaxed environment. Perhaps doing so would contribute to the reduction of the acrimony that we sometimes witness in meetings.

We will make final decisions in October.

George

Item 10

35

At 01:04 PM 9/20/2002 -0500, Mario V. Bonaca wrote:
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>----- Original Message -----
>From: "George Apostolakis" <apostola@MIT.EDU>
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Item 10
36

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CC: <sxd1@nrc.gov>, "John Larkins" <JTL@nrc.gov>

Item 10
37

From: "Mario V. Bonaca" <mvbonaca@snet.net>
To: <RANSOM@ECN.PURDUE#062#EDU>, "Powers, Dana A" <dapower@sandia.gov>, "FORD, F. Peter" <FPCTFord@aol.com>, "KRESS, T.S." <TSKress@aol.com>, "LEITCH, Graham" <gmleitch@aol.com>, "ROSEN, Steve" <historyart@computron.net>, "SHACK, Bill" <wjshack@anl.gov>, "sieber, jACK" <jdsieber@aol.com>, "WALLIS, Graham B." <Graham.B.Wallis@Dartmouth.edu>, "George Apostolakis" <apostola@MIT.EDU>
Date: 9/21/02 3:38PM
Subject: Re: Members' Issues for the ACRS Retreat: Mario's Charge

George,

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Cc: <sxd1@nrc.gov>; "John Larkins" <JTL@nrc.gov>
Sent: Friday, September 20, 2002 1:06 PM
Subject: Re: Members' Issues for the ACRS Retreat: Mario's Charge

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Item 10
38

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Item 10
39

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> >To: "Powers, Dana A" <dapower@sandia.gov>; "BONACA,Mario"
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Item 10
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CC: <SXD1@nrc.gov>, "John Larkins" <JTL@nrc.gov>

Item 10
41

From: George Apostolakis <apostola@MIT.EDU>
To: "Mario V. Bonaca" <mvbonaca@snet.net>, <RANSOM@ECN.PURDUE#062#EDU>, "Powers, Dana A" <dapower@sandia.gov>, "FORD, F. Peter" <FPCTFord@aol.com>, "KRESS, T.S." <TSKress@aol.com>, "LEITCH, Graham" <gmleitch@aol.com>, "ROSEN, Steve" <historyart@computron.net>, "SHACK, Bill" <wjshack@anl.gov>, "sieber, jACK" <jdsieber@aol.com>, "WALLIS, Graham B." <Graham.B.Wallis@Dartmouth.edu>
Date: 9/21/02 3:51PM
Subject: Re: Members' Issues for the ACRS Retreat: Mario's Charge

Mario:

I agree with you.

George

At 03:56 PM 9/21/2002 -0500, Mario V. Bonaca wrote:
>George,
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CC: <SXD1@nrc.gov>, "John Larkins" <JTL@nrc.gov>

Item 10
42

From: "Powers, Dana A" <dapower@sandia.gov>
To: "'sxd1@nrc.gov'" <sxd1@nrc.gov>
Date: 9/17/02 10:29AM
Subject: Comments on the proposed meeting dates

Sam,

You asked for comments on the proposed meeting dates for CY2003. I note that the meeting for June conflicts with the ANS meeting in San Diego. I have to attend these meetings for Sandia now. Shifting the meeting to the next week would be particularly good. If not that, starting the meeting on Thursday would be most helpful.

Dana

CC: "APOSTOLAKIS, George" <apostola@mit.edu>, "BONACA,Mario" <mvbonaca@snet.net>, "FORD, F. Peter" <FPCTFord@aol.com>, "KRESS, T.S." <TSKress@aol.com>, "LEITCH, Graham" <gmleitch@aol.com>, "ROSEN, Steve" <historyart@computron.net>, "SHACK, Bill" <wjshack@anl.gov>, "sieber, jACK" <jdsieber@aol.com>, "WALLIS, Graham B." <Graham.B.Wallis@Dartmouth.edu>

Item 11
43

From: <GMLeitch@aol.com>
To: <mvbonaca@snet.net>, <JDSIEBER@aol.com>, <dapower@sandia.gov>, <graham.b.wallis@dartmouth.edu>, <wjshack@anl.gov>, <historyart@computron.net>, <TSKress@aol.com>, <FPCTFord@aol.com>, <apostola@mit.edu>, <ransom@ecn.purdue.edu>
Date: 9/18/02 11:37AM
Subject: 2003 Meeting Schedule

I notice that there is an ACRS Meeting scheduled for Sept. 3, 4, and 5 2003. Assuming that we would not have a meeting Labor Day week, I scheduled a foreign trip that conflicts with these dates. I could do Sept. 10, 11, 12, and 13 (if Saturday is necessary). Would any of the members have a problem moving back a week? We can discuss this at our October 2002 meeting at which time we plan to finalize the schedule. Thanks Graham L.

CC: <MWW@nrc.gov>, <JNS@nrc.gov>, <AXS3@nrc.gov>, <RPS1@nrc.gov>, <MTM@nrc.gov>, <HJL@nrc.gov>, <JTL@nrc.gov>, <MME@nrc.gov>, <SXD1@nrc.gov>, <NFD@nrc.gov>, <AWC@nrc.gov>, <PAB2@nrc.gov>, <SXB@nrc.gov>

Item 11
44

From: <GMLeitch@aol.com>
To: <dapower@sandia.gov>, <sxd1@nrc.gov>
Date: 9/18/02 11:39AM
Subject: Re: Comments on the proposed meeting dates

Either of Dana's proposals for the June 2003 meeting dates is OK with me.
Graham L.

CC: <apostola@mit.edu>, <mvbonaca@snet.net>, <FPCTFord@aol.com>, <TSKress@aol.com>, <historyart@computron.net>, <wjshack@anl.gov>, <JDSIEBER@aol.com>, <Graham.B.Wallis@Dartmouth.edu>

Item 11
45

From: "Steve Rosen" <historyart@computron.net>
To: <GMLeitch@aol.com>, <mvbonaca@snet.net>, <JDSIEBER@aol.com>, <dapower@sandia.gov>, <graham.b.wallis@dartmouth.edu>, <wjshack@anl.gov>, <TSKress@aol.com>, <FPCTFord@aol.com>, <apostola@mit.edu>, <ransom@ecn.purdue.edu>
Date: 9/18/02 2:20PM
Subject: Re: 2003 Meeting Schedule

I would also prefer September 10, 11 and 12.

----- Original Message -----
From: GMLeitch@aol.com
To: mvbonaca@snet.net. ; JDSIEBER@aol.com ; dapower@sandia.gov ; graham.b.wallis@dartmouth.edu ; wjshack@anl.gov ; historyart@computron.net ; TSKress@aol.com ; FPCTFord@aol.com ; apostola@mit.edu ; ransom@ecn.purdue.edu
Cc: MWW@nrc.gov ; JNS@nrc.gov ; AXS3@nrc.gov ; RPS1@nrc.gov ; MTM@nrc.gov ; HJL@nrc.gov ; JTL@nrc.gov ; MME@nrc.gov ; SXD1@nrc.gov ; NFD@nrc.gov ; AWC@nrc.gov ; PAB2@nrc.gov ; SXB@nrc.gov
Sent: Wednesday, September 18, 2002 10:36 AM
Subject: 2003 Meeting Schedule

I notice that there is an ACRS Meeting scheduled for Sept. 3, 4, and 5 2003. Assuming that we would not have a meeting Labor Day week, I scheduled a foreign trip that conflicts with these dates. I could do Sept. 10, 11, 12, and 13 (if Saturday is necessary). Would any of the members have a problem moving back a week? We can discuss this at our October 2002 meeting at which time we plan to finalize the schedule. Thanks Graham L.

CC: <MWW@nrc.gov>, <JNS@nrc.gov>, <AXS3@nrc.gov>, <RPS1@nrc.gov>, <MTM@nrc.gov>, <HJL@nrc.gov>, <JTL@nrc.gov>, <MME@nrc.gov>, <SXD1@nrc.gov>, <NFD@nrc.gov>, <AWC@nrc.gov>, <PAB2@nrc.gov>, <SXB@nrc.gov>

Item 11
46

From: George Apostolakis <apostola@MIT.EDU>
To: "Steve Rosen" <historyart@computron.net>, <GMLeitch@aol.com>, <mvbonaca@snet.net>, <JDSIEBER@aol.com>, <dapower@sandia.gov>, <graham.b.wallis@dartmouth.edu>, <wjshack@anl.gov>, <TSKress@aol.com>, <FPCTFord@aol.com>, <ransom@ecn.purdue.edu>
Date: 9/18/02 9:54PM
Subject: Re: 2003 Meeting Schedule

I don't mind moving the September meeting to the second week. Please remember that it should be 9/11-13 (Thursday - Saturday). If it stays in the first week, it should be 9/4-6.

George

At 01:20 PM 9/18/2002 -0500, Steve Rosen wrote:
>I would also prefer September 10, 11 and 12.
>----- Original Message -----
>From: <mailto:GMLeitch@aol.com>GMLeitch@aol.com
>To: <mailto:mvbonaca@snet.net.>mvbonaca@snet.net. ;
><mailto:JDSIEBER@aol.com>JDSIEBER@aol.com ;
><mailto:dapower@sandia.gov>dapower@sandia.gov ;
><mailto:graham.b.wallis@dartmouth.edu>graham.b.wallis@dartmouth.edu ;
><mailto:wjshack@anl.gov>wjshack@anl.gov ;
><mailto:historyart@computron.net>historyart@computron.net ;
><mailto:TSKress@aol.com>TSKress@aol.com ;
><mailto:FPCTFord@aol.com>FPCTFord@aol.com ;
><mailto:apostola@mit.edu>apostola@mit.edu ;
><mailto:ransom@ecn.purdue.edu>ransom@ecn.purdue.edu
>Cc: <mailto:MWW@nrc.gov>MWW@nrc.gov ; <mailto:JNS@nrc.gov>JNS@nrc.gov ;
><mailto:AXS3@nrc.gov>AXS3@nrc.gov ; <mailto:RPS1@nrc.gov>RPS1@nrc.gov ;
><mailto:MTM@nrc.gov>MTM@nrc.gov ; <mailto:HJL@nrc.gov>HJL@nrc.gov ;
><mailto:JTL@nrc.gov>JTL@nrc.gov ; <mailto:MME@nrc.gov>MME@nrc.gov ;
><mailto:SXD1@nrc.gov>SXD1@nrc.gov ; <mailto:NFD@nrc.gov>NFD@nrc.gov ;
><mailto:AWC@nrc.gov>AWC@nrc.gov ; <mailto:PAB2@nrc.gov>PAB2@nrc.gov ;
><mailto:SXB@nrc.gov>SXB@nrc.gov
>Sent: Wednesday, September 18, 2002 10:36 AM
>Subject: 2003 Meeting Schedule
>
>I notice that there is an ACRS Meeting scheduled for Sept. 3, 4, and 5
>2003. Assuming that we would not have a meeting Labor Day week, I
>scheduled a foreign trip that conflicts with these dates. I could do Sept.
>10, 11, 12, and 13 (if Saturday is necessary). Would any of the members
>have a problem moving back a week? We can discuss this at our October 2002
>meeting at which time we plan to finalize the schedule. Thanks Graham L.

CC: <MWW@nrc.gov>, <JNS@nrc.gov>, <AXS3@nrc.gov>, <RPS1@nrc.gov>, <MTM@nrc.gov>, <HJL@nrc.gov>, <JTL@nrc.gov>, <MME@nrc.gov>, <SXD1@nrc.gov>, <NFD@nrc.gov>, <AWC@nrc.gov>, <PAB2@nrc.gov>, <SXB@nrc.gov>

Item 11
47

January 2003

ACRS/ACNW CALENDAR YEAR

SUNDAY	MONDAY	TUESDAY	WEDNESDAY	THURSDAY	FRIDAY	SATURDAY
			1 New Years Day (Holiday)	2	3	4
5 PayPeriod 3	6	7	8	9	10	11
12	13	14	15	16	17	18
19 PayPeriod 4	20 Birthday of Martin Luther King Jr. (Holiday)	21	22	23	24	25
HPS 36th Mid Year Topical Meeting (San Antonio,TX)				ACRS Retreat		
26	27	28	29	30	31	

84

9/18/2002

February 2003

ACRS/ACNW CALENDAR YEAR

SUNDAY	MONDAY	TUESDAY	WEDNESDAY	THURSDAY	FRIDAY	SATURDAY
						1 Chinese New Year
2 PayPeriod 5	3	4	5	6 499th ACRS Meeting	7 499th ACRS Meeting	8 499th ACRS Meeting
9	10	11	12 Lincoln's Birthday	13	14 Valentine's Day	15
16 PayPeriod 6	17 Presidents Day (Holiday)	18	19 140th ACNW Meeting	20 140th ACNW Meeting	21 140th ACNW Meeting	22
23 WM03, Tucson, AZ	24 WM03, Tucson, AZ	25 WM03, Tucson, AZ	26 WM03, Tucson, AZ	27 WM03, Tucson, AZ	28 WM03, Tucson, AZ	

49

9/18/2002

March 2003

ACRS/ACNW CALENDAR YEAR

SUNDAY	MONDAY	TUESDAY	WEDNESDAY	THURSDAY	FRIDAY	SATURDAY
						1
2 PayPeriod 7	3	4 500th Celebration	5 Ash Wednesday	6 500th ACRS Meeting	7	8
9	10	11	12	13	14	15
16 PayPeriod 8	17 St. Patrick's Day	18	19	20	21	22
23	24	25 141st ACNW Meeting (CNWRA)	26	27	28	29
30 PayPeriod 9 10th Ntl HLW Conference, Las Vegas, NV	31					

50

9/18/2002

April 2003

ACRS/ACNW CALENDAR YEAR

SUNDAY	MONDAY	TUESDAY	WEDNESDAY	THURSDAY	FRIDAY	SATURDAY
		1	2	3	4	5
		10th Ntl HLW Conference, Las Vegas, NV	10th Ntl HLW Conference, Las Vegas, NV	10th Ntl HLW Conference, Las Vegas, NV		
6	7	8	9	10	11	12
				501st ACRS Meeting	501st ACRS Meeting	501st ACRS Meeting
13 Palm Sunday PayPeriod 10	14	15	16	17 Passover	18 Good Friday	19
			Regulatory Information Conference	Regulatory Information Conference	Regulatory Information Conference	
20 Easter Sunday	21 Patriot's Day	22	23 Secretaries Day	24	25	26
		142nd ACNW Meeting	142nd ACNW Meeting	142nd ACNW Meeting		
27 PayPeriod 11	28	29	30			

May 2003

ACRS/ACNW CALENDAR YEAR

SUNDAY	MONDAY	TUESDAY	WEDNESDAY	THURSDAY	FRIDAY	SATURDAY
				1	2	3
4	5	6	7	8 Natl Academy Board on Radioactive Waste ... 502nd ACRS Meeting	9	10
11 Mother's Day PayPeriod 12	12	13	14	15	16	17
18	19 Victoria Day	20	21	22	23	24
25 PayPeriod 13	26 Memorial Day (Holiday)	27	28 143rd ACNW Meeting	29	30	31

9/18/2002

June 2003

ACRS/ACNW CALENDAR YEAR

SUNDAY	MONDAY	TUESDAY	WEDNESDAY	THURSDAY	FRIDAY	SATURDAY
1 ANS Annual Meeting, San Diego, CA	2	3	4	5	6	7
8 PayPeriod 14	9	10	11 CHANGED FROM 4–6. 503rd ACRS Meeting	12	13	14 Flag Day
15 Father's Day	16	17	18	19	20	21
22 PayPeriod 15	23	24 144th ACNW Meeting	25	26	27	28
29	30					

53

July 2003

ACRS/ACNW CALENDAR YEAR

SUNDAY	MONDAY	TUESDAY	WEDNESDAY	THURSDAY	FRIDAY	SATURDAY
		1 Canada Day	2	3	4 Independence Day (Holiday)	5
6 PayPeriod 16	7	8	9 504th ACRS Meeting	10	11	12
13	14	15	16	17	18	19
20 PayPeriod 17 HPS Society 48th Annual Meeting (San Diego, CA)	21	22	23	24	25	26
27	28	29 145th ACNW Meeting	30	31		

54

9/18/2002

August 2003

ACRS/ACNW CALENDAR YEAR

SUNDAY	MONDAY	TUESDAY	WEDNESDAY	THURSDAY	FRIDAY	SATURDAY
					1	2
3 PayPeriod 18 PayPeriod 18	4 Civic Holiday	5	6	7	8	9
10	11	12	13	14	15	16
17 PayPeriod 19	18	19	20	21	22	23
24	25	26	27	28	29	30
31 PayPeriod 20						

9/18/2002

September 2003

ACRS/ACNW CALENDAR YEAR

SUNDAY	MONDAY	TUESDAY	WEDNESDAY	THURSDAY	FRIDAY	SATURDAY
	1 Labor Day (Holiday)	2	3 Natl Academy Board on Radioactive Waste ...	4 Natl Academy Board on Radioactive Waste ...	5	6
7	8	9	10	11 CHANGED 505th ACRS Meeting	12 FROM 3-5 505th ACRS Meeting	13 505th ACRS Meeting
14 PayPeriod 21	15	16 146th ACNW Meeting (Las Vegas, NV)	17 146th ACNW Meeting (Las Vegas, NV)	18 146th ACNW Meeting (Las Vegas, NV)	19	20
21	22	23	24	25	26	27 Rosh Hashanah
28 Payperiod 22	29	30				

9/23/2002

25

October 2003

ACRS/ACNW CALENDAR YEAR

SUNDAY	MONDAY	TUESDAY	WEDNESDAY	THURSDAY	FRIDAY	SATURDAY
			1	2 506th ACRS Meeting	3 506th ACRS Meeting	4 506th ACRS Meeting
5	6 Yom Kippur	7	8	9	10	11 Sukkot
12 PayPeriod 23	13 Columbus Day (Holiday)	14	15	16 National Bosses Day	17	18
19	20	21 147th ACNW Meeting	22 147th ACNW Meeting	23 147th ACNW Meeting	24	25
26 PayPeriod 24	27	28	29	30	31 Halloween	

9/18/2002

57

November 2003

ACRS/ACNW CALENDAR YEAR

SUNDAY	MONDAY	TUESDAY	WEDNESDAY	THURSDAY	FRIDAY	SATURDAY
						1
2	3	4 Election Day	5	6 507th ACRS Meeting	7 507th ACRS Meeting	8 507th ACRS Meeting
9 PayPeriod 25	10	11 Veterans Day (Holiday)	12	13	14	15
16	17	18 148th ACNW Meeting	19 148th ACNW Meeting	20 148th ACNW Meeting	21	22
23 PayPeriod 26	24	25	26	27 Thanksgiving Day Holiday)	28	29
30						

9/18/2002

85

December 2003

ACRS/ACNW CALENDAR YEAR

SUNDAY	MONDAY	TUESDAY	WEDNESDAY	THURSDAY	FRIDAY	SATURDAY
	1	2	3 Natl Academy Board on Radioactive Waste ...	4 508th ACRS Meeting	5 508th ACRS Meeting	6 508th ACRS Meeting
7 PayPeriod 1	8	9	10	11	12	13
14	15	16	17	18	19	20 Hanukkah
21 PayPeriod 2	22	23	24	25 Christmas Day (Holiday)	26 Kwanza	27
28	29	30	31			

9/18/2002

59

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

October 18, 2001

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: LETTER FROM DR. KENNETH D. BERGERON REGARDING TENNESSEE VALLEY AUTHORITY LICENSE AMENDMENT REQUEST TO ALLOW TRITIUM PRODUCTION AT THE WATTS BAR NUCLEAR POWER PLANT

The purpose of this memorandum is to forward information received from Dr. Kenneth D. Bergeron, formerly associated with Sandia National Laboratories, and now a member of the public, concerning the Tennessee Valley Authority license amendment request to allow tritium production at the Watts Bar Nuclear Power Plant. Dr. Bergeron has raised issues concerning tritium production in commercial nuclear power plants and deterministic versus risk-informed approach in NRC review process. He encourages use of probabilistic methods to supplement traditional analysis in evaluating the Watts Bar license amendment request.

I understand that the staff is reviewing the issues in Dr. Bergeron's letter. The ACRS would like to be kept informed of the staff's disposition of this matter.

Attachment:
Letter dated September 13, 2001, from Kenneth D. Bergeron to Dana A. Powers, Advisory Committee on Reactor Safeguards

cc w/atts: A. Vietti-Cook, SECY
J. Craig, EDO
I. Schoenfeld, EDO
S. Collins, NRR
B. Sheron, NRR
R. Correia, NRR
M. Padovan, NRR
A. Thadani, RES

cc w/o attach: K. Bergeron

Item 14
60

Kenneth D. Bergeron, PhD
17 Tierra Monte NE
Albuquerque, NM 87122
e-mail: kenberg@flash.net

September 13, 2001

Dr. Dana Powers
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Mailstop T-2 E26
Washington, DC 20555

Dear Dana,

Enclosed is a letter I just sent to Brian Sheron at NRR. Its message is intended as much for the ACRS as for him, since I believe that the staff reviewing the Watts Bar License Amendment Request will need guidance from the most senior levels of the NRC to understand that probabilistic methods should play a central role in their review.

If the ACRS or one of its subcommittees includes the Watts Bar LAR in a future meeting agenda, please let me know (e-mail is fine), as I would like to have some input into the meeting.

Sincerely,

Ken Bergeron

61

Kenneth D. Bergeron, PhD
17 Tierra Monte NE
Albuquerque, NM 87122
e-mail: kenberg@flash.net

September 13, 2001

Dr. Brian W. Sheron
NRR/ADPT
US Nuclear Regulatory Commission
Mailstop O-5 E7
Washington, DC 20555

Dear Dr. Sheron,

I am writing to you about the ongoing staff review of TVA's License Amendment Request that would allow production of tritium at the Watts Bar plant. I have a specific suggestion in that regard, but before getting into it, I'd like to re-introduce myself to you. In the late 1980s, I worked for you as a manager of one of the groups at Sandia doing research on severe accidents. My group's principal focus was the CONTAIN code and performing studies with it for NRC. I remember a number of very stimulating meetings with you after you took over the severe accident program for RES. Around 1989, not long after you moved into RES, I got out of NRC work in order to manage Sandia's support to DOE's New Production Reactor, which was intended to replace Savannah River's K reactor as the source of tritium for the US nuclear arsenal. For a containment specialist like me, this was a very exciting time, because the government and its industry partners on the Heavy Water design were committed to building the most severe-accident-proof containment in history. I had the job of coordinating severe-accident-related work at Sandia, Argonne, Brookhaven and Savannah River, and it was very satisfying to be able to apply some of the lessons from TMI to the design of a reactor that was actually going to be built (or so we thought).

All that changed in 1992 when progress on nuclear arms reductions allowed President Bush to defer the tritium production program (the reason being that the tritium from decommissioned weapons could be used to replenish the weapons that remained in the arsenal). I then found other work at Sandia in international programs, but in 1994 Nestor Ortiz asked me to return to his program and manage all NRC work on severe accident computer codes. So I was responsible for not only CONTAIN, but also MELCOR, VICTORIA, IFCI, RADTRAD (actually an NRR project) and a number of analysis projects for RES and NRR. I continued in that role until I retired in 1999 after 25 years at Sandia.

This little biography is relevant to the Watts Bar LAR because it shows that I'm pretty knowledgeable about tritium production and severe reactor accidents, particularly from the perspective of containment. It turns out, too, that I know quite a bit about TVA's ice condenser plants, since they were a big focus for the CONTAIN project during the 62

Containment Performance Improvements program in the '80s, and since one of the last projects I worked on at Sandia was the project to resolve DCH for Ice Condensers. In that project I found myself in the unusual position of actually doing the CONTAIN calculations for the project leader, Marty Pilch. This is because most of the people who knew how to run CONTAIN had left the program or retired.

My professional experience with ice condensers and tritium production lead me to have grave misgivings about DOE's plans to obtain weapons tritium by having TVA produce it in the normal course of electricity production at their Watts Bar and Sequoyah plants. I believe that the modifications to the reactor and the added mission for the nuclear management team at TVA will add significantly to the already serious safety problems with these plants. I will, of course, detail the reasons for my concerns in my comments to the licensing Project Manager, Mark Padovan. What I want to ask you is on a higher level than such details. I want to encourage you to insist that the powerful new tools of Risk-Informed Regulation be brought to bear fully on this license amendment.

I was alarmed to see the schedule Mr. Padovan distributed at the August 20 meeting at White Flint. He showed the NRC review process being complete by early March 2002. Such a compressed schedule is completely inconsistent with a thorough assessment even if no element of Reg Guide 1.174 is brought to the review. As an aside, if the schedule is said to have actually begun in April 2001 well I have to cry "foul," since in May I asked NRC by e-mail when the LAR was expected and was informed that it would not be until late summer. I had asked to be kept informed about this and received no notification until Padovan e-mailed me on August 13 about the August 20 meeting.

In other recent public information, NRC has indicated they were planning for a yearlong review, so perhaps I should not focus too much on Padovan's handout. But what that document suggests to me is that the staff is assuming that this license amendment will be reviewed only via deterministic methods, with no additional insight brought in from risk methods.

For this LAR, I strongly encourage you to take full advantage of the authority the Commission has given your staff to use probabilistic methods to supplement the incomplete picture that traditional analysis provides. There are many important reasons:

1. For most containment types, Design Basis analysis is not a bad surrogate for assessing the overall level of protection that the containment adds to the safety of the plant. For ice condensers, the DBA is almost irrelevant as a test for robustness. The ice does a great job with a DEGB LOCA, if you ever were to have one, but it has the effect of increasing hydrogen concentrations in more risk-significant accidents, making the real safety problems worse. Put simply, it is impossible to gauge the effectiveness of the ice condenser containment system with traditional deterministic analysis.

2. It is also impossible to evaluate the true effect of the core modifications on the safety of the plant via deterministic analysis. It is my guess that the principal effect will be on the complexity of fuel handling, and that new event pathways will be important contributors to increased risk. I also think that the likelihood of accidents induced by

63

sabotage may be increased because of the plant's new defense mission. Obviously, only level II PRA can address such effects.

3. A significant source of added risk is the burden that this new military mission places on the overall management of the plant. There will be many new ways that management commitment to a safety culture at the plant could be compromised. A top-rung utility might be able to rise to such challenges and ensure that the commitment to safety remains the highest priority, but TVA has shown itself not to be in this class. Moreover, TVA's motivation for cooperating with DOE in this partnership is troubling. Most knowledgeable observers believe that TVA is cooperating only because by becoming effectively a part of the nuclear weapons complex the agency will be less vulnerable to those in Congress who for years have been trying to disband and privatize it. The conflicted motivational situation at the highest management level does not bode well for maintaining an adequate safety culture at the plant. It may be difficult to assess the subtle effects of compromised management commitment, but we all know that such effects are real and can be large. It is incumbent upon the NRC to address the issue, and it is only through risk methods that this can be done.

4. Normally, the staff might hesitate to apply risk methods when the licensee doesn't volunteer such analyses, because the NRC has a responsibility to avoid imposing unnecessary burdens on the licensee. The streamlining of many processes and regulations in recent years has been motivated by this philosophy because of the concern that over-regulation might threaten the viability of the nuclear industry itself. Such reasoning is irrelevant in this case. The nuclear industry gets no benefit from these changes (in fact, I believe it will be damaged by it in the long run because of public concerns about mixing military and civilian missions). The cost of the LAR and its review is not coming from ratepayers but from the DOE, which is saving billions by not having to build a dedicated production facility.

5. Time is not of the essence. DOE's schedule for producing tritium by 2005 is a ridiculous exaggeration. It ignores the arms reductions dictated by START-II, which has been ratified by both Russia and the US. The respected physicist Frank von Hippel (former Assistant Director for National Security at OSTP) estimates that we won't really need new tritium until 2029 or later.

6. This is an extraordinarily sensitive Federal interagency issue. Never before have two giant agencies, each with complex agendas quite different from NRC's, joined forces to demand concurrence from your licensing organization on an operating license change. All possible resources should be made available to your reviewers, and the overall process should come under the most intense scrutiny by senior management and the Commission itself. I believe firmly that this license amendment request satisfies the criterion cited in RIS 2001-02, that the change "could create 'special circumstances' under which compliance with existing regulations may not produce the intended or expected level of safety and plant operation may pose an undue risk to public health and safety." Therefore use of risk-informed methods is appropriate. I would go farther and say that not to use the much-vaunted RG-1.174 methods in these extraordinary circumstances would be irresponsible in the highest degree. It would

64

certainly strengthen the case of critics who see risk-informed regulation as nothing but a way for licensees to be relieved of any safety requirements they dislike.

I recognize that the NRC is in a very uncomfortable position because of this License Amendment Request. But the recent, terrible events of this week show only too clearly that the price of regulatory complacency can be incalculably high. I suggest to you that the only rational way for you to proceed is cautiously, using the best scientific tools available.

I would be glad to discuss this matter with you or your staff further, if you so desire.

I have taken the liberty of sharing this letter with some of my former colleagues who are members of the ACRS.

Sincerely,

Ken Bergeron

Kenneth Bergeron

Copies to:

D. Powers
T. Kress
G. Apostolakis

65

September 10, 2002

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

FROM: Herbert N. Berkow, Project Director
Project Directorate II /RA/
Division of Licensing Project Management

SUBJECT: LETTER FROM DR. KENNETH D. BERGERON REGARDING TENNESSEE VALLEY AUTHORITY'S LICENSE AMENDMENT REQUEST TO PRODUCE TRITIUM AT THE WATTS BAR NUCLEAR POWER PLANT

Your memorandum of October 18, 2001, to William D. Travers forwarded Dr. Kenneth D. Bergeron's letter of September 13, 2001. Dr. Bergeron was concerned about the ongoing U.S. Nuclear Regulatory Commission's review of Tennessee Valley Authority's (TVA's) license amendment request to irradiate tritium-producing burnable absorber rods (TPBARs) at the Watts Bar nuclear plant. You requested that the Advisory Committee on Reactor Safeguards be kept informed of the staff's disposition of this matter.

Attached is Brian W. Sheron's letter of September 6, 2002, to Dr. Bergeron responding to his letters of September 13, 2001, and January 16, 2002. We expect to issue license amendments to TVA this month in response to TVA's amendment requests of August 20, and September 21, 2001, to irradiate TPBARs in the Watts Bar and Sequoyah reactors.

Attachment: Letter to Dr. Bergeron dated 9/6/02

cc w/ attachment:
A. Vietti-Cook, SECY
J. Craig, EDO
I. Schoenfeld, EDO
A. Thadani, RES

CONTACT: L. Mark Padovan, NRR
415-1423

Item 14
66

September 6, 2002

Dr. Kenneth D. Bergeron
17 Tierra Monte NE
Albuquerque, NM 87122

SUBJECT: NRC STAFF RESPONSE TO YOUR SUGGESTIONS TO RISK-INFORM THE REVIEW OF THE SEQUOYAH AND WATTS BAR TRITIUM PRODUCTION LICENSE AMENDMENT REQUESTS

Dear Dr. Bergeron:

I am responding to your letters of September 13, 2001, and January 16, 2002, requesting that we risk-inform our process for reviewing Tennessee Valley Authority's (TVA's) license amendment requests to produce tritium at Sequoyah and Watts Bar, and expressing other safety concerns. We reviewed your written requests and evaluated your concerns expressed during the November 7, 2001, meeting held at One White Flint North.

As you are aware, RIS-2001-002 "Guidance on Risk-Informed Decision Making in License Amendment Reviews," addresses our process for determining when requests for risk information are justified as part of our review of a license amendment request. We conducted a technical assessment of the issues you identified following the guidance in RIS-2001-002. We were not able to substantiate that there would be a significant increase in risk if the U.S. Nuclear Regulatory Commission (NRC) approved TVA's amendment requests. However, we elevated your concerns to the risk informed licensing panel (RILP) even though our staff's assessment did not identify any issues that would raise questions about TVA's ability to maintain adequate protection of public health and safety. The RILP convened on July 11, 2002, and unanimously agreed that gathering additional risk information to evaluate TVA's amendment requests was not necessary. However, in our July 29, 2002, letter to TVA, we did ask TVA to send us some risk-informed background information to confirm our decision. In your email of August 10, 2002, to Mark Padovan of the NRC, you asked for a copy of TVA's response to the staff's request for information. TVA's August 9, 2002, response is enclosed.

Your letters noted numerous safety concerns. NRC staff considered each of your concerns against the guidance of RIS-2001-002, but grouped the concerns into the following broad categories:

- historical safety performance of Sequoyah and Watts Bar

- postulated increased risk from internal events, external events, and security concerns stemming from the dual-purpose civilian and military-related uses of the TVA reactors

- potential ice condenser plant design vulnerabilities to severe accident conditions, in particular, under station blackout (SBO) scenarios

67

- other issues, such as NRC's legal authority to issue the amendments, Advisory Committee on Reactor Safeguards (ACRS) participation in the amendment reviews, and more time for public comments

The staff's assessment of your concerns is provided below.

Regarding your concerns about TVA's performance, the staff does not use overall plant performance as a criterion for approving amendment requests. The NRC's reactor oversight process (ROP) continuously monitors licensee performance to provide assurance that licensees are operating plants safely and in accordance with the regulations and licensing bases. The ROP allows for a graded, predictable agency response commensurate with licensee performance. This can result in agency actions up to and including ordering the plant to shut down should NRC determine performance to be unacceptable.

The ROP relies on objective performance indicators (PIs) along with risk-informed inspections using 39 inspection procedures to monitor and evaluate plant performance. As discussed in the most recent Annual Assessment Letters for Watts Bar and Sequoyah, the results of the PIs and inspections are in the "licensee response band" of the ROP Action Matrix. This means that both plants have acceptable performance that does not require additional oversight beyond the baseline level of inspection. Plant performance results are available for public view on the NRC's external website at http://www.nrc.gov/NRR/OVERSIGHT/ASSESS/index.html.

You also expressed concern that Watts Bar may not be capable of operating with a tritium production core. The staff notes that TVA has successfully demonstrated its ability to irradiate and handle tritium producing burnable absorber rods (TPBARs). TVA successfully irradiated 32 lead test assemblies for one cycle as part of TPBAR efficacy testing at Watts Bar. Therefore, the staff does not have any basis to question TVA's capability to manage such a change.

You postulated several new accident scenarios in your letter of January 16, 2002. In particular, you were concerned that a TPBAR ejection was not evaluated. Each TPBAR has a threaded end plug that is connected to a hold-down plate. The TPBAR is also secured in place via a crimping device as described in TVA's submittal of August 20, 2001. The TPBARs are inserted into fuel assemblies, similar to traditional burnable poison rod assemblies, and do not contain fissile material. Immediately above the fuel assemblies containing TPBARs is the upper core plate and reactor vessel upper internals package. Therefore, the staff does not agree that a realistic scenario exists for TPBARs to be ejected, or that there is a significant increase in initiating event transients.

You also noted that you believed it was not an appropriate neutronic practice to offset, by soluble boron poisoning, additional reactivity from higher fuel enrichment. Changes to the core design and core reactivity issues will be fully addressed in the staff's safety evaluation. However, the staff did not identify in its deterministic design basis review any reactivity issues that would warrant probabilistic treatment of TVA's amendment requests.

You suggested that a potential TPBAR drop accident "during the TPBAR consolidation process" was not adequately addressed. You note that the TPBAR drop accident could occur with the plant at full power. The rod consolidation process is performed in the spent fuel pool and, as such, does not increase the likelihood of a reactor trip. From a dose perspective, TVA addressed dropping a TPBAR and NRC staff evaluated it for (1) fuel movement in the reactor cavity and (2) spent fuel pool operations. All of the fuel rods in an irradiated fuel assembly, and 24 TPBARs, are assumed to rupture, releasing the radionuclides within the fuel-clad gap to the fuel pool or reactor cavity water. TVA's analyses show the offsite consequences of this event are well within Title 10, *Code of Federal Regulations* (10 CFR), Part 100, dose guidelines. A complete radiological assessment of potentially dropped TPBARs will be addressed in the staff's safety evaluations.

Previous performance issues with the ice condenser system were also noted in your January 16, 2002, letter. For example, you noted problems with lower inlet door binding for both plants. These issues have been corrected, and the staff is not aware of any recent door failures due to floor upheaval/door binding within the past few years. The lower inlet doors continue to be tested in accordance with each plant's Technical Specifications and are monitored under several licensee programs, including the regulatory-required 10 CFR 50.65 maintenance rule program. More important, there is no direct nexus between a change in the core design and any effect on the reliability or availability of the ice condenser system. Therefore, overall, given no demonstrated significant increase from the baseline core damage frequency, and no demonstrated significant change in containment systems performance, the staff could not substantiate that there would be a significant increase in the baseline severe accident large early release frequency because of tritium production.

In your letters, and during the November 7, 2001, meeting, you noted concerns that safeguards measures at Sequoyah and Watts Bar may be inadequate once tritium production begins at these stations, especially in view of the events of September 11, 2001. The NRC and its licensees have taken a number of actions following the terrorist attack of September 11, 2001, to increase security at NRC-licensed facilities, including a heightened security stance pursuant to safeguards advisories. On February 25, 2002, the NRC issued Orders to all commercial nuclear power plants to implement interim compensatory measures for the current threat environment. Some of the requirements made mandatory by the Orders formalize the security measures that NRC licensees had taken in response to NRC's advisory letters. The specific actions are sensitive, but generally include requirements as follows:

- increased patrols
- augmented security forces and capabilities
- additional security posts
- installation of additional physical barriers
- checks at greater stand-off distances
- enhanced coordination with law enforcement and military authorities
- restrictive site access for all personnel

- additional security measures pertaining to waterways and the owner-controlled land outside the plants' protected areas

During our meeting of November 7, 2001, you raised a specific terrorist scenario against Watts Bar. Further, you alluded to this postulated vulnerability in your letter of January 16, 2002. Although the exact scenario you described is not evaluated in the plant's Updated Final Safety Analysis Report (UFSAR), the effects of the scenario had been analyzed for design basis considerations and are documented in the UFSAR. Under such a scenario, the specific plant structures and systems of interest to your concern are protected from such a phenomenon. The analysis used bounding design-basis assumptions and conditions beyond the nominal conditions that would be present from the scenario that you postulated. This phenomenon was also evaluated in the licensee's individual plant examination (IPE) of external events submittal. The staff concludes that the outcome of the scenario you postulated during our meeting and in your letter is not credible.

On the matter of the NUREG/CR-6427, "Assessment of the DCH [Direct Containment Heating] Issue for Plants with Ice Condenser Containments," the staff is in the process of resolving Generic Safety Issue (GSI)-189, "Susceptibility of Ice Condenser Plants and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident." Although NUREG/CR-6427 notes a 0.97 conditional containment failure probability (CCFP) for Sequoyah under SBO conditions, this CCFP value results from assumptions that were appropriate for resolving the Direct Containment Heating issue and must be used in the correct context. The NUREG gives no credit for offsite power recovery, and provides no evaluation of recovery of one of several simultaneously failed emergency diesel generators. The NUREG also does not reflect plant improvements since the licensee's original IPE submittal that reduces the frequency of SBO and reduces the likelihood of core damage during SBO conditions such as the following:

- maintaining high emergency diesel generator reliability
- a maintenance rule program
- high-temperature reactor coolant pump seals
- modifications to the turbine driven auxiliary feedwater operation procedures
- improved emergency operating procedures

More realistic treatment of SBO scenarios would probably reduce the core damage frequency, containment failure frequency, and CCFP. Also, the tritium amendment requests would not result in an increase in core damage frequency or large early release frequency above the current values. The CCFP value, as it stands today, is appropriate for its intended purpose of resolving the direct containment heating issue and use as a screening value for GSI-189 regulatory backfit analysis.

In summary, the staff evaluated your suggestions and concerns against the special circumstances criteria noted in RIS-2001-002 and against standards defined in NRC Regulatory Guide 1.174. The staff was not able to substantiate that there would be a significant increase in risk of internal or external events because of tritium production. The staff concluded this primarily because a tritium production core in itself does not:

- increase the likelihood of an initiating event
- affect the probabilistic risk assessment (PRA) success criteria
- affect the functionality, availability, or reliability of equipment and structures necessary to prevent core damage (Level I PRA) or mitigate core damage effects (Level II PRA)

The staff determined that the only salient issue relevant to Sequoyah and Watts Bar is GSI-189, which is unaffected by TVA's amendment requests. The Office of Nuclear Regulatory Research is completing GSI-189 regulatory analysis, and will forward it to Office of Nuclear Reactor Regulation for final resolution. However, the staff does not believe that approving the amendment requests depends on resolving GSI-189 for reasons previously noted.

You commented on the NRC's legal authority to issue the amendments in light of 42 USC 7272. This very issue was analyzed by the Atomic Safety and Licensing Board in the recent consolidated tritium license amendment proceedings. In a decision issued on July 2, 2002 (LBP-02-14), the Board concluded that Public Law 106-65, section 3134(a), which provides that the Secretary of Energy shall produce tritium at Watts Bar or Sequoyah, and its legislative history "clearly show that Congress intended for the NRC to entertain" TVA's tritium license amendment applications, notwithstanding 42 USC 7272. Thus, there should be no doubt that the NRC has the legal authority to issue the amendments.

The ACRS determines what involvement it will have reviewing licensing actions. It received your letter of October 18, 2001, on the subject of allowing tritium production at Watts Bar. The ACRS has not asked to participate in the review of TVA's amendment requests, but wanted to be informed of our response to you. Accordingly, we are forwarding a copy of this letter to the ACRS.

You also suggested that the NRC should allow more than 30 days for public comment on the staff's proposed no significant hazards consideration determinations. On January 15, 2002, Mr. David Lochbaum of the Union of Concerned Scientists sent us a letter requesting a 60-day extension of the public comment period. The letter of January 17, 2002, from the Secretary of the Commission, denied that request. However, the Secretary's letter said that the NRC staff would consider additional comments as it received them while reviewing other comments. Likewise, we continued to assess the information in your letters of September 13, 2001, and January 16, 2002, and are now responding to your concerns.

71

We appreciate your comments and suggestions regarding the amendment requests for tritium production and we hope that our response addresses your concerns. Please feel free to contact L. Mark Padovan at (301) 415-1423 or me should you have any questions.

Sincerely,

/RA/

Brian W. Sheron, Associate Director
for Project Licensing and Technical Analysis
Office of Nuclear Reactor Regulation

Enclosure: TVA letter to NRC dated 8/9/02

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72

TVA

Tennessee Valley Authority, 1101 Market Street, Chattanooga, Tennessee 37402-2801

August 9, 2002

10 CFR 50.90

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Gentlemen:

In the Matter of	)	Docket Nos.50-327, 328, 390
Tennessee Valley Authority	)	

SEQUOYAH (SQN) AND WATTS BAR (WBN) NUCLEAR PLANTS - REQUEST FOR RISK-INFORMED INFORMATION RE: TRITIUM PRODUCTION PROGRAM (TAC NO. MB1884)

The purpose of this letter is to respond to NRC questions provided in a letter dated July 29, 2002. This information is being provided to support the ongoing NRC review of WBN and SQN License Amendment Requests submitted by TVA on August 20, 2001, and September 21, 2001, respectively. TVA has separated the responses into two enclosures. Enclosure 1 provides the SQN responses. Enclosure 2 provides the WBN responses.

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73

There are no regulatory commitments made by this letter. The delay in submitting this information was coordinated via telecon with the NRC staff on August 7, 2002. If you have any questions, please contact me at (423) 751-2508.

Sincerely,

Mark J. Burzynski

Mark J. Burzynski
Manager
Nuclear Licensing

Subscribed and sworn to before me
on this 9th day of August

Lillian G. Martin
Notary Public

My Commission expires 1-25-03

Enclosures
cc: See page 3

74

cc (Enclosures):

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75

ENCLOSURE 1
SEQUOYAH NUCLEAR PLANT (SQN)
RESPONSES

1. **Please provide the SQN maintenance rule program (a)(2) performance criteria for the following systems:**

   A. Emergency Diesel Generators (EDGs)
   B. Turbine Driven Auxiliary Feedwater Pump
   C. Emergency 125 VDC Supply
   D. Emergency 120 VAC Supply
   E. Hydrogen Igniters
   F. Containment Air Return Fans
   G. Emergency Raw Cooling Water (ERCW)
   H. Ice Condenser

   TVA RESPONSE

   The maintenance rule program (a)(2) performance criteria for the systems listed above is as follows:

   A. Emergency Diesel Generators - Please note that the term Valid Failure is equivalent to Functional Failure (FF) and Valid Test is equivalent to valid Demand.

   - Unavailability - No more than 2.5% for each DG average over a rolling 24 months (438 hrs/24 months).

   - Function Level Unreliability - The DG target reliability of 97.5% is met provided the following trigger values are not reached:
     - 3 combined functional failures (FFs) (start demand and/or load run demand) out of 20 combined demands (all DGs combined)
     - 4 combined FFs out of 50 combined demands (all DGs combined)
     - 5 combined FFs out of 100 combined demands (all DGs combined)
     - 4 FFs out of 25 demands (for each DG)

   - Component Level Unreliability - No more than 2 Component (Pump) Failures (CFs) per Fuel Oil Transfer Pump per rolling 24 months.

76

B. Turbine Driven Auxiliary Feedwater Pump

- Unavailability - No more than 2.5% per train or 219 hrs/year, based on a 24 month rolling average when risk significant.

- Unreliability - No more than 1 FF per 24 months per train.

C. Emergency 125 VDC Supply

- Unavailability - No more than 0.194% or 17 hours/year, based on a 12 month rolling average (all modes and all Outage Risk Assessment Management (ORAM) states).

- Unreliability - No more than one FF of a vital battery or vital battery board per 24 months.

D. Emergency 120 VAC Supply

- Unavailability - No more than 16.4% or 60 days/year, based on a 12 month rolling average (all modes and all ORAM states).

- Unreliability - No more than four FFs of a 120 VAC vital instrument power board per 24 months.

E. Hydrogen Igniters

- Unavailability - No more than 0.95% average unavailability per unit during a rolling 24-month interval when risk significant (Modes 1&2). The function is unavailable whenever there are no functional igniters in one or more of the 34 zones.

- Unreliability - No more than 1 FF per unit during a rolling 24-month interval. A FF in Modes 1 & 2 is 1) a loss of two igniters in the same zone, or 2) a loss of any combination of three or more igniters in any combination of zones. When in State 11 or 12, a FF is the loss of either Train A or Train B.

F. Containment Air Return Fans

- Unavailability - No more than 0.28% per train every 24 months when risk significant (Modes 1&2 and ORAM States 1&2).

- Unreliability - No more than one FF per train every 24 months. A FF is defined as a failure of the train to start or operate as required.

G. Emergency Raw Cooling Water (ERCW)

- Unavailability - Train Level - No more than 2.7% per train per 24-month rolling average.

- Unreliability

  - Train/Functional Level - No more than two FFs per train per 24 months.

  - Component (ERCW Pump) Level - No more than one failure per pump per 24 months.

H. Ice Condenser

- Unavailability - In Mode 1, no actual unplanned capability loss events attributable to the ice condenser system are permitted in a rolling 24 month interval. In Modes 1 and 2 or ORAM states 1 and 2, no unavailability that if it had occurred at 100% power, it would have caused a greater than 20% power loss.

- Unreliability - No failure of a required flow path is permitted in a rolling 24 month interval.

- Condition -

  - No more than one failure to maintain the ice bed temperature at or below 27°F during Modes 1 and 2, ORAM States 1 and 2, and States 11 and 12 when required is permitted in a rolling 24 month period.

78

- No failure to maintain the design basis ice mass is permitted in a rolling 24 month interval when required.
- No failure to maintain the minimum sodium tetraborate concentration and proper range of pH as defined in LCO 3.6.5.1.a is permitted in a rolling 24 month interval.

2. **Are any of the above systems currently in maintenance rule program (a)(1) status and if so why?**

TVA RESPONSE

None of the systems listed in Item 1 are currently in maintenance rule program (a)(1) status.

3. **How many EDG failures (failure-to-start and failure-to-run) have occurred in the previous 100 starts for each of the EDGs?**

TVA RESPONSE

As of June 30, 2002, the number of EDG valid failures which have been recorded for the last 100 starts are as follows:

Generator	Number of Failures
EDG 1A	1
EDG 1B	0
EDG 2A	6*
EDG 2B	0

*This data is consistent with the response to Question 2. As indicated in the response to Question 1, the trigger criteria for each individual Sequoyah EDG is 4 FF out of 25 demands. The maximum number of valid failures per 25 demands EDG 2A has reached in the past is 2.

4. **Are any of the above EDG failures a common-mode failure of the SQN EDGs (i.e. were the other EDGs actually**

unavailable because the root cause of the failed EDG also actually affected the other EDGs)?

TVA RESPONSE

None of the EDG failures listed in Item 3 resulted from a common-mode failure.

5. **Do all reactor coolant pumps (RCPs) at SQN have the newer style high-temperature O-ring seals? If not, how many do not and on which unit? For those RCPs that do not have the new O-ring design, what is the schedule to replace them?**

TVA RESPONSE

All Sequoyah RCPs currently have the high temperature O-ring seals installed.

6. **Does SQN conduct Severe Accident Management Guidelines (SAMG) drills and how often?**

TVA RESPONSE

Severe Accident Management Guidelines (SAMG) training for SQN emergency preparedness teams normally consists of classroom instruction and a table top drill and are conducted annually with the teams being trained based on a four year rotation.

7. **How many failures of the ice condenser lower inlet doors have occurred during the previous two operating cycles (i.e. did not meet technical specifications surveillance requirements)? Are any of these failures attributed to floor upheaval/buckling causing door binding? Does Tennessee Valley Authority continue to monitor ice condenser floor growth from cycle-to-cycle?**

TVA RESPONSE

Surveillance instructions performed during the past two refueling outages, Cycle 10 and Cycle 11, for both Sequoyah Unit 1 and Unit 2 were evaluated for failures. Based on the data packages reviewed, all lower inlet doors met the Technical Specification surveillance requirements.

No lower inlet door surveillance requirement failures during the specified time period were due to floor upheaval/buckling.

SQN continues to monitor ice condenser floor movement during operation under Procedure No. 0-PI-SXX-061-001.0 "Ice Condenser Lower Plenum Floor Monitoring". This Instruction provides detailed steps for monitoring vertical movement of the ice condenser lower plenum floor to ensure lower inlet door operability.

8. Please provide the following information for SQN based on the current PRA model:

A. total core damage frequency (CDF) from internal events
B. total CDF from external events (if modeled)
C. percentage of CDF due to station blackout
D. loss of offsite power frequency and basis
E. probabilities of non-recovery of offsite AC power for various times in the model and basis for numbers used.
F. probability of EDG/emergency AC bus recovery (if modeled) and the basis for the number(s)

TVA RESPONSE

Based on Revision 01 of the Sequoyah Probabilistic Safety Analysis (PSA) model, the requested information has been established as follows.

A. The total CDF from internal events is 3.77E-05/yr.

B. The total CDF from external events has not been quantified. In the IPEEE (Individual Plant Examination for External Events) no vulnerabilities from external events were identified.

C. The percentage of CDF due to station blackout is 10.5%.

D. The loss of offsite power frequency is 0.0485/yr based on a Baysian update of generic industry data using site specific experience.

E. The probability of non-recovery from a loss of offsite AC power at 1 hour is 0.255. At 1 hour the steam generator secondary side inventory is depleted when no makeup is available. The probability of non-recovery at 1.7 hours is 0.604. At 1.7 hours core damage occurs when no secondary side makeup is available. The probability of non-recovery at 4 hours is 0.275. At 4 hours the station batteries are depleted. These non-recovery probabilities are based on the information in NUREG/CR-5032.

F. The probability of EDG/emergency AC bus recovery within 1.7 hours of 1/1 EDG is 0.39 and of 1/2 EDGs is 0.536. The probability of recovery within 4 hours of 1/1 EDG is 0.60 and of 1/2 EDGs is 0.80. The basis for these probabilities is a site specific EDG recovery model. This model is described in detail in Section 3.3.3.4.3.2 of the individual plant evaluation.

9. **What are the normal and emergency power supplies for the ERCW (intake structure) sump pumps?**

TVA RESPONSE

All of the sump pumps at the ERCW pumping station are powered from the various ERCW Motor Control Center (MCC) boards. The building basement sump pumps (not safety related) are powered from the MCC in their respective bays, the deck sump pump 1A is powered from the 1A ERCW 480v MCC, the deck sump pump 1B is powered from the 1B ERCW 480v MCC. All of the ERCW 480v MCC receive power from the 6.9 Kv Shutdown Boards, and are therefore Diesel backed. The deck sump pumps are safety related and remain loaded to the Diesel after blackout, the building basement sump pumps are non-safety related and are therefore load-stripped upon blackout.

82

ENCLOSURE 2
WATTS BAR NUCLEAR PLANT (WBN)
RESPONSES

1. **Please provide the WBN maintenance rule program (a)(2) performance criteria for the following systems:**

   A. Emergency Diesel Generators (EDGs)
   B. Turbine Driven Auxiliary Feedwater Pump
   C. Emergency 125 VDC Supply
   D. Emergency 120 VAC Supply
   E. Hydrogen Igniters
   F. Containment Air Return Fans
   G. Emergency Raw Cooling Water (ERCW)
   H. Ice Condenser

   TVA RESPONSE

   The maintenance rule program (a)(2) performance criteria for the systems listed above is as follows:

   A. Emergency Diesel Generators

   - Unavailability

     - No more than 2% for each DG averaged over a rolling 24 months (approximately 350 hours/24 months).

     - No more than 0.1% for the fuel oil transport support function for each EDG set averaged over a rolling 24 months (approximately 17 hours/24 months).

   - Unreliability

     - No more than 1 failure of any of the fuel oil transfer pumps within a 24-month period.

     - Unreliability performance criteria for the EDG function is based on trigger values established as a result of 10CFR50.63. Nuclear Engineering established a target reliability of 97.5%. These trigger values are used as unreliability performance criteria for the Maintenance Rule as follows:

83

- 3 combined functional failures (FFs) (start demand and/or load run demand) out of 20 combined demands (all DGs combined)
- 4 combined FFs out of 50 combined demands (all DGs combined)
- 5 combined FFs out of 100 combined demands (all DGs combined)
- 4 FFs out of 25 demands (for each DG)

B. Turbine Driven Auxiliary Feedwater Pump

- Unavailability - No more than 2% per train or 350 hours/24 months based on a 24 month rolling average.

- Unreliability - No more than two FFs per train in a 24-month interval.

C. Emergency 125 VDC Supply

- Unavailability (Battery Board) - No unavailability of the boards are allowed during power operation (0 hours). Additionally, no unavailability is planned at other times. This does not include swapping the battery with the spare battery, which includes a momentary loss of backup power.

- Unreliability - No more than one FF of a vital battery or vital battery board per 24 month period.

D. Emergency 120 VAC Supply

- Unavailability - No more than 0.274% or 48 hours/inverter/24 months interval. The inverters are not required available during certain pre-analyzed conditions during outages.

- Unreliability - No more than one FF per channel per 24-month interval.

E. Hydrogen Igniters

- Unavailability - No more than 7 days (168 hours) during a 24 month period (Modes 1 & 2). The system will be considered unavailable during periods in which there are no functional igniters in one or more of the 34 zones.

- Unreliability - No more than one FF within a 24-month interval. Functional failure is defined as any failure or combination thereof that results in the loss of ignition capability in any of the 34 zones.

- Supplemental component level performance criteria is no more than three igniter failures in a 24-month interval.

F. Containment Air Return Fans

- Unavailability - No more than 1% per train per 24-months (approximately 175 hrs/train/24-months) reporting period.

- Unreliability - No more than one FF per train per 24-month interval. Functional failure is defined as a failure of the fans to start or operate as required.

G. Emergency Raw Cooling Water (ERCW)

- Unavailability - The train unavailability performance criteria for modes 5 and 6 is 1.4% (approximately 245 hours/24-months). Risk considerations preclude the elective removal of either ERCW train from service during power operation. However, routine pump surveillance testing involves cross-tying of the trains for brief periods. The test instructions have been reviewed against the requirements for operator recovery from planned maintenance. It was determined that cross-tying of trains for performance of the pump test does not require maintenance rule unavailability.

- Unreliability -

    - Train Level - No FFs per train within a 24-month interval.

    - Component level - No more than three component failures within a 24-month interval (ERCW pumps, strainers, and traveling water screens).

H. Ice Condenser

- Unavailability - No unplanned capability loss attributable to the ice condenser is permitted in a rolling 24-month interval.

- Unreliability -

    - No FF due to loss of the minimum required flow path through the ice bed within an operating cycle.

    - No FFs within an operating cycle where the minimum total ice mass is found to be less than that specified by the Technical Specification, and

    - No instances within an operating cycle in which the average boron concentration or pH of the sample is found to be less than that specified by the Technical Specification

- Condition

    - Not more than one failure to maintain the mean ice bed temperature below 27°F is permitted within a 24 month interval.

2. Are any of the above systems currently in maintenance rule program (a)(1) status and if so why?

TVA RESPONSE

The Auxiliary Feedwater (AFW) system is in (a)(1) status. However, this is due to a start logic issue on the motor driven AFW pumps which has since been resolved. At this time, this equipment is being monitored for removal from (a)(1) status which is projected for 4th quarter FY03. The Turbine Driven AFW Pump is not in (a)(1) status.

3. **How many EDG failures (failure-to-start and failure-to-run) have occurred in the previous 100 starts for each of the EDGs?**

TVA RESPONSE

As of June 30, 2002, the number of EDG valid failures which have been recorded for the last 100 starts are as follows:

Generator	Number of Failures
DG 1A-A:	1
DG 1B-B:	0
DG 2A-A:	1
DG 2B-B:	0

4. **Are any of the above EDG failures a common-mode failure of the WBN EDGs (i.e., were the other EDGs actually unavailable because the root cause of the failed EDG also actually affected the other EDGs)?**

TVA RESPONSE

None of the EDG failures listed in Item 3 resulted from a common-mode failure.

5. **Do all reactor coolant pumps (RCPs) at WBN have the newer style high-temperature O-ring seals? If not, how many do not and on which unit? For those RCPs that do not have the new O-ring design, what is the schedule to replace them?**

TVA RESPONSE

All Watts Bar RCPs currently have the high temperature O-ring seals installed.

87

6. **Does WBN conduct Severe Accident Management Guidelines drills and if so how often?**

TVA RESPONSE

Severe Accident Management Guidelines (SAMG) training for WBN emergency preparedness teams normally consists of classroom instruction and a table top drill and are conducted annually with the teams being trained based on a four year rotation.

7. **How many failures of the ice condenser lower inlet doors have occurred during the previous two operating cycles (i.e. did not meet technical specifications surveillance requirements)? Are any of these failures attributed to floor upheaval/buckling causing door binding? Does Tennessee Valley Authority continue to monitor ice condenser floor growth from cycle-to-cycle?**

TVA RESPONSE

Surveillance instructions performed during the 3rd and 4th refueling outages were reviewed. Both of these performances were successfully completed with no doors failing their Technical specifications requirements.

No lower inlet door surveillance requirement failures during the specified time period were due to floor upheaval/buckling.

WBN performs a maintenance instruction 1-STRU-661-5000, "Ice Condenser Wear Slab Floor Inspection," each refueling outage which monitors floor growth to ensure that any floor movement does not impair the opening of the lower inlet doors and prevent them from fulfilling their accident function.

8. **Please provide the following information for WBN based on the current PRA model for each plant:**

   A. total core damage frequency (CDF) from internal events
   B. total CDF from external events (if modeled)
   C. percentage of CDF due to station blackout
   D. loss of offsite power frequency and basis
   E. probabilities of non-recovery of offsite AC power for various times in the model and basis for numbers used.

88

F. probability of EDG/emergency AC bus recovery (if modeled) and the basis for the number(s)

TVA RESPONSE

Based on Revision 2A of the Watts Bar Probabilistic Safety Analysis (PSA) model, the requested information has been established as follows:

A. The total CDF from internal events of 4.48E-5/yr.

B. The CDF from external events is not currently modeled in the WBN-PSA. In the IPEEE (Individual Plant Examination for External Events) no vulnerabilities from external events were identified.

C. WBN has not calculated the percentage of CDF due to Station Blackout, we do calculate the percentage of CDF due to Loss of Offsite Power (LOOP) which is 14% of the CDF.

D. The loss of offsite power frequency is 0.0259/yr based on a Baysian update of a generic industry data using site specific experience. Specifically, WBN has experienced no LOOP.

E. The non-recovery of the 161-kv Grid for WBN has a mean value of 0.255. Offsite power can also be restored to the WBN systems through the Unit 2 500-KV grid. The non-recovery of the Unit 2 500-KV grid has a mean value of 0.205.

A Monte Carlo simulation is used in the electric power recovery analysis at WBN. This recovery analysis for the WBN PSA model is an integrated, time dependent model that looks at several parameters and conditions. These parameters include the recovery of offsite power, the recovery of one or two diesels, and the availability of auxiliary feedwater for heat removal. The result of the recovery analysis is a recovery factor that is the ratio of two conditional frequencies, given a LOOP initiating event: the conditional frequency of the loss of onsite power in a mission time of 24 hours and the failure to restore onsite or offsite power before core damage occurs, and the conditional frequency of onsite power failure in a 24 hour period without recovery. Factors that influence the time available to restore AC power include the availability of 125V DC power (i.e., battery lifetime) and the length of time to core damage due to pump seal

89

leakage or power-operated relief valve (PORV) discharge following a loss of all onsite AC power.

The time to recover off-site power at nuclear power plants has been documented in NUREG/CR-5032, "Modeling Time to Recovery of Loss of Off-site Power at Nuclear Power Plants." The Model for Group I2 in this NUREG was chosen as best representing WBN and was used in the WBN recovery analysis.

F. As described above, a Monte Carlo simulation (PLG STADIC program is used in the electric power recovery (offsite and DG) analysis at WBN.

Some of the assumptions used in this time dependent model are:

- The diesels generators are assumed to be unrecoverable after the depletion of the DC batteries
- The turbine-driven AFW pump is also assumed to be unavailable after DC control power is lost

Examples of the non-recovery factors used for various conditions is provided in the following table:

Case	Number Of Unit 1 Diesel Generators Available For Recovery	Unit 1 Dgs Known To Initially Be In Maintenance	Auxiliary Feedwater Available	Operators Cooldown And Depressurize RCS	Number Of Recoverable Unit 1 Plus Unit 2 Diesel Generators	Probability Of Onsite Power Failure And Offsite Nonrecovery	Diesel Generator Unavailability	Sequence Recovery Factor
1	2	Unknown	Yes	No	2	1.17525-4	8.30762-3	5.5072-2
2	2	Unknown	Yes	Yes	2	8.73629-5	8.30762-3	4.09382-2
3	2	Unknown	No	N/A	2	6.21547-4	8.30762-3	0.291257
4	1	Unknown	Yes	No	1	9.14894-4	6.48421-2	5.4847-2
5	1	Unknown	Yes	Yes	1	6.75282-4	6.48421-2	4.04825-2
6	1	Unknown	No	N/A	1	4.95071-3	6.48421-2	0.29679
7	0	Unknown	Yes	No	0	3.7518-2	1.0	0.14712
8	0	Unknown	Yes	Yes	0	2.6999-2	1.0	0.105879
9	0	Unknown	No	N/A	0	0.159522	1.0	0.625577

90

9. **What are the normal and emergency power supplies for the ERCW (intake structure) sump pumps?**

TVA RESPONSE

There are two sump pumps per ERCW Strainer Room. The normal power supplies for the pumps in ERCW Strainer Room A are from the safety-related Control & Auxiliary Building Vent Board 1A1-A and 2A1-A (respectively). The normal power supplies for the pumps in ERCW Strainer Room B are from the safety-related Control & Auxiliary Building Vent Board 1B1-B and 2B1-B (respectively).

These boards receive diesel power; however, these pumps are load shed from their respective board in the event of Loss of Offsite Power.

91