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

May 21, 2007

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

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

SUBJECT: ACRS MEETING WITH THE U.S. NUCLEAR REGULATORY COMMISSION, JUNE 7, 2007 - SCHEDULE AND BACKGROUND INFORMATION

The ACRS is scheduled to meet with the U.S. Regulatory Commission between 1:30 and 3:30 p.m. on Thursday, June 7, 2007 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	20 minutes
2.	Framework for Future Plant Licensing	Thomas S. Kress	10 minutes
3.	Digital I&C Activities	George E. Apostolakis	10 minutes
4.	License Renewal/Extended Power Uprates	Mario V. Bonaca	10 minutes
5.	Human Reliability Analysis Models	George E. Apostolakis	10 minutes

Attachment: As stated

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

ACRS MEETING WITH THE U.S. NUCLEAR REGULATORY COMMISSION

June 7, 2007

OVERVIEW

William J. Shack

Accomplishments

- **Since our last meeting with the Commission on October 20, 2006, we issued 24 Reports:**
- **Topics included:**
  - **Draft Final Rule to Risk-Inform 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors"**

- **Draft Final Regulatory Guide DG-1145, "Combined License Applications for Nuclear Power Reactors"**
- **Draft Final NUREG-1824, "Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications"**
- **Development of the TRACE Thermal-Hydraulic System Analysis Code**

- **Draft Final Revision 3 to Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident"**
- **Development of an Integrated Long-term Regulatory Research Plan**
- **License Renewal and Extended Power Uprate Applications**

Future Plant Designs

- Established design-specific Subcommittees
- Completed review of revisions to high-priority SRP Sections and Regulatory Guides
- Reviewing ESBWR probabilistic risk assessment
- Reviewing the licensing framework for future plant designs

The ACRS will:

- **Perform pre-application review of the EPR design**

- **Review SER for the ESBWR design certification, chapter-by- chapter, as requested by the staff**

- **Review Vogtle early site permit application**

Dissimilar Metal Weld Issue

- Support staff and industry agreement on the resolution of pressurizer nozzle weld issues
  - Allow the final nine plants to complete inspection and mitigation activities in spring 2008, contingent on additional industry analysis results

- **Industry developing advanced finite element analysis to provide basis for leak-before-break**

- **Licensees committed to enhanced leakage detection as compensatory action**

- **Staff should encourage industry to perform inspections prior to mitigation activities**

- **Plan to review results of advanced finite element analysis when available**

Ongoing/Future Activities

- **Advanced reactor design certifications**
- **Assessment of research quality**
- **Combined license applications**
- **Commission paper on rulemaking to make risk-informed change to loss-of-coolant accident technical requirements, 10 CFR 50.46a**

- **Digital instrumentation and control systems**
- **Early site permit applications**
- **Extended power uprates**
- **Fire protection**
- **High-burnup fuel and cladding issues**
- **Human reliability analysis**
- **License renewal applications**

- **Operating plant issues**
- **Report on the NRC Safety Research Program**
- **Resolution of GSI-191, "Assessment of Debris Accumulation on PWR Sump Performance"**
- **Revisions to Regulatory Guides**
- **Risk-Informing 10 CFR Part 50**
- **Safeguards and security matters**

- **SPAR models development program**
- **State-of-the-art reactor consequence analysis**
- **Technology-neutral regulatory framework**
- **Thermal-hydraulic codes**

UNITED STATES
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ACRSR-2223

November 16, 2006

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

SUBJECT: DRAFT FINAL RULE TO RISK-INFORM 10 CFR 50.46, "ACCEPTANCE CRITERIA FOR EMERGENCY CORE COOLING SYSTEMS FOR LIGHT-WATER NUCLEAR POWER REACTORS"

Dear Chairman Klein:

During the 537th meeting of the Advisory Committee on Reactor Safeguards, November 1-3, 2006, we met with representatives of the NRC staff and the Boiling Water Reactor (BWR) Owners' Group to discuss the draft final rule to risk-inform 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," (the Rule). We also had the benefit of the documents referenced.

RECOMMENDATIONS

1. The Rule to risk-inform 10 CFR 50.46 should not be issued in its current form. It should be revised to strengthen the assurance of defense in depth for breaks beyond the transition break size (TBS). Such assurance would reduce concerns about uncertainties in determining the TBS.

2. The revision of draft NUREG-1829, "Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process," to include changes resulting from the resolution of public comments should be completed before the revised Rule is issued. This state-of-the-art review on the estimation of break size frequencies is an essential part of the technical basis for the Rule.

3. The interpretation that the Rule limits the total increase in core damage frequency (CDF) resulting from all changes in a plant that adopts the Rule to be "small" (i.e., $<10^{-5}$/yr) represents a significant departure from the current guidance for risk-informed regulation and should be reviewed for its implications.

DISCUSSION

In response to a Staff Requirements Memorandum (SRM) dated July 1, 2004, the staff has developed an alternative set of risk-informed requirements for emergency core cooling systems (ECCS). Licensees may voluntarily choose to comply with these requirements in lieu of meeting the existing requirements in 10 CFR 50.46. The Rule divides the spectrum of LOCA break sizes into two regions. The demarcation between the two regions is called a "transition break size." The first region includes small breaks up to and including the TBS. The second region includes breaks larger than the TBS up to and including the double-ended guillotine break (DEGB) of the largest reactor coolant system pipe.

Because pipe breaks in the smaller break size region are considered more likely than pipe breaks in the larger break size region, each region would be subject to different ECCS requirements. Loss-of-coolant accidents in the smaller break size region would be analyzed using the methods, assumptions, and criteria currently used for LOCA analysis; accidents in the larger break size region would be analyzed using less stringent methods, assumptions, and criteria due to their lower likelihood of occurrence. Although LOCAs for break sizes larger than the TBS would become "beyond design-basis accidents," the Rule requires that licensees maintain the ability to mitigate all LOCAs up to and including the DEGB of the largest reactor coolant system pipe.

The fundamental principles of a risk-informed regulation should be to ensure that any increases in risk associated with a change are small, that changes are consistent with the defense-in-depth philosophy, and that adequate safety margins are maintained. Regulatory Guide 1.174 provides quantitative criteria for assessing changes in risk, but its guidance on ensuring consistency with the defense-in-depth philosophy and maintaining adequate safety margins is more subject to engineering judgment.

Probabilistic risk assessments of internal events typically show that large-break LOCAs (LBLOCAs) are relatively small contributors to CDF. The results in draft NUREG-1829 suggest that the contribution to CDF from breaks larger than the TBS proposed in the Rule is a small fraction of the already small contribution to CDF due to all LBLOCAs. Thus, the requirements for mitigation capabilities for breaks beyond the TBS should be based on defense-in-depth considerations to provide margin against unanticipated degradation phenomena, human errors, extremely large loads such as those associated with earthquakes beyond the safe shutdown earthquake, and other unanticipated events. The degree of defense in depth required can only be determined by judgment based on experience and best attempts to quantify uncertainties.

The Rule requires an analysis to demonstrate mitigation for breaks greater than the TBS, up to the DEGB of the largest pipe in the reactor coolant system. The requirements in the Rule provide a degree of assurance of this mitigation. It is our judgment, however, that the Rule should impose additional requirements to strengthen this assurance.

Because the Rule now defines pipe breaks greater than the TBS as "beyond design basis," any equipment required solely to mitigate such breaks may no longer be safety-related and could be subject to less stringent maintenance and inspection requirements that could adversely affect its reliability. Such equipment could even be removed from technical specifications that control its availability. We agree that the low likelihood of breaks greater than the TBS justifies a relaxation in the requirements for mitigating such events, but this relaxation should instead result from the removal of additional requirements that make such events even more unlikely, such as the simultaneous loss-of-offsite-power (LOOP) and the assumption of the worst single failure. Confidence in the reliability and availability of the equipment needed to mitigate such breaks is important not only for defense in depth, but also for maintaining safety margins for breaks smaller than the TBS.

The Rule also provides restrictions on the unavailability of the non-safety-related equipment needed to mitigate breaks beyond the TBS, but it imposes no other requirements. We believe that the equipment needed to mitigate these breaks deserves some special treatment and control. The staff has dealt with the regulatory treatment of non-safety systems in other contexts, and similar approaches would be appropriate here.

The Rule should also increase confidence in the ability to mitigate breaks greater than the TBS by requiring licensees to submit the codes used for the analyses of breaks beyond the TBS to the NRC for review and approval.

The Rule is an enabling rule that will permit licensees to make changes that increase operational flexibility and reduce regulatory burden, which could result in increases or decreases in risk. The Rule contains a risk-informed change process that will control all changes in risk that occur after a licensee adopts the Rule. The risk-informed change process in the Rule uses the current 10 CFR 50.59 change process and the 10 CFR 50.65 maintenance rule categorization to screen changes that can impact risk. However, as currently envisioned by the staff, it allows the licensee in some cases to implement changes that have a ΔCDF greater than 10^{-6}/yr but less than 10^{-5}/yr without prior review by the staff. Regulatory Guide 1.174 would typically allow such changes only if the total CDF, including external events and low-power/shutdown events, is less than 10^{-4}/yr. Licensees should submit such changes to the staff for prior review and approval. Licensees could still implement changes that result in a ΔCDF < 10^{-6}/yr without prior review and should track the quantified changes in CDF in the 24 month report.

The Rule requires that the total increase in CDF resulting from all changes in a plant that adopts the Rule be "small" (i.e., < 10^{-5}/yr). This "cap" on the increase in risk applies regardless of whether the changes in CDF result from changes related to 10 CFR 50.46. This represents a significant departure from the current guidance for risk-informed regulation and should be reviewed for its implications.

Maintaining sufficient safety margin is another important element of risk-informed regulation that is not treated quantitatively in Regulatory Guide 1.174. It is likely that, with this Rule, the NRC will find requests for additional power uprates at pressurized water reactors (PWRs) acceptable. However, the uprates will clearly decrease safety margins, even for breaks below the TBS. The Rule currently contains acceptance criteria for fuel cladding performance under LOCA conditions based on the current 10 CFR 50.46. The Office of Nuclear Regulatory Research is now completing an examination of the adequacy of these criteria for high-burnup fuel. The adequacy of the acceptance criteria for cladding performance is important to maintain adequate safety margins. The Rule should not be finalized until the fuel cladding acceptance criteria for LOCAs involving breaks at or below the TBS are reviewed and/or revised to assure their adequacy for the higher burnup fuel and more demanding conditions of current reactor operating conditions. Alternatively, the acceptance criteria in the Rule could be expressed in terms of general requirements, such as a high degree of confidence in maintaining a coolable geometry and retaining some ductility in the cladding. Specific cladding and core criteria could be placed in the associated regulatory guide.

An important element in the selection of the TBS is the state-of-the-art review of break size frequencies conducted by the Office of Nuclear Regulatory Research, documented in draft NUREG-1829. There is substantial uncertainty in the determination of these frequencies. If there is a high degree of assurance that breaks greater than the TBS can be mitigated, the impact of this uncertainty on the selection of the TBS is substantially reduced. The selection of the TBS could then include consideration of the benefits of small changes in the break size. For example, the current TBS for BWRs inhibits implementation of longer diesel start-up times, which are almost universally agreed to lead to improved emergency diesel generator operability. If the staff strengthens the defense in depth for breaks greater than the TBS, the TBS proposed by the BWR Owners' Group could be acceptable and would not be inconsistent with the results in draft NUREG-1829.

Although the Rule defines TBSs for BWRs and PWRs, licensees should not presume that these automatically apply to all plants. As part of the adoption of the Rule, licensees should have to demonstrate that the results in draft NUREG-1829 are applicable to their plants. The staff should provide guidance for this demonstration in the associated regulatory guide. As part of this demonstration, licensees should demonstrate that the reactor coolant system piping of diameter corresponding to the TBS or larger meets the deterministic requirements currently used to credit leak-before-break for dynamic analysis of reactor coolant piping. Such demonstrations will provide additional assurance of the very low likelihood of failures greater than the TBS. Many plants should have already performed such analyses.

The staff is revising draft NUREG-1829 to incorporate, as appropriate, the changes resulting from the resolution of public comments. This revision should be completed prior to issuing the revised Rule.

For internal events, the occurrence of a LBLOCA and a LOOP can generally be considered as independent events, and thus the simultaneous occurrence of a break greater than the TBS and a LOOP is a very unlikely event. However, a LOOP is very likely for any seismic event that is large enough to induce failures in reactor piping systems. As part of its effort to establish the TBS, the staff performed a study of the likelihood of seismically induced failures in unflawed piping, flawed piping, and indirect failures of other components and component supports that could lead to piping failure. The study focused on piping systems in PWRs east of the Rocky Mountains. We have not yet completed our review of the staff's study in this area. However, the results of the study indicate that for these plants the likelihood of seismically induced failures in unflawed piping of size greater than the TBS is very low for earthquakes with 10^{-5} and 10^{-6} annual probabilities of exceedance. Even for pipes with long surface flaws, the depths of these flaws must be greater than 30-40% of the wall thickness for a high likelihood of failure during such earthquakes. Inspection programs, leak detection systems, and other measures taken to eliminate failure mechanisms such as stress corrosion cracking should make the likelihood of such cracks very low. Because seismic hazards are very plant specific, licensees adopting the Rule will have to demonstrate that the results developed by the staff bound the likelihood of seismically induced failure in their plants. For unflawed piping, the results of the individual plant examination of external events (IPEEE) program may provide the needed information. Licensees may have to perform additional calculations to demonstrate a comparable robustness of flawed piping.

Although substantial progress has been made in the development of a risk-informed 10 CFR 50.46, the Rule should not be issued in its current form. It would be significantly strengthened by addressing the issues raised in this report.

Additional comments by ACRS Member Graham B. Wallis and ACRS Member Sanjoy Banerjee are presented below.

Sincerely,

/RA/

Graham B. Wallis
Chairman

Additional comments from ACRS Member Graham B. Wallis

My colleagues have suggested some significant improvements to the draft final rule, which I support, if it should be issued as final.

However, I am not persuaded that an adequate case has been made for this rule or that its consequences have been sufficiently explored.

The probabilities for breaks of various sizes, as assessed in draft NUREG-1829, can be accommodated within the framework of the existing rule's "realistic (best estimate)" alternative without any new rulemaking. This can be done in numerous ways while preserving suitable caution and defense in depth. The details can be worked out between the staff and licensees through an evolutionary process that includes thorough consideration of practicality, enforcement, technical uncertainties, benefits, and risks.

Additional comments from ACRS Member Sanjoy Banerjee

I support the Recommendations in the ACRS letter regarding the draft final rule to risk inform 10 CFR 50.46, but would add the further Recommendation that the draft NUREG-1829 be externally peer reviewed before being issued.

I have arrived at this Recommendation after reviewing NUREG-1829 and transcripts of 5 meetings regarding the work contained in it, held by the ACRS Regulatory Policies and Practices Subcommittee from 11/21/03 to 11/16/04. Based on this, it is my opinion that the quality of the NUREG and the credibility of its conclusions, would be substantially enhanced by eliciting, and responding to, comments from external and independent peer reviewers. This point was also raised at several of the ACRS Subcommittee meetings, but no substantive external peer review appears to have been conducted.

Amongst the several issues which, in my opinion, may be elucidated by such a review are the wide divergence in the initial estimates for various LOCA frequencies, and the methods used to narrow the range of uncertainty in the final results from which the conclusions are drawn.

References:

1. Memorandum from Michael Marshall Jr., Acting Branch Chief, Financial, Policy, and Rulemaking Branch, Division of Policy and Rulemaking, Office of Nuclear Reactor Regulation, to Dr. Graham B. Wallis, Chairman, Advisory Committee on Reactor Safeguards, "Advisory Committee on Reactor Safeguards Review of the Draft Final Rule to Amend 10 CFR 50.46, 'Risk-informed changes to loss-of-coolant accident technical requirements'," dated October 26, 2006.

References (continued)

2. Report from Graham B. Wallis, Chairman, Advisory Committee on Reactor Safeguards, to Nils. J. Diaz, Chairman, U.S. Nuclear Regulatory Commission, "Proposed Rulemaking to Modify 10 CFR 50.46, 'Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements'," dated March 14, 2005.
3. Report from Mario V. Bonaca, Chairman, Advisory Committee on Reactor Safeguards, to Nils. J. Diaz, Chairman, U.S. Nuclear Regulatory Commission, "SECY-04-0037, 'Issues Related to Proposed Rulemaking to Risk-Inform Requirements Related to Large Break Loss-of-Coolant Accident (LOCA) Break Size and Plans for Rulemaking on LOCA with Coincident Loss-of-Offsite Power'," dated April 27, 2004.
4. Staff Requirement Memorandum from Annette L. Vietti-Cook, Secretary, U.S. Nuclear Regulatory Commission, to Luis A. Reyes, Executive Director for Operations, U.S. Nuclear Regulatory Commission, "Staff Requirements - SECY-04-0037 - Issues Related to Proposed Rulemaking to Risk-Inform Requirements Related to Large Break Loss-of-Coolant Accident (LOCA) Break Size and Plans for Rulemaking on LOCA with Coincident Loss-of-Offsite Power," dated July 1, 2004.
5. U.S. Nuclear Regulatory Commission, NUREG-1829, "Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process," Draft Report for Comment, June 2005.
6. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," November 2002.
7. U.S. Nuclear Regulatory Commission, "Seismic Considerations for the Transition Break Size," December 2005, ADAMS ML053470439.
8. Letter from Randy C. Bunt, Chair, BWR Owners' Group, to Graham B. Wallis, Chairman, Advisory Committee on Reactor Safeguards, "Draft Final Rule Language, Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements, ADAMS Accession NO. ML062760146, dated October 3, 2006," dated October 13, 2006.

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ACRSR-2227

December 12, 2006

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

SUBJECT: DRAFT FINAL REGULATORY GUIDE DG-1145, COMBINED LICENSE APPLICATIONS FOR NUCLEAR POWER PLANTS (LWR EDITION)

Dear Chairman Klein:

During the 538th meeting of the Advisory Committee on Reactor Safeguards, December 7-8, 2006, we met with representatives of the NRC staff to discuss draft final Regulatory Guide DG-1145, Combined License Applications for Nuclear Power Plants (LWR Edition). Our Subcommittee on Future Plant Designs also reviewed this Guide and related matters on November 30, 2006. We also had the benefit of the documents referenced.

Recommendations

1. The final rule, 10 CFR Part 52, should retain the requirements that a design-specific probabilistic risk assessment (PRA) be submitted with the design certification application and that a plant-specific PRA be submitted with the combined license (COL) application.

2. DG-1145 should be issued as a final Regulatory Guide after the staff ensures that it is consistent with the final rule 10 CFR Part 52 and with the Regulatory Guides and Standard Review Plan (SRP) Sections/Chapters being revised or developed in support of new reactor licensing.

Background and Discussion

DG-1145, Combined License Applications for Nuclear Power Plants (LWR Edition), provides detailed guidance on the content of a COL application. The development of DG-1145 was done in parallel with the development of a proposed revision to 10 CFR Part 52 and the development of revisions to Regulatory Guides and (SRP) Sections/Chapters in support of new reactor licensing.

The proposed 10 CFR Part 52 (SECY-05-0203), that we reviewed, required that PRAs be submitted as part of the design certification and COL applications. In the draft final rule (SECY-06-0220), this requirement has been eliminated. The staff stated that DG-1145 has to be revised to reflect this change. We disagree with this change in Part 52. To certify a design or approve a COL, it will be necessary to have a detailed review of the PRA. The information needed for this review includes event trees, fault trees, support system dependencies, initiating events, data (reliabilities/probabilities of failure), human reliability, common-cause failure analysis, fire risk, flooding risk, seismic risk, minimal cutsets, and uncertainty and importance measures. Unless the PRA is submitted, such a review will have to be done at the applicant's office. This will be extremely difficult for the staff and not feasible for the ACRS. The requirements to submit the PRA with a design certification application and with a COL application should be retained in Part 52. After issuance of the COL, updates to the PRA need not be submitted.

Before publishing DG-1145 as a final Regulatory Guide, the staff should ensure that it is consistent with 10 CFR Part 52 and with other Regulatory Guides and SRP Sections/Chapters associated with future plant designs. In addition, the staff should ensure that the scope and level of detail of the information within the various sections of DG-1145 are complete and consistent.

Section C.II.1 of DG-1145 identifies nine objectives that the COL applicant must address in its risk evaluation. Neither the ASME PRA Standard (ASME RA-S-2002) nor Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," provides guidance on how to meet these objectives. SRP Chapter 19.0, which is being revised, should include review guidelines for determining whether an applicant's risk evaluation meets these objectives.

We would like to be informed of any significant changes made to this Guide prior to publishing it in final form.

Sincerely,

/RA/

Graham B. Wallis
Chairman

References
1. Memorandum dated September 1, 2006, from David B. Matthews, Director, Division of New Reactor Licensing, NRR, to John T. Larkins, Executive Director, ACRS, Subject: Transmittal of Draft Regulatory Guide DG-1145 "Combined License Applications for Nuclear Power Plants (LWR Edition)"
2. SECY-05-0203, Revised Proposed Rule to Update 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," dated November 3, 2005
3. SECY-06-0220, Final Rule to Update 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," dated December 3, 2006
4. Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications (ASME RA-S-2002), Issued April 5, 2002
5. U. S. Nuclear Regulatory Commission, Regulatory Guide 1.200 For Trial Use, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," February 2004
6. Standard Review Plan NUREG-0800, Chapter 19.0, Probabilistic Risk Assessment

UNITED STATES
NUCLEAR REGULATORY COMMISSION
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ACRSR-2219

October 25, 2006

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

SUBJECT: DRAFT FINAL NUREG-1824, "VERIFICATION AND VALIDATION OF SELECTED FIRE MODELS FOR NUCLEAR POWER PLANT APPLICATIONS"

Dear Mr. Reyes:

During the 536th meeting of the Advisory Committee on Reactor Safeguards, October 4-6, 2006, we met with representatives of the NRC staff, Electric Power Research Institute (EPRI), and the National Institute of Standards and Technology (NIST) to discuss the draft final NUREG-1824 (EPRI 1011999), "Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications." Our Subcommittee on Reliability and Probabilistic Risk Assessment (PRA) also reviewed this matter during its meeting on September 21, 2006. During our review, we had the benefit of the documents referenced.

CONCLUSION AND RECOMMENDATIONS

1. The report provides a systematic evaluation of the predictive capability of five commonly used compartment fire models. It should be published.

2. The user's guide to be developed by the staff should include:

   a. Estimates of the ranges of normalized parameters to be expected in nuclear plant applications.
   b. Quantitative estimates of the uncertainties associated with each model's predictions, preferably in the form of probability distributions.

BACKGROUND

Fire models are used in a number of safety evaluations, including fire risk analysis; demonstrating compliance with, and exemptions to, the regulatory requirements for fire protection in 10 CFR Part 50, Appendix R; the significance determination process of the Reactor Oversight Process; and establishing the risk-informed, performance-based voluntary fire protection licensing basis under 10 CFR 50.48(c) and the referenced 2001 Edition of the National Fire Protection Association (NFPA) Standard, NFPA 805, "Performance-Based Standard for Fire Protection for Light-Water Reactor Electric Generating Stations." NFPA 805 requires that "only fire models that are acceptable to the authority having jurisdiction shall be used in fire modeling calculations." NFPA 805 further requires that the fire models be verified and validated, and be applied only within their domains of validity.

The NRC Office of Nuclear Regulatory Research (RES) and EPRI sponsored a collaborative project for the verification and validation of selected fire models that are commonly used in the nuclear industry. NIST participated in this work. Report NUREG-1824 (EPRI 1011999) is the result of this collaborative project.

The selected models are:

- Fire Dynamics Tools (FDTs) developed by the NRC
- Fire-Induced Vulnerability Evaluation, Revision 1 (FIVE-Rev1) developed by EPRI
- Consolidated Model of Fire Growth and Smoke Transport (CFAST) developed by NIST
- MAGIC developed by Electricité de France (EdF)
- Fire Dynamics Simulator (FDS) developed by NIST

The verification and validation study was based on the methodology described in the American Society for Testing and Materials (ASTM) International Standard E 1355 - 05a "Standard Guide for Evaluating the Predictive Capability of Deterministic Fire Models."

A draft version of NUREG-1824 was issued for public comment on January 31, 2006. The comment period closed on March 31, 2006. The project team responded to all of the public comments.

DISCUSSION OF THE NUREG REPORT

Ever since the Browns Ferry fire in 1975 and the publication of several PRAs that demonstrated the risk significance of fires, there has been a great deal of interest in modeling the effects of fire on nuclear power plants. A number of deterministic models have been proposed focusing primarily on compartment fires. These are based on varying assumptions and calculational methods ranging from simple hand calculations (FIVE-Rev1 and FDTs) to two-zone models (CFAST and MAGIC) to sophisticated detailed models (FDS). This study is the first systematic evaluation of the ability of fire models to predict experimental results and will be very useful to both the NRC and the industry.

The project team identified 13 parameters that are likely to be required in safety assessments involving fires. These parameters were selected by reviewing potentially risk-significant scenarios from a variety of sources and are limited to those that describe the environment created by a fire in a compartment, e.g., the height and temperature of the hot gas layer, the flame height, the smoke concentration, and the radiant heat flux. This set of parameters does not characterize other important fire phenomena that are out of the scope of the present work, such as fire propagation in cable trays.

The ability of the selected models to estimate numerical values for the chosen parameters was evaluated by comparing their results with experimental measurements. The measured heat release rates from the fires were used as input to the analyses. Twenty-six experiments were selected from five test series that were judged to be relevant to nuclear plant applications and for which sufficient information was available to allow quantitative evaluations. The experiments were performed using pool fires with a variety of hydrocarbon fuels and a wide range of heat release rates.

The model predictions for each experiment were compared with the experimental results. There are uncertainties associated with these comparisons because of uncertainty in model input (primarily the heat release rate) and uncertainty in the measurements themselves. The experimental *measurement uncertainty* and the experimental *model input uncertainty* are used to develop a range of possible values of the scenario parameter of interest. The accuracy of the model predictions is qualitatively characterized by a simple color code.

DISCUSSION OF THE USER'S GUIDE

The staff plans to develop a user's guide to complement NUREG-1824. A user will have to determine whether the results of the verification and validation study are applicable to the situation to be analyzed. This is done using "normalized parameters" (i.e., governing non-dimensional groups, not to be confused with the13 scenario parameters discussed above) that allow users to compare results from scenarios of different scales by normalizing physical characteristics of the scenario. These normalized parameters are traditionally used in fire modeling applications and are included in the NUREG report. The user's guide should provide estimates of the ranges of normalized parameters to be expected in nuclear plant applications. These estimates would allow a determination of whether risk-significant fires fall within or outside the parameter ranges covered by the verification and validation process.

The user's guide should also provide probability distributions for the model predictions due to the intrinsic model uncertainty, i.e., the uncertainty associated with the model's physical and mathematical assumptions. These distributions should not include the uncertainties in the heat release rate since the latter will be an input specified by the user. The color designations provide no quantitative estimate of the intrinsic uncertainty. This uncertainty is an important input in risk-informed applications. Even in non-risk-informed applications, a quantitative assessment of the tendency of a model to over- or under-predict would be valuable. The staff told us that such quantitative estimates will be provided in the user's guide. We look forward to reviewing this document.

CONCLUDING REMARKS

We commend the RES staff and EPRI for undertaking this project and providing the basis for the evaluation of fire models. The NUREG report and the user's guide will significantly improve the technical basis supporting the fire safety evaluations.

This commendable effort to validate models of compartment fires is an important first step in developing the fire models needed by the NRC to assess fire risks and licensee proposals. Validated models of the effects of fires on equipment and cables are needed. Also needed are models of smoke transport within plants and the effects of deposited smoke on equipment and structures. We look forward to interacting with the staff as this research progresses.

Sincerely,

/RA/

Graham B. Wallis
Chairman

References:

1. *Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications, Vol 1: Main Report,* U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research (RES), Rockville, MD, and Electric Power Research Institute (EPRI), Palo Alto, CA, NUREG-1824 and EPRI 1011999, August 2006.

2. *Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications, Vol 2: Experimental Uncertainty,* U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research (RES), Rockville, MD, and Electric Power Research Institute (EPRI), Palo Alto, CA, NUREG-1824 and EPRI 1011999, August 2006.

3. *Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications, Vol 3: Fire Dynamics Tools (FDT^s),* U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research (RES), Rockville, MD, and Electric Power Research Institute (EPRI), Palo Alto, CA, NUREG-1824 and EPRI 1011999, August 2006.

4. *Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications, Vol 4: Fire-Induced Vulnerability Evaluation (FIVE-Rev1),* U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research (RES), Rockville, MD, and Electric Power Research Institute (EPRI), Palo Alto, CA, NUREG-1824 and EPRI 1011999, August 2006.

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6. Verification and Validation of Selected Fire Models for Nuclear Power Plant Application, EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities: Vol. 6: MAGIC, U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research (RES), Rockville, MD, and Electric Power Research Institute (EPRI), Palo Alto, CA, NUREG-1824 and EPRI 1011999, August 2006

7. Verification and Validation of Selected Fire Models for Nuclear Power Plant Application, Vol. 7: Fire Dynamics Simulator (FDS), U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research (RES), Rockville, MD, and Electric Power Research Institute (EPRI), Palo Alto, CA, NUREG-1824 and EPRI 1011999, August 2006

8. NFPA 805, "Performance-Based Standard for Fire Protection for Light-Water Reactor Electric Generating Stations," 2001 Edition, National Fire Protection Association, Quincy, MA.

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

ACRSR-2242

March 22, 2007

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

SUBJECT: DEVELOPMENT OF THE TRACE THERMAL-HYDRAULIC SYSTEM ANALYSIS CODE

Dear Chairman Klein:

During the 540th meeting of the Advisory Committee on Reactor Safeguards (ACRS), March 8-9, 2007, we completed our report on the development of the TRACE thermal-hydraulic (T/H) system analysis code. We also discussed this matter during our 539th meeting, February 1-3, 2007. Our Thermal-Hydraulic Phenomena Subcommittee discussed this matter on December 5, 2006. During these reviews, we had the benefit of discussions with representatives of the NRC staff and its contractors. We also had the benefit of the document referenced.

RECOMMENDATIONS

1. The schedule for documenting, validating, and peer reviewing TRACE should be accelerated and the work completed expeditiously.

2. The development of a representative set of TRACE plant models and user testing on applications should also be accelerated to facilitate timely incorporation of TRACE into the regulatory process.

BACKGROUND AND DISCUSSION

In the mid-1990s, the Office of Nuclear Regulatory Research, working with the Office of Nuclear Reactor Regulation, determined that the four primary reactor system T/H codes that were in use at that time should be consolidated into one code. These codes included RELAP5 (LOCA), TRAC-P(PWR-LOCA), TRAC-B(BWR LOCA), and RAMONA (BWR Stability).

The models, correlations, and solution methodologies in these codes did not reflect the state-of-the-art and required in-depth review and modification. It was also recognized that they had been designed at a time when computer capabilities were limited and included many structural aspects, such as memory management, that were no longer needed and increased the cost of code maintenance and development. The availability of graphical user interfaces and their wide acceptance also suggested the desirability of incorporating similar capability in NRC codes. All these considerations led to extensive code consolidation, model improvements, and implementation efforts culminating in the development of TRACE .

TRACE is intended to serve as the main tool for confirmatory analyses of a broad range of thermal-hydraulic problems for current and future reactor designs. It has the potential to offer significantly enhanced capabilities for state-of-the-art analyses of thermal-hydraulic issues. Applications include certification of new reactor designs and the regulatory review of power uprates for currently operating reactors. Therefore, the schedule for documenting, validating, and peer reviewing TRACE, as well as the development of plant input decks, should be accelerated. The work should be completed expeditiously to enable the incorporation of the code into the regulatory process.

Sincerely,

/RA/

William J. Shack
Chairman

Reference:

1. Memorandum from Farouk Eltawila, Director, Division of Risk Assessment and Special Projects, Office of Nuclear Regulatory Research, to Frank Gillespie, Executive Director, ACRS, "TRACE V5.0 Documentation and Support", January 31, 2007

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

ACRSR-2226

November 17, 2006

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

SUBJECT: DRAFT FINAL REVISION 3 TO REGULATORY GUIDE 1.7, "CONTROL OF COMBUSTIBLE GAS CONCENTRATIONS IN CONTAINMENT FOLLOWING A LOSS-OF-COOLANT ACCIDENT," AND STANDARD REVIEW PLAN SECTION 6.2.5, "COMBUSTIBLE GAS CONTROL IN CONTAINMENT"

Dear Mr. Reyes:

During the 537th meeting of the Advisory Committee on Reactor Safeguards, November 1-3, 2006, we completed our review of the draft final Revision 3 to Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," and a proposed revision to Standard Review Plan (SRP) Section 6.2.5, "Combustible Gas Control in Containment." During our 536th meeting, October 4-6, 2006, we met with representatives of the NRC staff to discuss these documents. We had the benefit of the documents referenced.

RECOMMENDATIONS

1. Regulatory Guide 1.7, Revision 3, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," along with the corresponding SRP Section 6.2.5 should be issued after including References 19-22 from the SRP in the Regulatory Guide.

2. The staff should develop additional guidance on acceptable methods for demonstrating the effective achievement of a mixed atmosphere in the containment. Such guidance should caution that current analytical codes may overestimate mixing and that applicants will need to substantiate the applicability of these codes to their analyses.

DISCUSSION

10 CFR 50.44, "Combustible Gas Control for Nuclear Power Reactors," was revised in 2003. The revised rule recognizes that sufficient combustible gas to pose a risk-significant threat to containment integrity is generated only during a beyond-design-basis accident. The requirements in the prior version of the rule for systems to mitigate hydrogen release during a

design-basis loss-of-coolant accident were eliminated. For currently licensed plants, all boiling water reactor (BWR) Mark I and Mark II containments must have an inerted atmosphere. BWRs with Mark III containments and pressurized water reactors (PWRs) with ice condenser containments must have the capability for controlling combustible gas generated from a metal-water reaction involving 75 percent of the fuel cladding surrounding the active fuel region (excluding the cladding surrounding the plenum volume) so that there is no loss of containment structural integrity. Future water-cooled reactor applicants and licensees are required to have either an inerted containment or must limit hydrogen concentrations in containment during and following the release of an amount of hydrogen equivalent to that generated from a 100 percent fuel clad-coolant reaction, uniformly distributed, to less than 10 percent (by volume) and maintain containment structural integrity and appropriate accident mitigating features.

The revised rule also retains the requirement to monitor hydrogen concentrations in the containment atmosphere for all containment designs and includes a requirement for oxygen monitors in containments with inerted atmospheres. However, monitors are no longer classified as safety-related components.

The revised Regulatory Guide provides guidance for the design of combustible gas control systems. It also provides guidance for design, qualification criteria, and functional requirements for hydrogen and oxygen monitors. Although the combustible gas control systems are no longer considered safety related, the Regulatory Guide notes that systems installed and approved by the NRC prior to October 16, 2003, the effective date of the revised10 CFR 50.44, are sufficient to meet these criteria. The guidance provided is appropriate and consistent with the requirements for severe-accident mitigation equipment in evolutionary and passive plant designs.

The revised rule requires that containment structural integrity be demonstrated. The Regulatory Guide identifies criteria of the ASME Boiler and Pressure Vessel Code that provide an acceptable method for demonstrating that the requirements are met. These requirements are appropriate and consistent with current ASME code analyses used by licensees for this purpose.

The revised rule requires that all containments have a capability for ensuring a mixed atmosphere to avoid the potential for detonation of combustible gases. The Regulatory Guide provides general guidance on how this may be achieved. It allows this capability to be provided by an active, passive, or combination system. Active systems may consist of a fan, fan cooler, or containment spray. For passive or combination systems that use convective mixing to mix the combustible gases, it recognizes that the containment internal structures can have significant effects on the mixing in the containment and that the containment should have design features that promote the free circulation of the atmosphere. References 19-22 in the SRP Section 6.2.5 provide important insights into the potential for detonation of hydrogen-air mixtures and should be included as references in the Regulatory Guide prior to issuance.

Additional guidance on acceptable methods for demonstrating the effective achievement of a mixed atmosphere would be helpful and should be developed. Such guidance should caution that current analytical codes widely used to evaluate mixing and transport within containments may overestimate mixing and that applicants will need to substantiate the applicability of these codes to their analyses.

The revised SRP Section 6.2.5 has been updated to be consistent with the revised 10 CFR 50.44. It provides appropriate acceptance criteria and review procedures. Revision 3 to Regulatory Guide 1.7 and the revised SRP Section 6.2.5 should be issued.

Sincerely,

/RA/

Graham B. Wallis
Chairman

References:

1. Memorandum from Jimi T. Yerokun, Chief, Risk Applications and Special Projects Branch, Division of Risk Assessment and Special Projects, Office of Nuclear Regulatory Research, to Michael R. Snodderly, Chief, Technical Support Branch, Advisory Committee on Reactor Safeguards, "Additional Information - Regulatory Guides 1.7, 'Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident,' and 1.196, 'Control Room Habitability at Light-Water Nuclear Power Plants'," September 6, 2006.

2. Memorandum from Thomas O. Martin, Director, Division of Safety Systems, Office of Nuclear Reactor Regulation, to John T. Larkins, Executive Director, Advisory Committee on Reactor Safeguards, "Transmittal of Proposed Revision to Standard Review Plan NUREG-0800 Section 6.2.5, 'Combustible Gas Control in Containment'," October 2, 2006.

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

May 16, 2007

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

SUBJECT: DEVELOPMENT OF AN INTEGRATED LONG-TERM REGULATORY RESEARCH PLAN

Dear Chairman Klein:

During the 542nd meeting of the Advisory Committee on Reactor Safeguards (ACRS), May 3-5, 2007, we discussed the status of staff's efforts associated with the development of an integrated, long-term regulatory research plan. Our Subcommittee on the Safety Research Program also discussed this matter on May 2, 2007. During these meetings, we had the benefit of discussions with representatives of the NRC staff and of the documents referenced.

In our recent biennial reports on review and evaluation of the NRC safety research program, we have noted the need for long-term research not tied to the near-term issues of the regulatory process. Our focus was on long-term research to modernize the way NRC conducts its regulatory and safety mission. We called attention to critical areas where the NRC will need to maintain long-term technical competencies including neutronics, criticality analysis, reactor fuels, and probabilistic risk assessment (PRA). We also called attention to needs for access to experimental facilities and adequate computational tools for the regulatory process.

As directed by the Commission, the staff has undertaken an examination of the agency's long-term research needs. The focus of the work proposed by the staff differs from that emphasized for long-term research in our biennial reports. What the staff proposes does include some work directed toward the modernization and core expertise stressed in our reports. The staff has searched for emerging technologies and programs that NRC may have to address in regulatory processes sometime in the future. The staff has been careful to distinguish between research addressing current needs that will take a long time to complete and research to meet needs anticipated in the future

The staff has identified four broad areas of long-term research:

- Research to support licensing of nuclear facilities developed for the U.S. Department of Energy's Global Nuclear Energy Partnership (GNEP)
- Research to support reactor license renewal beyond 60 years
- Test facilities
- Long-term research activities for cross-cutting and emergent technologies

The process adopted by the staff has led to the identification of aspects of long-term research that the agency could pursue. The development of this long-term plan is a considerable departure from the staff's focus in recent years on immediate regulatory needs. We comment below on several of the specific long-term research activities identified by the staff. We understand that in step 2 of the staff's development process, the staff will solicit input from other stakeholders to further develop its long-term research plan. We will comment in a separate forum on the broader scope of long-term research the agency needs to consider.

Research for Licensing GNEP Facilities

The U.S. Department of Energy (DOE), as part of its Global Nuclear Energy Partnership, is exploring the possibility of using a sodium-cooled fast reactor to transmute actinides as a stage in the reprocessing of spent water reactor fuel. Associated with the advanced "burner" reactor (ABR) will be facilities for processing both water reactor fuel and fast reactor fuel. A decision to proceed with the development of these capabilities is not anticipated until June 2008 and there may be delays beyond that date.

NRC has substantial experience dealing with sodium-cooled reactors, including its work on the Fast Flux Test Facility (FFTF), the Clinch River Breeder Reactor (CRBR), and preliminary work on the PRISM reactor. NRC is currently in the process of licensing a Mixed-Oxide (MOX) Fuel Fabrication Facility that will include most of the elements of an aqueous fuel reprocessing system. NRC has no significant experience licensing pyrometallurgical fuel reprocessing systems that might be included in the GNEP projects. We agree with the staff that there is merit in NRC maintaining some cognizance of work under way in GNEP. There would be merit in collection and organization of the documentation of past NRC work with sodium-cooled reactors.

Many involved in this past work on sodium-cooled reactors are nearing retirement age and their experience needs to be preserved at the agency. There is, however, no need to undertake a detailed research program until it is clear the Department of Energy will pursue the development for licensing of an advanced burner reactor and associated fuel reprocessing facilities. The staff will want to monitor the ongoing licensing of the MOX Fuel Fabrication Facility and search for ways to improve review and evaluation of the associated Integrated Safety Assessment.

Research to Support Reactor License Renewal Beyond 60 Years

Extension of reactor operating licenses from 40 years to 60 years is an important NRC regulatory activity today. The staff noted that there have been serious discussions about the possibility of license renewal beyond 60 years. New issues may arise in such further license extensions, especially since many of the plants will have been operating for 10 years to 30 years at power levels higher than originally licensed. We do not find that these new issues are likely to be so different than those encountered in the current license extension process that they merit a separate and distinct long-term research project. It would appear that any issues likely to arise could be identified and addressed in current research efforts, including the Proactive Materials Degradation Assessment project.

Test Facilities

In its long-term research plan, the staff has identified two activities associated with possible development of experimental facilities:

- Integrated Digital Instrumentation and Control and Human Machine Interfaces Test Facility
- Integral Effects Test Facilities for Advanced Non-Light Water Reactors

The staff will explore the feasibility of developing a facility for testing digital instrumentation and control systems and the human/machine interface. This will not be the first time such a feasibility study has been undertaken. In the past, it has been found that such a facility would not be cost effective. As a result, NRC has continued its association with the somewhat less satisfactory capability at the Halden project. If the staff again concludes that a new facility in the U.S. is not cost effective, it should consider collaboration with other nations to better meet the needs foreseen by the agency.

The staff also plans to identify experimental facilities throughout the world that can be used to investigate phenomena associated with advanced reactors that do not rely on water technology, including gas-cooled reactors and liquid metal-cooled reactors. A study with somewhat similar objectives has been conducted by Organization for Economic Cooperation and Development/Nuclear Energy Agency (OECD/NEA). Again, there may well be international interest in collaboration on this proposed research.

Cross-Cutting and Emergent Technologies

In the fourth category of possible, long term, research projects, staff has identified quite a number of individual topics. We provide synoptic comments on several of these below:

<u>Advanced Fabrication Techniques</u>

The Department of Energy and elements of the industry are aggressively searching for methods to facilitate and improve the construction of nuclear power plants. Some of the methods now being considered could greatly affect the NRC's processes for monitoring nuclear facility construction. The new plant fabrication methods could affect the long-term performance of structures and components. It is appropriate for the staff to develop an understanding of modern construction methods that may be applied to nuclear facilities and how they may affect the regulatory processes.

<u>Advanced Computational Methods</u>

The staff has called attention to advanced computational methods now available. The issue of advances in computer technology is much larger than just improved numerical algorithms. We foresee far greater use of virtual methods for design and evaluation in future applications. The staff is now ill-equipped to evaluate submissions using such advanced design technologies that are becoming ever more widespread within the engineering community. The evaluation and adaptation of computer technologies that the staff will need to have over the next 20 years could well be a major thrust in NRC's long-term research program.

Extended In Situ and Real-Time Inspection and Monitoring Techniques

Non-destructive inspection and testing are important tools for managing aging degradation at nuclear power plants. Such inspection and testing methods are expensive and can result in significant personnel exposures to radiation. In many cases, non-destructive inspections can only be done when plants are shutdown. There is, then, the possibility that unexpected, rapid degradation would not be detected in time to prevent an accident. In situ, real-time monitoring techniques could lead to more effective and reliable management of aging-related degradation of reactor materials. The staff needs to maintain cognizance of developments of such in situ, real-time methods.

Multiphase Computational Fluid Dynamics Capability

Commercial computer codes are available for computational fluid dynamics but have limited predictive capability for multi-dimensional single phase flows in nuclear systems. Virtually no such capability exists for multi-dimensional, multi-phase flows. However, two phase computational fluid dynamics is an emerging field which may prove useful in many regulatory applications. We are heartened that the staff is looking beyond commercially available computer codes in its search for future computational resources to support regulatory activities.

Offsite Mitigation Strategies

The staff proposes to ascertain if mitigation strategies developed by other industries and agencies are applicable to plants having accidents. We support this effort, but also agree that the staff should not be developing these mitigation strategies.

Nanotechnology for Nuclear Power Applications

Nanotechnology has caught the imagination of the technical community and there may well be future applications to nuclear power including structural materials, sensors, and advanced coolants. In addition, other advanced material technologies may also be useful, for example, surface modifications of reactor material including ion implantation. We encourage the staff to expand this area to include advanced materials in general that may have applications in the nuclear field.

Fire Effects on Fiber Optic Cables

We agree with the staff that it is very likely fiber optic systems may one day be used in nuclear plants and these fiber optic systems may be subject to the effects of fire. It seems, however, a very simple step to add fiber optic qualification as a task in the ongoing research on fire effects on cables. We see no need for a separate long-term research program in this area.

Advanced Quantitative Risk Assessment Methods and Advanced Modeling Techniques for Level 2/3 PRA

NRC's research has developed for the most part the PRA techniques in use today. PRA has become an essential element of the regulatory process. It is essential that NRC not allow development of PRA methods to stagnate. We certainly endorse continued examination of improved methods (including those for Level 1 PRA) to develop these methods and to improve the utility of these risk-assessment methods for the regulatory process.

Advanced Offsite Consequences Code

Consequence analysis computer codes used by NRC both for risk analysis and for accident response have limited capabilities to realistically portray dispersal of radioactive material in the actual environments surrounding nuclear power plants. Staff proposes research to develop codes better able to simulate radioactive material dispersal. The plan for such research should be deferred until completion of the ongoing State-of-the-Art Reactor Consequence Analysis activity to better define agency needs in this area.

Formal Decision Analysis Methods

In the past, ACRS has recommended that the staff should make greater use of formal decision making techniques. The staff should focus on adapting existing methods to support regulatory decisionmaking.

We support the staff's efforts to develop long-range research programs. We understand that the staff plans to prioritize the various research efforts it has identified and to update both the list and the prioritization episodically. We look forward to continued discussions with the staff on these long-term research projects.

Sincerely,

/RA/

William J. Shack
Chairman

References:

1. Memorandum to The Commissioners from Luis A. Reyes, Executive Director for Operations, Subject: SECY-07-0068, "Candidate Agency Long-Term Research Activities for Fiscal Year 2009," April 6, 2007. (Official Use Only Document - Sensitive Internal Information - Limited to NRC Unless the Commission Determines Otherwise)

2. Office of Nuclear Regulatory Research Report, Subject: "U.S. Nuclear Regulatory Commission Long-Term Research: FY 2009 Activities," March 2007. (Official Use Only - Sensitive Internal Information - Draft)

3. U.S. Nuclear Regulatory Commission, NUREG-1635,Vol. 7, "Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program, A Report to U.S. Nuclear Regulatory Commission," Advisory Committee on Reactor Safeguards, May 2006.

4. U.S. Nuclear Regulatory Commission, NUREG-1635, Vol. 6, "Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program, A Report to U.S. Nuclear Regulatory Commission," Advisory Committee on Reactor Safeguards, March 2004.

5. OECD/NEA, "Nuclear Safety Research in OECD Countries, Support Facilities for Existing and Advanced Reactors (SFEAR)," Committee on the Safety of Nuclear Installations (CSNI), Organization for Economic Cooperation and Development (OECD) Nuclear Energy Agency (NEA), NEA/CSNI/R(2007)6, 2007.

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

ACRSR-2241

March 22, 2007

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

SUBJECT: PROPOSED NRC STAFF AND INDUSTRY ACTIVITIES FOR ADDRESSING DISSIMILAR METAL WELD ISSUES RESULTING FROM THE WOLF CREEK PRESSURIZER WELD INSPECTION RESULTS

Dear Mr. Reyes:

During the 540th meeting of the Advisory Committee on Reactor Safeguards, March 8-9, 2007, we discussed the proposed NRC staff and industry activities for addressing the dissimilar metal weld issues resulting from the Wolf Creek pressurizer weld inspection results. Our Subcommittee on Materials, Metallurgy, and Reactor Fuels also reviewed this matter on March 6, 2007. During these meetings, we had the benefit of discussions with representatives of the NRC staff, Nuclear Energy Institute, and FirstEnergy and of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

1. We support the agreement reached between the staff and the industry on the resolution of dissimilar metal weld issues on pressurizer nozzles.

2. In the upcoming outages, the staff should encourage the industry to inspect all inspectable dissimilar metal welds on pressurizer nozzles before performing mitigation activities.

BACKGROUND AND DISCUSSION

In October 2006, ultrasonic examination revealed five indications in the welds on three pressurizer nozzles at the Wolf Creek Generating Plant. These indications were interpreted by the nondestructive examination experts as large circumferential cracks in nickel-based dissimilar metal welds joining the ferritic steel nozzles to the austenitic stainless steel coolant piping. No metallurgical samples were taken at Wolf Creek to confirm that the ultrasonic indications were actually stress corrosion cracks rather than weld flaws associated with the original fabrication.

The licensee repaired the nozzles with weld overlays of a nickel alloy with a higher chromium content (Alloy 152) that is much more resistant to primary water stress corrosion cracking (PWSCC). The repairs provide a structural capability equivalent to an original unflawed weld even assuming that the original weld has a through-wall, 360° circumferential crack.

The nickel-based alloys (Alloy 82 and 182) used for these welds are known to be susceptible to PWSCC in the primary coolant environment of pressurized water reactors (PWRs). Because the adjoining base metals are resistant to stress corrosion cracking in the primary coolant environment, axial cracks will be limited in length to a size no greater than the width of the weld.

While they may lead to leakage, such cracks are unlikely to lead to rupture or significant loss of coolant. Circumferential cracks can potentially grow to sizes that could lead to rupture.

Prior to the Wolf Creek finding, the staff and industry had recognized the potential for cracking in these dissimilar metal welds, and the industry had instituted a program to inspect these welds and apply weld overlays similar to those used at Wolf Creek. Most licensees of PWRs with susceptible nozzle welds will complete inspections or apply weld overlays during 2007. However, nine plants plan to perform these activities during outages in the spring of 2008. The plants that have not yet completed inspections or mitigation activities have committed to enhanced leakage detection as a compensatory action until these activities are completed.

Scoping analyses performed by the staff indicate that for the safety valve and relief valve nozzles, the times required for the cracks to grow through-wall and leak are relatively short (1.3 to 2.6 years) and are close to the times required for the cracks to grow large enough to result in gross rupture of the nozzles. Under some assumptions, the relief valve nozzle is predicted to rupture before any leakage occurs. There are substantial uncertainties associated with these analyses.

Only about 15 percent of the dissimilar metal pressurizer nozzle welds in PWRs have been inspected. Consequently, the current state of most welds is unknown. Therefore, the staff has taken the position that mitigation activities should be completed by the end of 2007 rather than the spring of 2008.

The industry has presented arguments that suggest the likelihood that nozzles can rupture without prior warning is sufficiently low, and the increased risk associated with a schedule for completing inspection and mitigation activities in the spring of 2008 is acceptably small. The industry is undertaking a program to develop an advanced finite element analysis capability that can provide a more rigorous basis for its arguments. Licensees have also committed to accelerate the schedule for inspection and mitigation and complete the work by the end of 2007, if this analysis effort is unsuccessful in demonstrating the likelihood of leak-before-break for these nozzles. They have further stated that this schedule could also be accelerated if new information were obtained during upcoming plant inspections that challenges current industry assumptions. In upcoming outages, the staff should encourage the industry to inspect all inspectable dissimilar metal pressurizer nozzle welds before performing mitigation activities.

The work being undertaken by the industry addresses the simplifying approximation usually imposed on fracture mechanics analyses that the crack shape is either elliptical or constant depth. The refined analysis considers crack growth at each point along the crack front and allows the crack to change shape as dictated by the stress distribution and appropriate crack growth correlations. Preliminary results provided by the industry suggest that such analyses may be able to show that crack growth will be such that the leak-before-break principle is preserved. The industry and the staff recognize the need for validation of the analytical models and comparisons of the predictions of the models with experimental data. This work could provide a very significant increase in the capability to realistically model the growth of flaws in reactor components and would be useful in a variety of applications. We support the agreement reached between the staff and the industry on the resolution of pressurizer nozzle weld issues.

Even with this increased capability to model the growth of cracks, there will still be large uncertainties in important variables that affect the results such as the welding residual stresses, the applied loads on the welds, and the population of cracks that could be present in nozzle welds that have not been inspected. It may eventually be possible to formalize the evaluations of these uncertainties through Monte Carlo simulation, but the present problem will have to be addressed through sensitivity studies. The staff and the industry have not yet settled on how to determine what will constitute an acceptable demonstration that the likelihood of violation of the leak-before-break principle is acceptably low, and this may not be possible until some of the results of the planned analyses are available.

Sincerely,

/RA/

William J. Shack
Chairman

References:

1. Memorandum from Mark A. Cunningham, Director, Division of Fuel, Engineering, and Radiological Research, RES to Frank P. Gillespie, Executive Director, ACRS, dated February 13, 2007, "Transmittal of (Proprietary) Draft Summary Report, 'Evaluation of Circumferential Indications in Pressurizer Nozzle Dissimilar Metal Welds at the Wolf Creek Power Plant'" (ADAMS ML070460127)

2. Letter from Christine King, Electric Power Research Institute, Materials Reliability Program (MRP), to Tanya Mensah, U.S. Nuclear Regulatory Commission, dated January 22, 2007, transmitting MRP 2007-003, "Implications of Wolf Creek Pressurizer Butt Weld Indications Relative to Safety Assessment and Inspection Requirements" (ADAMS ML070240140)

FRAMEWORK FOR FUTURE PLANT LICENSING

Thomas S. Kress

The ACRS should provide its views to the Commission with respect to staff's work on technology neutral licensing framework with a focus on ensuring the value of such an approach versus the development of a licensing framework for specific designs, such as high temperature gas cooled reactor or a liquid metal cooled reactor (SRM, November 8, 2006)

General Views on Framework

- **Develop top-down technology-neutral approach**

- **Test framework concepts on PBMR design**

Risk-Informed Performance-Based Part 50

- Concur with the staff's recommendation to defer rulemaking until after development of the licensing strategy for NGNP
- Completed framework should help guide the licensing strategy for NGNP
- Framework is incomplete and needs modification

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

ACRSR-2244

April 20, 2007

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

SUBJECT: TECHNOLOGY-NEUTRAL FRAMEWORK FOR FUTURE PLANT LICENSING

Dear Chairman Klein:

In a November 8, 2006 Staff Requirements Memorandum, the Commission requested that the Advisory Committee on Reactor Safeguards (ACRS) "provide its views to the Commission with respect to staff's work on technology-neutral licensing framework with a focus on ensuring the value of such an approach versus the development of a licensing framework for specific designs, such as a high temperature gas cooled reactor or a liquid metal cooled reactor." During the 540th meeting of the ACRS, March 8-9, 2007, we met with the NRC staff and discussed the development of a technology-neutral licensing framework versus the development of a licensing framework for specific designs. The staff's technology-neutral licensing framework is documented in draft NUREG-1860, "Framework for Development of a Risk-Informed, Performance-Based Alternative to 10 CFR Part 50." Our Subcommittee on Future Plant Designs had previously reviewed this document on March 7, 2007. We continued our discussions during the 541st ACRS meeting, April 5-7, 2007. We had the benefit of the documents referenced.

RECOMMENDATIONS

1. The staff should complete work on a technology-neutral framework rather than proceed with the development of technology-specific frameworks.

2. The completed framework should be tested on the Pebble Bed Modular Reactor (PBMR) design.

DISCUSSION

The current regulations evolved over many years and addressed issues as they arose (a largely bottom-up approach). The prospect of applications for licensing non-light-water reactor designs presents an opportunity to produce a regulatory system that utilizes modern technology such as probabilistic risk assessment, incorporates lessons learned from the past, and is based on general principles (i.e., following a top-down approach). This top-down approach should be developed on a technology-neutral basis from which technology-specific requirements will be derived. This will ensure consistency among requirements for different designs and among requirements for a specific design, as well as make the intent of the regulations more transparent. Without a common technology-neutral framework, it will be necessary to develop a similar regulatory basis for each separate technology, an alternative that would be significantly less efficient. In the near term, an additional benefit would be derived for licensing applications that use existing regulations with some modifications. These modifications could be guided by the technology-neutral framework.

The framework represents a major advancement in the development of a coherent risk-informed approach to establishing regulatory requirements for either future or current reactors. At this critical juncture, the staff should complete the framework. We look forward to continuing to work with the staff to resolve certain issues associated with the framework.

Pebble Bed Modular Reactor (Pty) Ltd. has submitted a number of white papers (References 3 through 6) that outline potential elements of an approach to certifying the PBMR design. Since the PBMR design also represents a significant departure from a light-water reactor design, it is a logical choice on which to test the completed framework.

Sincerely,

/RA/

William J. Shack
Chairman

References:

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

2. Memorandum dated April 3, 2007, from Farouk Eltawila, Director, Division of Risk Assessment and Special Projects, RES, to Frank P. Gillespie, Executive Director, ACRS, Subject: Transmittal of Proposed "Technology Neutral Framework" for Advisory Committee on Reactor Safeguards Review.

3. Letter dated June 13, 2006, from Edward G. Wallace, Senior General Manager - US Programs, PBMR (Pty) Ltd., to NRC Document Control Desk, Subject: PBMR White Paper: PRA Approach.

4. Letter dated July 3, 2006, from Edward G. Wallace, Senior General Manager - US Programs, PBMR (Pty) Ltd., to NRC Document Control Desk, Subject: PBMR White Paper: LBE Selection.

5. Letter dated August 28, 2006, from Edward G. Wallace, Senior General Manager - US Programs, PBMR (Pty) Ltd., to NRC Document Control Desk, Subject: PBMR White Paper: SSC Classification.

6. Letter dated December 13, 2006, from Edward G. Wallace, Senior General Manager - US Programs, PBMR (Pty) Ltd., to NRC Document Control Desk, Subject: PBMR White Paper: Defense-in-Depth Approach.

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

May 16, 2007

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

SUBJECT: DRAFT COMMISSION PAPER ON STAFF PLAN REGARDING A RISK-INFORMED AND PERFORMANCE-BASED REVISION TO 10 CFR PART 50

During the 542nd meeting of the Advisory Committee on Reactor Safeguards (ACRS), May 5-7, 2007, we met with representatives of the NRC staff to discuss the draft Commission Paper on the staff's plan regarding a risk-informed and performance-based revision to 10 CFR Part 50 (i.e., a new 10 CFR Part 53). We had the benefit of the documents referenced.

RECOMMENDATIONS

1. We concur with the staff's recommendation that the Commission defer development of a new 10 CFR Part 53 until the licensing strategy for the Next Generation Nuclear Plant (NGNP) is completed.

2. Work on the technology-neutral regulatory framework should continue so the framework can help guide the development of the licensing strategy for the NGNP.

3. There are important issues, critical to the development of the framework, that are still being debated within the ACRS. While we strongly support the continued development of the framework, NUREG-1860 should not be finalized until we reach a position on these issues and discuss our positions with the staff.

DISCUSSION

The staff notes that "all near-term combined license applications will be limited to LWRs, which can be licensed using the existing regulations ...." The Committee concurs that there is no compelling reason to proceed with development of a new 10 CFR Part 53 at this time.

The technology-neutral regulatory framework is in an advanced state of development, but it is still incomplete and needs modification. This effort should be continued so the framework can help guide the development of the licensing strategy for the NGNP. In our letter dated April 20, 2007, we recommended that the completed framework be tested on the pebble bed modular reactor design since it differs significantly from a light-water reactor.

There are important issues, critical to the development of the framework, that are still being debated within the ACRS. The Committee needs to develop a position on these issues before it can make recommendations to the staff on the framework, as presented in NUREG-1860.

While we strongly support the continued development of the framework, NUREG-1860 should not be finalized until we reach a position on these issues and discuss our positions with the staff.

We look forward to further interaction with the staff on the technology-neutral regulatory framework and anticipate issuing another letter that will document our specific recommendations.

Sincerely,

/RA/

William J. Shack
Chairman

Reference:

1. Memorandum dated April 17, 2007, from Michael J. Case, Director, Division of Policy and Rulemaking, NRR to Frank P. Gillespie, Executive Director, ACRS, Subject: Transmittal of Draft Commission Paper on Staff Plan Regarding a Risk-Informed and Performance-Based Revision to 10 CFR Part 50.

2. Report dated April 20, 2007, from William J. Shack, Chairman, ACRS to Dale E. Klein, Chairman, NRC, Subject: Technology-Neutral Framework for Future Plant Licensing.

DIGITAL I&C ACTIVITIES

George E. Apostolakis

The Committee should provide its view to the Commission on staff's efforts related to digital instrumentation and controls. The Committee should consider potential means for providing reasonable backup, if appropriate

(SRM, November 8, 2006)

- **Concur with the staff's approach to developing a project plan that defines a process to improve deployment of digital I&C technology for new and operating reactors**

- **The staff should develop an inventory and classification, (e.g., by function and other characteristics, of the various types) of digital and software systems that are being used and are likely to be used in the near future in nuclear power plants**

- **The staff should evaluate the operating experience with digital systems in the nuclear and other industries to obtain insights regarding potential failure modes**

- **The information obtained through evaluation of operating experience and the development of an inventory and classification of digital and software systems should be used in developing regulatory guidance on defense in depth and diversity for digital I&C systems**
- **This information is necessary to develop our response regarding the need and potential means for backup**

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

May 18, 2007

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

SUBJECT: ACTIVITIES RELATED TO DIGITAL INSTRUMENTATION AND CONTROL SYSTEMS

Dear Chairman Klein:

In a November 8, 2006 Staff Requirements Memorandum (SRM), the Commission requested the Committee provide its views on the staff's efforts related to digital instrumentation and controls (I&C) and consider potential means for providing reasonable backup, if appropriate. During the 542nd meeting of the Advisory Committee on Reactor Safeguards, May 3–5, 2007, we met with representatives of the NRC staff and the Nuclear Energy Institute to discuss the ongoing staff and industry activities associated with digital I&C systems. Our Subcommittee on Digital I&C Systems reviewed this matter on April 18, 2007. We also had the benefit of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

1. We concur with the staff's approach to developing a project plan that defines a process to improve deployment of digital I&C technology for new and operating reactors.

2. The staff should develop an inventory and classification (e.g., by function or other characteristics) of the various types of digital and software systems that are being used and are likely to be used in nuclear power plants.

3. The staff should evaluate the operating experience with digital systems in the nuclear and other industries to obtain insights regarding potential failure modes.

4. The information obtained through performing activities in Recommendations 2 and 3 should be used in the development of regulatory guidance on defense in depth and diversity for digital I&C systems.

BACKGROUND AND DISCUSSION

The staff is responding to a December 6, 2006 SRM, in which the Commission directed that senior NRC managers engage industry to establish an NRC project plan with specific milestones and deliverables to address deployment of digital I&C technology. The staff's draft project plan builds on the ongoing digital I&C research program. The staff has formed a Steering Committee consisting of senior managers to provide oversight and guidance on six key technical and regulatory issues and to interface with the industry. Each key issue is assigned to a Task Working Group that reports to the Steering Committee. The staff has identified specific deliverables and is in the process of specifying the due dates for these deliverables. We agree with the staff's approach to the development of a process that will facilitate the deployment of digital I&C technology for new and operating reactors.

One of the key technical and regulatory issues is the determination of the degree of redundancy and diversity necessary to protect a safety function against the occurrence of a digital system failure due to a common cause. The staff stated that the principal means for reducing the potential for common-cause failures (CCFs) is the high quality that is demanded of the digital system design process. Even with the assumption of high quality, CCFs cannot be excluded and, therefore, are postulated in the analysis of particular accidents; a judgment is then made regarding the need for diverse means of protection. An example of the latter is the provision of diverse displays and controls in the main control room to enable the manual actuation of components.

Protecting against CCFs is a subjective exercise that relies on the experience and imagination of the analysts. A critical element is the specification of the postulated CCFs. The set of postulated CCFs may be overly conservative in some cases and incomplete in others.

The quality of this process depends on the information available to the analysts regarding the functionality and susceptibility to failures of the digital system. For example, the evaluation of the performance of systems with simple actuation software is expected to be simpler than the evaluation of systems with software used for feedback and control. Therefore, there is a basic need for an inventory and classification (e.g., by function) of the various types of digital software systems that are being used and are likely to be used in the near future in nuclear power plants.

The search for potential CCFs would be enhanced by lessons learned from relevant operating experience for each type of digital system. For example, there have been several well-known accidents in the aerospace industry involving digital software. An evaluation of this experience and its applicability to nuclear systems should provide very valuable insights into potential failure modes. These failure modes would be associated with the various layers of components and functions that constitute a specific type of digital I&C system, including: (1) the host computer or microprocessor hardware; (2) software performing critical background functions, such as timers and clocks, self-test routines, and network communications; and (3) application-specific software (i.e., software that receives input from plant sensors and implements the logic and/or algorithmic functions necessary to control external devices that are part of the physical plant).

While the development of an inventory of the various types of digital software systems and the evaluation of the operating experience will provide useful insights into all six key technical and regulatory issues, we consider it essential in the formulation of regulatory guidance on defense in depth and diversity strategies. This information is also necessary to develop our response regarding the need and potential means for backup.

We look forward to working with the staff as the development and implementation of the project plan proceeds.

Sincerely,

William J. Shack
Chairman

References

1. Letter from A.L. Vietti-Cook, Secretary, NRC, to John T. Larkins, Executive Director, ACRS, "Staff Requirements - Meeting with ACRS, October 20, 2006, Commissioners' Conference Room, One White Flint North, Rockville, MD (Open to Public Attendance)," November 8, 2006

2. U.S. Nuclear Regulatory Commission, Digital I&C Project Plan (draft), April 13, 2007

3. Letter from A.L. Vietti-Cook, Secretary, NRC, to Luis A. Reyes, Executive Director for Operations, "Staff Requirements - Briefing on Digital I&C, November 8. 2006, Commissioners' Conference Room, One White Flint North, Rockville, MD (Open to Public Attendance)," December 6, 2006

4. National Research Council, "Digital Instrumentation and Control Systems in Nuclear Power Plants: Safety and Reliability Issues," National Academy Press, Washington, DC, 1997

5. Branch Technical Position 7-19, "Guidance for Evaluation of Diversity and Defense-in-Depth in Digital Computer-Based Instrumentation and Control Systems," Revision 5, modified February 15, 2007

6. U.S. Nuclear Regulatory Commission/Lawrence Livermore National Laboratory NUREG/CR-6303, "Method for Performing D3 Analyses of Reactor Protection System," December 1994

7. SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary, and Advanced Light-Water Reactor Designs," Section Q, "Defense Against Common-Mode Failures in Digital Instrumentation and Control Systems," April 2, 1993

8. U.S. Nuclear Regulatory Commission/ The Ohio State University NUREG/CR-6901, "Current State of Reliability Modeling Methodologies for Digital Systems and Their Acceptance Criteria for Nuclear Power Plant Assessments," February 28, 2006

LICENSE RENEWAL/ EXTENDED POWER UPRATES

MARIO V. BONACA

License Renewal

- **Performed interim (Subcommittee) reviews of three applications and final (Full Committee) review of two applications since October 2006**
- **Will perform interim review of one application and final review of two applications during the remainder of CY 2007**
- **Will perform three interim reviews and four final reviews in CY 2008**

- **Recommended continued operation of Palisades during the entire period of extended operation contingent on the resolution of three time-limited aging analysis issues associated with reactor pressure vessel integrity**

License conditions for Oyster Creek license renewal:

- Identify options to eliminate or reduce leakage in the refueling cavity liner
- Perform 3-dimensional finite element analysis of the drywell shell
- Increase frequency of drywell inspections and monitor two drywell trenches

Extended Power Uprates

- **Reviewed 5% power uprate amendment for Browns Ferry Unit 1**
  - **Containment overpressure credit during long-term LOCA and Appendix R fire scenarios at 120% power will require more complete evaluations**

- **Will review the extended power uprates for Browns Ferry Units 1, 2, and 3 after receiving complete safety evaluation reports**
- **Will review extended power uprates for Hope Creek and Susquehanna in CY 2007**

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

ACRSR-2224

November 17, 2006

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

Subject: REPORT ON THE SAFETY ASPECTS OF THE LICENSE RENEWAL APPLICATION FOR THE PALISADES NUCLEAR POWER PLANT

Dear Chairman Klein:

During the 537th meeting of the Advisory Committee on Reactor Safeguards, November 1-3, 2006, we completed our review of the license renewal application for the Palisades Nuclear Plant (PNP) and the final Safety Evaluation Report (SER) prepared by the NRC staff. Our Plant License Renewal Subcommittee also reviewed this matter during a meeting on July 11, 2006. During our review, we had the benefit of discussions with representatives of the NRC staff and the applicant, Nuclear Management Company, LLC (NMC). In addition, we had the benefit of input from the public. We also had the benefit of the documents referenced. This report fulfills the requirements of 10 CFR 54.25 that the ACRS review and report on all license renewal applications.

CONCLUSION AND RECOMMENDATION

The programs established and committed to by the applicant to manage age-related degradation provide reasonable assurance that PNP can be operated in accordance with its current licensing basis for the period of extended operation without undue risk to the health and safety of the public.

The NMC application for renewal of the operating license for PNP should be approved. Continued operation during the entire period of extended operation is contingent on the resolution of the issues associated with three Time-Limited Aging Analyses (TLAAs) related to reactor pressure vessel (RPV) integrity.

BACKGROUND AND DISCUSSION

PNP is a Combustion Engineering 2-loop pressurized water nuclear plant with a large, dry, ambient-pressure containment. PNP is located five miles south of South Haven, Michigan, on the eastern shore of Lake Michigan. The current power rating of the PNP is 2566 MWt, for a gross electrical output of 767 MWe. PNP was originally licensed to operate on February 21, 1971. NMC requested renewal of the PNP operating license for 20 years beyond the current license term, which expires on February 20, 2011.

In the final SER, the staff documented its review of the license renewal application and other information submitted by NMC and obtained during the audit and inspection conducted at the plant site. The staff reviewed the completeness of the applicant's identification of structures, systems, and components (SSCs) that are within the scope of license renewal; the integrated plant assessment process; the applicant's identification of the plausible aging mechanisms associated with passive long-lived components; the adequacy of the applicant's Aging Management Programs (AMPs); and the identification and assessment of TLAAs requiring review.

The NMC application is largely consistent with NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," issued in July 2001. All deviations from the GALL Report are documented in the application. The applicant identified the SSCs that fall within the scope of license renewal and performed a comprehensive aging management review for these SSCs. Based on the results of this review, the applicant will implement 24 AMPs for license renewal including existing, enhanced, and new programs. In the final SER, the staff concluded that the applicant has appropriately identified the SSCs within the scope of license renewal and that the AMPs described by the applicant are appropriate and sufficient to manage aging of long-lived passive components that are within the scope of license renewal. We concur with this conclusion.

The staff conducted an inspection and an audit. The inspection verified that the scoping and screening methodologies are consistent with the regulations and are adequately reflected in the application. The audit verified the appropriateness of the AMPs and the aging management reviews. Based on the inspection and audit, the staff concluded that these programs are consistent with the descriptions contained in the NMC license renewal application. The staff also concluded that the existing programs, to be credited as AMPs for license renewal, are generally functioning well and that an implementation plan has been established in the applicant's commitment tracking system to ensure timely completion of the license renewal commitments.

During our meetings with the staff and the applicant, we discussed the adequacy of programs proposed by NMC to manage aging of certain components that are projected to exceed acceptance limits during the period of extended operation.

The applicant identified the systems and components requiring TLAAs and reevaluated them for 20 additional years of operation. As required by 10 CFR Part 54, the applicant must identify any exemptions granted under 10 CFR 50.12 which rely on a TLAA and determine if that exemption should be continued for an additional 20 years of operation. No such exemption currently exists in the PNP licensing basis. The applicant reexamined 23 TLAAs. All of these TLAAs are valid, without restriction, for 20 more years of operation, except for three TLAAs associated with reactor vessel neutron embrittlement, namely: reactor vessel upper shelf energy, reactor vessel pressurized thermal shock, and reactor vessel pressure-temperature curves. In each of these cases, PNP will exceed the acceptance limits prior to the end of the extended period of operation.

To analyze the reactor vessel neutron fluence for purposes of RPV integrity evaluations, the applicant uses the methodology described in WCAP-15353, which is consistent with Regulatory Guide 1.190.

The applicant began using low neutron leakage cores in 1988 to reduce the neutron embrittlement of the reactor vessel to extend the time before exceeding the acceptance limits. However, the applicant predicts that the following acceptance limits will be exceeded:

- Upper Shelf Energy limit – exceed in 2021.
- Reactor Vessel Pressurized Thermal Shock (PTS) screening criterion – exceed in 2014.
- Pressure-Temperature limit curves – expire in 2014.

The staff's confirmatory calculations show reasonable agreement with the applicant's findings.

Upper Shelf Energy Limit. The applicant predicts this criterion will be exceeded in 2021. Appendix G of 10 CFR 50 requires RPV beltline materials to have Charpy upper shelf energy values no less than 50 ft-lb in the transverse direction in the base metal and along a weld for weld material. However, in accordance with Appendix G, Charpy upper shelf energy values below 50 ft-lb may be acceptable if it is demonstrated that lower Charpy upper shelf energy values will provide margins of safety against fracture (ductile tearing) equivalent to those required by ASME Code, Section XI, Appendix G. Regulatory Guide 1.99 describes two acceptable methods for determining the upper shelf energy values for RPV beltline materials.

Because the reactor vessel upper shelf energy limit will be exceeded prior to the end of the extended period of operation, the applicant must provide an analysis in accordance with 10 CFR Part 50, Appendix G at least three years prior to exceeding the upper shelf energy limit.

PTS Screening Criterion. The applicant predicts the criterion for axial welds and plates will be exceeded in 2014. 10 CFR 50.61 provides the fracture toughness requirements for protecting reactor vessels from the effects of PTS events. The end of life reference temperature (RT_{PTS}) value is the sum of a reference value for an unirradiated material, a shift in the reference value caused by exposure to high-energy neutron irradiation, and an additional margin to account for uncertainties.

If an applicant determines that the RPV will not meet the PTS screening criterion through the end of the facility's current license term, several actions must be taken. 10 CFR 50.61(b)(3), requires that an applicant implement a reasonably practicable flux reduction program in an effort to avoid exceeding the PTS screening criterion. If no reasonably practicable flux reduction program will meet this objective (as is true in the case of PNP) the applicant has several options. The applicant may submit a safety analysis in accordance with 10 CFR 50.61(b)(4) to demonstrate that the RPV can be operated beyond the 10 CFR 50.61 screening criterion. This safety analysis may include plant modifications. Such an analysis must be submitted three years prior to the time the RPV is projected to exceed the PTS screening criterion. In accordance with 10 CFR 50.61(b)(7), the applicant could propose to anneal the RPV in order to improve its material properties and permit continued operation. In accordance with 10 CFR 50.66, the applicant's thermal annealing plan would have to be submitted three years prior to when the facility's RPV is projected to exceed the PTS screening criterion.

Pressure-Temperature Limit Curves. Pressure-temperature limit curves are contained in the PNP technical specifications and are assessed against the limits in 10 CFR 50.60, Appendix G to 10 CFR 50, and Appendix G to Section XI of the ASME Code. The current pressure-temperature limits approved by the staff are valid beyond the current license term, but not through the extended period of operation. Based on the neutron fluence expected to be accumulated, the pressure-temperature limit curves will expire in 2014. Prior to entering the period of extended operation, the applicant must submit an amendment requesting a technical specification change and approval of new limits covering the period of extended operation beyond 2014.

The staff has concluded that the applicant has provided an adequate list of TLAAs. Further, the staff has concluded that the applicant has met the license renewal rule by demonstrating that the TLAAs have been projected to the end of the period of extended operation. In those cases where the current TLAAs do not cover the entire period of extended operation, the applicant must provide additional information in a timely manner and submit a license amendment for a technical specification change to extend these three TLAAs to cover the entire period of extended operation. We concur with the staff that the applicant has properly identified the applicable TLAAs, reviewed the associated analyses and licensing bases, and identified those instances where additional measures are needed to modify the TLAAs to cover the entire period of extended operation. We concur with the staff's conclusions and the resulting license conditions and commitments.

During our Plant License Renewal Subcommittee meeting on July 11, 2006, members of the Public provided comments and raised several questions. These comments and questions were recorded and are contained in the transcript of that meeting. The reference to the transcript that contains these comments and questions was provided to the Executive Director for Operations. Subsequently, the staff has responded to these questions and comments.

We agree with the staff that there are no issues related to the matters described in 10 CFR 54.29(a)(1) and (a)(2) that preclude renewal of the operating license for PNP. The programs established and committed to by NMC provide reasonable assurance that PNP can be operated in accordance with its current licensing basis for the period of extended operation without undue risk to the health and safety of the public. Continued operation during the entire period of extended operation is contingent on the resolution of the issues associated with three TLAAs related to RPV integrity. The NMC application for renewal of the operating license for PNP should be approved.

Sincerely,

/RA/

Graham B. Wallis
Chairman

References:

1. Safety Evaluation Report Related to the License Renewal of the Palisades Nuclear Power Plant, September 2006.
2. Palisades Nuclear Power Plant - Application for Renewed Operating Licenses, March 22, 2005
3. Safety Evaluation Report with Open Items Related to the License Renewal of the Palisades Nuclear Power Plant, June 2006
4. Audit and Review Report for Plant Aging Management Reviews and Programs (AMPs) (AMRs) - Palisades Nuclear Power Plant, October 20, 2005
5. Palisades Nuclear Power Plant, Inspection Report 05000255/2005009, December 28, 2005
6. Memorandum dated September 13, 2006, from John T. Larkins, Executive Director, ACRS, to Luis A. Reyes, Executive Director for Operations, Subject: Questions Raised by Members of the Public During the ACRS Subcommittee Meeting on Palisades Nuclear Plant License Renewal Application
7. Regulatory Guide 1.99 Revision 2, Radiation Embrittlement of Reactor Vessel Materials, May 1988
8. Regulatory Guide 1.190, Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence, March 2001
9. Palisades Reactor Pressure Vessel Fluence Evaluation, WCAP-15353, January 2000

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

ACRSR-2233

February 8, 2007

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

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE LICENSE RENEWAL APPLICATION FOR THE OYSTER CREEK GENERATING STATION

Dear Chairman Klein:

During the 539th meeting of the Advisory Committee on Reactor Safeguards, February 1-3, 2007, we completed our review of the license renewal application for the Oyster Creek Generating Station (OCGS) and the updated Safety Evaluation Report (SER) prepared by the NRC staff. Our Plant License Renewal Subcommittee also reviewed this matter during meetings on October 3, 2006 and January 18, 2007. During these reviews, we had the benefit of discussions with representatives of the NRC staff and its contractor Sandia National Laboratories (SNL), members of the public, and AmerGen Energy Company, LLC (AmerGen) and its contractors. We also had the benefit of the documents referenced. This report fulfills the requirements of 10 CFR 54.25 that the ACRS review and report on all license renewal applications.

RECOMMENDATIONS

1. With the incorporation of the conditions described in Recommendations 2, 3, and 4, the application for license renewal for OCGS should be approved.

2. We concur with the staff's proposal to impose license conditions to increase the frequency of the drywell inspections and to monitor the two drywell trenches to ensure that the sources of water are identified and eliminated.

3. The staff should add a license condition to ensure that the applicant fulfills its commitment to perform an engineering study prior to the period of extended operation to identify options to eliminate or reduce the leakage in the OCGS refueling cavity liner.

4. The staff should add a license condition to ensure that the applicant fulfills its commitment to perform a 3-D (dimensional) finite-element analysis of the drywell shell prior to entering the period of extended operation.

DISCUSSION

The Oyster Creek Generating Station is located in Lacey Township, Ocean County, New Jersey, approximately 2 miles south of the community of Forked River, 2 miles inland from the shore of Barnegat Bay, and 9 miles south of Toms River, New Jersey. The NRC issued the provisional operating license for OCGS on April 9, 1969 and the operating license on July 2,

1991. OCGS is a single unit facility with a single cycle, forced circulation boiling water reactor (BWR)-2 with a Mark 1 containment. The nuclear steam supply system was furnished by General Electric and the balance of the plant was originally designed and constructed by Burns & Roe. The licensed power output is 1930 MWt with a design electrical output of approximately 650 MWe. The applicant, AmerGen requested renewal of the OCGS operating license for 20 years beyond the current license term, which expires on April 9, 2009.

During the 1980s, the licensee discovered corrosion on the outside wall of the OCGS drywell shell. Although some corrosion had occurred in the upper shell region, the majority had occurred in a region near the base of the shell where the shell was partially supported by a sand bed. The licensee determined that water had been leaking through flaws in the refueling cavity liner during refueling operations. This water had migrated down the outside of the drywell shell and into the sand bed. As part of the corrective actions, the licensee removed the sand and applied an epoxy coating to the outside of the shell in the sand bed region. In addition, repairs were made to the refueling pool liner and the concrete drain trough under the refueling seal. These repairs reduced the leakage and routed any leakage to a drain line rather than down the outside of the drywell shell. To further reduce leakage, the licensee applied strippable coatings to the liner during all but one of the subsequent refueling outages. The licensee performed ultrasonic testing (UT) to determine the as-found condition of the drywell shell and performed a structural analysis in 1992 to demonstrate acceptability of the containment in the degraded condition.

The 1992 structural analysis was reviewed and approved by the NRC staff. This analysis included a determination of the stresses in the thinned region under the design pressure loads and an evaluation of the potential for buckling during normal operations and postulated accident conditions. The buckling analysis utilized American Society of Mechanical Engineers (ASME) Code Case N-284, Revision 1. The staff accepted the use of this Code Case in the 1992 analysis. In support of the review of the OCGS license renewal application, the staff had SNL perform a confirmatory structural analysis. Both analyses demonstrated that the drywell shell met the minimum ASME Code requirements for buckling. However, the amount of margin above the Code minimum depended on the applicability of the increase in the buckling capacity due to tensile stresses orthogonal to the applied compressive stresses computed according to the Code Case. During the January 18, 2007 meeting, the Subcommittee requested additional justification for using the increased capacity factor. At our February meeting, Dr. C. Miller, the author of the ASME Code Case, described the technical basis for the Code Case and presented test results to demonstrate that the increased capacity factor was applicable to OCGS. The increased capacity factor used in the 1992 analysis provided by the applicant was based on results for metal cylinders. Dr. Miller showed results of tests conducted on metal spheres which demonstrated that the results for cylinders were conservative for spherical shells. The staff reaffirmed its position that the use of the increased capacity factor is appropriate for the analysis of the OCGS drywell shell. We concur with this position.

The 1992 structural analysis was based on the assumption that the shell is uniformly thinned in the sand bed region. The applicant has committed to perform a 3-D finite-element analysis of the OGCS drywell to determine the margin of the shell in the as-found condition using modern methods. This analysis will provide a more accurate quantification of the margin above the Code required minimum for buckling. The applicant has committed to complete the analysis prior to the period of extended operation. We commend the applicant for this action and would like to be briefed by the staff on the results when they become available. Although it is anticipated that the analysis will demonstrate additional margin above the Code required minimum, the applicant should complete this analysis in a timely manner prior to entering the period of extended operation in order to identify and resolve any unexpected results. The analysis should include sensitivity studies to determine the degree to which uncertainties in the size of thinned areas affect the Code margins. The staff should impose a license condition to ensure that the applicant completes the analysis prior to entering the period of extended operation.

In 2006, the applicant performed additional UT and visual inspections of the drywell shell. When compared to the previous UT, the 2006 results confirmed that the corrective actions taken in the sand bed region had been effective and that the corrosion had been arrested or at least that the corrosion rates were very low (i.e., within the data scatter). The epoxy coating appeared in very good condition with no evidence of degradation which is also consistent with the conclusion that the corrosion has been effectively arrested. These examinations also demonstrated that the corrosion rate in the upper shell region and the embedded floor regions remained sufficiently low to demonstrate structural integrity during the period of extended operation. The applicant has committed to perform UT and visual inspections of the drywell shell during the period of extended operation. Because of the relatively small margin above the Code minimum against buckling in the sand bed region shown by current analyses, the staff is proposing a license condition to increase the frequency of drywell inspections and UT in the sand bed region to all 10 bays every other refueling outage for the extended period of operation. Increased inspections will result in additional radiation exposure to personnel involved in the inspections. Therefore, the applicant should be allowed to increase the period between inspections if it demonstrates increased margin through analysis or if the ongoing inspections continue to demonstrate that the corrosion has been sufficiently arrested. With this provision, we agree with this license condition.

The 2006 examinations revealed that when the cavity was flooded for refueling, water leakage was still occurring. This leakage of approximately 1 gallon per minute is well within the capacity of the drain as long as the drain system is working properly. The purpose of the drain system is to catch water that may leak past a failed refueling seal or liner and divert the water to sumps, and prevent it from coming into contact with the outside of the drywell shell. Leakage is not expected to occur as part of normal operation with properly maintained equipment and structures. The applicant has committed to continue monitoring for leakage of the refueling cavity liner and other water sources associated with the drywell. The applicant has also committed to complete an engineering study to identify cost-effective repair or replacement options to eliminate the refueling cavity liner leakage. The engineering study will be completed prior to entering the period of extended operation. We agree that efforts should be made to eliminate routine leakage in order to provide increased protection against further degradation. The staff should impose a license condition to ensure the study is completed by the applicant prior to the period of extended operation.

During the 2006 refueling outage, the applicant discovered water in two trenches that had been previously excavated to allow access to and inspection of the inside of the shell in the embedded region. The applicant determined that the water had come from normal operation and maintenance activities. The water had migrated to the trenches due to a blocked drain tube in the sub-pile area and the lack of a seal between the shell and concrete curb. The applicant repaired the drain tube and installed a seal in the gap between the shell and concrete curb. The applicant intends to fill these trenches after two consecutive outages in which no water is observed. Having the trenches open is beneficial for identifying drainage issues, but it increases the risk of additional corrosion because it provides an open area in which water can be trapped against the shell. The staff is proposing a license condition that would require the applicant to leave the trenches open and monitor them during each refueling outage until such time that the applicant can demonstrate that the water sources have been identified and eliminated. We agree with the monitoring of the trenches to ensure the elimination of the sources of water. However, leaving the trenches open longer than necessary increases the risk of future corrosion. Therefore, the applicant should not be unnecessarily delayed in repairing the trenches. With this provision, we agree with the license condition proposed by the staff.

In the updated SER, the staff documents its review of the license renewal application and other information submitted by AmerGen and obtained during an audit and inspections conducted at the plant site. The staff reviewed the completeness of the applicant's identification of structures, systems, and components (SSCs) that are within the scope of license renewal; the integrated plant assessment process; the applicant's identification of the plausible aging mechanisms associated with passive, long-lived components; the adequacy of the applicant's aging management programs (AMPs); and the identification and assessment of time-limited aging analyses (TLAAs) requiring review.

The OCGS application either demonstrates consistency with the Generic Aging Lessons Learned (GALL) Report or documents deviations from the approaches specified in the GALL Report. The staff reviewed this application in accordance with NUREG-1800, the "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants."

The applicant identified those SSCs that fall within the scope of license renewal. For these SSCs, the applicant performed a comprehensive aging management review. Based on the results of this review, the applicant will implement 57 AMPs for license renewal including existing, enhanced, and new programs. In the SER, the staff concludes that the applicant has appropriately identified SSCs within the scope of license renewal and that the AMPs described by the applicant are appropriate and sufficient to manage aging of long-lived passive components that are within the scope of license renewal. With the incorporation of the license conditions described in Recommendations 2, 3 and 4, we agree with this conclusion.

The staff conducted inspections and an audit of the license renewal application. The purpose of the inspections was to verify that the scoping and screening methodologies are consistent with the regulations and are adequately reflected in the application. In addition, the inspectors personally examined selected areas of the sand bed region to verify the condition of the epoxy coating. The audit confirmed the appropriateness of the AMPs and the aging management reviews. Based on the inspections and audit, the staff concluded that these programs are consistent with the descriptions contained in the OCGS license renewal application. The staff also concluded that the existing programs, to be credited as AMPs for license renewal, are generally functioning well and that the applicant has established an implementation plan in its commitment tracking system to ensure timely completion of the license renewal commitments.

The applicant identified those systems and components requiring TLAAs and reevaluated them for 20 more years of operation. Affected TLAAs include those associated with neutron embrittlement, metal fatigue, irradiation-assisted stress corrosion cracking, environmental qualification of electrical equipment, and stress relaxation of hold-down bolts. The staff concluded that the applicant has provided an adequate list of TLAAs. Further, the staff concluded that in all cases the applicant has met the requirements of the license renewal rule by demonstrating that the TLAAs will remain valid for the period of extended operation, or that the TLAAs have been projected to the end of the period of extended operation, or that the aging effects will be adequately managed for the period of extended operation. With the incorporation of the license conditions described in Recommendations 2, 3 and 4, we concur with the staff that OCGS TLAAs have been properly identified and that criteria supporting 20 more years of operation have been met.

With the incorporation of the license conditions described in Recommendations 2, 3, and 4, no issues related to the matters described in 10 CFR 54.29(a)(1) and (a)(2) preclude renewal of the operating license for OCGS. The programs established and committed to by AmerGen provide reasonable assurance that OCGS can be operated in accordance with its current licensing basis for the period of extended operation without undue risk to the health and safety of the public and the NRC should approve the AmerGen application for renewal of the operating license for OCGS.

Sincerely,

/RA/

William J. Shack
Chairman

References:

1. Updated Safety Evaluation Report Related to the License Renewal of Oyster Creek Generating Station, December 29, 2006.

2. Safety Evaluation Report with Open Items Related to the License Renewal of the Oyster Creek Generating Station, August 18, 2006.

3. Oyster Creek Generating Station- Application for Renewed Operating Licenses, July 22, 2005.

4. Supplemental Information Related to the Aging Management Program for the Oyster Creek Drywell Shell, Associated with AmerGen's License Renewal Application, June 20, 2006.

5. Audit and Review Report for Plant Aging Management Reviews and Programs- Oyster Creek Generating Station August 18, 2006.

6. Supplemental Response to NRC Request for Additional Information (RAI 2.5.1.19-1), dated September 28, 2005, Related to Oyster Creek Generating Station License Renewal Application, November 11, 2005.

7. Oyster Creek Generating Station - NRC License Renewal Inspection Report 05000219/2006007, September 21, 2006

8. Memorandum dated December 14, 2006 from Louise Lund to John Larkins, Subject: Review Background Materials for the Meeting of the License Renewal Subcommittee Scheduled on January 18, 2007, Related to the Interim Review of the License Renewal of the Oyster Creek Generating Station. ML063470557

9. Memorandum date December 8, 2006 from Michael P. Gallagher to the U.S. Nuclear Regulatory Commission, Subject: Submittal of Information to ACRS Plant License Renewal Subcommittee Related to AmerGen's Application for Renewed Operating License for Oyster Creek Generating Station. ML063470532

10. Sandia National Laboratories Report "Structural Integrity Analysis of the Degraded Drywell Containment at the Oyster Creek Nuclear Generating Station," January 2007

11. ASME Code Case N-284-1, "Metal Containment Shell Buckling Design Methods, Class MC, Section III, Division one, March 14, 1995."

12. Letter dated January 31, 2007, from Senator Frank Lautenberg, Senator Robert Menendez, Representative Christopher H. Smith, and Representative Jim Saxton to The ACRS.

13. Letter dated January 31, 2007 from Richard Webster, Rutgers Environmental Law Clinic to the ACRS, regarding the Safety Evaluation Report for Oyster Creek Nuclear Power Plant.

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

ACRSR-2235

February 16, 2007

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

SUBJECT: BROWNS FERRY NUCLEAR PLANT, UNIT 1, 5-PERCENT POWER UPRATE

Dear Chairman Klein:

During the 539th meeting of the Advisory Committee on Reactor Safeguards, February 1–3, 2007, we discussed the Tennessee Valley Authority's (TVA's) application for a 5-percent power uprate for Browns Ferry Nuclear Plant, Unit 1 (BFN1) and the associated safety evaluation. Our Subcommittee on Power Uprates had previously reviewed this matter on January 16–17, 2007. During our review, we had the benefit of discussions with representatives of the staff and the licensee. We also had the benefit of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

1. The TVA application for a 5-percent power uprate at BFN1 should be approved.

2. Granting of containment overpressure credit during long-term loss-of-coolant accident (LOCA) and 10 CFR Part 50 Appendix R fire scenarios at 120-percent of the original licensed thermal power (OLTP) will require support by more complete evaluations.

BACKGROUND

BFN1 is a General Electric boiling-water reactor (BWR/4) with a Mark-1 containment, located in Decatur, Alabama. BFN1 is one of three BWR/4 units at the Decatur site. All three units were originally licensed for operation at 3293 MWt. Units 1, 2, and 3 commenced operation in 1973, 1974, and 1976, respectively, and were shut down in 1985 to address management, technical, and regulatory issues. Units 2 and 3 (BFN2 and BFN3) were restarted in 1991 and 1995, respectively, and have been in operation since then. Units 2 and 3 were authorized to increase their maximum power by 5-percent (to 3458 MWt) in 1998. Unit 1 has been shut down since 1985, and TVA plans to restart it in 2007. The license expiration dates for all three units have been extended for 20 more years.

TVA has submitted applications for an extended power uprate (EPU) for each of the three units to 120 percent of OLTP. This will involve a 20-percent power uprate for Unit 1 and a 15-percent power uprate for Units 2 and 3. TVA has performed analyses and evaluations at 120-percent to support the EPU applications.

However, because all of the information necessary to support the review of the EPU is not yet available, the licensee has requested a two-step approach. This approach involves, as a first step, an interim approval of a 5-percent power uprate for Unit 1 to raise its power to the same level and operating conditions as Units 2 and 3. The second step will involve a 15-percent EPU for all three units later this year.

For the Unit 1 uprate to 105-percent, a higher steam and feedwater flow is achieved by increasing the reactor power along specified control rod and core flow lines and increasing reactor operating pressure by approximately 30 psi. This increase in steam flow will enable the plant to increase its electrical output. Additionally, the TVA application shows that Unit 1 requires containment overpressure credit to ensure adequate net positive suction head (NPSH) for the emergency core cooling system pumps during LOCAs, anticipated transients without scram (ATWS), station blackouts (SBOs), and Appendix R events. It is important to note that the amount of the requested overpressure credit and its requested duration for each of the above events is based on analyses performed at 120-percent power.

With few exceptions, such as core design and steam dryers, the licensee has used the analyses performed at EPU conditions to support the 5-percent power uprate application.

DISCUSSION

When design-basis large-break LOCA, ATWS, SBO, and Appendix R events were analyzed at BFN1 at 120-percent power using current design-basis assumptions and methodologies, the available NPSH was found to be insufficient to avoid cavitation of the residual heat removal (RHR) and core spray pumps. The need for increased NPSH occurs because, at the higher power level, the suppression pool heats up more than at the OLTP. In the calculations performed to support the OLTP at BFN1, containment pressure was assumed to be atmospheric when computing the available NPSH.

In its application, TVA requests approval to change its licensing-basis methodology to include credit for containment accident pressure in determining available NPSH for emergency core cooling pumps for the above scenarios. Using conservative methods and a containment leak rate consistent with its technical specifications, TVA has calculated a conservative lower bound for the time-dependent pressure in containment that would result from each of these scenarios at 120-percent power. The incremental pressure credits that are requested for these scenarios are less than these computed pressures, except for the first 10 minutes of the LOCA scenario.

During the first 10 minutes of the LOCA scenario, a design-basis analysis indicates that the containment accident pressure is not sufficient to prevent cavitation of the RHR pumps for a period of about 4 minutes. The RHR pump manufacturer has stated that this period of cavitation should not inflict sufficient damage to disable the pumps, and a pump cavitation test performed on a similar RHR pump at BFN3 under similar accident conditions has confirmed this assertion. The licensee has also shown that if just two of the many conservatisms associated with this analysis were relaxed, containment overpressure would be sufficient to prevent cavitation of the RHR pumps.

After 10 minutes into the design-basis LOCA scenario, the operator throttles and realigns RHR and core spray pumps. During this long-term phase of the LOCA, the RHR pumps do not need containment overpressure, but the core spray pumps will continue to need an overpressure of up to 3 psi for 23 hours.

The ATWS and SBO scenarios need an overpressure of less than 2 psi for a little more than 1 hour. The licensee indicates that overpressure is not needed in the ATWS scenario when the analysis employs a more realistic model. For the SBO scenario, the licensee indicates that containment overpressure would not be needed for the first 3 hours of the required 4-hour coping duration. The availability of 4 offsite power lines and 8 onsite diesels with interconnecting capability makes it probable that power can be recovered within the first 3 hours of the event.

The Appendix R fire scenario is intended to demonstrate that safe shutdown can be achieved during a fire in any plant location using only protected plant equipment and operator actions that can be taken from several plant locations. This scenario at BFN1 includes the assumptions of loss of offsite power, loss of ultimate heat sink, evacuation of the control room, start of the one protected RHR pump from the remote shutdown panel, and blowdown to the suppression pool. In this bleed-and-feed scenario, up to 10 psi of overpressure are needed for 69 hours. To develop sufficient containment overpressure to allow continued operation of the RHR pumps without cavitation, the licensee will modify procedures to direct the operator to turn off the drywell coolers during the first 2 hours of this scenario. This is the limiting scenario for which overpressure credit is required. Although the thermal-hydraulic evaluations for this scenario are intended to be realistic, the licensee argues that this is an overly conservative scenario, because more equipment is likely to be available than postulated in the scenario. If two or more RHR pumps are available, credit for containment overpressure is not needed.

In determining whether credit for containment overpressure should be granted, we have noted in previous reports a number of important considerations. They include whether practical alternatives exist, such as the replacement of pumps with new pumps with less restrictive NPSH requirements; whether the containment design provides a positive indication of integrity before the event, as is the case in inerted containments; and the length of time for which containment pressure credit is required and the margin between the containment pressure required and the expected minimum containment pressure. The ultimate consideration is the risk significance of granting credit for containment overpressure.

Because of the plant configuration, the extent of modification required, and the worker dose that would be involved, we conclude that there are no practical design modifications that would preclude the need to consider the request for containment overpressure credit for most of the scenarios. However, for the Appendix R scenario, protecting a second RHR pump would eliminate the need for the credit and may be a feasible alternative.

The BFN1 containment is inerted. There is, then, a lower likelihood of significant preexisting containment leakage.

For the ATWS and SBO scenarios, the magnitude of credit and the period of time for which credit is required are small. For the short-term LOCA scenario, pump cavitation is unlikely, and the duration of containment overpressure credit is small.

For the long-term LOCA scenario, the licensee's design-basis analysis assumed the worst single failure, which is the loss of one train of emergency power. Allowing no credit for containment overpressure is equivalent to assuming an additional failure that causes loss of the overpressure. Thus, for all scenarios involving only a single failure, sufficient NPSH is available to ensure that pump cavitation will not occur. The licensee and the staff have also identified many conservatisms associated with the design-basis analysis of the long-term LOCA scenario. Limited sensitivity calculations provided by the licensee suggest that it is possible that, on a more realistic but still conservative basis, the temperature of the suppression pool would not become high enough to require credit for containment overpressure. Because of the smaller amount of credit required for operation at 105-percent power, these analyses are sufficient to support operation at that power level, but operation at EPU conditions should be supported by more defensible calculations. When realistic analyses are used as an alternative to conservative design-basis calculations, explicit consideration of uncertainties should be included in the analysis.

Because of the amount of time for which credit is required and the amount of credit required, the Appendix R fire scenario is the limiting event for which containment overpressure credit is required. The staff presented a risk evaluation of containment overpressure credit for this scenario that showed that the contribution to core damage frequency associated with this scenario is small and represents a small fraction of the BFN1 core damage frequency. However, this assessment did not include fires initiated by external events, such as earthquakes and tornados. The inclusion of these initiators in the risk evaluation is likely to increase the risk associated with the Appendix R scenario. To use risk arguments to justify overpressure credit for this scenario, the licensee and the staff need to provide a more complete analysis including all initiators. Because of the smaller amount of credit required for operation at 105-percent power, this extensive analysis is not needed to support operation at that power level.

TVA has implemented substantial hardware changes to support successful restart and operation under EPU conditions. Significant portions of several major systems have been replaced, including all recirculation piping and the entire reactor water cleanup system. The licensee stated that a large number of valves and all valve packing materials have been replaced. It has upgraded the condensate and feedwater system to satisfy EPU feedwater demands and replaced all three feedwater pumps and all three condensate booster pumps, as well as the condensate pump impellers. After these changes, sufficient margin exists so that a trip of one feedwater pump will not result in a reactor trip.

As part of the BFN1 restart test program, condensate, condensate booster, and feedwater pump trip tests will be performed at 105-percent power to demonstrate the adequacy of the condensate and feedwater system and to test the integrated response of control systems associated with feedwater level control and reactor pressure control. The staff has made these tests a license condition for this application. In addition, the licensee has agreed to perform two large transient tests at 105-percent power. They are the main steam isolation valve closure and the generator load-reject tests. These tests are also a license condition for this application. Given the number of modifications and upgrades implemented at BFN1, we agree that these large transient tests need to be performed.

Higher steam and feedwater flow rates at uprated conditions may lead to an increase in flow-accelerated corrosion (FAC) for some components. TVA has developed a CHECWORKS-based FAC model that has been successfully used to predict FAC rates in susceptible locations of BFN2 and BFN3 piping. Because Unit 1 is in the process of a recovery effort following an extended shutdown, plant-specific FAC data are not yet available. The licensee argued that Unit 2 and Unit 3 data are reasonably representative of Unit 1 because of similarities in piping geometry and materials. Based on Unit 2 and Unit 3 data and industry experience with the current materials, the first cycle of operation of Unit 1 is not expected to experience FAC-related failures. Measurements to be performed at the end of the first cycle of operation will provide baseline data for the Unit 1 plant-specific CHECWORKS model for future prediction of FAC rates. We concur with the staff that this represents an appropriate course of action to manage FAC at BFN1.

Increased flow rates have the potential to induce vibrations that could lead to failure of components. Because of the previous experience at Quad Cities, the steam dryers have been the primary focus of attention. After receiving the 5-percent power uprate license amendment, TVA will collect steam dryer data and perform an analysis of the steam dryers.

Confidence in the capability of the BFN1 steam dryers to maintain structural integrity at 105-percent OLTP conditions is based on the similarity of the three BFN units and the successful operating experience of the dryers of BFN2 and BFN3 at 105-percent OLTP since 1998. Before restart, the licensee will add structural reinforcements to the Unit 1 steam dryers based on the operating experience at Units 2 and 3.

The reactor operating domain is defined so that (1) the core will not be operated in an unstable regime, (2) the minimum critical power ratio is low enough to prevent dryout of the fuel pins, and (3) the linear heat generation rate is low enough to ensure the integrity of the fuel cladding during steady-state and transient conditions. The boundaries of this operating domain are based on neutronic and thermal-hydraulic calculations performed by General Electric. The staff has reviewed and approved the computer codes used in these analyses, which were performed at EPU conditions. TVA has submitted to the staff the results of the Supplemental Reload Licensing Topical Report for 105-percent power specifically addressing the safety limit minimum critical power ratio. The staff will review this report to confirm that the analyses remain applicable for the operation throughout the upcoming operating cycle at 105-percent power. Unit 1 will implement the BWR Owners Group Long-Term Stability Solution Option III for the automatic detection and suppression of stability-related power oscillations.

Only minor changes have been made in the emergency and abnormal operating procedures to accommodate the power uprate. A significant change to operator actions required to support the power uprate is the request that credit be given for manual action to terminate drywell cooling within 2 hours of entry into the safe shutdown procedure during an Appendix R event. This manual action can be performed in the control room or in two remote shutdown locations outside of the control room. The operator has 2 hours to perform this action, and the time required to terminate drywell cooling is well within this time frame. We concur with the staff that the revision to the Appendix R fire safe-shutdown operating instructions to terminate drywell cooling within 2 hours of entry into the procedure is acceptable.

TVA has performed a systematic assessment of the time available to perform the actions credited in the Updated Final Safety Analysis Report versus the time necessary to complete such actions before and after the power uprate. The uprate has not significantly affected operator action times. The main impact on operator action time at the EPU level is a decrease in the time available to complete initiation of the containment atmospheric dilution system. The time available at the current power level is 42 hours. The time available at EPU conditions is 32 hours. Therefore, there is ample time to perform an operator action that requires no more than 5 minutes to complete.

We conclude that the TVA application for a 5-percent power uprate at BFN1 should be approved.

Sincerely,

/RA/

William J. Shack
Chairman

References:

1. Memorandum from J.E. Dyer, Director, Office of Nuclear Reactor Regulation, to John Larkins, Executive Director, Advisory Committee on Reactor Safeguards, "Transmittal of the Browns Ferry 105-Percent Power Level Uprate Safety Evaluation for Unit 1," December 1, 2006.

2. Letter from T. Abney, Manager of Licensing and Industry Affairs, Tennessee Valley Authority, to the NRC, "Browns Ferry Unit 1—Proposed Technical Specifications Change TS–431, Request for License Amendment, Extended Power Uprate Operation," June 28, 2004.

3. Letter from W. Crouch, Manager of Licensing and Industry Affairs, Tennessee Valley Authority, to the NRC, "Browns Ferry Unit 1—Technical Specification Change TS-431, Supplement 1, Extended Power Uprate," September 22, 2006.

4. Letter from G. Wallis, Chairman, Advisory Committee on Reactor Safeguards, to Nils J. Diaz, Chairman, NRC, "Vermont Yankee Extended Power Uprate," January 4, 2006.

5. Letter from G. Wallis, Chairman, Advisory Committee on Reactor Safeguards, to Nils J. Diaz, Chairman, NRC, "R.E. Ginna Extended Power Uprate Application," May 22, 2006.

6. Letter from G. Wallis, Chairman, Advisory Committee on Reactor Safeguards, to Nils J. Diaz, Chairman, NRC, "Beaver Valley Extended Power Uprate Application," May 22, 2006.

HUMAN RELIABILITY ANALYSIS MODELS

George E. Apostolakis

The Committee should work with the staff and external stakeholders to evaluate the different Human Reliability models in an effort to propose either a single model for the agency to use or guidance on which model(s) should to be used in specific circumstances (SRM, November 8, 2006)

- **The staff and EPRI are making progress in developing a plan to evaluate several human reliability analysis models**
- **The goals and important milestones of the project will need to be clearly articulated**

- **The objective should be to develop a common understanding of the relative importance of factors affecting human performance and ways in which they could be integrated into analyses**
- **Achieving this objective will allow the staff to develop guidance on which model(s) should be used for specific regulatory applications**

- **The staff is currently organizing an HRA Empirical Study to perform model-to-model comparisons to assess the strengths and weaknesses of HRA models using the simulator in Halden, Norway**
- **ACRS views this study as part of the broader effort to collect evidence regarding the validity of HRA models**

- **The Empirical Study by itself will not be sufficient to develop meaningful quantitative estimates of the probability of errors**
- **Additional evidence should be collected from operating experience, especially the Augmented Inspection Team reports on past incidents**

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

ACRSR-2247

April 23, 2007

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

SUBJECT: HUMAN RELIABILITY ANALYSIS MODELS

Dear Chairman Klein:

In a November 8, 2006 Staff Requirements Memorandum (SRM), resulting from the October 20, 2006 meeting with the Advisory Committee on Reactor Safeguards (ACRS), the Commission requested that the ACRS "work with the staff and external stakeholders to evaluate the different Human Reliability models in an effort to propose either a single model for the agency to use or guidance on which model(s) should be used in specific circumstances." During the 541st meeting of the ACRS, April 5-7, 2007, we discussed proposed plans by the NRC staff and the Electric Power Research Institute (EPRI) for moving forward to evaluate several human reliability analysis (HRA) models used by staff and industry. Our Subcommittee on Reliability and Probabilistic Risk Assessment also reviewed this matter on March 22, 2007. During these meetings, we had the benefit of discussions with representatives of the NRC staff and EPRI, and of the documents referenced.

CONCLUSION

The staff and EPRI are making progress in developing a plan to evaluate several human reliability analysis models.

DISCUSSION

Insights and results from probabilistic risk assessments (PRAs) and models for human performance provide valuable input to many regulatory decisions. NUREG-1842, "Evaluation of Human Reliability Analysis Methods Against Good Practices," lists several different methods for HRA in support of PRAs. Even within the agency, the staff uses multiple approaches to human reliability analysis for actions following an initiating event.

ATHEANA (A Technique for Human Event Analysis) is the result of a multi-year research effort supported by the Office of Nuclear Regulatory Research. Its underlying premise is that significant human errors occur as a result of a combination of influences associated with plant conditions and specific human-centered factors that may lead plant personnel to perform unsafe acts. ATHEANA provides a systematic process for the identification of these combinations of influences (the "error-forcing contexts"). Given an error-forcing context, ATHEANA requires that a group of experts develop the probabilities of unsafe acts.

In 1994, an NRC HRA model was developed to support the Accident Sequence Precursor (ASP) Program. This is the Accident Sequence Precursor /Standardized Plant Analysis Risk (ASP/SPAR) HRA Model. This model was updated in 1999 and renamed SPAR-H. NRC staff analysts use it for risk-informed regulatory activities such as: the Significance Determination Process Phase 3, the development of an integrated risk-informed performance measure for the Reactor Oversight Process, and the analysis of operating experience data to identify precursors to severe accident sequences. SPAR-H relies on eight performance shaping factors (PSFs) that are deemed to be capable of influencing human performance. These include available time for action, stress level, and experience and training. An error probability is provided that corresponds to all the PSFs being at their nominal values. Adjustment factors for this baseline probability are given for PSF values other than nominal.

During the discussion of draft NUREG-1852, "Demonstrating the Feasibility and Reliability of Operator Manual Actions in Response to Fire," the staff presented another approach for evaluating human actions. This approach is "deterministic" in the sense that it requires a demonstration that the sum of the time to detect the fire and the time to implement actions to control it is less than the available time (i.e., before an undesirable consequence occurs) by a "time margin." This time margin accounts for the unquantified uncertainties. Besides being "deterministic," this approach focuses on time in contrast with the ATHEANA and SPAR-H models, which treat time as one of the PSFs.

EPRI has developed an HRA Calculator, which is a software tool designed to facilitate a standardized approach to HRA. The HRA methodologies implemented in the Calculator follow the framework established in EPRI's Systematic Human Action Reliability Procedure (SHARP), the Cause-Based Decision Tree Method (CBDTM), Human Cognitive Reliability/Operator Reactor Experiments (HCR/ORE), the Accident Sequence Evaluation Program (ASEP) Human Reliability Analysis Procedure, and the Technique for Human Error Rate Prediction (THERP). The EPRI models focus on the available time for action to a greater extent than ATHEANA and SPAR-H.

Evaluations of extended power uprate amendments have shown that the most significant impact on core damage frequency is the reduction of the time available for operator action. Estimates are made of the changes in human error probabilities as a result of the shorter times. We understand that these estimates are usually produced using the EPRI Calculator, which, to our knowledge, has not been evaluated by the staff.

Analysts can obtain widely different results for human error probabilities. In mid 1980s, the Ispra Joint Research Center of the European Commission organized a benchmark exercise in which 15 teams from 11 countries used a number of HRA models available at the time to estimate the probability of the crew not responding correctly to a transient. The results produced by the teams using the same HRA model differed by orders of magnitude. The results produced by a single team using a number of HRA models also differed by orders of magnitude. Although these results are fairly old now, it is important to understand whether they are still representative of the model uncertainties present in HRA.

The staff and EPRI are in the process of developing a plan that is intended to lead to an integrated approach to evaluate various HRA models. The goals and important milestones of the project will need to be clearly articulated.

The staff should compare the NRC and EPRI models with respect to their basic assumptions and intended use. An evaluation of the validity of these assumptions and the supporting evidence should be performed. The objective should be to develop a common understanding of the relative importance of factors affecting human performance and ways in which they could be integrated into analyses. Achieving this objective will allow the staff to develop guidance on which model(s) should be used for specific regulatory applications.

The staff is currently organizing an HRA Empirical Study whose objective is to perform model-to-model comparisons to assess the strengths and weaknesses of HRA models. Various operator crews will run scenarios, similar to those appearing in PRAs, at the simulator in Halden, Norway. Teams of analysts will then analyze the human actions appearing in these scenarios using HRA models of their choice. The results will be compared to produce insights with respect to the assumptions the teams made and how the models were applied.

We view this Empirical Study as part of the broader effort to collect evidence regarding the validity of HRA models. Its anticipated benefits should be defined carefully. The study may provide useful qualitative information on crew performance and the factors that influence it. However, the Empirical Study by itself will probably not be sufficient to develop meaningful quantitative estimates of the probability of errors.

Additional evidence should be collected from operating experience, especially the Augmented Inspection Team reports on past incidents. The staff is already evaluating the operating experience in the Human Event Repository and Analysis System (NUREG/CR-6903). These sources of information should be used to enhance the insights gained from the Empirical Study. A large amount of data can be collected from licensee simulators. These data could complement the Halden results. It may also be beneficial to compare the Halden results against data from similar experiments at a U.S. plant simulator.

We look forward to working with the staff in formulating the details of the plan and its implementation.

Sincerely,

William J. Shack

William J. Shack
Chairman

References:

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ABBREVIATIONS

ACRS	Advisory Committee on Reactor Safeguards
CFR	Code of *Federal Regulations*
CY	Calendar year
DG	Draft guide
EPR	Evolutionary Power Reactor
EPRI	Electric Power Research Institute
ESBWR	Economic Simplified Boiling Water Reactor
GSI	Generic Safety Issue
HRA	Human reliability analysis
I&C	Instrumentation and control
LOCA	Loss-of-coolant accident
NGNP	Next Generation Nuclear Plant
NRC	Nuclear Regulatory Commission
PBMR	Pebble Bed Modular Reactor
PRA	Probabilistic risk assessment
PWR	Pressurized water reactor
SER	Safety evaluation report
SPAR	Standardized Plant Analysis Risk Model
SRM	Staff Requirements Memorandum
SRP	Standard Review Plan
U.S.	United States