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

May 29, 2008

MEMORANDUM TO: Annette L. Vietti-Cook
Secretary of the Commission

A. Dias for

FROM: Frank P. Gillespie, Executive Director **/RA/**
Advisory Committee on Reactor Safeguards

SUBJECT: ACRS MEETING WITH THE U.S. NUCLEAR REGULATORY COMMISSION, - JUNE 5, 2008, SCHEDULE AND BACKGROUND INFORMATION

The ACRS is scheduled to meet with the U.S. Nuclear Regulatory Commission between 1:30 and 3:30 p.m. on Thursday, June 5, 2008 to discuss the topics listed below. Background materials related to these items are attached.

TOPICS		PRESENTERS	PRESENTATION TIME
1.	Overview	William J. Shack ACRS Chairman	10 minutes
2.	Safety Research Program	Dana A. Powers	10 minutes
3.	Digital I&C Matters	George E. Apostolakis	10 minutes
4.	State-of-the-Art Reactor Consequence Analyses (SOARCA)	William J. Shack	10 minutes
5.	ESBWR Design Certification	Michael L. Corradini	10 minutes
6.	Extended Power Uprates and Related Technical Issues	Mario V. Bonaca	10 minutes

Attachment: As stated

Note: Presentation time does not include time for Commissioners' questions and answers by ACRS members

U.S.NRC

United States Nuclear Regulatory Commission

Protecting People and the Environment

ACRS MEETING WITH THE U.S. NUCLEAR REGULATORY COMMISSION

June 5, 2008

OVERVIEW

William J. Shack

Accomplishments

- Since our last meeting with the Commission on June 7, 2007, we issued 29 Reports:
- Topics included:
  - Review and evaluation of the NRC Safety Research Program
  - Quality assessment of selected NRC research projects

– Selected Chapters of the ESBWR design certification application
– State-of-the-Art Reactor Consequence Analyses (SOARCA) Project
– Digital I&C research project plan and interim staff guidance
– Dissimilar metal weld issue in pressurizer nozzles

- Cable Response to Live Fire (CAROLFIRE) Testing and Fire Model Improvement Program
- AREVA Detect and Suppress Stability Solution and Methodology
- License Renewal, Extended Power Uprate, and Early Site Permit Applications

New Plant Activities

- Established design-specific Subcommittees
- Reviewed technology-neutral licensing framework for future plant designs
- Performed interim review of the Vogtle early site permit application
- Reviewed proposed licensing strategy for Next Generation Nuclear Plant (NGNP)

- Reviewing the SER for the ESBWR design certification application, chapter-by-chapter, as requested by the staff. Provided interim letters on several Chapters
- Interacting with NRO staff periodically to establish schedule for ACRS review of design certification and COL applications to ensure timely completion of ACRS review

License Renewal

- Completed review of three license renewal applications (Vermont Yankee, Pilgrim, Fitzpatrick)
- Completed interim review of two applications (Wolf Creek and Shearon Harris)
- Will complete final review of two applications and interim review of three applications (Indian Point, Vogtle, Beaver Valley) during the remainder of CY 2008

- Recent license renewal applications have exhibited a trend toward an increasing number of exceptions to the Generic Aging Lessons Learned (GALL) Report
- In future updates of the GALL Report, the staff plans to incorporate alternative approaches used by the industry and approved by the staff to reduce the number of exceptions to the GALL Report

Radiation Protection and Nuclear Materials Issues

- No issues carried over from ACNW&M to ACRS
- New Subcommittee to be established to focus on radiation protection and nuclear materials issues

Ongoing/Future Activities

- Advanced reactor design certifications
- Combined license applications
- Design Certification applications
- Digital instrumentation and control systems

- Early site permit application (Vogtle)
- Extended power uprates
- Fire protection
- High-burnup fuel and cladding issues
- Human reliability analysis
- License renewal applications
- Next generation nuclear plant (NGNP) project

- Operating plant issues
- PWR sump performance issue
- Report on the NRC Safety Research Program
- Research Quality Assessment
- Resolution of Generic Safety Issues
- Revisions to Regulatory Guides and SRPs
- Risk-Informing the Regulations
- Safeguards and security matters

- State-of-the-Art Reactor Consequence Analyses (SOARCA) Project
- Waste management, radiation protection, decommissioning, and materials issues

NRC SAFETY RESEARCH PROGRAM

Dana A. Powers

Scope

- The current safety research projects organized by the Office of Nuclear Regulatory Research (RES)
- The long-term, sustained research at the NRC
- Research on security and safeguards, nuclear materials, and waste management not addressed

General Observation

- The current safety research program is well focused in support of near term regulatory activities of NRC line organizations

- The research program is generally aligned with the DOE/Nuclear Industry Strategic Plan for LWR R&D
  - Greater use of risk information
  - Support the development of a regulatory process for deployment of DI&C technology
  - Improve understanding of materials degradation and plant aging
  - Higher fuel burnup

Advanced Non-LWR Research

- An appropriate level of research activity for advanced reactor concepts:
  - Gas-cooled reactors
  - Liquid metal-cooled reactors

International Collaboration

- The current research program is making good use of international collaborations:

  - Severe accident research
  - Fire research
  - Seismic research
  - Human reliability research

Long-Term Research

- The challenge posed by a re-energized nuclear industry in the U.S.
- RES must address HOW NRC staff will work in the future not just WHAT issues staff will have to address

- International collaborations offer opportunities to the NRC to develop over the longer term its capabilities in the areas of advanced reactor safety as well as the safety of allied technologies

DIGITAL I&C MATTERS

George E. Apostolakis

ACRS Report, October 16, 2007

- The staff's three ISGs on diversity and defense in depth, communications, and human factors will help with the review of anticipated near-term licensing actions related to digital I&C

- In the longer term, the staff should develop an alternative process to the 30-minute criterion to determine the conditions under which operator manual actions can be credited as a diverse protective function

ACRS Report, April 29, 2008

- The draft ISG on the Review of New Reactor DI&C PRAs should be revised to emphasize the importance of the identification of failure modes, deemphasize sensitivity studies that deal with probabilities, and discuss the current limitations in DI&C PRAs

ACRS Report, May 19, 2008

- NUREG/CR-6962, Approaches for Using Traditional PRA Methods for Digital Systems, should be revised before publication to state clearly that its methods do not address software failures and that it employs simulation in addition to traditional PRA methods. The revised NUREG/CR report should focus on failure mode identification only

- The staff should establish an integrated program that focuses on failure mode identification of DI&C systems and takes advantage of the insights gained from the investigations on traditional PRA methods and on advanced simulation methods

- The quantification of the reliability of DI&C systems should be deferred until a good understanding of the failure modes is developed

The Committee will continue to provide its views to the Commission on the staff’s activities related to digital I&C

STATE-OF-THE-ART REACTOR CONSEQUENCE ANALYSES

William J. Shack

ACRS REPORT, FEBUARY 25, 2008

- Level-3 PRAs should be performed for the pilot plants before extending the analyses to other plants. The PRAs should address the impact of mitigative measures using realistic evaluations of accident progression and offsite consequences. The core damage frequency (CDF) should not be the basis for screening accident sequences

- The process for selecting the external event sequences in SOARCA needs to be made more comprehensive. The impacts from these events on containment mitigation systems, operator actions, and offsite emergency responses should be evaluated realistically

- Consequences should be expressed in terms of ranges calculated using the threshold recommended by the Health Physics Society Position Statement and some lower thresholds. A calculation with linear, no-threshold (LNT) should also be performed, which would facilitate comparison with historical results

ACRS Letter to EDO, April 21, 2008

- The staff did not agree with the ACRS recommendation that a limited set of level-3 PRAs be performed to benchmark the SOARCA approach developed by the staff

- The Committee continues to believe that the credibility of the SOARCA Project cannot rely on confidence in the judgment of the staff and on a novel analysis procedure that differs substantially from previous state-of-the-art analyses of the consequences of severe reactor accidents

ESBWR DESIGN CERTIFICATION

Michael L. Corradini

Design Features

- Direct-cycle power conversion system
- Natural circulation in the reactor vessel
- Passive emergency core cooling system
- Passive containment cooling

- Severe Accident Mitigation
  - Core retention device in the lower drywell
  - Passive drywell flooding
- ESBWR does not need emergency AC power for 72 hours after a transient or accident

Design Certification Review

- Reviewing the SER with open items for the ESBWR design certification chapter-by-chapter, as requested by the staff, to aid effective resolution of ACRS issues
- Completed interim review of 15 SER chapters during three full committee meetings and six Subcommittee meetings
- Issued three interim letters (November 20, 2007, March 20, and May 23, 2008)

Some Committee Issues

- Further examine system interactions
- Address containment response to design basis accidents
- Develop sound technical basis for performance of passive systems
- Assure proper operation of the vacuum breaker system
- Confirm coupled neutronic and thermal-hydraulic stability, including interactions between the core and chimney

Future Plans

The ACRS will:

- Perform interim review of the remaining SER chapters
- Review the staff's resolution of open items and ACRS issues
- Review the final SER and issue a final report to support the Agency schedule

EXTENDED POWER UPRATES AND RELATED TECHNICAL ISSUES

Mario V. Bonaca

EPU Review Status

- Completed **review of** EPU applications for Susquehanna Units 1&2 (20%) and Hope Creek (17%)

- Will review EPU applications for Browns Ferry Units 1, 2, & 3 (20%) and Millstone Unit 3 (7%)

EPU Technical Issues

- Steam Dryer Integrity

- Containment Overpressure Credit

- Validation of Analytical Methods

Steam Dryer Integrity

- Dryer Integrity Resolutions
  - Steam dryer replacement / Instrumentation
  - Use of new and evolving analytical methods
  - Installation of branch lines
  - Reliance on careful power ascension testing

- Only Quad Cities Unit 2 and Susquehanna Unit 1 steam dryers instrumented

- Other licensees measure steam line strain data and depend on analytical acoustic-circuit model to infer steam dryer pressure loads

- To date, acoustic circuit model was benchmarked only against Quad Cities Unit 2 measured pressures
- This is limited validation for model addressing such a complex set of conditions
- ACRS accepted Hope Creek EPU application steam dryer evaluations in part because of predicted large margin to the stress limit

Containment Overpressure Credit

- For some plants, demonstrating adequate NPSH for safety systems for EPU operation requires:
  - Containment backpressure credit
  - Termination of drywell cooling to maximize backpressure

- ACRS Position - Overpressure credit may be granted in small amounts and only for short duration when the risk is low

- Staff Position – No limit in amount of credit granted and duration is needed, provided it is supported by conservative backpressure calculations

Browns Ferry Unit 1 Containment Overpressure Credit

- For Browns Ferry Units 1, 2, & 3 EPU (20%) Appendix R scenario, containment backpressure credit of up to 9.3 psig needed for 69 hours
- Drywell cooling is terminated to maximize available backpressure
- Margin between available and required backpressure is as low as 1.6 psi

- In the February 16, 2007 Browns Ferry Unit 1 report on 5% power uprate, ACRS recommended that granting of containment overpressure credit during long-term loss-of-coolant accident and 10 CFR Part 50 Appendix R fire scenarios at 120-percent of the original licensed thermal power will require support by more complete evaluations

- Viable solutions minimizing need for overpressure credit
  - Protect a second RHR train for Appendix R scenario
  - Use best-estimate calculation, with appropriate uncertainty and biases applied
  - Use more rigorous risk assessment for fire scenario to demonstrate low risk

Validation of Analytical Methods

- Susquehanna EPU and applicability of core response analysis methods at EPU conditions were reviewed concurrently
  - ACRS expressed concern regarding treatment of uncertainties and biases in methods
  - Staff took exception to ACRS recommendation and accepted limited sensitivity analysis

- Reduced margin to thermal limits for EPU operation warrants re-evaluation of the fidelity of the analytical methods, codes, and the supporting validation data

Abbreviations

AC	Alternating current
ACNW&M	Advisory Committee on Nuclear Waste & Materials
ACRS	Advisory Committee on Reactor Safeguards
CAROLFIRE	Cable Response to Live Fire (Testing Program)
CDF	Core damage frequency
COL	Combined license
CY	Calendar year
DBA	Design-basis accident
DI&C	Digital instrumentation and control
DOE	Department of Energy
EDO	Executive Director for Operations
EPU	Extended Power Uprate
ESBWR	Economic Simplified Boiling Water Reactor
GALL	Generic Aging Lessons Learned (Report)
I&C	Instrumentation & control
ISG	Interim staff guidance
LNT	Linear, no-threshold
LWR	Light water reactor
NGNP	Next Generation Nuclear Plant
NPSH	Net positive suction head
NRC	Nuclear Regulatory Commission
NRO	Office of New Reactors
PRA	Probabilistic risk assessment
PSIG	Pounds per square inch gauge
PWR	Pressurized water reactor
RES	Office of Nuclear Regulatory Research
RHR	Residual heat removal
R&D	Research & development
SRP	Standard Review Plan
SER	Safety evaluation report
SOARCA	State-of-the-Art Reactor Consequence Analyses
U.S.	United States

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See previous concurrence
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DATE	05/29/08	05/29 /08

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