COMMISSION BRIEFING SLIDES/EXHIBITS

MEETING WITH ACRS

DECEMBER 8, 2005

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

November 21, 2005

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

FROM: John T. Larkins
Executive Director, ACRS

SUBJECT: ACRS MEETING WITH THE NRC COMMISSIONERS DECEMBER 8, 2005 - SCHEDULE AND BACKGROUND INFORMATION

The ACRS is scheduled to meet with the NRC Commissioners between 1:00 and 3:00 p.m. on Thursday, December 8, 2005 to discuss the topics listed below. Background materials related to these items are attached:

TOPICS	PRESENTERS	ESTIMATED TIME
INTRODUCTION	Nils J. Diaz NRC Chairman	
ACRS PRESENTATIONS		
1. Overview • Major Accomplishments • License Renewal • Early Site Permits • Future ACRS Activities	Graham B. Wallis ACRS Chairman	10 minutes
2. Proposed Alternative Embrittlement Criteria in 10 CFR 50.46	Dana A. Powers	10 minutes
3. Issues Related to New Plant Licensing (including Technology-Neutral Framework)	Thomas S. Kress	15 minutes
4. Fire Protection Matters	George E. Apostolakis	10 minutes
5. Power Uprate Technical Issues	Richard S. Denning	15 minutes

CLOSING REMARKS Nils J. Diaz
NRC Chairman

Attachment:
As stated

NOTE: Estimated times are for presentation only and do not include the time set aside for Commissioners' questions and answers by the ACRS.

ACRS MEETING WITH THE U.S. NUCLEAR REGULATORY COMMISSION

December 8, 2005

OVERVIEW

Graham B. Wallis

Accomplishments

- **Since our last meeting with the Commission on April 7, 2005, we issued 25 Reports.**
- **Topics included:**
  - **Risk-informed alternatives to the single failure criterion**
  - **Assessment of the quality of selected NRC research projects**

- **Digital Instrumentation and Control Systems Research Plan**
- **Proposed Revision 4 to Reg. Guide 1.82, "Water Sources for Long-Term Recirculation Cooling Following a LOCA"**
- **Generic Letter on Grid Reliability and the Impact on Plant Risk and Operability of Offsite Power**

License Renewal

Since April 2005:

- **Completed review of five applications (Millstone, D.C. Cook, Arkansas Nuclear One, Unit 2, Point Beach, and Farley)**
- **Performed Interim Review of Browns Ferry Units 1, 2, and 3**

- Completed review of the generic license renewal guidance documents (SRP, GALL Report, and Regulatory Guide)
- Will perform final review of five applications and interim review of one application in CY2006

Early Site Permits

- Completed review of the North Anna ESP application
- Performed interim review of the Grand Gulf and Clinton ESP applications
- Will complete review of the Grand Gulf application in December 2005 and Clinton application in March 2006

Future Activities

- Advanced reactor designs
- Assessment of research quality
- Early site permit applications
- Extended core power uprates
- Fire protection
- High-burnup fuel and cladding issues

Future Activities

- Human factors and human reliability assessment
- License renewal applications
- Materials and Metallurgy
- Operating plant issues
- PWR sump performance
- Report on the NRC Safety Research Program

Future Activities

- Resolution of GSIs
- Revisions to SRP
- Risk-informing 10 CFR 50.46
- Rules and regulatory guidance
- Safety management
- Standardized plan analysis risk (SPAR) models
- Technology-neutral framework
- Thermal-hydraulic codes

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

June 10, 2005

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

SUBJECT: DRAFT COMMISSION PAPER ON "RISK-INFORMED ALTERNATIVES TO THE SINGLE FAILURE CRITERION"

Dear Chairman Diaz:

During the 523rd meeting of the Advisory Committee on Reactor Safeguards, June 1-3, 2005, we reviewed the draft Commission Paper, "Risk-Informed Alternatives to the Single Failure Criterion." During our review, we had the benefit of discussions with the NRC staff and the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

1. The staff has conducted a useful review of the role of the single failure criterion in the current regulatory system, defined desirable attributes of risk-informed alternatives, and developed some potential alternatives to the single failure criterion.

2. We concur with the staff that it is premature to select any particular alternative at the present time.

3. Additional input from stakeholders should be sought to determine if there is sufficient benefit to justify the resources that will be required to proceed with development of a risk–informed alternative.

4. We concur with the staff that any follow-up activities to risk-inform the single failure criteria should be included and prioritized in the program plan being developed for a risk-informed, performance-based revision to 10 CFR Part 50.

DISCUSSION

In response to a Staff Requirements Memorandum (SRM), dated March 31, 2003, the staff and its contractors have prepared a report, "Technical Work to Support Evaluation of a Broader Change to the Single Failure Criterion," that examines risk-informed alternatives to the single failure criterion. Although the Commission directive was associated with General Design Criterion (GDC) 35 and the emergency core cooling system (ECCS) acceptance criteria, the staff has examined alternatives to the single failure criterion that could apply to all safety (and non-safety) functions of the plant.

Single failure criterion requirements are part of the GDC. They are also addressed in the guidance for the analysis of some of the Design-Basis Accidents (DBAs) in Chapter 15 of Regulatory Guide 1.70 and the Standard Review Plan. The intent of the single failure criterion requirements is to achieve high safety system reliability through redundancy. The search for the most limiting single failure leads to a systematic study of design weaknesses and has generally resulted in robust designs.

However, it is evident from operating experience and risk analyses that the single failure criterion has not always succeeded in assuring adequate reliability. Common-cause failures, multiple independent failures, failures of support systems, multiple failures caused by spatial dependencies, and multiple human errors may not be mitigated by redundant system design alone. The NRC has imposed additional requirements for diversity and redundancy to increase system reliabilities through the station blackout rule, the anticipated transient without scram rule, and the post-Three Mile Island accident requirement to increase the availability of the auxiliary feedwater systems of pressurized water reactors.

The requirements for redundancy imposed by the single failure criterion may result in unnecessary burden with little risk benefit. Studies carried out by the staff with the Standardized Plant Analysis Risk (SPAR) models to examine the effect of system and functional redundancy on core damage frequency (CDF) showed that the impact of the redundancy of different systems on CDF varied by two orders of magnitude. Reducing redundancy in some cases led to large increases in CDF, and in others to virtually no change in CDF. Similarly, the single failure requirements in the analysis of some DBAs sometimes focus attention on events with very low frequency that may in fact have low risk significance.

Currently, changes in single failure criterion requirements are considered in the context of specific licensing issues as they arise (e.g., large-break loss-of-coolant accident (LBLOCA) redefinition). One of the alternatives the staff has considered is to continue with this current approach, which focuses resources on the most important issues. In the draft Commission Paper, this is referred to as the "baseline alternative." A related topic, the LOCA/loss-of-offsite power requirement, is already being dealt with as a separate issue.

The staff's Alternative 1 attempts to risk-inform DBA analyses. Sequence frequencies, obtained using probabilistic risk assessment (PRA) models and data, would be used to determine the failure events to be postulated in DBA analyses. Both removals and additions to the current set of design-basis sequences would be possible. Failure events associated with sequences with sufficiently low frequency would no longer have to be postulated. Eliminated failure events could include both initiating events and the assumed single failure postulated in current DBA analyses. The licensee would be required to demonstrate using the plant PRA that the collective frequency of design-basis sequences excluded from DBA analyses is small. Plant changes proposed based on Alternative 1 would have to be consistent with Regulatory Guide 1.174 guidelines.

Alternative 2 would risk-inform the application of the single failure criterion to safety systems based on their safety significance. A risk-informed process would be defined to categorize the safety significance of all plant systems. Taking advantage of current categorization processes, this alternative would expand on the 10 CFR 50.69 approach. Various reductions in the requirements for redundancy for RISC–3 (safety-related, low safety significance) components would be considered.

Alternative 3 is a more systematic approach to evaluating reliability requirements that recognizes the importance of diversity as well as redundancy in assuring high reliability. It would provide quantitative measures of the reliability that has been achieved. More redundancy and diversity would be required in response to more frequent events, and less in response to infrequent events. Licensees would choose target reliability values for each safety function (typically at the train level), and would show that these targets satisfy the functional objectives and the top-level objectives (CDF and large early release frequency). Each safety function would be analyzed using the PRA to show that the function-level reliability target is met. Methods would have to be developed to define the concept of "noncompliance" with set reliability targets. This is a generic challenge for performance-based requirements.

The resources required for Alternatives 1, 2, and 3 are more substantial than proceeding with the current approach, but more systematic approaches could lead to a greater coherency in requirements. As the staff has noted, other alternatives are possible, and not all the technical and implementation difficulties with these alternatives have been addressed. For example, Alternatives 2 and 3, which focus on the role of the single failure criterion in increasing reliability, may have to address the resulting impact on the role of the single failure criterion in DBAs. Thus Alternatives 2 and 3 may not be independent of Alternative 1 or some variation of it. Because of the preliminary nature of the work, the staff does not recommend any particular alternative at the present time. We concur with the staff that such a selection would be premature.

The staff has carried out this effort in response to the SRM without sufficient input from stakeholders. Before further work is performed, the staff should seek additional stakeholder input to determine if there is sufficient benefit to justify the resources that will be needed to proceed with development beyond that needed for the baseline alternative. As directed in the SRM dated May 9, 2005, the Office of Nuclear Regulatory Research will work with the Office of Nuclear Reactor Regulation to develop a formal program plan to make a risk-informed, performance-based revision to 10 CFR Part 50. We agree with the staff that any follow-up activities to risk-inform the single failure criterion should be included and prioritized in this program plan.

Sincerely,

/RA/

Graham B. Wallis
Chairman

REFERENCES:

1. Memorandum dated May 19, 2005, from Charles E. Ader, Director, Division of Risk Analysis and Applications, RES, to John T. Larkins, Executive Director, ACRS, Subject: Transmittal of Draft Commission Paper Entitled, "Risk-Informed Alternatives to the Single Failure Criterion," (Pre-Decisional For Internal ACRS Use Only).

2. Memorandum dated May 6, 2005, from Charles E. Ader, RES, to John T. Larkins, Executive Director, ACRS, Subject: Transmittal of Draft Report Entitled, "Technical Work to Support Evaluation of Broader Change to the Single Failure Criterion," (Pre-Decisional For Internal ACRS Use Only).

3. Staff Requirements Memorandum dated March 31, 2003, from Annette L. Vietti-Cook, Secretary, to William D. Travers, EDO, Subject: Staff Requirements - SECY-02-057 - Update to SECY-01-0133, "Fourth Status Report on Study of Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50 (Option 3) and Recommendations on Risk-Informed Changes to 10 CFR 50.46 (ECCS Acceptance Criteria)."

4. Regulatory Guide 1.174, Revision 1, November 2002, An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis.

5. 10 CFR § 50.69 Risk-Informed categorization and treatment of structures, systems and components for nuclear power reactors.

6. Memorandum to L. Reyes, EDO, from A. Vietti-Cook, SECY, dated May 9, 2005, Subject: Staff Requirements - Briefing on RES Programs, Performance, and Plans, 9:30 am, Tuesday, April 5, 2005, Commissioners' Conference Room, One White Flint North, Rockville, Maryland (Open to Public Attendance) [Refer to: M050405]

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

July 15, 2005

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

SUBJECT: ASSESSMENT OF THE QUALITY OF THE NRC RESEARCH PROJECTS

Dear Chairman Diaz:

In its April 25, 2005 Staff Requirements Memorandum, the Commission requested the ACRS to "provide the Commission a list of research projects it intends to review in the short term as part of its assessment of research quality, with an indication of the methodology the Committee will use for the reviews." This report responds to this Commission request.

Throughout its history, an essential activity of the ACRS has been reviewing the research sponsored by the NRC. Currently, we conduct review of research in four ways:

- Review of research conducted in support of specific regulatory activities
- Episodic review of particularly important ongoing research
- Biennial review of the technical and programmatic aspects of the overall reactor safety research program
- Review of the quality of selected research projects

Our assessments of supporting research and episodic review of significant ongoing research are discussed in individual reports. Our biennial review of the overall reactor safety research program is provided in a report to the Commission (successive volumes of NUREG-1635).

We have recently undertaken the in-depth assessment of the quality of selected research projects in response to a request from the Director of the Office of Nuclear Regulatory Research (RES). The Director requested us to do these reviews to meet the requirement of the Government Performance and Results Act (GPRA) that there be an independent quality review of Government-sponsored research. This independent review is required to include quantitative assessments so that research sponsors can demonstrate improvements in research quality over the years. We have undertaken this review in partial fulfillment of the role we assumed when we replaced the Nuclear Safety Research Review Committee as directed by the Commission.

During fiscal year (FY) 2004, we conducted a trial review of the quality of selected research projects. Based on the outcomes of this trial review, we have established the following review process:

- RES submits to us a list of research projects that are candidates for review because they have reached sufficient maturity that meaningful technical review can be conducted.
- We select a maximum of four projects for detailed review in the fiscal year.
- A panel of three ACRS members is established to assess the quality of each research project.
- The panel follows the guidance developed by the ACRS full Committee in conducting the technical review. This guidance is discussed further below.
- Each panel assesses the quality of the assigned research project and presents an oral and a written report to the ACRS full Committee for review. This review is to ensure uniformity in the evaluations by the various panels.
- The Committee revises these reports and provides them promptly to the cognizant research manager, as appropriate.
- The Committee submits an annual summary report to the RES Director.

The definition of quality research we have adopted includes two major characteristics:

- Results meet the objectives
- Documentation of research results and methods is adequate

The first of these major characteristics is weighted 75% in the scoring of the work. The documentation characteristic is weighted 25%. The measures and associated weights within the first characteristic are:

- Justification of major assumptions (12%)
- Soundness of technical approach and results (52%)
- Uncertainties and sensitivities addressed (11%)

The measures and weights within the general category of documentation are:

- Clarity of presentation (16%)
- Identification of major assumptions (9%)

These measures and associated weights for assessing the quality of research projects were defined by the ACRS full Committee and are addressed explicitly in the reports of the review panels. Scoring is based on a 10-point scale. A score of five is assigned to sound, professional performance of research. Exceptional performance is required to raise scores above this standard. Identifiable deficiencies must be cited to justify lower scores.

In our FY 2004 trial review, we assessed the quality of the following research projects:

- Effects of chemical reactions on head loss in debris beds that may block sump screens

- Experimental studies of loss-of-coolant accident generated debris accumulation and head loss on sump screens

- Improvements to the MACCS computer code, plume model adequacy

We submitted a summary report of our review of these research projects to the RES Director on November 18, 2004.

During FY 2005, we are assessing the quality of the research projects associated with:

- Standardized Plant Analysis Risk (SPAR) model development program

- Thermal-hydraulic experiments at the Pennsylvania State University

- Steam generator tube integrity research being performed at the Argonne National Laboratory

A fourth research project on reactor containment performance being conducted at Sandia National Laboratories will be evaluated later in the year, once a particularly pivotal report on the research becomes available. We plan to submit a summary report on our quality review of three research projects to the RES Director in the fall of 2005.

Sincerely,

/RA/

Graham B. Wallis
Chairman

References:

1. U.S. Nuclear Regulatory Commission, "Staff Requirements Memorandum (SRM), April 7, 2005 Meeting with the Advisory Committee on Reactor Safeguards (ACRS)," April 25, 2005.

2. Letter dated November 18, 2004, from Mario V. Bonaca, Chairman, ACRS, to Carl J. Paperiello, Director, Office of Nuclear Regulatory Research, NRC, Subject: ACRS Assessment of the Quality of Selected NRC Research Projects.

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

November 21, 2005

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

SUBJECT: DRAFT NRC DIGITAL SYSTEM RESEARCH PLAN FOR FY 2005 - FY 2009

Dear Mr. Reyes:

During the 527th meeting of the Advisory Committee on Reactor Safeguards, November 3-4, 2005, we met with representatives of the NRC staff to discuss the draft NRC Digital System Research Plan for FY 2005 - FY 2009. Our Subcommittee on Digital Instrumentation and Control Systems discussed the details of the research plan during meetings on June 14-15 and October 20-21, 2005. We also had the benefit of the documents referenced.

CONCLUSION

The Digital System Research Plan for FY 2005 - FY 2009 is well directed toward meeting agency needs. The plan can be further refined by considering the following recommendations.

RECOMMENDATIONS

1. The plan should include a research project to develop an inventory and classification, e.g., by function, of the various types of digital systems that are used and are likely to be used in nuclear power plants in the future.

2. The research plan should include a more detailed identification of current and future regulatory needs and possible benefits of the planned research to the regulatory system.

3. The plan should acknowledge the existence of two different aspects of software safety. The overall thrust of the proposed research is "software-centric." The "system-centric" aspect should receive more consideration than it is currently given.

4. Research in Section 3.6, Advanced Nuclear Power Plant Digital Systems, should be given higher priority.

BACKGROUND

Analog instrumentation and control systems in nuclear power plants are becoming obsolete and replacement parts are difficult to obtain. Licensees are replacing these systems with digital systems that are more flexible and have the potential to increase reliability and improve operational performance. Digital technology, however, brings a number of challenges. It can introduce new failure modes to the system, the rapid pace of change in digital technology requires the agency to update its knowledge base frequently, and new methods and acceptance criteria are needed to assess the safety and security of the systems.

The Office of Nuclear Regulatory Research has developed a plan for digital instrumentation and control systems research for Fiscal Years FY 2005 - FY 2009. This plan updates the previous plan for Fiscal Years FY 2000 - FY 2004. The plan has been reviewed by the Office of Nuclear Reactor Regulation, the Office of Nuclear Material Safety and Safeguards, and the Office of Nuclear Security and Incident Response.

DISCUSSION

The draft plan divides the research into six areas:

- System aspects of digital technology
- Software quality assurance
- Risk assessment of digital systems
- Security aspects of digital systems
- Emerging digital technology and applications
- Advanced nuclear power plant digital systems

The proposed research areas are comprehensive.

The applicability of the methods being investigated can vary greatly across the spectrum of possible systems. There is, therefore, a need for an inventory and classification, e.g., by function, of the various types of digital systems that are used or likely to be used in nuclear power plants in the future. Such a classification, along with a concurrent examination of the failures that have occurred in digital systems, should provide information on what types of tools may be best suited for different assessments. This classification could be the key to understanding the limitations of current methods of assessment and to guiding future efforts. For example, the analytical tools required to evaluate the performance of systems with simple actuation software are expected to be simpler than those required to evaluate systems with feedback and control software.

The plan discusses the shortcomings of the current regulations and the potential improvements that the proposed research is expected to produce. The plan would benefit by better identifying regulatory needs and anticipated benefits across all research areas. During our meetings, it was evident that the staff had thought through most of these issues, but its thinking was not well documented in the plan. Such documentation should be included.

As stated in the additional comments to our June 9, 2004, letter, the literature on digital software indicates that there are two main approaches to software reliability. The first approach views "failure" as a property of the software itself, just as the failure modes of hardware are considered properties of the components. This first approach is "software-centric." The second approach is "system-centric," in that the software is considered part of the system and the focus is on system failures.

Although the staff is aware of the two approaches to digital system reliability, the plan appears to be heavily focused on the software-centric view. For example, one objective of the research project described in Section 3.3.3, "Investigation of Digital System Characteristics Important to Risk," is said to be the calculation of the risk-importance of generic digital systems. This project seems to focus on the software more than the overall system. Although such a calculation may be meaningful for software in actuation systems such as the reactor protection system, it is unclear whether this can be done in more complex cases. Similarly, the term "digital system reliability" is used repeatedly in Section 3.3.4, "Investigation of Digital System Reliability Assessment Methods." A system-centric analysis focuses on the reliability of the broader system, not just the digital part. Such an approach to reliability should receive more consideration in the plan. The digital system classification in Recommendation 1 will assist the staff in determining when each approach is appropriate.

The research plan includes a program to investigate advanced nuclear power plant digital systems (Section 3.6), but this work has not begun. Due to the rapidly increasing interest in new reactors and the anticipated regulatory needs, this research should be given higher priority than it currently has.

In conclusion, we found the Digital System Research Plan for FY 2005 - FY 2009 to be well developed. The planned research programs should provide important inputs to the regulatory process. We look forward to continuing discussions with the staff on these programs as work progresses.

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References:

1. Memorandum from Michelle G. Evans, Chief, Engineering Research Applications Branch, Division of Engineering Technology, Office of Nuclear Regulatory Research, to John T. Larkins, Executive Director, Advisory Committee on Reactor Safeguards, "Transmittal of Material to Support the November 3 and 4, 2005, ACRS Meeting," September 29, 2005. (Pre-decisional).
2. Letter dated June 9, 2004 from Mario V. Bonaca, Chairman, Advisory Committee on Reactor Safeguards, to Luis A. Reyes, Executive Director for Operations, Nuclear Regulatory Commission, Subject: Digital Instrumentation and Control Research Program.

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

September 20, 2005

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

SUBJECT: PROPOSED REVISION 4 TO REGULATORY GUIDE 1.82, "WATER SOURCES FOR LONG-TERM RECIRCULATION COOLING FOLLOWING A LOSS-OF-COOLANT ACCIDENT"

Dear Mr. Reyes:

During the 525th meeting of the Advisory Committee on Reactor Safeguards, September 8-10, 2005, we reviewed the proposed Revision 4 to Regulatory Guide (RG) 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," and the supporting Standard Review Plan (SRP) Section 6.2.2, "Containment Heat Removal Systems." The review focused mainly on the issue of granting containment overpressure credit for calculation of net positive suction head (NPSH) for emergency core cooling and containment heat removal system pumps. During our review, we had the benefit of presentations by and discussion with representatives of the NRC staff and members of the public. We also had the benefit of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

1. Revision 4 to RG 1.82 should not be issued for public comment at this time and should be revised to improve clarity and reflect the following recommendation.

2. Containment overpressure credit to ensure sufficient NPSH for emergency core cooling and heat removal system pumps should only be selectively granted.

DISCUSSION

One purpose of the proposed Revision 4 to RG 1.82 is to make it consistent with current regulatory practice for crediting containment accident pressure in calculating available NPSH for boiling water reactor (BWR) and pressurized water reactor (PWR) systems. As a part of this effort, SRP Section 6.2.2 would also be revised to reference RG 1.82 rather than RG 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps." RG 1.1 would be designated as applicable only to those plants for which it was used as the basis for the original license.

RG 1.82 was first issued in 1974 to provide guidance on the design of PWR sumps which serve as a source of water during the recirculation core cooling phase of postulated design-basis loss-of-coolant accidents (LOCAs). Three revisions to RG 1.82 have been issued, one in November 1985, another in May 1996, and the most recent in November 2003. These revisions have addressed issues associated with containment emergency sump performance, particularly debris blockage on the emergency core cooling system suction strainers and granting credit for containment overpressure in determining NPSH available for the emergency core cooling and containment heat removal pumps.

Even though containment overpressure credit had been granted on an ad hoc basis before RG 1.1 was issued in 1974, Revision 3 to RG 1.82 issued in November 2003 was the first version to provide explicit guidance for granting limited use of containment accident pressure for calculating available NPSH. This guidance conflicts with the original guidance in RG 1.1, still in effect, which states that no such credit should be used. Not granting credit preserves the independence of the performance of the ECCS and containment systems.

The proposed Revision 4 to RG 1.82 includes provisions that permit licensees to use either a conservative deterministic approach or a best estimate with uncertainty analysis to establish the amount of containment overpressure to be credited.

We previously stated our position on granting containment overpressure credit in our December 12, 1997 letter (i.e., "selectively granting credit for small amounts of overpressure for a few cases may be justified") and more recently in our letter dated September 30, 2003. In that letter we recommended issuing Revision 3 to RG 1.82. That RG included a provision to grant, only where necessary, some containment accident pressure credit for some operating reactors with the caveat that "this should be minimized to the extent possible."

The position that the overpressure should be conservatively calculated is the only explicit restriction on the use of overpressure credit given in the proposed revision of the RG. In addition, the guidance describing what factors to consider in conservatively calculating containment overpressure, in Sections 1.3.1 and 2.1.1 of the proposed RG is confusing.

We believe that additional restrictive guidance should be placed on the granting of overpressure credit. Before such credit can be granted, licensees should demonstrate that there are no practical alternative approaches that can eliminate the need for such credit. Such credit should be granted only for robust containments for which there are positive means for indication of containment integrity such as inerted and sub-atmospheric containments. The time intervals for which such credit is needed should be limited to a few hours, commensurate with the demonstrated capability of all associated equipment to perform its intended functions during this time period. The RG should be revised to include such restrictions before it is released for public comment.

Dr. William Shack did not participate in the Committee's deliberations regarding this matter.

Sincerely,

/RA/

Graham B. Wallis
Chairman

References:

1. Letter from Suzanne Black to John Larkins, "Proposed Revision to Regulatory Guide 1.82, Revision 3, "Water Sources for Long-term Recirculation Cooling Following a Loss-of-Coolant Accident (LOCA)", June 3, 2005
2. Letter from James E. Lyons to John Larkins, "Proposed Revision to Regulatory Guide 1.82, Revision 3, "Water Sources for Long-term Recirculation Cooling Following a Loss-of-Coolant Accident (LOCA)", September 6, 2005
3. Letter from David O'Brien to Mario Bonaca, "State of Vermont Request to Consider the Containment Overpressure Credit Policy", September 17, 2004
4. B. R. Hobbs, et. al., "Vermont Yankee Extended Power Uprate Feasibility Study", June 28, 2002
5. "Learning about Pump NPSH Margin", http://www.pumps.org/public/pump resources, February 28, 2005
6. T. Henshaw, "How Much NPSH Does Your Pump Really Require?", www.pump-zone.com, September 2001, page 42
7. P. Cooper, et. al., "Checking In,", www.pump-zone.com, January, 2002, p. 8
8. R. Lueneberg, Sulzer-Bingham Pumps Inc., "NPSH/Minimum Flow Study - Summary, F-97-10782(30P59)", May 1, 1998
9. L. Lukens, "MSIV As-Found LLRTs Show An Adverse Trend - Adverse Trend Common Cause Analysis", CR-VTY-2004-0918, May 5, 2004

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

November 18, 2005

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

SUBJECT: DRAFT FINAL GENERIC LETTER 2005-XX, "GRID RELIABILITY AND THE IMPACT ON PLANT RISK AND THE OPERABILITY OF OFFSITE POWER"

Dear Mr. Reyes:

During the 527th meeting of the Advisory Committee on Reactor Safeguards, November 3-5, 2005, we reviewed the draft final Generic Letter 2005-XX, "Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power." During our review, we had the benefit of the document referenced and discussions with representatives of the staff and the Nuclear Energy Institute.

RECOMMENDATION

Generic Letter 2005-XX, "Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power," should be issued.

BACKGROUND AND DISCUSSION

The blackout of much of the Northeastern United States and parts of Canada on August 14, 2003, highlighted the extent to which changed conditions in the electric utility infrastructure could affect the probability of a station blackout event at nuclear power plants (NPPs). During the August 14 2003, event, nine NPPs lost all offsite power for periods ranging from 1 hour to 6.5 hours. Emergency diesel generators at these plants started and operated to supply emergency power, as designed. Adequate core cooling was maintained at all plants. Nonetheless, this event was significant because of the number of plants affected and the duration of the power outage. The severity and duration of this event called into question the bases for determining the risk impacts to the fleet of NPPs due to grid reliability issues.

Concerns about the reliability of the Nation's electrical grid prompted the U.S. Congress to enact the Electricity Modernization Act of 2005, which was signed on August 8, 2005. This Law added Section 215 to the Federal Power Act (FPA). Section 215 requires the Federal Energy Regulatory Commission (FERC) to enact regulations to improve and enforce the reliability of the electric power transmission infrastructure. FERC is currently amending its regulations to implement the requirements of the amended FPA. Among the changes under the amended FPA, FERC is charged with approving enforceable reliability standards. The North American Electric Reliability Council (NERC) is currently developing these reliability standards. The establishment of a national Electric Reliability Organization and the implementation of enforceable grid reliability standards are expected to be completed by December 31, 2006. The NRC has entered into a Memorandum of Agreement with both FERC and NERC which allows the staff to observe and participate in this important ongoing work. The continued cooperation between the staff and FERC and NERC is important in achieving the objectives of enhanced grid reliability without duplication of effort or conflicting goals, rules, or strategies. This cooperation should continue until satisfactory resolution of the grid reliability issue is achieved.

Even though FERC is taking important steps to improve grid reliability, the staff is rightly concerned as to how licensees are operating their NPPs in compliance with the rules and technical specifications relevant to grid operability. The NRC staff has developed Generic Letter 2005-XX, "Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power," to obtain information needed to assess whether licensees are in compliance with technical specifications, license conditions, and regulations regarding the operability and reliability of offsite power sources. Specifically, the Generic Letter requests licensees to provide detailed information, under oath or affirmation, regarding the details of their compliance with the following regulations:

- 10 CFR 50.63 (Station Blackout Rule)
- 10 CFR 50.65 (Maintenance Rule)
- 10 CFR Part 50, Appendix A, General Design Criterion 17 (Electric Power Systems)
- Technical Specification 3.8.1 (Operability of Offsite Power Systems)

The questions posed in the Generic Letter are appropriate and the staff should issue the Generic Letter to the licensees. The staff may need to explore these same questions with licensees after the Electric Reliability Organization is established and functioning, and the electric reliability standards are approved and in full force and effect. Also, the staff should continue to interact with FERC and NERC on grid reliability issues. We would like to hear a briefing from the staff after it has evaluated the information submitted by the licensees in response to this Generic Letter.

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

<u>Reference</u>:
Memorandum from M. Mayfield, NRR, to J. Larkins, ACRS, dated October 6, 2005, Subject: Request for Review and Endorsement by the Advisory Committee on Reactor Safeguards (ACRS) Regarding the Proposed Generic Letter 2005-XX, "Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power," (ADAMS Accession No. ML052790683).

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

April 14, 2005

The Honorable Nils J. Diaz
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 2005-0001

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE LICENSE RENEWAL APPLICATION FOR THE JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2

Dear Chairman Diaz:

During the 521st meeting of the Advisory Committee on Reactor Safeguards (ACRS), April 7-8, 2005, we completed our review of the license renewal application for the Joseph M. Farley Nuclear Plant (FNP), Units 1 and 2, and the related final Safety Evaluation Report (SER) prepared by the NRC staff. Our Plant License Renewal Subcommittee also reviewed this matter during a meeting on November 3, 2004. During our review, we had the benefit of discussions with representatives of the NRC staff and Southern Nuclear Operating Company, Inc. (SNC). We also had the benefit of the documents referenced. This report fulfills the requirement of 10 CFR 54.25 that the ACRS review and report on all license renewal applications.

CONCLUSIONS AND RECOMMENDATIONS

1. The programs established and committed to by the applicant will provide reasonable assurance that FNP Units 1 and 2 can be operated in accordance with their current licensing basis for the period of extended operation without undue risk to the health and safety of the public.

2. The SNC application for renewal of the operating licenses for FNP Units 1 and 2 should be approved.

BACKGROUND AND DISCUSSION

FNP Units 1 and 2 are 2775 MW_{th}, three-loop Westinghouse pressurized water reactors housed in pre-stressed/post-tensioned dry containment buildings. SNC requested renewal of the units' operating licenses for 20 years beyond the current license terms, which expire on June 25, 2017, for Unit 1, and March 31, 2021, for Unit 2.

In the final SER, the staff documents its review of the license renewal application and other information submitted by SNC and obtained during the audits and inspections conducted at the plant site. The SER also includes commitments identified by the staff and agreed to by the applicant. 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; and the identification and assessment of time-limited aging analyses (TLAAs) requiring review.

The FNP application either demonstrates consistency with the Generic Aging Lessons Learned (GALL) Report, or documents deviations to the specified approaches in the GALL Report. The FNP application is the first to be evaluated using a new audit and review process intended to confirm consistency with the GALL Report, and the acceptability of deviations from that report. This approach, which requires more review activities at the site, has resulted in improved communications and more effective interactions between the applicant and the staff, and a significant reduction in requests for additional information. During our meeting, the staff presented a well-structured and effective overview of its reviews, audits, and inspections.

Several scoping issues that in previous applications resulted in significant disagreement between the staff and applicants were promptly resolved at FNP due to the clear interim staff guidance. Among these issues were fuse holders, equipment required to recover from station blackout, and fire protection equipment. The staff disagreed with SNC in some areas, such as the scoping criteria for spray interactions in low-energy lines. We agree with the resolution of these issues, and the staff and SNC should be commended for promptly resolving them.

The applicant performed a comprehensive aging management review of all SSCs within the scope of license renewal. In the application, SNC describes 22 aging management programs for license renewal including existing, enhanced, and new programs. We agree that these programs are adequate.

We reviewed plant-specific operating experience to assess how effectively the applicant has dealt with age-related degradation. In 1987, FNP Unit 2 experienced a throughwall leak in an unisolable portion of the emergency core cooling system piping. The leak was attributed to thermal cycling due to valve leakage. This event led to the issuance of NRC Bulletin 88-08, "Thermal Stresses in Piping Connected to the Reactor Coolant System." Since then, FNP has established accurate baseline cycle counts. For license renewal, the applicant developed a new fatigue monitoring program consistent with the GALL Report for monitoring fatigue of metal piping in components of the reactor coolant pressure boundary. The program will automatically monitor cycles using installed plant equipment.

As in previous reviews, we questioned the adequacy of opportunistic inspections of inaccessible buried piping and tanks, in lieu of periodic inspections at a plant-specific frequency, as specified in the GALL Report. The applicant has committed to enhancing its Buried Piping and Tank Inspection Program by performing an inspection within 10 years of entering the period of extended operation unless an opportunistic inspection has occurred within this 10-year period. This program enhancement is appropriate. The staff has also included this 10-year inspection as new generic guidance in the proposed revision to the GALL Report.

The applicant has also committed to perform an engineering evaluation before the 10th year of extended operation to determine whether sufficient inspections have been conducted to draw a conclusion regarding the ability of the coatings to protect underground piping and tanks from degradation. If not, a focused inspection will be conducted to allow a conclusion to be reached.

We agree with the staff that the applicant has identified and properly addressed systems and components requiring TLAAs. The staff has independently verified the applicant's calculations of reactor vessel upper shelf energy and has confirmed that the limiting beltline materials at 60 years satisfy the acceptance criteria. We also note that the most limiting beltline materials satisfy the pressurized thermal shock criterion with ample margin based on both the applicant's and the staff's calculations.

When environmental factors are applied and projected to 60 years, cumulative usage factors (CUFs) for some piping locations may exceed a CUF of 1.0. For these locations, the applicant has committed to take corrective action prior to the period of extended operation. This action might include a more refined analysis, repair, replacement, and/or an inspection program approved by the NRC. We are satisfied with this commitment.

The licensee is improving FNP Units 1 and 2. New steam generators with Alloy 690 tubing, quatrefoil support plates, and full depth rolls were installed in both units in 2000 and 2001. Although control rod drive mechanism (CRDM) inspections have not identified leaks, both units are susceptible to CRDM cracking due to high head temperatures. Therefore, reactor vessel heads are being replaced with new heads that contain Alloy 690 penetrations without thermal sleeves. The licensee has also replaced the cooling towers and installed a dry cask storage facility.

Recent inspections of the reactor pressure vessel lower head penetrations of both units revealed no degradation. Bare metal visual inspections of Alloy 600/182/82 pressure boundary locations were also performed and did not reveal any degradation.

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 licenses for Farley Units 1 and 2. The programs established and committed to by SNC provide reasonable assurance that the plant 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 SNC application for renewal of the operating licenses for FNP Units 1 and 2 should be approved.

Sincerely,

/RA/

Graham B. Wallis
Chairman

References

3. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the License Renewal of the Joseph M. Farley Nuclear Plant, Units 1 and 2," March 2005
4. Southern Nuclear Operating Company, Inc. "Joseph M. Farley Nuclear Plant License Renewal Application," September 2003
5. U.S. Nuclear Regulatory Commission, "Draft Safety Evaluation Report Related to the License Renewal of the Joseph M. Farley Nuclear Plant, Units 1 and 2," October 2004
6. U.S. Nuclear Regulatory Commission Inspection Report 50-348/2004-007, 50-364/2004-007, Scoping and Screening, June 22, 2004
7. Information Systems Laboratories, Inc., "Audit and Review Report for Plant Aging Management Reviews and Programs, Joseph M. Farley Nuclear Plant, Units 1 & 2," September 10, 2004

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

ACRSR-2124

May 13, 2005

The Honorable Nils J. Diaz
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 2005-0001

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE LICENSE RENEWAL APPLICATION FOR ARKANSAS NUCLEAR ONE, UNIT 2

Dear Chairman Diaz:

During the 522nd meeting of the Advisory Committee on Reactor Safeguards, May 5-6, 2005, we completed our review of the license renewal application for Arkansas Nuclear One, Unit 2 (ANO-2), and the associated final Safety Evaluation Report (SER) prepared by the NRC staff. Our Plant License Renewal Subcommittee also reviewed this matter during a meeting on December 1, 2004. During our review, we had the benefit of discussions with representatives of the NRC staff and Entergy Operations, Inc. (Entergy). 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

1. The programs established and committed to by the applicant to manage age-related degradation provide reasonable assurance that ANO-2 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.

2. The Entergy application for renewal of the operating license for ANO-2 should be approved.

BACKGROUND AND DISCUSSION

ANO-2 is a Combustion Engineering pressurized water reactor rated at 3026 MWt, enclosed in a large dry containment building. The current power rating includes a 7.5% power uprate implemented in 2002. The ANO-2 steam generators were replaced with new Westinghouse Delta steam generators with Alloy 690 tubing in conjunction with this power uprate.

Entergy requested renewal of the ANO-2 operating license for 20 years beyond the current license term, which expires on July 17, 2018. In the final SER, the staff documents its review of the license renewal application and other information submitted by Entergy and obtained during the audits and inspections 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 plausible aging mechanisms associated with passive, long-lived components; the adequacy of the applicant's aging management programs; and the identification and assessment of time-limited aging analyses (TLAAs).

The ANO-2 application demonstrates consistency with the Generic Aging Lessons Learned (GALL) Report or documents deviations from the approaches specified in that report. The ANO-2 application is the second one evaluated by the staff using the new audit and review process developed to confirm consistency with, and the acceptability of deviations from, the GALL Report. The new process requires that more review activities be conducted at the site. As in the first application, this approach has resulted in more effective interactions between the applicant and the staff and has significantly reduced requests for additional information (RAIs).

During its review, the staff identified several components that the applicant should have included in the scope of license renewal but did not. The applicant subsequently brought them into scope. The staff concluded that these omissions were the result of minor oversights or different interpretations of the scoping methodology, and not an indication of process problems. The staff also concluded that the applicant's scoping and screening processes have successfully identified SSCs within the scope of license renewal and subject to an aging management review. We agree with these conclusions.

The applicant performed a comprehensive aging management review of all SSCs within the scope of license renewal. In the application, Entergy describes 34 aging management programs for license renewal, including existing, enhanced, and new programs. We agree with the staff's conclusion that these programs are adequate.

Implementation is key to effective aging management programs. Although the applicant's Structures Monitoring-Masonry Wall Program is consistent with the GALL Report, the staff's audit of this program revealed that the initial baseline examinations were not documented properly, the first 5-year reexamination was not performed, and qualifications for personnel responsible for walkdowns were not established. The Annual Assessment Letter for ANO, Units 1 and 2, dated March 3, 2004, had already identified a substantive cross-cutting issue concerning problem identification and resolution. Based on the Annual Assessment Letter dated March 2, 2005, the weaknesses in the ANO-2 Problem Identification and Resolution Program appear to have been corrected. Maintaining an effective problem identification and resolution program is critical to the success of the aging management programs.

As in previous reviews, we questioned the adequacy of relying on opportunistic inspections of inaccessible buried piping and tanks, in lieu of periodic inspections at a plant-specific frequency, as specified in the GALL Report. The applicant has committed to enhance its Buried Piping Inspection Program by performing an inspection within 10 years of entering the period of extended operation unless an opportunistic inspection has occurred within this 10-year period. This program enhancement is appropriate.

The applicant identified and reevaluated systems and components requiring TLAAs for 20 more years of operation. The applicant's analyses of reactor vessel embrittlement (upper shelf energy, pressurized thermal shock, and pressure-temperature limits), independently verified by the staff, demonstrate that the limiting beltline materials will satisfy the acceptance criteria at 48 effective full-power years (EFPYs). This value corresponds to a constant capacity factor of 80% for 60 years. We questioned the use of 48 EFPYs, rather than the 54 EFPYs used by other applicants to bound 60 years of operation. Given the current performance of the fleet, 54 EFPYs seems to be a more appropriate value for 60 years of operation. The staff independently verified that the upper shelf energy and pressurized thermal shock acceptance criteria would still be met at 54 EFPYs.

In 2000, nondestructive examinations revealed a number of leaks in pressurizer and hot-leg penetration nozzles. The applicant implemented repairs using the half-nozzle repair technique. The applicant evaluated the potential for existing flaws in the remaining pressurizer and hot-leg penetration welds to propagate into the pressurizer or hot leg. The applicant has performed a TLAA to bound the period of extended operation and has demonstrated that stress corrosion cracking will not cause existing flaws to propagate into the carbon steel or low-alloy steel base metal.

Since a shroud prevents a complete 360° bare metal visual inspection of some of the control rod drive mechanism (CRDM) penetrations, the applicant performed alternative eddy current and volumetric inspections. Although these inspections did not identify any cracking or leakage, ANO-2 is ranked as highly susceptible to CRDM cracking. The applicant has scheduled the procurement of a new reactor vessel head in 2006. Meanwhile, the applicant plans to modify the shroud to allow increased access for visual examinations.

We agree with the staff that no issues related to the matters described in 10 CFR 54.29(a)(1) and (a)(2) preclude renewal of the operating license for ANO-2. The programs established and committed to by Entergy provide reasonable assurance that ANO-2 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 Entergy application for renewal of the operating license for ANO-2 should be approved.

Sincerely,

/RA/

Graham B. Wallis
Chairman

References
3. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the License Renewal of the Arkansas Nuclear One, Unit 2," April 2005
4. Entergy Operations Inc., "License Renewal Application Arkansas Nuclear One - Unit 2," October 2003
5. U.S. Nuclear Regulatory Commission, "Draft Safety Evaluation Report Related to the License Renewal of the Arkansas Nuclear One, Unit 2," November 2004
6. U.S. Nuclear Regulatory Commission, "Arkansas Nuclear One, Unit 2 - NRC License Renewal Scoping and Screening Inspection Report 05000368/2004-06," April 19, 2004
7. Information Systems Laboratories, Inc., "Audit and Review Report for Plant Aging Management Reviews and Programs, Arkansas Nuclear One - Unit 2," July 29, 2004

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

June 9, 2005

Mr. Luis A. Reyes
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 2005-0001

SUBJECT: INTERIM REPORT ON THE SAFETY ASPECTS OF THE LICENSE RENEWAL APPLICATION FOR THE POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

Dear Mr. Reyes:

During the 523rd meeting of the Advisory Committee on Reactor Safeguards, June 1-3, 2005, we reviewed the license renewal application for the Point Beach Nuclear Plant (PBNP), Units 1 and 2, and the associated Safety Evaluation Report (SER) with open items prepared by the NRC staff. Our Plant License Renewal Subcommittee also reviewed this matter on May 31, 2005. During these reviews, we had the benefit of discussions with representatives of the NRC staff, including Region III personnel, and the Nuclear Management Company, LLC. We also had the benefit of the documents referenced.

We recognize that the license renewal rule does not include specific consideration of current operating performance. However, aspects of current performance may affect the development of license renewal programs and commitments as well as the effectiveness of the implemented programs.

The Confirmatory Action Letter (CAL) issued to the PBNP on April 21, 2004 will remain open until improvements are demonstrated in the areas of human performance, engineering design control, the engineering/operations interface, emergency preparedness, and the Corrective Action Program (CAP).

An adequate CAP is a key element in the successful implementation of the aging management programs critical to license renewal. A review of the events leading to the issuance of the CAL leads to the conclusion that the applicant's CAP has been in a degraded condition for a long time. The Region III staff stated that the problems are not in the design of the program but in its implementation. The inspections have also identified other weaknesses in the area of human performance. Errors in engineering calculations have been identified and are being corrected, but this work is not yet complete. These errors may have an impact on long-lived passive components.

It often takes a long time to successfully implement improvements in human performance, and we note that the current operating license for Unit 1 expires on October 5, 2010. The March 2, 2005 Annual Assessment Letter to the PBNP notes that some improvements in the human performance area have been observed. However, problems continue to be identified in the CAP, and the PBNP remains in the Multiple/Repetitive Degraded Cornerstone column of the Reactor Oversight Process Action Matrix. The resources needed to address the CAL compete with the effective development, tracking, and implementation of license renewal programs and commitments.

In support of its final SER, the staff normally audits and inspects only a fraction of the license renewal programs and commitments. In the case of the PBNP, the staff should take additional actions to increase confidence that the requirements of the license renewal rule have been met and that there is reasonable assurance that aging degradation can be adequately managed. These actions may include, for example, an expanded inspection of license renewal commitments and a focused review of the effectiveness of the CAP before the PBNP enters the period of extended operation. We would like to hear about such planned actions during our review of the final SER.

Sincerely,

/RA/

Graham B. Wallis
Chairman

References
1. Nuclear Management Company, LLC, "Application for Renewed Operating Licenses Point Beach Nuclear Plant Units 1 & 2," February 2004
2. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the License Renewal of the Point Beach Nuclear Plant, Units 1 and 2," May 2005
3. Letter from J. Caldwell, Regional Administrator, to G. Van Middlesworth, Site Vice President, Point Beach Nuclear Plant, Nuclear Management Company, LLC, "Confirmatory Action Letter," April 21, 2004
4. Letter from J. Caldwell, Regional Administrator, to D. Koehl, Site Vice President, Point Beach Nuclear Plant, Nuclear Management Company, LLC, "Annual Assessment Letter - Point Beach Nuclear Plant (Report 05000266/200501; 05000301/200501)," March 2, 2005
5. Letter from J. Dyer, Regional Administrator, to M. Warner, Site Vice President, Kewaunee and Point Beach Nuclear Plants, Nuclear Management Company, LLC, "Point Beach Special Inspection - NRC Inspection Report 50-266/01-17(DRS); 50-301/01-17(DRS), Preliminary Red Finding," April 3, 2002
and Preliminary Red Finding - Auxiliary Feedwater Orifice Plugging Issue; NRC Inspection Report 50-266/02-15(DRP); 50-301/02-15(DRP)," April 2, 2003

6. Letter from J. Caldwell, Regional Administrator, to A. Cayia, Site Vice President, Point Beach Nuclear Plant, Nuclear Management Company, LLC, "Point Beach Nuclear Plant, Units 1 and 2 Final Significance Determination for a Red Finding and Notice of Violation (NRC Inspection Report No. 50-266/02-15(DRP); 50-301/02-15(DRP))," December 11, 2003
7. Letter from G. Van Middlesworth, Site Vice President, Point Beach Nuclear Plant, Nuclear Management Company, LLC, to U.S. Nuclear Regulatory Commission Document Control Desk, "Commitments in Response to 95003 Supplemental Inspection," March 22, 2004
8. Pacific Northwest National Laboratory, "Audit and Review Report for Plant Aging Management Reviews and Programs, Point Beach Nuclear Plant Units 1 and 2," April 11, 2005
9. U.S. Nuclear Regulatory Commission, "Point Beach Nuclear Plant, Units 1 and 2 NRC License Renewal Scoping, Screening, and Aging Management Inspection Report 05000266/2005005 (DRS); 05000301/2005005 (DRS)," May 2, 2005

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

July 18, 2005

The Honorable Nils J. Diaz
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 2005-0001

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE LICENSE RENEWAL APPLICATION FOR THE DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2

Dear Chairman Diaz:

During the 524th meeting of the Advisory Committee on Reactor Safeguards, July 6-8, 2005, we completed our review of the license renewal application for the Donald C. Cook Nuclear Plant (CNP), Units 1 and 2, 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 February 9, 2005. During these reviews, we had the benefit of discussions with representatives of the NRC staff and Indiana Michigan Power Company, the applicant. 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

1. The programs committed to and established by the applicant to manage age-related degradation provide reasonable assurance that CNP Units 1 and 2 can be operated in accordance with their current licensing basis for the period of extended operation without undue risk to the health and safety of the public.

2. The Indiana Michigan Power Company's application for renewal of the operating licenses for CNP Units 1 and 2 should be approved.

BACKGROUND AND DISCUSSION

CNP Units 1 and 2 are Westinghouse pressurized water reactors with ice condenser containment buildings. Licensed power output is 3304 MWt for Unit 1 and 3468 MWt for Unit 2. The Indiana Michigan Power Company requested renewal of the operating licenses of Units 1 and 2 for 20 years beyond their current license terms, which expire on October 25, 2014 and December 23, 2017, respectively.

In the final SER, the staff documented its review of the license renewal application and other information submitted by the applicant and obtained during the staff's audits and inspections 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 plausible aging mechanisms associated with passive, long-lived components; the adequacy of the applicant's aging management programs; and the identification and assessment of time-limited aging analyses (TLAAs).

The CNP application demonstrates consistency with, or justifies deviations from, the approaches specified in the Generic Aging Lessons Learned Report.

During its review, the staff identified several components that should have been included in the scope of license renewal. The applicant brought them into scope. With these inclusions, the staff concluded that the applicant's scoping and screening processes have successfully identified the SSCs within the scope of license renewal and subject to an aging management review. We agree.

The applicant performed a comprehensive aging management review of all SSCs within the scope of license renewal. The application contains descriptions of 46 aging management programs for license renewal, including existing, enhanced, and new programs. We agree with the staff's conclusion that these programs are adequate and consistent with accepted practices for aging management.

To be effective, the aging management programs need to be appropriately implemented. During the aging management program inspections, the staff found that walkdowns performed as part of the System Walkdown Program were not conducted quarterly as stated in the license renewal application. Also, the applicant noted that it had not evaluated two coupons from the Boral Surveillance Program. This program monitors the performance of absorber materials in the spent fuel pool by periodically measuring the physical and chemical properties of coupon samples that receive a higher radiation dose than the functional boral panels. The applicant has implemented corrective actions to ensure that the commitments will not be missed in the future.

The applicant identified and reevaluated systems and components requiring TLAAs for 20 more years of operation. Analyses of reactor vessel neutron embrittlement (upper shelf energy, pressurized thermal shock screening criteria, and pressure-temperature limits) performed by the applicant and independently verified by the staff demonstrate that the limiting reactor vessel beltline materials will satisfy the acceptance criteria for the period of extended operation.

The applicant showed that the current fatigue analysis of the ice condenser lattice frame, which conservatively assumes 400 operating basis earthquakes, bounds 60 years of operation. This analysis also bounds the effects of loads due to temperature fluctuations. The Structures Monitoring Program manages aging of this structure. Operating experience indicates that the lattice frame is not subject to significant age-related degradation.

The final SER documents the closure of confirmatory items addressing fatigue of Class 1 components. These confirmatory items were closed by the applicant's commitments to perform additional actions to address fatigue of the auxiliary spray line piping and environmentally assisted fatigue of the pressurizer surge line, safety injection nozzles, charging nozzles, and residual heat removal line. These commitments will ensure that the effects of fatigue are appropriately managed.

Reactor vessel head inspections identified flaw indications in two nozzle penetrations of Unit 2. Weld repairs were performed. No leakage was identified in the reactor vessel head penetrations of Unit 1. Both reactor vessel heads are scheduled for replacement by 2007. Inspections of bottom-mounted instrumentation nozzles in both units have not identified any leakage, and the applicant has committed to follow the recommendations the industry is developing for aging management of Alloy 600 components.

No issues related to the matters described in 10 CFR 54.29(a)(1) and (a)(2) preclude renewal of the operating licenses for CNP Units 1 and 2. The programs committed to and established by the applicant provide reasonable assurance that CNP Units 1 and 2 can be operated in accordance with their current licensing basis for the period of extended operation without undue risk to the health and safety of the public. The application for renewal of the operating licenses for CNP Units 1 and 2 should be approved.

Sincerely

/RA/

Graham B. Wallis
Chairman

References

1. Indiana Michigan Power Company, "Donald C. Cook Nuclear Plant License Renewal Application," October 2003
2. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the License Renewal of the Donald C. Cook Nuclear Plant, Units 1 and 2," May 2005
3. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report with Open Items Related to the License Renewal of the Donald C. Cook Nuclear Plant, Units 1 and 2," December 2004
4. U.S. Nuclear Regulatory Commission, "Donald C. Cook Nuclear Power Plant, Units 1 and 2 NRC License Renewal Scoping/Screening Inspection Report 05000315/2004003 (DRS); 05000316/2004003 (DRS)," June 22, 2004
5. U.S. Nuclear Regulatory Commission, "D.C. Cook Nuclear Power Plant, Units 1 and 2 NRC Aging Management Program Inspection Report No. 05000315/2004013 (DRS); 05000316/2004013 (DRS)," January 10, 2005
6. Information Systems Laboratories, Inc., "Audit and Review Report for Plant Aging Management Reviews and Programs, Donald C. Cook Nuclear Plant, Units 1 & 2," September 22, 2004

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

September 22, 2005

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

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE LICENSE RENEWAL APPLICATIONS FOR THE MILLSTONE POWER STATION, UNITS 2 AND 3

Dear Chairman Diaz:

During the 525th meeting of the Advisory Committee on Reactor Safeguards (ACRS), September 8-10, 2005, we completed our review of the license renewal applications for the Millstone Power Station (MPS), Units 2 and 3 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 April 6, 2005. During these reviews, we had the benefit of discussions with the staff, Dominion Nuclear Connecticut, Inc. (DNC), and a member of the public representing the Connecticut Coalition Against Millstone. We also had the benefit of the documents referenced. This report fulfills the requirements of 10 CFR 54.25, which requires that the ACRS review and report on all license renewal applications.

CONCLUSION AND RECOMMENDATION

1. The programs committed to and established by the applicant to manage age-related degradation provide reasonable assurance that MPS, Units 2 and 3 can be operated in accordance with their current licensing basis for the period of extended operation without undue risk to the health and safety of the public.

2. DNC's applications for renewal of the operating licenses for MPS, Units 2 and 3 should be approved.

BACKGROUND AND DISCUSSION

Millstone Power Station consists of three nuclear units on a 500-acre site on the north shore of Long Island Sound in the town of Waterford, Connecticut. Each of the three Millstone units was supplied by a different nuclear steam supply system vendor. Unit 1, a Mark 1 boiling water reactor which was shut down in the late 1990s, is not the subject of the license renewal applications being considered here. Unit 2 is a 2700 MWt (895 MWe) 4-loop (two steam generators) Combustion Engineering pressurized water reactor (PWR). Unit 3 is a 3411 MWt (1195 MWe) 4-loop Westinghouse PWR. The applicant has requested renewal of the current operating licenses for Units 2 and 3 for an additional 20 years beyond their current terms, which expire on July 31, 2015, and November 25, 2025, respectively.

Those long-lived passive structures, systems, and components (SSCs) from Unit 1 that service Units 2 and 3 fall within the scope of this license renewal. Although DNC submitted separate license renewal applications for Unit 2 and Unit 3, the staff consolidated its SER to address both applications. Since the applicant will apply identical aging management programs (AMPs) to both units, the staff's consolidation of the SER is appropriate.

In the final SER, the staff documented its review of the DNC's license renewal applications and other information submitted by the applicant or obtained during the staff's audits and inspections at the plant site. The staff reviewed the completeness of the applicant's identification of SSCs that are within the scope of license renewal; the integrated plant assessment process; the applicant's identification of plausible aging mechanisms associated with passive, long-lived components; the adequacy of the applicant's aging management programs; and the identification and assessment of time-limited aging analyses (TLAAs).

The DNC applications demonstrate consistency with, or justify deviations from, the approaches specified in the Generic Aging Lessons Learned Report.

In its draft SER the staff identified a number of issues requiring further definition, analysis, or modification by the applicant to satisfy the requirements of the license renewal rule. Among these issues were the following:

> The staff questioned the applicant's definition of the "first equivalent anchor point" for determining the endpoint of nonsafety-related piping attached to safety-related systems to be included within the scope of the rule. The applicant resolved this issue by changing the definition of the "first equivalent anchor point" to be consistent with the current licensing basis.
>
> The staff questioned the applicant's neglect of effects other than thermal cycling, such as vibration, that could lead to age-related loss of preload of bolting. The applicant modified its Bolting Integrity Program to reflect such aging effects.
>
> The staff questioned the exclusion of the reactor vessel flange leak detection lines in Units 2 and 3 from the scope of license renewal. The applicant initially argued that the break flow through a failed leak detection line would be limited by a restriction in the reactor vessel flange geometry to a flow less than the makeup capability of the chemical and volume control system. However, the applicant finally decided to include the reactor vessel flange leak detection lines within the scope of aging management, satisfying the staff's concern.
>
> The staff questioned the adequacy of the leak-before-break (LBB) analyses for Units 2 and 3 for the period of extended operation. The applicant submitted additional information on the methods and assumptions used to update these analyses for the period of extended operation. The current LBB analyses are for the reactor coolant system loop piping and components, the pressurizer surge line, and portions of the safety injection and shutdown cooling lines of Unit 2, and for the reactor coolant system loop piping and components of Unit 3. The analyzed systems and components were constructed of carbon and low-alloy steel, stainless steel [including cast austenitic stainless steel (CASS)], and nickel-based alloys. TLAAs were performed that account for fatigue crack growth, the thermal aging of CASS, and the corrosion of nickel-based alloys. DNC demonstrated that the analyses for fatigue crack growth and thermal aging of CASS, assuming fully aged materials, are adequate for the period of extended operation. The corrosion of nickel-based alloys will be managed by the use of the Inservice Inspection Program. In addition, DNC has committed to follow the industry recommendations regarding the aging effects and appropriate aging management of nickel-based alloys for Units 2 and 3 and to submit an aging management program at least 24 months prior to entering the period of extended operation. These commitments are documented in the SER.

Analyses of reactor vessel neutron embrittlement (upper shelf energy, pressurized thermal shock screening criterion, and pressure-temperature limits) performed by the applicant and independently verified by the staff demonstrate that the limiting reactor vessel beltline welds and plate materials will satisfy the acceptance criteria for the period of extended operation.

Both the applicant and the staff chose to use a conservative lifetime capacity factor of 90 percent for determining neutron fluence. We agree.

The staff requested confirmatory analyses or other technically justifiable responses to six confirmatory issues. DNC has supplied information regarding these confirmatory items and the staff has determined that the applicant's responses are satisfactory to close these confirmatory items.

We agree with the resolution of all open items identified in the draft SER. DNC has made appropriate commitments to carry out the tasks identified by the staff and agreed to by DNC to satisfy outstanding issues related to these applications. The staff has included appropriate license conditions in the SER to satisfy remaining documentation issues and action items.

DNC's applications for renewal of the licenses for MPS, Units 2 and 3 are of high quality and DNC's responses to the staff's requests for additional information are thorough, timely, and complete. The staff's evaluation is technically comprehensive and well documented in the SER. The inspections and audits performed by the NRC staff for evaluating the applicant's proposed and existing programs and analyses are effective. They reduce the amount of paperwork and staff and applicant time needed to prepare and respond to written requests for additional information.

No issues related to the matters described in 10 CFR 54.29(a)(1) and (a)(2) preclude renewal of the operating licenses for MPS, Units 2 and 3. The programs committed to and established by the applicant provide reasonable assurance that MPS, Units 2 and 3 can be operated in accordance with their current licensing basis for the period of extended operation without undue risk to the health and safety of the public. The applications for renewal of the operating licenses for MPS, Units 2 and 3 should be approved.

Drs. Mario Bonaca and George Apostolakis did not participate in the Committee's deliberations regarding this matter.

Sincerely,

/RA/

Graham B. Wallis
Chairman

References:
1. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the License Renewal of the Millstone Power Station, Units 2 and 3," August 2005
2. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report with Open Items Related to the License Renewal of the Millstone Power Station, Units 2 and 3," February 2005
3. Dominion Nuclear Connecticut, Inc., "Millstone Power Station Unit 2 Application for Renewed Operating License Technical and Administrative Information," January 2004
4. Dominion Nuclear Connecticut, Inc., "Millstone Power Station Unit 3 Application for Renewed Operating License Technical and Administrative Information," January 2004
5. U.S. Nuclear Regulatory Commission, "Millstone Power Station Unit 2 and Unit 3 - License Renewal Application Inspection Report Nos. 05000336/2004009, 05000423/2004009," December 3, 2004
6. U.S. Nuclear Regulatory Commission, "Millstone Power Station Unit 2 and Unit 3 - License Renewal Application Inspection Report Nos. 05000336/2004010, 05000423/2004010," December 3, 2004

7. Information Systems Laboratories, Inc., "Audit and Review Report for Plant Aging Management Reviews and Programs, Millstone Power Station - Units 2 & 3," February 2, 2005
8. Letter to Graham Wallis, Chairman, Advisory Committee on Reactor Safeguards, from Nancy Burton, Connecticut Coalition Against Millstone, Subject: Millstone Nuclear Power Station, September 7, 2005
9. Letter to the Advisory Committee on Reactor Safeguards from Nancy Burton, Connecticut Coalition Against Millstone, Subject: Millstone Nuclear Power Station Application for License Renewal, April 5, 2005
10. Letter to Paul G. Kroh, Chief Inspector, Region I, U.S. Nuclear Regulatory Commission, from Nancy Burton, Connecticut Coalition Against Millstone, Subject: Millstone Nuclear Power Station, April 1, 2005

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

ACRSR-2159

October 19, 2005

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

SUBJECT: INTERIM REPORT ON THE SAFETY ASPECTS OF THE LICENSE RENEWAL APPLICATION FOR THE BROWNS FERRY NUCLEAR PLANT UNITS 1, 2, AND 3

Dear Mr. Reyes:

During the 526th meeting of the Advisory Committee on Reactor Safeguards, October 6-7, 2005, we reviewed the license renewal application (LRA) for the Browns Ferry Nuclear Plant (BFN) Units 1, 2, and 3, and the associated Safety Evaluation Report (SER) with open items prepared by the NRC staff. On August 23, 2005 we visited the Browns Ferry site and reviewed activities under way for license renewal, power uprate, and restart. Our Plant Operations and Plant License Renewal Subcommittees also reviewed these matters on September 21, 2005. Our Plant License Renewal Subcommittee reviewed the LRA and the staff's associated SER on October 5, 2005. During these reviews, we had the benefit of discussions with representatives of the NRC staff, including Region II personnel, and the Tennessee Valley Authority (TVA). We also had the benefit of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

1. We agree with the open items identified by the staff and concur with the staff's interim evaluation of the LRA for BFN Units 2 and 3.

2. The plant-specific operating experience for BFN Unit 1, by itself, does not fully meet the intent of the license renewal rule. In addition, many components have been subjected to an extended period of layup that is unusual in plant experience. The SER documents in several places how the applicant plans to compensate for the lack of plant-specific operating experience. The final SER should include a cohesive discussion of the applicability of BFN Units 2 and 3 operating experience to Unit 1 and the compensating actions taken where such experience is not sufficient.

3. The final SER should include a description of the attributes of the new Periodic Inspection Program for BFN Unit 1 components that will not be replaced before restart.

4. If the extended power uprate (EPU) is implemented, the staff should require that TVA evaluate Units 1, 2, and 3 operating experience at the uprated power level and incorporate lessons learned into their aging management programs prior to entering the period of extended operation.

BACKGROUND

TVA has requested renewal of the BFN Units 1, 2, and 3 operating licenses for 20 years beyond their current license terms, which expire on December 10, 2013, June 28, 2014, and July 2, 2016, respectively.

All three BFN units are General Electric boiling water reactors (BWR 4) with Mark I containments. Units 1 and 2 commenced operation in 1973 and 1974 respectively, and were both shut down after the March 22, 1975 fire in Unit 1. Both units were returned to service in 1976, the same year Unit 3 commenced operation. All three units operated until 1985, when they were shut down to address management, technical, and regulatory issues. Units 2 and 3 were restarted in 1991 and 1995 respectively, and have been in operation since then. Unit 1 has been shut down since 1985. The approximate durations of power operation of the three units are 10 years for Unit 1, 23 years for Unit 2, and 18 years for Unit 3. TVA plans to restart Unit 1 in 2007. As part of an extensive restart program for Unit 1, components that have been in different states of "layup" for the past 20 years will be either replaced or requalified. Layup provides a controlled environment intended to limit corrosion of plant components.

In addition to license renewal, TVA is requesting power uprates for the three units. The original power rating of the three units was 3293 MWt. Units 2 and 3 have been uprated to their current power level of 3458 MWt. TVA is implementing physical modifications in Unit 1 to support an EPU of 20%, which TVA plans to implement at restart. This will raise the Unit 1 power level to 3952 MWt. Units 2 and 3 will then be uprated to the same power level as Unit 1. However, the license renewal application is based on the current power levels, since the planned EPUs are separate licensing actions which have not yet been approved.

DISCUSSION

The multiple licensing actions involved in the implementation of TVA's strategy for BFN make this LRA more complex than usual. These actions include license renewal and EPU for all three units as well as restarting Unit 1 as the lead plant at the highest power level after having been idle for 22 years. The SER discusses the work done by the applicant on these multiple licensing actions; however, its focus is on the LRA.

We agree with the staff's interim evaluation of the LRA for BFN Units 2 and 3. We have some issues with those portions of the LRA and the SER for Unit 1.

The intent of the license renewal rule is that plants applying for license renewal accumulate substantial operating experience to disclose any plant-specific concerns with regard to age-related degradation and to ensure that the aging management programs instituted to manage aging during the license renewal period will adequately address such concerns. The Statement of Considerations of the rule clearly sets forth this intent and states that 20 years is an appropriate operating period to support license renewal. Exceptions to the 20-year time limit for filing license renewal applications have been granted, but not to the intent of the rule that substantial plant-specific operating experience be available. BFN Unit 1 does not have the substantial operating experience intended by the rule. Therefore, the applicant has relied on the BFN Units 2 and 3 operating experience, plus generic operating experience from plants of similar designs, to provide the operating experience intended by the rule.

By the time BFN Unit 1 enters the period of extended operation, the plant will have experienced 10 years of early operation, 22 years in layup conditions, major equipment replacement and requalifications to support restart, a planned EPU to 3952 MWt, and six years of operation at this new power level. It is not clear how representative the current operating experience of Units 2 and 3 is of the Unit 1 operating history. The application acknowledges this on page B-4: "During the performance of the Aging Management Review activities, there was recognition that the operating experience on Unit 1 may not be the same as the operating experience on Units 2 and 3 due to the layup program implemented on Unit 1 during its extended outage."

In several places in the SER, the staff documents how the applicant plans to compensate for the lack of plant-specific operating experience. Examples include a commitment to perform periodic inspections of components that were in layup and have been requalified without replacement, and use of Unit 3 layup experience that appears applicable to Unit 1. However, the SER does not provide a cohesive discussion of the applicability of Units 2 and 3 operating experience to Unit 1 and of the compensating actions taken where such experience is not sufficient. Without such discussion, it is not clear that the issue has been consistently recognized and evaluated.

Section 3.7 of the SER documents the staff's aging management review of Unit 1 systems that were in layup for extended outage. This section identifies several instances of deficient layup conditions during the early phase of the extended outage and raises the possibility of potential latent effects that may result in accelerated aging once the plant restarts and operates at power. In the application, the applicant proposed to perform one-time inspections of systems in layup before plant restart to address this issue. The staff concludes that these restart inspections, whose purpose is to assess conditions of components in layup prior to restart, are not equivalent to the one-time inspections used in license renewal to confirm the absence of significant degradation in areas of expected low susceptibility, but not usually subject to inspection. Furthermore, the staff concludes that for portions of Unit 1 systems that have not been replaced, there is insufficient operating history or data to conclude that one-time inspections are appropriate substitutes for periodic inspections.

In response to these concerns raised by the staff, the applicant has agreed to perform periodic inspections of certain Unit 1 systems that were kept in layup during the extended outage and will not be replaced. Restart inspections will still be performed and will provide a baseline for comparison with the subsequent periodic inspections. During our meetings, the applicant stated that targeted locations will be subjected to at least three inspections, one prior to startup, one prior to entering the period of extended operation, and one during the license renewal period. The frequency of subsequent inspections would be determined based on the results of these inspections.

We agree with the staff that periodic inspections of systems and components that were not replaced are appropriate and necessary. However, it is not clear which systems will be included in the scope of the periodic inspection program. For example, in Section 3.7 of the SER, the staff agrees with the applicant's proposal to perform only a one-time inspection of the high-pressure coolant injection system and the containment system prior to Unit 1 restart.

No further attributes of this future program have been provided in the SER. The main attributes of the program, including the intended scope, need to be defined in the final SER. Periodic inspections are the most significant compensating actions for the lack of plant-specific operating experience of BFN Unit 1. It is not possible to judge the adequacy of this important program since insufficient information has been provided. As a result of our review, the staff elevated this issue from a confirmatory item to an open item and requested the applicant to provide details of the periodic inspection program prior to issuance of the final SER.

Some restart inspections continue to be referred to as "one-time" inspections. "One-time" inspections have a specific intent and meaning when performed for license renewal purposes. To avoid confusion, the term "one-time" inspection should be used only for license-renewal-related inspections.

According to current plans, all three BFN units will be subjected to an EPU that will raise their power output to 3952 MWt prior to entering the period of extended operation. However, the license renewal application and the associated SER reflect operating experience only at the current power level. If this EPU is implemented, the staff should require that, prior to entering the period of extended operation, TVA conduct an evaluation of operating experience of BFN Units 1, 2, and 3 at the EPU level and incorporate lessons learned into their aging management programs.

Sincerely,

/RA/

William J. Shack
Acting Chairman

References:

5. Tennessee Valley Authority, "Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 - Application for Renewed Operating Licenses," December 31, 2003
6. Tennessee Valley Authority, "Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 - January 28, 2004 Meeting Follow-Up - Additional Information," February 19, 2004
7. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report with Open Items Related to the License Renewal of the Browns Ferry Nuclear Plant, Units 1, 2, and 3," August 2005
8. Brookhaven National Laboratory, "Audit and Review Report for Plant Aging Management Programs (AMPs) and Aging Management Reviews (AMRs), Browns Ferry Nuclear Plant Units 1, 2, and 3," April 26, 2005
9. U.S. Nuclear Regulatory Commission, "Browns Ferry Nuclear Plant - Inspection Report 05000259/2004012, 05000260/2004012, and 05000296/2004012," January 27, 2005
10. U.S. Nuclear Regulatory Commission, "10 CFR Parts 2, 50, 54, and 140, Nuclear Power Plant License Renewal," *Federal Register*, Vol. 54, No. 240, December 13, 1991, pp. 64943-64980
11. U.S. Nuclear Regulatory Commission, "10 CFR Parts 2, 51, and 54, Nuclear Power Plant License Renewal; Revisions," *Federal Register*, Vol. 60, No. 88, May 8, 1995, pp. 22461-22495

current power level. If this EPU is implemented, the staff should require that, prior to entering the period of extended operation, TVA conduct an evaluation of operating experience of BFN Units 1, 2, and 3 at the EPU level and incorporate lessons learned into their aging management programs.

Sincerely,

William J. Shack
Acting Chairman

References:

12. Tennessee Valley Authority, "Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 - Application for Renewed Operating Licenses," December 31, 2003
13. Tennessee Valley Authority, "Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 - January 28, 2004 Meeting Follow-Up - Additional Information," February 19, 2004
14. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report with Open Items Related to the License Renewal of the Browns Ferry Nuclear Plant, Units 1, 2, and 3," August 2005
15. Brookhaven National Laboratory, "Audit and Review Report for Plant Aging Management Programs (AMPs) and Aging Management Reviews (AMRs), Browns Ferry Nuclear Plant Units 1, 2, and 3," April 26, 2005
16. U.S. Nuclear Regulatory Commission, "Browns Ferry Nuclear Plant - Inspection Report 05000259/2004012, 05000260/2004012, and 05000296/2004012," January 27, 2005
17. U.S. Nuclear Regulatory Commission, "10 CFR Parts 2, 50, 54, and 140, Nuclear Power Plant License Renewal," *Federal Register*, Vol. 54, No. 240, December 13, 1991, pp. 64943-64980
18. U.S. Nuclear Regulatory Commission, "10 CFR Parts 2, 51, and 54, Nuclear Power Plant License Renewal; Revisions," *Federal Register*, Vol. 60, No. 88, May 8, 1995, pp. 22461-22495

DOCUMENT NAME: **E:\Filenet\ML052920483.wpd**
To receive a copy of this document, indicate in the box: "C" = Copy without attachment/enclosure "E" = Copy with attachment/enclosure "N" = No copy Accession #: ML052920483

OFFICE	ACRS/ACNW	Y	ACRS/ACNW	Y	ACRS/ACNW	Y	ACRS/ACNW	Y	ACRS/ACNW	Y	ACRS/ACNW	Y
NAME	CSantos		MSnodderly		MScott		AThadani		JLarkins		JTL for WJS	
DATE	10/18/05		10/18/05		10/18/05		/ /05		10/19/05		10/19/05	

OFFICIAL RECORD COPY

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

September 22, 2005

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

SUBJECT: DRAFT FINAL REVISIONS TO GENERIC LICENSE RENEWAL GUIDANCE DOCUMENTS

Dear Chairman Diaz:

During the 525th meeting of the Advisory Committee on Reactor Safeguards, September 8-10, 2005, we reviewed the draft final revisions to NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," and Regulatory Guide 1.188, "Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses," as well as NEI 95-10, Rev.6, "Industry Guideline for Implementing the Requirements of 10 CFR Part 54 — The License Renewal Rule," which is endorsed by Regulatory Guide 1.188. These documents provide guidance for preparing and reviewing license renewal applications (LRAs). During our review, we had the benefit of discussions with representatives of the NRC staff and the Nuclear Energy Institute (NEI). We also had the benefit of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

1. The draft final revisions to the generic license renewal guidance documents should be approved for issuance.

2. The staff should continue to evaluate the need for revisions to the guidance documents in order to maintain them current.

DISCUSSION

The generic license renewal guidance documents were first issued in 2001. Since then, many license renewal applications have been reviewed and approved by the staff. The guidance documents have proven effective in guiding and simplifying preparation and review of the industry applications. However, in preparing and reviewing individual applications, the applicants and the staff have identified numerous opportunities for improvements. The current revisions incorporate such improvements.

Some of the improvements stem from eliminating excessive specificity and prescriptiveness in the guidance that resulted in unnecessary exceptions to the GALL Report. Components with similar materials, environments, and aging management programs (AMPs) were "rolledup" into a single line item in the aging management review (AMR) tables. Technical criteria such as temperature and fluence thresholds for aging effects were added to permit screening out components in relatively benign environments. The use of more practical component groupings, material nomenclature, and more detailed environmental definitions should make it simpler for applicants to demonstrate consistency with the GALL Report. Chapter IX was added to the GALL Report to standardize and define the terminology used in the document.

Another, more technical category of changes and additions incorporated into the guidance documents are some of the NRC positions established in the final Safety Evaluation Reports (SERs) (approved precedent). Final SERs and staff comments for improving the license renewal process identified over 400 items that were evaluated for inclusion in the guidance documents. Approved interim staff guidance was also incorporated in the guidance documents.

NEI proposed additional AMR line items for new material, environment, aging effect, and aging management program (MEAP) combinations that are common to most LRAs.

The staff reviewed domestic and foreign operating experience to identify potential new AMR line items. The review of foreign experience did not lead to any changes. The review of domestic experience resulted in changes to one AMR and the addition of a new AMR. An AMR was modified to emphasize the need to manage stress corrosion cracking in nozzle safe end welds. An AMR was added to manage primary water stress corrosion cracking in pressurizer steam space nozzles.

During our review of the Dresden and Quad Cities LRA, we recommended that steam dryers be included in the scope of license renewal. We also recommended that the staff require that, prior to entering the period of extended operation, the applicant conduct an evaluation to ensure that operating experience at extended power uprate (EPU) levels is properly addressed in aging management programs and that the staff review and approve this evaluation. The staff added a new line item to the GALL Report that calls for plant-specific AMPs to manage the effects of flow-induced vibration on steam dryers. Section 3.0.2 of the Standard Review Plan states that applicants with recently approved EPUs are to commit to perform an operating experience review at the EPU level prior to entering the period of extended operation.

The current revisions to the guidance documents have been a major undertaking. The changes to the guidance documents are comprehensive and appropriate. They will facilitate the demonstration of the consistency of applications with the GALL Report and staff reviews, and will reduce the number of requests for additional information that are required to support the staff's review. We have previously commented on the value and significance of the GALL Report as a source of information that is critical to managing aging. The current revisions are major improvements of this important document as well as the other guidance documents and should be approved. The staff should continue to evaluate the need for revisions to the guidance documents in order to maintain them current. The contributions of the staff, the industry, and the public to these revisions should be recognized.

Drs. William Shack and George Apostolakis did not participate in the Committee's deliberations regarding this matter.

Sincerely,

/RA/

Graham B. Wallis
Chairman

References:
1. U.S. Nuclear Regulatory Commission, NUREG-1801, Revision 1, Volume 1, "Generic Aging Lessons Learned (GALL) Report Summary," August 2005
2. U.S. Nuclear Regulatory Commission, NUREG-1801, Revision 1, Volume 2, "Generic Aging Lessons Learned (GALL) Report Tabulation of Results," August 2005
3. U.S. Nuclear Regulatory Commission, NUREG-1800, Revision 1, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," August 2005
4. U.S. Nuclear Regulatory Commission Regulatory Guide 1.188, Revision 1, "Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses," August 2005
5. U.S. Nuclear Regulatory Commission, draft NUREG-1832, "Analysis of Public Comments on the Revised License Renewal Guidance Documents," August 2005
6. U.S. Nuclear Regulatory Commission, draft NUREG-1833, "Technical Bases for Revision to the License Renewal Guidance Documents," August 2005
7. Nuclear Energy Institute, "Industry Guideline for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule," NEI 95-10, Revision 6, June 2005

July 18, 2005

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

SUBJECT: DOMINION NUCLEAR NORTH ANNA, LLC, EARLY SITE PERMIT APPLICATION AND THE ASSOCIATED NRC FINAL SAFETY EVALUATION REPORT

Dear Chairman Diaz:

During the 524th meeting of the Advisory Committee on Reactor Safeguards, July 6-8, 2005, we met with representatives of the NRC staff and Dominion Nuclear North Anna, LLC (Dominion) and discussed the final safety evaluation report of the Dominion application for the North Anna early site permit (ESP). Our reviews of the application and the staff's safety evaluation report were conducted to fulfill the requirement of 10 CFR 52.23, which states that the ACRS shall report on those portions of an early site permit application that concern safety. We had the benefit of the documents referenced.

CONCLUSIONS

- The proposed site, subject to the permit conditions recommended by the NRC staff, can be used for up to two nuclear power units each of up to 4300 MW_{th} without undue risk to the public health and safety.
- The staff's final safety evaluation report of the Dominion early site permit application will contribute to the documentary basis for the mandatory public hearing concerning the proposed early site permit.

DISCUSSION

Dominion has submitted a first-of-a-kind application for an early site permit pursuant to the requirements of Subpart A, "Early Site Permits," of 10 CFR Part 52. The proposed site is entirely within the current North Anna Power Station site about 40 miles north-northwest of Richmond, Virginia. Years ago, this site was approved for four units, but only two units (3-loop Westinghouse pressurized water reactors) were constructed. Both of these units are now operating.

The Dominion application is to locate up to two nuclear power units on the proposed site. Each unit is to have a power of up to 4300 MW_{th}. The Dominion application is based on a set of conservative, enveloping parameters defined to allow flexibility in the selection of reactor technology should a decision be made in the future to actually develop the site.

- Nature of the Proposed Site

The vicinity of the proposed site is rural in nature. There are no significant industrial, transportation, or military facilities within five miles of the site center. The major water sources available to the site are the North Anna river and an artificial lake adjacent to the site. The dam for this lake is under the control of the applicant. The applicant has recognized that water availability may be insufficient for two water-cooled units and proposes air cooling for one unit on the proposed site. The staff proposes that this be made a permit condition.

- Population in the Vicinity of the Site

The permanent population around the site is quite low. The nearest population center, Mineral, Virginia, has a population of less than 500. The nearest significant cities are Fredericksburg (projected year 2065 population 20,950) at a distance of 22 miles, Charlottesville (year 2000 population 45,069) at 36 miles, and Richmond (year 2000 population 197,790) at 40 miles. The applicant used methods found acceptable by the staff to show that projected populations in the vicinity of the site through the year 2065 will still be within acceptable limits.

- Geology and Seismicity of the Site

The proposed site will have reactors founded on hard rock. Dominion has undertaken a thorough effort to update geologic and seismic information concerning the site and has made use of methods that are new since the construction of reactors now operating on the North Anna site to characterize the proposed site. The staff has approved these analyses as they have been amended in four revisions of the initial application. Because of the hard rock foundations, reactors on the site would be subject to significant seismically-induced accelerations at frequencies in excess of 10 Hz. Dominion originally proposed to use a new "performance-based" method described in its application to derive a safe shutdown earthquake spectrum that bounds what was determined by the staff using its own methods. The staff has not endorsed the proposed performance-based applicant's methods. Dominion has ultimately elected to use the staff's method as identified in Regulatory Guide 1.165. The staff concurs with conclusions reached by the applicant.

- Meteorology

The applicant has done a thorough examination of historical meteorological data to set design constraints for such things as maximum rainfall, wind velocities, snow pack and temperature extremes. The staff has found these findings to be acceptable. The design constraints posed by the proposed site meteorology are not severe in comparison to design parameters for candidate reactor technologies considered in the development of the early site permit application.

- Potential Radionuclide Releases

For the studies of radiological source terms at the proposed site, Dominion has selected two advanced reactors that could be located on the site. These example plants (AP1000 and the Advanced Boiling Water Reactor) have very low predicted core damage frequencies relative to those predicted for the extant plants on the North Anna site. Dominion has used staff-approved methods to deduce that consequences of radionuclide release at the proposed site will be less than considered in the applications for the design certifications of the example plants. The staff has verified these conclusions with its own evaluations.

- Emergency Plans

The applicant has elected to submit for review just the "major features" of emergency planning for the proposed site as is allowed by the regulations. The staff has found these major features to be acceptable and concludes that the proposed site does not pose significant impediments to the development of adequate emergency plans should a decision be made to develop the site.

The staff has identified a number of items that are treated either as permit conditions or as actions that must be addressed at the combined license (COL) stage. The staff has developed criteria to identify permit conditions. Permit conditions are recommended by the staff when:

- evaluations of the site rest on an assumption that can be justified only after a site permit has been issued,
- a physical attribute exists for the site that is not acceptable for the design of systems, structures and components important to safety, or
- evaluations can be completed only after some future act has taken place.

We conclude that these are appropriate criteria for the imposition of permit conditions.

The staff has prepared a high-quality, detailed, yet readable, safety evaluation report on the Dominion application. All open items have been resolved. The staff concludes that the site is adequate for the proposed use subject to eight permit conditions.

The staff has also identified 30 items that need to be considered in conjunction with reviews of a COL application should the early site permit be granted and a decision to develop the site be made.

We concur with the staff's conclusions concerning the Dominion application for an early site permit. This first use of the early site permit process has revealed several areas where the process can be refined and streamlined. We look forward to working with the staff to improve the early site permit process.

Sincerely,

/RA/

Graham B. Wallis
Chairman

References :
1. U.S. Nuclear Regulatory Commission, Final Safety Evaluation Report, "Safety Evaluation of Early Site Permit Application in the Matter of Dominion Nuclear North Anna, LLC, for the North Anna Early Site Permit', June 16, 2005.

2. North Anna Early Site Permit Application, Revision 3, September 2004, NRC Docket No. 51-008.

3. U.S. Nuclear Regulatory Commission, Review Standard, RS-002, "Processing Applications for Early Site Permit Applications", May 3, 2004.

4. Memorandum from Luis A. Reyes, NRC Executive Director for Operations, to Graham B. Wallis, Chairman, ACRS, Subject: Interim Letter: Draft Safety Evaluation Report on North Anna Early Site Permit Application, dated June 3, 2005.

5. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.165, "Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion," dated March 1997.

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

March 11, 2005

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

SUBJECT: INTERIM LETTER: DRAFT SAFETY EVALUATION REPORT ON NORTH ANNA EARLY SITE PERMIT APPLICATION

Dear Mr. Reyes:

During the 520th meeting of the Advisory Committee on Reactor Safeguards (ACRS), March 3-5, 2005, we met with representatives of the NRC staff and Dominion Nuclear North Anna, LLC (Dominion) and discussed the NRC staff's draft safety evaluation report and the application related to North Anna early site permit (ESP). This matter was also discussed during our ESP Subcommittee meeting on March 2, 2005. We are conducting such reviews to fulfill the requirement of 10 CFR 52.23, which states that the ACRS shall report on those portions of an early site permit application that concern safety. We also had the benefit of the documents referenced.

CONCLUSIONS

Staff is preparing a quality safety evaluation of a first-of-a-kind application for an early site permit.

DISCUSSION

Dominion has submitted a first-of-a-kind application for an early site permit. Dominion seeks to locate up to two nuclear power units, each with a thermal power of up to 4300 MW, entirely within the current North Anna power station site about 40 miles north-northwest of Richmond, Virginia. Years ago, this site was approved for four units, but only two units (3-loop Westinghouse pressurized water reactors) were constructed. Both of these units are now operating on the site.

The application by Dominion and the safety evaluation report are lengthy, but nevertheless very readable documents that have been well prepared by their respective authors and represent significant amounts of effort.

At the time of our review, several open items remained under discussion between Dominion and the staff. We determined that none of these open items precluded our review of the application and the safety evaluation report and the preparation of this interim letter.

Applications for early site permits are subject to the requirements of 10 CFR 52.17. Staff's review of these applications is guided by the Review Standard (RS-002) "Processing Applications for Early Site Permits," which we previously reviewed. Major elements required in an early site permit application and staff's findings concerning these elements are discussed below:

- Nature of the Proposed Site

The vicinity of the proposed site is rural in nature. There are no significant industrial, transportation, or military facilities within five miles of the site center. The major water sources available to the site are the North Anna river and the artificial lake adjacent to the site. The dam for this lake is under the control of the applicant. The applicant has recognized that water availability may be insufficient for two water-cooled units and proposes air cooling for one unit on the proposed site.

- Population in the Vicinity of the Site

The permanent population around the site is quite low. The nearest population center, Mineral, Virginia, has a population of less than 500. The nearest significant cities are Fredericksburg (projected Year 2065 population 20,950) at 22 miles and Charlottesville (Year 2000 census population 45,049) at 36 miles. A significant transient population makes use of the recreational opportunities afforded by the lake. The applicant has used methods found acceptable by the staff to show that projected populations in the vicinity of the site through Year 2065 will remain within acceptable limits.

- Geology and Seismicity of the Site

Since construction of the units now on the North Anna site, new methods of seismic hazard analysis have been developed and are recommended by NRC for site characterization. Dominion has undertaken a thorough effort to update geologic and seismic information concerning the site and has made use of the new methods to characterize the site. Staff has approved these analyses as they have been amended in three revisions of the initial application. We are skeptical of accepting categorization of possible quaternary seismic features published in archival documents without scrutinizing the bases for the categorization to ensure these bases are consistent with the needs of safety regulation. The categorization done for this application is not consequential because the applicant has adopted conservative seismic sources.

The proposed North Anna site will have reactors founded on hard rock. Consequently, seismically induced accelerations of interest extend to frequencies in excess of 10 Hz. The applicant has used a "performance based" method described in its application to derive a safe shutdown earthquake spectrum that bounds what was determined by the staff using its own methods. Staff has not endorsed the applicant's methods, but concurs with the conclusion. The safe shutdown earthquake for the site exceeds the design-basis earthquakes for the example plants considered in the development of the early site permit application (the AP1000 pressurized water reactor and the ABWR boiling water reactor). Such discrepancies will have to be addressed when the election is made to actually build nuclear units on the site. The site safe shutdown earthquake also exceeds at frequencies above about 5 Hz the safe shutdown earthquake for the plants currently on the site and it exceeds the limiting earthquake found in the individual plant examination of external events (IPEEE) assessments for these plants at frequencies above about 10 Hz. The staff is pursuing the issues these findings raise. Staff anticipates that displacements associated with the high frequency motions will not pose safety threats to the operating plants.

- Meteorology

The applicant has done a thorough examination of historical meteorological data to set design constraints for such things as maximum rainfall, wind velocities, snowpack, and temperature extremes. Staff has approved these findings. Despite active scientific research and popular interest in the evolution of weather and climate, there is no discussion either in the application or in the safety evaluation report of how weather and climate patterns may be changing. The application and the safety evaluation report should discuss these matters. Indeed, it appears that staff's own guidance (RS-002) indicates that it should do this by stating, "The applicability of these data to represent site conditions during the expected period of reactor operations should be substantiated."

- Potential Radiological Source Terms

For the radiological source term studies, the applicant has selected two advanced reactors as example power plants that could be located on the site. These example plants (AP1000 and the ABWR) have very low predicted core damage frequencies relative to those predicted for the extant plants on the North Anna site. The applicant has used staff-approved methods to deduce that consequences of radionuclide release at the proposed site will be less than considered in the applications for design certification of the example plants. Staff's evaluations verified these conclusions. Neither the application nor the safety evaluation report provides sufficient information for the interested reader to reproduce these analyses or to judge the reasonableness of the conclusions.

- Emergency Plans

The applicant has elected to submit for review just the "major features" of emergency planning for the proposed site, as is allowed by the regulations. Unfortunately, the regulations do not provide a clear definition of what is meant by the term "major features" as it applies to emergency plans. As a result, both the applicant and the staff reviewers have delved into details of emergency plans that will change undoubtedly by the time any decision is made to construct a plant on the site. We question the need for such detailed examinations of emergency plans for proposed sites that are on or adjacent to sites with operating plants having approved emergency plans.

In conclusion, we see a promising start to the first application of the early site permit process both on the part of the applicant and on the part of the staff reviewing the application. We look forward to examining a final version of the staff's safety evaluation report. Furthermore, we hope to work with the staff in the development of "lessons learned" from the review of this and the next few applications for early site permits.

Sincerely,

/RA/

Graham B. Wallis
Chairman

References:

1. U.S. Nuclear Regulatory Commission, Draft Safety Evaluation Report, "Safety Evaluation of Early Site Permit Application in the Matter of Dominion Nuclear North Anna, LLC, for the North Anna Early Site Permit," December 2004.

2. North Anna Early Site Permit Application, Revision 3, September 2004, NRC Docket No. 51-008.

3. U.S. Nuclear Regulatory Commission, Review Standard, RS-002, "Processing Applications for Early Site Permit Applications", May 3, 2004.

4. Report from Mario V. Bonaca, ACRS Chairman, to Richard A. Meserve, NRC Chairman, Subject: Draft Review Standard, RS-002: "Processing Applications For Early Site Permits", dated March 12, 2003.

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

June 14, 2005

Mr. Luis A. Reyes
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-00001

SUBJECT: INTERIM LETTER: DRAFT SAFETY EVALUATION REPORT ON GRAND GULF EARLY SITE PERMIT APPLICATION

Dear Mr. Reyes:

During the 523rd meeting of the Advisory Committee on Reactor Safeguards (ACRS), June 1-3, 2005, we met with representatives of the NRC staff and System Energy Resources, Inc. (SERI), the applicant for an early site permit (ESP) for the Grand Gulf site, and discussed the application and the NRC staff's draft Safety Evaluation Report (SER). This matter was also discussed during the meeting of our Early Site Permit Subcommittee on May 16, 2005. We are conducting our review of early site permit applications to fulfill the requirement of 10 CFR 52.23 that the ACRS report on those portions of an ESP application that concern safety. We had the benefit of the documents referenced.

CONCLUSION AND RECOMMENDATION

The staff has prepared a quality draft SER of the SERI application for the Grand Gulf early site permit. The draft SER should be augmented with a more complete exposition on threats posed by transportation accidents on the river adjacent to the proposed site.

DISCUSSION

The SERI application is the second early site permit application we have reviewed. At the time of our review, 23 items remained under discussion between the staff and the applicant. We determined that none of these open items precluded our review of the application and the draft SER for the purpose of preparing this interim report.

SERI seeks a site permit for a reactor or a set of reactor modules of total power up to 4300 MW_{th} on a site adjacent to the current Grand Gulf Nuclear Power Station Unit 1, a BWR/6 with a Mark III containment. With the additional unit or modules, the total nuclear generating capacity would be 8600 MW_{th}. The Grand Gulf site had previously been approved for two units, but the second unit was never completed.

- Nature of the Proposed Site

The proposed site is located on the eastern side of the Mississippi River about 25 miles south of Vicksburg, Mississippi. The site is quite rural in nature. There is little industrial activity near the site and no nearby military bases. There is a natural gas pipeline somewhat more than 4 miles from the site.

The nearest major airport is at Jackson, Mississippi, about 65 miles from the proposed site. Air traffic corridors near the site have been determined by the staff to pose no undue risk. There is a highway 4½ miles from the site. The principal ground transportation hazard, however, is thought to involve the delivery of hydrogen to the site for use in the currently operating boiling water reactor.

There is, of course, an important river transportation corridor 1.1 miles from the site. The staff should provide a more explicit characterization of the proposed site in terms of accidents on the river. The staff needs to augment the treatment of explosion and fire events with a discussion of the potential for accidents involving release of toxic chemicals such as chlorine and ammonia.

- Population in the Vicinity of the Site

The permanent population around the site is low. The nearest town, Port Gibson, Mississippi, is about 6 miles away and has a population of about 1750. The nearest population center, Vicksburg, Mississippi, is 25 miles to the north and has a current population of 27,000. Projected population growth in the area to year 2070 is expected to be small, perhaps less than 20%.

- Geology and Seismicity of the Site

The proposed site is located on consolidated river sediments. Geological investigations show no evidence of significant ground deformations for at least the last 500,000 years and perhaps for the last 5 million years. Salt domes in the area are 6 and 8 miles from the proposed reactor location.

The site is in an area of little seismic activity. The nearest historical seismic event occurred more than 25 miles away. The limiting earthquake source is the New Madrid seismic zone over 200 miles away. SERI has undertaken a probabilistic seismic hazard analysis that takes into account recent revisions made by the U.S. Geological Survey to the frequencies and intensities of events in the New Madrid seismic center. The analysis also considers the possibility of seismic activity along the suspected faults on the Saline River which may not be capable faults. The proposed site is a deep soil site (bed rock is at a depth of about 10,000 feet). SERI has done sufficient characterization of the site to produce analyses of the soil amplification factors. The probabilistic seismic hazard curve developed for the site is bounded by the design safe shutdown earthquake curves adopted in the plant parameter envelope developed by SERI.

- Meteorology

Weather at the proposed site is mild relative to many reactor sites. Vigorous storms such as hurricanes and tornados are the principal weather threats. SERI and the staff have used historical information to characterize these and other weather features of the site. We note that the staff has done a good job critically reviewing and correcting the applicant's historical weather data. We continue to question the defensibility of the methods used by the staff and the applicant to prognosticate the weather at the site over the next 65 years based just on historical frequencies of severe weather events. At the very minimum, staff should review current literature on possible changes in weather in the upper Gulf of Mexico to be confident that the methods used for weather predictions are defensible.

Flooding is a concern about the site given its location adjacent to a major river. The proposed reactor site is, however, on a "bluff" some 65 feet above the normal river levels. Land on the opposite bank of the river is more easily flooded and it is expected, therefore, that river flooding is not a significant threat to the site. Local, onsite flooding will have to be addressed if the permit is granted and a decision is made to construct a power plant on the site.

- Emergency Plans

The applicant has elected to submit for review just the "major features" of emergency planning for the proposed site, as is allowed by the regulations. These major features appear adequate should a new plant be built on the site.

We conclude this report by noting that the staff's draft SER is comprehensive, and, though lengthy, is a well constructed, readable document.

Sincerely,

/RA/

Graham B. Wallis
Chairman

References:

1. U.S. Nuclear Regulatory Commission, Draft Safety Evaluation Report, "Safety Evaluation of Early Site Permit Application in the Matter of System Energy Resources, Inc., a Subsidiary of Entergy Corporation, for the Grand Gulf Early Site Permit Site," April 2005.

2. System Energy Resources, Inc., Grand Gulf Early Site Permit Application, Revision 0, October 2003.

3. U.S. Nuclear Regulatory Commission, Review Standard, RS-002, "Processing Applications for Early Site Permit Applications," May 3, 2004.

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

November 18, 2005

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

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE LICENSE RENEWAL APPLICATION FOR THE POINT BEACH NUCLEAR PLANT UNITS 1 AND 2

Dear Chairman Diaz:

During the 527th meeting of the Advisory Committee on Reactor Safeguards, November 3-5, 2005, we completed our review of the license renewal application for the Point Beach Nuclear Plant (PBNP) Units 1 and 2, and the final Safety Evaluation Report (SER) prepared by the NRC staff. We issued an interim report on the safety aspects of this application and the draft SER on June 9, 2005. Our Plant License Renewal Subcommittee also reviewed this matter during a meeting on May 31, 2005. During these reviews, we had the benefit of discussions with representatives of the NRC staff and the Nuclear Management Company, LLC (NMC). 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 inclusion of the conditions in Recommendation 2, the NMC application for license renewal of PBNP Units 1 and 2 should be approved.

2. The staff should expand the scope of its post-approval site inspection to verify that all license renewal programs have been implemented and commitments have been met. In addition, the staff should review the effectiveness of the PBNP corrective action program (CAP) before PBNP enters the period of extended operation.

BACKGROUND AND DISCUSSION

The PBNP Units 1 and 2 are two-loop Westinghouse pressurized water reactors housed in dry ambient containments. Originally, each unit was licensed at a power level of 1519 MWt. Each unit has undergone a low-pressure turbine modification and a measurement uncertainty recapture power uprate to increase the power level to 1540 MWt. NMC has requested renewal of the operating licenses of Units 1 and 2 for 20 years beyond their current license terms, which expire on October 5, 2010, and March 8, 2013, respectively.

In the final SER, the staff documents its review of the license renewal application and other information submitted by the applicant and obtained through the audits and inspections 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; and the identification and assessment of time-limited aging analyses (TLAAs).

The PBNP application demonstrates consistency with, or documents deviations from, the approaches specified in the Generic Aging Lessons Learned Report. The staff questioned the applicant's approach to identifying nonsafety-related components whose failure could affect safety-related components. The applicant modified its scoping methodology to address the staff's questions. An inspection completed on August 17, 2005 confirmed that this methodology has been appropriately implemented. In the final SER, the staff concludes that the scoping and screening processes implemented by the applicant have successfully identified SSCs within the scope of license renewal and subject to an aging management review. We agree with this conclusion.

The applicant performed a comprehensive aging management review of all SSCs within the scope of license renewal. In the application, the applicant describes 26 aging management programs for license renewal, including existing, enhanced, and new programs. The draft SER identified 5 open items and 15 confirmatory items. The final SER describes the resolution of these items. We agree with the resolution of these items and with the staff's conclusion that the applicant's proposed aging management programs are adequate.

One of the open items relates to plant-specific operating experience of the two units. Containment liner corrosion due to borated water leakage has been identified in both units. The applicant has committed to performing augmented inspections in accordance with ASME Section XI Subsection IWE to monitor the extent of corrosion. The Boric Acid Corrosion Program is also credited with assessing and managing loss of material in the containment liner. The augmented inspection program does not include specific criteria for evaluation, repair, or replacement. At the staff's request, the applicant has agreed to include in the acceptance criteria element of the aging management program, "ASME Section XI, Subsections IWE and IWL Inservice Inspection Program," an appropriate discussion of the evaluation, repair or replacement criteria, and reexamination requirements necessary to ensure leak-tightness and structural integrity of the liner.

The applicant identified and reevaluated systems and components requiring TLAAs for 20 more years of operation. The upper shelf energy for both vessels and the reference temperature for pressurized thermal shock (PTS) for the Unit 2 vessel failed to meet the screening criteria.

To address the low upper shelf energy, the applicant performed equivalent margin analyses allowed by 10 CFR Part 50, Appendix G. These analyses yielded acceptable results through the end of the period of extended operation. The staff performed independent analyses to confirm the applicant's conclusion.

The intermediate-to-lower shell circumferential weld of the Unit 2 vessel is projected to exceed the PTS screening criterion in 2017. Consistent with the requirements of 10 CFR 54.21(c)(1)(iii), the applicant has chosen to manage the effects of aging of this weld during the period of extended operation. The applicant's commitments for PTS include implementing a low-low leakage fuel management pattern, using hafnium absorber assemblies, and documenting a flux reduction plan. This documentation will include any required safety analyses supporting continued operation. Other options the applicant may pursue include a more refined analysis of PTS or thermal annealing of the reactor pressure vessel.

In our June 9, 2005 interim report on the PBNP application, we expressed concern with the effectiveness of the PBNP CAP and the applicant's ability to effectively implement license renewal programs and meet commitments. We were concerned that the resources needed to address the staff's April 21, 2004 Confirmatory Action Letter to PBNP would compete with the effective development, tracking, and implementation of license renewal programs and commitments. We recommended that, prior to the units entering the period of extended operation, the staff take additional actions to increase confidence that the requirements of the license renewal rule have been met. We suggested, for example, an expanded inspection of license renewal commitments and a focused review of the effectiveness of the CAP. The PBNP remains in the Multiple/Repetitive Degraded Cornerstone column of the Reactor Oversight Process Action Matrix, and there are still weaknesses in the CAP.

In its July 15, 2005 response to the Committee, the staff described the inspections being conducted at PBNP to verify that license renewal programs and commitments are appropriate and consistent with the rule. However, detailed development and implementation of many of these programs and commitments will occur after the license is renewed and prior to the license renewal period. The staff plans to perform a post-approval site inspection in accordance with Inspection Procedure 71003 before the period of extended operation begins.

Inspection Procedure 71003 is the standard inspection that the staff performs prior to the period of extended operation. This inspection evaluates only a sample of the license renewal commitments and programs. In light of the applicant's weakness in managing commitments, as discussed in our interim report, the staff should expand the scope of the post-approval site inspection to verify that all license renewal programs have been implemented and commitments have been met. In addition, before PBNP enters the period of extended operation, the staff should review the effectiveness of the CAP. These actions are necessary to ensure that there is reasonable assurance that aging degradation can be adequately managed.

With a commitment to perform the expanded inspections described above, the application for renewal of the operating licenses of the PBNP Units 1 and 2 should be approved.

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References:
1. Nuclear Management Company, LLC, "Application for Renewed Operating Licenses Point Beach Nuclear Plant Units 1 & 2," February 2004.
2. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the License Renewal of the Point Beach Nuclear Plant, Units 1 and 2," May 2005.
3. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the License Renewal of the Point Beach Nuclear Plant, Units 1 and 2," October 2005.
4. Letter from Graham B. Wallis, Chairman, ACRS, to Luis A. Reyes, Executive Director for Operations, NRC, "Interim Report on the Safety Aspects of the License Renewal Application for the Point Beach Nuclear Plant, Units 1 and 2," June 9, 2005.
5. Pacific Northwest National Laboratory, "Audit and Review Report for Plant Aging Management Reviews and Programs, Point Beach Nuclear Plant Units 1 and 2," April 11, 2005.
6. U.S. Nuclear Regulatory Commission, "Point Beach Nuclear Plant, Units 1 and 2 NRC License Renewal Scoping, Screening, and Aging Management Inspection Report 05000266/2005005 (DRS); 05000301/2005005 (DRS)," May 2, 2005.
7. U.S. Nuclear Regulatory Commission, "Point Beach Nuclear Plant, Units 1 and 2 NRC License Renewal Followup Inspection Report 05000266/2005015 (DRS); 05000301/2005015 (DRS)," September 9, 2005.
8. U.S. Nuclear Regulatory Commission, "Point Beach Nuclear Plant, Units 1 and 2 NRC Special Inspection Report 05000266/2005011; 05000301/2005011," September 23, 2005.
9. U.S. Nuclear Regulatory Commission, "Point Beach Nuclear Plant, Units 1 and 2 NRC Special Emergency Preparedness Inspection Report 05000266/2005009 (DRS); 05000301/2005009 (DRS)," August 2, 2005.

10. Letter from J. Dyer, Regional Administrator, to M. Warner, Site Vice President, Kewaunee and Point Beach Nuclear Plants, Nuclear Management Company, LLC, "Point Beach Special Inspection - NRC Inspection Report 50-266/01-17(DRS); 50-301/01-17(DRS), Preliminary Red Finding," April 3, 2002.
11. Letter from J. Dyer, Regional Administrator, to M. Warner, Site Vice President, Kewaunee and Point Beach Nuclear Plants, Nuclear Management Company, LLC, "Point Beach Nuclear Plant Final Significance Determination for a Red Finding and Notice of Violation NRC Special Inspection Report No. 50-266/01-17(DRS; 50-301/01-17(DRS)," July 12, 2002.
12. Letter from J. Dyer, Regional Administrator, to A. Cayia, Site Vice President, Point Beach Nuclear Power Plant, Nuclear Management Company, LLC, "Point Beach Nuclear Plant Special Inspections: Resolution of Auxiliary Feedwater Old Design Issue and Preliminary Red Finding - Auxiliary Feedwater Orifice Plugging Issue; NRC Inspection Report 50-266/02-15(DRP); 50-301/02-15(DRP)," April 2, 2003.
13. Letter from J. Caldwell, Regional Administrator, to A. Cayia, Site Vice President, Point Beach Nuclear Plant, Nuclear Management Company, LLC, "Point Beach Nuclear Plant, Units 1 and 2 Final Significance Determination for a Red Finding and Notice of Violation (NRC Inspection Report No. 50-266/02-15(DRP); 50-301/02-15(DRP))," December 11, 2003.
14. Letter from G. Van Middlesworth, Site Vice President, Point Beach Nuclear Plant, Nuclear Management Company, LLC, to U.S. Nuclear Regulatory Commission Document Control Desk, "Commitments in Response to 95003 Supplemental Inspection," March 22, 2004.
15. Letter from J. Caldwell, Regional Administrator, to G. Van Middlesworth, Site Vice President, Point Beach Nuclear Plant, Nuclear Management Company, LLC, "Confirmatory Action Letter," April 21, 2004.
16. Letter from J. Caldwell, Regional Administrator, to D. Koehl, Site Vice President, Point Beach Nuclear Plant, Nuclear Management Company, LLC, "Annual Assessment Letter - Point Beach Nuclear Plant (Report 05000266/200501; 05000301/200501)," March 2, 2005.
17. Letter from D. Koehl, Site Vice-President, Point Beach Nuclear Plant, Nuclear Management Company, LLC, to U.S. Nuclear Regulatory Commission Document Control Desk, "License Renewal Application Revised Information," September 10, 2004.
18. Memorandum from L. Reyes, EDO, to Chairman Diaz, Commissioner McGaffican, and Commissioner Merrifield, "Pressurized Thermal Shock Analyses for Renewal of Certain Nuclear Power Plant Operating Licenses," May 27, 2004.

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

September 22, 2005

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

SUBJECT: INTERIM LETTER: EXELON GENERATION COMPANY, LLC, APPLICATION FOR EARLY SITE PERMIT AND THE ASSOCIATED NRC STAFF'S DRAFT SAFETY EVALUATION REPORT

Dear Mr. Reyes:

During the 525th meeting of the Advisory Committee on Reactor Safeguards, September 8-10, 2005, we met with representatives of the NRC staff and Exelon Generation Company, LLC (the applicant) to discuss the application for an early site permit for the Clinton site, and the associated NRC staff's draft Safety Evaluation Report. We reviewed the application and the draft Safety Evaluation Report to fulfill the requirement of 10 CFR 52.23 that the ACRS report on those portions of an early site permit application that concern safety. Our Subcommittee on Early Site Permits also discussed this matter during a meeting on September 7, 2005. We also had the benefit of the documents referenced.

RECOMMENDATION

A thorough, expeditious review of the applicant's performance-based seismic hazard analysis methodology should be conducted, recognizing that this methodology may be used by applicants for purposes other than early site permits.

DISCUSSION

Exelon Generation Company, LLC (Exelon) has applied for an early site permit for locating nuclear power plants or modules having a total power generation rate of 2400 to 6800 MW_{th} on the site where the Clinton plant, a BWR6 within a Mark III containment, is currently operating. The early site permit application is based on the now familiar "plant parameter envelope" approach since the applicant has not identified the particular reactor technology that will be adopted. The plant parameter envelope is based on the characteristics of designs such as the AP1000 and Advanced Boiling Water Reactor (ABWR) as well as other designs such as International Reactor Innovative and Secure (IRIS), Economic and Simplified Boiling Water Reactor (ESBWR), Gas-Turbine Modular Helium Reactor (GT-MHR), and Pebble Bed Modular Reactor (PBMR). The staff has prepared a draft Safety Evaluation Report of this application.

This is an interim review of the application and the draft Safety Evaluation Report. This is the third early site permit application we have reviewed this year.

Nature of the Site

The proposed site is located in a rural setting in central Illinois. The terrain is essentially flat with some rolling hills. Nearby population centers with populations in excess of 25,000 include Springfield (74 km away), Peoria (75 km), Champaign (49 km), Urbana (65.5 km), Decatur (36 km), and Bloomington (36 km). Nearer the site (< 16 km away) are the small towns Clinton (population 7000), as well as DeWitt, Weldon, and Wapella each with a population of less than 1000.

Population trends in the larger cities near the site have been estimated based on census data. Modest growth in population over the next 60 years is anticipated in these population centers. Interestingly, data obtained from other sources led the applicant to anticipate that populations in the rural regions around the site will decline modestly over the next 60 years.

Three highways and a railroad run near and through the site. Threats to the plant safety posed by accidents involving hazardous materials on these transportation routes or accidents at agriculture supply facilities in the area have been characterized well by the applicant and do not pose significant safety issues.

Weather

Weather at the proposed site is well characterized in recent years as would be expected for a site with an operating nuclear power plant. The weather is marked by rather warm summer periods and harsh winters. Weather extreme characteristics of the site have been based on historical data. Neither the applicant nor the staff have taken account of literature suggesting that there are cycles in weather that may complicate the prediction of future weather extremes based on historical records.

Seismicity

The essential issue of the proposed site is associated with seismic hazards and related risks. The site can be affected by the New Madrid seismic source (320 km), the Wabash Valley seismic source (209 km) and the central Illinois source zone associated with historic as well as prehistoric earthquakes. The first of these seismic sources has received much study. The U.S. Geological Survey has found that major earthquakes similar to those of the New Madrid seismic source in 1811-1812 recur at intervals of 200 to 800 years. Also evidence indicates that the maximum magnitude of earthquakes at the Wabash Valley source could be larger than had been anticipated at the time the plant now operating at the Clinton site was approved.

The central Illinois seismic source zone is poorly defined. It is thought to be responsible for a large magnitude earthquake in the area of the nearby population center at Springfield about 6700 years ago and perhaps a more recent prehistoric earthquake. There is no particular geologic structure associated with these earthquakes. The Springfield earthquake is known through examinations of prehistoric soil liquefaction evidence. Consequently, the seismic epicenter cannot be as precisely localized as the better known seismic events that are used to characterize the seismic risk at the Clinton site.

The applicant has chosen to characterize the seismic hazard using a methodology that differs from that utilized in previous early site permits and recommended in the agency's Regulatory Guide 1.165. The alternative, American Standards for Civil Engineers (ASCE) Standard 43-05, "Seismic Design Criteria for Systems, Structures and Components in Nuclear Facilities", is an industry standard with a quality pedigree. It may well be used by other applicants in the future for early site permits and other purposes. The alternative has many features in common with the more familiar method recommended by the NRC staff. These features include requirements for surveying literature data and conducting a probabilistic seismic hazard assessment. The methods differ in the target acceptance criterion. The alternative method seeks to find the ground motion spectrum that will result in a 10^{-5} yr^{-1} probability for the onset of inelastic deformation of safety significant systems, structures, and components. The mean ground motion spectrum (plot of peak ground acceleration against vibrational frequency) for the proposed Clinton site calculated using this alternative methodology is quite similar to that derived by the NRC-endorsed methodology for a recurrence frequency of 10^{-4} yr^{-1}. On the other hand, the applicant claims that its results are bounded by results using the NRC-approved methodology for frequencies less than 16 Hz and exceed the results of the approved methodology only modestly at the higher, less important, frequencies. The applicant asserts that the result yields a core damage frequency (CDF) of 1-4 $x10^{-6}$ yr^{-1}. Documentation to substantiate this assertion is not available now for review. The applicant further asserts that the alternative will promote greater regulatory stability in the face of continuing improvements in our understanding of the seismicity of the site though it is not immediately apparent why this is so.

The performance-based treatment of the seismic hazard of the Clinton site proposed by the applicant is an industry standard and merits consideration as an alternative to the methods currently found acceptable by the staff. Thorough review of the proposed methodology is complicated by some discrepancy between inputs to the methodology cited by the applicant and the references from which the inputs were derived. These inputs are, of course, issues in staff requests for additional information that are being considered by the applicant now.

Acceptance of this methodology by the staff for use in connection with the early site permits may have implications for other regulatory activities involving seismic hazard analyses. A thorough, prompt review of the proposed methodology recognizing the breadth of possible applications is needed.

Most open items in the staff review of the non-seismic portion of the Clinton early site permit application have been satisfactorily resolved. The staff is now re-examining the list of 15 permit conditions in light of criteria the staff established during the review of the North Anna early site permit application. It is anticipated that some of the permit conditions will evolve into action items for the combined license stage. The applicant is preparing responses to seven open items identified in connection with the seismic aspects of the application. It is anticipated that a more nearly finalized safety evaluation report will be available for review in early 2006.

Sincerely,

/RA/

Graham B. Wallis
Chairman

References:

1. U.S. Nuclear Regulatory Commission, Draft Safety Evaluation Report, "Safety Evaluation of Early Site Permit Application in the Matter of Exelon Generation Company, LLC, for the EGC Early Permit Site," February 2005.
2. U.S. Nuclear Regulatory Commission, Supplemental Draft Safety Evaluation Report, "Safety Evaluation of Early Site Permit Application in the Matter of Exelon Generation Company, LLC, for the EGC Early Permit Site," August 2005.
3. Exelon Generation Company, LLC, Early Site Permit Application, September 23, 2003.
4. U.S. Nuclear Regulatory Commission, Review Standard, RS-002, "Processing Applications for Early Site Permit Applications," May 3, 2004.
5. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.165, "Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion," March 1997.

PROPOSED ALTERNATIVE EMBRITTLEMENT CRITERIA IN 10 CFR 50.46

Dana A. Powers

Revising 10 CFR 50.46 Fuel and Cladding Issues

- Existing regulatory requirements are clad -specific: Zircaloy and Zirlo™
- Fuel and clad are evolving as we go to higher burnups and will continue to do so
- Regulatory safety objective is to maintain core coolability in the event of an accident

Core Coolability Logic

- Maintain coolability by maintaining core geometry
- Maintain geometry by keeping fuel within clad
- Keep fuel in clad by retaining some clad ductility
- Preserve ductility in zirconium alloy clad by limiting hydrogen uptake during operations and oxygen uptake in accidents

Research Results

- **Synergism between clad hydriding during operations and oxygen uptake makes clad brittle during quenching by ECCS**
- **Research results have led staff to an alloy-independent process for assessing clad embrittlement**

ACRS Conclusions

- Good piece of research
- Important to update regulatory requirements, but they should be independent of cladding or fuel technology
- Regulatory requirement should be to preserve core coolability in design basis accidents
- Acceptable technology-specific methods to demonstrate core coolability should be described in Regulatory Guides

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

September 23, 2005

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

SUBJECT: REPORT ON A PROPOSED TECHNICAL BASIS FOR REVISION OF THE EMBRITTLEMENT CRITERIA IN 10 CFR 50.46

Dear Mr. Reyes:

During the 525th meeting of the Advisory Committee on Reactor Safeguards (ACRS), September 8-10, 2005, the staff made a presentation concerning a proposed technical basis for revision of the embrittlement criteria in 10 CFR 50.46. Our Reactor Fuels Subcommittee also heard a presentation on this matter during a meeting on July 28, 2005. During these reviews, we had the benefit of discussions with representatives of the staff, the Electric Power Research Institute, Westinghouse, and Framatome. We also had the benefit of the document referenced.

RECOMMENDATIONS

- The requirements of 10 CFR 50.46(b) concerning the coolability and geometric integrity of a reactor core during a design-basis loss-of-coolant accident (LOCA) and the aftermath of such an accident should be updated to facilitate the use of better reactor materials and improved understanding of phenomena and processes that affect core integrity and core coolability.
- The updated requirements should be written at a high level so that they are as technology-neutral and materials-neutral as practicable. Methods acceptable to the staff for demonstrating that specific cladding materials meet the high-level requirements of the regulations should be described in regulatory guides.
- The process developed by the staff for the qualification of zirconium alloy cladding provides a basis for a regulatory guide for such materials. The research needed to validate this process should be completed.

DISCUSSION

Regulations dealing with reactor core behavior during design-basis LOCAs (10 CFR 50.46(b)) require that core coolability be maintained during the accident and its aftermath. Coolability can be maintained if the overall core geometry is maintained and reactor fuel is retained within the fuel cladding, which may be "ballooned" and even ruptured. These requirements for core coolability can be achieved if the fuel cladding is not extensively embrittled and retains some ductility once cooled.

Regulatory requirements to achieve these ends are written currently with reference to specific cladding materials (Zircaloy and ZIRLO) and particular oxidation rate correlations. Since the regulations were written, technology has progressed and our understanding of accident phenomena has advanced. New cladding materials are being introduced that allow fuel to be taken safely to higher levels of burnup. Because of the material specificity of the current regulations, exemption requests must be prepared by licensees and reviewed by the NRC staff to take advantage of the newer, better materials. Fortunately, these additional burdens have not stifled the adoption of newer, better fuel cladding, though the potential for such inhibition exists. This is, of course, the danger posed by anachronistic safety regulations. They can create burdens that inhibit licensees and even regulators from adopting improved technology and forego opportunities for having safer nuclear power plants.

The NRC's Office of Nuclear Regulatory Research (RES) has undertaken, in cooperation with the nuclear industry, a confirmatory research program to understand the behavior of fuel cladding at the higher levels of fuel burnup that are becoming common within the nuclear power industry. This research has identified new mechanisms of cladding embrittlement and has improved the understanding of embrittlement mechanisms known at the time the current regulations were written. Based on these early research findings, the RES staff is proposing a revision to the embrittlement criteria that support the regulations that would eliminate reference to specific types of zirconium alloy cladding. The proposed changes would include in the revised regulation a six-step process for the qualification of new fuel cladding:

(1) Determine the extent of oxidation, at 1477 K, of unirradiated cladding that reduces residual ductility to a critical level (nominally 2%), when measured at 408 K.
(2) Determine the time to "breakaway" oxidation rates at lower temperatures (about 1073-1477 K) during accident transients.
(3) Establish the corrosion kinetics of cladding during normal operations.
(4) Calculate the extent of cladding oxidation, including pre-existing corrosion, during design-basis LOCAs accidents to show that residual ductility is retained.
(5) Assure that the duration of cladding exposure to high temperatures in excess of about 1073 K do not lead to "breakaway" oxidation and the absorption of hydrogen that will exacerbate embrittlement during clad cooling.
(6) Use the Cathcart-Pawel correlation for the analysis of the kinetics of steam oxidation of zirconium alloy cladding.

The proposed changes to the embrittlement criteria have a good technical foundation. They would be relatively easy to implement and would not result in major changes to current practices by either the licensee or the regulatory staff. The proposed changes are supported by several representatives of the nuclear industry.

To be sure, the proposed changes to the current regulatory requirements do eliminate reference to specific cladding materials. However, any particular requirements for qualification of cladding may well become anachronistic and burdensome as technology improves and technical understanding advances in the future. Utilization of reactor fuel to ever higher burnups yields both economic and societal benefits. Development of new cladding materials to facilitate this trend in fuel usage is anticipated to continue. Certainly, industry representatives have assured us that new cladding alloys are under development and will be introduced to the fuel market before current plant licenses and extended licenses expire. It would be better to revise the current regulations at a high level, emphasizing the safety needs for retention of coolability and core geometry without codifying methods for qualifying specific fuel claddings based on currently available clad materials and current understanding of the phenomena and processes that affect these materials. Methods acceptable to the staff for the qualification of specific reactor materials and reactor technologies can be developed in regulatory guides.

There is not an urgent safety issue prompting this recommendation for updating the regulations. Current discussions of other aspects of 10 CFR 50.46 make it opportune to consider proposed changes at the high level advocated here.

The database that supports the proposed steps for the qualification of zirconium-based cladding alloys is not extensive. Further investigations are warranted and are being proposed in the cooperative research effort being undertaken by RES and the nuclear power industry. We suggest also that the staff:

- consider requiring that clad oxidation and measurements of residual ductility be done with hydrogen-loaded cladding alloys that better replicate clad that has been exposed to normal operating conditions prior to an accident, and
- investigate the effects on ductility of clad cooling or quenching in tests versus the cooling rates in hypothesized design-basis LOCA.

Dr. William Shack did not participate in the Committee's deliberations regarding this matter.

Sincerely,

/RA/

Graham B. Wallis
Chairman

Reference:
R. Meyer, "Technical Basis for Revision of Embrittlement Criteria in 10 CFR 50.46," September 8, 2005 (Power Point Slides)

ISSUES RELATED TO NEW PLANT LICENSING (INCLUDING TECHNOLOGY-NEUTRAL FRAMEWORK)

Thomas S. Kress

SECY-05-0130: Two Policy Issues

- **Minimum level of safety to achieve enhanced safety for new plants**
- **How shall risk from multiple reactors at a single site be accounted for?**

Staff's Recommendations

- Minimum level of safety
  - Plants must meet QHOs by design
- Risk from multiple plants
  - Only new plants added to a site would be required to meet QHOs

ACRS Disagreed

- QHOs cannot be used as reactor design parameters
- Site risk status should include all sources of risk from the site – not just the new plants

ACRS Views

- **CDF and LRF are appropriate design safety parameters that can express an enhanced level of safety**
- **Site suitability should relate to all sources of risk from the site**
- **QHOs may not be sufficient. Should consider criterion to more directly address societal risk**

CDF and LRF for Modular Plants?

- **ACRS Was Of Two Minds**
  - **Modular designs should be viewed as a package**
  - **Each module should be treated as an individual reactor**

ACRS Highlighted Two Issues

- **Evaluation of risk for advanced reactor designs**
- **Consideration of societal risk acceptance measures**

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

September 21, 2005

The Honorable Nils J. Diaz
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: REPORT ON TWO POLICY ISSUES RELATED TO NEW PLANT LICENSING

Dear Chairman Diaz:

During the 523rd meeting of the Advisory Committee on Reactor Safeguards, June 1-3, 2005, we met with the NRC staff and discussed two policy issues related to new plant licensing. We also discussed this matter during our 524th, July 6-8, 2005, and 525th, September 8-10, 2005 meetings. We had the benefit of the documents referenced. These policy issues were:

- What shall be the minimum level of safety that new plants need to meet to achieve enhanced safety?

- How shall the risk from multiple reactors at a single site be accounted for?

In SECY-05-0130, the staff recommends that the expectation for enhanced safety be met by requiring that new plants meet the Quantitative Health Objectives (QHOs), i.e., by applying the QHOs to individual plants. The staff maintains that this would represent an enhancement in safety over current plants, which are now required to meet adequate protection, but may not meet the QHOs. The staff argues that this position is consistent with the Commission's Policy Statement on Regulation of Advanced Nuclear Power Plants.

The staff proposes to address the risk of multiple reactors at a single site by requiring that the integrated risk associated with only new reactors (i.e., modular or multiple reactors) at a site not exceed the risk expressed by the QHOs. The risk from existing plants, which may already exceed the QHOs, is not considered.

We discussed these issues and concluded that use of the existing QHOs is not sufficient to resolve either of these issues. In considering the overall scope of the issues raised by the staff, we found it more apt and effective to reframe the two issues into the following questions:

1. What are the appropriate measures of safety to use in the consideration of the certification of a new reactor design?

2. Should quantitative criteria for these measures be imposed to define the minimum level of safety?

3. How should these measures be applied to modular designs?

4. How should risk from multiple reactors at a site be combined for evaluation by suitable criteria?

5. How should the combination of new and old reactors at a site be evaluated by these criteria?

6. What should these criteria be?

7. How should compliance with these criteria be demonstrated?

DISCUSSION

Question 1. What are the appropriate measures of safety to use in the consideration of the certification of a new reactor design?

The QHOs are criteria for the risk at a site and thus involve not only the design and operation of the reactor(s), but also the site characteristics, the number and power level of plants on the site, meteorological conditions, population distribution, and emergency planning measures. By themselves, the QHOs do not express the defense-in-depth philosophy that the Commission seeks to limit not only the risk from accidents, but also the frequency of accidents.

Although core damage frequency (CDF) and large, early release frequency (LERF) have been viewed by the NRC as light water reactor (LWR)-specific surrogates for the QHOs, they have come to be accepted as metrics to gauge the acceptable level of safety of certified designs and the acceptability of proposed changes in the licensing basis. They are measures of reactor design safety that incorporate a defense-in-depth balance between prevention and mitigation. Currently used values of these metrics have been derived from the QHOs. If they were no longer to be viewed as surrogates, acceptance values for these metrics could be independently specified and need not be derived from the QHOs. Thus, they would be fundamental characteristics of reactor design independent of siting and emergency planning requirements.

If these measures are no longer viewed as surrogates for the QHOs, the appropriate measure of a large release need not be restricted to "early" but could be a "large release frequency" (LRF) which would apply to the summation of all large release frequencies regardless of the time of occurrence. The LRF would thus have broader applicability to designs in which the release is likely to occur over an extended period.

A majority of the Committee members favors the use of CDF and LRF as fundamental measures of the enhanced safety of new reactor designs and not simply as surrogates for the QHOs.

In SECY-05-0130, the staff argues that it will be difficult to derive such measures for different technologies, although the staff proposes to include them as subsidiary goals in their technology-neutral framework document. Although the processes and mechanisms for failure and release will differ greatly for different reactor technologies, technology-neutral definitions in terms of a release from the fuel (the accident prevention/CDF goal) and from the containment/ confinement (the large release goal) seem feasible to us. For example, the CDF of a Pebble Bed Modular Reactor (PBMR), would be an indicator of the success criteria for the design measures intended to prevent release from the fuel of that module. It could be defined in terms of the frequency of exceeding a fuel temperature of 1600 °C.

Question 2. Should quantitative criteria for these measures be imposed to define the minimum level of safety?

In the current Policy Statement on the Regulation of Advanced Nuclear Power Plants, the Commission decided not to set numerical criteria for enhanced safety but rather focused on aspects which might make designs more robust. In addition, the Safety Goal Policy Statement was intended to provide a definition of "how safe is safe enough." If a plant would meet the QHOs at a proposed site, then the additional risk it imposes is already very low compared to other risk in society. It now seems possible to build economically competitive reactors with risks at most sites that would be much lower than implied by the QHOs. The Electric Power Research Institute (EPRI) and European Utility Requirements Documents specify CDF and LERF values that would provide large margins to the QHOs for virtually all sites. An explicit commitment to lower values of CDF and LRF would be responsive to the Commission's desire for enhanced safety and may have significant impact on public perceptions and confidence.

We considered the following alternatives, identifying arguments in favor of each. Since such a decision has broad practical implementation and policy implications, we recommend that the staff further explore the consequences of these (and possibly other) choices as a basis for an eventual Commission decision.

a. Set maximum values for CDF and LRF at 10^{-5}/yr and 10^{-6}/yr for new reactor designs. This would make more explicit the Commission's stated expectation that future reactors provide enhanced safety. This could also provide a basis for establishing multinational design approval (as these would now be independent of U.S. QHOs). The suggested values are consistent with those in the EPRI and the European Utility Requirements Documents, the EPR Safety Document, and those used in the certification of advanced reactors (the ABWR, AP600 and CE-System 80+). These values are also consistent with the generic values for an accident prevention frequency and a LRF in the staff's draft technology-neutral framework document.

b. Leave the values unspecified. CDF and LRF would be considered along with other aspects of the design, such as defense-in-depth and passive safety features, in reaching a decision about design certification. This would give the staff more flexibility to respond to technology-specific features.

On a preliminary basis, the majority of the Committee members favor Alternative (a), but is not ready to make a recommendation until more is understood about the likely consequences and policy implications of the decision.

Question 3. How should these measures be applied to modular designs?

The staff's considerations of integrated risk do not distinguish between criteria for modular reactor designs and criteria for the risk due to multiple plants on a site. Thus, the staff treats CDF and LRF (or LERF) for modular designs and/or multiple plants on a site as still being QHO risk surrogates. In our view, the CDF and LRF metrics are design criteria that are to be "imposed" at the plant design certification stage independent of any site considerations.

New reactors could include PBMR, AP600, AP1000, Economic and Simplified Boiling Water Reactor (ESBWR), and EPR, and the number of new reactors at a site could vary by an order of magnitude.

Some Committee members believe that to get consistency in expectations of enhanced safety in all cases, the integrated risk from all new reactors on a site is the appropriate measure. This is true both for the risk metric LRF and the defense-in-depth accident prevention metric CDF. Thus, for the PBMR, which is proposed in terms of an eight-module package, the CDF and LRF goals (e.g., 10^{-5}/ry and 10^{-6}/ry) would be applied to the package. In effect each module would have to have a somewhat lower CDF and LRF. Because of the potential for interactions, analysis of individual modules may not be meaningful and the analysis should focus on the "eight pack."

Other Committee members prefer CDF and LRF design specifications that are independent of the number of modules. These members believe the specified acceptable CDF for enhanced safety (e.g. 10^{-5}/yr) should be applied to each module at the design stage and would be an indicator of the success criteria for the design measures provided for each module intended to prevent release from the fuel of that module. Similarly, LRF would be on a modular basis. As it may be possible to restrict the total power of a given module to a level that the quantity of fission products releasable cannot exceed the acceptance LRF value (e.g. 10^{-6}/yr), a modular design implicitly represents a kind of defense-in-depth (given appropriate consideration of common-mode failures and module interactions).

Question 4. How should risk from multiple reactors at a site be combined for evaluation by suitable criteria?

The QHOs address the risk to individuals that live in the vicinity of a site. Logically, the risk to these individuals should be determined by integrating the risk from all the units at the site. The manner by which the risks of different units at a site are to be integrated must address the treatment of modular designs, units with differing power levels, and accidents involving multiple units.

Question 5. How should the combination of new and old reactors at a site be evaluated by these criteria?

Any new plant that meets the independent safety criteria discussed in Questions 1 through 3 would be expected to add substantially less risk to an existing site than that already provided by existing plants on the site. If a proposed site already exceeds the QHOs, it should not be approved for new plants. For existing sites not being proposed for the addition of new plants, there would be no need to assess their risk status because they provide adequate protection. These sites would, thus, be grandfathered in the new framework.

Question 6. What should these criteria be?

Use of the QHOs for evaluating the site suitability for new reactors is attractive because the QHOs represent a fundamental statement about risk independent of any particular technology. The current QHOs (prompt and latent fatalities), however, only address individual risk and do not directly address societal risks such as total deaths, injuries, non-fatal cancers, and land contamination. These societal impacts are addressed somewhat in the current regulations by the siting criteria on population.

Some ACRS members believe that measures of societal risk need to be an explicit part of any new technology-neutral framework. The staff argues in the technology-neutral framework document that the limits proposed there for CDF and LRF limit societal risks such as land contamination and dose to the total population. However, these members recognize that CDF and LRF are not equivalent to risk and disagree with the staff's position.

Other ACRS members believe that the current siting criteria have served to limit societal risks. In addition, societal risks are considered in the environmental impact assessments of license renewal. The estimates presented in NUREG-1437 Vol. 1 indicate that the risk of early and latent fatalities from current nuclear power plants is small. The predicted early and latent fatalities from all plants (that is, the risk to the population of the United States from all nuclear power plants) is approximately one additional early fatality per year and approximately 90 additional latent fatalities per year, which is a small fraction of the approximately 100,000 accidental and 500,000 cancer fatalities per year from other sources. The evaluation of Severe Accident Mitigation Alternatives (SAMAs) as part of the license renewal process also considers societal risk measures and monetizes them to perform cost benefit studies. Based on current NRC regulatory analysis guidance, very few of these SAMAs appear cost beneficial.

Environmental impact statements (EISs) also assess the societal costs of probabilistic accidents at the current sites. The results, although very approximate, indicate that the societal costs at many current reactor sites would likely exceed a reasonable societal cost risk acceptance criterion. For example, these would exceed the cost associated with 0.1% of the above noted 100,000 early fatalities due to all accidents.

Thus, the inclusion of a quantitative societal risk acceptance measure appears important and could add to greater public confidence and understanding of the risks of nuclear power. It may be worthwhile for the staff to consider supplementing the current QHOs with additional risk acceptance measures that relate directly to societal risks.

7. *How should compliance with these criteria be demonstrated?*

The establishment of goals or criteria of various kinds cannot be divorced from the ability to demonstrate compliance. Considerable improvement in PRA practice will be needed to provide confidence that the goals on CDF and LRF for future plants will be met in a meaningful way. Operating experience has been crucial for the analysts to appreciate the significance of potential errors/faults. For example, before TMI, it was assumed that operators would not have problems diagnosing what is going on under certain conditions.

Some of the challenges that new plants will create for PRA analysts are:

i. Operating experience on component failure rate distributions and frequencies developed for light-water reactors has limited applicability to other reactor types.
ii. Some designs are considering components, e.g., microturbines and fuel cells, for which reliability data are nearly non-existent.
iii. Digital Instrumentation and Control systems are expected to be an integral part of future reactor designs. The risk consequences of such practice are difficult to quantify at this time.

Thus, in addition to the imposition of design goals for low CDF and LRF, it will be important to maintain sufficient defense-in-depth in the technology-neutral framework.

We look forward to additional discussion with the staff on these issues.

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

Additional comments from ACRS Members Dana A. Powers and John D. Sieber

We disagree with our colleagues on the matter of this letter. The Commission has indicated a laudable expectation that future reactors will be safer than current reactors. The question that our colleagues should have addressed first is whether a quantitative metric is needed to substantiate this expectation. It is by no means obvious that such a metric is essential. We can well imagine future plants designed in conjunction with far more comprehensive probabilistic safety analyses that realistically address all known accident hazards during all modes of operation to a depth far greater than is attempted now for elements of the fleet of operating reactors. Our experience has been that whenever improvements are made in quantitative risk analysis methods, unforeseen, hazardous, plant configurations, systems interactions and operations become apparent. Hidden, these configurations, interactions and operations may arise unexpectedly with undesirable consequences. Revealed, they can be avoided often with modest efforts. This is exploitation of the full potential of quantitative risk analysis to achieve greater safety in nuclear power plants. It contrasts with the more effete pursuit of the "bottomline" results of PRA to compare with arbitrarily proliferated safety metrics.

Our objective should be to foster the voluntary development of quantitative risk analysis methods both in scope and depth in order to improve the safety of nuclear power plants. Fostering voluntary development of methods by nuclear community is especially important now when methods developments have stagnated at NRC relative to the situation a decade ago.

Our colleagues seem to presume it essential that future reactors meet the Quantitative Health Objectives (QHOs). These QHOs define a very stringent safety level that has always been viewed as an "aiming point" or a benchmark and not as some minimum standard that cannot be exceeded. Indeed, the definition of the QHOs was undertaken to define "how safe is safe enough" so that no additional regulatory requirements for greater safety would be needed. Requiring such a stringent standard as the QHOs as a minimum level of safety for advanced reactors appears to go well beyond the authority granted by the Atomic Energy Act that requires adequate protection of the public health and safety. We are unaware that the Commission has made such a demand for advanced reactors. Were the Commission to make such a demand, we would question the wisdom of doing so. By demanding such a stringent level of safety, our colleagues appear to be willing to forego great strides in safety that can be achieved with advanced plants if these plants fail to live up to what can only be viewed as an extreme safety standard.

The demands our colleagues appear to make on the safety of advanced reactors lack a critical dimension of practicality since we do not believe the technology now exists to do the calculations needed to compare a plant's safety profile to the QHOs. By the very definitions of the QHOs, such calculations would entail analyses of modes of operation only very crudely addressed today by most (fire risk, shutdown risk and natural phenomena risk) and the conduct of uncertainty analyses dealing with both parameters and models that to our knowledge have been done by no one.

Because of the limitations of risk assessment technology available today for the evaluation of the current fleet of nuclear power plants, surrogate metrics such as core damage frequency (CDF) and large early release frequency (LERF) have been introduced and widely used. Our colleagues seem to believe that there are known critical values of these surrogate metrics that mark the point at which a plant meets the QHOs. We know of no defensible analysis that establishes such critical values of these surrogate metrics. We are, of course, quite aware of very limited analyses considering only risk during normal operations that purport to show existing reactors meet the QHOs. Such limited analyses are simply not pertinent. They do not meet the exacting standards required by the definitions of the QHOs. Should defensible analyses ever be done, we are sure that they will show the critical values of the surrogate metrics are technology dependent. Indeed, more defensible analyses will show in all likelihood that better surrogate measures can be defined for advanced reactor technologies.

Our colleagues are sufficiently enamored with the existing surrogate metrics that they recommend these surrogates be enshrined on a level equivalent to QHOs. More remarkable, our colleagues want to establish critical values of the metrics that are a factor of ten less than the values they assert mark a plant meeting the rather stringent level of safety defined by the QHOs. They do this, apparently, for no other reason than the fact that clever engineers can design plants meeting these smaller values at least for a limited number of operational states. While we are willing to congratulate the engineers on their designs, we can see no reason why such stringent safety requirements should be made regulatory requirements to be imposed on the designers' efforts. Again, we worry that doing so may create unnecessary burdens that cause our society to sacrifice for practical reasons great improvements in power reactor safety simply because these improvements fall short of our colleagues unreasonably high safety expectations.

Though surrogate metrics have been useful, it is important to remember that they are only expedients. The full promise of risk-informed safety assessment will not be realized until it is possible to do routinely risk assessments of sufficient scope and depth so it is possible to dispense with surrogate metrics. Enshrining these surrogates along with the QHOs will only delay efforts to reach this preferred status.

The potential of our colleagues recommendations have to stifle new technology and forego improved safety reaches a crisis when they speak to the location of modern, safer plants on sites with older but still adequately safe plants. Our colleagues have no tolerance for a single older plant if a newer, safer plant is to be collocated on the site. They are willing to tolerate any number of similarly old plants on a site if a new, safer plant is not added to this site. We find this remarkable. Our colleagues' recommendations give no credit for experience with a site. They fail to recognize the finite life of older plants even when licenses have been renewed. We fear that our colleagues have failed to assess the integral safety consequences of their stringent demands on this matter. A very great concern is that our colleagues pursuit of ideals in risk avoidance may well arrest the current, healthy quest for improved safety among those exploring advanced reactor designs.

<u>References:</u>

1. U.S. Nuclear Regulatory Commission, SECY-05-130," Policy Issues Related to New Plant Licensing and Status of the Technology Neutral Framework for New Plant Licensing," dated July 21, 2005

2. U.S. Nuclear Regulatory Commission, "Safety Goals for the Operations of Nuclear Power Plants, Policy Statement," Federal Register, Vol. 51, (51 FR 30028), August 4, 1986

3. U.S. Nuclear Regulatory Commission, "Commission's Policy Statement on the Regulation of Advanced Nuclear Power Plants," 59 FR 35461, July 12, 1994

4. U.S. Nuclear Regulatory Commission, NUREG-1437, Volume 1, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants," May 1996

FIRE PROTECTION MATTERS

George E. Apostolakis

Draft Final Reg. Guide On Fire Protection

- Should not be issued in its present form
- Demonstration of a success path should not be used as an alternative to estimates of ΔCDF and ΔLERF
- Staff should not endorse methods for evaluating ΔCDF and ΔLERF that are not based on fire PRA

NUREG/CR-6850 On EPRI/RES Fire PRA Methodology

- Issue NUREG/CR-6850
- Complete full-scope pilot fire PRAs
- Further identify fire PRA uncertainties

Post Fire Operator Manual Actions Rule

- The proposed rule would not satisfy the objective of significantly reducing the number of future exemption requests
- Agree with the staff's decision to withdraw the rule
- Alternative approach is to transition to a risk-informed fire protection program under 10 CFR 50.48(c)

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

June 14, 2005

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

SUBJECT: DRAFT FINAL REGULATORY GUIDE, "RISK-INFORMED, PERFORMANCE-BASED FIRE PROTECTION FOR EXISTING LIGHT-WATER NUCLEAR POWER PLANTS"

Dear Mr. Reyes:

During the 523rd meeting of the Advisory Committee on Reactor Safeguards, June 1-3, 2005, we met with representatives of the NRC staff to review the draft final Regulatory Guide, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," which endorses, with certain exceptions, the Nuclear Energy Institute (NEI) document NEI 04-02, "Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program Under 10 CFR 50.48 (c)." Our Subcommittee on Fire Protection met with representatives of the NRC staff and NEI on May 17, 2005 to review this matter. During these reviews, we had the benefit of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

1. The Regulatory Guide should not be issued in its present form.

2. The acceptability of changes in a fire protection program that is based on the National Fire Protection Association (NFPA) Standard 805 should be determined using methods consistent with Regulatory Guide 1.174. In particular:

   - The "initial fire modeling" approach should not be used as an alternative to estimates of changes in core damage frequency (ΔCDF) and large early release frequency (ΔLERF). Identification of a success path does not necessarily assure that ΔCDF and ΔLERF are negligible (Section 5.3.4.1 of NEI 04-02).

   - The staff should not endorse methods for evaluating ΔCDF and ΔLERF (Section 5.3.5.1 of NEI 04-02) that are not based on a fire probabilistic risk assessment (PRA).

3. NEI 04-02 contains many statements that are inconsistent with the Commission's policy of promoting the use of PRA methods. In the Regulatory Guide, the staff should make it clear that it does not endorse such statements.

4. The staff should ensure that the parts of NEI 04-02 that it endorses use correct methodology and language.

BACKGROUND

NFPA issued a performance-based standard for fire protection for light-water reactors (LWRs) in 2001 (NFPA 805). This standard specifies the minimum fire protection requirements for existing LWRs and offers the choice of a "deterministic" and a "performance-based" methodology for determining fire protection features and demonstrating that nuclear safety performance criteria are met.

Effective July 16, 2004, the Commission amended its fire protection requirements in 10 CFR 50.48 to add 10 CFR 50.48(c), which incorporates the 2001 edition of NFPA 805 by reference, with certain exceptions. Section 50.48(c) allows licensees to voluntarily adopt and maintain a fire protection program that meets the requirements of NFPA 805 as an alternative to meeting the requirements of 10 CFR 50.48(b). Adopting NFPA 805 requires the submission of a license amendment request to the NRC.

NEI has worked with representatives of the industry and the NRC staff to develop implementing guidance for the specific provisions of NFPA 805 and 10 CFR 50.48(c). In April 2005, NEI published this guidance as NEI 04-02, Revision 0. The Regulatory Guide endorses NEI 04-02, with certain exceptions, and offers guidance to licensees in meeting the Commission's requirements.

DISCUSSION

The Regulatory Guide endorses the guidance provided in NEI 04-02 regarding the transition to an NFPA 805-based fire protection program. This transition process is essentially deterministic. It "brings forward" a significant portion of the existing licensing basis to the new NFPA 805-based licensing basis and adds some new requirements, such as one for investigating fires occurring during non-power operational modes.

After this transition phase, NFPA 805 requires that any request for changes to the approved fire protection program be risk-informed. Paragraph 2.4.4 of NFPA 805 states: "The evaluation process shall consist of an integrated assessment of the acceptability of risk, defense-in-depth, and safety margins." Paragraph 2.4.3.1 states further that: "The PSA [probabilistic safety assessment] evaluation shall use core damage frequency (CDF) and large early release frequency (LERF) as measures for risk." Regulatory Guide 1.174 provides guidance and acceptability criteria for implementing a risk-informed approach to changes in the licensing basis, including changes in the fire protection program. This is acknowledged in Section 5.3.5 of NEI 04-02. However, NEI 04-02 deviates from Regulatory Guide 1.174 by appearing to allow:

- Demonstration of the existence of a success path as an alternative to an assessment of the change in risk (Section 5.3.4.1)
- Risk-informed judgments to be made about the acceptability of changes without a defensible assessment of the CDF and LERF of the plant (Section 5.3.5.1).

NEI 04-02 includes an approach based on the concepts of a Maximum Expected Fire Scenario and Limiting Fire Scenario (Figure 5-1 and Section 5.3.4.1). Figure 5-1 suggests that this approach is intended to simplify the calculation of ΔCDF and ΔLERF in some cases. The statement in Section 5.3.4.1 that "This approach eliminates the need for additional risk assessment because it effectively demonstrates that target damage does not occur and that a success path remains free of fire damage"[1] suggests that NEI 04-02 is confusing the identification of a success path with an estimate that ΔCDF and ΔLERF are small. Even when it can be demonstrated that a success path free of fire damage exists, a proposed change may result in ΔCDF and ΔLERF that exceed the guidelines in Regulatory Guide 1.174. The staff should state in the Regulatory Guide that it is unacceptable to interpret Section 5.3.4.1 of NEI 04-02 in a way that confuses the identification of a success path free of fire damage with a demonstration that ΔCDF and ΔLERF are small.

While the use of simplified calculations can be acceptable, the definitions of the Maximum Expected Fire Scenario and Limiting Fire Scenario in NFPA 805 and NEI 04-02 are sometimes contradictory and confusing. The Regulatory Guide should be revised to provide definitions of the Maximum Expected Fire Scenario and Limiting Fire Scenario that are acceptable.

Comparison of the Maximum Expected Fire Scenario and Limiting Fire Scenario is supposed to determine whether sufficient margin exists to assume that fire damage is negligible and therefore the change is acceptable. The Regulatory Guide should note that the definition of sufficient margin should include the uncertainties in the fire model being used in the analysis.

The staff should ensure that the parts of NEI 04-02 that it endorses use correct methodology and language. For example, Section 5.3.5.1 states: "If the ΔCDF satisfies the ΔLERF acceptance criteria, a specific assessment for ΔLERF is not required." This statement erroneously assumes that the relationship between ΔCDF and ΔLERF is the same as that between CDF and LERF. Another example of confused logic is the following: "If the fire-induced consequences do not disable the containment isolation function, then the ΔLERF criterion can be considered satisfied" (NEI 04-02, Section 5.3.5.1).

We look forward to reviewing the revised Regulatory Guide.

Sincerely,

/RA/

Graham B. Wallis
Chairman

---

[1] Statements such as this one are also inconsistent with the stated policy of the Commission that "the use of PRA technology in NRC regulatory activities should be increased to the extent supported by the state-of-the-art in PRA methods and data..."

REFERENCES

1. Regulatory Guide X.XXX, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," July 2005.

2. Nuclear Energy Institute (NEI), "Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program Under 10 CFR 50.48(c)," NEI 04-02, Revision 0, April 2005.

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

4. Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk Informed Decision on Plant-Specific Changes to the Licensing Basis," Revision 1, November 2002.

5. U.S. Nuclear Regulatory Commission, Final Policy Statement on Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities, Dated August 16, 1995.

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

July 7, 2005

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

/RA/

FROM: John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards

SUBJECT: PROPOSED GENERIC LETTER 2005-XX, "IMPACT OF POTENTIALLY DEGRADED HEMYC/MT FIRE BARRIER MATERIALS ON COMPLIANCE WITH APPROVED FIRE PROTECTION PROGRAMS"

During the 524th meeting of the Advisory Committee on Reactor Safeguards, July 6-8, 2005, the Committee considered the proposed Generic Letter 2005-XX, "Impact of Potentially Degraded HEMYC/MT Fire Barrier Materials on Compliance with Approved Fire Protection Programs." The Committee plans to review the draft final version of this Generic Letter after reconciliation of public comments. The Committee has no objection to the staff's proposal to issue the proposed Generic Letter for public comment.

Reference:
Memorandum dated June 9, 2005, from Brian W. Sheron, Associate Director for Project Licensing and Technical Analysis, NRR, to John T. Larkins, Executive Director, ACRS, Subject: Request for the Advisory Committee on Reactor Safeguards (ACRS) to Defer Initial Review of the Proposed Draft Generic Letter Entitled, "Impact of Potentially Degraded HEMYC/MT Fire Barrier Materials on Compliance with Approved Fire Protection Programs," (ADAMS Accession No. ML051510350).

cc:
A. Vietti-Cook, SECY
W. Dean, OEDO
J. Dixon-Herrity, OEDO
J. Lyons, NRR
J. Hannon, NRR
S. Weerakkody, NRR
D. Frumpkin, NRR
A. Markley, NRR
A. Lavretta, NRR
M. Crutchley, NRR

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

June 10, 2005

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

SUBJECT: DRAFT FINAL NUREG/CR-6850, "EPRI/NRC-RES FIRE PRA METHODOLOGY FOR NUCLEAR POWER FACILITIES"

Dear Mr. Reyes:

During the 523rd meeting of the Advisory Committee on Reactor Safeguards, June 1-3, 2005, we met with representatives of the NRC staff and Electric Power Research Institute (EPRI) to discuss the draft final NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities." Our Subcommittee on Fire Protection also reviewed this matter during its meeting on May 4, 2005. During our review, we had the benefit of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

1. NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities," will be useful to both the industry and the staff and should be issued.

2. Full-scope pilot fire probabilistic risk assessments (PRAs) based on the procedures and methods in NUREG/CR-6850 should be completed, and the insights provided by these applications should be used to enhance the methodology.

3. Efforts should continue to further identify, quantify, and document remaining fire PRA uncertainties.

DISCUSSION

The NRC Office of Nuclear Regulatory Research (RES) and EPRI have completed a cooperative program to consolidate the fire PRA research and development activities, conducted over the past few years, into a single state-of-the-art methodology for fire PRA. The results, documented in NUREG/CR-6850, provide a structured framework for the overall analysis as well as specific recommended practices to address key aspects of the analysis. This work was conducted under the terms of an EPRI/RES memorandum of understanding and an accompanying fire research addendum.

While the primary objective of the project was to consolidate state-of-the-art methods, in many areas the newly documented methods represent a significant advancement over those previously documented. Several new methods and approaches were developed. These methods specifically address and resolve previously identified methodological issues. The participants should consider publication of some of the more innovative material in appropriate archival journals.

At some nuclear plants, risk from fire-initiated accidents is commensurate with risk from internal events. Despite the valuable contribution and advances in fire risk analysis described in NUREG/CR-6850, the body of knowledge and the tools supporting fire risk analysis are still not comparable with the state-of-the-art PRAs for internal events. Further development of fire PRA methods is needed. Ultimately, internal events and fire PRAs should be integrated.

Industry participants provided an extensive peer review of the project. A peer-review panel was formed from the six nonpilot utility participants. Two nuclear plants participated as pilot plants and supported demonstration studies conducted by the technical development teams. RES and EPRI intended that these demonstration studies would be complemented by full-scope fire PRAs at the pilot plants. Neither of the two pilot plants has completed its fire PRA. This represents a missed opportunity to gain experience with the procedures and new approaches in NUREG/CR-6850. Full-scope pilot fire PRAs based on the procedures and methods in NUREG/CR-6850 should be completed, and the insights provided by these applications should be used to enhance the methodology.

We have often emphasized the need for thorough uncertainty analyses to support licensee and regulatory decisionmaking. NUREG/CR-6850 prescribes methods for conducting these analyses as part of fire PRAs. Appendix V to Chapter 15 identifies uncertainty issues associated with each task in the methodology for conducting a fire PRA and suggests a strategy for addressing these uncertainties. While the uncertainties in fire ignition frequencies and post-fire human reliability will be quantified, many of the other uncertainties are to be relegated to a quality review rather than elucidated and made visible by estimation or analysis. Although a reasonable attempt has been made to require the identification of the key sources of uncertainty, efforts should continue to develop new approaches to further identify, quantify, and document the remaining uncertainties.

A formal issue resolution process was incorporated into the project to ensure that divergent technical views were fully considered. Although EPRI or RES could have maintained separate positions, no such cases were encountered, and consensus was reached. NUREG/CR-6850 will be useful to both the industry and the staff. We commend the organizations and the individuals involved in the preparation of this document.

Sincerely,

/RA/

Graham B. Wallis
Chairman

REFERENCES

1. EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities: Vol I: Summary and Overview, Electric Power Research Institute(EPRI), Palo Alto, CA, and U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research (RES), Rockville, MD, EPRI-TR-1008239 and NUREG/CR-6850, Draft Final, April 2005.

2. EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities: Vol 2: Detailed Methodology, Electric Power Research Institute(EPRI), Palo Alto, CA, and U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research (RES), Rockville, MD, EPRI-TR-1008239 and NUREG/CR-6850, Draft Final. April 2005.

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

November 18, 2005

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

SUBJECT: STAFF RECOMMENDATION TO WITHDRAW THE PROPOSED RULE ON POST-FIRE OPERATOR MANUAL ACTIONS

Dear Chairman Diaz:

During the 527th meeting of the Advisory Committee on Reactor Safeguards, November 3-5, 2005, we discussed the staff's recommendation to withdraw the proposed rule on post-fire operator manual actions. During our review, we had the benefit of discussions with representatives of the staff and the Nuclear Energy Institute. We also had the benefit of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

- The proposed rule on post-fire operator manual actions would not satisfy the objective of significantly reducing the number of future exemption requests.

- We concur with the staff's decision to withdraw the proposed rule.

DISCUSSION

A proposed rule that would modify Appendix R of 10 CFR 50 to include the regulation of post-fire operator manual actions was issued for public comment on March 7, 2005. After evaluating the public comments, the staff concluded that the final rule would not achieve the objective of reducing the number of exemption evaluations required and that it should be withdrawn.

Section III.G of Appendix R provides requirements that assure the protection of at least one path of achieving safe shutdown during a fire at any location in the plant. Plants that received their licenses after 1979 are not subject to Appendix R, but comply with similar requirements. Because the plants to which Appendix R applies were constructed in the absence of standards addressing separation and protection, some fire areas contain equipment from more than one safe shutdown train. Section III.G.2 of Appendix R identifies three alternative means of protecting at least one train of safe shutdown equipment within a fire area:

- A 3-hour rated fire barrier (for fire areas outside containment)
- Separation by at least 20 feet with no intervening material, in combination with fire detection and automatic fire suppression equipment
- Enclosure of one train of equipment with a 1-hour rated fire barrier, in combination with detection and automatic fire suppression equipment.

Some plants have had difficulty in complying with Section III.G.2 and have sought exemptions in which operator manual actions compensate for an inability to satisfy one of the alternatives. Some plants relied on compensatory operator manual actions without receiving regulatory approval. To achieve compliance, either plants can obtain exemption from Section III.G.2 requirements or the requirements can be modified by rulemaking to cover those conditions for which manual actions represent an acceptable alternative. The staff developed the proposed rule for this purpose.

In our letter dated November 19, 2004, we recommended that the draft rule be published for public comment. In approving publication of the proposed rule, the Commission directed the staff to "engage stakeholders to get a clear understanding of the likelihood that the proposed rule would achieve its underlying purpose, including the number of plants for which the proposed rule would address the operator manual actions issue. This information should be considered in deciding whether to proceed to final rulemaking."

Comments were received from the public, licensees, and the Nuclear Energy Institute. Based on its evaluation of the comments, the staff has concluded that the proposed rule would not lead to a significant reduction in the number of exemption requests. We concur with the staff's recommendation to withdraw the proposed rule.

In the absence of the final rule, the staff will proceed with enforcement of the existing regulations and the case-by-case resolution of exemption requests. An alternative available to licensees is to transition to a risk-informed fire protection program under 10 CFR 50.48(c). Appendix R sets forth an established deterministic approach for assuring the ability to safely shut down a nuclear plant during a fire. However, when a licensee seeks an exemption from Appendix R, risk insights may be useful to determine that adequate safety is preserved.

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References:
1. Memorandum from J. Lyons, NRR, to J. Larkins, ACRS, dated October 28, 2005, "Proposed Withdrawal of Rulemaking Allowing Use of Post-Fire Operator Manual Actions," (ADAMS Accession No. ML052970102).

2. Letter from M. Bonaca, ACRS, to N. Diaz, Chairman, dated November 19, 2004, "Draft Proposed Rule on Post-Fire Operator Manual Actions," (ADAMS Accession No. ML043240215).

3. Memorandum from E. Merchoff acting for EDO, to M. Bonaca, ACRS, dated December 22, 2004, "Draft Proposed Rule on Post-Fire Operator Manual Actions," (ADAMS Accession No. ML043380177).

4. Staff Requirements, SECY-04-0233 - Proposed Rulemaking - Post-Fire Operator Manual Actions, January 18, 2005.

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

July 25, 2005

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

/RA/

FROM: John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards

SUBJECT: PROPOSED GENERIC LETTER 2005-XX, "POST-FIRE SAFE-SHUTDOWN CIRCUIT ANALYSIS SPURIOUS ACTUATIONS"

The Advisory Committee on Reactor Safeguards (ACRS) members considered the proposed Generic Letter 2005-XX, "Post-Fire Safe-Shutdown Circuit Analysis Spurious Actuations." They plan to review the draft final version of this Generic Letter after reconciliation of public comments. The ACRS members have no objection to the staff's proposal to issue the proposed Generic Letter for public comment.

Reference:
Memorandum dated July 5, 2005, from Brian W. Sheron, Associate Director for Project Licensing and Technical Analysis, NRR, to John T. Larkins, Executive Director, ACRS, Subject: Request for the Advisory Committee on Reactor Safeguards (ACRS) to Defer Initial Review of the Proposed Draft Generic Letter Entitled, "Post-Fire Safe-Shutdown Circuit Analysis Spurious Actuations," (ADAMS Accession No. ML051590450).

cc: A. Vietti-Cook, SECY
W. Dean, OEDO
J. Dixon-Herrity, OEDO
J. Lyons, NRR
J. Hannon, NRR
S. Weerakkody, NRR
R. Wolfgang, NRR
A. Markley, NRR
C. Patel, NRR
M. Crutchley, NRR

POWER UPRATE TECHNICAL ISSUES

Richard S. Denning

Containment overpressure credit

- **Credit is sought when net positive suction head is not sufficient to prevent pump cavitation**
- **ACRS maintains position that credit should only be granted on case by case**

Large Transient Testing

- Turbine trip
- MSIV closure
- Specified by GE topical report
- Typically an exception is granted
- Should be performed unless adequate justification is provided

- <u>**Risk Issues**</u>
- **LERF treatment does not address increased source term**
- **Risk impact of reduced thermal margins is difficult to assess**
- **Reduced time for operator action**

Increased Flow Effects

- Flow accelerated corrosion
- Steam dryer vibration and damage
- Others

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

September 20, 2005

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

SUBJECT: PROPOSED REVISION 4 TO REGULATORY GUIDE 1.82, "WATER SOURCES FOR LONG-TERM RECIRCULATION COOLING FOLLOWING A LOSS-OF-COOLANT ACCIDENT"

Dear Mr. Reyes:

During the 525th meeting of the Advisory Committee on Reactor Safeguards, September 8-10, 2005, we reviewed the proposed Revision 4 to Regulatory Guide (RG) 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," and the supporting Standard Review Plan (SRP) Section 6.2.2, "Containment Heat Removal Systems." The review focused mainly on the issue of granting containment overpressure credit for calculation of net positive suction head (NPSH) for emergency core cooling and containment heat removal system pumps. During our review, we had the benefit of presentations by and discussion with representatives of the NRC staff and members of the public. We also had the benefit of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

1. Revision 4 to RG 1.82 should not be issued for public comment at this time and should be revised to improve clarity and reflect the following recommendation.

2. Containment overpressure credit to ensure sufficient NPSH for emergency core cooling and heat removal system pumps should only be selectively granted.

DISCUSSION

One purpose of the proposed Revision 4 to RG 1.82 is to make it consistent with current regulatory practice for crediting containment accident pressure in calculating available NPSH for boiling water reactor (BWR) and pressurized water reactor (PWR) systems. As a part of this effort, SRP Section 6.2.2 would also be revised to reference RG 1.82 rather than RG 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps." RG 1.1 would be designated as applicable only to those plants for which it was used as the basis for the original license.

RG 1.82 was first issued in 1974 to provide guidance on the design of PWR sumps which serve as a source of water during the recirculation core cooling phase of postulated design-basis loss-of-coolant accidents (LOCAs). Three revisions to RG 1.82 have been issued, one in November 1985, another in May 1996, and the most recent in November 2003. These revisions have addressed issues associated with containment emergency sump performance, particularly debris blockage on the emergency core cooling system suction strainers and granting credit for containment overpressure in determining NPSH available for the emergency core cooling and containment heat removal pumps.

Even though containment overpressure credit had been granted on an ad hoc basis before RG 1.1 was issued in 1974, Revision 3 to RG 1.82 issued in November 2003 was the first version to provide explicit guidance for granting limited use of containment accident pressure for calculating available NPSH. This guidance conflicts with the original guidance in RG 1.1, still in effect, which states that no such credit should be used. Not granting credit preserves the independence of the performance of the ECCS and containment systems.

The proposed Revision 4 to RG 1.82 includes provisions that permit licensees to use either a conservative deterministic approach or a best estimate with uncertainty analysis to establish the amount of containment overpressure to be credited.

We previously stated our position on granting containment overpressure credit in our December 12, 1997 letter (i.e., "selectively granting credit for small amounts of overpressure for a few cases may be justified") and more recently in our letter dated September 30, 2003. In that letter we recommended issuing Revision 3 to RG 1.82. That RG included a provision to grant, only where necessary, some containment accident pressure credit for some operating reactors with the caveat that "this should be minimized to the extent possible."

The position that the overpressure should be conservatively calculated is the only explicit restriction on the use of overpressure credit given in the proposed revision of the RG. In addition, the guidance describing what factors to consider in conservatively calculating containment overpressure, in Sections 1.3.1 and 2.1.1 of the proposed RG is confusing.

We believe that additional restrictive guidance should be placed on the granting of overpressure credit. Before such credit can be granted, licensees should demonstrate that there are no practical alternative approaches that can eliminate the need for such credit. Such credit should be granted only for robust containments for which there are positive means for indication of containment integrity such as inerted and sub-atmospheric containments. The time intervals for which such credit is needed should be limited to a few hours, commensurate with the demonstrated capability of all associated equipment to perform its intended functions during this time period. The RG should be revised to include such restrictions before it is released for public comment.

Dr. William Shack did not participate in the Committee's deliberations regarding this matter.

Sincerely,

/RA/

Graham B. Wallis
Chairman

References:

1. Letter from Suzanne Black to John Larkins, "Proposed Revision to Regulatory Guide 1.82, Revision 3, "Water Sources for Long-term Recirculation Cooling Following a Loss-of-Coolant Accident (LOCA)", June 3, 2005
2. Letter from James E. Lyons to John Larkins, "Proposed Revision to Regulatory Guide 1.82, Revision 3, "Water Sources for Long-term Recirculation Cooling Following a Loss-of-Coolant Accident (LOCA)", September 6, 2005
3. Letter from David O'Brien to Mario Bonaca, "State of Vermont Request to Consider the Containment Overpressure Credit Policy", September 17, 2004
4. B. R. Hobbs, et. al., "Vermont Yankee Extended Power Uprate Feasibility Study", June 28, 2002
5. "Learning about Pump NPSH Margin", http://www.pumps.org/public/pump resources, February 28, 2005
6. T. Henshaw, "How Much NPSH Does Your Pump Really Require?", www.pump-zone.com, September 2001, page 42
7. P. Cooper, et. al., "Checking In,", www.pump-zone.com, January, 2002, p. 8
8. R. Lueneberg, Sulzer-Bingham Pumps Inc., "NPSH/Minimum Flow Study - Summary, F-97-10782(30P59)", May 1, 1998
9. L. Lukens, "MSIV As-Found LLRTs Show An Adverse Trend - Adverse Trend Common Cause Analysis", CR-VTY-2004-0918, May 5, 2004

Abbreviations

ACRS	Advisory Committee on Reactor Safeguards
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
CY	Calendar Year
ECCS	Emergency Core Cooling System
EPRI	Electric Power Research Institute
EPU	Extended Power Uprate
ESP	Early Site Permit
GALL	Generic Aging Lessons Learned Report
GE	General Electric
GSI	Generic Safety Issue
I&C	Instrumentation and Control
LERF	Large Early Release Frequency
LOCA	Loss-of-Coolant-Accident
LRF	Large Release Frequency

Abbreviations

MSIV	Main Stream Isolation Valve
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
PRA	Probabilistic Risk Assessment
PWR	Pressurized Water Reactor
QHO	Quantitative Health Objective
RES	Office of Nuclear Regulatory Research
RG	Regulatory Guide
RS	Review Standard
SPAR	Standardized Plant Analysis Risk Model
SRP	Standard Review Plan
U.S.	United States