ABSTRACT

This compilation contains 57 Advisory Committee on Reactor Safeguards reports submitted to the U. S. Nuclear Regulatory Commission (NRC), or to the NRC Executive Director for Operations, during calendar year 2006. All reports have been made available to the public through the NRC Public Document Room, the U. S. Library of Congress, and the Internet at http://www.nrc.gov/reading-rm/doc-collections. The reports are organized in chronological order.

******

NUREG-1125
Volume 28

A Compilation of
Reports of

The Advisory Committee on Reactor Safeguards

2006 Annual

U.S. Nuclear Regulatory Commission

June 2007

PREFACE

The enclosed reports, issued during calendar year 2005, contain the recommendations and comments of the U. S. NRC's Advisory Committee on Reactor Safeguards on various regulatory matters. NUREG-1125 is published annually. Previous issues are as follows:

Volume	Inclusive Dates
1 through 6	September 1957 through December 1984
7	Calendar Year 1985
8	Calendar Year 1986
9	Calendar Year 1987
10	Calendar Year 1988
11	Calendar Year 1989
12	Calendar Year 1990
13	Calendar Year 1991
14	Calendar Year 1992
15	Calendar Year 1993
16	Calendar Year 1994
17	Calendar Year 1995
18	Calendar Year 1996
19	Calendar Year 1997
20	Calendar Year 1998
21	Calendar Year 1999
22	Calendar Year 2000
23	Calendar Year 2001
24	Calendar Year 2002
25	Calendar Year 2003
26	Calendar Year 2004
27	Calendar Year 2005

ACRS MEMBERSHIP (2006)

CHAIRMAN:	Dr. Graham B. Wallis Dartmouth College
VICE CHAIRMAN:	Dr. William J. Shack Argonne National Laboratory
MEMBERS:	Mr. John D. Sieber, Retired Duquesne Light Company
	Dr. George E. Apostolakis, Professor Massachusetts Institute of Technology
	Dr. J. Sam Armijo University of Nevada, Reno
	Dr. Sanjoy Banerjee University of California, Santa Barbara
	Dr. Mario V. Bonaca, Retired Northeast Utilities
	Dr. Michael L. Corradini University of Wisconsin
	Dr. Said Abdel-Khalik Georgia Institute of Technology
	Dr. Thomas S. Kress, Retired Oak Ridge National Laboratory
	Mr. Otto L. Maynard Kansas Gas & Electric Company (Retired)
	Dr. Dana A. Powers Sandia National Laboratories

TABLE OF CONTENTS

Page

TABLE OF CONTENTS (CONT'D)

Page

TABLE OF CONTENTS (CONT'D)

Page

TABLE OF CONTENTS (CONT'D)

Page

TABLE OF CONTENTS (CONT'D)

Page

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

February 14, 2006

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: DRAFT FINAL REVISION 2 TO REGULATORY GUIDE 1.92, "COMBINING MODAL RESPONSES AND SPATIAL COMPONENTS IN SEISMIC RESPONSE ANALYSIS"

During the 529th meeting of the Advisory Committee on Reactor Safeguards, February 9-11, 2006, the Committee considered the draft final Revision 2 to Regulatory Guide 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis." The Committee decided not to review this document and has no objection to the staff's proposal to issue this Guide.

Reference:
Memorandum dated January 13, 2006, from Carl J. Paperiello, Director, Office of Nuclear Regulatory Research, to John T. Larkins, Executive Director, ACRS, Subject: Request for ACRS Review of Regulatory Guide 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis," Revision 2.

cc: A. Vietti-Cook, SECY
W. Dean, OEDO
J. Dixon-Herrity, OEDO
M. Cunningham, RES
M. Evans, RES
A. Hsia, RES
T. Chang, RES
R. Assa, RES
T. Meek, NRR

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

February 14, 2006

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: PROPOSED REVISIONS TO REGULATORY GUIDES REGARDING ASME CODE CASES

During the 529th meeting of the Advisory Committee on Reactor Safeguards, February 9-11, 2006, the Committee considered proposed revisions to the following Regulatory Guides and decided not to review them:

- Proposed Revision 34 of Regulatory Guide 1.84 (DG-1133), "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III, Division 1"
- Proposed Revision 15 of Regulatory Guide 1.147 (DG-1134), "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1"
- Proposed Revision 2 of Regulatory Guide 1.193 (DG-1135), "ASME Code Cases Not Approved for Use"

The Committee has no objection to the staff's proposal to issue these documents for public comment.

The Committee would prefer to review proposed Regulatory Guides and any associated rulemaking as a package.

Reference:
Memorandum dated January 10, 2006, from Mark A. Cunningham, Director, Division of Engineering Technology, to John T. Larkins, Executive Director, ACRS, Subject: Request for ACRS Review of Draft ASME Code Case Regulatory Guides

cc: A. Vietti-Cook, SECY
W. Dean, OEDO
J. Dixon-Herrity, OEDO
M. Cunningham, RES
J. Uhle, RES
D. Jackson, RES
W. Norris, RES
R. Assa, RES
T. Meek, NRR

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

February 22, 2006

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

SUBJECT: STANDARD REVIEW PLAN, SECTION 14.2.1, "GENERIC GUIDELINES FOR EXTENDED POWER UPRATE TESTING PROGRAMS"

Dear Mr. Reyes:

During the 529th meeting of the Advisory Committee on Reactor Safeguards, February 9-10, 2006, we reviewed the Standard Review Plan (SRP) Section 14.2.1, "Generic Guidelines for Extended Power Uprate Testing Programs." During our review, we had the benefit of discussions with representatives of the NRC staff and the documents referenced.

RECOMMENDATION

Paragraph III.c. of SRP Section 14.2.1 should be rewritten to provide more structured and explicit guidance defining those conditions under which large transient tests would be exempted or required.

BACKGROUND AND DISCUSSION

The staff has revised SRP Section 14.2.1, making changes that are largely editorial. However, in the review of the extended power uprate (EPU) applications, it has become apparent that more clearly defined criteria are needed to specify those conditions under which it is acceptable to exempt a plant from performing a large transient test. A similar comment was made in our September 24, 2003 report to Chairman Diaz regarding the "Draft Final Review Standard for Extended Power Uprates, RS-001" in which we stated that "the criteria for integral system transient testing were vague." SRP Section 14.2.1 properly identifies the factors that would support such a decision but does not provide explicit guidance on how the decision should be made.

Large transient tests have specific objectives. They are conducted not only to test the performance of individual components and structures but also the integrated response of the system, including its control functions. Because large transient tests impose substantial hydrodynamic and thermal loads on the plant, they have associated risks and impacts on the plant. Although these risks are not high, it is appropriate to exempt the licensee from performing these tests if they provide essentially no benefit. Conversely, transient tests can identify the unexpected. It would be preferred to uncover problems during a controlled test, rather than under the conditions of an unplanned transient.

Draft SRP Section 14.2.1 identifies seven factors to be considered in determining whether a licensee should be exempted from performing a test. Although these are the appropriate factors to be considered, more explicit guidance should be provided to the reviewer as a basis for decisionmaking. Section III.c. should be rewritten to provide more structured and explicit guidance defining those conditions under which large transient tests would be exempted or required. We would like to be kept informed of the changes to SRP Section III.c. to address our concern.

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References:

1. Memorandum from D. Thatcher, NRR, to J. Larkins, ACRS, dated January 18, 2006, Subject: Request for the Advisory Committee on Reactor Safeguards Final Review of the Standard Review Plan 14.2.1, "Generic Guidelines for Extended Power Uprate Testing Programs"

2. Standard Review Plan (SRP) 14.2.1, "Generic Guidelines for Extended Power Uprate Testing Programs," Rev. 1 - XXXX 2005 (ADAMS Accession No. ML051100780)

3. Letter from M. Bonaca, ACRS, to N. Diaz, Chairman, dated September 24, 2003,Subject: Draft Final Review Standard for Extended Power Uprates, RS-001 (ADAMS Accession No. ML032681204)

4. Memorandum from J. Larkins, ACRS, to L. Reyes, EDO, dated November 9, 2005 Subject: Standard Review Plan, Section 14.2.1, "Generic Guidelines for Extended Power Uprate Testing Programs," (ADAMS Accession No. ML053170009)

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

February 22, 2006

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

SUBJECT: DRAFT NUREG REPORT, "EVALUATION OF HUMAN RELIABILITY ANALYSIS METHODS AGAINST GOOD PRACTICES"

Dear Mr. Reyes:

During the 529th meeting of the Advisory Committee on Reactor Safeguards, February 9-10, 2006, we met with representatives of the NRC staff to discuss the draft NUREG report, "Evaluation of Human Reliability Analysis Methods Against Good Practices" (Reference 1). Our Subcommittees on Reliability & Probabilistic Risk Assessment and Human Factors also discussed this report with the staff during a joint meeting on December 15-16, 2005. We also had the benefit of the documents referenced.

RECOMMENDATION

The draft NUREG report, "Evaluation of Human Reliability Analysis Methods Against Good Practices," should be issued for public comment.

DISCUSSION

Probabilistic risk assessment (PRA) guidance documents are essential to implementing the Commission's phased approach to PRA quality. The American Society of Mechanical Engineers PRA standard (Reference 2) and Regulatory Guide 1.200 (Reference 3) provide high-level guidance on what items should be addressed in a PRA without specifying methods for implementation. This lack of specific guidance is particularly acute in the area of human reliability analysis (HRA), especially for human actions under accident conditions where several models are being used by various groups. An early benchmark exercise by the European Commission's Joint Research Centre at Ispra showed substantial variability in the results produced by the same group of analysts using different HRA models, as well as substantial variability in the results from the same model used by different teams (Reference 4).

At the present time, there is no documented systematic evaluation of the assumptions, strengths, and weaknesses of the many HRA models. The staff is remedying this situation in two phases. First, a document was prepared to identify a set of good practices (Reference 5). HRA analysts should follow those practices regardless of the particular model used. In the second phase, several HRA methods were reviewed and evaluated against these good practices. These are documented in the draft NUREG report. This review is limited to models used in the United States, although the staff plans to expand its review to include international methods during the next round of evaluations.

The purpose of the draft NUREG report is to aid reviewers of HRAs in evaluating analyses submitted to the NRC. Since the report highlights the strengths, limitations, and bases of various commonly applied HRA models, it should also be useful to analysts preparing HRAs and other submittals requiring considerations of human performance.

The staff and its contractors performed most of the evaluations, but arranged for outside experts to evaluate models developed under NRC sponsorship (ATHEANA, SPAR-H, and SLIM/FLIM) in order to get a more objective assessment. We commend the staff for this action.

The draft NUREG report is an important step toward improving the consistency and quality of the application of HRA. Including the evaluations of several models against a common set of criteria in one document will be very useful to future work on the resolution of the significant model uncertainties that now exist in HRA. The report should be issued for public comment. We plan to review the draft final report after resolution of public comments.

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References:

1. Memorandum from Charles E. Ader, Director, Division of Risk Analysis and Applications, Office of Nuclear Regulatory Research, to John T. Larkins, Executive Director, Advisory Committee on Reactor Safeguards, "Transmittal of the Draft Report 'Evaluation of Human Reliability Analysis Methods Against Good Practices'," January 12, 2006.

2. "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," ASME RA-S-2002 (including the Addenda to Standard RA-SA-2003), American Society of Mechanical Engineers, April 5, 2002.

3. "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Regulatory Guide 1.200 For Trial Use, U.S. Nuclear Regulatory Commission, Washington, DC, February 2004.

4. A. Poucet, "The European Benchmark Exercise on Human Reliability Analysis," Proceedings of the American Nuclear Society International Topical Meeting on Probability, Reliability, and Safety Assessment (PSA '89), Pittsburgh, PA, April 2-7, 1989, pp. 103-110.

5. "Good Practices for Implementing Human Reliability Analysis", NUREG-1792, US Nuclear Regulatory Commission, Washington, DC, 2005.

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

March 15, 2006

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

SUBJECT: REVIEW AND EVALUATION OF THE NRC SAFETY RESEARCH PROGRAM

Dear Chairman Diaz:

Enclosed is an advance copy of the 2006 ACRS report entitled, "Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program." This report presents the Committee's observations and recommendations concerning the NRC Safety Research Program. The final report will be issued as NUREG-1635, Vol. 7.

This report focuses on that portion of the NRC research program dealing with the safety of existing nuclear reactors and advanced light water reactor designs, such as ESBWR. In its review of the NRC research activities, the Committee considered the programmatic justification for the research as well as the technical approaches and progress of the work. This review attempts to identify research crucial to the NRC mission. This review also attempts to identify research activities that have progressed sufficiently to meet current and anticipated regulatory needs so that they can be curtailed in favor of more important activities. The report does not address research on the security of nuclear power plants. Comments on such research will be reported separately. The report also does not comment on research conducted in support of the regulatory activities of the Office of Nuclear Material Safety and Safeguards. The Advisory Committee on Nuclear Waste (ACNW) will report on these research activities.

As agreed to by the Commission, the ACRS will provide its next report to the Commission on the overall NRC Safety Research Program in March 2008.

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

Enclosure: As stated
[Final version attached]

NUREG-1635, Vol. 7

Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program

A Report to the U.S. Nuclear Regulatory Commission

March 2006

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

ABSTRACT

This report to the U.S. Nuclear Regulatory Commission (NRC) presents the observations and recommendations of the Advisory Committee on Reactor Safeguards (ACRS) concerning the NRC Safety Research Program being carried out by the Office of Nuclear Regulatory Research (RES). These observations and recommendations focus on that portion of the NRC research program dealing with the safety of existing nuclear reactors and advanced light water reactor designs, such as the Economic Simplified Boiling Water Reactor (ESBWR) submitted for certification. The research strategy for more advanced reactors that are not based on water reactor technology such as the Generation IV reactors being studied by the Department of Energy is also discussed. In its evaluation of the NRC research activities, the ACRS considered the programmatic justification for the research as well as the technical approaches and progress of the work. The evaluation identifies research crucial to the NRC missions. The ACRS also attempts to identify research that had progressed sufficiently to meet current and anticipated regulatory needs so that it could be curtailed in favor of more important activities. This report does not address research on the security of nuclear power plants. Comments on such research will be reported separately. Also, the ACRS does not comment on research activities dealing with nuclear waste issues. The Advisory Committee on Nuclear Waste (ACNW) will report on these research activities.

TABLE OF CONTENTS

TABLES

ABBREVIATIONS

ACRS	Advisory Committee on Reactor Safeguards
ACNW	Advisory Committee on Nuclear Waste
ACR-700	Advanced CANDU Reactor-700
ANL	Argonne National Laboratory
ANS	American Nuclear Society
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
ASP	Accident Sequence Precursor
ATHEANA	A Technique for Human Event Analysis
ATWS	Anticipated Transients Without Scram
BWR	Boiling Water Reactor
CAMP	Code Applications and Maintenance Program
CFR	Code of Federal Regulations
CRDM	Control Rod Drive Mechanism
CSARP	Cooperative Severe Accident Research Program
DOE	Department of Energy
ECCS	Emergency Core Cooling System
EMI	Electro Magnetic Interference
EPIX	Equipment Performance and Information Exchange System
EPR	Evolutionary Power Reactor
EPRI	Electric Power Research Institute
ESBWR	Economic Simplified Boiling Water Reactor
FY	Fiscal Year
GDC	General Design Criterion
GSI	Generic Safety Issue
HERA	Human Event Repository and Analyses
HRA	Human Reliability Analysis
HSST	Heavy Section Steel Technology
I&C	Instrumentation and Control
IAEA	International Atomic Energy Agency
IASCC	Irradiation Assisted Stress Corrosion Cracking
ICET	Integrated Chemical Effects Tests
IEC	International Electrotechnical Commission
IEEE	Institute of Electrical and Electronics Engineers
INL	Idaho National Laboratory
INPO	Institute of Nuclear Power Operations
IPEEE	Individual Plant Examination of External Events
IRIS	International Reactor Innovative and Secure
ITAAC	Inspections, Tests, Analyses, and Acceptance Criteria
LANL	Los Alamos National Laboratory
LERF	Large Early Release Frequency
LOCA	Loss-of-Coolant Accident
LPSD	Low Power and Shutdown
LWR	Light Water Reactor
MACCS	MELCOR Accident Consequence Code System
MOX	Mixed Oxide

ABBREVIATIONS (Cont'd)

NDE	Non-Destructive Examination
NEA	Nuclear Energy Agency
NEI	Nuclear Energy Institute
NFPA	National Fire Protection Association
NIST	National Institute os Standards and Technology
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
NSIR	Office of Nuclear Security and Incident Response
OECD	Organization for Economic Cooperation and Development
PARCS	Purdue Advanced Reactor Core Simulator
PBMR	Pebble Bed Modular Reactor
PIRT	Phenomena Identification and Ranking Table
PRA	Probabilistic Risk Assessment
PSF	Performance Shaping Factor
PSHA	Probabilistic Seismic Hazard Analysis
PTS	Pressurized Thermal Shock
PUMA	Purdue University Multidimensional Integral Test Assembly
PWR	Pressurized Water Reactor
RCS	Reactor Coolant System
RES	Office of Nuclear Regulatory Research
RG	Regulatory Guide
ROP	Reactor Oversight Process
SDP	Significance Determination Process
SMIRT	Structural Mechanics in Reactor Technology
SNAP	Symbolic Nuclear Analysis Package
SPAR	Standardized Plant Analysis Risk Model
SRM	Staff Requirements Memorandum
SSHAC	Senior Seismic Hazard Analysis Committee
TRACE	TRAC-RELAP Advanced Computational Engine
UNM	University of New Mexico
U.S.	United States

1 INTRODUCTION

The Nuclear Regulatory Commission (NRC) maintains a Safety Research Program to:

- Ensure its regulations and regulatory processes have sound technical bases.
- Prepare for anticipated changes in the nuclear industry that could have safety implications.
- Develop improved methods to carry out its regulatory responsibilities.
- Maintain an infrastructure of expertise, facilities, analytical capabilities, and data to support regulatory decisionmaking.

These essential missions for the research effort were defined when the NRC was established and there was limited experience with the operation of light water nuclear power plants. The need for research remains today, despite the growth of experience with existing power plants, because:

- Nuclear power plants age and encounter challenges of material degradation not anticipated when the plants were designed.
- The NRC considers applications for extending licenses, uprating the operating power levels of plants, and new plant licenses.
- Reactor fuels are used to higher levels of fuel burnup and new cladding alloys for the fuels are introduced.
- Mixed-Oxide (MOX) fuel is considered for the disposal of excess weapons-grade plutonium.
- The NRC evolves its regulations from a deterministic foundation to a risk-informed basis that makes ever greater use of best-estimate analyses to assess safety.
- New technologies including software-based digital instrumentation and control (I&C) systems are backfit into the existing nuclear power plants.
- New water reactor designs such as the ESBWR, which uses passive systems, have been submitted for certification.

There are on the horizon new power reactor concepts that are not based on the water reactor technologies used in the current fleet of power reactors. The U.S. Department of Energy is studying power reactors that use gas cooling, liquid metal cooling, and molten salt cooling. Reactors that use fast rather than thermal neutrons for fission are being studied with the intent of development. These new reactors make it important for the NRC to consider evolution of its regulatory system from one that is specific to water reactor technologies to one that is not specific to particular reactor technologies, but still lead to adequate protection of the public health and safety. This will require substantial research not only for the early development of technology-neutral regulations, but also, in the longer term, for the development of technology-specific regulatory guidance and plans for reviewing specific license applications.

In this report, the Advisory Committee on Reactor Safeguards (ACRS) presents its observations and recommendations concerning that portion of the NRC Safety Research Program devoted to regulation of existing light water reactors (LWRs) and the certification of advanced water reactor designs submitted for certification such as the ESBWR. The ACRS also makes observations on the need for research in anticipation of more advanced power reactor concepts. Observations and recommendations on research dealing with the security of existing nuclear power reactors and nuclear facilities will be provided in separate reports and are not discussed here. The ACRS does not comment on research activities dealing with nuclear waste issues. The Advisory Committee on Nuclear Waste (ACNW) will address such research separately.

In its review of the NRC Safety Research Program, the ACRS considered the programmatic justification for the research as well as the technical approach and progress of the work. The ACRS supports research that:

- Provides support to the identification and resolution of current safety and regulatory issues.
- Provides the technical basis for the resolution of foreseeable safety issues.
- Develops the capabilities of the agency to independently review risk-significant proposals and submittals by licensees and applicants.
- Supports initiatives of the agency such as the development of "technology-neutral" regulatory systems.
- Improves the efficiency and effectiveness of the regulatory process.
- Maintains technical expertise within the agency and associated facilities in disciplines crucial to the agency mission and that are not readily available from other sources.

This review of the NRC Safety Research Program identifies some research activities that have made valuable contributions to the agency mission in the past, but now have reached the point where additional research is not needed for efficient and effective safety regulation. This review also identifies research activities that could benefit by greater collaboration with research activities elsewhere in the world, including collaboration with researchers in Asia and Europe.

General observations and recommendations concerning NRC research activities are presented in Chapter 2. Observations and recommendations regarding research activities in specific technical disciplines are discussed in detail in Chapters 3 through 14:

- Advanced Reactor Research
- Digital Instrumentation and Control Systems
- Fire Safety Research
- Reactor Fuel Research
- Neutronics and Criticality Safety
- Human Factors and Human Reliability Research
- Materials and Metallurgy
- Operational Experience
- Probabilistic Risk Assessment
- Seismic Research
- Severe Accident Research
- Thermal-Hydraulics Research

2 GENERAL OBSERVATIONS AND RECOMMENDATIONS

The NRC Safety Research Program is largely focused on addressing near-term regulatory needs of the agency. Current activities are especially concentrated in three disciplines:

- Materials and Metallurgy
- Probabilistic Risk Assessment
- Thermal Hydraulics

This is an appropriate focus of the current NRC research activities. These activities are discussed further below along with other major aspects of the research program.

The incident at the Davis-Besse Nuclear Power Plant has emphasized, among other things, how important it is for the agency to have a better understanding of the corrosion of metallic systems in the aging fleet of currently operating nuclear power plants. Aging degradation research is necessary to ensure effective aging management for plants operating for extended periods under license renewal and to assess the effect that operation under extended power uprate conditions may have on margins against degradation. Continued challenges posed by stress corrosion cracking of steam generator tubes in pressurized water reactors (PWRs) and systems within boiling water reactor (BWR) vessels further support such focus in the research effort.

Probabilistic risk assessment (PRA) is the basic technology for the risk-informed regulatory system envisaged by the Commission. Research activities are focused now largely on the application of current PRA technology to reactor regulation through the Reactor Oversight Process (ROP). PRA insights are essential to develop and implement revisions to such central regulations as 10 CFR 50.46. They also will play a key role in the development of "technology-neutral" regulatory systems that will have applications to power reactors that are not based on the LWR technology used in the current fleet of operating plants.

The Standardized Plant Analysis Risk (SPAR) models are fundamental tools for risk-informed regulation. A stronger commitment should be made to the improvement of these models and their extension on a timely basis to include fire, seismic, and shutdown risks. The development of these capabilities for the SPAR models will not only provide a regulatory capability but will also encourage industry to more aggressively develop their own capabilities in these areas.

The quality of PRA results depends on good phenomenological models and there are important areas where such models still need further development. Approximate and often bounding risk analyses done for individual plants suggest that the risk of core damage as a result of events initiated by fires can be comparable to risks from other accidents initiated during normal operations. It is important to know if similar results would also be obtained using fire risk assessments of sophistication comparable to the risk assessments possible for normal operations. Such a finding would have ramifications on both regulatory attentions and licensee attentions to safety. The ACRS continues to believe that based on the potential risk significance of fires, fire safety research merits strong consideration in the NRC research program. The collaboration with Electric Power Research Institute (EPRI) is providing a good understanding of the current state-of-the-art methodology for fire risk assessment. This work provides a basis for determining the need for further development.

Thermal hydraulics is a fundamental feature of safety analyses of nuclear power plants.

The NRC allows licensees to do either bounding or best-estimate analysis of plant thermal hydraulics for design basis accidents. Confirmatory review of licensee analyses requires that the agency have high quality thermal-hydraulic analytical tools. Need for such tools is even greater for the analysis of advanced light water reactors that rely on passive systems to achieve safe configurations following accidents.

NRC has consolidated several models of the thermal-hydraulic transient analysis codes into a single code called TRACE. The TRACE code should be subjected to an independent technical review to assess its range of validity. The TRACE code then should be at a point at which it can be used as the primary thermal-hydraulic tool for regulatory analyses. A plan should be developed for its integration into the regulatory process. This integration will require strong support from the management of the NRC user organizations since such a change in the short run will create additional burden on the staff.

The potential for blockage of sump screens by debris dispersed into the sumps during depressurization of the reactor coolant system during an accident remains an unresolved issue. The complexity of the interactions between fibrous and particulate debris, as well as the chemical interactions that can occur among debris materials and solutes in the coolant, make predictions of blockage and consequently screen size requirements difficult. Research needed to reach a prompt resolution of this issue should receive the required resources.

International Collaboration

Reactor safety is an international undertaking. It is important that there not be great differences in safety regulations among the nations making major use of nuclear power generation. The NRC research is making good use of collaborations with other countries on reactor safety research. Much of this collaboration has been in the nature of information exchange. Such information exchanges are important and should continue to be encouraged and supported. They provide access to information and a kind of peer review that might not otherwise be obtained. However, there are other important cases where NRC has gone farther and formally partnered with other countries to leverage resources for experimental investigations of important reactor safety research issues. Such collaborations are especially noteworthy in the disciplines of reactor fuel research and in severe accident research. The combined resources of the partners in these collaborations are yielding higher quality and more extensive results than would be possible in research programs sponsored by individual countries.

Other areas of NRC research could benefit from more extensive collaborations. Such areas include fire safety research and thermal-hydraulics research. The benefits of such collaborations become more apparent as NRC moves to more realistic analyses which may require validation by costly large-scale, integral tests. Collaborations of this type may become even more important in the future as new types of reactors are proposed for certification internationally. To be effective and efficient in dealing with future challenges, NRC should look for opportunities to increase significantly collaboration with other countries. The ongoing collaborative efforts are very extensive with European countries. More collaboration with Asian countries having active nuclear power plant programs should be pursued.

Support for Future Licensing Activities

There has been a recent resurgence in interest in the use of nuclear reactors for the generation of electrical power. Innovative reactor designs are being suggested to sustain uranium resources and to generate electrical energy at much greater efficiency. The U.S. Department of Energy is studying very high temperature gas reactors, supercritical water reactors, sodium-cooled reactors, lead-bismuth cooled reactors, and molten salt cooled reactors. Some of these reactors will use fast neutrons rather than moderated neutrons for fission. These reactors use technologies quite different than those used for the currently operating fleet of reactors. The current regulatory framework is not well suited for the licensing, regulation, or monitoring of such different reactor technologies. Several years ago, it appeared that a substantial portion of NRC resources might need to be devoted to the development of the capabilities to address these very advanced reactor technologies. Today, this is not the case. NRC advanced reactor research resources are focused on addressing issues associated with advanced water reactors such as the ESBWR and the EPR.

This seems to be an appropriate use of NRC's limited resources for advanced reactor safety research. Very advanced reactor concepts have not reached a sufficient state of development that productive use of regulatory research resources can be made. However, work should continue on the development of a technology-neutral framework for regulation, although the development of technology- specific guides can be delayed until it is clearer which alternate reactor technologies will be of the greatest interest.

Development of the framework is not only important for the licensing of non-light-water reactors, but also may provide insights that are useful in developing a more efficient regulatory program for advanced reactors of all types.

There are some indications that certifications may be sought for advanced designs with minimal experimental study of plant response under accident conditions. NRC needs to provide clear guidance on its expectations for the experimental validation of computer models used in the licensing of advanced reactors that do not use familiar technologies. Development of such guidance is an area of advanced reactor research that can be pursued at relatively low cost, but which can play an important role in timely and efficient licensing of advanced reactors with new technologies.

Opportunities for Independent Research

In recent years, a strong effort has been made to ensure that NRC research is supportive of the needs of the line organizations. Focusing NRC research entirely on the immediate needs of the line organizations does, however, entail an important risk. This focus reduces the opportunities for independent thought by the research staff and the opportunities to conduct research that could make more dramatic improvements in the regulatory process, for example, in the tools that support it at a time when there is a rapid increase in workload. The risk is magnified by the diversion of so much research talent to address issues of security of nuclear facilities. There is the further risk of a loss of prestige in the research program focused as it is on issues of implementation. This could eventually lead to a loss in the credibility of the technical basis that underlies regulatory decisions.

It is important that NRC research stay abreast of technological developments that can enhance safety. Areas where developments in the larger technical community can be important to the NRC include reactor fuels, corrosion and materials degradation, man-machine interfaces, technologies for monitoring component performance, inspection techniques, and virtual facility inspections. Where NRC can adopt or adapt developments in other industries, safety can be improved and the efficiencies of NRC reviews enhanced.

One mechanism for RES to interact with the larger technical community is by sharing its own research plans. This has been done for research into digital instrumentation and control. Investigators did creditable reviews of the state-of-the-art, presented them at appropriate professional society meetings as a kind of public peer review, and developed from these state-of-the-art reviews a research plan that is well directed to address agency needs. Sharing research plans with a larger technical community is a strategy that would benefit other NRC research activities. Such interactions also help provide visibility for and help sustain the prestige of the NRC research program.

A Vision for the Future

Nuclear energy will remain an important and perhaps growing component in the mix of energy generating technologies used in this Country. There is the potential that many new reactors could be built in the next 15 to 20 years. It is unlikely that agency resources of either manpower or funds will experience a similar growth. Indeed, the experience level of the NRC staff is likely to decrease due to retirements just when the new plant licensing activities accelerate. A portion of the research program needs to be devoted to the development of a regulatory infrastructure for regulatory work in the next 20 years that supports a staff with less experience dealing with more tasks. Computerization will be undoubtedly an important element of such an infrastructure. The ACRS can foresee, for example, a time when regulatory staff have routine access to superior analysis tools for systems analysis, phenomenological analysis, and risk assessment. Development of such validated and verified tools for routine use by non-specialists will require a research program that is not tied exclusively to the near-term issues of the regulatory process. Appropriate attention will have to be paid to the agency's analytical tools, its access to facilities, and its ability to provide recently recruited staff with a sound understanding of past safety decisions. Availability of good infrastructure will enhance safety and allow for much more efficient and effective NRC review of new reactor designs and licensing applications based on realistic evaluations of safety.

Observations and Recommendations on Specific Research Activities

NRC research has made substantial progress since the last ACRS report, NUREG-1635, Vol. 6, on the research program. This progress has occurred despite the diversion of substantial research talent in the agency to address issues of reactor security that are not reviewed here. Notable accomplishments of the research program in recent years include:

- Multidisciplinary review of pressurized thermal shock criteria
- Performance of high-burnup fuel during reactivity transients
- Embrittlement of zirconium alloy cladding when taken to high burnup.

The ACRS applauds these high technical quality research accomplishments. The ACRS is, however, disappointed at the pace with which these important research results are being used to modify regulations.

Other major observations and recommendations concerning the NRC research activities are summarized below and also discussed in more detail in individual Chapters.

Advanced Reactor Research

Highest priority should be given to those research activities that support the ESBWR design certification process. The importance of tasks associated with the ACR-700 or a related design with higher power depends on whether the certification review for such a reactor is resumed.

Digital Instrumentation and Control Systems

Software-based digital electronic systems are inevitable for both current and more advanced design nuclear power plants. The staff has developed a research plan that addresses the challenges associated with the use of digital technology that will face the agency in the next five years.

The ACRS has recently reviewed and reported favorably on the research plan for digital systems. The ACRS was impressed by the technical quality in the development of the research plan, the scope and content of the plan, and the prioritization of activities in the plan. The ACRS recommends a number of improvements to an already quality research plan, including addition of an explicit element to the plan to study the acceptability of international standards in comparison to Institute of Electrical and Electronics Engineers (IEEE) standards for meeting regulatory requirements concerning digital instrumentation and control systems. This study will be an important element of efforts to develop a multi-national design approval process.

Fire Safety Research

There have been a number of important accomplishments by NRC research in the area of fire protection since the last ACRS report on NRC safety research program in 2004. Fire safety research continues to merit emphasis in the NRC research program.

RES, in cooperation with EPRI has taken some important steps 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 risk assessment.

There are a variety of methods that can be used to model the progression of fires. Some of these have been used in fire protection programs for non-nuclear facilities for many years. The ranges of applicability of these methods have not been well studied or documented. In cooperation with EPRI, a program is in progress to verify and validate a set of fire progression modeling tools. The accuracies of these tools are being examined for different fire conditions and applications by comparison with benchmark tests.

RES has worked closely with the U.S. industry in undertaking generic fire risk research activities. Fire risk is, however, an issue of world-wide concern. RES has not aggressively sought collaborations with the international community to advance NRC capabilities for fire risk assessment. Collaborations with other countries especially in experimental studies may be essential to leverage resources of all partners sufficiently to achieve fire risk assessment capabilities commensurate with what can now be done for risk from normal plant operations.

Reactor Fuel Research

The NRC research on reactor fuel has been concentrated in recent years on the confirmation of regulatory decisions that allow licensees to take light water reactor fuels to burnups of nominally 62 GWd/t. The research on high-burnup fuel is reaching a substantial level of maturity. Some major confirmatory experiments remain to be done - notably experiments on reactivity insertion to be done in a water loop at the CABRI reactor. Since the last ACRS report on NRC safety research program, plans for these experiments have been revised so the experiments which are part of an international collaborative effort now better meet the agency needs. It is important that this work that is so well coordinated both with agency needs and with international partners be taken to completion. Still major findings of the research effort can be reduced to regulatory practice now. This reduction to regulatory practice needs to be initiated and pursued aggressively.

It is evident that high-burnup fuel research will soon achieve results that are adequate for agency needs. The NRC has made clear that it will expect the nuclear industry to provide necessary safety analyses and experimental data should the industry want to take fuel to burnups that exceed the current regulatory maximum. NRC needs to make these expectations more explicit, particularly its expectations for the experimental data needed to support the analyses of high-burnup fuel behavior under accident conditions.

Neutronics and Criticality Safety

The neutronics and criticality safety research program is small but appears adequate to ensure that the NRC has capabilities to meet immediately foreseen regulatory needs.

In the future, more innovative core designs for advanced reactors may be submitted to the NRC. Confirmatory analyses of reactor core physics will be an essential part of the regulatory process for these advanced reactors. The capabilities now available to the NRC in the area of core physics may well be stretched. It will be useful to the agency to understand these future needs. If long-term development activities are identified, such as those that might be needed for analysis of the PBMR, additional research may be needed in this area.

Human Factors and Human Reliability Research

As new reactor designs, likely dependent on a higher degree of automation than the current fleet, are introduced, the need for revised guidance and tools for the NRC staff in human factors and human reliability analysis will increase. RES has initiated a project to develop regulatory guidance and analytical techniques to review human factors for advanced nuclear power plants. The ACRS views this five-year project essential for preparing the staff in reviewing advanced reactor designs.

The quantification of human reliability continues to be a challenge in risk assessments. Human reliability modeling introduces large uncertainties in probabilistic risk assessments. The NRC staff needs guidance in its review of the human reliability models used by the industry in risk-informed licensing applications. Progress has been made with the publication of NUREG-1792, "Good Practices for Implementing Human Reliability Analysis (HRA)." Still, further guidance is needed for reviewers of licensing applications.

Materials and Metallurgy

The NRC is investing heavily in the better understanding of materials degradation issues in the currently operating fleet of nuclear power plants. Such investment is justified in view of significant agency regulatory activities that aging degradation research supports.

The current program is well focused on improving the agency's ability to independently evaluate licensees' efforts to prevent, detect, and mitigate environmentally assisted stress corrosion cracking.

The nuclear industry and the NRC have often been surprised by unexpected material degradation problems. As a result, they have responded to such problems in a reactive mode which has proven to be inefficient. The Proactive Materials Degradation Assessment project is an effort to identify potential material degradation problems before they manifest in operating nuclear power plants. The ACRS admires the vision of this undertaking and supports its continuation. The ACRS looks forward to reviewing the initial results of this ongoing effort soon and learning whether the admirable goal of this project is, in fact, feasible.

RES needs to reevaluate the need for continued research into heavy section steel components. This research may be justified if there is a clear need for NRC to develop its capabilities in the area of probabilistic fracture mechanics (PFM) so that it can evaluate licensees' applications. If this is the case, the research needs to be clearly focused on this objective and not the research that the industry should perform to meet its responsibilities to ensure reactor pressure vessel integrity. It appears now, however, that it is NRC that is advancing the state-of-the-art and making available information that allows licensees to reduce conservatism in their analyses.

Operational Experience

The ACRS is supportive of the research activities in the area of operational experience and recommends that these activities be continued. In light of the limited resources allocated to these tasks, RES has done a commendable job in producing outputs in well-documented and thorough fashion. Tasks that are currently in the 2005 research plan related to operational experience should remain funded and should be continued for the foreseeable future.

Probabilistic Risk Assessment

Altogether the scope and the number of activities in the NRC's PRA research program is quite impressive. The ACRS cautions, however, that NRC should not allow its work in such a crucial technology as risk assessment to become totally devoted to the support of line activities. Methods development is still important. As an example, the ACRS notes that considerable research is being reported in the literature regarding Binary Decision Diagrams as tools for solving large fault tress without resort to cutoff frequencies as is now done. The staff needs to review the literature concerning Binary Decision Diagrams and evaluate the need to adopt this technology. The growing importance of the SAPHIRE code and the SPAR models in the regulatory process warrants such an investigation.

Seismic Research

Seismic hazard analysis and structural response are not areas where NRC must maintain state-of-the-art expertise. Such expertise is available to the NRC on a contractual basis. As ACRS noted in its previous report on NRC safety research program, research activities at the agency can be confined to support needed updates to regulatory guides and collaborative work with the international community to stay abreast of developments in other countries. The current research program is, indeed, largely focused on needs of the regulatory process and a few important international collaborations.

Severe Accident Research

The ACRS is very supportive of the strategy NRC has developed to maintain and update its capabilities for severe accident analyses. The leveraging of resources through international collaborative experimental research is especially important. The planned extensions and continuations of current collaborations are well worth the investment.

Thermal-Hydraulics Research

Highest priority should be given to the integration of TRACE code into the regulatory process. As this integration progresses, the research staff can continue its efforts to improve and further develop TRACE on a "time available" basis. The ACRS is concerned now that efforts to improve TRACE lack prioritization and defensible organization. Prioritization of technical improvements might be aided substantially by commissioning a detailed peer review of TRACE. To do this, the staff will have to have available code documentation of outstanding scope and quality. Such high quality code documentation will also be needed if the code is to become part of the regulatory process. Code documentation, then, is a task that ought to take precedence in the thermal-hydraulic research effort.

3 ADVANCED REACTOR RESEARCH

The agency is already engaged in various activities related to a number of new plant designs, including ESBWR, PBMR, IRIS, and ACR-700. The staff has begun its review of ESBWR design certification application. It is anticipated that requests for design certification reviews will be received for EPR, and PBMR. Of these, the ESBWR, ACR-700, IRIS, and EPR can be certified in all likelihood under the current requirements in 10 CFR Part 52 using the design basis accidents as they are now defined. Nevertheless, there will be the need for NRC to verify the thermal-hydraulic assessments made by the applicants for the various designs. This will require review and approval of the computer codes that were used by the applicants for assessing the design basis accidents. Confirmatory analyses will require that design-specific versions of the computer codes TRACE and CONTAIN be available to the staff for audit calculations and independent assessment of separate effects and integral system experiments. Highest priority should be given to those research activities that make such tools available for the ESBWR design certification review. This includes tasks Y6857, Y6898, N6018, and Y6804. The importance of tasks associated with the ACR-700 or a related design with higher power, Y6831, Y6812, Y6899, Y6489, Y6748 and Y6933, depends on whether the certification process for such a reactor is resumed.

Certification reviews for designs such as the PBMR and the 4S that do not use water reactor technology will be more challenging. Although significant efforts were undertaken in the past to license such non-LWR designs under the current regulatory system designed for light water reactors, it would be far more appropriate, effective, and efficient to have the "technology-neutral-framework" for certification of such designs. For timely application to these reactor types (and possibly even more unusual designs in later years), the development of the technology-neutral framework needs to be given high priority and provided sufficient resources to complete the job in 2006 and to allow two years for rulemaking. High priority, then, should be given to the tasks N6205 and Y6487 that will develop a technology-neutral framework for the regulation of advanced nuclear power plants.

The Commission has expressed a desire for "enhanced safety" for new reactor designs. To ensure that new designs have reached enhanced levels of safety, the NRC will require each of the applicants for design certification to submit a full-scope PRA with consideration of uncertainties. The staff must be prepared to review these PRAs, to validate the results and to compare the results with acceptance criteria for "enhanced safety." This evaluation will include undoubtably a complete Level-2 evaluation of accident source terms since LERF (large early release frequency) will no longer be an appropriate safety metric. To review and independently assess the Level-2 analyses of source terms, the regulatory organizations will need design-specific versions of the MELCOR computer code. There is, then, the potential need to develop MELCOR versions specific for the PBMR and 4S designs. Development of such code versions will take time. Second priority should be given then to tasks K6703, Y6801, and Y6619. Again, the importance of developing an accident progression model for ACR-700 depends on resumption of its certification process.

Table 1. Advanced Reactor Research Activities

Job Code	Title	Comment
Y6857	*ESBWR Input Deck Development*	Analysis of DBAs in ESBWR using the TRACE code; This project should have high priority.
Y6898	*ESBWR Design Certification Report*	Support for review of PRA for ESBWR; This project should have high priority.
N6018	*Separate Effects Experiments*	Separate effects tests in support of TRACE model development for ESBWR; This project should have high priority
Y6804	*ESBWR Containment Support*	Analysis of experiments with CONTAIN and MELCOR. This is a high priority task for ESBWR design certification review.
Y6489	*PRA for ACR-700*	Support for review of ACR-700 PRA. This project can be deferred until certification application becomes active again.
Y6899	*ACR-700 Design Certification Support*	Support for review of PRA for ACR-700. This project can be deferred until the certification application becomes active again.
Y6748	*Review ACR-700 Support*	Support for thermal hydraulics review of ACR-700. This project can be deferred until the certification application becomes active again.
Y6831	*Methods Development for ACR-700*	TRAC code upgrades needed for ACR-700 certification calculations. This project can be deferred until the certification application becomes active again.
Y6812	*ACR-700 Input Model Development*	Develop RELAP5 and TRAC-M input models for ACR-700. This project can be deferred until the certification application becomes active again.
Y6933	*Evaluate Severe Accident Phenomena in ACR-700*	Analysis of risk dominant sequences for ACR-700. This project can be deferred until the certification application becomes active again.
K6703	*Coop. Agreement with Center for Advanced Nuclear Energy Systems*	Improve NRC's knowledge and information on advanced reactors. This project is useful but can have a second level priority.
Y6619	*Advanced Reactor PRA Development*	Develop knowledge needed to review advanced reactor PRAs. Second priority work for non-LWR design certifications.

Table 1. Advanced Reactor Research Activities (Continued)

Job Code	Title	Comment
Y6801	***Advanced Reactor/Severe Accident Code Development***	Develop a version of MELCOR code for advanced reactors. This project can have a second level priority.
Y6755	***Materials Evaluations for Advanced LWR Reactors***	Research materials engineering issues for advanced LWRs especially effect of coolant environment on fatigue and in-service inspection and monitoring. This project can have a second level priority.
N6205	***Assistance for Development of a Regulatory Structure for New Plant Licensing***	Development of a technology-neutral regulatory framework. This project should have high priority.
Y6487	***Advanced Reactor Regulatory Framework Development***	Development of a regulatory framework for advanced reactors. This project should have high priority.
Y6741	***Environmental Effect on Containment***	Develop understanding of the properties of concrete in high temperature gas cooled reactors. This project can have a low priority.

[Intentionally Blank]

4 DIGITAL INSTRUMENTATION AND CONTROL SYSTEMS

Software-based digital electronic systems are inevitable for both current and advanced design nuclear power plants. Already such software-based digital electronics appear ever more frequently in systems for plant control. Eventually, they will appear in safety systems. The reliability of digital systems especially when using commercial, "off the shelf" hardware and software has become an issue because they cannot be comprehensively tested. The quality of the requirements for the software cannot be assessed fully through testing. Quality in the software-based systems is achieved through the control of the process of software development. Particular attention has to be given to the requirements for the system software. Failure to specify adequate requirements has often been found to be the root cause of digital system failures. Review and approval of licensee applications to incorporate software-based digital systems in its facility is, then, time-consuming for both the regulator and the licensee. New failure modes that arise in digital systems need to be recognized. Such failures can depend on the operational state of the system at the time of failure. Indeed, testing and maintenance as well as normal operations of digital systems can create the opportunities for their own unique kinds of failures.

Security of digital systems has become a major concern and there needs to be regulatory guidance and acceptance criteria for the security aspects of digital systems. Codes, Standards, and regulations must prompt the designer of digital safety systems to avoid system communications outside of the controlled areas of the plant and the use of wireless technology must be carefully evaluated to prevent interception, interdiction, or interference in communications to digital systems.

Current licensing guidelines provide information on what to review in digital systems. They do not necessarily provide sufficient guidance on how to review submittals or the acceptance criteria to apply. The NRC staff needs a firm technical basis for deciding when review of submittals is adequate and when confirmatory analyses are necessary. The situation will get worse with time. Digital systems in nuclear power plants are expected to become more numerous. The complexity of these systems will increase. There is the potential for the consolidation of what are now discrete analog safety systems into a single digital system. At the same time, there is interest both within the agency and on the part of licensees to adopt risk-informed techniques for the review of digital software systems. NRC lacks the technical basis to support risk-informed reviews of digital systems. Currently, the ability to model the reliability of software-based digital systems in PRAs is very limited. Without quantitative risk information, a much less defensible, qualitative, "graded approach" to the review of digital systems is likely to emerge.

If the use of digital protection systems and control systems becomes as widespread as now predicted, review of digital systems as part of ITAAC (Inspections, Tests, Analyses, and Acceptance Criteria) may eventually become a burdensome, time-consuming aspect of the licensing process. Methods and tools to facilitate confirmation that "as built" systems conform to accepted designs are going to be needed. As use of digital systems becomes more extensive in nuclear facilities, NRC may find it necessary to reconsider its current positions on defense-in-depth and diversity in instrumentation and control systems.

The nuclear industry is not a major user of digital technology relative to many other industries. Yet, the consequences of failure of digital systems in nuclear power plants are likely to be less acceptable to the public than are failures of such systems in other industries even when consequences are significant. Greater rigor in the review of digital systems is necessary for nuclear applications of these systems. It is expected then that NRC will have to "blaze new paths" in this area through research. In particular, the usual industrial practice of separately considering hardware and software reliabilities may not be adequate for nuclear systems and a more integrated or systems approach may be needed.

The staff has developed a research program plan that addresses these challenges that will face the agency in the next five years. Critical reviews of the state of the art in several areas were completed, documented and presented before audiences in professional societies. Recommendations made to the NRC by independent bodies, including the National Academy of Sciences were considered in the development of the plan. Inputs from the program offices at NRC (NRR, NSIR, and NMSS) were also obtained. The research plan is well directed toward meeting the agency needs and is intended to provide:

- Improved technical guidance for review of digital systems
- Technical support for developing improved acceptance criteria for assessing the safety and security of the systems
- Tools and methodologies for improved review of digital systems
- Technical bases for including models of digital systems in PRAs

The research plan has six major elements:

- Systems aspects of digital technology
- Risk assessment of digital systems
- Emerging digital technology with application to nuclear facilities
- Software quality assurance
- Security aspects of digital systems
- Advanced nuclear power plant digital systems

Within each of these major elements of the plan, there are a number of subelement. The staff has prioritized work on the subelement basis. Now, the major focus of the work is on collection of data on the failure modes of digital systems, including international experience with digital system failures, software quality assurance, environmental stressors on digital systems, modeling digital systems in PRAs and cyber security of digital systems. Within the general element of emerging digital technologies applicable to nuclear facilities, attentions are on system diagnosis, prognosis and on-line monitoring as well as wireless technology. Research on digital systems for advanced nuclear power plants was given a low priority. Perhaps, future new orders for advanced plants (AP1000, ESBWR, etc.) may create new regulatory demands and cause this priority to be re-evaluated.

The ACRS has recently reviewed and reported favorably on the research plan for digital systems. The ACRS was impressed by the technical quality in the development of the research plan, the scope and content of the plan, and the prioritization of activities in the plan. Indeed, it would help better understanding of other research programs if they were also based on such thorough planning efforts. The ACRS recommends the following to further improve an already quality research plan:

- The plan is currently focused very much on the software aspects of digital systems. Eventually, the research will have to be expanded to recognize the entire system of interest. Though the focus on software is appropriate now, the plan should reflect the need for expansion in scope in the longer term.

- There should be an explicit element of the plan to study the acceptability of international standards in comparison to IEEE standards (such as IEC 60780 in comparison to IEEE 323) for meeting regulatory requirements concerning digital instrumentation and control systems. This study will be an important element of efforts to develop a multi-national design approval process.

- As data on digital system failures are collected and analyzed, the research staff should prepare episodic papers or presentations to professional societies of their interpretations and "lessons learned" for peer review by the larger digital system reliability community.

Table 2. Research Activities in Digital Instrumentation and Control Systems

Job Code	Title	Comment
N6116	*Secure Network Design Techniques*	Develop technical guidance for mitigating cyber vulnerabilities in secure networks
N6095	*Assignment Robert Edwards*	Support analysis of digital systems failures and consequences
Y6962	*Emerging Technologies*	Conduct periodic surveys of the state of the art for a wide range of technology issues in the I&C field
Y6873	*International Cooperative Research Program on Digital I&C*	Search for opportunities to collaborate in the safety assessment of digital systems
N6010	*COMPSYS*	OECD/NEA international program to develop data-base on digital systems failures
K6472	*Risk Importance of Digital Systems*	Develop methods to include digital systems in PRAs
Y6332	*Digital Systems Risk*	Develop a PRA method for modeling failures of digital I&C systems.
Y6591	*Software Reliability Code Measurements*	Large-scale validation of NRC methodology for predicting software reliability in digital systems
N6080	*Interactions with Industry on Standards*	Development of standards on EMI/RFI
Y6475	*Wireless*	Confirmatory research on effects of wireless communications
N6113	*Security of Digital Platforms*	Study in laboratory digital systems generically qualified for nuclear safety applications
N6114	*Site-specific Protocol Analysis*	Study power plant implementation of digital systems generically qualified for nuclear safety applications
N6124	*Digital System Dependability Performance*	Qualify safety of a digital system using a process developed in NRC research
W6851	*Review Guidance for Lightning*	Support for response to public comments on draft regulatory guide; Program completed.
Y6924	*SPACE Engineering Workstation for Review of TXC Applications*	Evaluate the use of the RETRAN tool for review of TELEPERM-based digital instrumentation and control upgrades
Y6349	*Halden Environmentally Assisted Cracking* (The title of this program is amazingly misleading!)	Despite the name this is research on COS operating experience, ranking software engineering practices and testing digital reliability assessment methods

5 FIRE SAFETY RESEARCH

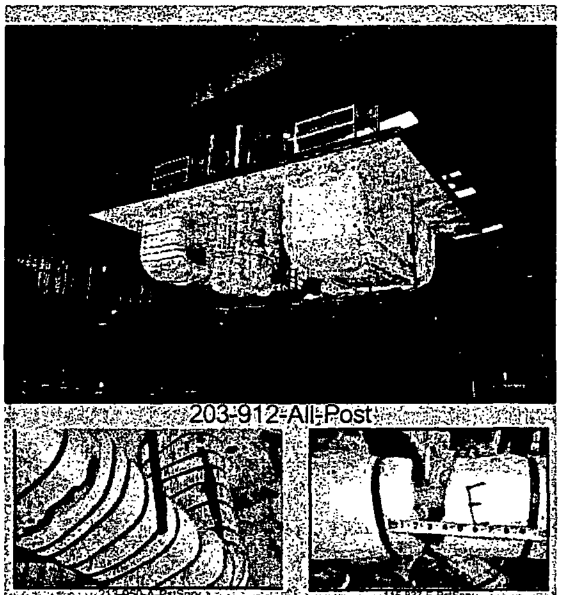
The fire safety research program can be divided into three technical areas:

- Fire Risk Assessment
- Fire Modeling
- Fire Testing

Each of these areas is discussed below.

Fire Risk Assessment: The nuclear industry has made substantial progress over the past thirty years in the development and standardization of internal events risk assessment. Progress in the development of the methods of fire risk assessment has been much slower. Only a few nuclear power plants currently have full-scope fire risk assessments. The requirements placed by the NRC on the industry for performing Individual Plant Examinations of External Events (IPEEE) permitted the use of simplified and qualitative techniques. Most analyses of fire risk at nuclear power plants were performed with these less quantitative techniques..

As the NRC moves from deterministic regulations to risk-informed and performance-based regulations, the need for quality risk information increases greatly. It is expected that many nuclear power plants will transition from their current fire protection programs to the risk-informed, performance-based fire protection programs that meet the requirements of 10 CFR 50.48(c) and the referenced 2001 Edition of National Fire Protection Association (NFPA) standard, NFPA 805, "Performance-Based Standard for Fire Protection for Light-Water Reactor Electric Generating Stations." This is only possible if a full-scope fire risk assessment is performed for each transitioning nuclear power plant. NRC will need appropriate standard to assess the quality of such fire risk assessments and inspectors will need tools



Confirmatory Testing of Hemyc/MT Fire Barriers

The Hemyc and MT electrical raceway fire barrier systems are used in a number of plants to provide a fire barrier between two trains of safe shutdown equipment within a fire area. In the performance of fire protection inspections at nuclear plants, questions raised regarding the fire resistance capability of these systems. NRC conducted confirmatory testing of Hemyc and MT fire barriers at the Omega Point Test Facility in 2005. All of the configuration tested failed to meet acceptance criteria. A Generic Letter was issued requiring licensees to identify where Hemyc and MT fire barriers are used in their plants and to provide a plan and schedule for corrective actions.

and the knowledge to assess the validity of changes to the licensing basis made at the plants.

RES in cooperation with EPRI has taken some important steps 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 risk assessment. In 2005, the final NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities," was issued. This document provides a structured framework for the overall fire risk assessment as well as specific recommended practices to address key aspects of the analysis. 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. Although some utilities have used parts of the improved approach, no utility has completed a fire risk assessment using the methodology and submitted the assessment for critical peer review.

Areas of fire risk analysis where further development in methodology is needed have been recognized by RES. These include spurious equipment actuations, post-fire human reliability analysis, aging effects, and low power and shutdown fire risk.

Fire Modeling: Deterministic criteria for fire protection are typically very conservative in their treatment of fire progression. Fire risk assessment, on the other hand, requires a realistic assessment of fire progression. There are a variety of methods that can be used to model the progression of fires. Some of these have been used in fire protection programs for non-nuclear facilities for many years. The ranges of applicability of these methods have not been well studied or documented. In cooperation with EPRI, a Project (Y6688) is in progress to verify and validate a set of fire progression modeling tools. The accuracies of these tools are being examined for different fire conditions and applications by comparison with benchmark tests performed by National Institute of Standards and Technology (NIST) The phenomena identification and ranking table (PIRT) process is being used by RES to identify potential limitations of the fire progression modeling tools. Preliminary draft of multi volumes NUREG-1826, "Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications," was issued for Public Comment in January 2006.

Fire Testing: Confirmatory testing is another critical element of the fire safety program. During the past year, tests were performed at the Omega Point Test Facility on the Hemyc and MT electrical raceway fire barrier systems (see side column). The test results indicated that these fire barrier systems are not capable of satisfying regulatory requirements. It is somewhat distressing that confirmatory testing of these fire barriers did not occur until sixteen years after problems were identified with a similar fire barrier material, Thermo-lag, and five years after inspection teams raised specific concerns about the Hemyc and MT fire barriers. The results of these tests provide further evidence of the continuing value of NRC's confirmatory testing program.

There have been a number of important accomplishment by NRC research in the area of fire protection since the last ACRS report on NRC safety research program in 2004. Fire safety research continues to merit emphasis in the NRC research program. Approximate, and often bounding risk analysis, performed for individual plants indicate that the risk of core damage from fire- initiated events is comparable to or greater than the risk from other accidents initiated during normal operations. It is important to know whether the same conclusion would be drawn if fire risk assessments were performed using tools of comparable sophistication as those used for assessing risk of accidents initiated by internal events. Conclusions based on more realistic fire risk assessments could have ramifications on both regulatory attention and licensee attention to safety. In the interim, risk-informed regulatory decisions are being made with an incomplete understanding of the impact of fire on risk.

RES has worked closely with the U.S. industry in undertaking generic fire risk research activities. Fire risk is, however, an issue of world-wide concern. France, for example, has recently initiated a fire research program in a multi-volume test facility. RES has not aggressively sought collaborations with the international community to advance NRC capabilities for fire risk assessment. Collaborations with other countries especially in experimental studies may be essential to leverage resources of all partners sufficiently to achieve fire risk assessment capabilities commensurate with what can now be done for risk from normal plant operations.

Table 3. Fire Safety Research Activities

Job Code	Title	Comment
N6107	***10 CFR 50.48C - related Technical Activities***	Develop fire PRA methods, tools, and data. Perform demonstration studies. This is a collaborative effort between NRC and EPRI.
N6108	***Fire Risk Assessment and Risk Applications***	Improve fire PRA approaches. Develop test plan to address spurious equipment actuation issues.
N6134	***LPSD Level 1 & Fire Risk Standard***	Supports NRC staff in the development of industry standards.
Y6651	***Effects of Switchgear Aging on Energetic Faults***	Assess the aging of medium voltage switch gear as it affects the potential for energetic electrical faults. Such faults are thought to contribute significantly to fire initiation. The work addresses how aging affects fire risk.
Y6688	***Fire Model Benchmarking and Validation***	Benchmark fire model computer codes against fire experiments performed by NIST. Such validation is necessary to ensure that appropriate tools are used for regulatory applications.
Y6817	***Fire Protective Wrap Performance Testing***	Test Hemyc and MT fire wrap materials. These important tests conducted in 2005 showed there to be significant issues associated with these fire barrier materials.
Y6955	***Fire Incident Records Exchange***	Collect and analyze international fire events data. This is a long-term collaborative effort with OECD.

6 REACTOR FUEL RESEARCH

Reactor fuel is an important element of safety technology. NRC must maintain expertise in the area of reactor fuel because of both the importance to safety and because of the limited availability of expertise outside the agency that is independent of licensees. Research is an important vehicle for maintaining expertise in reactor fuel. NRC research on reactor fuel during normal operations and design basis accidents has been concentrated in recent years on the confirmation of regulatory decisions that allow licensees to take light water reactor fuels to burnups of nominally 62 GWd/t. This research has largely resolved the issue of the vulnerability of high-burnup fuel and cladding to reactivity transients though some confirmatory tests need to be completed. Research results will allow regulatory changes to better reflect the degraded capacity of high-burnup fuel to sustain reactivity insertion events.

The reactor fuel research has remained quite productive as examinations of high-burnup fuel behavior under loss-of-coolant accidents have been initiated. An important discovery has been the synergistic effect on clad ductility of hydrogen absorption during normal operation and steam oxidation of the cladding during an accident. Based on the research, revised embrittlement criteria have been developed that could be incorporated into 10 CFR 50.46.

The research on high-burnup fuel is reaching a substantial level of maturity. Some major confirmatory experiments remain to be done - notably experiments on reactivity insertion to be done in a water loop at the CABRI reactor. Plans for these experiments have been revised since our last report on reactor fuels research so the experiments which are part of an international collaborative effort now better meet agency needs. It is important that this work that is so well coordinated both with agency needs and with international partners be taken to completion. Still, major findings of the research effort can be reduced to regulatory practice now. This reduction to regulatory practice needs to be initiated and pursued aggressively.

It is evident that high-burnup fuel research will soon achieve results that are adequate for agency needs. The NRC has made clear that it will expect the nuclear industry to provide necessary safety analyses and experimental data should the industry want to take fuel to burnups that exceed the current regulatory maximum. NRC needs to make these expectations more explicit, particularly its expectations for the experimental data needed to support the analyses of high-burnup fuel behavior under accident conditions.

Completion of NRC's research on high-burnup fuel raises the question of how NRC will maintain expertise in fuel. Continued evolution in fuel cladding alloys can be anticipated. Interest is developing within the industry in fuels with enrichments exceeding 5% ^{235}U. These higher enrichment fuels may necessitate NRC research. If use of MOX fuel becomes more widespread than the planned disposal of excess weapons-grade plutonium, additional research on MOX fuel with reactor grade plutonium may be needed. Research on both higher enrichment fuel and MOX fuel can be done with substantial collaboration with international partners. Such collaboration will further the ideal of international safety evaluations of nuclear power plants.

Table 4. Reactor Fuel Research Activities

Job Code	Title	Comment
Y6586	*Fuel Code Assessment for MOX*	Improve FRAPCON and FRAPTRAN for calculating the behavior of MOX fuel rods; An important activity for licensing core loads for excess weapons-grade plutonium disposal.
Y6580	*Fuel Code Applications for High Burnup Fuel*	Improve FRAPCON and FRAPTRAN for calculating the behavior of high burnup fuel rods; an important activity as licensees press limits on allowable fuel burnup.
Y6788	*Halden Fuel Experiments Under Transient Conditions*	Data on fuel behavior under operational transient conditions for code development.
N6074	*STUDSVIK Cladding Integrity Project*	Stress corrosion cracking, hydride embrittlement and delayed hydride cracking study of ZIRLO clad. Defueled clad segments provided for NRC research.
Y6849	*ZIRLO Cladding Performance*	Adequacy of criteria for ZIRLO cladding performance in a LOCA; an important study of cladding used for high burnup fuel and the synergism between hydriding and oxidation on clad ductility.
Y6850	*M5 Cladding Performance*	Adequacy of criteria for M5 cladding performance in a LOCA; an important study of cladding used for high burnup fuel and the synergism between hydriding and oxidation on clad ductility.
G6923	*Failure of Hydrided Zircaloy under Severe Loading Conditions*	Develop theoretical model of mechanical failure of hydrided Zircaloy cladding.
W6832	*CABRI Water Loop*	NRC support for the CABRI water loop for RIA testing of high burnup fuel; confirmatory testing of high burnup clad and fuel vulnerability to reactivity transient events.
Y6367	*High Burnup Cladding Performance*	LOCA testing of high burnup cladding behavior; important study of cladding used for high burnup fuel and the synergism between hydriding and oxidation on clad ductility.
Y6723	*International Agreement on Fuel Behavior and Materials Science Research*	Data report on BIGR pulse reactor tests.
Y6847	*Clad Performance in ATWS*	Determine the adequacy of criteria and analysis of clad performance in BWR power oscillations; NRC needs to see if this problem can be solved by analysis with minimal experimental confirmation.
Y6195	*Dry Cask Storage License for High Burnup Fuel*	Develop criteria for dry-cask storage and transportation of spent high burnup fuel.

7 NEUTRONICS AND CRITICALITY SAFETY

Neutronics and criticality safety are areas in which NRC must maintain exceptional capabilities through its research program. The neutronics and criticality safety research program is small but appears adequate to ensure that the NRC has capabilities to meet immediately foreseen regulatory needs. The current NRC research activities in neutronics analysis, core physics, and criticality safety are listed in Table 5. Maintenance of the SCALE suite of codes is essential for the analysis of reactor core physics. These codes are complemented by the PARCS code which is part of the TRACE code and is discussed in more programmatic detail in the Chapter 14 of this report dealing with Thermal Hydraulics Research. The availability of the NEWT lattice code is important to licensees since it will be essential for the use of more advanced computer models in future regulatory processes. Currently, this lattice code is being used for the analysis of reactor cores fueled in part with MOX fuel for the disposition of excess weapons-grade plutonium. Several other activities are under way to support the licensing of MOX fuel core at the Catawba reactor for this plutonium disposition activity. These are appropriate programs at the current time. It is noted that NRC is taking advantage, to the extent feasible, of the considerable European experience with MOX fuel made with reactor-grade plutonium.

In the future, more innovative core designs for advanced reactors may be submitted to the NRC. Confirmatory analyses of reactor core physics will be an essential part of the regulatory process for these advanced reactors. The capabilities now available to the NRC in the area of core physics may well be stretched. It will be useful to the agency to understand these future needs. If long-term development activities are identified, such as those that might be needed for analysis of the PBMR, additional research may be needed in this area.

Table 5. Research Activities in Neutronics Analysis, Core Physics, and Criticality Safety

Job Code	Title	Comment
Y6846	***SCALE Code Development for Reactor Physics***	Essential code for neutronics analysis to audit licensee submittals and other regulatory needs.
Y6320	***NEWT Lattice Code***	Generate lattice cross-sections for safety analysis of MOX cores to support licensing of cores for Pu disposal.
N6162	***MOX Benchmark***	Confirmation of uncertainties in PARCS code predictions of MOX core neutronics; Also supports the licensing of Pu disposal activities.
Y6403	***Reactor Core Analysis***	Analysis to predict details of reactivity transient in MOX core. Again, this research supports regulatory activities associated with the DOE program to dispose of excess weapons-grade plutonium.
Y6685	***Experimental Data for High Burnup Spent Fuel Validation***	This project provides NRC with foreign and domestic data on high burnup fuel and MOX fuel for assessment of analytical tools used to predict fuel inventories, decay heating, and radiation shielding.

8. HUMAN FACTORS AND HUMAN RELIABILITY RESEARCH

Human performance plays a critical role in the safe operation of nuclear plower plants. Human performance issues have been main contributors to accidents and unsafe conditions experienced by the current fleet of operating reactors. They can be expected to continue to have a major impact on nuclear power plant safety. As licensees increasingly rely on risk-informed licensing applications that require the quantification of human reliability under accident conditions, the staff needs to be able to evaluate the treatment of operator actions in such applications. As new reactor designs, likely dependent on a higher degree of automation than the current fleet, are introduced, the need for revised guidance and tools for the NRC staff in human factors and human reliability analysis will increase. Therefore, it is very important that the NRC maintain research programs in these areas.

The current NRC research activities in the areas of human factors and human reliability analysis are:

- Human Factors
  (B7488, N6207, Y6843, N6137, Y6529)

- Human Reliability Analysis
  (Y6497, Y6496, N6248)

Current research in the human factors area includes a continuing international collaborative research program at the Halden project (B7488). The ACRS is supportive of this collaborative program and recommends continued NRC participation.

The project "Development of a Regulatory Guide and Analytical technique for Assessing NPP Staffing" (N6207) supports the development of guidance for staffing exemption requests to 10 CFR 50.54(m). This project is almost complete. Guidance is now provided in the recently issued NUREG-1791, "Guidance for Assessing Exemption Requests from the NPP Licensed Operator Staffing Requirements Specified in 10 CFR 50.54(m)." Publication of this guidance is a significant accomplishment that provides a more flexible approach to staffing of current and future reactors.

Human performance issues, including organizational issues are of great importance to nuclear reactor safety. Inspectors at nuclear power plants currently have limited guidance or means with the Reactor Oversight Process (ROP) to characterize problems associated with human performance. This issue has been highlighted in a recent report from the Inspector General. In response to a Commission request, the project Y6843, "Develop Human Performance Indicators,"has been initiated to study the feasibility of establishing the technical bases for indicators of human performance that would be used to supplement indicators currently used in the ROP. This research is appropriate and very important. It may lead to significant improvements in the NRC inspection program and the ROP.

There is evidence of degrading performance of operations personnel in the nuclear and other industries due to operator overload. The research project N6137, "Impact of Operator Workload on Human Performance," is a five-year effort to assess the impact of operator overload on performance. The plan is to develop licensing requirements as well as inspection guidance and techniques for reviewing the impact of workload on operator performance and plant safety. This is an important new project that deserves support both for the current fleet of operating reactors and for advanced reactor designs.

Advanced reactor designs are likely to introduce much greater automation than exists in current reactors. Certainly, advanced digital control and instrumentation methods as well as new human-system interfaces can be anticipated. These new features of plants are likely to have some effects on human performance. The NRC staff needs to prepare itself to review new concepts and designs proposed by licensees. The project "Human Factors of Advanced Reactors" (Y6529) has been initiated to address this issue and to develop regulatory guidance and analytical techniques to review human factors for advanced nuclear power plants. The ACRS views this five-year project essential for preparing the staff in reviewing advanced reactor designs.

The quantification of human reliability continues to be a challenge in risk assessments. Many approaches to the quantification of human reliability have been proposed. However, the benchmark exercise conducted by the Ispra Laboratory of the European Union demonstrated that the choice of model has a significant impact on the results obtained. Not much progress to improve this situation has been made since that exercise was performed. The NRC staff has recently completed an assessment of the strengths and weaknesses of the various methodologies now available for assessing human reliability. The ACRS has been quite impressed with this assessment and hope the work leads to the identification of best methods for the quantification of human reliability in PRA.

Human reliability modeling introduces large uncertainties in PRAs. The NRC staff needs guidance in its review of the human reliability models used by the industry in licensing applications. The project Y6497, "HRA Application and ATHEANA Maintenance," is intended to improve NRC's ability to independently model human reliability and to provide guidance concerning risk-informed regulatory applications. Progress has been made with the publication of NUREG-1792, "Good Practices for Implementing Human Reliability Analysis (HRA)." Still, further guidance is needed for reviewers of licensing applications. The NRC has applied ATHEANA model to the human performance issues associated with its recent pressurized thermal shock study. The NRC is also planning to apply the ATHEANA model to a number of ongoing risk assessments, including those for fire and steam generator tube rupture to develop lessons learned on human reliability analysis and to develop guidance for the staff. If needed, modifications to the Standard Review Plan for licensee's applications will be devised. The ACRS believes that this effort is needed. ATHEANA is a state-of-the-art model of human performance and is complicated to use. Application of the tool will show whether benefits derived from the analyses are commensurate with the enhanced complexity. Application may also show how the complexity of ATHEANA can be reduced. Application of ATHEANA is, however, very much behind schedule. Resources and management attention are needed to either accelerate the efforts or to revise the scope of the application efforts.

Both ATHEANA and SPAR-H (the HRA model used in SPAR) quantify the probability that a human unsafe act will be committed. This probability depends on a number of performance shaping factors (PSFs) that determine the context within the crew operates. The available time for action is one of the PSFs estimated from thermal-hydraulic considerations. The evaluated failure probability ia understood to be the probability that the required action will not be completed within the available time.

An alternative approach to HRA is to recognize the importance of time taken by the crew to complete a task and to develop a probability distribution for this time. The failure probability, then, is calculated from this distribution as the probability that this time will exceed the available time.

Recent experiments performed at Halden, Norway, have shown that there may be significant variability in the time that crews take to perform a given task. Such evidence is very difficult to account for in ATHEANA and SPAR-H. The alternative approach could accommodate such evidence. In addition, the staff is currently supporting research at Idaho National Laboratory (INL) that develops "time lines" for past accidents. This evidence can also be accommodated in the alternative approach.

The staffshould evaluate the merits of an HRA model that focuses on the time required for action.

The project Y6496 is a continuing effort to develop an event database called Human Event Repository and Analyses. This database and analysis capability should significantly improve the treatment of human reliability in nuclear reactors and provide a realistic, performance-based database to assess licensee's quantification of human performance. This effort should be sustained and made an ongoing part of the research program.

The project N6248, "Advanced Reactor HRA Development," is the first year of a proposed five-year effort to develop HRA methods and tools to support an independent staff review of human reliability analyses submitted as part of new reactor licensing applications. Given the importance of human factors to reactor safety and the likelihood that new reactor designs may significantly alter the role of operators and the human-system interface, this project is valuable and should be continued to completion.

Table 6. Human Factors and Human Reliability Research Activities

Job Code	Title	Comment
Y6497	*HRA Application and ATHEANA Maintenance*	Apply ATHEANA to Fire Risk Requantification; upgrade and improve ATHEANA. ATHEANA is NRC's tool for analysis of human reliability. Application of this tool will allow assessment of its worth.
Y6496	*Human Event Repository and Analysis*	Develop a human event repository and analysis tools. This program develops a useful data-base for comparison to model predictions of human events.
B7488	*Halden Reactor Project*	International collaborative research project that addresses man-machine interaction and verification and validation of software, surveillance and support systems, advanced control rooms and fuels and materials. This international effort helps keep staff aware of international developments in human factors and human reliability.
N6207	*Develop Reg. Guide and Analytical Technique for assessing NPP staffing*	Support development of guidance for staffing exemption requests to 10 CFR 50.54 (m). This is an important program as licensees look at manpower costs associated with nuclear power plant operations.
Y6843	*Develop Human Performance Indicators*	Determine availability and viability of human performance indicators for assessing performance at nuclear power plants; This program was undertaken in response to a Commission SRM.
N6137	*Impact of Operator Workload on Human Performance*	An important new effort to assess the impact of operator overload on operator performance and plant safety.
N6248	*Advanced Reactor HRA Development*	The first year of a proposed five -year effort for addressing human performance issues for new reactors. This is a valuable project and should be continued to completion.
Y6529	*Human Factors of Advanced Reactors*	Develop regulatory guidance and analytical techniques to review human factors for advanced reactors. Essential work to prepare the staff in its review of advanced reactor designs.

9 MATERIALS AND METALLURGY

Research in the area of materials and metallurgy is an important focus of the NRC Safety Research Program. Current research activities are concentrated in five areas:

- Environmentally Assisted Cracking in Light Water Reactors
(Projects K6266, K6202, Y6270, Y6388, N6007)

- Steam Generator Tube Integrity
(Projects Y6536, Y6588)

- Non-destructive Examinations
(Projects Y6534, Y6604, Y6649, Y6869, Y6867, Y6541, N6019)

- Proactive Materials Degradation Assessment
(Project Y6868)

- Reactor Pressure Vessel Integrity
(Projects W6953, Y6533, Y6378, Y6638, Y6951, N6204, Y6870, N6223, Y6485, Y6656)

These projects represent a significant investment by the NRC to better understand the issues of materials degradation in the currently operating fleet of nuclear power plants. Such investment is justified in view of significant agency regulatory activities that aging degradation research supports. As plants age, known degradation mechanisms will continue to affect components and new degradation mechanisms may develop. The current program is well focused on improving the agency's ability to independently evaluate licensees' efforts to prevent, detect, and mitigate environmentally assisted stress corrosion cracking. The Proactive Materials Degradation Assessment project is an effort to identify potential material degradation problems before they manifest in operating nuclear power plants.

Unfortunately, the planning of NRC's research in materials and metallurgy is not well documented in the way planning for research on digital instrumentation and control systems has been documented. It is, then, difficult to explain the role and priority of each task within each of the five project areas. In aggregate, the activities in the first four project areas (Environmentally Assisted Cracking, Steam Generator Tube Integrity, Non-destructive Examinations, and Proactive Materials Degradation Assessment) seem to be appropriate. These are the very areas that most challenge the industry and its ability to detect component degradation. The agency must develop the capabilities to assess the acceptability of the industry's initiatives to deal with these degradation challenges. The five project areas are further discussed below.

Environmentally Assisted Cracking

Environmentally assisted cracking is a complicated technical issue that continues to afflict the industry as components age and irradiation effect increases. In recent years, the industry has experienced irradiation assisted stress corrosion cracking (IASCC) of components internal to the vessels of boiling water reactors (BWRs) and stress corrosion cracking of reactor vessel head penetration assemblies in pressurized water reactors (PWRs). Although the industry has responded to these events with initiatives to prevent and mitigate these types of degradation, the event at Davis-Besse makes it readily apparent that the NRC staff must be capable of independently evaluating the adequacy of licensees' initiatives. The research projects now under way seem well designed to ensure that the NRC has the needed technical understanding of the stress corrosion cracking issues.

The project Y6388, "Environmentally Assisted Cracking of LWRs," evaluates environmental effects on fatigue of steels used in light water reactors and provides the NRC with technical data and analytical methods to assess licensees' plans concerning mitigation. The large effort includes tests of neutron-irradiated specimens to improve the understanding of IASCC initiation and stress relaxation. It also provides data on the performance of probes and monitoring techniques in radiation environments. This work is essential and should be continued. A new project, "Investigation of Stress Corrosion Cracking in Selected Materials" (N6007), will develop a better understanding of stress corrosion cracking in PWRs. Such cracking occurs typically in the reactor coolant system boundary. Understanding of such cracking in this boundary is essential for maintaining the defense-in-depth.

Environmentally assisted corrosion of reactor materials is an international concern. The CIR-II Cooperative Agreement (K6202) is a collaboration with the international community for studying the susceptibility of stainless steel to IASCC. Certainly, this collaboration should be continued.

Steam Generator Tube Integrity

Rupture of steam generator tubes in PWRs can lead to accidents that allow radioactive materials released from the core to bypass the reactor containment and enter directly into the environment. Severe accidents involving containment bypass can be risk dominant at some PWRs. Through the years, many modes of corrosion of steam generator tubes have been experienced. Regulations on the corrosion were developed when erosion was the dominant concern. Careful water chemistry control by licensees has largely eliminated erosion as a safety concern. But, now, stress corrosion cracking has emerged as the dominant threat to the integrity of steam generator tubes. Incipient stress corrosion cracking is much more difficult to detect. NRC has two research projects to deal with the degradation mechanisms in steam generator tubes, "Steam Generator Tube Integrity Program" (Y6588) and "PWR Primary System Components Severe Accidents" (Y6536). The first project, Y6588, deals with potential tube degradation modes, their resulting leak rates, and the effectiveness of in-service inspections. The second project, Y6536, seeks to improve methods and models used to predict the behavior of degraded steam generators and other PWR components under severe accident loads. Both of these research efforts are important and should be continued.

Non-destructive Examinations

Non-destructive examinations are relied upon to monitor the integrity of the reactor coolant system. The reliability and effectiveness of existing non-destructive examination techniques remain open to question. Certainly, a steam generator tube cracking incident at the Indian Point reactor emphasizes this point. Four projects are under way to improve non-destructive examination techniques (Y6534, Y6604, Y6649, and Y6869) and this work should continue. Two of these projects deal with the effectiveness and reliability of non-destructive examination of reactor vessel penetration assemblies. As the ACRS noted in NUREG-1635, Vol. 6, this is an area that needs increased attention. A third project will provide destructive examination data that should be of tremendous value for the validation of non-destructive examination methods. The project, N6019, will examine non-destructive methods and leak monitoring techniques and the requirements for light water reactor components that have experienced degradation or have been identified as being susceptible to future degradation. The project "Evaluate Reliability and Effectiveness of Advanced NDE," Y6541, will support continued investigation of innovative methods to detect incipient amounts of wastage of ferritic steel. All of these projects are responsive to the NRC's needs and should be continued.

Proactive Materials Degradation Assessment

The nuclear industry and the NRC have often been surprised by unexpected material degradation problems. As a result, they have responded to such problems in a reactive mode which has proven to be inefficient. Reactive response does not enhance public confidence in the safe operations of nuclear power plants. The project "Proactive Material Degradation Assessment" (Y6868) is an NRC initiative to identify materials and locations in light water reactors where degradation can reasonably be expected in the future. The goal of this project is to develop the technical bases needed to implement regulatory actions to proactively address materials degradation problems. Current inspection and monitoring programs at plants can be reviewed and modified as needed to provide earlier identification of incipient degradation before it affects plant safety. The ACRS admires the vision of this undertaking and supports its continuation. The ACRS looks forward to reviewing the initial results of this ongoing effort soon and learning whether the admirable goal of this project is, in fact, feasible.

Reactor Pressure Vessel Integrity

The integrity of the reactor pressure vessels has been studied for decades. Maintaining the structural integrity of the reactor pressure vessel in a nuclear power plant during both routine operations and during postulated upset conditions, including pressurized thermal shock situations, is a longstanding obligation of licensees. This obligation is codified in three general design criteria (GDC 14, GDC 30 and GDC 31) as well as in 10 CFR 50.61 and the appendices G and H to 10 CFR Part 50. Technical bases for these requirements were largely established in the 1980s. NRC is continuing to devote substantial resources to the study of pressure vessel embrittlement though there does not seem to be a comparable interest within the industry who will have most of the research benefits. Indeed, the number of projects in this area seems to have grown since the ACRS last reviewed the NRC research program and questioned the need for research in the area of reactor pressure vessel integrity.

Some of the activities in this programmatic area deal with the finalization of the NRC's work on pressurized thermal shock which is nearing completion. These activities will contribute to the potential revisions of Regulatory Guide 1.99 on radiation embrittlement of reactor pressure vessel materials and Appendices G and H to 10 CFR Part 50 on fracture toughness requirements and reactor surveillance needed to ensure low probability of reactor vessel failure.

The project "International Pressure Vessel Technical Cooperative Program" (Y6378) will ensure NRC participation in the International Atomic Energy Agency (IAEA) deliberation on reactor pressure vessel integrity.

The NRC's comprehensive program on reactor pressure vessel integrity has produced significant results by providing better understanding of the available margin in reactor pressure vessel components. Revisions to PTS screening criterion in the PTS rule and the associated regulatory guides and Appendices G and H to 10 CFR Part 50 are likely to provide great benefit to licensees by relaxing current requirements and allowing longer life of reactor pressure vessels. These activities should be completed soon.

RES needs to reevaluate the need for continued research into heavy section steel components. This research may be justified if there is a clear need for NRC to develop its capabilities in the area of probabilistic fracture mechanics so that it can evaluate licensees' applications. If this is the case, the research needs to be clearly focused on this objective and not the research that the industry should perform to meet its responsibilities to ensure reactor pressure vessel integrity. It appears now, however, that it is NRC that is advancing the state-of-the-art and making available information that allows licensees to reduce conservatism in their analyses.

Table 7. Research Activities in Materials and Metallurgy

Job Code	Title	Comment
Environmentally Assisted Cracking in LWRs		
K6266	***CIR-II Cooperative Agreement***	NRC contribution to international research on irradiation assisted stress corrosion cracking.
K6202	***Extension of CIR-II Cooperative Agreement***	Assess the susceptibility of stainless steels to Irradiation Assisted Stress Corrosion Cracking. This program allows NRC to stay abreast of international developments.
Y6270	***Environmentally Assisted Cracking***	Provide neutron irradiated specimens for NRC research programs.
Y6388	***Environmentally Assisted Cracking of LWRs***	Develop data on irradiation assisted stress corrosion cracking in PWRs and BWRs. This program provides NRC staff with the data and analytical methods to review licensees' activities and plans to limit corrosion.
N6007	***Investigation of Stress Corrosion Cracking in Selected Materials***	User need for a better understanding of stress corrosion cracking in PWRs. This program supports the regulatory process.
Steam Generator Tube Integrity		
Y6536	***PWR Primary System Components Severe Accidents***	Methods and models to predict PWR reactor coolant system component behaviors under severe accident loads; This is an essential research program.
Y6588	***Steam Generator Tube Integrity Program***	Wide-ranging program in support of the steam generator integrity action plan. ACRS supports this action plan and regularly monitors its progress.

Table 7. Research Activities in Materials and Metallurgy (Continued)

Job Code	Title	Comment
Non-destructive Examinations		
Y6534	***Piping NDE Reliability***	Program addresses Inconel cracking in weld metal and base metal. This is an essential program to ensure licensees adequately monitor nickel alloys in plants.
Y6604	***Evaluate Reliability of NDE Techniques***	Addressing the inspection of cast stainless steel components and dissimilar metal welds; evaluation of reliability and accuracy of in-service inspection. This is an essential program to facilitate NRC monitoring of licensee activities.
Y6649	***Phase II - Alloy 600 Cracking***	Independent assessment of industry analyses of CRDM nozzle cracking. This is a classic NRC program of confirmatory research.
Y6869	***Barrier Integrity Research Program***	Evaluate RCS leakage experience and leak detection capabilities. This is an essential program to facilitate NRC monitoring of licensee activities.
Y6867	***Cooperative Activities Reactor Coolant System Pressure Boundary Components***	Complete non-destructive examinations of nozzles from vessel heads. Plan destructive tests. This is an important program to validate analyses NRC uses in its regulation of licensee activities.
Y6541	***Evaluate Reliability and Effectiveness of Advanced NDE***	Identify innovative NDE techniques in coordination with industry and international community. This program allows NRC staff to stay abreast of international developments in NDE.
N6019	***NDE & Leak Monitoring Requirements***	Assess adequacy of current inspection and monitoring requirements. Assemble data on probabilities of failure of passive components. This is an essential program to facilitate NRC monitoring of licensee activities.

Table 7. Research Activities in Materials and Metallurgy (Continued)

Job Code	Title	Comment
Proactive Materials Degradation Assessment		
Y6868	***Proactive Materials Degradation Assessment***	Identify materials and locations in LWRs where degradation can reasonably be expected. This program is intended to better equip NRC to anticipate materials degradation problems at nuclear power plants. This program should be continued. The ACRS looks forward to reviewing the initial results.
Reactor Pressure Vessel Integrity		
N6204	***Review and Revisions of Pressurized Thermal Shock Reports NUREGs 1806 and 1809***	Support documentation of thermal hydraulics analyses for pressurized thermal shock, and document Calvert Cliffs RELAP5 calculations to support FAVOR calculations. This program should be completed.
Y6485	***Technical Support - Pressurized Thermal Shock Rulemaking***	Support for the pressurized thermal shock rulemaking effort. This is essential support for the regulatory process.
W6953	***Heavy-Section Steel Irradiation Program***	Evaluation of Master Curve methodology for reactor pressure vessels. The ACRS questions the need for the large investment in heavy section steel research.
Y6870	***Cooperative Program on Irradiation***	Development of a cooperative program with DOE to study reactor pressure vessel materials.
Y6378	***International Pressure Vessel Technical Cooperative Program***	International cooperative effort to understand embrittlement of reactor pressure vessels and other components. This program will keep staff aware of international developments in reactor pressure vessel integrity.
Y6533	***HSST-3 (Heavy Section Steel Technology)***	Development of fracture mechanics methodologies; The ACRS questions the need for the large investment in heavy section steel research.

Table 7. Research Activities in Materials and Metallurgy (Continued)

Job Code	Title	Comment
Y6951	***Fracture Mechanics Technology for LWR***	Fracture mechanics of heavy section steel. The ACRS questions the need for the large investment in heavy section steel research.
Y6638	***Statistical Analysis of RPV Steels***	Assist NRC staff in developing a revision to Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials." This research directly supports the regulatory process.
N6223	***FAVOR 4.1 Sampling Validation***	Validation of new features of the FAVOR computer code for fracture analysis of vessels. FAVOR is NRC's computer code for fracture mechanics analysis and is used extensively.
Y6656	***Risk Inform Appendices G & H***	Develop a risk-informed revision to 10 CFR 50, Appendix G on Fracture Toughness Requirements and Appendix H on Reactor Vessel Material Surveillance Program.
N6227	***SMIRT-18 Conference Registration***	Costs associated with presentation of papers on NRC research projects at the Structural Mechanics in Reactor Technology meeting.
N6097	***SMIRT 18***	Financial support to publish proceedings of the 18th International SMIRT conference.

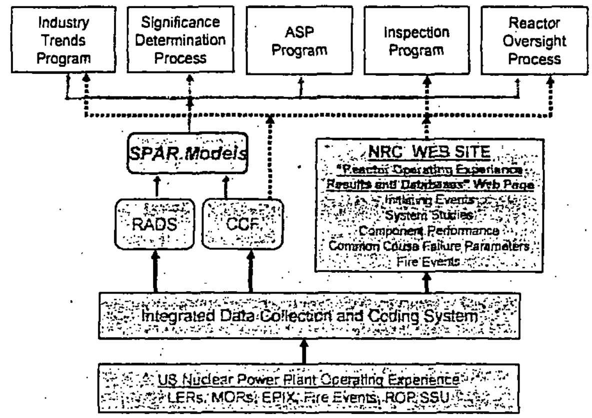
10 OPERATIONAL EXPERIENCE

The analysis of operating data is a cornerstone in the NRC's increased use of risk information in regulatory processes. Such analysis provides current information on initiating events, component failure data, and the risk profiles of licensees. Comparison of these results to goals in the agency's Strategic Plan provides a measure of regulatory effectiveness and inputs for the agency's annual report to Congress on significant operating events.

The NRC research activities associated with operational experience are listed in Table 8. The Accident Sequence Precursor (ASP) Program, Y6815, and the Industry Trends Program, Y6546, alert the staff and industry to component failures as old or replacement components age or operations change. Data derived from operating experience will validate or refute the assumption that aging management programs are sufficient to ensure the operability of both active and passive components. The operating experience programs provide data that can be the bases for regulatory decisions to improve safety. These programs also support the Reactor Oversight Process, including the determination of the safety significance of inspection findings and the development of industry performance indicators.

Two tasks in the research of operational events, "Method to assess Effect of Design and Operations Margins," N6082, and "Procedure Development for External Events," Y6814, are important efforts to extend the use of quantitative risk assessment into external events, including fire, and low power and shutdown operations.

ACRS is supportive of the research activities in the area of operational experience and recommends that these activities be continued. In light of the limited resources



Uses of Operational Data and Analyses in Regulatory Activities

allocated to these tasks, RES has done a commendable job in producing outputs in well-documented and thorough fashion. Tasks that are currently in the 2005 Research Plan related to Operational Experience should remain funded and should be continued for the foreseeable future.

Staff engaged in the collection and analysis of operating experience data might also be able to improve the state-of-the-art in PRA modeling. Specifically, they might be able to use operating experience data to derive higher resolution models of system and component operability. Currently, PRAs use success criteria models. A system or component that meets the success criteria is deemed operable This "go/no go" model is not entirely realistic. There is no assessment of margins, equipment aging, changing plant conditions, etc. Success criteria models may not provide adequate answers for some applications such as power uprates, containment overpressure credit, license renewal, sump screen clogging, or any set of plant conditions that are in some way off-normal or even outside the design specifications of the equipment. There have been several events that were surprises because the phenomena that caused or contributed to the failure mode had not been realistically modeled. Certainly, the recent Davis-Besse event involving corrosion of the reactor pressure vessel head penetrations comes to mind. Staff granted a small extension to ordered shutdown date for reactor pressure vessel penetration inspections. They did so, in part, because the calculated risk was small. Unfortunately, the phenomenological modeling of the head penetrations and their corrosion was incorrectly used in the risk assessment.

Development of improved models of system and component operability models will require that choices be made concerning areas where improved modeling will yield useful improvements in the risk predictions. The issues of interest may themselves dictate where choices for improved modeling should be made. Some modeling improvements are being made now on an *ad hoc* basis. There is no need to continue to do so if a more structured approach could result in better models with wider applications.

Table 8. Research Activities in Operational Experience

Job Code	Title	Comment
N6082	***Method to Assess Effect of Design and Operations Margins***	Provides a methodology to assess the effects of changes to design and operation on plant safety margins. This program provides direct support for the regulatory process.
Y6468	***Reactor Operating Experience Data for Risk Applications***	Collect operational data for reactor systems, components, initiating events, common-cause failures and fire events. Data collected in this program is of use for validation of PRA models.
Y6546	***Industry Trends Program***	Includes grid concerns. This is an essential program for NRC.
Y6864	***Operating Event Technical Support***	Support for technical expertise in operating events.
Y6816	***SDP/ASP Standardization***	Develop analysis guidelines for operating events during low power/shutdown conditions. This program will extend the ASP program to include events during shutdown operations.
Y6815	***Accident Sequence Precursor Analysis***	Systematically screen, review and evaluate operating events. This is a flagship program at NRC.
Y6987	***Expert Elicitation Process - Accident Sequence Precursor Program***	Develop guidelines for obtaining and using expert opinion in ASP analyses. The useful elicitation of expert opinion is of growing importance in the risk-informed regulatory system.
Y6814	***Procedure Development for External Events***	Expand the scope of ASP analyses to include the calculation of risk from external events and from low power and shutdown modes of operation. This program will help extend the scope of the ASP program.

11 PROBABILISTIC RISK ASSESSMENT

Probabilistic risk assessment has become an essential technology for NRC as it evolves the regulatory system to make greater use of risk information. The NRC research activities in probabilistic risk assessment are shown in Table 9. Probabilistic risk assessment has become pervasive within the research program. Other activities nominally part of the development of PRA are addressed in other Chapters of this report. See especially the discussions of Digital Instrumentation and Control Systems (Chapter 4), Fire Safety Research (Chapter 5), Human Factors and Human Reliability Research (Chapter 8), and Operational Experience (Chapter 10). The staff involved in PRA research has been extraordinarily productive since the last ACRS report the NRC research program. A major focus of the current PRA research is to support the ROP, which uses risk information for monitoring the operations of nuclear power plants and acting on inspection findings and deviation of performance indicators from established thresholds.

The ROP makes heavy use of the SAPHIRE computer code and the SPAR models of specific plants. The SPAR model development program has become an essential element of the ROP. The ability to develop a SPAR model for each nuclear power plant has only been feasible because of the existence of Level I, internal events, PRAs for each plant. Each SPAR model begins with a basic model of a plant system for a generic category of plants (e.g., a BWR4 reactor with a Mark I containment). The SPAR model is then made plant specific through upgrades based on discussions with the licensee. NRC has found it essential to develop its own risk-assessment model for each plant as a matter of practicality. It would be difficult for the NRC staff to take a variety of plant PRAs, which use different platforms and approaches, make them operational at NRC, and have knowledgeable staff available to execute and update each plant model. NRC development of SPAR models for individual plants has also enhanced the plants' risk assessments.

A major issue that confronts the use of risk information in nuclear power plant regulation is the question of incompleteness of individual plant risk assessments. The Individual Plant Examination (IPE) program and subsequent evolutions at the nuclear power plants led to development of Level I, internal events, PRA models of all of the operating. These PRAs meet (or with modest effort can meet) the requirements of industry standards for internal events PRAs. The same is not true for the assessment of risk from fires, floods, seismic events and for plant modes of operation that differ from full power operations. Furthermore, the capabilities to assess risk at Level II, radionuclide release and source terms , lag far behind the Level I capabilities.

The NRC staff has plans to expand the scope of the SPAR models to include treatment of risks from fire-initiated events, seismic events and shutdown modes of operations. These plans are, however, not well developed. There is furthermore the question of availability of resources needed to undertake these efforts. The expansions of the scope of SPAR models will be challenging because all licensees do not have sophisticated risk assessments in these areas for comparison and validation of NRC's SPAR models with expanded scope. The NRC staff could develop generic models accounting for the major features of the plant designs, but the staff would not be able to upgrade the generic models to become plant-specific models as was done for the treatments of risk from internal events. In addition, fire and seismic risk assessments differ qualitatively from internal events risk assessments since the events occur in "areas" of a plant and affect multiple systems rather than just specific components in specific systems. Fire and seismic risk assessments require detailed knowledge of spatial relationships in addition to functional relationships. Spatial relationships, of course, vary substantially even among plants of the same generic type. Despite these challenges, the regulatory oversight value of full-scope SPAR models is very high. Over the next year, the staff should develop its approach and plans for the expansion of the scope of the SPAR models to treat external events, shutdown modes of operation and even to go to Level II analyses that include accident progression and the release of radionuclides to the environment. Even if it is not possible to have plant-specific models in the near term, the generic shells should be available and can be adapted to be plant specific in the future or can be upgraded in particular areas to address specific regulatory issues.

Another barrier to the greater use of risk assessment in the regulatory process is the question of uncertainty in the risk predictions. There are, of course, parametric uncertainties and the agency has active programs to better understand the important parametric uncertainties (See especially Chapter 10, Operational Experience). There are also issues of uncertainty in the models adopted in PRA. Uncertainties in the models of human reliability and passive system reliability are significant examples. It has become common now for the NRC and the licensee to agree upon a model appropriate for particular regulatory activities. This agreement can often be based on familiarity or expedience. The disturbing trend is for the staff to conclude, then, that there are no longer uncertainties associated with the results predicted by the agreed upon models. Staff needs to ensure that it treats uncertainty in risk assessments in a more defensible manner. Research needs to provide the tools and understanding so that this can be done.

The staff has also been revising 10 CFR 50.46 to account better for risk information. This is challenging and important work. Even more challenging is the effort to develop a "technology-neutral" alternative to the current regulatory framework. The ACRS views such a technology-neutral regulatory framework as essential in the future and feels that it needs more attention.

Altogether the scope and the number of PRA research activities are quite impressive. The ACRS cautions, however, that NRC should not allow its work in such a crucial technology as risk assessments become totally devoted to the support of line activities. Methods development is still important. As an example, the ACRS notes that considerable research is being reported in the literature regarding Binary Decision Diagrams as tools for solving large fault tress without resort to cutoff frequencies as is now done. Some researchers report that the unavailability of highly redundant systems could be underestimated significantly when cutoff frequencies are used for the analysis. Although no definitive evidence has yet been produced to show that methods used in the NRC's SAPHIRE code are inadequate, the staff needs to review the literature concerning Binary Decision Diagrams and evaluate the need to adopt this technology. The growing importance of the SAPHIRE code and the SPAR models in the regulatory process warrants such an investigation.

Table 9. Probabilistic Risk Assessment Research Activities

Job Code	Title	Comment
N6027	*PRA for Dry Cask Storage Follow Up*	A variety of tasks including uncertainty analysis and extension to multiple casks. This program supports licensing and inspection oversight of cask vendors.
N6105	*Guidelines for the Communication of Risk Information*	Complete the technical basis for the internal risk communication guidelines. This task completes the technical basis for internal risk communication guidelines. The ACRS remains concerned that publically available information on risk analyses may not be sufficient to ensure public confidence in a risk-informed regulatory process.
Y6842	*Guidance for the Development of Latent Errors*	Quantitatively assess the importance of latent errors and the treatment of latent errors in PRAs. This project has been deferred until FY2007. The ACRS cautions that operating experience shows that latent errors may be four times more common than active errors in important reactor events. The work should not be deferred further.
J8263 Y6370	*Reactor Oversight Process Support* *Development of Risk-based Performance Indicators*	Development of performance indicators to be incorporated into the ROP. Support for the Mitigating Systems Performance Index. These programs support the ROP.
Y6626	*Access to INPO's EPIX System*	Data-base on equipment performance and reliability.
J8258	*International Common Cause Exchange Project*	Sharing of data on common-cause failures with the international reactor safety community. This program keeps staff abreast of international findings concerning common-cause failures.
N6008	*Passive Components Conditional Core Damage Probability*	This program should prioritize passive components for consideration in the proactive materials degradation assessment (Project Y6868, Materials and Metallurgy, Chapter 9).

Table 9. Probabilistic Risk Assessment Research Activities (Continued)

Job Code	Title	Comment
Y6153	*SPAR Model Development: Level2/LERF*	Develop SPAR models for evaluation of large early release frequencies.
N6090	*SPAR Model Development: Shutdown Models*	Develop logic models for analyzing low power and shutdown internal events.
W6355	*SPAR Model Development: Low Power Shutdown*	Identify methods to characterize risk during low power or shutdown operations.
W6467	*SPAR Model Development: Level 1 Rev. 3 Models*	Revision of Level 1 SPAR models to better reflect as built and operated plants.
Y6595	*SPAR Model Development: External Events Analysis*	Development models of external events for the SPAR codes
N6075	*SPAR Model Development: Enhanced Level 1, Revision 3 Models*	These are important programs to support the expanded scope of the SPAR models.
Y6394	*Maintain and Support SAPHIRE Code and Library of PRA*	Testing to ensure that SAPHIRE is a state-of-the-art PRA code.
N6172	*Participate in the MERIT Program (Maximizing Enhancements in Risk Informed Technology)*	Base program supports risk informing 10 CFR 50.46 and includes development of a probabilistic LOCA code, non-piping component degradation, and pressurized water stress corrosion cracking. This international program supports one of the important NRC initiatives.

Table 9. Probabilistic Risk Assessment Research Activities (Continued)

Job Code	Title	Comment
N6111 Y6538	*Technical Support for 10 CFR 50.46 Task Order 3* *Technical Development of LOCA Frequency Distributions*	Quantification of the effect of break size reduction and alternative break locations on margin to existing alternate acceptance criteria Provide LOCA frequency estimates for use in revision of 10 CFR 50.46. These programs are needed to support risk informed revisions to 10 CFR 50.46.
K6081	*PRA Techniques in Risk-informed and Performance-based Regulation*	Develop methods for uncertainty analysis for risk-informed purposes. This is a cooperative agreement with a broad scope. In addition to potential methodological contributions it has an educational value.
N6107	*10 CFR 50.48c related Technical Activities*	In collaboration with EPRI, develop a comprehensive set of risk methods, tools and data to understand and evaluate risks from fires.
W6224 Y6492	*Risk-informing Part 50* *Assess Possible Part 50 Risk-informed Changes*	Develop recommendation on changes to 10 CFR Part 50 to make it risk-informed. Develop recommendations to specific requirements in 10 CFR Part 50 to make them risk-informed. These program support the initiative to risk inform 10 CFR Part 50.
W6970 W6971	*Support to Develop Consensus PRA Standards* *Support in Development of Consensus PRA Standards*	Provide guidance on the use of industry standards for PRA. Revise Regulatory Guide 1.200 based on industry pilots and Revision 1 to ASME PRA standard. These program support the Commission's phased approach to PRA quality.

Table 9. Probabilistic Risk Assessment Research Activities (Continued)

Job Code	Title	Comment
Y6103	***Low Power and Shutdown Risk Study - Level 2***	Program to extend the scope of SPAR models to include accident progression for accidents initiated during shutdown operations. Premature at this point.
N6133	***Development of Consensus on PRA***	Support for staff in development of ANS Low Power and Shutdown operations PRA Standard.
N6134	***Low Power/Shutdown Level 1 and Fire Risk Standard***	Project provides support for staff involvement in the development of ANS standards on PRA for low power/shutdown operations and fire-initiated events.
Y6371	***Risk Associated with Cable Aging***	Addresses the inclusion of aging effects into PRA.

12 SEISMIC RESEARCH

As the design of nuclear power plants improves, the seismic hazard and seismic response of the plants can make an increasingly important contribution to risk. Seismic hazard analysis and structural response are not areas where NRC must maintain state-of-the-art expertise. Such expertise is available to the NRC on a contractual basis. As noted in our previous report, seismic research activities at NRC can be confined to support needed updates to regulatory guides and collaborative work with the international community to stay abreast of developments in other Countries. The current research program is, indeed, largely focused on needs of the regulatory process and a few important international collaborations.

Table 10. Seismic Research Activities

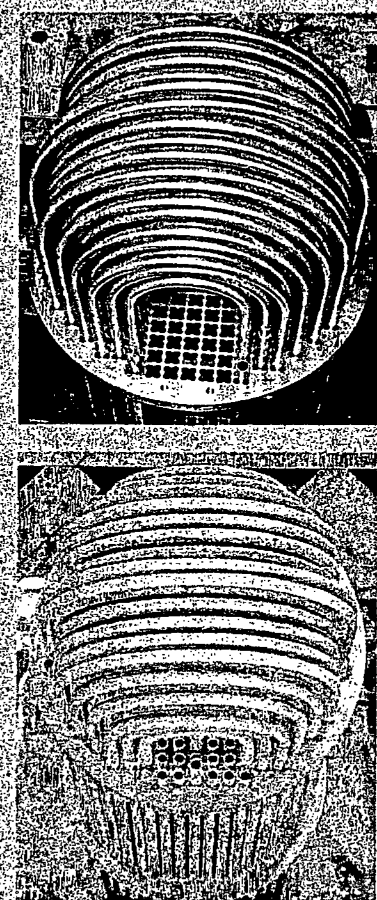
Job Code	Title	Comment
N6020	*Seismic-induced Passive Component LOCA Frequencies*	Review of work by national laboratories and industry on piping degradation and failure under earthquake loads; Work being done to upgrade Regulatory Guides.
Y6481	*SSHAC Method*	10-year update of the Probabilistic Seismic Hazard Assessment used in evaluation of early site permits; work to support update required by regulations.
Y6718	*Soil-structure Interaction for Buried Structures*	Review adequacy of current NRC guidelines concerning soil-structure interactions; work to update Regulatory Guides.
N6112	*Evaluation of Seismic Siting*	Review of ASCE Standard 43-05, "Seismic Design Criteria for Structures, Systems and Components in Nuclear Facilities."
N6076	*Japanese Collaboration on Seismic Issues*	Collaboration with Japan on seismic tests and analyses; Collaborative work give NRC access to extensive work underway in Japan.
W6081	*Japanese Collaboration on Seismic Issues*	Supports work in U.S. in connection with collaboration.
N6102	*Reg. Guide 1.165 Update Technical Basis*	Review of technical advances in the development of seismic response spectra; prepare draft revision to Regulatory Guide 1.165.
N6103	*Enhancement of the CARES Code (Computer Analyses for Rapid Evaluation of Structures)*	The CARES computer code is used to predict the free field and structural response to seismic input.
N6219	*Resolve Regulatory Guide 1.92 Public Comments*	Regulatory Guide provides up-to-date guidance for using the response spectrum and time history methods for estimating seismic response of power plants.
N6104	*Ground Motion Seismic Hazard Studies*	Collection and review of new data on the propagation of earthquake motion in the Central and Eastern U.S.; work to support required update in regulations.
Y6796	*IAEA Coordinated RES Project on Seismic Ground Motion*	NRC contribution to international effort to understand earthquake effects on nuclear power plants. Collaborative effort keeps NRC staff abreast of any international developments.
Y6757	*Containment Capacity Studies*	Confirmatory analyses of structural response and failure modes of containments under extreme loading including seismic loads.

13 SEVERE ACCIDENT RESEARCH

In the past, NRC invested heavily in the experimental and analytical characterization of severe reactor accidents. A substantial technology has been established to understand the progression of severe reactor accidents and the radiological consequences of such accidents. Once its immediate needs were met to understand severe reactor accidents sufficiently well to estimate risks to the level of confidence needed to provide assurance of adequate protection, the NRC substantially curtailed its investments in severe reactor accident research. The current NRC research activities in the severe accident area are listed in Table 11.

Research on severe accidents has been continuing in other countries. Substantial programs are under way in both Europe and Japan. NRC has developed an effective strategy to maintain the technology for severe accident analysis and to update this technology with research results from international programs. The body of knowledge coming the NRC's past work and the ongoing international work are systematized in the useable form in the MELCOR accident analysis code. At the same time, the NRC is entering into international cooperative research programs to obtain data for validating the MELCOR code and improving its accuracy and realism. NRC provides the Cooperative Severe Accident Research Program (CSARP) as a forum for the exchange of severe reactor accident information among Countries. One outcome of this focus of the NRC's research into severe reactor accidents is that many Countries and institutions have adopted the MELCOR code as the preferred tool for the severe accident analysis.

A new version of the MELCOR code has been released to users. NRC is collaborating with researchers in Russia to modernize MELCOR to use FORTRAN 95 coding. MELCOR is



Aerosol Trapping in a Steam Generator (ARTIST)

NRC is participating in ARTIST international cooperative research program to conduct an experimental study in Paul Scherrer Institute in Switzerland to measure the aerosol removal on secondary sides of steam generators during severe accidents at PWRs that bypass reactor containments. Such bypass accidents are often risk dominant for PWRs. The high risks associated with such accidents may stem from conservatism in the aerosol decontamination assumed in accident analysis models for steam generators. Test results are expected to provide the basis for more realistic analyses of these accidents.

being used for licensing actions. The capabilities developed to perform detailed parametric uncertainty analyses with the code are especially attractive.

RES is also maintaining the MACCS code for the analysis of consequences of accidents at nuclear facilities. This code is widely accepted in the U.S. as a tool for consequence analysis. Its maintenance at near the state-of-the-art is important to the agency and the ACRS is supportive of the current research programs.

Collaborative severe reactor accident research programs that NRC has joined are making good technical progress and there have been notable accomplishments in the last 2 years.

- **PHÉBUS - FP**

The Phébus-FP program consists of large-scale prototypic experiments involving the degradation of irradiated reactor fuel, release of fission products as vapors and aerosols, and transport of these fission products through a model of a reactor coolant system into a model of a reactor containment. These are the most prototypic and most comprehensive severe accident experiments that have ever been performed. The last of these tests was completed recently. The experiments have proved to be invaluable for the validation and improvement of the MELCOR code and the validation of the alternative source term used for a large number of licensing actions. The program has revealed a number of unanticipated phenomena and refined understanding of other phenomena. NRC has joined a second-generation program that will involve about 15 Nations to conduct separate effects tests to further understand the important accident phenomena revealed in the PHÉBUS-FP test program. This follow-on program addresses the containment chemistry of radioactive iodine, fission product chemistry in the reactor coolant system, the effects of boron carbide control rods on core degradation and fission product chemistry, and the release of fission products from high-burnup fuel and MOX fuel.

- **ARTIST**

The ARTIST test program is an international collaborative effort undertaken in Switzerland to ascertain the amount of decontamination that can occur in the secondary side of steam generators in PWR accidents initiated by steam generator tube ruptures or initiated by other means but involving steam generator tube ruptures. Such accidents have been found to be risk dominant for some PWRs. During last year, the scoping test program has been completed. Results of the tests show that decontamination is modestly larger than what had been anticipated in accident analyses. Plans are being formulated now to conduct integral system tests and additional tests to support modeling of secondary side decontamination.

- **MASCA**

The MASCA test program and its predecessor the RASPLAV program were undertaken to understand the technical feasibility of retaining core debris within reactor pressure vessels, especially with water flooding the outside of the vessel. These programs were conducted in Russia and involved the development of technology to produce large scale melts of prototypic core debris involving UO_2, ZrO_2, and Zr. The major tests in the program have now been completed. Efforts are under way to identify and maintain the experimental capabilities that have been developed for the MASCA program since these capabilities may be essential for the investigation of severe accidents in reactors that do not use light water technology.

OECD-MCCI

This is an international collaborative experimental study being conducted at the Argonne National Laboratory to investigate the viability of using an overlying layer of water to cool core debris interacting with structural concrete. This program is nearing completion.

Planned modifications of the MELCOR code to address the ACR-700 have been curtailed since the application for certification of this reactor has not been submitted. There still may a need to upgrade the modeling of iodine chemistry in reactor containments to respond to recent findings concerning the effects of tri-sodium phosphate buffer in reactor sumps on sump pump screen blockage.

The ACRS is very supportive of the strategy NRC has developed to maintain and update its capabilities for severe accident analyses.

The leveraging of resources through international collaborative experimental research is especially important. The planned extensions and continuations of current collaborations are well worth the investment. This type of collaboration in experimental research could be emulated in other NRC research areas such as fire safety research and thermal-hydraulics research.

Table 11. Severe Accident Research Activities

Job Code	Title	Comment
Y6321	***Benchmark, MOX Fuel Release, Source Term Experiments***	International Collaborative follow-on to the PHEBUS-FP experiments.
Y6328	***Assessment and Analysis of PHEBUS-ST***	In-kind support for the follow on to the PHEBUS-FP experiments. This work is providing data on fission product behavior during reactor accidents for use in MELCOR development.
Y6628	***Consequence Models and Uncertainty Assessment***	Uncertainty analysis of the MACCS code for computing reactor accident consequences.
Y6313	***OECD-MCCI Program***	International collaborative research on the interactions of core debris with concrete. This program should be completed next year
Y6690	***Analysis Support for OECD-MCCI Program***	In-kind and financial support for the international collaborative research on ex-vessel core debris interactions with concrete.
Y6312	***MASCA Program***	International collaborative research on the behavior of molten core debris in the lower plenum of a reactor vessel. This program has resolved safety issues with respect to invessel retention of core debris. The program has developed the capability to produce and test large-scale melts of uranium dioxide that may be of use in advanced reactor safety model development and validation.
Y6802	***MELCOR Severe Accident Code Development and Assessment***	Computer model for the analysis of severe reactor accident and repository for severe accident research results. This is the agency tool for Level 2 PRA including source term characterization; MELCOR is the repository for severe accident research results obtained by the agency.
Y6721	***AGT W/IBRAE-RAS on Nuclear Safety Analysis Codes***	Support for Russian investigators in the development of a FORTRAN-95 version of MELCOR. This program is modernizing the coding in MELCOR by cost-effective use of expertise in Russia.

Table 11. Severe Accident Research Activities (Continued)

Job Code	Title	Comment
Y6848	*High Burnup Fission Product Release Data*	Refine release models in MELCOR for the effects of high fuel burnup; code analyses will be used to create a licensing source term applicable to high-burnup fuel and reflecting improved modeling of severe accidents.
Y6517	*High Burnup Source Term for Storage*	Establish the technical basis for the extension of regulatory guide on spent fuel heat generation in a spent fuel storage facility to include high-burnup fuel
Y6504	*Steam Generator Fission Product Retention*	International collaborative research on the retention of aerosols on the secondary sides of steam generators in containment bypass accidents (ARTIST program). This program provides an experimental resolution of a long-standing issue of source terms from accidents that bypass containments.
Y6607	*Support ARTIST Tests*	In-kind support for the ARTIST program - see Y6504 above.
Y6486	*Severe Accident Initiated Steam Generator Tube Rupture Sequences*	Investigation of the potential for induced steam generator tube failure during severe accidents leading to containment bypass. This is an important part of the Steam Generator Action plan and the analysis of plant behavior under accident conditions.
Research Programs to Maintain the MACCS Code for Consequence Analysis		
Y6785	*Plume Model Adequacy Evaluation*	Test the assumption that simple plume treatments in MACCS code are adequate by comparing with the state-of-the-art dispersion model. This activity is important to show MACCS is adequate for regulatory needs.
Y6628	*MACCS Uncertainty Assessment for Consequence Models*	Support for emergency planning.
Y6469	*Evaluation of Radionuclide Pathways and Uptakes*	Upgrade information on uptake pathways. This project upgrades the code to take advantage of more recent information.

14 THERMAL-HYDRAULICS RESEARCH

Thermal hydraulics, especially the dynamics of two-phase flow, have always been essential elements of the regulatory evaluation of design basis accidents. NRC confirmatory evaluation of licensees' submittals in the area of thermal hydraulics has long been a major element of many licensing actions. Thermal-hydraulic analyses have grown ever more sophisticated. This trend is likely to continue for existing plants as licensees seek power uprates and take advantage of NRC's willingness to allow best-estimate analyses (with scrupulous attention to uncertainties) in the place of deliberately bounding, conservative analyses. To evaluate the adequacy of the licensees' analyses, NRC must have state-of-the-art thermal-hydraulic computational tools and equally sophisticated understanding of both thermal-hydraulic phenomena and the limitations of computer codes. NRC attempts to maintain its competence in the thermal-hydraulic field through its research program.

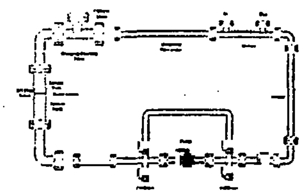
Major elements of the current NRC thermal-hydraulics research program can be grouped into three general areas:

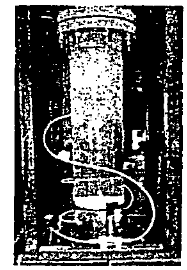
- PWR sump screen blockage issues
- TRACE computer code development
- Experimental studies of thermal-hydraulic phenomena

These major features of the current thermal-hydraulics research program are discussed below.

PWR Sump Screen Blockage

The sump screen blockage issue for PWRs is the analog of a previous issue identified for BWRs. Debris from coatings and insulation can be generated during the high-pressure blowdown of the reactor coolant system





Chemical Effects/Head-Loss Tests in a Simulated PWR Sump Pool Environment

GSI-191 addresses the potential for debris accumulation on PWR sump screens to affect emergency core cooling system (ECCS) pump net positive suction head margin. In response to a concern expressed by the ACRS, RES has initiated a program to investigate the potential for chemical reactions that can occur in the containment pool to produce chemical products that can increase the head losses over those due to the physical debris alone.

NRC and the nuclear utility industry jointly developed an Integrated Chemical Effects Tests (ICET) program to determine if chemical reaction products can form in representative PWR post-LOCA containment sump environment. These tests were conducted by Los Alamos National Laboratory (LANL) at the University of New Mexico (UNM). Chemical products were observed in all five test series.

A head–loss loop was set up at Argonne National Laboratory (ANL) to investigate the potential head loss associated with the chemical products observed in the ICET tests.

These recent research results indicate that a simulated pool environment containing phosphate and dissolved calcium can rapidly produce a calcium phosphate precipitate that, if transported to a fiber bed covered screen, produces significant head loss.

following a major pipe break. This debris can clog the screens protecting the intake pumps for the emergency cooling system and prevent adequate coolant flow. Blockage issues have been exasperated by the discovery of mechanical and chemical effects that magnify the blocking effects of debris trapped on the sump screens. As a result, it is difficult to design screens that are of sufficient size to ensure emergency core cooling. The industry is looking to the NRC for guidance on acceptable methods for sizing screens to protect the sump intakes of the cooling pumps.

The NRC is still in the exploratory phase of research on sump screen blockage. It is still identifying phenomena that affect blockage. It is far from developing tools and methods that can be used with confidence for making predictions. NRC staff is now analyzing the licensees' responses to Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors." These responses should reveal the licensees' views of current predictive techniques and their applicability, as well as indicate what methods they expect to use to assess the adequacy of their current and modified screen systems. The NRC staff needs to have sufficient technical knowledge to evaluate these methods. Current NRC research is focused on significant gaps in knowledge, establishing what phenomena play significant roles, and on developing general awareness of what analytical steps are needed to describe the phenomena adequately. The ACRS would expect that many of the details of predictive methods, such as the coefficients in correlations, computational schemes, and methods for developing suitable conservatism to account for uncertainty, could be left to the licensees or to industry-sponsored organizations such as EPRI. This is possible, however, only when the phenomena are well understood and a technical basis has been established for their prediction. When this is not the case, the NRC may need to develop sufficient predictive ability of its own to achieve authoritative competence to evaluate licensees' submittals.

For example, the NRC-sponsored research has revealed the "thin bed effect". This appears to involve a dense agglomeration of fine particles that fill the pores in a layer of debris, such as fiberglass, but the mechanism by which it occurs and how it influences the pressure drop are not understood. Previous NRC acceptance of pertinent Nuclear Energy Institute (NEI) guidance now appears premature in light of confirmatory research that has revealed much larger influence of the bed structure (e.g. up to a factor of about 100 on pressure drop for the same mix of fibers and particles) than had previously been thought to be possible. Research in this area should be continued and expanded as needed in order to reduce the very large uncertainties surrounding these effects and to determine if a predictive capability is feasible.

Other important phenomena, such as chemical and downstream (of the screen) effects are now being investigated by RES. These are essentially exploratory studies that have uncovered some significant effects, but have yet to reveal their scope and magnitude. Predictive capability remains to be demonstrated. The NRC needs to evaluate the results of these studies and determine how much it can rely on the nuclear industry to develop reliable predictive tools and how much independent predictive capability it requires. Development of a predictive capability may require investment of substantial resources and time.

TRACE Computer Code Development

Several years ago, the NRC recognized that it could not sustain the continued maintenance of several thermal-hydraulic codes for each general type of nuclear power plant. It elected to consolidate its existing codes for the confirmatory analysis of licensee submittals on design basis thermal-hydraulic issues into a single code now called TRACE. The consolidation is now largely completed. The TRACE computer code is viewed by the NRC research staff as "as good as anything else that is out there." The long-term validation and improvement phase of code development is at hand. Current research is devoted to improving features of the TRACE code, making it easier to use and validating it against available data. Some of the data already exist and other data are being generated. In addition, the integration of the TRACE code, coupled with the CONTAIN code to model containment response and the PARKS code for neutronic analyses into the regulatory processes of the agency has begun.

The TRACE code is reputed to now be able to serve as the "workhorse" thermal-hydraulic analysis code for the agency. In the course of its work to consolidate thermal-hydraulics codes into TRACE, the research staff has found many ways to improve the code. Such improvements should be done. Now, however, it is far more important that the integration of TRACE into the regulatory process be completed in an expeditious manner. The research staff working on the development needs to have input from users of the code on needed features and capability of the code. Inevitably, the introduction of a new computational tool will slow and detract the regulatory process for some transient period. There is no way to counter this difficulty associated with the introduction of a new computer code. It must be endured and the sooner this is done, the sooner the challenges associated with the use of a new code in the regulatory process can be overcome. Once TRACE is integrated into the regulatory process, the developers will receive valuable advice on how their efforts to improve the code should be directed to enhance the regulatory process.

Highest priority should be given to the integration of TRACE code into the regulatory process. As this integration progresses, the research staff can continue its efforts to improve and further develop TRACE on a "time available" basis. The ACRS is concerned that efforts to improve TRACE lack prioritization and defensible organization. Placing the TRACE code in the hands of users will also identify a host of needed improvements. Prioritization of technical improvements might be aided substantially by commissioning a detailed peer review of TRACE. To do this, the staff will have to have available code documentation of outstanding scope and quality. Such high quality code documentation will also be needed if the code is to become part of the regulatory process. Code documentation, then, is a task that ought to take precedence in the thermal hydraulic research effort.

Experimental Studies of Thermal-Hydraulic Phenomena

Thermal-hydraulic phenomena involving the flow of two-phase mixtures of steam and water are very complicated especially those involving blowdown from high pressure systems. Thermal-hydraulic phenomena that arise in advanced light water reactor designs that emphasize passive response to accidents are driven by subtle forces that require sophisticated understanding to ensure plant safety. As a consequence, NRC has long felt that it cannot rely solely on computer code projections of thermal-hydraulic phenomena to ensure adequate protection of the public health and safety. Experimental confirmation is also required. As the computer models used to analyze thermal-hydraulic phenomena have become more sophisticated, the experiments needed to validate model predictions have become progressively more integral in nature. Experimental facilities have become larger and more complex. RES has an interest in maintaining these facilities for use in addressing future as well as current regulatory issues. Maintenance of large, complex experimental facilities has become a significant expense in this research area. The major experimental facilities used by NRC in the U.S. are the APEX and PUMA facilities as well as RBHT facility at Penn State University. Abroad, NRC is conducting tests at the PKL facility, the SETH tests and tests at the ROSA facility. Additional experimental needs may arise in connection with the design certification of the ESBWR.

APEX is a medium-size, scaled, integral test facility that proved useful for the certification of the AP600 and AP1000 reactor designs. It has been modified to provide data crucial to the analysis of thermal shock to reactor vessels. It is proposed now that the APEX facility be used for confirmatory analyses for AP1000 and for some "thermal hydraulic integral experiments." These proposed applications would benefit from review to assess their focus and applicability.

PUMA is a medium size, scaled facility especially suited for evaluating passive emergency core cooling systems. It is being modified to be applicable to testing the emergency core cooling systems for the ESBWR.

The RBHT test program has been under way for a number of years with the purpose of improving core reflood models that are a key part of evaluating the adequacy of pressurized water reactor emergency core cooling systems. The reflood models may become critical if applications are submitted for large power uprates in PWRs. The proposed research program at the RBHT facility needs evaluation to see if the quality, scope and detail of the data are properly matched to the proposed uses of these data.

NRC has wisely not sought to duplicate large test facilities available overseas. Use of these facilities is possible through international programs. The SETH program was useful for resolving Generic Safety Issue (GSI) 185 and assessing the emergency heat removal systems in the ESBWR. Future work under this program at the ROSA and the PKL facilities in support of the TRACE code needs to be more clearly focused.

It is essential for NRC to maintain an ability to assess thermal-hydraulic phenomena that occur both in existing reactors and in future reactors. It is evident that the development of computer codes to predict thermal hydraulic phenomena and the experimental validation of these predictions will grow more burdensome with time. Major development efforts can be anticipated if very innovative designs using coolants other than water are brought forward for certification. It is not likely that the nuclear institutions of any one country will be able to develop adequate codes and conduct sufficient validation of these codes alone. International cooperative development of codes and conduct of experiments appear essential as NRC research moves beyond TRACE with its current capabilities and especially if analyses are needed for coolants other than water. NRC already takes substantial advantage of international experimental capabilities. Extending this international flavor in thermal-hydraulics research to include the development of computer codes will contribute to current ideas of multi-national design approval process. It may slow code development. It also may ensure that sufficient resources for code development are available so that it is feasible to meet the more exacting standards that are likely to be demanded in the future.

Table 12. Thermal-Hydraulics Research Activities

Job Code	Title	Comment
N6106	*Confirmatory Head Loss Testing*	Experiments to measure head loss across sump pump strainers in PWRs.
Y6871	*PWR Sump Screen Penetration and Throttle Valve Testing*	Experiments to determine the type and quantity of debris that can pass through typical PWR sump screens.
N6100	*Head Loss Testing*	Assess the susceptibility of recirculation screens to debris blockage during design basis accidents.
Y6999	*Integrated Chemical Effects Tests*	Five tests to determine representative chemical and material environments in PWRs that can contribute to sump blockage.
N6121	*GSI-191 Chemical Effects Simulations*	Experiments to determine chemical effects that can contribute to sump screen blockage.
N6198	*Transportability of Coatings*	Parametric study to ascertain if coatings can be transported to sumps under accident conditions.
N6083	*BWR ECCS Suction Concerns*	Technical assessment of Generic Issue 193 "BWR Suction Concerns."
Y6769	*PUMA Test Facility*	Facility for the conduct of thermal hydraulics tests. This facility can produce data for natural circulation systems for use in ESBWR design certification.
Y6852	*PWR Thermal-Hydraulics Integral Experiments*	Tests at the APEX facility at Oregon State University.
N6042	*OECD/ROSA Program*	International collaborative tests of reactor accident thermal hydraulic phenomena.
Y6945	*Rod Bundle Heat Transfer Test Program - Phase 3*	Experiments at Penn State University in support of TRACE code analyses of small and large break loss of coolant accidents. To date, there is little evidence that data from this facility can be of value for TRACE code development. Further work in this facility should be scrutinized carefully to assure that it meets agency needs.

Table 12. Thermal-Hydraulics Research Activities (Continued)

Job Code	Title	Comment
Y6589	***Thermal-Hydraulic Research***	Perform analytical and small-scale experimental work in support of the TRACE code. Neutronic work in this program in nearly complete. Long-range thermal hydraulic work needs to be shown necessary for agency needs.
N6043	***Thermal-Hydraulic Sub-channel International Standard***	Analysis for international standard problem for a BWR subchannel benchmark.
Y6571	***SETH Program - Test Facilities***	Thermal-hydraulics tests in two international efforts: PKL on boron dilution and PANDA in support of ESBWR certification.
Y6974	***OECD-PKL Program and Test Facility***	International collaborative research on boron dilution accidents including mid-loop operation.
N6213	***TRACE Verification and Validation***	Verification and validation of the TRACE thermal-hydraulics analysis code. This work is viewed as vital to the verification and validation of TRACE.
Y6673	***TRAC-M Development and Assessment - Small LOCA Processes (In the past, the TRACE code was called TRAC-M)***	Simulate separate effects tests with the TRACE code and show acceptable agreement with predecessor codes. Good progress has been made in this important work.
Y6666	***Advanced Numerical Methods in TRAC-M (In the past, the TRACE code was called TRAC-M)***	Advanced numerical methods for the TRACE code. This work is not essential for the current range of efforts to make TRACE useful to the agency.
N6147	***TRACE Development and Assessment Against Specified Tests***	Use TRACE code to evaluate level swell tests done at several facilities. This is a small part of the TRACE validation and verification effort.
N6201	***Gravity Reflood and SBLOCA TRACE Assessment***	Use the TRACE code to assess PUMA facility tests. This work necessary to lend credibility to TRACE for ESBWR analysis.

Table 12. Thermal-Hydraulics Research Activities
(Continued)

Job Code	Title	Comment
Y6525	***TRAC-M Code Maintenance (In the past, the TRACE code was called TRAC-M)***	Maintenance of the TRACE code. This is an essential activity.
N6040	***Data Acquisition***	Recover old input decks for the TRAC-PWR model.
N6072	***Implementation of ACR-700 (Misleading title, Project deals with PUMA input deck)***	This work is no longer necessary.
Y6198	***Continuation of Support for System Code Analysis***	Support for the SCDAP/RELAP5 computer code and the analysis of steam generator tube rupture accidents.
Y6392	***Maintenance, Application, Assessment and Development of NRC Computer Codes***	Consolidation of RELAP5 capabilities into TRACE. This work appears to overlap most of the TRACE development tasks. Incorporation of RELAP capabilities into TRACE has proven difficult because of code philosophy differences.
Y6667	***SNAP Implementation***	Graphical user interface for TRACE and other NRC computer codes. This work is important because of poor direct input methods inherited in TRACE from the underlying TRAC models.
Y6662	***AP1000 Confirmatory Thermalhydraulics Analysis***	Confirmatory thermal hydraulic analyses of a wide range of design basis accidents hypothesized to occur in AP1000. This work is complete.
Y6526	***Administer CAMP Meeting***	Meeting of users of NRC thermal-hydraulics codes. This program will assist in the international acceptance of TRACE.
N6030	***Flow-induced Vibrations and Effects on BWR components***	Analysis of component vibration that can lead to fatigue failure in BWRs.

15 REFERENCES

1. U.S. Nuclear Regulatory Commission, "Review and Evaluation of the Nuclear regulatory Commission Safety Research Program," Advisory Committee on Reactor Safeguards (ACRS), NUREG-1635, Vol. 6, March 2004.

2. Code of Federal Regulations, Title 10, Part 50 (10 CFR 50), "Domestic Licensing of Production and Utilization Facilities," U.S. Government Printing Office, Washington D.C., 2005.

3. 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," U.S. Government Printing Office, Washington D.C., 2005.

4. 10 CFR 50.48, "Fire Protection," U.S. Government Printing Office, Washington D.C., 2005.

5. 10 CFR 50.54, "Conditions of Licenses," U.S. Government Printing Office, Washington D.C., 2005.

6. 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," U.S. Government Printing Office, Washington D.C., 2005.

7. Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements," U.S. Government Printing Office, Washington D.C., 2005.

8. Appendix H to 10 CFR Part 50, "Reactor Vessel Material Surveillance Program Requirements," U.S. Government Printing Office, Washington D.C., 2005.

9. 10 CFR Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants," U.S. Government Printing Office, Washington D.C., 2005.

10. U.S. Nuclear Regulatory Commission, Supplement 4 to Generic Letter (GL) 88-20, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10 CFR 50.54(f)," June 28, 1991.

11. U.S. Nuclear Regulatory Commission, "Procedure and Submittals Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," NUREG-1407, June 1991.

12. U.S. Nuclear Regulatory Commission, "Staff Requirements Memorandum (SRM), SECY-4-0111, Recommended Staff Actions Regarding Agency Guidance in the Areas of Safety Conscious Work Environment and Safety Culture," August 30, 2004.

13. Pouchet, A., "The European Benchmark Exercise on Human reliability Analysis, "Proceedings of American nuclear society International Topical Meeting on probability, reliability, and safety Assessment (PSA '89), pp. 103-110, Pittsburgh, PA, April 2-7, 1989.

14. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Rev. 2, May 1988.

15. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," February 2004.
16. 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.
17. Generic Safety Issue 191, "Assessment of Debris Accumulation on PWR Sump Performance."
18. Generic Safety Issue 185, "Control of Recriticality Following Small-break Locas in PWR."

March 17, 2006

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: RESOLUTION OF GENERIC SAFETY ISSUE 188, "STEAM GENERATOR TUBE LEAKS OR RUPTURES CONCURRENT WITH CONTAINMENT BYPASS FROM MAIN STEAMLINE OR FEEDWATER LINE BREACHES"

During the 530th meeting of the Advisory Committee on Reactor Safeguards, March 9-11, 2006, the Committee considered the technical basis for resolving Generic Safety Issue 188, "Steam Generator Tube Leaks or Ruptures Concurrent with Containment Bypass from Main Steam or Feedwater Line Breaches." The Committee agrees with the staff's resolution of this generic safety issue.

<u>References</u>:
Memorandum dated February 15, 2006, from Mark A. Cunningham, Director, Division of Engineering Technology, to John T. Larkins, Executive Director, ACRS, Subject: Closeout of Generic Safety Issue 188, "Steam Generator Tube Leaks or Ruptures Concurrent with Containment Bypass from Main Steam Line or Feedwater Line Breaches."

NUREG-XXXX, "Resolution of Generic Safety Issue 188: Steam Generator Tube Leaks or Ruptures Concurrent with Containment Bypass from Main Steam Line or Feedwater Line Breaches," U.S. Nuclear Regulatory Commission, Washington, DC.

cc: A. Vietti-Cook, SECY
W. Dean, OEDO
B. Sosa, OEDO
M. Cunningham, RES
J. Uhle, RES
A. Lee, RES
T. Mintz, RES
R. Emrit, RES
R. Assa, RES
K. Karwoski, NRR
A. Hiser, NRR
T. Meek, NRR

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

March 23, 2006

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 BROWNS FERRY NUCLEAR PLANT UNITS 1, 2, AND 3

Dear Chairman Diaz:

During the 530th meeting of the Advisory Committee on Reactor Safeguards, March 9-11, 2006, we completed our review of the license renewal application (LRA) for the Browns Ferry Nuclear Plant (BFN) Units 1, 2, and 3 and the associated final Safety Evaluation Report (SER) 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 SER with Open Items on October 5, 2005. We issued an interim letter on the safety aspects of this application on October 19, 2005. During our 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. This report fulfills the requirements of 10 CFR 54.25 that the ACRS review and report on all license renewal applications.

CONCLUSIONS AND RECOMMENDATIONS

1. With the inclusion of the conditions in Recommendations 2 and 3, the application for license renewal for BFN Units 1, 2, and 3 should be approved.

2. The drywell refueling seals should be included within the scope of license renewal and be subjected to periodic inspections. Alternatively, as proposed by the staff, the drywell shells should be subjected to periodic volumetric inspections to detect external corrosion.

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

DISCUSSION

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

The BFN site is located in Limestone County, Alabama on the north shore of the Wheeler Reservoir. All three BFN units are General Electric boiling water reactors (BWR 4) with Mark 1 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 and TVA plans to restart it in May 2007. The approximate duration of power operation of the three units is 10 years for Unit 1, 23 years for Unit 2, and 18 years for Unit 3. As part of an extensive restart program for Unit 1, components that have been in "layup" for the past 20 years will be either replaced or requalified. Layup is intended to provide a controlled environment to limit corrosion of plant components.

BFN Unit 1 is currently not identical to Units 2 and 3. TVA has committed to implement all of the physical and programmatic improvements to Unit 1 that have been made to Units 2 and 3. By the time of restart, the Unit 1 licensing basis will be identical to that of the other two units. The three units will have nearly identical components, materials, environments, operating procedures, and technical specifications. The Corrective Action Program applies to all three units, so that any condition identified in one unit will be reviewed for generic implications to the other units. The applicant states that, because all three units contain the same materials and have experienced the same conditions, the aging mechanisms during the layup and recovery periods are similar among the three units. Since the aging effects of the Unit 1 shutdown are similar to those experienced in Units 2 and 3, the applicant has used operating experience from the restart of Units 2 and 3 in the recovery of Unit 1. Based on these considerations, TVA has submitted a common license renewal application for all three units.

In part because it is not clear to what extent the layup experience of Units 2 and 3 parallels the experience of Unit 1, in our interim report we questioned the extent of applicability of Units 2 and 3 operating experience to the unique operating history of Unit 1. The SER states that a 1987 NRC inspection report identified several instances of deficient layup conditions during the early phase of the extended outage. This raises the possibility of potential latent effects that could result in accelerated aging once the plant restarts and operates at power. The applicant acknowledges this concern by stating on page B-4 of the LRA that "During the performance of the Aging Management Review activities, there was recognition that the operating experience of 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 response to this concern, TVA added the Unit 1 Periodic Inspection Program to those aging management programs described in the LRA. Although this inspection program has not been fully defined, significant attributes of this program have been provided to the staff and are discussed in the final SER. This program requires periodic inspections of those components in layup that will not be replaced before restart. The scope of this program covers carbon steel, low-alloy steel, and stainless steel pipes and fittings from 25 plant systems. Samples are grouped by common material types and environments.

The applicant has agreed to use an inspection sampling size that would reflect a 95/95 confidence level that unacceptable degradation can be detected. Inspections will be performed at susceptible locations and in areas where degradation is not expected. Baseline inspections will be performed before restart. Additional inspections will be performed after Unit 1 is restarted and again within the first ten years of the period of extended operation. The inspection frequency will depend on the results of each inspection. The acceptance criteria are that the pipe wall remains above the minimum acceptable thickness until the next inspection and no unacceptable weld cracks exist. We concur with the staff's conclusion that this program will adequately manage the aging effects for which it is credited.

In the original BFN LRA, the applicant requested renewed licenses at EPU conditions for all three units. In a letter dated January 7, 2005, TVA requested that the EPU and the LRA be separated. Even though the staff reviewed the LRA based on current licensed power levels for each unit, the final SER has several references to EPU conditions. The steam dryers are included in the scope of license renewal, but their aging management review will be performed as part of the safety evaluation of the EPU application. The time-limited aging analyses (TLAAs) associated with neutron embrittlement, reactor vessel fatigue, radiation degradation of drywell expansion gap foam, and stress relaxation of the core plate hold-down bolts were performed assuming EPU conditions.

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

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

The staff also performed inspections and an audit of AMPs and aging management reviews (AMRs). A recent inspection found that the applicant had made significant progress in developing the AMP implementation packages but identified errors in them. The applicant initiated a Problem Evaluation Report to identify the causes of the errors and determine corrective actions to prevent recurrence. Inspections performed before BFN enters the period of extended operation should verify that implemented corrective actions have been effective.

The audit of the AMPs and AMRs is documented in a report by the Brookhaven National Laboratory. The audit examined 28 AMPs and the associated AMRs and verified that the AMPs are consistent with the GALL Report or concluded that they would adequately manage aging during the period of extended operation. Several of the existing AMPs will be enhanced to include Unit 1 prior to the period of extended operation. Appendix F of the LRA describes TVA's plan to resolve the differences between the licensing bases of Unit 1 and Units 2 and 3 before Unit 1 restart. The staff's review of Appendix F did not identify any omissions or discrepancies.

The staff concluded that the scoping and screening processes implemented by the applicant have successfully identified SSCs within the scope of license renewal and subject to an AMR. With the inclusion in the scope of license renewal of those Unit 1 systems and components that were in layup and have not been replaced, we agree with this conclusion.

Open Item 2.4-3 in the SER concerns aging management of drywell shell corrosion. The staff was concerned that leakage through refueling seals at the top of the drywell could lead to corrosion of the drywell shell in a location that cannot be inspected. This aging effect has been observed in several Mark I containments and is the subject of Generic Letter 87-05 and Information Notice 86-99 on the potential for corrosion of BWR Mark I steel drywells in the sandpocket region. The staff has concluded that the refueling seals should be within the scope of license renewal because they are nonsafety-related components whose failure can affect the integrity of the safety-related containment steel liner. We concur with this conclusion.

The applicant acknowledges that water was observed below the refueling seals at BFN Unit 3 during the 1998 refueling outage, but maintains that the refueling seals should not be within the scope of license renewal. As an alternative to the inclusion of the seals, the staff proposed that TVA periodically perform ultrasonic testing of the drywell shells as part of the containment inservice inspection program. Such an approach has been used by previous license renewal applicants, and we agree that it is an acceptable alternative. As an alternative to the staff's proposal, the applicant committed to perform a one-time confirmatory inspection of the Unit 1 drywell shell prior to restart and of the Units 2 and 3 shells prior to entering the period of extended operation. Based on this commitment, the staff closed out this open item. We do not agree with this resolution. One-time inspections are intended to confirm that an unexpected aging effect is not occurring or is occurring at such a slow rate that no further inspections are required. This aging effect has been observed in several Mark I containments, and we are aware of at least one instance of through-wall corrosion. One-time inspection of the shell does not provide assurance that leakage of the refueling seals after the one-time inspection is performed will not create an environment that could result in future drywell degradation. Unless the applicant can demonstrate that the resulting corrosion rate would not be sufficient to degrade the pressure retaining function during the period of extended operation, the refueling seals should be within the scope of license renewal and subject to periodic inspections, or the drywell shells should be subjected to periodic volumetric inspections.

During our March 9, 2006 meeting, we were told that the staff has reopened this item based on discussions with the applicant regarding drywell inspection results. Ultrasonic inspections performed in 1999, 2002, and 2004 identified a small inclusion in the drywell liner of Unit 1. The applicant will submit this information to the staff in writing. The staff plans to document its evaluation of this information in a supplemental SER. Based on our discussions with the applicant and staff, the resolution of this issue does not affect our recommendations regarding this LRA.

In our interim letter we noted that in the draft SER some restart inspections were referred to as "one-time" inspections. We suggested that, to avoid confusion, the term "one-time" inspection should be used only for license-renewal-related inspections. For clarification purposes, the final SER now provides definitions of one-time inspections, restart inspections, and Unit 1 periodic inspections. Section 3.7 of the final SER still refers to some restart inspections as one-time inspections. The final SER should be revised to be consistent with these definitions.

The applicant has identified systems and components requiring a TLAA and reevaluated them for 20 more years of operation. The SER concludes that the TLAAs are valid for the period of extended operation, the TLAAs are projected to the end of the period of extended operation, or that aging effects will be adequately managed for the period of extended operation. We concur with this assessment.

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 LRA and the associated SER reflect operating experience only at the current power level. If the EPU is implemented before the period of extended operation, the staff should require that TVA evaluate the operating experience of Units 1, 2, and 3 at the uprated power level and then incorporate lessons learned into their aging management programs prior to entering the period of extended operation. The EDO response to our interim letter stated that the staff's SER for the EPU would include a commitment to perform such an evaluation.

With the inclusion of commitments to perform periodic inspections of BFN Units 1, 2, and 3 drywell refueling seals or drywell shells and to perform an evaluation of operating experience at the EPU level and incorporate lessons learned into their aging management programs prior to entering the period of extended operation, the application for license renewal of Browns Ferry Units 1, 2, and 3 should be approved.

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References:
1. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the License Renewal of the Browns Ferry Nuclear Plant, Units 1, 2, and 3," January 2006.
2. 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.
3. Tennessee Valley Authority, "Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 - Application for Renewed Operating Licenses," December 31, 2003.
4. Tennessee Valley Authority, "Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 - January 28, 2004 Meeting Follow-Up - Additional Information," February 19, 2004.
5. 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, Docket Nos.: 05000259, 05000260, 05000296," April 26, 2005.
6. U.S. Nuclear Regulatory Commission, "Browns Ferry Nuclear Plant - Inspection Report 05000259/2004012, 05000260/2004012, and 05000296/2004012," January 27, 2005.
7. U.S. Nuclear Regulatory Commission, "Browns Ferry Nuclear Plant - Inspection Report 05000259/2005013, 05000260/2005013, and 05000296/2005013," November 7, 2005.
8. Tennessee Valley Authority, "Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application (LRA) - Annual Update (TAC Nos. MC1704, MC1705, and MC1706)," January 31, 2006.
9. Letter from William J. Shack, Acting Chairman, ACRS, to Luis A. Reyes, Executive Director for Operations, NRC, "Interim Report on the Safety Aspects of the License Renewal Application for the Browns Ferry Nuclear Plant, Units 1, 2, and 3," October 19, 2005.
10. Letter from Luis A. Reyes, Executive Director for Operations, NRC, to William J. Shack, Acting Chairman, ACRS, "Response to Advisory Committee on Reactor Safeguards - Interim Report on the Safety Aspects of the License Renewal Application for Browns Ferry Nuclear Plant, Units 1, 2, and 3," November 28, 2005.
11. Tennessee Valley Authority, "Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 - Summary of NRC Site Visit and Meeting Regarding Extended Power Uprate (EPU) and License Renewal Application (LRA)," January 7, 2005.
12. 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.
13. 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.

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

REVISED

April 10, 2006

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

SUBJECT: GENERIC SAFETY ISSUE 191 - ASSESSMENT OF DEBRIS ACCUMULATION ON PWR SUMP PERFORMANCE

Dear Chairman Diaz:

During the 530th meeting of the Advisory Committee on Reactor Safeguards, March 9-11, 2006, we considered several reports by the NRC staff regarding their efforts to resolve Generic Safety Issue 191(GSI-191), "Assessment of Debris Accumulation on PWR Sump Performance." The staff discussed licensee responses to Generic Letter 2004-02 (GL 2004-02), "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors," and presented the results of efforts by the Office of Nuclear Regulatory Research (RES) to understand several phenomenological issues that have arisen as part of the GSI-191 effort, including chemical effects, downstream effects, and head loss correlations through debris beds. The results were presented to our Thermal-Hydraulics Phenomena Subcommittee on February 14-16, 2006. 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. In response to GL 2004-02, many licensees plan to increase the size of their sump screens as quickly as feasible. Based on the current state of knowledge, we concur with this intent. However, it is not evident that this measure will be sufficient to resolve all long-term core cooling issues.

2. Results of prototypical experiments planned by industry to validate screen effectiveness will be difficult to extrapolate to plant conditions. Further work is required to provide the technical basis by which the staff can assess the adequacy of the planned modifications to the plants. Guidance should be developed to support the staff's review.

3. Recent research has revealed significant influences of particle/fiber mixtures and chemical reaction products on screen pressure drop for which improved predictive methods and guidance should be developed.

4. Increasing screen size to reduce the pressure drop may increase the amount of fine debris and chemical products that passes through the screen. Methods for predicting the quantity and properties of this bypassed debris should be developed. Potential adverse effects on downstream components, including pumps, valves, the core entrance regions, and the core itself, should be evaluated.

5. There has been some success at using adjustable parameters in an equilibrium chemistry model to match the chemical species that form in sumps. The methods should be validated further and guidance should be developed for their use.

6. The results of tests of coating debris formation and transport should be included in the assessment of core coolability as they become available. Future work should include the development of adequate predictive capability for the effects of coating debris on screen pressure drop and bypass.

OVERVIEW

At our meeting with the Commission on December 8, 2005, several Commissioners expressed the view that the sump screen issue should receive high priority. This was formally stated in the Commission's staff requirements memorandum of December 20, 2005: "... The ACRS shall make among its highest priorities its role in the resolution of GSI-191. ..." At the Commission meeting we indicated that we were waiting to hear status reports from the staff. We have now received several reports, some of them preliminary, and this has enabled us to form an opinion on progress towards resolving GSI-191.

We have written previous letters on the sump screen issue. In particular we raised the matter of chemical effects and questioned some aspects of the NEI guidance which the staff had endorsed.

The staff issued GL 2004-02 on September 13, 2004, and has received responses from all licensees. Though all licensees responded to the generic letter, the staff has concluded that none of the responses was complete. Gaps were evident in all important areas, particularly chemical and downstream effects. The staff has issued requests for additional information (RAIs) relating to several significant effects. Many licensees are finalizing plans to replace the screens before these RAIs are resolved.

While progress has been made in all areas of research, much remains to be done. These programs have produced significant results and are making important contributions to understanding the issues related to PWR sump performance. Many relevant physical and chemical phenomena are being explored. Assessments of other important effects may need to be added to the program.

This research has yet to lead to an ability to develop and validate predictive methods. Much of the work is exploratory in nature, in response to indications that existing analytical capabilities were incomplete and inadequate. The results from some programs are not yet available or are awaiting staff review.

The GL 2004-02 responses and recent research have raised new questions. Present plans by licensees to make hardware changes in their plants are driven by the need to reduce the potential for excessive head loss across sump screens during recirculation. Increasing the screen size will reduce this head loss, but the staff's ability to assess the adequacy of the reduction may be limited by uncertainties in the available knowledge base. In addition, downstream effects may be exacerbated by some screen designs and configurations. The staff needs effective means to evaluate these downstream effects and their influence on core coolability.

DISCUSSION

Industry Response to Generic Letter 2004-02

In general, licensees intend to address the sump screen issue by making a significant increase in the flow areas of the screens. Some designs may also have smaller openings and/or active debris removal mechanisms. Physical changes have already been made in some plants. Modifications to almost all plants are planned to be completed by the end of calendar year 2007. Some licensees have requested extensions until the spring outage of 2008. Each of the five vendors of the new sump screens plans to undertake integrated-effect "proof tests" with screens or segments of screens to demonstrate the ability of the screens to accommodate the anticipated loading of debris with an acceptable pressure drop.

The prediction of debris formation, transport, and impact on core coolability is a very complex technical problem. A number of phenomenological issues must be addressed, either by the development of a predictive capability or by the implementation of engineering solutions that circumvent the more difficult issues. The industry is focusing on engineering approaches that maximize screen area to the extent practical, control of materials that affect the quantity and character of debris generation, and the control of sump chemistry to minimize chemical effects.

Regulatory Approach

The staff intends to undertake eight to ten audits of plant modifications. The scope of the audits will be expanded if the staff encounters problems with the technical adequacy of the planned resolutions.

Because of the "proof test" nature of the planned industrial testing program, it is essential that the staff have a level of understanding and a modeling capability for the underlying phenomena adequate to support their technical review of the licensee results. It is doubtful that the current understanding of these phenomena will be adequate to support such a review. The results of recent research have served to call into question some previous guidelines and assumptions without replacing them with validated, improved methods.

Research Efforts

Research is being performed to address the following phenomena:

- Chemical effects – experiments (Los Alamos National Laboratory (LANL) and Argonne National Laboratory (ANL)) and model development for speciation (Center for Nuclear Waste Research Activities (CNWRA))

- Head loss from debris buildup on screens – experiments (Pacific Northwest National Laboratory (PNNL)) and model development (RES)

- Downstream effects – experiments (LANL)

- Coating debris formation and transport – experiments (Electric Power Research Institute (EPRI), Naval Surface Warfare Center (NSWC))

We have seen only the preliminary results from some of these research efforts. It is premature for us to perform a comprehensive evaluation until all the work is complete. However, several research projects have developed important new quantitative information which reveals the significance of certain phenomena. Understanding of those phenomena has not yet been established to the point where validated predictive tools are available. RES has set a target of the spring of 2006 to bring these activities to a conclusion. This schedule is unrealistic in view of the many unresolved issues.

Chemical effects

Exploratory integrated chemical effects tests (ICET) revealed that some species, particularly aluminum oxyhydroxide and calcium phosphate, can be produced under certain conditions. It was concluded that plant-specific evaluations would be required.

ANL is investigating the interaction between calcium silicate insulation (CalSil) and trisodiumphosphate (TSP), which forms calcium phosphate. A qualitative understanding of the chemical processes has been achieved. Studies of head loss on screens using debris quantities that duplicated earlier LANL tests with no chemical additives showed some variability. When calcium phosphate was produced by adding TSP to CalSil, or calcium chloride to TSP, the pressure drop increased substantially. For example, in one test (ICET3-9) the pressure drop through a fiberglass bed was 0.14 psi at a flow velocity of 0.1 ft/s. When calcium chloride was added in stages to the solution of TSP, the pressure drop eventually rose to 5.2 psi at a flow velocity below 0.02 ft/s. Since the flow regime was probably laminar, for which pressure loss is proportional to flow velocity, this corresponds to an increase in bed resistance by a factor of about 200, amounting essentially to blockage of the screen. Similar results were obtained in Tests 1 and 2.

The results of chemical speciation prediction by codes using chemical equilibrium models and measured corrosion rates are encouraging over the range of species that have been studied. CNWRA found that some ICET results could be matched by adjusting the speciation parameters.

Head Loss Tests

PNNL has been conducting head loss tests with mixtures of fiberglass and CalSil in amounts corresponding to those used in earlier LANL tests. The results in some cases differ significantly from the results obtained by LANL. No distinct pattern is evident though some trends might be inferred. In an extreme case, when the constituents were introduced in a particular way, the head loss was roughly 100 times more than the head loss with a well-mixed debris bed of the same overall composition. These results indicate that the structure of the debris bed and the way in which it is formed can have a huge influence on the head loss. Unless the assumption of a homogeneous bed can be justified, it will be necessary to develop an adequate model for these effects (for plants that intend to retain CalSil) or to find a way to scale them in the proof tests now planned by industry. The alternative of developing theoretical models for the way in which the bed builds up in different parts of the screen over time during a variety of accidents is probably unrealistic and may be beyond the capabilities of present state-of-the-art.

RES has begun development of a theoretical model to predict the head loss in a nonhomogeneous debris bed. Substantiation and validation of such a model would be a major undertaking.

Downstream Effects

Tests conducted by LANL revealed that fine debris, of a size characteristic of the debris expected during energetic loss-of-coolant accidents (LOCAs), would pass through a typical sump screen under some conditions. Unless a debris bed has been established, most particles of CalSil and fine fiberglass pass through the screen. Significant quantities of reflective metallic insulation were observed to pass through under some conditions. In the absence of a detailed model for the history of debris bed development on a screen and the arrival of various constituents as functions of location and time, there are considerable uncertainties about how to apply such results to an actual plant. An order of magnitude calculation, with 5000 ft^3 of debris produced, indicates that about 6% of the debris would fill the typical lower plenum of a reactor vessel, if it settled there and was not transported to the core or filtered by debris catchers below the fuel. The larger the screen, the more open area there is likely to be through which fine debris can pass. Chemical reaction products are also likely to pass through open areas of the screen.

In reply to our subcommittee's questions about the effects of such debris on core coolability, the staff and representatives of the Westinghouse Owners Group (WOG) stated that they thought the core would be adequately cooled in a number of scenarios. However, they presented no physical models or analytical predictions to show a validated, quantitative basis for such conclusions.

Tests by LANL of debris transported to throttle valves have revealed a significant effect on pressure drop. Adequate predictive methods are therefore needed for the amount of this debris which actually reaches these valves, and for the resulting consequences.

Coatings

EPRI is conducting experiments on the formation of debris from qualified and unqualified coatings. The results were not presented at our meetings.

NSWC is conducting some basic tests of terminal velocity and transport of paint chips of various shapes, sizes, and composition. Guidance for use of these data remains to be developed.

What Is Missing

We are not aware of research efforts in several important areas.

The most significant omission appears to be an adequate understanding of the effects of the various debris species which enter the reactor vessel and reach the core. These effects are likely to depend on the LOCA scenario, particularly the location and size of the break, and on the screen design. Although guidance developed by the WOG describes several of the phenomena to be modeled to represent these effects, the WOG apparently leaves the evaluation to engineering judgment and ad hoc model development. Unless these effects can somehow be avoided, there is a need for a comprehensive set of validated tools for representing them. Developing the tools would involve significant experimental and model development efforts.

The proof tests being developed by industry to evaluate new screen designs involve the phenomena described earlier in this letter, as well as others. Synthesizing these evaluations into a defensible method for scaling test results to the actual LOCA scenario is no trivial matter. We have yet to see scaling laws, methods of extrapolation, or theoretical representations (e.g. computational codes) which can make a convincing case that the test results can be applied to the actual plant. For example, one issue is how to use tests on a single module to predict the performance of an array of modules. The Office of Nuclear Reactor Regulation (NRR) may need to draw on further research results in order to evaluate submissions based on these proof tests.

Formation and transport of coating debris are being studied. We have not seen results of work on the effects of this debris on screen head loss. In view of the difficulty of predicting head loss with the existing mix of ingredients, and the surprises that have been encountered, it is necessary to establish a knowledge base for the effects of coatings on head loss by means of an adequate set of experiments and predictive methods.

Research has already revealed that the structure of a debris bed influences head loss and the bypass of fine material. As screens become larger and perhaps have more complex geometry, the variability of bed structure over the surface of the screen is likely to increase. Some areas, such as the base of vertical screens or the outer layers of multiple screens, may be covered by a pile of coarse debris, other areas may support "thin beds" that are blocked by chemical products or fine debris, while some areas may be clear of debris, providing paths through which fine material can pass. There is a need to reduce uncertainty in predicting the performance of these screens under a wide variety of scenarios. Since modeling everything theoretically is impractical, the emphasis should be placed on designing for predictability, supported by data.

THE PATH FORWARD

In response to GL 2004-02, licensees have undertaken the task of showing that they satisfy the requirements of recirculation core cooling. In most cases, the response has been to plan the replacement of sump screens by those with significantly larger area. The hole size and other characteristics of these screens may also be changed.

These changes are in the right direction to alleviate the potential for excessive head loss. However, in view of uncertainties introduced by new research results, the incomplete response by industry to the generic letter, the difficulties of validating the "proof tests" planned by industrial consortia, and downstream effects, NRR will need to develop assurance that it has the capability to evaluate the effects of these changes. The staff anticipates that, if sufficient uncertainty is encountered, supplemental actions may be required. These may include the following measures:

- Removal from containment of constituents that are known to cause problems with head loss and lack of predictability.

- Development of screen designs that are insensitive to the plethora of uncertainties associated with many existing designs. These designs may include active screens or similar devices that can handle many forms of debris without the need for knowing the details of the debris characteristics.

- Design of screens for minimum bypass of fine debris. Emphasis is currently being placed on reducing head loss, but downstream effects should also be considered.

- Identification of other solutions to core cooling that get around the manifold uncertainties associated with the present range of screen designs and can more confidently demonstrate success in meeting specifications.

- Use of probabilistic analysis to show that the most undesirable debris bed configurations are highly unlikely. Evaluation would be based on realistic analysis rather than on a conservative approach.

We endorse the immediate plans to increase the size of sump screens because this will alleviate the potential for excessive head loss. This action by itself may not be sufficient to resolve all long-term core cooling issues.

We anticipate working further with the staff on these important matters.

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

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References:

1. U.S. Nuclear Regulatory Commission Generic Letter 2004-02: "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors", September 13, 2004.
2. U.S. Nuclear Regulatory Commission Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors", June 9, 2003.
3. Letter from Mario V. Bonaca, Advisory Committee on Reactor Safeguards, "Proposed Draft Final Generic Letter on Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at PWRs," July 19, 2004.
4. "Draft NRC Staff Review Guidance for Evaluation of Downstream Effects of Debris Ingress into the PWR RCS On Long Term Core Cooling Following a LOCA", undated.
5. "Evaluation of Downstream Sump Debris Effects in Support of GSI-191," WCAP-16406-P, Westinghouse Owners Group, June 2005.
6. Information Notice 2005-26: "Results of Chemical Effects Head Loss Tests in a Simulated PWR Sump Pool Environment," September 16, 2005.
7. NRC Information Notice 2005-26, Supplement 1: "Additional Results of Chemical Effects Tests in a Simulated PWR Sump Pool Environment," January 20, 2006.
8. "Integrated Chemical Effects Test Project: Test #1 Data Report," LA-UR-05-124, June 2005.
9. "Integrated Chemical Effects Test Project: Test #2 Data Report," LA-UR-05-6146, September 2005.
10. "Integrated Chemical Effects Test Project: Test #3 Data Report," LA-UR-05-6996, October 2005.
11. "Integrated Chemical Effects Test Project: Test #4 Data Report," LA-UR-05-8735, November 2005.
12. "Integrated Chemical Effects Test Project: Test #5 Data Report," LA-UR-05-9177, January 2006.
13. Memorandum from Michele G. Evans to John N. Hannon, "Final Transmittal of Information Summarizing Integrated Chemical Effects Results and Implications", October 25, 2005.
14. "Corrosion Rate Measurements and Chemical Speciation of Corrosion Products Using Thermodynamic Modeling of Debris Components to Support GSI-191," NUREG/CR-6873, April 2005.
15. "Screen Penetration Test Report," NUREG/CR-6885, LA-UR-04-5416, October 2005.
16. Memorandum from Ralph Architzel to James Lyons, "Report on Results of Staff Pilot Plant Audit–Crystal River Analyses Required for the Response to Generic Letter 2004-02 and GSI-191 Resolution," June 29, 2005.
17. Memorandum from Ralph Architzel to Thomas Martin, "Report on Results of Staff Pilot Plant Audit– Fort Calhoun Station Analyses Required for the Response To Generic Letter 2004-02 and GSI-191 Resolution," January 26, 2006.

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

March 24, 2006

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

SUBJECT: FINAL REVIEW OF THE EXELON GENERATION COMPANY, LLC, APPLICATION FOR EARLY SITE PERMIT AND THE ASSOCIATED NRC STAFF'S FINAL SAFETY EVALUATION REPORT

Dear Chairman Diaz:

During the 530th meeting of the Advisory Committee on Reactor Safeguards, March 9-11, 2006, we completed our review of the early site permit application for the Clinton site and the associated final Safety Evaluation Report (SER) prepared by the NRC staff. We reviewed the application and the final SER 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. We issued an interim letter on this application and the associated draft SER on September 22, 2005. This matter was also discussed during our Subcommittee meeting on March 8, 2006. During these reviews, we had the benefit of discussions with representatives of the NRC staff and Exelon Generation Company, LLC (Exelon). We also had the benefit of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

- The early site permit application and the staff's final SER show that the proposed nuclear power plant site adjacent to the existing Clinton Nuclear Power Station is an acceptable site for nuclear power plants that meet the plant parameter envelope proposed by the applicant.

- The staff has thoroughly reviewed a performance-based method proposed by the applicant for determining the safe shutdown earthquake (SSE) ground motion. This method is an attractive alternative to methods endorsed in current regulatory guides.

- The staff should consider development of a regulatory guide dealing with the alternative, performance-based, method for assessing the seismic hazard of a site.

DISCUSSION

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 MWt on a site adjacent to the currently operating Clinton plant, which is a BWR 6 within a Mark III containment. 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 certified designs such as the AP1000 and Advanced Boiling

Water Reactor (ABWR) as well as other designs such as the International Reactor Innovative and Secure (IRIS), Economic Simplified Boiling Water Reactor (ESBWR), Gas-Turbine Modular Helium Reactor (GT-MHR), and Pebble Bed Modular Reactor (PBMR).

The staff's review of this application included a detailed review of the alternative, performance-based method proposed by the applicant for determining the SSE ground motion spectrum. The staff identified six permit conditions for the proposed site. The staff has used technically sound, objective criteria for identifying these permit conditions. The staff and the applicant have agreed to 32 combined license (COL) action items. The action items for the proposed Clinton site can be compared to 30 action items for the North Anna early site permit and 26 action items for the Grand Gulf early site permit.

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 populations centers with populations in excess of 25,000 include Springfield (74 km), Peoria (75 km), Champaign (49 km), Urbana (66 km), Decatur (36 km), and Bloomington (36 km). Near the site (<16 km) are the small towns Clinton (population 7,000), as well as DeWitt, Weldon, and Wapella each with a population of less than 1,000.

Population trends in the larger cities near the site have been estimated based on census data. Modest growth in population is anticipated in these cities over the next 60 years. 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.

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 has considered the potential for cycles in weather that may complicate the prediction of future weather extremes based on historical records. Nevertheless, we believe that the applicant has adequately characterized the site weather for the purposes of an early site permit.

Seismicity

The proposed site is affected by the New Madrid seismic zone and the Wabash Valley seismic zone. Since the nuclear power plant at the Clinton site was licensed, the estimated frequency of major earthquakes at the New Madrid seismic zone has been increased. The estimate of the maximum potential magnitude of earthquakes at the Wabash Valley seismic zone has also been increased. There is a background seismicity of the site represented by the Springfield earthquake estimated to have occurred at a location about 70 km from the site, approximately 6,000 years ago and to have had a magnitude of 6.2 to 6.8 on the Richter scale.

In other applications for early site permits, the applicants have adopted the methods recommended in Regulatory Guide 1.165 to estimate the SSE ground motion spectrum. Exelon has adopted an alternative method. This alternative is based on an industry standard (ASCE 43-05) that itself is based on work done by the Department of Energy for assessing the seismic safety of its nuclear facilities. The alternative is considered "performance based" because it uses a target probability for the maximum acceptable facility damage from an earthquake.

Exelon has selected the frequency of 10^{-5}/yr for the onset of significant inelastic deformation of systems, structures, and components. This target provides a rather substantial margin to core damage and containment failure.

The staff has reviewed thoroughly the proposed alternative method for estimating the seismic hazard at the proposed site. The staff's review included examination of the credibility of parametric quantities in the models and an independent assessment of the analysis results by direct integration of the seismic risk equation. Also, the staff has reviewed carefully the applicant's assessment of the local seismic hazard. We concur with the staff that the alternative approach adopted by Exelon for this application provides a high level of safety. The seismic core damage frequency that can be inferred from the proposed ground motion spectrum (~2×10^{-6}/yr) is significantly less than the median found in seismic probabilistic risk assessments for 29 existing nuclear power plants. The performance-based alternative method yields results that are in concert with the Commission's expectation that advanced reactors will provide enhanced margins of safety and/or utilize simplified, inherent, passive, or other innovative means to accomplish their safety functions.

The alternative, performance-based, method uses a target frequency that does not change with time as new information on the seismicity of power plant sites changes. In this sense, the alternative method provides some additional regulatory stability. For this reason, if no other, we expect that the alternative method will be attractive to licensees and applicants for a variety of purposes. The staff may want to consider developing a regulatory guide on the use of the alternative methodology. Certainly, the detailed review of the method conducted by the staff for this early site permit would provide a substantial technical basis for the development of such a regulatory guide.

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References:

1. Exelon Generation Company, LLC, Early Site Permit Application, September 23, 2003.
2. ACRS Interim Letter, Exelon Generation Company, LLC, Application for Early Site Permit and the Associated NRC Staff's Draft Safety Evaluation Report, dated September 22, 2005.
3. EDO response to ACRS Interim Letter, "Interim Letter: Exelon Generation Company, LLC, Application for Early Site Permit and the Associated NRC Staff's Draft Safety Evaluation Report on the Clinton Early Site Permit Site," dated October 26, 2005.
4. Final Safety Evaluation Report for Exelon Early Site Permit Application, dated February 17, 2006.
5. Exelon Generation Company, LLC, letter to the U.S. Nuclear Regulatory Commission Subject: "Seismic Risk (Performance Goal) Based Approach Primer Revision," dated January 14, 2005.
6. NRC Regulatory Guide 1.165, "Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion," March 1997.
7. American Society of Civil Engineers, Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities, ASCE/SEI 43-05 (ASCE Standard 43-05), 2005.

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

March 28, 2006

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

SUBJECT: DRAFT FINAL REVISION 4 TO REGULATORY GUIDE 1.97, "CRITERIA FOR ACCIDENT MONITORING INSTRUMENTATION FOR NUCLEAR POWER PLANTS"

Dear Mr. Reyes:

During the 530th meeting of the Advisory Committee on Reactor Safeguards, March 9-11, 2006, we reviewed draft final Revision 4 to Regulatory Guide 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants." During our review, we had the benefit of discussions with representatives of the NRC staff and the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

1. Revision 4 to Regulatory Guide 1.97 should not be issued in its present form.

2. The staff should revise Regulatory Position 1 to allow licensees to adopt the IEEE 497-2002 Standard to modify individual accident monitoring instruments without a complete analysis of all accident monitoring instrumentation.

3. We agree that licensees should not be allowed to use the IEEE 497-2002 Standard to eliminate or reclassify accident monitoring instrumentation required by previous editions of this Standard unless Revision 4 to Regulatory Guide 1.97 is adopted in its entirety.

DISCUSSION

Draft final Revision 4 to Regulatory Guide 1.97 endorses, with certain exceptions, IEEE 497-2002, "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations." IEEE Standard 497-2002 supersedes IEEE 497-1981 and IEEE 497-1983, both of which are now inactive standards. This Standard provides a consolidated source of post-accident monitoring requirements and the associated bases for a new generation of advanced nuclear plant designs. This Standard also contains appropriate guidance and a flexible basis for making changes to such systems in operating plants. In addition to incorporating requirements from previous editions of this Standard, Revision 4 to Regulatory Guide 1.97 is designed to consider the current state-of-the-art digital design technology for accident monitoring displays, and incorporates user experience and feedback. This Standard addresses some important aspects of the design, installation, and qualification of digital technology for accident monitoring instrumentation.

The staff has reviewed this Standard and, after consideration of public comments, endorsed it, subject to eight regulatory positions. The staff's positions are technically sound. However, the staff has adopted a position that could frustrate the application of this Standard to modifying and upgrading portions of the accident monitoring instrumentation in existing plants.

Regulatory Position (1) states: "If a current operating reactor licensee voluntarily converts to the criteria in Revision 4 of this guide, the licensee should perform the conversion on the plant's entire accident monitoring program to ensure a complete analysis."

In this position, the staff sets forth its intentions with regard to the applicability of IEEE Standard 497-2002 to current operating reactors. Clause 1.1 of IEEE Standard 497-2002 states that the Standard is intended for new plants, although current plants may find its guidance useful in performing design-basis evaluations or implementing design modifications.

In Revision 4 to Regulatory Guide 1.97, the staff states that conversion means adapting the plant's entire accident monitoring program from the current licensing basis (Revision 2 or 3 of Regulatory Guide 1.97), to the guidance in Revision 4. This adaptation could include physical changes (e.g., replacing an instrument), licensing changes (e.g., technical specification changes), or both for each variable. The staff also recognizes that Revisions 3 and 4 of this Regulatory Guide differ in several ways, including variable type definitions and associated criteria, removal of design and qualification categories, removal of prescriptive tables of monitored variables, analysis required to produce the necessary design-basis documentation, and related changes in licensing basis and/or commitments. These differences could involve modifications to existing instrumentation and could impose unnecessary regulatory burden on current operating reactor licensees, inhibiting the adoption of the IEEE 497-2002 Standard.

Regulatory Position 1 is too restrictive. In the case where a licensee desires to upgrade a portion of its accident monitoring instrumentation, the licensee should be allowed to apply the IEEE 497-2002 Standard to perform such upgrades without being required to perform a complete analysis of the entire set of accident monitoring instruments at the plant.

We agree that in some cases where a licensee may want to eliminate or reclassify an instrument (variable) from its list of accident monitoring variables, the licensee should then be required to adopt the IEEE 497-2002 Standard in its entirety. This will ensure that operators have the necessary information to mitigate any accident, consistent with the Emergency Operating Procedures, Abnormal Operating Procedures, and Emergency Response Guidelines.

We look forward to reviewing the staff's resolution of this matter.

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References:

1. Memorandum from J. Wiggins, RES, to J. Larkins, ACRS, Subject: Request for ACRS Review of Regulatory Guide 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," Revision 4, January 30,2006 (ADAMS Accession No. ML053640127).

2. Regulatory Guide 1.97 (Draft was issued as DG-1128, dated June 2005), "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," Revision 4, December 2005 (ADAMS Accession No. ML053640151).
3. Staff Responses to Public Comments on DG-1128, January 31, 2006 (ADAMS Accession No. ML053640161).
4. IEEE Standard 497-2002, "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Generating Stations," September 2002.
5. Memorandum from J. Larkins, ACRS, to L. Reyes, EDO, Subject: Proposed Regulatory Guide (DG) -1128, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants (Revision 4 Regulatory Guide 1.97)," July 8, 2005 (ADAMS Accession No. ML051950526).

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

April 14, 2006

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

SUBJECT: REVIEW OF THE 1994 ADDENDA TO THE ASME CODE FOR CLASS 1, 2, AND 3 PIPING SYSTEMS AND THE RESOLUTION OF THE DIFFERENCES BETWEEN THE NRC STAFF AND ASME

Dear Mr. Reyes:

During the 531st meeting of the Advisory Committee on Reactor Safeguards, April 5-7, 2006, we reviewed the resolution of the differences between the NRC staff and the American Society of Mechanical Engineers (ASME) regarding the 1994 Addenda to Section III of the ASME Boiler and Pressure Vessel Code for Class 1, 2, and 3 piping systems. During our reviews, we had the benefit of discussions with representatives of the NRC staff and ASME. We also had the benefit of the documents referenced.

RECOMMENDATION

Most of the differences between the staff and ASME are resolved. The staff proposes to address the one remaining issue related to dynamic strain aging of certain carbon steels at temperatures greater than 300 °F by placing a restriction on the endorsement of the ASME Code in 10 CFR 50.55a. This approach is practical; however, we encourage the staff to work with ASME to resolve the one remaining issue.

DISCUSSION

The NRC staff initially did not endorse the revised seismic design criteria in the 1994 Addenda to the ASME Code because of concerns with the technical basis used to establish these criteria. Since that time, the ASME has initiated changes to the Code to address the staff's concerns. These changes include eliminating the application of the seismic rules to flow-transient loads, eliminating the NB-3200 strain criteria, modifying the Class 2 and 3 Level B limits to be consistent with the Level D limits, eliminating changes specifying the methods to generate seismic loads in the evaluation of reversing dynamic loads, and adding provisions to address potential strain concentrations. The staff agrees with these changes.

The remaining unresolved issue between ASME and the staff relates to the effects of dynamic strain aging on the ultimate tensile capacity of certain carbon steels at temperatures greater than 300 °F. The staff proposes to address this issue by placing a restriction in the 10 CFR 50.55a endorsement of the ASME Code. This approach is practical; however, we encourage the staff to work with ASME to resolve the one remaining issue.

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References:

1. U.S. Nuclear Regulatory Commission, "Seismic Analysis of Piping," NUREG/CR-5361, June 1998.
2. Letter to G.M. Eisenberg, Director, Nuclear Codes and Standards, ASME, from Brian W. Sheron, NRR, "ASME Code Revisions to the Design Rules for Piping Systems," May 24, 1995.
3. Presentation by John R. Fair, NRR, to the ACRS Subcommittee on Materials and Metallurgy, "Piping Seismic Design Criteria," March 25, 1999.
4. Presentation by John R. Fair, NRR, to William J. Shack, ACRS, "Status of ASME Code Piping Seismic Design Criteria," October 3, 2003.

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

April 14, 2006

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

SUBJECT: GRAND GULF EARLY SITE PERMIT APPLICATION: EVALUATION OF TRANSPORTATION ACCIDENTS ON THE MISSISSIPPI RIVER

Dear Mr. Reyes:

During the 531st meeting of the Advisory Committee on Reactor Safeguards, April 5-7, 2006, 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 evaluations performed by the applicant and the NRC staff of the hazards posed to the proposed site by transportation accidents on the Mississippi River as well as the proposed changes to the NRC staff's final Safety Evaluation Report. We provided an interim letter on the Grand Gulf ESP application and the draft Safety Evaluation Report on June 14, 2005, and a final letter on December 23, 2005. The Committee reviewed this application to fulfill the requirement of 10 CFR 52.23 that the ACRS report on those portions of an ESP application that concern safety. We also had the benefit of the documents referenced.

In our December 23, 2005, letter concerning the staff's final Safety Evaluation Report for the Grand Gulf early site permit application, we asked for clarification on risks associated with transportation accidents and possible explosions on the Mississippi River, which is approximately 1.8 kilometers from the proposed site. We asked particularly for a more complete explanation of the attenuation of shock waves that was attributed to the location and elevation of the site relative to the river. The staff asked the applicant to provide this clarification.

In response, the applicant adopted an alternative approach to the analysis of accidental explosions during transportation accidents on the river. This approach is centered on the low probability of an explosion that could produce a pressure pulse that exceeded about 7 kPa at the proposed site. To do this, the applicant examined three types of explosions that might occur should there be an accident involving barge traffic on the river:

- explosions contained within a barge
- explosions near a barge that had spilled volatile, combustible cargo so that a vapor cloud developed
- explosions of vapor clouds that drifted toward the proposed site

The staff independently evaluated the probabilities of these three classes of explosions. The staff was careful to use shipment frequencies, accident frequencies, spill frequencies, and the like that could be justified based on data applicable to barge traffic on the Mississippi River. The staff adopted conservative probabilities in those instances where sufficient data were not available to justify lower probabilities used in some cases by the applicant. Nevertheless, the staff concluded that the probability of an explosion producing a pressure pulse in excess of 7 kPa at the proposed power plant site was on the order of 10^{-6}/yr. The staff concluded that explosions of such low probability posed negligible risk to power plant facilities that might be located on the proposed site.

We found the staff's analyses of river transportation accidents acceptable and support the staff's proposed changes to the Safety Evaluation Report to describe these analyses.

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References:
1. Memorandum dated March 27, 2006, from David A. Matthews, NRR/ADRA/DNRL to John T. Larkins, Executive Director, Advisory Committee on Reactor Safeguards, Subject: ACRS Review of the Grand Gulf Early Site Permit Application - Final Safety Evaluation Report Changed Pages.
2. U.S. Nuclear Regulatory Commission, Final 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," October 21, 2005.
3. Letter dated June 14, 2005, from G.B. Wallis, Chairman, ACRS, to L.A. Reyes, Executive Director for Operations, NRC, Subject: Interim Letter: Draft Safety Evaluation Report on Grand Gulf Early Site Permit Application.
4. Letter dated December 23, 2005, from Graham B. Wallis, ACRS, to L.A. Reyes, Executive Director for Operations, NRC, Subject: Early Site Permit Application for the Grand Gulf Site and the Associated Final Safety Evaluation Report.
5. Letter dated February 1, 2006, from L.A. Reyes, Executive Director for Operations, NRC, to Graham B. Wallis, ACRS, Subject: Early Site Permit Application for the Grand Gulf Site and the Associated Final Safety Evaluation Report.
6. System Energy Resources, Inc. (SERI), letter dated February 22, 2006, from George A. Zinke, SERI, to NRC Document Control Desk, Subject: Response to Request for Additional Information Regarding the Grand Gulf Early Site Permit Final Safety Evaluation Report.
7. System Energy Resources, Inc. (SERI), letter dated March 7, 2006, from George A. Zinke, SERI, to NRC Document Control Desk, Subject: Supplemental Information, Response to Request for Additional Information Regarding the Grand Gulf Early Site Permit Final Safety Evaluation Report.
8. Regulatory Guide 1.91, Revision 1, "Evaluations of Explosions Postulated To Occur on Transportation Routes Near Nuclear Power Plants," dated February 1978.

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

April 19, 2006

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

SUBJECT: RESPONSE TO YOUR MARCH 29, 2006 LETTER REGARDING STANDARD REVIEW PLAN, SECTION 14.2.1, "GENERIC GUIDELINES FOR EXTENDED POWER UPRATE TESTING PROGRAMS"

Dear Mr. Reyes:

In our letter dated February 22, 2006, we provided the following recommendation on Standard Review Plan (SRP) Section 14.2.1, "Generic Guidelines for Extended Power Uprate Testing Programs:"

> Paragraph III.C of SRP Section 14.2.1 should be rewritten to provide more structured and explicit guidance defining those conditions under which large transient tests would be exempted or required.

In your March 29, 2006 response, you stated that plant-specific issues can influence a decision for large transient testing. As a result, the staff concluded that it is not practical or even feasible, to improve the SRP decision logic.

Large transient tests have special objectives. They test not only the performance of individual components and structures but also the integrated response of the system, including control functions. Because large transient tests impose substantial hydrodynamic and thermal loads on the plant, they have impacts on the plant risks. Although these risk impacts are not substantial, it is appropriate to exempt the licensee from performing the tests if they provide little benefit. Conversely, transient tests can identify the unexpected. It would be preferred to uncover issues within the context and precautions of a controlled test, rather than during an unplanned transient.

Section 14.2.1 of the SRP identifies the following seven factors to consider in determining whether a licensee should be exempted from performing a test:

- Power uprate operating experience
- Introduction of new thermal-hydraulic phenomena or identified system interactions
- Facility conformance to limitations associated with computer modeling and analytical methods
- Plant operator familiarization with facility operation and trial use of operating and emergency operating procedures
- Minimal reductions in the margin of safety
- Guidance contained in vendor topical reports
- Risk implications

Although it is appropriate to consider these factors, there is little guidance provided to the reviewer as to standards of acceptance.

We understand that plant-specific considerations could impact the decision process. However, a structured decision process does not have to be rigid. The process does not make the decision; it is an aid to the decision. It is practical and feasible to develop such a logical structure without constraining the ability of the staff to include plant specific considerations. An example of such a structure follows:

- Identify each large transient test and associated objectives from the initial startup program.
- Determine which systems, operations, system interactions, and procedures are changed by the uprate.
- Assess whether the plant modifications or changes affect the conclusions of the initial start-up tests. If not, these tests would not have to be performed.
- Identify any new tests that would be required to verify the proper operation of any modified or new equipment.
- Determine whether other tests will be performed that will ensure that each modified component will perform as intended. If not, a transient test would be expected.
- Assess whether there are multiple modified components, such that the system is effectively new. If so, a transient test would be expected.
- Assess whether analytic modeling capability encompasses the changed range of parameters. If not, a transient test would be expected.
- Assess whether physical phenomena or system interactions could be substantially affected by the change (e.g., potential lifting of relief valves or water level rising to steamline). If so, a transient test would be expected.
- Determine whether the range of system conditions falls within the history of previous power uprates. If not, a transient test would be expected.

We would appreciate the opportunity to meet with the staff to discuss approaches to improving SRP Section 14.2.1.

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References:

1. Letter from L. Reyes, EDO, to G. Wallis, ACRS, Subject: Standard Review Plan, Section 14.2.1, "Generic Guidelines for Extended Power Uprate Testing Programs," dated March 29, 2006 (ADAMS Accession No. ML060680235).

2. Letter from G. Wallis, ACRS, to L. Reyes, EDO, Subject: Standard Review Plan, Section 14.2.1, "Generic Guidelines for Extended Power Uprate Testing Programs," dated February 22, 2006 (ADAMS Accession No. ML060530320).

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

April 20, 2006

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

SUBJECT: DRAFT FINAL REGULATORY GUIDE 1.205, "RISK-INFORMED, PERFORMANCE-BASED FIRE PROTECTION FOR EXISTING LIGHT-WATER NUCLEAR POWER PLANTS"

Dear Mr. Reyes:

During the 531st meeting of the Advisory Committee on Reactor Safeguards, April 5 - 7, 2006, we reviewed draft final Regulatory Guide (RG) 1.205, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants." We issued a letter on a previous version of this Regulatory Guide on June 14, 2005, and discussed the staff's proposed response to this letter during the 526th meeting on October 6-8, 2005. 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. RG 1.205 should be issued after the peer-review guidance is clarified.

2. RG 1.205 should be revised to make clear that in cases where licensees elect to rely on information contained in an internal-event Probabilistic Safety Assessment (PSA)[1] or other analyses such as Individual Plant Examinations of External Events (IPEEE) to quantify risk associated with fires, these analyses should be peer reviewed.

3. The staff should develop models for human performance that focus on the probability distribution of the time to complete a recovery action under specified conditions.

BACKGROUND AND DISCUSSION

The National Fire Protection Association (NFPA) issued a performance-based standard for fire protection for light-water reactors in 2001 (NFPA 805). 10 CFR 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). 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. By memorandum dated May 3, 2005, the staff sent to us the draft final Regulatory Guide for our review.

In our June 14, 2005 letter, we recommended that the draft final Regulatory Guide not be issued. The main reason for this recommendation was that the proposed methods in NEI 04-02, Revision 0 for risk-informed decisionmaking were not based on a fire PSA. In a letter dated August 2, 2005, the staff agreed with the principal argument of our letter and stated that it would work with NEI to ensure that the parts of NEI 04-02, Revision 0 that the staff endorses use correct methodology and language.

[1]The terms "Probabilistic Safety Assessment" and "Probabilistic Risk Assessment" (PRA) are treated as synonymous in the regulatory guide.

NEI issued Revision 1 to NEI 04-02, in September 2005. The March 2006 version of the draft final RG 1.205 endorses the revised NEI report with the exception of Section 6. These documents have satisfactorily addressed the principal concerns that we expressed in our June 14, 2005 letter.

Plant-specific fire PSAs have shown that fires can be among the major contributors to risk. We believe that any changes to the fire protection program that claim to be risk informed should be based on a rigorous peer-reviewed, plant-specific fire PSA.

In the Background Section of RG 1.205, the staff states that it anticipates that licensees will develop a fire PSA and that, without it, licensees "will not realize the full safety and cost benefits of transitioning to NFPA." In Section 3.2.3, the staff states that, "for PSA-based methodologies," license amendment requests should include an explanation of why the fire PSA is considered technically adequate, as well as a description of the associated peer review. However, 10 CFR 50.48 (c) permits license amendment requests that are not based on a fire PSA. Such requests will have to be based on information in an internal-event PSA or an IPEEE to quantify risk associated with fires. RG 1.205 now appears to indicate that the staff would accept such alternative analyses without a peer review. The staff has agreed to clarify the RG to make clear that a peer review should be conducted for these alternative analyses. After clarifying the guidance for peer review, RG 1.205 should be issued.

RG 1.205 also addresses operator manual actions. If such actions are credited in lieu of Appendix R requirements and have not been approved by the NRC, then they must be treated as plant changes. Section B.2.2.4 of NEI 04-02, Revision 1 states: "The reliability of the recovery action should be commensurate with its risk-significance." The NEI document specifies that, in evaluating this reliability, "the amount of time available to the licensee to complete the recovery action versus the time to actually complete the action should be considered and evaluated." The evaluation should also consider the uncertainties associated with "(i) human performance, (ii) the difference between field verification conditions and actual environmental and fire conditions, and (iii) design basis (e.g., thermal hydraulic analysis) versus actual time constraints."

We agree with these statements. However, we note that their implementation would be facilitated by human reliability models that focus on the probability distribution of the time required to complete a certain action under specified conditions. Neither of the NRC models for human performance (ATHEANA and SPAR-H) focuses on this distribution. They instead treat the available time as just one of many performance shaping factors. The staff should work with the human reliability analysis experts in the Office of Nuclear Regulatory Research to develop appropriate models for evaluating the reliability of operator recovery actions.

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References:
1. Regulatory Guide 1.205, "Risk-informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," March 2006 (ADAMS Accession No. ML060600183).
2. NEI 04-02, Revision 1, "Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program Under 10 CFR 50.48 (c)," September 2005 (ADAMS Accession No. ML052590476).
3. Letter from the EDO to Dr. Wallis, dated August 2, 2005, Subject: Draft Final Regulatory Guide, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants" (ADAMS Accession No. ML051940255).
4. Letter from Dr. Wallis to the EDO, dated June 14, 2005, Subject: Draft Final Regulatory Guide, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants" (ADAMS Accession No. ML051650432).
5. Memo from M. Salley, RES, to S. Weerakkody, NRR, "Transmittal of Fire Risk Analysis Review Guidance in Support of NFPA 805 Based Changes to the Fire Protection Program" dated January 12, 2006 (ADAMS Accession No. ML060120449).

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

April 21, 2006

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

SUBJECT: NRC STAFF'S PROPOSED APPROACH TO ENHANCE THE REACTOR OVERSIGHT PROCESS TO ADDRESS SAFETY CULTURE ISSUES

Dear Chairman Diaz:

During the 531st meeting of the Advisory Committee on Reactor Safeguards, April 5-7, 2006, we met with representatives of the NRC staff to review the staff's proposed approach to enhance the Reactor Oversight Process (ROP) to more explicitly address safety culture issues. Our Subcommittees on Human Factors and Reliability & Probabilistic Risk Assessment discussed the proposed approach during a joint meeting on January 25, 2006. We also had the benefit of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

1. The staff's proposed approach enhances significantly the ability of the Agency to identify and address safety culture issues.

2. After gaining experience with the enhanced process, the staff should reassess the adequacy of the guideline that specifies about 30-minutes daily for resident inspectors to review entries to the Corrective Action Program.

3. The revision to Inspection Procedure 95003, "Supplemental Inspection for Repetitive Degraded Cornerstones, Multiple Degraded Cornerstones, Multiple Yellow Inputs, or One Red Input," should include criteria for safety culture assessments, clear thresholds for evaluating crosscutting aspects of findings, and clear expectations for the resolution of the staff's concerns with the licensee's safety culture.

DISCUSSION

In a Staff Requirements Memorandum (SRM) dated August 30, 2004, the Commission directed the staff to "enhance ROP treatment of crosscutting issues to more fully address safety culture." In the SRM, the Commission directed the staff not to use surveys of licensee personnel, but to rely on inspector observations and other indicators already available to the NRC. The staff was asked to consider enhanced problem identification and resolution initiatives as part of this effort. In addition, as part of their enhanced inspection activities for plants in the Degraded Cornerstone column of the ROP Action Matrix, the staff was also directed to include a determination of the need for a specific evaluation of the licensee's safety culture and a process for making the determination and conducting the evaluation.

The staff has nearly completed its revision to applicable Inspection Manual Chapters and associated inspection procedures that provide the tools for treating safety culture in the ROP as directed by the Commission. However, we have not reviewed the revision to Inspection Procedure 95003, which will govern how an independent assessment of safety culture is to be performed, because it was not available when we met with the staff.

The staff's approach preserves the three existing ROP crosscutting areas (Problem Identification and Resolution, Human Performance, and Safety Conscious Work Environment). To help inspectors identify causal factors related to safety culture, the staff modified the existing inspection framework to include expanded definitions and descriptions of components of safety culture that align with each crosscutting area. The staff proposes a graded approach for regulatory intervention as a licensee's performance moves from left to right in the Regulatory Response Columns of the ROP Action Matrix. We view this as a prudent evolutionary approach that enhances the existing ROP's ability to identify and address safety culture issues.

In developing this approach, the staff has interacted with internal and external stakeholders. There is general stakeholder agreement with the approach proposed by the staff. During a public meeting on December 8, 2005, the NRC and the industry representatives agreed that the NRC would use the definition of safety culture developed by the International Atomic Energy Agency (Safety Series No. 75-INSAG-4, "Safety Culture," Vienna, 1991):

"That assembly of characteristics and attitudes in organizations and individuals which establishes that, as an overriding priority, nuclear plant safety issues receive the attention warranted by their significance."

Participants in that meeting also agreed on two sets of attributes, which they called "components," that appropriately characterize safety culture. The components in the first set align with at least one crosscutting area of the ROP:

Corrective Action Program
Self- and independent assessment
Operating experience
Decisionmaking
Resources
Work control
Work practices
Environment for raising nuclear safety concerns
Preventing, detecting, and mitigating perceptions of retaliation

The second set of components do not align directly with the ROP crosscutting areas:

Accountability
Continuous learning environment
Organizational change management
Safety policies

Information on the first set of components is readily accessible through baseline inspections and therefore can be gathered as part of the inspection procedures that support the ROP. Information on the second set is typically not available through baseline inspection procedures, and evaluations of such components would be part of the supplemental inspection procedures for plants in the Multiple/Repetitive Degraded Cornerstone column. We agree that the above components are appropriate.

The staff has assigned components from the first set to each crosscutting issue. Although some components could be aligned with more than one crosscutting issue, the staff assigned each component to only one crosscutting issue to avoid entering a performance deficiency into multiple locations. Following this approach, the staff identified the Corrective Action Program, self- and independent assessment, and operating experience as components of problem identification and resolution. Decisionmaking, resources, work control, and work practices are identified as components of human performance. Environment for raising safety concerns and preventing, detecting, and mitigating perception of retaliation are identified as components of safety conscious work environment.

We generally agree with this approach. However, a component may be relevant to more than one crosscutting issue. For example, resources and decisionmaking are also important attributes of problem identification and resolution. Among the key performance indicators of problem identification and resolution are backlog, time to correct identified conditions, and the threshold for entering conditions into the Corrective Action Program. For these indicators, performance depends significantly on resources and conservative decisionmaking. The revision to Inspection Procedure 71152, "Identification and Resolution of Problems," should expand on this issue to improve inspectors' ability to recognize the possible impact of decisionmaking and resources on problem identification and resolution.

Draft revision to Inspection Procedure 71152 gives instructions for the resident inspector's daily review of each item entered into the Corrective Action Program. This procedure is important because it focuses on early detection of safety culture problems. The staff has made the procedure more effective by including the crosscutting issue component descriptions and associated resident inspector training. However, in spite of the importance of this activity, the procedure states that the inspection time should be generally less than 30 minutes per day. After gaining experience with safety culture enhancements to the ROP, the staff should revisit the less-than-30-minutes-a-day guideline to make sure this is enough time.

As mentioned above, the revised procedures reflect a graded regulatory response as a licensee's performance moves from left to right across the ROP Action Matrix. Once a plant enters into the Multiple Repetitive Degraded Cornerstone column, the NRC expects that an independent assessment of safety culture will be performed. Under certain circumstances, this assessment is also required for a plant with a single degraded cornerstone or a substantive crosscutting issue. When an independent assessment is required, revised Inspection Procedure 95003 will guide the assessment. When the safety culture of the licensee is to be independently evaluated, all components will be tested, irrespective of where the findings were identified. Although we have not seen a draft revision of Inspection Procedure 95003, it should include criteria on assessing performance in each component, so that different organizations performing the assessment will produce consistent results. The procedure should include clear thresholds for crosscutting aspects of findings and clear expectations for the resolution of the staff's concerns with the licensee's safety culture.

The staff has developed a performance-based structured approach, to identify safety culture issues. With the inclusion of the criteria discussed above in the revised Inspection

Procedure 95003, the proposed changes to the ROP are appropriate and will enhance the agency's ability to address safety culture issues. We look forward to additional discussions with the staff on the revised Inspection Procedure 95003 and its application within the ROP.

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References:
1. Memorandum dated July 1, 2004, from Luis A. Reyes, Executive Director for Operations, NRC, for the Commissioners, SECY-04-0111, Subject: Recommended Staff Actions Regarding Agency Guidance in the Area of Safety Conscious Work Environment and Safety Culture.
2. Memorandum dated August 30, 2004 from Annette Vietti-Cook, Secretary of NRC, to Luis A. Reyes, Executive Director for Operations, NRC, Subject: Staff Requirements - SECY-04-0111 - Recommended Staff Actions Regarding Agency Guidance in the Area of Safety Conscious Work Environment and Safety Culture.
3. Memorandum dated October 19, 2005, from Luis A. Reyes, Executive Director for Operations, NRC, for the Commissioners, SECY-05-0187, Subject: Status of Safety Culture Initiatives and Schedule for Near-Term Deliverables.
4. Memorandum dated December 21, 2005, from Annette Vietti-Cook, Secretary of NRC to Luis A. Reyes, Executive Director for Operations, NRC, Subject: Staff Requirements - SECY-05-0187 - Status of Safety Culture Initiatives and Schedule for Near-Term Deliverables.
5. U.S. Nuclear Regulatory Commission Revised Inspection Procedure 93800, "Augmented Inspection Team," (03/22/06).
6. U.S. Nuclear Regulatory Commission Revised Inspection Procedure 71152, "Identification and Resolution of Problems," (03/22/06).
7. U.S. Nuclear Regulatory Commission Revised Inspection Procedure 71153, "Event Followup," (03/22/06).
8. U.S. Nuclear Regulatory Commission Revised Inspection Procedure 93812, "Special Inspection," (03/22/06).
9. U.S. Nuclear Regulatory Commission Revised Inspection Procedure 95001, "Inspection for One or Two White Inputs in a Strategic Performance Area," (03/22/06).
10. U.S. Nuclear Regulatory Commission Revised Inspection Procedure 95002, "Inspection for One Degraded Cornerstone or Any Three White Inputs in Strategic Performance Area," (03/22/06).
11. U.S. Nuclear Regulatory Commission Inspection Procedure 95003, "Supplemental Inspection for Repetitive Degraded Cornerstones, Multiple Degraded Cornerstones, Multiple Yellow Inputs, or One Red Input," Issue Date: 01/17/02.
12. U.S. Nuclear Regulatory Commission Revised Manual Chapter 0305, "Operating Reactor Assessment Program," (03/22/06).
13. U.S. Nuclear Regulatory Commission Revised Manual Chapter 0612, "Power Reactor Inspection Reports," Issue Date: 09/30/06.
14. U.S. Nuclear Regulatory Commission Revised Manual Chapter 0612, Appendix D, "Guidance for Documenting Inspection Procedure 71152, Identification and Resolution of Problems," (03/22/06).
15. Safety Culture Initiative Narrative, Revision 1, February 9, 2006.

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

April 21, 2006

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

SUBJECT: APPLICATION OF THE TRACG COMPUTER CODE TO EVALUATE THE STABILITY OF THE ECONOMIC SIMPLIFIED BOILING WATER REACTOR

Dear Mr. Reyes:

During the 531st meeting of the Advisory Committee on Reactor Safeguards, April 5-7, 2006, we reviewed the staff's draft Safety Evaluation Report related to the use of the TRACG computer code to evaluate the stability of the Economic Simplified Boiling Water Reactor (ESBWR). This issue was reviewed by our Thermal-Hydraulic Phenomena Subcommittee on January 19 and March 14, 2006. During our reviews, we had the benefit of presentations by and discussions with representatives of the NRC staff and General Electric (GE). We also had the benefit of the documents referenced.

RECOMMENDATIONS

The staff should approve the use of TRACG to analyze the stability of the ESBWR during normal operation, anticipated operational occurrences, and the low-power phase of reactor startup.

DISCUSSION

TRACG has been validated for the analysis of anticipated operation occurrences in boiling water reactors (BWRs) and for loss-of-coolant accident analyses of the ESBWR. It has also been used as a basic computational tool for predicting the performance of BWRs in commercial service. It is currently under review for use in addressing stability-related issues for operating BWRs.

The question we addressed was whether TRACG can adequately model those ESBWR features that affect stability. The main difference between the ESBWR and current operating BWRs is the use of natural circulation, rather than forced circulation, to provide flow to the core during full-power operation. This leads to a number of design changes, including the use of a subdivided "chimney" section above the core.

Our evaluation was limited to the capabilities of TRACG to represent the major physical phenomena and was not a detailed assessment of the performance of an ESBWR.

Our review focused on several questions:

- How well does TRACG model the phenomena that have an important influence on ESBWR stability?

- Do the data from operating reactors and the test facilities accurately represent the phenomena that will exist in the ESBWR?
- Does TRACG adequately model two-phase flow in the chimney?
- Are the nodalization of the chimney and the associated computational scheme adequate to represent unsteady flow in the chimney?
- Does TRACG adequately model natural circulation oscillations?
- Are the predictions of pressure drop fluctuations in the core, the chimney, and other parts of the natural circulation loop reasonable?
- Is the interaction between criticality conditions and the void fraction, flow rate, and heat transfer fluctuations reasonably represented?
- Are the predicted transient responses and decay ratios credible?

In response, GE and the staff presented detailed calculations. There are several sources of data from operating BWRs that have experienced oscillatory behavior. Limited experimental data relevant to the ESBWR are also available. These data include void fraction measurements by Ontario Hydro in large-diameter pipes and transient tests at SIRIUS/CRIEPI which were specifically designed to model some features of the ESBWR.

GE presented several comparisons between TRACG predictions and data recorded at operating BWRs (Peach Bottom, La Salle, Leibstadt, and Dodewaard). These comparisons included scenarios during which the plants were operating at or close to natural circulation conditions. The comparisons indicated that the code has the ability to model the phenomena that are relevant to the ESBWR, and that it represents these oscillations with reasonable accuracy.

Based on comparisons with the Ontario Hydro tests, TRACG appears to provide a reasonable representation of the average void fraction in a large duct, such as the ESBWR chimney, as a function of flow rate and steam quality. At a meeting with our Thermal-Hydraulic Phenomena Subcommittee, GE also presented predictions for the ESBWR response to random void fraction fluctuations that were observed in some tests.

GE explored various nodalizations of the chimney. GE demonstrated that the computational scheme can describe void propagation without significant distortion, numerical diffusion, or artificial mixing. However, they presented other results which indicated that there could be significant numerical diffusion, leading to artificial attenuation of void waves, if the Courant number was not close to 1. GE was able to argue that the effects of this distortion were not significant for the particular case of the ESBWR response that they presented. However, GE and the staff will need to evaluate these effects carefully when more complete analyses are performed in support of the ESBWR design certification.

GE showed that TRACG modeled the main features of low-pressure (startup) oscillations observed in the CRIEPI/SIRIUS tests. These results were consistent with qualitative descriptions of the governing physical processes. High-pressure oscillations in CRIEPI/SIRIUS were also successfully modeled by TRACG. Natural circulation instability in the FRIGG tests, which used electrical heating and lacked the damping introduced by neutronic feedback in the ESBWR, was also successfully modeled by TRACG.

TRACG simulations of ESBWR transients displayed the usual density-wave oscillations that are familiar from BWR experience, but did not reveal significant natural circulation oscillations. GE and the staff provided detailed calculations and physical arguments to explain the absence of these oscillations to our satisfaction. This included presentation of the interaction between components of pressure drop and buoyancy fluctuations in components of the system. They also explained why criticality feedback tended to induce density-wave oscillations but suppress natural circulation oscillations.

The staff performed several useful confirmatory analyses. These included runs of the LAPUR and RELAP5 codes, and the use of a drift-flux void propagation model. In addition, the staff performed several sensitivity studies using TRACG to confirm the code's robustness. They also confirmed that the use of a low Courant number could lead to numerical diffusion.

On the basis of these detailed explanations we found the predicted transient responses to be credible and concluded that TRACG was able to model them adequately. We expect to consider them further during our review of the ESBWR design certification application.

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References:

1. Memorandum from David B. Matthews, Director, Division of New Reactor Licensing, Office of Nuclear Reactor Regulation, to John T. Larkins, Executive Director, Advisory Committee on Reactor Safeguards, "Draft Safety Evaluation for the Application of TRACG for ESBWR Stability," January 12, 2006.
2. Memorandum from Frank M. Akstulewicz to Laura A. Dudes, Safety Evaluation by the Office of Nuclear Reactor Regulation, "Application of the TRACG Computer Code to Stability Analysis for the ESBWR Design — NEDE-33083P, Supplement 1," March 28, 2006.
3. NEDE-33083P, Supplement 1, "TRACG Application for ESBWR Stability Analysis," General Electric Nuclear Energy, December 2004.
4. NEDE-32176P, Rev. 2, "TRACG Model Description," December 1999.
5. NEDE-33083P-A, "TRACG Application for ESBWR," March 2005.

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

May 5, 2006

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: DRAFT REGULATORY GUIDE DG-1144, "GUIDELINES FOR EVALUATING FATIGUE ANALYSES INCORPORATING THE LIFE REDUCTION OF METAL COMPONENTS DUE TO THE EFFECTS OF THE LIGHT-WATER REACTOR ENVIRONMENT FOR NEW REACTORS"

During the 532nd meeting of the Advisory Committee on Reactor Safeguards, May 4-5, 2006, the Committee considered draft regulatory guide DG-1144, "Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors." The Committee has no objection to the staff's proposal to issue DG-1144 for public comment. The Committee would like the opportunity to review the draft final version after reconciliation of public comments.

Reference:
Memorandum dated April 27, 2006, from Mark A. Cunningham, Director, Division of Fuel, Engineering, and Radiological Research, to John T. Larkins, Executive Director, ACRS, Subject: Request to Defer ACRS Review of Draft Regulatory Guide DG-1144, "Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors."

cc: A. Vietti-Cook, SECY
W. Dean, OEDO
B. Sosa, OEDO
M. Cunningham, RES
J. Uhle, RES
A. Lee, RES
H. Gonzales, RES
R. Assa, RES
J. Fair, NRR

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

May 8, 2006

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: CLINTON EARLY SITE PERMIT APPLICATION - FINAL SAFETY EVALUATION REPORT CHANGED PAGES PRIOR TO PUBLISHING AS A NUREG

During the 532nd meeting of the Advisory Committee on Reactor Safeguards, May 4-5, 2006, the Committee considered the changes reflected in Revision 4 of Exelon Generation Company (EGC), LLC, application for an early site permit (ESP). The changes included a revised analysis for determining the probable maximum flood (PMF) elevation at Clinton Lake for the EGC ESP. EGC also requested that the NRC staff consider adopting EGC's revised PMF elevation as the site characteristic in the NRC staff's final Safety Evaluation Report (SER) prior to issuing the final SER as a NUREG report. The staff has evaluated EGC's revised PMF analysis and the information in Revision 4 to the EGC ESP application and concluded that the revised analysis conservatively estimated the hydrostatic PMF elevation. The staff modified the final SER to document the basis for this conclusion. The changes to the final SER also include modifications to Section 2.4 to better describe the technical information in the application regarding EGC's ice thickness calculations and modifications to Appendix A to reflect the new PMF site characteristics. Finally, the staff revised the final SER to make some minor editorial changes. The Committee decided that the proposed changes do not affect its previous conclusions and recommendations with regard to issuing the ESP, and that additional review of this document prior to issuance is not necessary.

<u>References:</u>

1. Memorandum dated April 24, 2006, from David B. Matthews, Director, Division of New Reactor Licensing, NRR, to John T. Larkins, Executive Director, ACRS, Subject: ACRS Review of the Exelon Early Site Permit Application - Final Safety Evaluation Report Changed Pages Prior to Publishing as a NUREG.

2. U.S. Nuclear Regulatory Commission, Final Safety Evaluation Report, "Safety Evaluation of Early Site Permit Application in the Matter of Exelon Generation Company, LLC, for the Clinton Early Site Permit," dated February 17, 2006.

cc: A. Vietti-Cook, SECY
W. Dean, OEDO
B. Sosa, OEDO
S. Lee, OEDO
D. Matthews, NRR
W. Beckner, NRR
L. Dudes, NRR
J. Segala, NRR
C. Araguas, NRR
N. Patel, NRR
G. Wunder, NRR
S. (Min) Lee, NRR

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

May 17, 2006

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 BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2

Dear Chairman Diaz:

During the 532nd meeting of the Advisory Committee on Reactor Safeguards, May 4-5, 2006, we completed our review of the license renewal application for the Brunswick Steam Electric Plant (BSEP), 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 8, 2006. During these reviews, we had the benefit of discussions with representatives of the staff and the applicant, Carolina Power and Light (CP&L). We also had the benefit of the documents referenced. This report fulfills the requirement of 10 CFR 54.25, which requires that the ACRS review and report on all license renewal applications.

CONCLUSIONS AND RECOMMENDATIONS

1. The programs committed to and established by the applicant to manage age-related degradation provide reasonable assurance that BSEP, Units 1 and 2 can be operated in accordance with their current licensing basis for the period of extended operation with no undue risk to the health and safety of the public.

2. CP&L's application for renewal of the operating licenses for BSEP, Units 1 and 2 should be approved.

3. The staff's new two-tiered process for reviewing the scoping of balance of plant (BOP) systems was effective and improved the efficiency of the review. This process should be used by the staff in its review of future license renewal applications.

BACKGROUND AND DISCUSSION

BSEP consists of two boiling water reactor (BWR) units that were built on a site located south of Wilmington, NC at the mouth of the Cape Fear River in Brunswick County. The current operating licenses will expire on September 8, 2016 for Unit 1 and December 27, 2014 for Unit 2. The applicant has requested renewal of these licenses for an additional 20 years.
These units are General Electric BWRs with Mark I containments. Each unit is authorized to operate at 2,923 MWt. The main condensers are cooled by a once-through circulating water system using cooling water from the Cape Fear River estuary.

In the final SER, the staff documented its review of the license renewal application 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 systems, structures, 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 structures and components; the adequacy of the applicant's aging management programs (AMPs); and the identification and assessment of time-limited aging analyses.

The application demonstrates consistency with, or justifies deviations from, the approaches specified in the Generic Aging Lessons Learned Report. The applicant has correctly identified those long-lived passive SSCs from both units that fall within the scope of license renewal. The applicant performed an aging management review of components in scope. Based on the results of this review, the licensee will apply 20 Aging Management Programs (AMP) to both units and 14 additional AMPs which are specific to one unit or the other. Of the 34 AMPs, 26 are existing AMPs and 8 are new AMPs.

This application was the first to be reviewed using a new two-tiered process for the scoping of BOP systems. In Tier 1, the license renewal application and the Updated Final Safety Analysis Report were reviewed to identify apparent missing components for an aging management review. In Tier 2, the license renewal boundary drawings and other licensing basis documents were reviewed in addition to the license renewal application and the Updated Final Safety Analysis Report. The screening criteria used to identify systems for the detailed Tier-2 review are based on: safety importance/risk significance; potential for system failure to cause failure of redundant safety system trains; operating experience indicating likely passive failures; and experience from reviews of previous license renewal applications indicating likely omissions. For this license renewal application, 15 BOP systems received a Tier-1 review and 24 BOP systems received a Tier-2 review. The two-tiered review process was effective and the staff should continue to use this process in reviewing future license renewal applications.

The staff conducted an inspection and an audit of this license renewal application. The inspection was performed to verify that the scoping and screening methodology was consistent with the regulations and adequately reflected in the application. The audit verified that the AMPs and the Aging Management Reviews are adequate. Based on the inspection and audit, the staff concluded that the license renewal activities are consistent with the descriptions contained in the CP&L license renewal application. Also, the staff concluded that existing programs to be credited as AMPs for license renewal are generally functioning well and that an implementation plan had been established in the applicant's Action Request System to track license renewal commitments to ensure their timely completion.

Analyses of reactor vessel neutron embrittlement (upper shelf energy 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 construction details of the Mark I containments used in this plant are unique. The drywell uses reinforced concrete as the load bearing structural component with an inner liner of carbon steel which serves as a leak-tight membrane. While liner integrity is important to ensure leak tightness, the structural integrity of the liner is not important in maintaining the integrity of the pressure boundary. The applicant proposes a combination of visual inspections to detect liner bulges and corrosion as well as the integrated leak rate tests as an adequate containment liner AMP. The staff has accepted this approach. We concur.

No open items or confirmatory items have been identified in the SER. CP&L has made 31 commitments related to establishing AMPs to manage aging effects for structures and components identified during the scoping review. The staff has included appropriate license conditions in the final SER to satisfy remaining documentation issues and action items. No changes in the technical specifications for BSEP are required.

CP&L submitted a well prepared application for renewal of the licenses for BSEP, Units 1 and 2, which resulted in a reduction in the number of Requests for Additional Information (RAIs). CP&L's responses to the staff's RAIs were thorough and timely. The staff's evaluation was technically comprehensive and well documented in the final SER.

No issues related to the matters described in 10 CFR 54.29(a)(1) and (a)(2) preclude renewal of the operating licenses for BSEP, Units 1 and 2. The programs committed to and established by the applicant provide reasonable assurance that BSEP, Units 1 and 2 can be operated in accordance with their current licensing basis for the period of extended operation with no undue risk to the health and safety of the public. The application for renewal of the operating licenses for BSEP, Units 1 and 2 should be approved.

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References:

1. U.S. Nuclear Regulatory Commission, "The Final Safety Evaluation Report Related to the License Renewal of the Brunswick Steam Electric Plant, Units 1 and 2," dated March 2006.
2. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the License Renewal of the Brunswick Steam Electric Plant, Units 1 and 2," dated December 2005.
3. Progress Energy Carolinas, Inc., "Brunswick Steam Electric Plant, License Renewal Application," dated October 18, 2004.
4. U.S. Nuclear Regulatory Commission, "Brunswick Steam Electric Plant - Inspection Report 05000325/2005008 and 05000324/2005008," dated July 22, 2005.
5. Brookhaven National Laboratory, "Audit and Review Report for Plant Aging Management Reviews and Programs Brunswick Steam Electric Plant, Units 1 and 2," dated June 21, 2005.
6. U.S. Nuclear Regulatory Commission, "Generic Aging Lessons Learned Report," NUREG-1801, Vol. 1-2, Rev. 1, dated September 2005.

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

May 17, 2006

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

SUBJECT: MODIFIED DRAFT FINAL REVISION 4 TO REGULATORY GUIDE 1.97, "CRITERIA FOR ACCIDENT MONITORING INSTRUMENTATION FOR NUCLEAR POWER PLANTS"

Dear Mr. Reyes:

During the 530th meeting of the Advisory Committee on Reactor Safeguards, March 9-11, 2006, we reviewed the draft final Revision 4 to Regulatory Guide 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," and provided comments in our letter dated March 28, 2006. During our 532nd meeting, May 4-5, 2006, we reviewed an alternative proposal by the staff to accommodate the comments and recommendations included in our March 28, 2006 letter. During our review, we had the benefit of discussions with representatives of the NRC staff and industry. We also had the benefit of the documents referenced.

RECOMMENDATION

The staff should issue the modified Revision 4 to Draft Regulatory Guide 1.97 as final.

DISCUSSION

Draft final Revision 4 to Regulatory Guide 1.97 endorses IEEE Standard 497-2002, "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations," with exceptions. IEEE Std 497-2002 is intended to supersede IEEE Std 497-1981 and IEEE Std 497-1983. This revised Standard provides a consolidated source of post-accident monitoring requirements, the associated bases, and a new method for selecting and applying criteria to accident monitoring instrumentation. It is primarily intended for new nuclear power plants, though it also contains appropriate guidance and provides a flexible basis for making changes to such systems in operating plants.

In our letter dated March 28, 2006, we recommended that Draft Final Revision 4 to Regulatory Guide 1.97 not be issued in its then current form. In particular, we stated, "The staff should revise Regulatory Position 1 to allow licensees to adopt the IEEE 497-2002 Standard to modify individual accident monitoring instruments without a complete analysis of all accident monitoring instrumentation." We agreed with the staff's position "that licensees should not be allowed to use the IEEE 497-2002 Standard to eliminate or reclassify accident monitoring instrumentation required by previous editions of this Standard unless Revision 4 to Regulatory Guide 1.97 is adopted in its entirety."

The staff has now proposed a more flexible alternative to Regulatory Position 1. Specifically, the staff deleted the previous guidance regarding partial conversions and added the following new guidance regarding modifications:

> "If the licensee voluntarily uses the criteria in Revision 4 of this guide to perform modifications that do not involve a conversion, the licensee should first perform an analysis to determine the complete list of accident monitoring variables and their associated types in accordance with the selection criteria in Revision 4."

The staff's proposed change to Regulatory Position 1 meets the intent of our recommendations. It provides assurance that the licensee and the staff will have the information needed to review the basis for proposed modifications. It provides sufficient flexibility to apply IEEE Std 497-2002 to accident monitoring instrument replacements and modifications in existing plants.

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References:
1. Memorandum from C. Paperiello, Director, Office of Nuclear Regulatory Research, to J. Larkins, Executive Director, Advisory Committee on Reactor Safeguards, Subject: Request for ACRS Review of Regulatory Guide 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," Revision 4, dated January 30, 2006.
2. Regulatory Guide 1.97 (Draft was issued as DG-1128, dated June 2005), "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," Revision 4, dated April 2006.
3. IEEE Standard 497-2002, "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Generating Stations," dated September 30, 2002.
4. Letter from G. Wallis, Chairman, Advisory Committee on Reactor Safeguards, to L. Reyes, Executive Director for Operations, NRC, Subject: Draft Final Revision 4 to Regulatory Guide 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," dated March 28, 2006.
5. Letter from L. Reyes, Executive Director for Operations, NRC, to G. Wallis, Chairman, Advisory Committee on Reactor Safeguards, Subject: Draft Final Revision 4 to Regulatory Guide 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," dated April 20, 2006.

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

May 22, 2006

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

SUBJECT: PROPOSED REVISIONS TO 10 CFR PART 52: LICENSES, CERTIFICATIONS, AND APPROVALS FOR NUCLEAR POWER PLANTS, AND CONFORMING AMENDMENTS TO APPLICABLE NRC REGULATIONS

Dear Chairman Diaz:

During the 532nd meeting of the Advisory Committee on Reactor Safeguards (ACRS), May 4–5, 2006, we reviewed the proposed revisions to 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," and conforming amendments to 10 CFR Parts 1, 2, 10, 19, 20, 21, 25, 26, 50, 51, 54, 55, 72, 73, 95, 140, 170, and 171. During our review, we had the benefit of discussions with representatives of the NRC staff and the documents referenced.

RECOMMENDATIONS

1. A level-3 probabilistic risk assessment (PRA) consequence analysis should not be required at the early site permit (ESP) stage.

2. We support a requirement for the combined license (COL) applicant to maintain an up-to-date PRA. Updates to the PRA need not be submitted to the NRC.

3. It should be sufficient for the ESP applicant to identify only the major features of the site emergency plan. The definitions of major features should be specified in regulatory guidance documents.

4. We agree with the staff that a new paragraph (e) to 10 CFR 50.47, "Emergency plans," should be added, along with a coordinated revision of 10 CFR 50.54, "Conditions of Licenses," to allow operation up to 5 percent power even with deficiencies in emergency preparedness identified by the Federal Emergency Management Agency (FEMA), as is currently allowed for nuclear power plants licensed under Part 50.

DISCUSSION

The ACRS considers a 10 CFR Part 100, "Reactor Site Criteria," radiological analysis to be an adequate characterization of a site for the purpose of an ESP. At the ESP stage, there is insufficient design detail to make a level-3 radiological consequence analysis meaningful.

We support a requirement for the COL applicant to maintain an up-to-date PRA. Updates to the PRA need not be submitted to the NRC. The updated PRA should be available at the licensee's site for inspection by the NRC.

One of the lessons learned from existing ESP applications is that significant impediments to emergency planning are not anticipated. This is because it is unlikely that a site with a significant impediment would be proposed for an ESP. It should be sufficient for an ESP applicant to identify the major features of the emergency plan. Experience has shown, however, that the definition of "major features" should be clarified in guidance documents available to ESP applicants.

We support the addition of a new paragraph (e) to 10 CFR Part 50.47 and the revision to 10 CFR 50.54. Even if FEMA has identified deficiencies in the emergency plan after the plant has been made ready for operation, operation at up to 5 percent power level for a limited period of time should be acceptable from a site risk viewpoint. The economic risk is that the plant subsequently may not be allowed to operate if the deficiency cannot be sufficiently remedied. It should be up to the licensee to decide whether to accept such a risk.

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

<u>References</u>:

1. SECY-05-0203, Revised Proposed Rule to Update 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," dated November 3, 2005.
2. Staff Requirements Memorandum - SECY-05-0203 - Revised Proposed Rule to Update 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," dated January 30, 2006.
3. *Federal Register* Notice: Proposed Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," 71 FR 12782-12932.
4. Letter dated December 14, 2005, from Marvin S. Fertel, Nuclear Energy Institute, to Nils J. Diaz, Chairman, NRC, Subject: Industry Comments on the Part 52 Rulemaking Proposal.

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

May 22, 2006

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

SUBJECT: BEAVER VALLEY EXTENDED POWER UPRATE APPLICATION

Dear Chairman Diaz:

During the 532nd meeting of the Advisory Committee on Reactor Safeguards, May 4-5, 2006, we discussed the Extended Power Uprate (EPU) Application for the Beaver Valley Power Station (Beaver Valley), Units 1 and 2 and the associated NRC staff's Safety Evaluation. Our Subcommittee on Power Uprates also discussed this application on April 24-25, 2006. During our review, we had the benefit of discussions with the staff and representatives of FirstEnergy Nuclear Operating Company (FENOC), the licensee. We also had the benefit of the documents referenced.

RECOMMENDATION

The application for a power uprate at Beaver Valley should be approved.

BACKGROUND

FENOC has applied for an upgrade of Beaver Valley Units 1 and 2 from the current power level of 2689 MWt to 2900 MWt, an increase of approximately 8 percent. The uprated power level will be comparable to that of similar units at North Anna, V.C. Summer, Shearon Harris, and Vandellos (Spain). The power increases can be implemented with minor changes in the plant configuration and operating practices.

In anticipation of its power uprate request, FENOC initiated a number of licensing actions. These include an enrichment limit increase for the new fuel storage racks, a slightly positive moderator temperature coefficient at low power, an increase in the boron concentration in the accumulators and refueling water storage tank, selective implementation of the alternative source term model, and a relaxation of the axial offset control requirements. The licensee replaced the reactor vessel head and steam generators in Unit 1 during the outage completed in April 2006. The new steam generator tubes are manufactured from Alloy 690, which has greater corrosion resistance than the Alloy 600 tubes used in the original steam generators. The steam generators in Unit 2 do not require replacement at this time. The additional plant modifications to enable the 8 percent upgrades include replacement of the high-pressure turbines, staking of the Unit 2 main condenser, modifications to the fill at the Unit 2 cooling tower, replacement of the turbine-generator rotors, rewinding of the Unit 1 turbine generator rotors, and modifications to some valves.

The licensee will continue to use the Westinghouse RFA fuel design for the EPU condition. This fuel design is based on a 17x17 assembly with intermediate flow mixing grids. These mixing grids provide enhanced margin to departure from nucleate boiling in the upper portions of the fuel rod. This enhanced cooling capability is part of the reason that the reactor can be operated at uprated conditions with minimal impact on thermal margins for anticipated operational transients. The plant has six operating cycles of experience with these fuel assemblies. The cores of the units have been completely converted to this fuel design.

Safety Analysis Results

The nominal core outlet temperature for Unit 1 will be increased for the EPU condition by 4 °F to 611 °F. For Unit 2, the core outlet temperature will be unchanged but the inlet temperature will be reduced by 5 °F. The core flow rates will be unchanged. On the secondary side, the mass flow rates will increase almost proportionally to the power uprate. The increased primary system temperature could increase the rate of corrosion of components. The increased secondary side flow rate could lead to accelerated corrosion and fluid/structure interactions.

A variety of transients have been analyzed for the EPU condition. The results of these analyses satisfy the regulatory criteria.

A spectrum of loss-of-coolant accidents (LOCA) was analyzed for EPU conditions. For the large-break LOCA, FENOC used a best-estimate methodology. The predicted peak clad temperatures have significant margin to the regulatory limit of 2200 °F. The limiting quantity of hydrogen generated is close to the regulatory limit of 1 percent but the methodology for calculating hydrogen generation is conservative. Small-break LOCA analyses were also performed for a spectrum of break sizes. The results satisfy regulatory criteria with substantial margin.

In addition to demonstrating compliance with acceptance criteria, analyses were performed to examine the potential for boric acid precipitation in the core region during the long-term cooling phase following a LOCA in the cold leg. As a result of these analyses, changes will be made in the emergency operating procedures to shorten the time at which the operators will initiate hot-leg injection of emergency coolant to flush the core region. With these changes, the analyses indicate that adequate margin to the boron solubility limit will exist.

In our report of February 24, 2005 related to the Waterford 3 uprate, we indicated the need for the staff to develop a better understanding of the properties of highly concentrated boric acid in a boiling system. A more detailed treatment of the thermal-hydraulic conditions within the core region is needed to better define the conditions leading to recirculation and mixing within the vessel and lower plenum. In its response to our letter, the staff stated that this issue should be addressed by the industry as part of satisfying the long-term cooling requirements of 10 CFR 50.46. We look forward to reviewing progress on this issue.

Containment Analysis

The containment systems for both units have been converted to a slightly higher, but still subatmospheric, operating pressure. Containment pressurization calculations that were performed for the design basis LOCA and steam line break confirm that the peak pressure is below the design limit.

For Unit 1, containment overpressure credit has been granted by the staff to provide net positive suction head for the containment spray pumps that recirculate coolant from the containment sump. Containment spray flow through heat exchangers provides long-term removal of heat during a LOCA. The duration of time for which overpressure credit is required is less than 20 minutes. FENOC provided results from tests performed on this pump design that demonstrate an ability to operate for this period without damage. Under EPU conditions, the amount of overpressure and duration of credit required are only slightly increased. We concur with the staff's decision to grant overpressure credit under these conditions. Because of a difference in the location of the pumps in Unit 2, no overpressure credit is required.

Reactor Vessel Integrity

The power uprate will lead to additional fluence and embrittlement of the reactor vessel at the end of life for the two units. Based on results obtained from surveillance capsules, FENOC has estimated the shift in the pressurized thermal shock reference temperature (RT_{PTS}) at the end of extended life. These estimates have been independently confirmed by the staff. The final value of RT_{PTS} for each vessel is less than the pressurized thermal shock screening criterion of 270°F. The upper shelf energies exceed 50 ft-lbs. We conclude that radiation-induced vessel embrittlement is a manageable issue at the power uprate conditions.

Component Vibration

FENOC has performed a systematic assessment of components for which vibration could be induced by higher velocities following the power uprates. The main steam condenser at Unit 2 will be staked; the Unit 1 condenser was staked previously. There is extensive industry operating experience with the steam generators in use at both units for the conditions that will be encountered at Beaver Valley without any indication of vibration-induced failures. The steam dryers in these units are subject to much lower flow velocities than those in boiling water reactors for which flow-induced vibrations have been a power uprate issue. FENOC has committed to performing pre-EPU and post-EPU walkdowns to identify vibration issues should they occur.

Flow-Accelerated Corrosion

FENOC has used the CHECWORKS code to predict the rate of wall thinning that could result from the higher flow rates following the EPU. The predicted changes in corrosion rates are small. These results are used primarily to prioritize monitoring activities. The affected components are on the secondary side of the plant. FENOC has a program in which components with materials that are subject to flow-accelerated corrosion are replaced with chromium-molybdenum steels, as the opportunities arise. Flow-accelerated corrosion under EPU conditions can be effectively managed under the existing monitoring program.

Risk Assessment

The licensee performed quantitative assessments of the changes in risk associated with EPU for internal events, fires, and seismic events for operation at full power. These assessments were confined to changes in core damage frequency (CDF) and the large early release frequency (LERF) and did not consider the impact of the increase in the radioactive inventory on risk. The changes associated with the power uprates at the two Beaver Valley units have very little impact on the CDF and LERF. Changes in the time periods available for critical operator actions were assessed using table-top and simulator exercises. These were then reflected as changes in human error rates in the probabilistic risk assessments. The assessed changes in failure probabilities are small.

Power Ascension and Testing

FENOC has developed a testing plan to assure the proper performance of modified components, settings and controls following power uprate. For each Unit, the power ascension will be performed in three steps. The first step of 3 percent will be made in the current operating cycle for Unit 1 and the next operating cycle for Unit 2. The plant will continue to operate at the 3 percent increased power level until the following refueling outage. In the subsequent outage, the final ascension to full EPU will be performed in two steps of 2.5 percent each. Following each step, the licensee will evaluate the plant operation and determine whether the unit is operating as expected. We concur with the staff's conclusion that large integral transient tests are not warranted.

Summary

The proposed power uprates at Beaver Valley Units 1 and 2 will have very little impact on the manner in which the units are operated. There are no identified areas in which safety margins would be substantially reduced or conflict with regulatory criteria. The Beaver Valley power uprate application should be approved.

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References:

1. Memorandum from Catherine Haney to John Larkins, "Beaver Valley Power Station, Unit Nos. 1 and 2 (BVPS-1 and 2) - Revised Draft Safety Evaluation for Proposed Extended Power Uprate (TAC Nos MC4645 and MC4646)," dated April 13, 2006.

2. Letter from L. William Pearce to U.S. Nuclear Regulatory Commission, "Beaver Valley Power Station, Unit No. 1 and No. 2, BV-1 Docket No. 50-334, License No. DPR-66, BV-2 Docket No. 50-412, License No. NPF-73, License Amendment Request Nos. 302 and 173," dated October 4, 2004.

3. Report dated February 24, 2005, from Graham B. Wallis, Chairman, ACRS, to Nils J. Diaz, Chairman, NRC, Subject: Waterford Steam Electric Station, Unit 3 - Extended Power Uprate.

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

May 22, 2006

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

SUBJECT: R.E. GINNA EXTENDED POWER UPRATE APPLICATION

Dear Chairman Diaz:

During the 532nd meeting of the Advisory Committee on Reactor Safeguards, May 4-5, 2006, we reviewed the Extended Power Uprate (EPU) application for the R.E. Ginna Nuclear Power Plant (Ginna) and the associated NRC staff's Safety Evaluation. Our Subcommittee on Power Uprates also reviewed this matter on March 14-15, 2006 and April 27, 2006. During these reviews, we had the benefit of discussions with the staff and representatives of Constellation Energy, the licensee. We also had the benefit of the documents referenced.

RECOMMENDATION

The application for a power uprate at the R.E. Ginna Nuclear Power Plant should be approved.

BACKGROUND

Ginna is a two-loop, Westinghouse-designed pressurized water reactor operating at a power level of 1520 MWt. Constellation Energy has applied for a power uprate of 17 percent to 1775 MWt. Kewaunee, a plant of similar design, is licensed to operate at a comparable level of 1772 MWt. The steam generators in Ginna were replaced in 1996, and the reactor vessel head was replaced in 2003.

A number of design changes are being made to support the power uprate. The current fuel design is being replaced by the Westinghouse 422V+ design. The newer design has a slightly longer fuel stack and a larger diameter fuel pin. These changes help preserve operating margins in the plant after the power uprate. The new fuel design also has improved resistance to vibration. This fuel will be introduced over the next three cycles. Other important changes to support the power uprate include replacement of the high-pressure turbine and turbine control valves.

Safety Analysis Results

At the uprated power, the nominal outlet temperature will increase from 590 °F to 607 °F. The primary coolant flow rate will be essentially unchanged. The secondary side flow rate will be increased by 18%, approximately equal to the percentage uprate in power. The increased primary system temperature could increase the rates of corrosion of components. The increased secondary side flow rate could lead to accelerated corrosion and fluid/structure interactions.

A number of transients have been analyzed at the uprated power to determine if they satisfy safety criteria for departure from nucleate boiling, overpressurization of the primary system, overpressurization of the secondary system, or conditions that could result in a more severe event, such as over-filling of the pressurizer. Constellation Energy provided information on the degree of conservatism in the analyses and on the validation of the analytical tools used. We concur with the staff's acceptance of these results.

A full spectrum of loss-of-coolant accident (LOCA) events has been analyzed at the uprated power. The results of these analyses show substantial margin to the established regulatory limits on peak clad temperature, oxidation, and hydrogen generation. The emergency core cooling system configuration at Ginna is somewhat different from later plant designs. The high-capacity, low-pressure system injects through two lines directly into the upper plenum. The high-pressure system also has high capacity and the accumulators inject at a relatively high pressure of 700 psia. This configuration of systems is quite effective in providing cooling over the entire spectrum of breaks.

At the time at which recirculation is initiated in a large LOCA, the sump temperature is too high to meet the net positive suction head limits for the high-pressure injection system. Thus, when recirculation is initiated, the low-pressure upper plenum injection system is switched from the injection mode to the recirculation mode but the high-pressure injection system is turned off. For a hot-leg break, there is some concern that, with injection occurring only on the hot side of the core, emergency core cooling water could escape out the break without effectively mixing in the core. Boric acid could concentrate within the vessel and potentially deposit within the core region. The licensee has performed analyses to determine when cold-leg injection should be reinitiated to flush the system and ensure that the concentration of boric acid does not approach saturation. The emergency operating procedures have been modified accordingly.

System Impacts

An assessment of the effect of the increased power output of the plant on grid stability indicates that the grid can withstand a trip of the unit from the EPU condition. The plant's ability to cope with a four-hour station blackout is virtually unaffected because the DC system loads are not significantly increased by the power uprate and substantial margin previously existed.

The CHECWORKS code was used to assess the impact of increased secondary side flow on flow accelerated corrosion. No components need to be replaced. Some inspection sampling rates will be increased based on this assessment. The plant's monitoring program is adequate to ensure the control of increased corrosion rates should they occur.

The potential for flow-induced vibration associated with higher secondary side flow rates has been assessed for the steam generators, feedwater heaters, condenser tubes, and moisture separator reheaters. Within the vibration monitoring program, a baseline will be established by a walkdown prior to EPU. After EPU plant modifications have been made, walkdowns will be performed at the initial power level and at the uprated power level.

Because the temperatures in the primary system will be somewhat higher after EPU, we requested that the licensee identify those components that contain Alloy 600 and its associated weld materials (Alloy 82/182) for which increased stress corrosion cracking might be expected. The licensee explained that these components are all located in regions of the primary system that will not experience high temperatures or are not load bearing.

Risk Assessment

The licensee performed a quantitative assessment of the change in risk associated with EPU for internal events, external events, and shutdown risk. This assessment was confined to changes in core damage frequency (CDF) and large early release frequency (LERF) and did not consider the impact of the increase in the radioactive inventory on risk. Changes were considered in initiating event frequencies, success criteria, equipment failure times, and operator response times. Some changes were required in success criteria. Significant reductions were identified in the time available for some key operator actions. In all cases, table top and simulator analyses indicated that the available time was sufficient for these actions. However, the human failure probabilities were increased.

The largest impacts on CDF and LERF were obtained for the internal events and shutdown risks, where the estimated increases were on the order of 20%. The post-uprate value for overall CDF is 7×10^{-5} per yr and for LERF is 5×10^{-6} per yr, which represents approximately a 10% increase in each. Although these changes fall within values that are typically considered acceptable, the licensee undertook an evaluation of plant changes that could be made at the time of the power uprate that would result in an overall decrease in CDF. The licensee has committed to undertaking a set of modifications that will have a net impact on CDF and LERF such that after the EPU, the CDF and LERF will be slightly less than the pre-EPU values.

Power Ascension and Testing

The power escalation test plan extends over an eleven day period. During the first day, a number of low-power tests will be performed including a manual turbine trip from 30% power. In the second day, the power level will be raised from 30% to the old full power, which is 85% of the uprated power. The remaining increases will be made in five steps of 3%. Each increase will be followed by one day of testing before proceeding to the next step. There are no large integral tests planned at full power. By design, a turbine trip at full power would lead to a reactor trip. The planned turbine trip from 30% power is more challenging than a full power trip would be to the systems that control rod position, steam dump, and pressurizer level.

Summary

Although the proposed power uprate at Ginna represents a significant change in operating conditions, similar pressurized water reactors are operating at comparable conditions. While the uprate will lead to a decrease in margins, the remaining safety margins are sufficient to ensure that safety limits will not be challenged by anticipated operating occurrences. The plant also satisfies regulatory criteria for loss of coolant accidents with substantial margin. The power uprate application should be approved.

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

ADDITIONAL COMMENTS BY ACRS MEMBER G.E. APOSTOLAKIS

I agree with the recommendations of the report. I am writing these comments to bring to the Commission's attention a general issue related to human reliability analysis (HRA).

The major impact of extended power uprates (EPUs) is on human performance. The higher power shortens the time available to the operators for action. This necessitates an estimation of the change in human error probabilities from the base case.

The licensees usually use the EPRI "calculator" to estimate these changes. We were told by EPRI and industry representatives at a Subcommittee meeting in December 2005 that the calculator itself is simply a computer program that facilitates the use of four HRA models. To my knowledge, the calculator and its models have not been reviewed by the NRC staff.

As an example, the Ginna EPU application states (Table 2.13-13) that, for the event FSHFDAWXX-2 (operator fails to manually align and start the turbine-driven auxiliary feedwater pump under certain conditions), the time for action is reduced from 47 minutes to 34. Table 2.13-14 lists a base value of the human error probability of 8.60×10^{-2} and an EPU value of 2.25×10^{-1}.

This change indicates that the EPRI model is remarkably accurate with respect to changes in the available time. Both of the NRC HRA models (ATHEANA and SPAR-H) treat the available time as one of many "performance shaping factors," i.e., factors that affect the judgements of experts when they evaluate human error probabilities.

The same issue arose when we reviewed Regulatory Guide 1.205, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants." The industry guidance document (NEI 04-02, Revision 1) recognized explicitly that, under fire conditions, it would be important to estimate the amount of time available to the licensee to actually complete a recovery action.

At this time, the agency does not have a tool to estimate human error probabilities as a function of available time for action. In addition, I don't believe that the staff should be accepting the results of the EPRI calculator without a review of its models.

My comments do not affect the recommendation to approve the Ginna EPU request. I agree with the staff's finding that "the licensee's HRA and its associated results are reasonable for this application."

ADDITIONAL COMMENTS BY ACRS MEMBERS T. KRESS, D. POWERS, AND G. E. APOSTOLAKIS

The assessed risk impacts of significant power uprate requests have been universally limited to ΔCDF and ΔLERF. The assessed changes to these metrics do not reflect the increase in fission-product inventory and invariably, turn out to be small. It is clear, however, that the real societal risk impact [total probabilistic deaths, injuries, and land contamination] due to the increased fission product inventory are at least as large as the total increase in site power.

With respect to applications for significant power uprates, PRA level-3 impacts are neither assessed nor reported. In addition, there are no criteria to judge the acceptability of such risk increases. These are regulatory shortcomings that need attention. Guidance on how to judge the acceptability of increases in societal risk is needed to be incorporated into the review standard. In a risk-informed regulatory system, a level-3 assessment should be part of the staff's review of the acceptability of any power uprate application.

<u>References</u>:

1. Memorandum from Catherine Haney to John Larkins, "R.E. Ginna Nuclear Power Plant - Draft Safety Evaluation for Proposed Extended Power Uprate (TAC No. MC7382)," dated March 6, 2006.
2. Memorandum from Catherine Haney to John Larkins, "R.E. Ginna Nuclear Power Plant - Draft Safety Evaluation for Proposed Extended Power Uprate (TAC No. MC7382)," dated April 6, 2006.
3. Letter from Mary G. Korsnick (Constellation Energy) to U.S. Nuclear Regulatory Commission, "License Amendment Request Regarding Extended Power Uprate," dated July 7, 2005.

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

June 6, 2006

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: DRAFT FINAL REVISION 1 TO REGULATORY GUIDE 8.38, "CONTROL OF ACCESS TO HIGH AND VERY HIGH RADIATION AREAS IN NUCLEAR POWER PLANTS"

During the 533rd meeting of the Advisory Committee on Reactor Safeguards, May 31- June 1, 2006, the Committee considered the draft final Revision 1 to Regulatory Guide 8.38, "Control of Access to High and Very High Radiation Areas in Nuclear Power Plants." The Committee decided not to review Revision 1 to Regulatory Guide 8.38 and has no objection to the staff's proposal to issue this Guide.

Reference:
Memorandum dated May 18, 2006, from Mark A. Cunningham, Director, Division of Fuel, Engineering and Radiological Research, RES to John T. Larkins, Executive Director, ACRS, Subject: Request to Waive ACRS Review of Revision 1 of Regulatory Guide DG-8028, "Control of Access to High and Very High Radiation Areas in Nuclear Power Plants."

cc: A. Vietti-Cook, SECY
W. Dean, OEDO
J. Lamb, OEDO
B. Sosa, OEDO
B. Sheron, RES
M. Cunningham, RES
N. Chokshi, RES
H. Karagiannis, RES
R. Assa, RES
J. Dyer, NRR
B. Boger, NRR
R. Pederson, NRR
M. Lee, NRR

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

June 15, 2006

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

SUBJECT: DRAFT FINAL GENERIC LETTER 2006-XX, "INACCESSIBLE OR UNDERGROUND CABLE FAILURES THAT DISABLE ACCIDENT MITIGATION SYSTEMS"

Dear Mr. Reyes:

During the 533rd meeting of the Advisory Committee on Reactor Safeguards, May 31 - June 1, 2006, we reviewed the draft final Generic Letter 2006-XX, "Inaccessible or Underground Cable Failures that Disable Accident Mitigation Systems." 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.

RECOMMENDATION

Generic Letter 2006-XX, "Inaccessible or Underground Cable Failures that Disable Accident Mitigation Systems," should be issued.

DISCUSSION

Cables in inaccessible locations such as buried conduits or direct-buried installations can be exposed to moisture from condensation and flooding. Cables in these environments can fail due to water treeing or other mechanisms that reduce the dielectric strength of the insulation material. Some of these inaccessible or underground cables are used to energize safety-related systems.

On March 21, 2002, the staff issued Information Notice (IN) 2002-12 to alert licensees about the effects of moisture on cable performance. IN 2002-12 described medium-voltage safety-related cable failures at several plants as a result of long-term flooding problems in trenches and conduits. Upon further review of operating experience, the staff found 23 licensee event reports and two morning reports since 1988 on failures of buried medium-voltage cables due to insulation failure. The staff believes that this represents a small fraction of the total number of failures since not all cable failures are reportable. None of the failed cables was designed or qualified for long-term wetting or submergence.

The staff is particularly concerned that more than one safety-related cable could fail on demand as a result of undetected degradation of inaccessible cables exposed to wet environments for which they have not been qualified. This could result in multiple equipment failures. In certain applications, the incipient failure of cables can go undetected because they are not energized during power operation. In addition, degraded cables may survive short-term, periodic functional tests, but fail during the extended duty imposed by operation during accident mitigation. The staff further believes that condition monitoring would provide early indication of degrading insulation. The Generic Letter will allow the staff to gather more information on power cable failures experienced by the plants to-date and on plant-specific programs to detect degradation or reasons why such programs are not necessary.

A number of licensees and NEI commented on the proposed Generic Letter. NEI also summarized operating experience with medium-voltage, wetted, and energized cables at a large fraction of domestic plants in NEI 06-05, "Medium Voltage Underground Cable White Paper." The industry contends that only a small number of plants have experienced cable failures and that there does not appear to be an increasing trend. The affected plants have promptly replaced the failed cables and addressed the conditions that caused the failures. Based on this operating experience, industry concludes that the likelihood of common-cause failure of multiple systems is extremely low and reliance on functional testing is sufficient.

Since cable degradation and failure are assisted by aging, the number of failures experienced to-date, by itself, is not necessarily a good predictor of future performance. Also, functional testing of equipment powered by the cables does not provide information on whether these cables, exposed to an adverse environment for which they are not qualified, are experiencing incipient degradation that could lead to failure in service. The population of cables in this condition may be significantly larger than the number of failures experienced to date.

In addition to the experience reported in IN 2002-12, during our reviews of license renewal applications, we have encountered several plants that have experienced failures of inaccessible cables, as well as flooding of inaccessible cable raceways and conduits. Many of these cables were in safety-related applications and were not qualified for this environment. Regulations require that these cables be capable of performing their intended functions in anticipated environmental conditions. Consequently, the Generic Aging Lessons Learned Report describes an acceptable program for managing aging of cables such that their intended functions will be maintained consistent with the current licensing basis for the period of extended operation. This program includes periodic inspections to address water collection problems and assessments of insulation condition.

Since failures are occurring during the current licensing term, information should be gathered to determine if existing licensee programs are sufficient to address these issues now.
The Generic Letter will allow the staff to better understand the extent of the problem with inaccessible or underground power cable failures and the current industry initiatives to detect degradation before failure occurs or the reasons why such initiatives are not needed. The Generic Letter should be issued.

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References:

1. Memorandum from Michael E. Mayfield, Director, Division of Engineering, Office of Nuclear Reactor Regulation to John T. Larkins, Executive Director, ACRS, dated May 15, 2006, Subject: Request for Review and Endorsement by the Advisory Committee on Reactor Safeguards (ACRS) Regarding the Proposed Generic Letter 2006-XX, "Inaccessible or Underground Cable Failures that Disable Accident Mitigation Systems."
2. U.S. Nuclear Regulatory Commission, Information Notice 2002-12, "Submerged Safety-Related Electrical Cables," March 21, 2002.
3. Nuclear Energy Institute, "Medium Voltage Underground Cable White Paper," NEI 06-05, April 2006.
4. U.S. Nuclear Regulatory Commission, "Generic Aging Lessons Learned (GALL) Report," NUREG-1801, Rev. 1, September 2005.

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

June 16, 2006

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

SUBJECT: DRAFT FINAL GENERIC LETTER 2006-XX: POST-FIRE SAFE-SHUTDOWN CIRCUIT ANALYSIS SPURIOUS ACTUATIONS

Dear Mr. Reyes:

During the 533rd meeting of the Advisory Committee on Reactor Safeguards, May 31-June 1, 2006, we reviewed the Draft Final Generic Letter (GL) 2006-XX: Post-Fire Safe-Shutdown Circuit Analysis Spurious Actuations. During our review, we had the benefit of discussions with representatives of the NRC staff, the Nuclear Energy Institute (NEI), Duke Energy, and Progress Energy. We also had the benefit of the documents referenced.

RECOMMENDATION

The Generic Letter 2006-XX: Post-Fire Safe-Shutdown Circuit Analysis Spurious Actuations should be issued after the scope of requested information is clarified and the submittal dates are made more realistic.

BACKGROUND

One of the consequences of the Browns Ferry fire in 1975 was a number of spurious actuations of equipment. The proper treatment of spurious actuations that could affect the ability of a nuclear power plant to safely shut down during a fire has been a long-standing source of differing opinion between the NRC staff and the nuclear industry. For many years, the industry contended that it was extremely unlikely that a cable fire would lead to multiple spurious actuations. They argued that it should only be necessary to consider one spurious actuation for a particular cable fire or that, if multiple actuations occurred, they would be spaced sufficiently in time to permit each actuation to be mitigated separately.

In 2001, cable fire tests performed by Electric Power Research Institute (EPRI)/NEI indicated not only that multiple spurious actuations are likely to occur but also that the time between actuations may be insufficient to allow the mitigation of each actuation separately.

If a licensee has not accounted for multiple spurious actuations in its circuits analysis, it may not be in compliance with 10 CFR 50.48 and 10 CFR Part 50, Appendix A, General Design Criterion 3, which require that a licensee provide and maintain free of fire damage one train of systems necessary to achieve and maintain safe shutdown. The intent of the GL is to obtain the information needed to ensure that licensees have adequately addressed the potential for spurious actuations that compromise the capability for safe shut down.

The GL requests that each licensee:

- Within 90 days, submit a description of the plant's licensing basis with respect to the regulatory requirement for protecting redundant safe shutdown trains from multiple simultaneous spurious actuations and maintaining one train free of fire damage and submit a conclusion regarding the compliance of the plant.

  a. If not in compliance, submit a functionality assessment of systems, structures, and components (SSCs) that affect ability to achieve and maintain safe shutdown.

  b. If not in compliance, submit a description of compensatory measures put in place.

- Within 6 months, submit a plan to return all affected SSCs to compliance with regulatory requirements.

Within 30 days of issuance of the GL, the licensee can submit a request for additional time.

DISCUSSION

There are three likely approaches that the licensee will take to bring its plant into compliance:

- Make the modifications necessary to ensure safe shutdown regardless of fire location and with multiple simultaneous spurious actuations.

- Use a risk-informed approach based on Regulatory Guide 1.174 to justify exemptions or license amendments in accordance with 10 CFR 50.12 or 10 CFR 50.90.

- Adopt a performance-based fire protection program in accordance with 10 CFR 50.48, National Fire Protection Association Standard (NFPA) 805.

Among the principal comments by the industry regarding the draft GL are that it: establishes a new regulatory position; does not allow risk-informed methods (as in NEI 00-01) to be used by licensees that are not adopting NFPA 805; and imposes an unreasonable schedule for providing information.

With regard to the question whether the GL establishes a new regulatory position, the NRC's Committee to Review Generic Requirements reviewed this issue and stated that it had no objection to issuing this GL. Consequently, we did not pursue this issue further.

The request for information within 90 days regarding the extent of compliance from licensees with the regulatory intent described in the GL is reasonable. However, it is unreasonable to expect the licensees to perform the requested analyses of multiple spurious actuations within that time period, as would be necessary to assess the functionality of SSCs and to identify appropriate compensatory measures. We agree with the staff's objective to bring the licensees into compliance with regulatory requirements expeditiously. However, we recognize the magnitude of the effort required and the potential benefit of additional experiments that will be performed over the next six months. The staff has agreed to more clearly define the scope of the information that is to be provided at each deadline and to extend the time by which affected SSCs are identified and compensatory measures are reported.

Many licensees will address multiple spurious actuations by adopting a performance-based fire protection program (NFPA 805). For licensees that do not adopt the performance-based approach, a large number of exemption requests and license modifications may be required. Some combinations of spurious actuations, although conceivable, would have an extremely low frequency of occurrence. In their response to public comments, the staff indicated that the industry should develop screening tools to eliminate low-frequency combinations. In NEI 00-01, Rev. 1, "Guidance for Post-Fire Protection for Existing Light-Water Nuclear Power Plants," NEI proposes such an approach. Regulatory Issue Summary 2004-003 was developed to provide a risk-informed approach to inspections to focus on risk-significant configurations. Similar guidance could be developed as an aid to the exemption or amendment process.

The staff has agreed to clarify the scope of information to be provided at each milestone in the schedule and to provide additional time for the functionality assessment of affected SSCs. The GL should be issued after making these changes.

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References:

1. Memorandum dated May 10, 2006, from James E. Lyons, Office of Nuclear Reactor Regulation to John T. Larkins, Advisory Committee on Reactor Safeguards, transmitting for final ACRS review of Draft NRC Generic Letter 2006-XX: Post-Fire Safe-Shutdown Circuit Analysis Spurious Actuations, and the Staff's Resolution of public comments.
2. NRC Regulatory Issue Summary 2004-03: Risk-informed Approach for Post-fire Safe-Shutdown Associated Circuit Inspections.
3. NRC Regulatory Issue Summary 2005-30: Clarification of Post-fire Safe-shutdown Circuit Regulatory Requirements.
4. Title 10 Code of Federal Regulations, 50.48 "Fire Protection".
5. U.S. Nuclear Regulatory Commission Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant Specific Changes to the Licensing Basis," July 1998.
6. Title 10 Code of Federal Regulations, 50.12 "Specific Exemptions."
7. Title 10 Code of Federal Regulations, 50.90 "Application for Amendment of License or Construction Permit."
8. NFPA 805 "Performance-Based Standard for Fire Protection for Light-Water Reactor Generating Plants."
9. NEI 00-01 "Guidance for Post-Fire Protection for Existing Light-Water Nuclear Power Plants."

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

June 21, 2006

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: RESULTS OF THE STAFF'S INITIAL SCREENING OF GENERIC ISSUE-197, "IODINE SPIKING PHENOMENA"

During the 533rd meeting of the Advisory Committee on Reactor Safeguards, May 31-June 1, 2006, the Committee considered the results of staff's initial screening of Generic Issue-197, "Iodine Spiking Phenomena." The Committee has no objection to the staff's proposal to drop this issue from further consideration.

Reference:
Memorandum dated June 16, 2006, from Mark A. Cunningham, Director, Division of Fuel, Engineering and Radiological Research, to John T. Larkins, Executive Director, ACRS, Subject: Forwarding the Summary from the Review Panel for Generic Issue-197, "Iodine Spiking Phenomena."

cc: A. Vietti-Cook, SECY
W. Dean, OEDO
J. Lamb, OEDO
B. Sosa, OEDO
B. Sheron, RES
M. Cunningham, RES
J. Uhle, RES
A. Lee, RES
R. Assa, RES
H. Vandermolen, RES
R. Emrit, RES
M. Lee, NRR

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

July 14, 2006

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: PROPOSED REVISION TO STANDARD REVIEW PLAN NUREG-0800, SECTION 3.9.4, "CONTROL ROD DRIVE SYSTEMS"

During the 534th meeting of the Advisory Committee on Reactor Safeguards, July 12-13, 2006, the Committee considered the proposed revision to the Standard Review Plan NUREG-0800, Section 3.9.4, "Control Rod Drive Systems," and decided not to review this revision. The Committee has no objection to the staff's proposal to issue this document.

Reference:
Memorandum dated June 20, 2006, from John A. Grobe, Director, Division of Component Integrity, Office of Nuclear Reactor Regulation, to John T. Larkins, Executive Director, ACRS, Subject: Transmittal of Proposed Revision to Standard Review Plan NUREG-0800, Section 3.9.4, "Control Rod Drive Systems."

cc: A. Vietti-Cook, SECY
W. Dean, OEDO
B. Sosa, OEDO
J. Grobe, NRR
W. Bateman, NRR
T. Liu, NRR
K. Poertner, NRR
S. Lee, NRR
R. Assa, RES

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

July 14, 2006

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: DRAFT FINAL RULE PACKAGE TO AMEND 10 CFR 50.68, "CRITICALITY ACCIDENT REQUIREMENT"

During the 534th meeting of the Advisory Committee on Reactor Safeguards, July 12-13, 2006, the Committee considered the draft final rule package to amend 10 CFR 50.68, "Criticality Accident Requirements." The Committee decided that it would like to review this draft rule with the staff. The next opportunity for such a discussion would occur during the Committee's meeting on September 7-9, 2006.

Reference:
Memorandum dated July 12, 2006, from Ho K. Nieh, Acting Director, Division of Policy and Rulemaking, Office of Nuclear Reactor Regulation, to John T. Larkins, Executive Director, ACRS, Subject: Draft Final Rule Package to Amend 10 CFR 50.68, "Criticality Accident Requirements."

cc: A. Vietti-Cook, SECY
W. Dean, OEDO
J. Dyer, NRR
H. Nieh, NRR
S. Lee, NRR
G. Tartal, NRR
B. Sosa, OEDO

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

August 1, 2006

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

SUBJECT: GENERIC SAFETY ISSUE 191 - ASSESSMENT OF DEBRIS ACCUMULATION ON PWR SUMP PERFORMANCE

Dear Mr. Reyes:

On April 10, 2006, we issued a report to Chairman Diaz discussing the resolution of Generic Safety Issue (GSI) 191, "Assessment of Debris Accumulation on PWR Sump Performance." On May 2, 2006, you responded that it is the staff's intent to terminate research activities related to GSI-191 in June 2006. You indicate that additional work by the industry and the staff may be needed to address some remaining issues such as chemical and downstream effects. The staff's current approach is to rely on large-scale integral tests of screens by the industry to demonstrate that the safety margin is sufficiently conservative to accommodate phenomenological uncertainties. Because of the complexity of the phenomena that affect the pressure drop across debris beds, particularly when chemical effects are included, the staff has concluded that the development of predictive models is a "challenging and long-term effort which may not achieve timely closure of GSI-191 issues."

The efforts that are being taken by the industry in response to Generic Letter 2004-02 to substantially increase screen size are appropriate. We also agree that the industry's integral experiments will help to support the safety case. However, it is important to recognize the limitations of these tests.

Historically, integral tests have been used to validate predictive analytical tools. These tools are used to evaluate the performance of safety systems. Integral tests have not been used as "proof tests" as an alternative to analytical tools because of the difficulty of achieving conditions that are truly prototypic. In addition, it is not practical to examine system behavior experimentally over the full range of variability of input conditions. The planned tests of full-size screen modules will be performed using conditions that vary substantially from prototypic, including differences in water temperature, water chemistry, pre-conditioning of insulation debris, and the actual system configuration, such as multiple modules. In order to understand the impact of these experimental non-typicalities, it is necessary to have some level of quantitative understanding of the phenomena. The staff must have the capability to perform an independent technical assessment of the approaches used by licensees to address GSI-191 issues.

During a meeting on June 13-14, 2006, our Thermal-Hydraulic Phenomena Subcommittee reviewed the status of the NRC's sump performance research program. Substantial progress has been made in a number of areas. Progress on developing a predictive tool for debris bed pressure drop without chemical effects is very promising but further work is required.

Experiments have been performed that indicate that chemical effects can be substantial. However, to date, the staff has not interpreted the experimental results from these tests within the context of a mechanistic model or even correlated them empirically. The staff only recently initiated calculations to assess potential downstream effects, particularly related to in-vessel flow blockages. These are examples of areas in which additional research is still warranted.

A continued regulatory research program to address key areas of uncertainty is a risk-management strategy for reducing the likelihood of erroneous regulatory conclusions. We recommend that confirmatory research on GSI-191 be continued.

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

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References:

1. Report dated April 10, 2006 from Graham B. Wallis, Chairman, Advisory Committee on Reactor Safeguards, to Nils J. Diaz, Chairman, Nuclear Regulatory Commission, Subject: Generic Safety Issue 191 - Assessment of Debris Accumulation on PWR Sump Performance.

2. Memorandum dated May 2, 2006 from Luis Reyes, Nuclear Regulatory Commission, for Graham B. Wallis, Chairman, Advisory Committee on Reactor Safeguards, Subject: Generic Safety Issue 191 - Assessment of Debris Accumulation on PWR Sump Performance.

3. U.S. Nuclear Regulatory Commission Generic Letter 2004-02: "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors," September 13, 2004.

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

August 2, 2006

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

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE LICENSE RENEWAL APPLICATION FOR THE NINE MILE POINT NUCLEAR STATION, UNITS 1 AND 2

Dear Chairman Klein:

During the 534th meeting of the Advisory Committee on Reactor Safeguards, July 12–13, 2006, we completed our review of the license renewal application for the Nine Mile Point Nuclear Station (NMPNS), 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 April 5, 2006. During these reviews, we had the benefit of discussions with representatives of the staff and the applicant, Constellation Energy Group, LLC (CEG). We also had the benefit of the documents referenced. This report fulfills the requirement 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 NMPNS, 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. CEG's application for renewal of the operating licenses for NMPNS, Units 1 and 2, should be approved.

BACKGROUND AND DISCUSSION

NMPNS consists of two General Electric (GE) boiling water reactor (BWR) Units on a site six miles northeast of Oswego, NY. The current operating licenses will expire on August 22, 2009 for Unit 1 and October 31, 2026 for Unit 2. The applicant has requested renewal of these licenses for an additional 20 years.

Unit 1 uses a Mark 1 containment design consisting of a drywell, a suppression chamber in the shape of a torus, and a vent system that connects the drywell to the torus. Unit 2 uses a Mark 2 containment structure of reinforced concrete with an inner liner of carbon steel. Unit 1 is authorized to operate at 1,850 MWt, and Unit 2 at 3,467 MWt. The Unit 1 main condenser is cooled by a once-through circulating water system using cooling water from Lake Ontario. Unit 2 has a closed cooling system that uses a natural draft cooling tower.

In the final SER, the staff documented its review of the license renewal application and other information submitted by the applicant or obtained during the staff's audit and inspection 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 structures and components; the adequacy of the applicant's Aging Management Programs (AMPs); and the identification and assessment of time-limited aging analyses (TLAAs).

The application demonstrates consistency with, or justifies deviations from, the approaches specified in the Generic Aging Lessons Learned (GALL) Report. The applicant has correctly identified those SSCs from both Units that fall within the scope of license renewal. The applicant performed an aging management review of SSCs within the license renewal scope. Based on the results of this review, the applicant will apply 43 AMPs. Of these, 9 are fully consistent with the GALL Report, 27 are consistent with the GALL Report with exceptions or enhancements, and 7 are plant specific. The staff determined that the AMPs described by the applicant are appropriate and sufficient to manage aging of long-lived passive components that are within the scope of license renewal. We concur.

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

Analyses of neutron embrittlement of the reactor vessels for both units were performed by the applicant and independently verified by the staff. These analyses demonstrate that the limiting reactor vessel beltline welds and plate materials will satisfy acceptance criteria for the periods of extended operation. Both the applicant and the staff chose to use a lifetime capacity factor of 90 percent for determining neutron fluence.

The staff identified no open items or confirmatory items in the final SER. CEG has made 56 license renewal commitments for NMPNS. Of these commitments, 26 are common to both Units with 16 commitments applying only to Unit 1 and 14 commitments applying only to Unit 2. The staff has included appropriate license conditions in the final SER to satisfy the remaining documentation issues and action items. No changes in the technical specifications are required for either Unit.

The applicant's initial license renewal application was not of adequate quality. In reviewing the application, the staff generated 323 Requests for Additional Information (RAIs) and 385 audit questions. The large number of RAIs prompted the applicant to request a delay to prepare an amended license renewal application. The amended license renewal application was more complete and of higher quality.

The staff's evaluation was comprehensive and well documented in the final SER. The inspection and audit performed by the staff were effective in evaluating the applicant's proposed and existing programs and TLAAs.

No issues related to the matters described in 10 CFR 54.29(a)(1) and (a)(2) preclude renewal of the operating licenses for NMPNS, Units 1 and 2. The programs committed to and established by the applicant provide reasonable assurance that NMPNS, 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 NMPNS, Units 1 and 2, should be approved.

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References:

1. Safety Evaluation Report-Final Related to the License Renewal of Nine Mile Point Nuclear Station, Units 1 and 2, dated May 30, 2006.

2. Nine Mile Point Nuclear Station, Units 1 and 2 — Application for Renewed Operating Licenses, dated May 26, 2004.

3. Audit and Review Report for Plant Aging Management Programs (AMPs) and Aging Management Reviews (AMRs) — Nine Mile Point Nuclear Station, dated January 5, 2006.

4. Nine Mile Point Nuclear Station Inspection Report 05000220/20050011 and 05000410/20050011, dated March 2, 2006.

5. Safety Evaluation Report with Open Items Related to the License Renewal of the Nine Mile Point Nuclear Station, Units 1 and 2, dated March 3, 2006.

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

August 2, 2006

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

SUBJECT: DRAFT NUREG REPORT, "INTEGRATING RISK AND SAFETY MARGINS"

Dear Mr. Reyes:

During the 534th meeting of the Advisory Committee on Reactor Safeguards, July 12-13, 2006, we met with representatives of the NRC staff to discuss the draft NUREG report "Integrating Risk and Safety Margins." We also had the benefit of the documents referenced.

RECOMMENDATIONS

1. This work could have substantial regulatory benefits by providing an approach to quantify changes in safety margins and defense in depth. It should be pursued in the context of the technology-neutral framework and for future revisions of Regulatory Guide (RG) 1.174.

2. The draft NUREG report is preliminary and exploratory and needs to be substantially revised before it is published to make its purposes, concepts, and conclusions clearer.

DISCUSSION

The licensing bases for nuclear power plants are currently established through deterministic analyses. These analyses show that for design basis accidents, a variety of safety limits (reactor coolant system pressure, containment pressure and temperature, peak cladding temperature, 10 CFR Part 100 doses, etc.) are met. Although these limits must be met in all design basis accidents, they will, in fact, be exceeded with some likelihood if a wider range of possible event sequences is considered. If a plant undergoes a modification, such as a power uprate, these limits must still be met in all design basis accidents, but such modifications can reduce margins. The likelihood of event sequences in which these limits will be exceeded increases after the power uprate.

In risk-informed amendments for changes to the licensing basis prepared using the guidance in RG 1.174, the effects for a broad range of event sequences are addressed in terms of core damage frequency (CDF) and large, early release frequency (LERF). Acceptance guidance for ΔCDF and ΔLERF is provided. As part of the integrated decisionmaking process, the decisionmaker is directed to consider whether the proposed changes maintain sufficient safety margins. However, RG 1.174 does not provide explicit guidance or a methodology for evaluating changes in safety margin.

The draft report is intended to provide a framework to address changes in safety margins. The basic approach is to consider a broad range of event scenarios such as is now done in a probabilistic risk assessment to assess CDF and to determine the frequency with which any safety limits of interest are exceeded after the plant changes. This frequency can then be compared to the comparable frequency of exceedance before the changes were introduced. This comparison could provide a measure of the impact of the change.

In addition to providing an approach to quantifying changes in safety margins, such an approach could provide a way to quantify defense in depth by considering the changes in the failure frequency of individual barriers such as the cladding, the reactor coolant system, and containment independent of whether they, in fact, lead to core damage or large releases of radioactive material. It could also be used to quantify the effect of plant changes on other NRC objectives, such as limiting the frequency of small releases, in a more comprehensive and realistic manner than is currently done through analysis of design basis events.

It is premature to judge whether the approach described to us by the staff can be successful and whether it could be implemented by a reasonable extension of current design basis analyses and probabilistic risk assessments or would require significant additional analysis. The selection of specific safety limits for such analysis will require careful consideration. However, the general approach appears to be worthwhile exploring both in the context of new approaches to regulation such as the technology-neutral framework and for future revisions of RG 1.174. This work should be continued. In the near term, it should focus on the potential for its use as part of the integrated decisionmaking process in RG 1.174 to quantify changes in risk and defense in depth. Specific examples of applications should be developed in order to assess the value of the approach. We would like to continue to hear from the staff about further developments.

The draft NUREG report reflects the preliminary, exploratory nature of the work and needs to be substantially revised before it is published to make its purposes, concepts, and conclusions clearer. During the meeting, members provided detailed comments to the staff for consideration in revising the report.

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References:

1. Memorandum from Farouk Eltawila, Director, Division of Risk Analysis and Special Projects, Office of Nuclear Regulatory Research, to John T. Larkins, Executive Director, Advisory Committee on Reactor Safeguards, Subject: ACRS Review of Draft NUREG for Framework Integrating Risk and Safety Margins, dated June 20, 2006.

2. U.S. Nuclear Regulatory Commission Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 1, dated November 2002.

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

September 13, 2006

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

Michael Snodderly for

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

SUBJECT: QUESTIONS RAISED BY MEMBERS OF THE PUBLIC DURING THE ACRS SUBCOMMITTEE MEETING ON PALISADES NUCLEAR PLANT LICENSE RENEWAL APPLICATION

During the July 11, 2006 ACRS Subcommittee meeting on Plant License Renewal that was held to review the license renewal application for the Palisades Nuclear Power Plant, members of the public raised several questions. These questions can be found in the transcript of the meeting (ADAMS Accession No. ML062080468). Since most, if not all, of these questions do not deal with license renewal issues, the Committee brings this matter to your attention for disposition.

cc: A. Vetti-Cook SECY
M. Johnson, OEDO
B. Sosa, OEDO
J. Lamb, OEDO
F. Gillespie, NRR
L. Lund, NRR
L. Padovan, NRR
J. Ayala, NRR
D. Collins, NRR
S. (Min) Lee, NRR

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

September 13, 2006

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: PROPOSED REVISION TO STANDARD REVIEW PLAN, NUREG-0800, SECTION 6.1.1, "ENGINEERED SAFETY FEATURES MATERIALS"

During the 535th meeting of the Advisory Committee on Reactor Safeguards, September 7-8, 2006, the Committee considered the proposed revision to Standard Review Plan (SRP), NUREG-800, Section 6.1.1, "Engineered Safety Features Materials." The Committee decided not to review this document. The Committee has no objection to the staff's proposal to issue the revised SRP Section 6.1.1.

Reference:
Memorandum dated July 26, 2006, from John A. Grobe, Director, Division of Component Integrity, Office of Nuclear Reactor Regulation, to John T. Larkins, Executive Director, ACRS/ACNW, Subject: Transmittal of Proposed Revision to Standard Review Plan NUREG-0800, Section 6.1.1, "Engineered Safety Features Materials."

cc: A. Vietti-Cook, SECY
M. Johnson, OEDO
B. Sosa, OEDO
J. Grobe, NRR
A. Keim, NRR
S. (Min) Lee, NRR
S. Koenick, NRR
R. Assa, RES

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

September 13, 2006

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

Michael Snodderly for

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

SUBJECT: PROPOSED REVISION TO REGULATORY GUIDE 1.23 (DG-1164), "METEOROLOGICAL MONITORING PROGRAMS FOR NUCLEAR POWER PLANTS"

During the 535th meeting of the Advisory Committee on Reactor Safeguards, September 7-8, 2006, the Committee considered proposed revisions to Regulatory Guide 1.23 (DG-1164), "Meteorological Monitoring Programs for Nuclear Power Plants." The Committee decided not to review this regulatory guide and has no objection to the staff's proposal to issue it for public comment. The Committee would like to be informed of any significant changes made to this Guide prior to publishing it in its final form.

Reference:
Memorandum dated August 30, 2006, from Jimi T. Yerokun, Chief, Risk Applications and Special Project Branch, RES to John T. Larkins, Executive Director, ACRS, Subject: Additional Information - Regulatory Guide 1.23 (DG-1164), "Meteorological Monitoring Programs For Nuclear Power Plants"

cc: A. Vietti-Cook, SECY
M. Johnson, OEDO
B. Sosa, OEDO
J. Lamb, OEDO
B. Sheron, RES
J. Monninger, RES
J. Yerokun, RES
S. Koenick, NRR
R. Harvey, NRR
S. (Min) Lee, NRR
R. Assa, RES

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

September 13, 2006

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: DRAFT NUREG-1852, "DEMONSTRATING THE FEASIBILITY AND RELIABILITY OF OPERATOR MANUAL ACTIONS IN RESPONSE TO FIRE"

During the 535th meeting of the Advisory Committee on Reactor Safeguards, September 7-8, 2006, the Committee considered the draft NUREG-1852, "Demonstrating the Feasability and Reliability of Operator Manual Actions in Response to Fire." The Committee plans to review the draft final version of this report after reconciliation of public comments. The Committee has no objection to the staff's proposal to issue the draft report for public comment.

Reference:
Memorandum dated September 8, 2006, from Farouk Eltawila, Director, Division of Risk Assessment and Special Projects, Office of Nuclear Regulatory Research, to John T. Larkins, Executive Director, Advisory Committee on Reactor Safeguards, Subject: Draft NUREG-1852, "Demonstrating the Feasibility and Reliability of Operator Manual Actions in Response to Fire."

cc: A. Vietti-Cook, SECY
M Johnson, OEDO
B. Sosa, OEDO
J. Lamb, OEDO
F. Eltawila, RES
P. Baranowsky, RES
J. Monninger, RES
E. Lois, RES
K. Hill, RES
R. Assa, RES
S. (Min) Lee, NRR

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

September 13, 2006

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: DRAFT FINAL REVISION TO REGULATORY GUIDE 1.76, "DESIGN BASIS TORNADO FOR NUCLEAR POWER PLANTS"

During the 535th meeting of the Advisory Committee on Reactor Safeguards, September 7-8, 2006, the Committee considered the Draft Final Revision 1 to Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants," and decided not to review this Guide. The Committee has no objection to the staff's proposal to issue this document.

Reference:
Memorandum dated September 5, 2006, from Jimi T. Yerokun, Chief, Risk Applications and Special Project Branch, RES to John T. Larkins, Executive Director, ACRS, Subject: Additional Information - Regulatory Guide 1.76, "Design Basis Tornado For Nuclear Power Plants"

cc: A. Vietti-Cook, SECY
M. Johnson, OEDO
B. Sosa, OEDO
J. Lamb, OEDO
B. Sheron, RES
J. Monninger, RES
J. Yerokun, RES
S. Koenick, NRR
R. Harvey, NRR
S. (Min) Lee, NRR
R. Assa, RES

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

September 19, 2006

The Honorable Dale E. Klein
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 2005-0001

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE LICENSE RENEWAL APPLICATION FOR THE MONTICELLO NUCLEAR GENERATING PLANT

Dear Chairman Klein:

During the 535th meeting of the Advisory Committee on Reactor Safeguards, September 7-8, 2006, we completed our review of the license renewal application for the Monticello Nuclear Generating Plant (MNGP) 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 May 30, 2006. During our review, we had the benefit of discussions with representatives of the NRC staff and the applicant, 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.

CONCLUSION AND RECOMMENDATION

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

The NMC application for renewal of the operating license for MNGP should be approved.

BACKGROUND AND DISCUSSION

MNGP is a General Electric Boiling Water Reactor-3 (BWR-3) within a Mark-I containment. The current power rating of 1775 MWt includes a 6.3% power uprate that was implemented in 1998. NMC requested renewal of the MNGP operating license for 20 years beyond the current license term, which expires on September 8, 2010.

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

The NMC application is largely consistent with the Generic Aging Lessons Learned (GALL) Report. All deviations from the approaches specified in the GALL Report are documented in the application. The applicant identified the SSCs that fall within the scope of license renewal and performed a comprehensive aging management review for these SSCs. Based on the results of this review, the applicant will implement 36 AMPs for license renewal including existing, enhanced, and new programs. In the SER, the staff concluded that the applicant has appropriately identified the SSCs within the scope of license renewal and that the AMPs described by the applicant are appropriate and sufficient to manage aging of long-lived passive components that are within the scope of license renewal. We concur with this conclusion.

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

During our meetings with the staff and the applicant, we discussed the adequacy of programs proposed by NMC to manage aging of certain components that are a current focus of the staff and the industry, as described below.

Aging of the drywell shell of MNGP will be managed through the use of the ASME Section XI, Subsection IWE Program. We agree with this approach. Even though this Program does not include ultrasonic testing, this approach was chosen by NMC and accepted by the staff because the plant has several design features that prevent water accumulation behind the shell. During each refueling outage, water leakage is monitored from the refueling seal bellows, the drywell air gap drains, and the sand-pocket drains. The refueling seal is within the scope of license renewal. Ultrasonic inspections performed in the past did not identify any degradation.

MNGP has experienced shroud cracking. This cracking was identified through the required licensee inspection process. Periodic inspections of up to 75% of the shroud welds are performed according to the guidelines of the Boiling Water Reactor Vessel and Internals Project (BWRVIP). Previously identified flaws have exhibited no significant crack growth since the introduction of hydrogen water chemistry at MNGP. Aging of the shroud will continue to be managed by using the guidelines in the BWRVIP-76. We find this AMP appropriate.

The MNGP steam dryers are within the scope of license renewal. A 1998 inspection identified an indication that was not structurally significant. A 2001 inspection revealed no change in this indication and no additional indications were identified. A comprehensive inspection conducted in 2005 to examine areas where steam dryer failures had occurred at other plants found new indications on the dryer shell. These indications were evaluated and determined to be acceptable by the applicant. Another inspection is planned for 2007. Aging of the steam dryers will continue to be managed in accordance with the guidelines in the BWRVIP-139 program. We find this AMP appropriate.

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

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

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References:

1) Safety Evaluation Report Related to the License Renewal of the Monticello Nuclear Generating Plant, dated August 2, 2006.
2) Monticello Nuclear Generating Plant- Application for Renewed Operating License, dated March 16, 2005.
3) Audit and Review Report for Plant Aging Management Programs (AMPs) and Aging Management Reviews (AMRs) - Monticello Nuclear Generating Plant, dated October 12, 2005.
4) Monticello Nuclear Generating Plant, Inspection Report 05000263/2006006, dated March 30, 2006.
5) BWR Vessel and Internals Project, BWR Core Shroud Inspection and Flaw Evaluation Guidelines (BWRVIP-76), EPRI Report TR-114232, November 1999.
6) BWR Vessel and Internals Project, Steam Dryer Inspection and Flaw Evaluation Guidelines (BWRVIP-139), EPRI Report TR-1011463, April 2005.

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

September 21, 2006

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

SUBJECT: PROPOSED DIRECT FINAL RULE TO AMEND 10 CFR 50.68, "CRITICALITY ACCIDENT REQUIREMENTS"

Dear Mr. Reyes:

During the 535th meeting of the Advisory Committee on Reactor Safeguards, September 7-8, 2006, we reviewed the proposed direct final rule to amend 10 CFR 50.68, "Criticality Accident Requirements." During our review, we had the benefit of discussions with representatives of the NRC staff and the Nuclear Energy Institute, and the documents referenced.

RECOMMENDATION

1. The proposed direct final rule to amend 10 CFR 50.68 should be issued for public comment.

2. The NRC staff should complete the research to quantify the reactivity effects of fission products in the fuel. The results of this research may enable additional burnup credit to be allowed in the guidance for 10 CFR Part 71 and 72.

BACKGROUND AND DISCUSSION

The staff has proposed to amend 10 CFR 50.68, so that the requirements governing criticality control for spent fuel pool storage racks do not apply to the fuel within a spent fuel transportation package or storage cask when a package or cask is in a spent fuel pool. 10 CFR 50.68 currently requires that spent fuel pools remain subcritical in an unborated, maximum moderation condition. The implementation of this regulation also allows credit for the operating history of the fuel (burnup credit) when analyzing the storage configuration of the spent fuel.

10 CFR Parts 71 and 72 govern the use of spent fuel storage casks and transportation packages. 10 CFR Part 71 requires that transportation packages be designed assuming they can be flooded with fresh water (unborated), and thus, are already analyzed in a manner that complies with 10 CFR 50.68. 10 CFR Part 72 requires that dry storage casks be designed to be subcritical when stored dry, but may rely on soluble boron to avoid criticality when filled with water when the cask is in a spent fuel pool.

On March 23, 2005, the NRC issued Regulatory Issue Summary (RIS) 2005-05 addressing spent fuel criticality analyses for spent fuel pools under 10 CFR 50.68 and independent spent fuel storage installations under 10 CFR Part 72. In the Statement of Considerations for the proposed direct final rule the staff stated that, "The intent of the RIS was to advise reactor licensees that they must meet both the requirements of 10 CFR 50.68 and 10 CFR Part 72 with respect to subcriticality during storage cask loading in spent fuel pools. In order to satisfy both requirements, an additional site-specific analysis according to 10 CFR 50.68 is required. In this analysis, the licensee can take credit for fuel burnup to determine the margin to criticality for the specific cask loading.

The NRC staff has determined that the requirement to perform multiple analyses is an unnecessary burden for both industry and the agency. As a result, the staff proposes to modify 10 CFR 50.68 to eliminate the requirement for redundant criticality analyses of fuel in a cask in a spent fuel pool. Under the proposed rule, the criticality requirements of 10 CFR Parts 71 and 72 would apply to fuel in these casks in a spent fuel pool. For fuel in the pool but outside the cask, the criticality analyses requirements of 10 CFR 50.68 would apply.

We agree with the staff's proposed revision to 10 CFR 50.68. The proposed direct final rule should be issued for public comment.

The staff's justification for their position is a qualitative analysis that scenarios that could result in criticality are very unlikely. The arguments regarding the likelihood of these scenarios discussed in Appendix A to the rule package are persuasive but the presentation is confusing. The use of simple event trees to display the scenarios would have been very helpful and could be beneficial if included in the final rule package.

The NRC staff should also consider revising the guidance associated with 10 CFR Parts 71 and 72 to allow for fuel burnup credit, as is now permitted in the guidance for 10 CFR Part 50. The staff stated that this has not been done because the uncertainty in fission product reactivity effects is large, and has not been quantified. Industry and the Office of Nuclear Regulatory Research are cooperating on a program to obtain the data needed to reduce uncertainties. The results of this research may enable additional burnup credit to be allowed for dry cask storage.

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References:
See next page

<u>References</u>

1. Memorandum from Ho K. Nieh to John Larkins, Revised Draft Final Rule Package to Amend 10 CFR 50.68, "Criticality Accident Requirements," August 22, 2006, ML062330162.

2. Memorandum from Ho K. Nieh to John Larkins, Draft Final Rule Package to Amend 10 CFR 50.68, "Criticality Accident Requirements," July 12, 2006, ML061790301 (Predecisional).

3. Memorandum from John Larkins to Luis Reyes, Draft Final Rule Package to Amend 10 CFR 50.68, "Criticality Accident Requirements", July 14, 2006

4. Spent Fuel Project Office, Interim Staff Guidance - 8, Revision 2, "Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transport and Storage Casks", ML022700416

5. Memorandum from C. Withee to M. Hodges, "ISG-8, REV. 2 Supporting Document", September 27, 2002, ML022700393

6. "Technical Recommendations for the Criticality Safety Review of PWR Storage and Transportation Casks That Use Burnup Credit", C. Witheee and C. Parks, September 4, 2002, ML022700412

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

September 22, 2006

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

SUBJECT: LESSONS LEARNED FROM THE REVIEW OF EARLY SITE PERMIT APPLICATIONS

Dear Mr. Reyes:

During the 535th meeting of the Advisory Committee on Reactor Safeguards (ACRS), September 7-8, 2006, we met with representatives of the NRC staff; Dominion Nuclear North Anna, LLC; System Energy Resources, Inc.; and, Southern Nuclear Operating Company, Inc. to discuss any lessons that may have been learned in the submission, evaluation, and review of the North Anna, Grand Gulf, and Clinton early site permit applications. This matter was also discussed by our Subcommittee on Early Site Permits on September 6, 2006. We had the benefit of the documents referenced.

In accordance with 10 CFR Part 52, Subpart A, early site permit applications address separately safety and environmental issues. The ACRS is required to report on those portions of the applications that concern safety. We have reported separately on each of the applications for North Anna, Grand Gulf, and Clinton. Generally, we have praised both the quality of the applications and the quality of the staff safety evaluation reports on these applications.

Based on our review of the applications and discussions with representatives of the NRC staff and the applicants, two lessons emerged that may have generic applicability, especially to the many Combined License (COL) applications now anticipated by the agency. One lesson concerned the development of a "common understanding" between the staff and the applicant regarding expectations for the application. The second concerned the use of data obtained from the internet to substantiate portions of an application and safety analysis.

The applications we have reviewed have been the first opportunity to exercise the early site permit regulations. Not all the guidance that might be desired has been in place. Some available guidance was written for rules in place in a previous era. Applicants found it important to establish through direct discussions with the staff a common understanding of staff expectations concerning portions of the early site permit applications. Where this common understanding had been established, the preparation of the application and review process were generally smooth. Where a common understanding was not established, the processes often were more time consuming. Time spent by the staff to establish guidance and develop a common understanding with the applicants should facilitate processing of anticipated COL applications.

In the current electronic age, ever more information is becoming available through the internet. This trend will continue and eventually the internet may replace libraries and other information repositories that support engineering and safety analyses. Internet resources have advantages in comparison to familiar printed resources. They also have vulnerabilities that are not suffered by printed resources. Though internet information sources were conservatively and appropriately handled for the three early site permit applications we have reviewed, it is evident that eventually the staff will have to establish guidance to ensure reliability of internet information and the continuing ability to retrieve such information.

Two of the applicants made specific note of the challenges they faced in the electronic submission of their applications and continuing challenges they face in the electronic submission of updates to these applications. The NRC staff is addressing these challenges in anticipation of electronic submissions of COL applications.

In the course of reviews of the first three early site permit applications, the staff found that it had to discipline the review process by defining criteria for the imposition of permit conditions and COL action items. We have reviewed the criteria staff established and reported favorably on these criteria in our March 24, 2006, report. The applicant for an early site permit application for the Clinton site surprised the staff by invoking a novel, performance-based, seismic hazard analysis. This new methodology deviated markedly from the staff-approved seismic analysis methodology. The staff was able to examine and approve this methodology as it applied to the Clinton early site permit. Again, we reviewed the staff's analysis and reported favorably in our March 24, 2006 report. Nevertheless, the new approach to seismic hazard analysis did strain staff resources. Timely processing of future early site permit applications and COL applications will depend on advance dialog between the staff and the applicants when new analysis methodologies are to be introduced.

The staff has identified other lessons from the review of the first three early site permit applications and is acting upon these lessons. Among the lessons are the needs for:

- definition and criteria for pertinent site characteristics,
- criteria for the controlling elements of the plant parameter envelope,
- guidance on the treatment of the high frequency (10-100 Hz) component of seismic ground motion,
- guidance on the depth of review of major features of the emergency plan for a proposed new site, and
- criteria and review guidance for the computation of the probable maximum flood at a proposed site.

The priority that staff ascribes to addressing these lessons is influenced by its anticipation that future applicants will adopt specific reactor technologies and will not rely on the plant parameter envelope option permitted under the current regulations. The staff also anticipates that future applicants will provide fully integrated emergency plans and will not ask for approval of just specific major features of an emergency plan.

During the review of the early site permit applications, a number of questions arose concerning the applicability of 10 CFR Part 21 and 10 CFR Part 50, Appendix B to the early site permit process and holders of early site permits. The staff did conclude that processes for reporting deficiencies and quality control of activities are needed. The staff now proposes rule changes to make these elements of the regulations applicable to the early site permit process.

Among the characteristics of a proposed site considered in the early site permit process are extremes of weather. There is an evolving understanding of climatic cycles that affect extremes of weather especially for sites on the east coast of the United States and near the Gulf of Mexico. Though it cannot be claimed that the understanding is well established, it is evident that there are weather cycles with periods on the order of decades that can affect site characteristics. The popular press ensures that the public is aware of this growing understanding of weather cycles. This public awareness may make it particularly important that the staff demonstrate some understanding of these processes and the likely effects of weather cycles on the suitability of proposed sites for nuclear power plants. The staff needs to ensure that historical weather data used to characterize a site extend over sufficient time intervals to capture cyclical extremes in the weather that will affect plant design.

In our meeting with the staff and applicants, a consensus developed that the experiences gained in the course of the early site permit process would aid considerably the preparation of applications for COLs at the sites. Applicants that have not been through the process will benefit from an effort to derive their own lessons to the extent they can from the review of these three early site permit applications. We anticipate that additional lessons will be learned should the staff undertake a review of an early site permit for a so-called "green field" site that is not adjacent to the site of a currently operating nuclear power plant.

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References:
See next page

References:

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

2. Letter dated March 11, 2005, from G. B. Wallis, Chairman, ACRS, to L. A. Reyes, Executive Director for Operations, NRC, Subject: Interim Letter: Draft Safety Evaluation Report on North Anna Early Site Permit Application.

3. Letter dated June 14, 2005, from G. B. Wallis, Chairman, ACRS, to L. A. Reyes, Executive Director for Operations, NRC, Subject: Interim Letter: Draft Safety Evaluation Report on Grand Gulf Early Site Permit Application.

4. Report dated July 18, 2005, from G. B. Wallis, Chairman, ACRS, to N. J. Diaz, Chairman, NRC, Subject: Dominion Nuclear North Anna, LLC, Early Site Permit Application and the Associated NRC Final Safety Evaluation Report.

5. Letter dated September 22, 2005, from G. B. Wallis, Chairman, ACRS, to L. A. Reyes, Executive Director for Operations, NRC, Subject: Interim Letter: Exelon Generation Company, LLC, Application for Early Site Permit and the Associated NRC Staff's Draft Safety Evaluation Report.

6. Letter dated December 23, 2005, from G. B. Wallis, Chairman, ACRS, to L. A. Reyes, Executive Director for Operations, NRC, Subject: Early Site Permit Application for the Grand Gulf Site and the Associated Final Safety Evaluation Report.

7. Report dated March 24, 2006, from G. B. Wallis, Chairman, ACRS, to N. J. Diaz, Chairman, NRC, Subject: Final Review of the Exelon Generation Company, LLC, Application for an Early Site Permit and the Associated NRC Staff's Final Safety Evaluation Report.

8. Report dated May 22, 2006, from G. B. Wallis, Chairman, ACRS, to N. J. Diaz, Chairman, NRC, Subject: Proposed Revisions to 10 CFR Part 52: Licenses, Certifications, and Approvals for Nuclear Power Plants, and Conforming Amendments to Applicable NRC Regulations.

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

October 6, 2006

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

Michael Snodderly for John T. Larkins

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

SUBJECT: DRAFT FINAL REVISIONS TO 10 CFR PART 26, "FITNESS-FOR-DUTY PROGRAMS"

During the 536th meeting of the Advisory Committee on Reactor Safeguards, October 4-6, 2006, the Committee considered the draft final revisions to 10 CFR Part 26, "Fitness for Duty Programs," and decided not to review this rule. The Committee has no objection to the staff's proposal to issue the final rule.

References:
Memorandum dated October 5, 2006, from Michael Marshall, Jr., Acting Branch Chief, Division of Policy and Rulemaking, NRR to Dr. Graham B. Wallis, Chairman, ACRS, Subject: Advisory Committee on Reactor Safeguards Review of the Draft Final Rule to 10 CFR PART 26, "Fitness-for-Duty Program"

cc: A. Vietti-Cook, SECY
M. Johnson, OEDO
B. Sosa, OEDO
J. Lamb, OEDO
J. Dyer, NRR
H. Nieh, NRR
D. Diec, NRR
D. Desaulniers, NRR
E. Skarpac, NSIR
S. (Min) Lee, NRR

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

October 13, 2006

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: SUPPLEMENT 1 TO FINAL SAFETY EVALUATION REPORT FOR NORTH ANNA EARLY SITE PERMIT (ESP) APPLICATION

During the 536th meeting of the Advisory Committee on Reactor Safeguards, October 4-6, 2006, the Committee considered the changes reflected in Revisions 6, 7, 8, and 9 of Dominion Nuclear North Anna LLC (Dominion) application for an early site permit (ESP). In its revised application, Dominion proposed: (1) to change the once-through cooling system planned for Unit 3 in previous versions of the safety site analysis report (SSAR) to a closed-cycle system; (2) to increase the power levels for Units 3 and 4 to match the designed maximum power (4500 MWt) of a General Electric Economic and Simple Boiling-Water Reactor (ESBWR), one of the reactor designs included in the plant parameter envelope; and (3) to reduce the bounding value for tritium activity release (associated with the ACR-700 design), to ensure that the tritium concentration in liquid effluent releases is less than both the 10 CFR Part 20 limit and the limit set in the EPA drinking water standards. By letter dated September 29, 2006, the staff transmitted Supplement 1 to its final Safety Evaluation Report (SER), which addresses Revisions 6 through 9 of the North Anna ESP application, to the ACRS for possible review.

The Committee decided that the proposed changes do not affect its previous conclusions and recommendations with regard to issuing the ESP, and that additional review of this document prior to issuance is not necessary.

<u>References</u>:

1. Memorandum dated September 29, 2006, from David B. Matthews, Director, Division of New Reactor Licensing, NRR, to John T. Larkins, Executive Director, ACRS, Subject: Transmittal of Supplement 1 to Final Safety Evaluation Report for North Anna Early Site Permit (ESP) Application.
2. U.S. Nuclear Regulatory Commission, Final Safety Evaluation Report, "Safety Evaluation Report for an Early Site Permit (ESP) at the North Anna ESP Site," dated September 2005 (NUREG-1835).

cc: A. Vietti-Cook, SECY
M. Johnson, OEDO
B. Sosa, OEDO
J. Lamb, OEDO
D. Matthews, NRR
T. Bergman, NRR
N. Patel, NRR
S. (Min) Lee, NRR

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

October 16, 2006

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: PROPOSED REVISIONS TO REGULATORY GUIDES IN SUPPORT OF NEW REACTOR LICENSING ACTIVITIES

During the 536th meeting of the Advisory Committee on Reactor Safeguards, October 4-6, 2006, the Committee considered proposed revisions to the following Regulatory Guides and decided not to review them. The Committee has no objection to the staff's proposal to issue these Guides for public comment. The Committee would like to be informed of any significant changes made to these Guides prior to publishing them in final form.

- Proposed Revision 4 to Regulatory Guide 1.9, Application and Testing of Safety-Related Diesel Generators in Nuclear Power Plants (DG-1172)
- Proposed Revision 2 to Regulatory Guide 1.13, Spent Fuel Storage Facility Design Basis (DG-1162)
- Proposed Revision 3 to Regulatory Guide 1.20, Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing (DG-1163)
- Proposed Revision 4 to Regulatory Guide 1.26, Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants (DG-1152)
- Proposed Revision 4 to Regulatory Guide 1.29, Seismic Design Classification (DG-1156)
- Proposed Revision 1 to Regulatory Guide 1.37, Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants (DG-1165)
- Proposed Revision 1 to Regulatory Guide 1.57, Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components (DG-1158)
- Proposed Revision 3 to Regulatory Guide 1.68, Initial Test Programs for Water-Cooled Nuclear Power Plants (DG-1166)

- Proposed Revision 1 to Regulatory Guide 1.71, Welder Qualification for Areas of Limited Accessibility (DG-1167)
- Proposed Revision 1 to Regulatory Guide 1.93, Availability of Electric Power Sources (DG-1153)
- Proposed Revision 2 to Regulatory Guide 1.124, Service Limits and Loading Combinations for Class 1 Linear-Type Component Supports (DG-1168)
- Proposed Revision 2 to Regulatory Guide 1.128, Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants (DG-1154)
- Proposed Revision 2 to Regulatory Guide 1.129, Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Nuclear Power Plants (DG-1155)
- Proposed Revision 2 to Regulatory Guide 1.130, Service Limits and Loading Combinations for Class 1 Plate-and-Shell-Type Component Supports (DG-1169)
- Proposed Revision 1 to Regulatory Guide 1.196, Control Room Habitability at Light-Water Nuclear Power Reactors (DG-1171)
- Draft Regulatory Guide DG-1142, Guidelines for Environmental Qualification of Safety-Related Computer-Based Instrumentation and Control Systems in Nuclear Power Plants

The Committee referred Proposed Revision 1 to Regulatory Guide 1.112, Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents From Light-Water-Cooled Nuclear Power Reactors (DG-1160), and Regulatory Guide 4.15, Quality Assurance for Radiological Monitoring Programs (Inception Through Normal Operations to License Termination) - Effluent Streams and the Environment (DG-4010), to the Advisory Committee on Nuclear Waste for possible review.

<u>Reference</u>:

1. Memorandum dated September 27, 2006, from Jimi T. Yerokun, Chief, Risk Applications and Special Project Branch, RES to Michael R. Snodderly, Chief, Technical Support Branch, ACRS, Subject: Information - Regulatory Guide 1.9, "Application and Testing of Safety-Related Diesel Generators in Nuclear Power Stations" (DG-1172).
2. Memorandum dated September 19, 2006, from Jimi T. Yerokun, Chief, Risk Applications and Special Project Branch, RES to Michael R. Snodderly, Chief, Technical Support Branch, ACRS, Subject: Additional Information - Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis" (DG-1162), 1.61, "Damping values for Seismic Design of Nuclear Power Plants" (DG-1157), 1.124, "Service Limits and Loading Combinations for Class 1 Linear-Type Component Supports" (DG-1168), and 1.130, "Service Limits and Loading Combinations for Class 1 Plate-and-Shell-Type Component Supports" (DG-1169).

3. Memorandum dated September 29, 2006, from Jimi T. Yerokun, Chief, Risk Applications and Special Project Branch, RES to Michael R. Snodderly, Chief, Technical Support Branch, ACRS, Subject: Additional Information - Regulatory Guide 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing" (DG-1163).
4. Memorandum dated September 19, 2006, from Jimi T. Yerokun, Chief, Risk Applications and Special Project Branch, RES to Michael R. Snodderly, Chief, Technical Support Branch, ACRS, Subject: Additional Information - Regulatory Guide 1.26 (DG-1152), "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants."
5. Memorandum dated September 29, 2006, from Jimi T. Yerokun, Chief, Risk Applications and Special Project Branch, RES to Michael R. Snodderly, Chief, Technical Support Branch, ACRS, Subject: Additional Information - Regulatory Guide 1.29, "Seismic Design Classification" (DG-1156).
6. Memorandum dated September 20, 2006, from Jimi T. Yerokun, Chief, Risk Applications and Special Project Branch, RES to Michael R. Snodderly, Chief, Technical Support Branch, ACRS, Subject: Additional Information - Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants" (DG-1165).
7. Memorandum dated September 29, 2006, from Jimi T. Yerokun, Chief, Risk Applications and Special Project Branch, RES to Michael R. Snodderly, Chief, Technical Support Branch, ACRS, Subject: Additional Information - Regulatory Guide 1.57, "Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components" (DG-1158).
8. Memorandum dated September 28, 2006, from Jimi T. Yerokun, Chief, Risk Applications and Special Project Branch, RES to Michael R. Snodderly, Chief, Technical Support Branch, ACRS, Subject: Additional Information - Regulatory Guide 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants" (DG-1166).
9. Memorandum dated September 29, 2006, from Jimi T. Yerokun, Chief, Risk Applications and Special Project Branch, RES to Michael R. Snodderly, Chief, Technical Support Branch, ACRS, Subject: Additional Information - Regulatory Guide 1.71, " Welder Qualification for Areas of Limited Accessibility" (DG-1167).
10. Memorandum dated September 22, 2006, from Jimi T. Yerokun, Chief, Risk Applications and Special Project Branch, RES to Michael R. Snodderly, Chief, Technical Support Branch, ACRS, Subject: Additional Information - Regulatory Guide 1.93, "Availability of Electric Power Sources" (DG-1153).
11. Memorandum dated September 25, 2006, from Jimi T. Yerokun, Chief, Risk Applications and Special Project Branch, RES to Michael R. Snodderly, Chief, Technical Support Branch, ACRS, Subject: Additional Information - Regulatory Guide 1.112, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents From Light-Water-Cooled Power Reactors" (DG-1160).
12. Memorandum dated September 7, 2006, from Jimi T. Yerokun, Chief, Risk Applications and Special Project Branch, RES to Michael R. Snodderly, Chief, Technical Support Branch, ACRS, Subject: Additional Information - Regulatory Guide 1.128, "Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants" (DG-1154).

13. Memorandum dated September 7, 2006, from Jimi T. Yerokun, Chief, Risk Applications and Special Project Branch, RES to Michael R. Snodderly, Chief, Technical Support Branch, ACRS, Subject: Additional Information - Regulatory Guide 1.129, "Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Nuclear Power Plants" (DG-1155).
14. Memorandum dated September 6, 2006, from Jimi T. Yerokun, Chief, Risk Applications and Special Project Branch, RES to Michael R. Snodderly, Chief, Technical Support Branch, ACRS, Subject: Additional Information - Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," and 1.196, "Control Room Habitability at Light-Water Nuclear Power Reactors."
15. Memorandum dated September 28, 2006, from Jimi T. Yerokun, Chief, Risk Applications and Special Project Branch, RES to Michael R. Snodderly, Chief, Technical Support Branch, ACRS, Subject: Additional Information - Draft Regulatory Guide 1142, "Guidelines for Environmental Qualification of Safety-Related Computer-Based Instrumentation and Control Systems in Nuclear Power Plants."
16. Memorandum dated September 22, 2006, from Jimi T. Yerokun, Chief, Risk Applications and Special Project Branch, RES to Michael R. Snodderly, Chief, Technical Support Branch, ACRS, Subject: Additional Information - Regulatory Guide 4.15, "Quality Assurance for Radiological Monitoring Programs (Inception Through Normal Operations to License Termination) - Effluent Streams and the Environment" (DG-4010).

cc:
A. Vietti-Cook, SECY
M. Johnson, OEDO
J. Lamb, OEDO
B. Sosa, OEDO
B. Sheron, RES
J. Yerokun, RES
S. Koenick, NRR
S. (Min) Lee, NRR
R. Assa, RES
S. Aggarwal, RES
H. Gonzalez, RES
J. Hixon, RES
A. Istar, RES
S. Jones, NRR
P. Kang, RES
H. Karagiannis, RES
W. Norris, RES
R. Pettis, NRR
J. Rajan, NRR
S. Shaukat, RES
A. V. du Bouchet, RES
J. Ridgely, RES

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

October 17, 2006

Dr. Brian Sheron
Director
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington DC 20555-0001

SUBJECT: ACRS ASSESSMENT OF THE QUALITY OF SELECTED NRC RESEARCH PROJECTS- FY 2006

Dear Dr. Sheron:

Enclosed is our report on the quality assessment of the following research projects:

- Melt Coolability and Concrete Interaction (MCCI) Program at the Argonne National Laboratory
  - This project was found to be satisfactory. The results meet the research objectives.
- Containment Integrity Research at Sandia National Laboratories
  - This project was found to be more than satisfactory. The results meet the research objectives.

These projects were selected from a list of candidate projects suggested by the Office of Nuclear Regulatory research.

We anticipate receiving your list of candidate projects for ACRS prior to our December 2006 Full Committee meeting.

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

Enclosure: As stated

Assessment of the Quality of Selected NRC Research Projects by the Advisory Committee on Reactor Safeguards for FY 2006

October 2006

U.S. Nuclear Regulatory Commission
Advisory Committee on Reactor Safeguards
Washington, DC 20555-0001

ABOUT THE ACRS

The Advisory Committee on Reactor Safeguards (ACRS) was established as a statutory Committee of the Atomic Energy Commission (AEC) by a 1957 amendment to the *Atomic Energy Act* of 1954. The functions of the Committee are described in Sections 29 and 182b of the Act. The *Energy Reorganization Act* of 1974 transferred the AEC's licensing functions to the U.S. Nuclear Regulatory Commission (NRC), and the Committee has continued serving the same advisory role to the NRC.

The ACRS provides independent reviews of, and advice on, the safety of proposed or existing NRC-licensed reactor facilities and the adequacy of proposed safety standards. The ACRS reviews power reactor and fuel cycle facility license applications for which the NRC is responsible, as well as the safety-significant NRC regulations and guidance related to these facilities. On its own initiative, the ACRS may review certain generic matters or safety-significant nuclear facility items. The Committee also advises the Commission on safety-significant policy issues, and performs other duties as the Commission may request. Upon request from the U.S. Department of Energy (DOE), the ACRS provides advice on U.S. Naval reactor designs and hazards associated with the DOE's nuclear activities and facilities. In addition, upon request, the ACRS provides technical advice to the Defense Nuclear Facilities Safety Board.

ACRS operations are governed by the *Federal Advisory Committee Act* (FACA), which is implemented through NRC regulations at Title 10, Part 7, of the *Code of Federal Regulations* (10 CFR Part 7). ACRS operational practices encourage the public, industry, State and local governments, and other stakeholders to express their views on regulatory matters.

MEMBERS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

Dr. Said Abdel-Khalik, Southern Nuclear Distinguished Professor, George W. Woodruff School of Mechanical Engineering, Georgia Institute of Technology, Atlanta, Georgia

Dr. Joseph S. Armijo, Adjunct Professor of Materials Science and Engineering at the University of Nevada, Reno

Dr. George E. Apostolakis, Professor of Nuclear Science and Engineering, Professor of Engineering Systems, Massachusetts Institute of Technology, Cambridge, Massachusetts

Dr. Sanjoy Banerjee, Professor of Chemical Engineering, University of California at Santa Barbara, Santa Barbara, California

Dr. Mario V. Bonaca, Retired Director, Nuclear Engineering Department, Northeast Utilities, Connecticut

Dr. Michael Corradini, Professor and Chairman of Department of Engineering Physics, University of Wisconsin, Madison, Wisconsin

Dr. Thomas S. Kress, Retired Head of Applied Systems Technology Section, Oak Ridge National Laboratory, Oak Ridge, Tennessee

Mr. Otto L. Maynard, Retired Chief Executive Officer, Wolf Creek Generating Station, Kansas

Dr. Dana A. Powers, Senior Scientist, Sandia National Laboratories, Albuquerque, New Mexico

Dr. William J. Shack, (Vice-Chairman), Associate Director, Energy Technology Division, Argonne National Laboratory, Argonne, Illinois

Mr. John D. Sieber, (Member-at-Large), Retired Senior Vice-President, Nuclear Power Division, Duquesne Light Company, Pittsburgh, Pennsylvania

Dr. Graham B. Wallis, (Chairman), Sherman Fairchild Professor Emeritus, Thayer School of Engineering, Dartmouth College, Hanover, New Hampshire

ABSTRACT

In this report, the Advisory Committee on Reactor Safeguards (ACRS) presents the results of its assessment of the quality of selected research projects sponsored by the Office of Nuclear Regulatory Research(RES) of the NRC. An analytic/deliberative methodology was adopted by the Committee to guide its review of research projects. The methods of multi-attribute utility theory were utilized to structure the objectives of the review and develop numerical scales for rating the project with respect to each objective. The results of the evaluations of the quality of the two research projects are summarized as follows:

- Melt Coolability and Concrete Interaction (MCCI) Program at the Argonne National Laboratory

  - This project was found to be satisfactory . The results meet the research objectives.

- Containment Integrity Research at Sandia National Laboratories

  - This project was found to be more than satisfactory. The results meet the research objectives.

CONTENTS

FIGURES

TABLES

ABBREVIATIONS

ACRS	Advisory Committee on Reactor Safeguards
AEC	Atomic Energy Commission
ANL	Argonne National Laboratory
BWR	Boiling Water Reactor
CCFL	Counter Current Flooding Limit
CCI	Core Concrete Interactions
CFR	Code of Federal Regulations
CHF	Critical Heat Flux
DOE	Department of Energy
FACA	Federal Advisory Committee Act
FY	Fiscal Year
GPRA	Government Performance and Results Act
LOCA	Loss-of-Coolant Accident
MAUT	Multi-Attribute Utility Theory
MACE	Melt Attack and Coolability Experiments
MCCI	Melt Coolability and Concrete Interaction
MET	Melt Eruption Test
NRC	Nuclear Regulatory Commission
OECD	Organization for Economic Cooperation and Developmen
PRA	Probabilistic Risk assessment
PWR	Pressurized Water Reactor
RES	Office of Nuclear Regulatory Research
SSWICS	SMALL Scale Water Ingression and Crust Strength

1 INTRODUCTION

The Nuclear Regulatory Commission (NRC) maintains a safety research program to ensure that the agency's regulations have sound technical bases. The research effort is needed to support regulatory activities and agency initiatives while maintaining an infrastructure of expertise, facilities, analytical tools, and data to support regulatory decisions.

The Office of Nuclear Regulatory Research (RES) is required to have an independent evaluation of the effectiveness (quality) and utility of its research programs. This evaluation is required by the NRC Strategic Plan that was developed as mandated by the Government Performance and Results Act (GPRA). Since fiscal year 2004, the Advisory Committee on Reactor Safeguards (ACRS) has been assisting RES by performing independent assessments of the quality of selected research projects [1,2]. The Committee has established the following process for conducting the review of the quality of research projects:

- RES submits to the ACRS a list of candidate research projects for review because they have reached sufficient maturity that meaningful technical review can be conducted.
- The ACRS selects no more than four projects for detailed review during the fiscal year.
- A panel of three to four 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, as needed, and provides them to the cognizant research manager, as appropriate.
- The Committee submits an annual summary report to the RES Director.

An analytic/deliberative decisionmaking framework was adopted for evaluating the quality of NRC research projects. The definition of quality research adopted by the Committee includes two major characteristics:

- Results meet the objectives
- The results and methods are adequately documented

Within the first characteristic, ACRS considered the following general attributes in evaluating the NRC research projects:

- Soundness of technical approach and results

- Has execution of the work used available expertise in appropriate disciplines?

- Justification of major assumptions
  - Have assumptions key to the technical approach and the results been tested or otherwise justified?
- Treatment of uncertainties/sensitivities
  - Have significant uncertainties been characterized?
  - Have important sensitivities been identified?

Within the general category of documentation, the projects were evaluated in terms of following measures:

- Clarity of presentation
- Identification of major assumptions

In this report, the ACRS presents the results of its assessment of the quality of the research projects associated with:

- Melt Coolability and Concrete Interaction (MCCI) Program at the Argonne National Laboratory
- Containment Integrity Research at Sandia National Laboratories

These two projects were selected from a list of candidate projects suggested by RES.

The methodology for developing the quantitative metrics (numerical grades) for evaluating the quality of NRC research projects is presented in Section 2 of this report. The results of assessment and ratings for the selected projects are discussed in Section 3.

2 METHODOLOGY FOR EVALUATING THE QUALITY OF RESEARCH PROJECTS

To guide its review of research projects, the ACRS has adopted an analytic/deliberative methodology [Ref. 3 and 4]. The analytical part utilizes methods of multi-attribute utility theory (MAUT) [Ref. 5 and 6] to structure the objectives of the review and develop numerical scales for rating the project with respect to each objective. The objectives were developed in a hierarchical manner (in the form of a "value tree"), and weights reflecting their relative importance were developed. The value tree and the relative weights developed by the full Committee are shown in Figure 1.

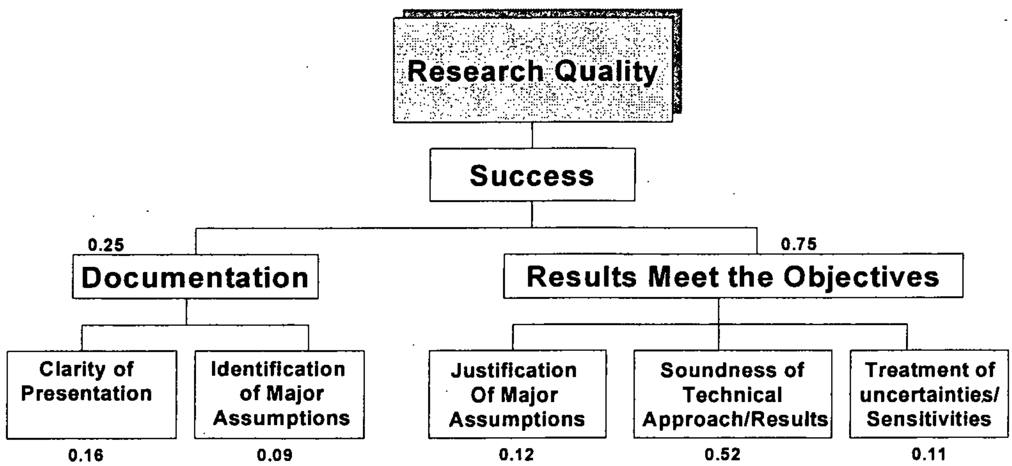


Figure 1 The value tree used for evaluating the quality of research projects

The quality of projects is evaluated in terms of the degree to which the results meet the objectives of the research and of the adequacy of the documentation of the research. It is the consensus of the ACRS that meeting the objectives of the research should have a weight of 0.75 in the overall evaluation of the research project. Adequacy of the documentation was assigned a weight of 0.25. Within these two broad categories, research projects were evaluated in terms of subsidiary "performance measures":

- justification of major assumptions (weight: 0.12)
- soundness of the technical approach and reliability of results (weight: 0.52)
- treatment of uncertainties and characterization of sensitivities (weight: 0.11)

Documentation of the research was evaluated in terms of the following performance measures:

- clarity of presentation (weight: 0.16)
- identification of major assumptions (weight: 0.09)

To evaluate how well the research project performed with respect to each performance measure, constructed scales were developed as shown in Table 1. The starting point is a rating of 5, Satisfactory (professional work that satisfies the research objectives). Often in evaluations of this nature, a grade that is less than excellent is interpreted as pejorative. In this ACRS evaluation, a grade of 5 should be interpreted literally as satisfactory. Although innovation and excellent work are to be encouraged, the ACRS realizes that time and cost place constraints on innovation. Furthermore, research projects are constrained by the work scope that has been agreed upon. The score was, then, increased or decreased according to the attributes shown in the table. The overall score of the project was produced by multiplying each score by the corresponding weight of the performance measure and adding all the weighted scores.

The value tree, weights, and constructed scales were the result of extensive deliberations of the whole ACRS. As discussed in Section 1, a panel of three ACRS members was formed to review each selected research project. Each member of the review panel independently evaluated the project in terms of the performance measures shown in the value tree. The panel deliberated the assigned scores and developed a consensus score, which was not necessarily the arithmetic average of individual scores. The panel's consensus score was discussed by the full Committee and adjusted in response to ACRS members' comments. The final consensus scores were multiplied by the appropriate weights, the weighted scores of all the categories were summed, and an overall score for the project was produced. A set of comments justifying the ratings was also produced.

Table 1. Constructed Scales for the Performance Measures

SCORE	RANKING	INTERPRETATION
10	Outstanding	Creative and uniformly excellent
8	Excellent	Important elements of innovation or insight
5	Satisfactory	Professional work that satisfies research objectives
3	Marginal	Some deficiencies identified; marginally satisfies research objectives
0	Unacceptable	Results do not satisfy the objectives or are not reliable

3. RESULTS OF QUALITY ASSESSMENT

3.1 MELT COOLABILITY AND CONCRETE INTERACTION PROGRAM AT THE ARGONNE NATIONAL LABORATORY

The Melt Coolability and Concrete Interaction (MCCI) research was conducted at the Argonne National Laboratory (ANL). The research was part of an international collaborative effort under the auspices of the Organization for Economic Cooperation and Development (OECD). Thirteen OECD countries including the United States participated in the research program. As the host organization, NRC coordinated the program with the following objectives:

1. Provide both confirmatory evidence and test data on coolability mechanisms identified in previous integral effect tests and resolve the ex-vessel coolability issue through an understanding of the synergistic effects of these coolability mechanisms and through development of analytical models.

2. Address remaining uncertainties related to long-term two-dimensional melt-concrete interaction under dry as well as flooded cavity conditions.

The MCCI experimental efforts built upon previous OECD sponsored (MACE) integral effect tests program that attempted to define conditions under which water was able to quench core debris interacting with concrete. This previous effort identified some mechanisms of debris cooling that had not previously been recognized.

The MCCI project consisted of three experimental efforts:

- Small-scale Water Ingression and Crust Strength Tests
- Melt Eruption Tests.
- Core Concrete Interaction Tests

The Small Scale Water Ingression and Crust Strength (SSWICS) tests were intended to measure the ability of water to cool the molten core material by mechanisms other than conduction limited heat transfer, and to measure the strength of the crust formed during flooding of the melt.

The Melt Eruption Test (MET) was intended to measure the influence of gas sparging on melt entrainment and cooling and to determine the effect of melt ejection on the core-concrete interaction.

The Core Concrete Interaction (CCI) tests were intended to resolve uncertainties in axial versus lateral power splits and respective concrete ablation rates. The tests were intended to replicate as closely as possible conditions at plant scale and contribute data to verify and validate predictive codes. These tests were augmented by flooding with water after partial ablation to obtain debris coolability at later stages in the accident process.

The MCCI project was completed in December 2005 and an OECD final report [7] was issued in February 2006.

GENERAL OBSERVATIONS

The project has met several of its goals. It has successfully demonstrated several valuable test techniques to simulate the complex phenomena that occur during molten core concrete interactions. It has explored phenomena of interest and provided data consistent with scoping level tests. The project however has been too ambitious in its scope (or claims), and has failed to work with fully prototypic materials. The study was more exploratory than confirmatory. Analysis and deductions are somewhat weak, tenuous, and may be wrong in some aspects. Though some qualitative understanding has been achieved and possible theoretical approaches have been developed, the work is far from having established reliable predictive tools for resolving any of the issues that led to the proposed research.

The consensus scores for this project are shown in Table 2. The score for the overall assessment of the work was found to be 5.22 which should be interpreted as "a professional job that satisfies the research objectives." The Committee identified areas for improvement in all of the evaluation categories. Comments and conclusions within the evaluation categories are:

Documentation

- Clarity of presentation **(Consensus score = 6.5)**

  The Committee is generally pleased with the documentation which is challenging for a long-term, multifaceted effort such as the MCCI project.

  The writing and descriptions are generally clear. Many observations were made. Some seem inconsistent (e.g. pictures and data plots do not confirm the text). Some necessary details and dimensions are missing. The text generally presents a reasonable story of an ambitious undertaking that was partially successful.

  The report is generally well written and understandable. However it was difficult to find the actual composition of the particular "thermite" mixture used to produce the molten core material in the tests. The information was found in supporting documents. The compositions of the various melts used in the tests were provided, although some (Table 3-4) were mislabeled as 'thermite compositions".

Table 2. Summary Results of ACRS Assessment of the Quality of the Project on Melt Coolability and Concrete Interaction (MCCI)

Performance Measures	Consensus Scores	Weights	Weighted Scores
Clarity of presentation	6.5	0.16	1.04
Identification of major assumptions	4.75	0.09	0.43
Justification of major assumptions	4.0	0.12	0.48
Soundness of technical approach/results	5.5	0.52	2.86
Treatment of uncertainties/sensitivities	3.75	0.11	0.41
Overall Score:			**5.22**

The authors are overly familiar with the results derived from the SSWICS tests and do not present these results as clearly as possible. However, they do a poor job explaining legends of figures such as the legends of Figure 2-2 and 2-4. It is especially difficult to ascertain the meaning of "F-integrated" in the legend of Figure 2-4. The pair-wise comparisons of test results discussed on page 11 and following would have been far more effective if figures of corresponding pair-wise results had been provided rather than the jumbles of multiple results found in Figures 2-9 and 2-10. Scatter and variations in the plotted results make it quite challenging to understand what the authors mean by cooling plateaus they discuss at length in the text. Indeed, on page 15 the authors acknowledge that the identification of a plateau is subjective. Why, then, don't they share explicitly with the reader their identifications of the plateaus so the reader can judge for himself what has drawn the authors' interests and attentions?

The authors do not provide the readers with any parametric quantities used for the evaluation of models they compare to data or parametric values used for the plant scale comparisons. This places an enormous burden on the reader to independently assess the asserted models and the experimental data. In some cases, the reader is quite challenged to do this. For instance, in the discussion of radiation heat transfer (p. 52) the authors invoke the core debris melting temperature as a point value even though they previously went to pains to note that core debris freezes over a very large temperature range (Figure 2-13 and associated text). What value did the authors in fact use for the radiation heat transfer calculations? Similar conceptual questions in addition to simple material properties questions arise in connection with many other correlations.

- Identification of major assumptions **(Consensus score = 4.75)**

Assumptions appear in the way in which physical models were hypothesized, developed, and accepted. For example, CCFL was claimed to limit heat transfer in a porous crust, gas was assumed to create eruptions and entrain particles, gas bubbles were assumed to prevent crust formation, the crust was modeled as being broken by its own weight and the weight of water above it without being supported from below. Though these assumptions were identified and described, they were not critically examined.

No explicit statement of assumptions for the tests was provided. The assumptions could only be inferred from the tests themselves, and the subsequent analyses of results. If major assumptions had been stated and reviewed by the parties planning and authorizing the tests, perhaps they would have been modified to provide prototypic test melts and base mats.

Results Meet Objectives

- Justification of major assumptions **(Consensus score = 4.0)**

No explicit justification of assumptions for any of the tests was provided. It is not clear how so many test parameters could have been varied in so few tests without justification. This suggests that the choice of variables was rather informal and consistent with exploratory testing and equipment checkout.

Assumptions are justified as being deduced from empirical evidence, mechanistic models and previous work. Critical examination of them by clear comparisons with data and evidence is weak. Equations are written down based on word descriptions which are sometimes vague and would benefit from a clearer, more rigorous approach, using sketches and control volumes. Radiation is invoked as a mechanism of heat transfer but is not evaluated numerically, so it is unclear what role it plays. CHF and CCFL are invoked rather loosely without being fully described or evaluated analytically, so it is not clear exactly how they were used and how well their appropriateness was validated.

- Soundness of technical approach and results **(Consensus score = 5.5)**

Design and conduct of the tests were quite good and represent an engineering achievement. The system "worked" (except for the MET test), allowing observations to be made and data obtained. Theoretical and "engineering modeling" conclusions are not so strong. On balance this is a reasonable piece of exploratory investigation, but it does not really meet the work-scope objectives, which seem overly ambitious. No mechanism nor theory is developed for the actual process of erosion of the concrete (it is hypothesized that the rate is proportional to heat flux, as if this were a phase change reaction, but this mechanism is not confirmed and the heat flux to the concrete is not predicted).

The approach to the simulation of real molten core concrete interaction phenomena was sound, but attempted to answer too many questions with too few experiments. Further, neither the most likely composition of the real molten cores, nor the structure and composition of real concrete base mats were simulated. Consequently the applicability of the test results to real events is limited. There is no reason why the approach taken cannot be improved by better simulation of core melts and base mats and by a more disciplined approach in the definition of each future test, the limitation of variables in each test, the performance of duplicate tests, and improvements in redundancy and reliability of instrumentation.

The absence of pre-test predictions for the various tests is troubling. Certainly the models exist, as well as some data. It would have been expected some pre-test analyses to help pinpoint the parameters with greatest uncertainty and to focus on the primary objectives of each experiment.

The absence of a thorough ceramographic examination of the solidified crust to understand the structure and composition (on a microscopic scale) of this highly heterogeneous material was a significant shortcoming in the approach. Gross chemical composition measurements were made indicating some variability across the solidified melt, but this level of analysis provides little if any knowledge of its physical properties. In the absence of a detailed understanding of the microstructure of the solidified corium, analytical models will have to rely on crude approximations of the property data required for predictive models.

The SSWICS tests were undertaken to evaluate the rate of core debris cooling by an overlying water pool and to obtain samples of solidified core debris for strength measurements. The "core debris" simulant used in the tests was laced with various fractions of concrete to simulate the effects of some period of concrete ablation prior to exposing the molten core debris to water. The tests were, however, conducted in an apparatus composed of cast zirconia and magnesia. The tests did not involve active attack on the concrete and the vigorous melt stirring and sparging that accompanies such attack. The assumption that the concrete attack would not affect cooling or crust formation cries out for justification but none is provided. The further, implicit, assumption that attention can focus on the oxide phase of core debris[1] and not consider the voluminous metallic phase that would be present in core debris that had penetrated a reactor vessel also calls for some justification and none is provided

One of the objectives of the SSWICS tests was to obtain solidified core debris samples for strength measurements. There is, of course, an implicit assumption that the mechanical properties of the solidified materials produced in so unprototypical a way are somehow similar to what would be expected of core debris. Ceramic materials are notorious for having mechanical properties that are quite sensitive to the details of microstructure. Indeed, the authors find that crack structure has more a bearing on strength than composition, but they do not explain why they think the crack structures of their samples are indicative of the crack structures of solidified core debris. Certainly, the challenges faced by those removing solidified core debris (which, of course, had zero concrete content but was quenched by water) from the Three Mile Island vessel suggest that real core debris may be much stronger than suggested by the test results for the samples from the SSWICS tests.

More troubling about SSWICS is the implicit assumption that a room temperature strength measurement is somehow useful in the prediction of the strength of a solid with a thermal gradient that goes from the saturation point of water where the solid should be brittle to the melting point and a zone where the crust will be quite plastic. No explanation is provided on why the authors think that a crack will propagate from the cool regions through this plastic zone which might be quite thick. Indeed, the authors in section 6.0 seem to feel a brittle failure model is appropriate even though they acknowledge the underside of the melt will be very close to the melting point of the core debris.

In the authors' defense, some of the material that is the basis of their technical approach for the SSWICS tests is to be found in ancillary documentation so well referenced in the report. Examination of this material does not resolve the issues raised here. The question is so central to the thesis of the document that it deserves exposition.

Results obtained by the authors show that the concrete content of the core debris affects the crust strength and fracture behavior. The exposition would have been greatly enhanced if the authors had demonstrated based on accident analyses that they were working in a relevant and meaningful range. Discussion below concerning the initial transient when core debris first contacts concrete suggests that they are not.

---

[1] The tests involved some small fraction of chromium metal in the core debris ~6 to 8 weight percent which is much less than the fraction of metal usually expected in ex-vessel core debris.

Overall, the soundness of the technical approach for the SSWICS tests is arguable, but not demonstratively flawed.

The MET failed because of inadequate tests apparatus design. Indeed, experience has shown it unlikely that the test would have met its objectives had it been possible to retain the melt within the experimental cavity. The tests were to examine melt entrainment by gas sparging. Gas was supplied not by the attack of the melt on concrete but through a porous plate at the base of the apparatus. Such porous plate designs seldom yield a uniform gas flux. The variable flow resistances near the wall cause preferential flow through regions of the melt. The investigators did no simulation tests to see if they could get uniform flow. Without a reasonably uniform flow across the diameter of the melt, entrainment results are difficult to interpret and nearly impossible to scale up to reactor dimensions. To get uniform flow through a porous plate, rather difficult variations in plate porosity must be engineered across the diameter.

The authors conclude, however, that the objectives of the MET were met by examining data from other experiments. They do this with no attention to uncertainty. They compare results to a model that has an uncertainty of (-25% to +50%) which is large enough. As noted below, it is remarkable that the authors were able to avoid comparing results to the widely used Kataoka-Ishii correlation.

The CCI tests were undertaken to ascertain the split between the horizontal heat flux to concrete from molten core debris and the downward heat flux to concrete. A rich literature on this issue developed very shortly after publication of WASH-1400 and it is unfortunate that the authors do not provide a precise of this literature involving both experiments and analyses. The technical approach adopted in the experimental effort minimizes the important effect of the metallic phase of core debris to the downward (and sideward) heat flux. The authors acknowledge this and even note their results "... may not be directly applicable to reactor accident sequences ..." They hope instead that the results may be useful for code validation but do not provide any evidence that this is the case such as might be derived by discussing the extent models of heat flux from core debris to concrete. It is consequently not evident at all why these tests were undertaken.

The CCI experiments were done in the direct electrical heating apparatus used for the MACE program. Core debris was used that contains some fraction of concrete as might be expected following the vigorous initial interaction with concrete by core debris containing some amount of metallic zirconium. The initial configuration of the core debris has not similarly been modified to reflect such an initial interaction and it is not at all evident why. Modern reactor analysis codes predict for many of the risk dominant accident sequences that core debris penetrating a reactor vessel will contain an important fraction of unoxidized zirconium metal. This is especially so for accidents at boiling water reactors that have a much higher initial core inventory of zirconium metal than do pressurized water reactors. Hot, metallic zirconium even when alloyed with very substantial quantities of steel structures from the reactor vessel is quite reactive. Heat liberated by the chemical reactions of metallic zirconium with the gaseous and condensed products of concrete decomposition is often predicted to raise the core debris temperature to quite high levels leading to more gas generation and more reaction with zirconium in the core debris[2]. There is the further radiation heat transfer to concrete not contacted by core debris. This concrete spalls and melts into the core debris. The geometry of the region occupied by core debris in an accident will be quite different than the regular geometry of the test and this will affect the heat flux partitioning between the horizontal and axial directions. The investigators could not simulate the initial vigorous interaction of core debris with concrete because of limitations of experimental methods. It is unclear why they did not address the geometric issues.

The CCI tests yielded results that indicate that the ratio of axial to radial ablation of concrete depend on concrete type. This, of course, has been known since the first tests of melt interactions with concrete were done in 1977. Most models attribute the differences to the higher gas production per unit of calcareous concrete ablated than gas production associated with ablation of siliceous concrete. Liquid concrete films at the interface also affect the heat transfer split. The authors draw attention to the core debris interface with concrete and note the differences in the interfaces for siliceous and calcareous concretes - differences that have also been known for 30 years. Siliceous concretes typically melt at lower temperatures to yield a more viscous product than do limestone concrete. Furthermore, the decomposition of calcium carbonate yield a decrepitated product. The heat transfer models, especially that developed by Bradley for the CORCON code, take these well known observations concerning the interface into account.

The report concludes with sections dealing with correlations of results and applications to reactor accidents. The titles of these sections are somewhat misleading. The authors do not really correlate their results. They assert models and compare model predictions to the results with scant attention to the uncertainty of the model predictions as a result of parameter uncertainty nor uncertainties in their experimental results. This technical approach has its merits, but it does not recognize that there are several models of core debris interactions with concrete being used today for accident analyses. It is remarkable that the authors elected not to compare their results to predictions of these models (CORCON, WECHSL, MAAP, etc.) Similarly, for plant applications, the authors did not analyze accidents. They set up stylized situations to examine how their correlations would relate to a larger scale. Again, more interesting would be to compare predictions of extant models modified to account for the new data to actual plant accidents. Do the new results change any of our current perceptions concerning accidents? Were this the first investigation of core debris interactions with concrete, the technical approach adopted for this work would be satisfactory. In fact, core debris

---

[2] The authors of the report seem not to recognize this prediction of modern accident analysis models as evidenced by their discussion of the initial transient interactions presented in section 5.1 concerning bulk cooling where they emphasize the temperature fall during the initial interaction.

interactions with concrete is a well-trodden field and there is a lot of both experimental and theoretical work that has been done that is not recognized in this documentation.

The technical approach does not defend the model selection. This is left to ancillary documentation. In many cases, this is quite acceptable. But, in other cases models of core debris interactions with concrete used in accident analyses are using different models. A comparison of the data to these models now in use would be most illuminating. For instance, it is not clear why the Kataoka-Ishii liquid entrainment correlation used in so many places is neglected in favor of the Rico-Spalding correlation.

- Treatment of uncertainties and characterization of sensitivities **(Consensus score = 3.75)**

There is no formal treatment of uncertainties. The conclusions are all qualitatively uncertain, since they are mostly descriptive. More efforts could have been usefully placed on giving the reader a crisper evaluation of uncertainty, based on the very small number of tests and many speculative effects. In particular, anyone wanting to use the theory and coefficients "C" and "E" would benefit from more direct warning about how uncertain they are, as well as the preliminary status of the equations in which they appear.

Obvious uncertainties were simply ignored in the analyses and conclusion. For example on pages 16 through 19 of the final report, data from the SSWICS tests were used to verify the Lister/Epstein dryout heat flux model. This required various thermal and mechanical property data for the solidified crust. There was no explicit discussion of the methods used to create the necessary data for such a heterogeneous material and consequently no treatment of uncertainties.

The report states that the crust mechanical property data were approximated using a volume-weighted method based on the properties of the individual constituents (uranium and zirconium oxides, chromium and concrete). However, there were no ceramographic examinations of the solidified melts identifying the microstructures and quantifying the compositions and amounts of the various phases present. Since the mechanical properties of a heterogeneous material are generally not controlled by a simple volumetric weighted average of constituents, this was an overly simplistic assumption.

As in all experiments there are uncertainties in the reliability of individual pieces of equipment and instruments used in the test system. These uncertainties are generally estimated by careful pre-test analyses, and ultimately confirmed by repeated testing of the entire test system with minimum variation of test parameters. The report does not indicate that any duplicate tests were performed in this project. Consequently it is not clear how much confidence one can have in the reproducibility of the results. An example is the behavior of the CCI-2 test in which the average melt temperature rose by approximately 100 C° after water addition (Figure 0-5). This was either an experimental error or a real phenomenon. A rationalization was provided in the report that suggested that the bulk melt temperature could have increased due to the quenching of the surface, formation of an insulating crust and loss of conductive heat transfer, but there were no redundant thermocouples available to support this supposition or resolve the question of experimental error. The fact that the phenomenon was not observed in the CCI-1 and CCI-3 tests leaves the issue open. This uncertainty undermines confidence in the temperature measurement throughout the CCI-2 experiment, and propagates through to any analysis that uses the data.

3.2 CONTAINMENT INTEGRITY RESEARCH AT SANDIA NATIONAL LABORATORIES

For nearly 30 years, significant research has been performed at Sandia National Laboratories (SNL), primarily under the sponsorship of the NRC, to improve the understanding of performance of nuclear power plant steel and concrete containment structures under severe accident pressure and temperature loads that exceed the design bases of containments. This work has consisted of experimental programs and analytical studies to investigate the response and capacity of containment structures for a wide variety of loading conditions with a primary emphasis on internal overpressurization. The report [8] selected for the present review and quality evaluation does not document a specific research effort, but summarizes the results obtained from all the earlier research activities and identifies common themes that have emerged. As stated in its foreword, the primary focus of the report is to comment on and tie the results of earlier experiments and analyses into current research. The scope also includes documenting the lesson learned during the containment model testing and analytical simulations, which are directly applicable to regulating and licensing the operation of the current fleet of nuclear power plants, as well as the design of new plants.

GENERAL OBSERVATIONS

This report summarizes the results of increasingly large and complex tests of scale models of containment structures and sub-components conducted at SNL between 1983 and 2001.These tests were intended to model existing containments in operation in the US. For most of these tests, various international nuclear research agencies and designers, operators, universities and consultants were invited to participate in test planning, pre-test predictions and post-test analysis. These efforts were meant to improve the ability to predict containment performance up to and including failure. The report effectively describes the results of these evaluations. The report describes a number of lessons learned and provides detailed recommendations and cautions regarding analytical modeling techniques of containments and their subcomponents.

The tests described in the report took place over a period of 20 years, with containment models increasing over time in scale and level of detailed simulation of critical components (penetrations, hatches, stiffeners, etc.) Although one can infer from the text the main reasons for the increasing complexity of the tests, it would have been valuable to have in the report a more explicit discussion of how lessons learned from tests were utilized to design future tests, to reduce uncertainties and to improve simulation and applicability of results to full scale containments. But it is recognized that such level of detail may have been beyond the intent of this report.

This report does not explicitly address uncertainties and their treatment. There are occasional discussions about uncertainties inherent in containment testing and analysis and sensitivity to certain effects and parameters, but an explicit discussion of uncertainties is lacking. Twenty-five years of research on containment deserve some discussion of uncertainties, or why such discussion is not provided or cannot be provided. The absence of such a section is a major detractor to the value of this report.

Table 3 Summary Results of the ACRS Assessment of the Quality of the Project on Containment Integrity Research at Sandia National Laboratories

Performance Measures	Consensus Scores	Weights	Weighted Scores
Clarity of presentation	6.5	0.16	1.04
Identification of major assumptions	5.5	0.09	0.495
Justification of major assumptions	5.0	0.12	0.60
Soundness of technical approach/results	6.0	0.52	3.12
Treatment of uncertainties/sensitivities	4.5	0.11	0.495
		Overall Score:	**5.75**

The consensus scores for this project are shown in Table 3. The score for the overall assessment of the work was found to be 5.75 which is more than satisfactory. Comments and conclusions within the evaluation categories are:

Documentation

- Clarity of presentation **(Consensus score = 6.5)**

  The report is well written and understandable, results are well communicated and explained. The sheer volume of information provided is a challenge to clarity that was well met by the authors.

  The scope of the work that is being summarized is enormous. The accompanying detailed report on the reinforced concrete containment test gives an example of the extraordinary amount of detailed information available on these tests. The presentation overall is very clear and readable. However, at least in the draft that the Committee was reviewing there were some annoying editorial problems. Through most of section 3 (text page 24) the references to figures in the text differed from the actual figure number by 6. In section 4 the references were again consistent.

Sometimes the summary can be somewhat misleading. On page 137 and again on page A–11, the authors state "The element choices must be assessed by verifying that the solutions do not violate fundamental mechanics (for example, force equilibrium), ..." In a finite element solution, force equilibrium is never exactly satisfied (because finite element solutions are only approximate), and as the authors correctly point out on page A–20 "a lack of internal element force convergence is not necessarily a good measure of the quality of solution, and in fact, the philosophy of ignoring internal element force convergence (but still enforcing global external force convergence and displacement convergence) has led to many good pretest predictions of containment large scale tests for many years."

Sometimes the presentation is too complete. On page 115 there is a fairly extensive reviews of work done on models for leakage through cracked concrete. It is not until page A–9 that we find out that even for intact concrete there is an enough shrinkage cracks and other defects that for containments with liners (all U.S. containments), the leakage is controlled for all practical purposes by the leakage through the liner.

In the course of the testing, fundamental issues (e.g.: What constitutes "failure?") needed to be defined. Other compromises (e. g. the choice of testing medium, construction details of the test models, lack of temperature and dynamic impulse effects (detonation and deflagration) and construction details (penetrations, hatches, and joints) were discussed but not explicitly treated. Further, the effect of large displacements of the containment on other structures and systems was not treated.

- Identification of major assumptions **(Consensus score = 5.5)**

Major assumptions utilized in the reported tests and analytical simulations are generally identified, discussed and documented.

A number of explicit and implied assumptions arise in the research and these were identified satisfactorily. In some cases, the effect of these simplifications and assumptions were estimated. On other cases, these simplifications were identified but not further evaluated. Considering the scope and limitations of these combined projects, the authors of the report properly identified these issues.

The report is very good at pointing out the limitations and assumptions of the testing program and the capability to analyze containment behavior.

Results Meet Objectives

- Justification of major assumptions **(Consensus score = 5.0)**

  Assumptions are well discussed and justified throughout the report

  Although the major assumptions were discussed in some detail, often the effect of these assumptions upon the analytical and/or test results were not numerically estimated. Many of these assumptions were driven by the practicalities inherent in scale model testing of complex phenomena. Other assumptions were driven by the practicalities and limitations of the analytical modeling tools used. Overall, insights into the effects of these assumptions led to a testing and analytical program that is least likely to be distorted by the effect of these assumptions.

  The report provides good justifications for major assumptions and limitations (e.g., the decision to not include temperature effects in the model test program). There is also a good discussion of how the intended use of an analytical model affects the modeling assumptions that must be made.

- Soundness of technical approach and results **(Consensus score = 6.0)**

  In general, the technical approach was appropriate for this type of study. The program gave reasonable results consistent with intuitive and experiential expectations and produced results that confirm the regulatory expectations of the program.

  While our capability to analyze containment behavior is not yet complete, the work done in the programs at Sandia and in complementary industry programs have provided a wealth of data with which to benchmark analyses of containment behavior. In particular the development and benchmarking of materials models for the behavior of reinforced concrete have been important contributions to the capability to analyze containment behavior.

  The work on steel containments where the failure is controlled by tensile instabilities probably would have benefited from input from processing engineers. Structural engineers tend to focus on load capability. Simple elastic–plastic models work quite well to predict loads. However, in cases where one is interested in the detailed understanding of the ductility of the material, the work done by processing engineers, such as for example, those doing sheet forming would be relevant. It would probably still be impossible to know the geometries and the material properties well enough to compute local failure but such interactions would help better understand the role of triaxiality and strain hardening on the failures.

  The approach taken by the authors is generally appropriate to meet the stated objectives. However, a more explicit discussion of how lessons learned from tests were used to design future tests to reduce uncertainties and to improve simulation and applicability of results to full scale containments would have further enhanced the value of this report.

- Treatment of uncertainties and characterization of sensitivities **(Consensus score = 4.5)**

  This report does not explicitly address uncertainty in the results.

  Although there is no quantitative discussion of uncertainties, there is extensive discussion of the potential limitations of our capability to do containment analyses and what we can compute with relative small uncertainty (gross structural failure) and what we can only compute with large uncertainties (onset and amount of leakage).

  The uncertainties in the results were estimated. Due to the cost of this research and the substantial margin to failure that the results appear to demonstrate, there is probably not a need to define these uncertainties in a more rigorous manner. However, uncertainties that arise from the lack of treatment of temperature effects and dynamic (impulse loading) may be important.

4. REFERENCES

1. 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.
2. Letter Dated November 5, 2005, from William J. Shack, Acting Chairman, ACRS, to Carl J. Paperiello, Director, Office of Nuclear Regulatory research, NRC, Subject: ACRS Assessment of the Quality of Selected NRC Research Projects - FY 2005.
3. National Research Council, *Understanding Risk: Informing Decisions in a Democratic Society.* National Academy Press, Washington, DC, 1996.
4. Apostolakis, G.E., and Pickett, S.E., "Deliberation: Integrating Analytical Results into Environmental Decisions Involving Multiple Stakeholders," *Risk Analysis*, 18:621-634, 1998.
5. Clemen, R., *Making Hard Decisions*, 2nd Edition, Duxbury Press, Belmont, CA, 1995.
6. Keeney, R.L., and H. Raiffa, *Decisions with Multiple Objectives: Preferences and Value Tradeoffs*, Wiley, New York, 1976.
7. Farmer, M.T., S. Lomperski, D. J. Kilsdonk, and R.W. Aeschlimann, "MCCI Project," OECD/MCCI-2005-TR06, February 28, 2006.
8. Hessheimer, M. F. , and F. A. Dameron, " Containment Integrity Research at sandia National Laboratories," NUREG/CR-xxxx, Draft March 2006.

October 23, 2006

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

SUBJECT: DRAFT REVISION 1 TO REGULATORY GUIDE 1.200 (DG-1161), "AN APPROACH FOR DETERMINING THE TECHNICAL ADEQUACY OF PROBABILISTIC RISK ASSESSMENT RESULTS FOR RISK-INFORMED ACTIVITIES," AND SRP SECTION 19.1, "DETERMINING THE TECHNICAL ADEQUACY OF PROBABILISTIC RISK ASSESSMENT RESULTS FOR RISK-INFORMED ACTIVITIES"

Dear Mr. Reyes:

During the 536th meeting of the Advisory Committee on Reactor Safeguards, October 4-6, 2006, we met with representatives of the NRC staff to discuss draft Revision 1 to Regulatory Guide 1.200 (DG-1161), "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-informed Activities," and a draft revision to Standard Review Plan (SRP) Section 19.1, "Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-informed Activities." We also had the benefit of the documents referenced.

RECOMMENDATIONS

1. Revisions to Regulatory Guide 1.200 and SRP Section 19.1 should be issued for use after reconciliation of public comments.

2. A regulatory guide on how to perform sensitivity and uncertainty analyses should be developed.

DISCUSSION

Regulatory Guide 1.200 and the associated SRP Section 19.1 describe an acceptable approach for determining whether the quality of a probabilistic risk assessment (PRA) that is used to support regulatory decisionmaking is sufficient to provide confidence in the results. In addition to their use in the regulation of operating reactors, the revised documents will support new reactor licensing activities and are planned to be issued by March 2007. They are intended to reflect guidance provided by standard setting and nuclear industry organizations and to be consistent with Regulatory Guide 1.174.

We reviewed the original version of Regulatory Guide 1.200 in September 2003. That version of the Guide was issued for trial use in February 2004. Together with industry, the staff conducted five pilot applications and has incorporated lessons learned into Revision 1 of the Guide. The revised documents were posted for public comment in mid-September. Because the staff has already had numerous interactions with the public regarding this Guide, it does not expect many additional comments. We would like to be informed of any significant changes made to this Guide and the associated SRP Section as a result of public comments.

In addition to a number of wording and other minor changes, Regulatory Guide 1.200 has been revised to include explicit definitions of core damage frequency and large early release frequency; additional information for internal flood, internal fire, and external hazard technical elements; and additional clarification of the regulatory position regarding consensus PRA standards. The revised SRP Section now includes descriptions of historical events in addition to a number of clarifications consistent with Regulatory Guide 1.200. We agree with these changes. The revised Regulatory Guide 1.200 and associated SRP Section 19.1 should be issued for use after reconciliation of public comments.

In our report dated September 22, 2003, we stated that we agreed with the staff's position that it would be more appropriate to discuss methods for performing uncertainty and sensitivity analyses in a separate regulatory guide than to include such a discussion in Regulatory Guide 1.200. In its November 7, 2003 response to our report, the EDO stated that the staff expected to provide a draft of such a Guide for our review in early 2004. We continue to believe that there is a need for guidance on acceptable methods for performing uncertainty and sensitivity analyses. The staff should develop a regulatory guide to provide such guidance in a timely manner.

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References:

1. Memorandum from Jimi T. Yerokun, Chief, Risk Applications and Special Projects Branch, Division of Risk Assessment and Special Projects, Office of Nuclear Regulatory Research, to John T. Larkins, Executive Director, Advisory Committee on Reactor Safeguards, "Request for ACRS Review of Regulatory Guide 1.200 (DG-1161), 'An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities,' and SRP 19.1, 'Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities'," September 1, 2006.

2. Report from Mario V. Bonaca, Chairman, Advisory Committee on Reactor Safeguards, to Nils J. Diaz, Chairman, U.S. Nuclear Regulatory Commission, "Draft Final Regulatory Guide x.xxx, 'An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities' (formerly DG-1122)," September 22, 2003.

3. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," November 2002.

4. Letter from William D. Travers, Executive Director for Operations, U.S. Nuclear Regulatory Commission, to Mario V. Bonaca, Chairman, Advisory Committee on Reactor Safeguards, "Draft Final Regulatory Guide x.xxx, 'An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities' (Formerly DG-1122)," November 7, 2003.

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

October 25, 2006

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

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

Dear Mr. Reyes:

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

CONCLUSION AND RECOMMENDATIONS

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

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

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

BACKGROUND

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

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

The selected models are:

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

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

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

DISCUSSION OF THE NUREG REPORT

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

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

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

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

DISCUSSION OF THE USER'S GUIDE

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

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

CONCLUDING REMARKS

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

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

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References

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

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

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

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

5. *Verification and Validation of Selected Fire Models for Nuclear Power Plant Application EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities: Vol 5: Consolidated Fire Growth and Smoke Transport (CFAST),* U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research (RES), Rockville, MD, and Electric Power Research Institute (EPRI), Palo Alto, CA, NUREG-1824 and EPRI 1011999, August 2006.

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

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

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

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

November 3, 2006

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: PROPOSED REVISIONS TO REGULATORY GUIDES IN SUPPORT OF NEW REACTOR LICENSING

During the 537[th] meeting of the Advisory Committee on Reactor Safeguards, November 1-3, 2006, the Committee considered proposed revisions to the following Regulatory Guides and decided not to review them. The Committee has no objection to the staff's proposal to issue these Guides for public comment. The Committee would like to be informed of any significant changes made to these Guides prior to publishing them in final form:

- Proposed Revision 1 to Regulatory Guide 1.61, Damping Values for Seismic Design of Nuclear Power Plants (DG-1157)
- Proposed Revision 3 to Regulatory Guide 1.136, Materials, Construction, and Testing of Concrete Containments (DG-1159)
- Draft Regulatory Guide DG-1146, A Performance-Based Approach to Define the Safe Shutdown Earthquake Ground Motion

References:
1. Memorandum dated September 19, 2006, from Jimi T. Yerokun, Chief, Risk Applications and Special Project Branch, RES, to Michael R. Snodderly, Chief, Technical Support Branch, ACRS, Subject: Additional Information - Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis" (DG-1162), 1.61, "Damping Values for Seismic Design of Nuclear Power Plants" (DG-1157), 1.124, "Service Limits and Loading Combinations for Class 1 Linear-Type Component Supports" (DG-1168), and 1.130, "Service Limits and Loading Combinations for Class 1 Plate-and-Shell-Type Component Supports" (DG-1169).
2. Memorandum dated September 29, 2006, from Jimi T. Yerokun, Chief, Risk Applications and Special Project Branch, RES, to Michael R. Snodderly, Chief, Technical Support Branch, ACRS, Subject: Additional Information - Regulatory Guide 1.136, "Materials, Construction, and Testing of Concrete Containments" (DG-1159).
3. Memorandum dated September 29, 2006, from Jimi T. Yerokun, Chief, Risk Applications and Special Project Branch, RES, to Michael R. Snodderly, Chief, Technical Support Branch, ACRS, Subject: Additional Information - Draft Regulatory Guide 1146, "A Performance-Based Approach to Define the Safe Shutdown Earthquake Ground Motion."

cc: A. Vietti-Cook, SECY
M. Johnson, OEDO
J. Lamb, OEDO
B. Sosa, OEDO
B. Sheron, RES
J. Yerokun, RES
S. Koenick, NRR
S. (Min) Lee, NRR
R. Assa, RES
J. Ridgely, RES
H. Graves, RES
S. Shaukat, RES
A. Murphy, RES
B. Tegeler, RES

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

November 6, 2006

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: PROPOSED REVISIONS TO STANDARD REVIEW PLAN SECTIONS IN SUPPORT OF NEW REACTOR LICENSING

During the 537th meeting of the Advisory Committee on Reactor Safeguards, November 1-3, 2006, the Committee considered proposed revisions to the following Standard Review Plan (SRP), NUREG-0800, Sections and decided not to review them.. The Committee has no objection to the staff's proposal to issue these SRP Sections. The Committee would like to be informed of any significant changes made to these SRP Sections prior to publishing them in final form:

- Proposed Revision 2 to SRP Section 9.1.3, Spent Fuel Pool Cooling and Cleanup System
- Proposed Revision 3 to SRP Section 10.3.6, Steam and Feedwater System Materials
- Draft Final Revision to SRP Section 17.5, Quality Assurance Program Description - Design Certification, Early Site Permit and New License Applicants

References:

1. Memorandum dated August 25, 2006, from Thomas O. Martin, Director, Division of Safety Systems, NRR, to John T. Larkins, Executive Director, ACRS, Subject: Transmittal of Proposed Revision to Standard Review Plan NUREG-0800, Section 9.1.3, Revision 2, "Spent Fuel Pool Cooling and Cleanup System."
2. Memorandum dated August 24, 2006, from John A. Grobe, Director, Division of Component Integrity, NRR, to John T. Larkins, Executive Director, ACRS, Subject: Transmittal of Proposed Revision to Standard Review Plan NUREG-0800, Section 10.3.6, "Steam and Feedwater System Materials."
3. Memorandum dated September 22, 2006, from Michael E. Mayfield, Director, Division of Engineering, NRR, to John T. Larkins, Executive Director, ACRS, Subject: Transmittal of Proposed Final Revision to Standard Review Plan NUREG-0800, Section 17.5, "Quality Assurance Program Description - Design Certification, Early Site Permit and New License Applicants."

cc: A. Vietti-Cook, SECY
M. Johnson, OEDO
J. Lamb, OEDO
B. Sosa, OEDO
B. Sheron, RES
J. Yerokun, RES
S. Koenick, NRR
S. (Min) Lee, NRR
R. Assa, RES
J. Ridgely, RES
M. Comar, NRR
R. Foster, NRR
J. Nguyen, NRR
W. Held, NRR
K. Parczewski, NRR
S. Jones, NRR
W. Koo, NRR
P. Prescott, NRR

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

November 7, 2006

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: BROWNS FERRY NUCLEAR PLANT, UNIT 1 - EXTENDED POWER UPRATE APPLICATION AND SUPPLEMENTAL APPLICATION

On June 28, 2004, the Tennessee Valley Authority (TVA) submitted an amendment request to raise the thermal power of Browns Ferry Nuclear Plant (BFN) Unit 1 from 3293 MWt to 3952 MWt, an increase of approximately 20% in original licensed thermal power (OLTP) (Reference 1). This is commonly referred to as an extended power uprate (EPU). Because of concerns with steam dryer operation at the EPU level, TVA will need to gather data and perform analyses to support the staff's completion of the related Safety Evaluation Report (SER). This will delay completion of the SER until the data and analyses are provided to the staff. The ACRS plans to review this EPU application after the final SER is provided.

On September 22, 2006, TVA submitted an amendment supplement (Reference 2) requesting approval of an increase in licensed thermal power of approximately 5% above the OLTP. TVA stated that it will use the analyses performed at 120% OLTP to license operation at 105% OLTP, whenever the results of the analyses performed at 120% OLTP bound plant operation at 105% OLTP. In its amendment supplement TVA stated that after review and approval of the 105% OLTP power uprate, the transition to 120% OLTP will "only be contingent upon NRC review and acceptance of the steam dryer stress report. All other safety evaluations that support operation at 105% OLTP would remain valid for operation at 120% OLTP."

Normally, the ACRS does not review power uprates less than about 105% OLTP (Reference 3). But in the case of BFN Unit 1, the licensee will use bounding arguments to demonstrate that safety analyses performed at 120% OLTP are valid to support operation at 105% OLTP. In view of the licensee's intended approach, the ACRS has decided to review the SER for the 105% power uprate for BFN Unit 1.

References:

1. Letter dated June 28, 2004 from T. Abney to Document Control Desk, "Browns Ferry Nuclear Plant (BFN) - Unit 1 - Proposed Technical Specifications (TS) Change TS-431 - Request for License Amendment - Extended Power Uprate (EPU) Operation"

2. Letter dated September 22, 2006 from W. Crouch to Document Control Desk, "Browns Ferry Nuclear Plant (BFN) - Unit 1 - Technical Specifications (TS) Change TS-431, Supplement 1 - Extended Power Uprate (EPU) (TAC No. MC3812)"

3. Memorandum dated October 9, 2003, from John T. Larkins to James E. Dyer, "Kewaunee Nuclear Power Plant - Advisory Committee on Reactor Safety Review of Stretch Power Uprate Amendment (TAC No. MB9031)"

cc: A. Vietti-Cook, SECY
M. Johnson, OEDO
B. Sosa, OEDO
J. Lamb, OEDO
J. Dyer, NRR
C. Haney, NRR
C. Holden, NRR
L. Raghavan, NRR
M. Chernoff, NRR
E. Brown, NRR
M. Zobler, OGC

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

November 16, 2006

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

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

Dear Chairman Klein:

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

RECOMMENDATIONS

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

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

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

DISCUSSION

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

Additional comments from ACRS Member Graham B. Wallis

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

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

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

Additional comments from ACRS Member Sanjoy Banerjee

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

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

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

References:

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

References (continued)

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

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

November 17, 2006

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

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

Dear Chairman Klein:

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

CONCLUSION AND RECOMMENDATION

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

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

BACKGROUND AND DISCUSSION

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References:

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

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

November 17, 2006

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

SUBJECT: PROPOSED REVISION 1 TO REGULATORY GUIDE 1.189 (DG-1170), "FIRE PROTECTION FOR NUCLEAR POWER PLANTS"

Dear Mr. Reyes:

During the 537th meeting of the Advisory Committee on Reactor Safeguards, November 1-3, 2006, we met with representatives of the NRC staff to discuss Proposed Revision 1 to Regulatory Guide 1.189 (DG-1170), "Fire Protection for Nuclear Power Plants." Our Subcommittee on Plant Operations and Fire Protection also reviewed this matter on October 31, 2006. We also had the benefit of the document referenced.

RECOMMENDATION

Proposed Revision 1 to Regulatory Guide 1.189 (DG-1170) should be issued for public comment. We would like to have an opportunity to review the draft final version of this Guide after resolution of public comments.

BACKGROUND AND DISCUSSION

The regulatory framework which establishes the design principles and detailed requirements for fire protection in nuclear power plants are contained in various regulations, including General Design Criterion-3, "Fire Protection," of 10 CFR Part 50, Appendix A. Over the last 30 years, experience gained through fire events, fire endurance testing, and improved analytical techniques has allowed the staff to identify areas where more detailed guidance is needed to fulfill the regulatory requirements. During this time the NRC has referenced or endorsed 72 Codes and Standards applicable to fire protection in nuclear power plants and published 91 other information or guidance documents to address the Commission's fire protection concerns and industry experience. These documents have been incorporated into, or referenced by DG-1170.

DG-1170 provides guidance for establishing the fire protection design bases and for the analysis, design, operation, and maintenance of fire protection features in existing and new nuclear power plants. In addition, this Guide provides guidance related to the organization, qualifications, duties, training, and staffing requirements of licensee personnel who administer the fire protection program and also the members of the shift fire brigades. DG-1170 contains a compendium of the current state of regulatory knowledge and guidance applicable to existing and future nuclear power plants, as well as those awaiting decommissioning or being decommissioned.

Proposed Revision 1 to Regulatory Guide 1.189 (DG-1170) should be issued for public comment. We would like to have an opportunity to review the draft final version of this Guide after resolution of public comments.

Sincerely,

Graham B. Wallis

Graham Wallis
Chairman

Reference:

1. Memorandum dated September 25, 2006, from Jimi T. Yerokun, Chief, Risk Applications and Special Projects Branch, Division of Risk Assessment and Special Projects, Office of Nuclear Regulatory Research, to Michael R. Snodderly, Chief, Technical Support Branch, ACRS, Subject: Additional Information - Regulatory Guide 1.189, "Fire Protection for Nuclear Power Plants"

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

November 17, 2006

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

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

Dear Mr. Reyes:

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

RECOMMENDATIONS

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

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

DISCUSSION

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

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

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

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

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

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

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

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

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References:

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

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

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

December 12, 2006

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

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

Dear Chairman Klein:

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

Recommendations

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

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

Background and Discussion

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

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

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

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

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

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References

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

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

December 15, 2006

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: PROPOSED REVISIONS TO STANDARD REVIEW PLAN SECTIONS IN SUPPORT OF NEW REACTOR LICENSING

During the 538th meeting of the Advisory Committee on Reactor Safeguards, December 7-8, 2006, the Committee considered proposed revisions to the following Standard Review Plan (SRP), NUREG-0800, Sections and decided not to review them. The Committee has no objection to the staff's proposal to issue these SRP Sections. The Committee would like to be informed of any significant changes made to these SRP Sections prior to publishing them in final form:

- Proposed Revision 3 to SRP Section 2.3.3, Onsite Meteorological Measurements Program
- Proposed Revision 2 to SRP Section 3.2.1, Seismic Classification
- Proposed Revision 2 to SRP Section 3.2.2, System Quality Group Classification
- Proposed New SRP Section 3.13, Threaded Fasteners - ASME Code Class 1, 2, and 3
- Proposed New SRP Section 17.4, Reliability Assurance Program

The Committee has completed its review and/or consideration of all of the high priority SRP Sections provided by the staff. The ACRS is awaiting receipt of additional high priority SRP Sections from the staff.

References:
1. Memorandum dated December 5, 2006, from Cornelius F. Holden, Director, Division of Risk Assessment, NRR, to John T. Larkins, Executive Director, ACRS, Subject: Transmittal of Proposed Revision to Standard Review Plan NUREG-0800, Section 2.3.3, Rev. 3, "Onsite Meteorological Measurements Program."
2. Memorandum dated November 6, 2006, from Patrick L. Hiland, Director, Division of Engineering, NRR, to John T. Larkins, Executive Director, ACRS, Subject: Transmittal of Proposed Revision to Standard Review Plan NUREG-0800, Section 3.2.1, Rev. 2, "Seismic Classification."

3. Memorandum dated November 6, 2006, from Patrick L. Hiland, Director, Division of Engineering, NRR, to John T. Larkins, Executive Director, ACRS, Subject: Transmittal of Proposed Revision to Standard Review Plan NUREG-0800, Section 3.2.2, Rev. 2, "System Quality Group Classification."
4. Memorandum dated November 6, 2006, from Michele Evans, Director, Division of Component Integrity, NRR, to John T. Larkins, Executive Director, ACRS, Subject: Transmittal of Proposed New Standard Review Plan NUREG-0800, Section 3.13, "Threaded Fasteners - ASME Code Class 1, 2, and 3."
5. Memorandum dated October 31, 2006, from Patrick L. Hiland, Director, Division of Engineering, NRR, to John T. Larkins, Executive Director, ACRS, Subject: Transmittal of Proposed Revision to Standard Review Plan NUREG-0800, Section 17.4, Rev. 0, "Reliability Assurance Program."

cc: A. Vietti-Cook, SECY
M. Johnson, OEDO
J. Lamb, OEDO
B. Sosa, OEDO
B. Sheron, RES
J. Yerokun, RES
S. Koenick, NRR
S. (Min) Lee, NRR
R. Assa, RES
J. Ridgely, RES
M. Comar, NRR
R. McNally, NRR
P. Patnaik, NRR
P. Prescott, NRR
B. Harvey, NRR

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

December 15, 2006

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

SUBJECT: PROPOSED REVISION TO STANDARD REVIEW PLAN SECTION 13.3, "EMERGENCY PLANNING"

Dear Mr. Reyes:

During the 538th meeting of the Advisory Committee on Reactor Safeguards, December 7-8, 2006, we reviewed the proposed revision to NUREG-0800, Standard Review Plan (SRP), Section 13.3, "Emergency Planning." During our review, we had the benefit of discussions with representatives of the NRC staff and the Nuclear Energy Institute representing the industry, and the documents referenced.

Recommendation

NUREG-0800, Standard Review Plan, Section 13.3, "Emergency Planning," should be issued.

Background and Discussion

The proposed revision to SRP Section 13.3, "Emergency Planning," incorporates the new reactor licensing processes codified in 10 CFR Part 52. Although this revision is a restructure and rewrite of the existing SRP Section 13.3, it does not contain any new or unreviewed staff positions in the area of emergency planning.

The revision also incorporates changes to comply with the Commission policy on the use of emergency planning inspections, tests, analyses, and acceptance criteria (EP-ITAAC) specified in the Staff Requirements Memorandum, dated February 22, 2006. The staff presented a version of the generic EP-ITAAC to the Commission in SECY-05-0197, "Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria," dated October 28, 2005.

In addition to complying with the Commission Policy on the use of the EP-ITAAC at the combined license (COL) application stage, the staff has provided an option of using the EP-ITAAC at the early site permit (ESP) stage consistent with 10 CFR Part 52. The license application review process in 10 CFR Part 52 utilizes the same existing emergency preparedness and planning requirements contained in 10 CFR 50.47 as well as Appendix E to 10 CFR Part 50.

The SRP Section 13.3 places major emphasis on the successful completion of a full-participation exercise to demonstrate reasonable assurance of an executable offsite plan. However, deficiencies in the offsite plan are not required to be resolved before operation up to 5 percent of full power is authorized. The staff considers that successful completion of periodic exercises at existing reactor sites provides a degree of assurance on the effectiveness of the offsite plans for new reactors to be built at or near the site. Green-field sites may lack this assurance and a review of the offsite plan at an early stage may be prudent for these sites. The staff is aware of this issue.

Recent evacuations related to non-nuclear events have shown weaknesses in evacuation plans. The staff is reviewing these experiences for any applicable lessons learned. We support the staff's plan to follow-up with the Department of Homeland Security's Federal Emergency Management Agency (DHS/FEMA) to identify applicable lessons learned from experiences of evacuating large numbers of people (e.g., fire at the hazardous waste site in Apex, North Carolina, and Hurricane Katrina).

We support the staff's efforts in gaining an understanding of emergency planning policies and activities of other Countries. We encourage expansion of these efforts to identify good practices in emergency planning activities in Countries with major nuclear programs.

SRP Section 13.3 should be issued.

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References
1. Memorandum from David B. Matthews, Director, Division of New Reactor Licensing, NRR to John Larkins, Executive Director, ACRS, Subject: "Transmittal of Proposed Draft Revision to Standard Review Plan, NUREG-0800, Section 13.3, "Emergency Planning", dated September 8, 2006, (Agencywide Documents Access and Management System (ADAMS) accession number ML061870206).

2. SRM SECY-05-0197, "Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria," dated February 22, 2006, (ADAMS accession number ML060530316).

3. SECY-05-0197, "Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria," dated October 28, 2005, (ADAMS accession number ML052770225).

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

December 18, 2006

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

SUBJECT: DRAFT FINAL REGULATORY GUIDE 1.207 (DG-1144), "GUIDELINES FOR EVALUATING FATIGUE ANALYSES INCORPORATING THE LIFE REDUCTION OF METAL COMPONENTS DUE TO THE EFFECTS OF THE LIGHT-WATER REACTOR ENVIRONMENT FOR NEW REACTORS"

Dear Mr. Reyes:

During the 538th meeting of the Advisory Committee on Reactor Safeguards, December 7-8, 2006, we met with representatives of the NRC staff, American Society of Mechanical Engineers (ASME), and AREVA to discuss the draft final Regulatory Guide (RG) 1.207, "Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors." Our Subcommittee on Materials, Metallurgy, and Reactor Fuels reviewed this matter on December 6, 2006. We had the benefit of the documents referenced.

RECOMMENDATION

Regulatory Guide 1.207 should be issued as final.

BACKGROUND AND DISCUSSION

The ASME Boiler and Pressure Vessel Code Section III fatigue design curves, developed in the late 1960s and early 1970s, are based on tests conducted in laboratory air environments at ambient temperatures. In the Code, adjustments are made to strain and cyclic life to account for variations in material properties, surface finish, data scatter, and unknown effects. The Code does not explicitly account for potential degradation in the fatigue properties attributable to exposure to light water reactor (LWR) coolant environments. Recent fatigue test data and analyses have demonstrated conclusively that LWR environments have a significant impact on the fatigue life of reactor structural materials. Although the ASME Code Committee has recognized this issue for many years, it has been unable to reach consensus on how to resolve the matter. The staff has therefore taken the initiative to develop this Regulatory Guide.

Given that the fatigue life of ASME Class 1 components in LWR coolant environments is a function of several parameters, the NRC staff has selected an environmental correction factor, F_{en}, to account for LWR environments. By definition, F_{en} is the ratio of fatigue life of the material in a room temperature air environment to its fatigue life in a LWR coolant environment at operating temperature. To incorporate environmental effects into the fatigue evaluation, the fatigue usage is calculated using ASME Section III Code procedures, and the fatigue usage is multiplied by the correction factor. In license renewal applications, applicants have used this methodology to evaluate the fatigue usage of materials in Class 1 components.

The F_{en} methodology that the staff considers acceptable is described in RG 1.207. NUREG/CR-6909, "Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials," provides the technical basis for this methodology. In developing the underlying models, Argonne National Laboratory (ANL) researchers analyzed existing data to predict fatigue life as a function of temperature, strain rate, dissolved oxygen level in water, and sulfur content of the steel. They identified a strain threshold below which environmental effects on fatigue life do not occur. Using this guidance, only the types of stress cycles or load set pairs that exceed strain threshold criteria for carbon steels, low-alloy steels, austenitic stainless steels, and Ni-Cr-Fe alloys need to be considered for F_{en} calculations. The evaluation options depend on the complexity of the analyzed transient condition and the level of detail in the analysis. Detailed analyses may be used to reduce the conservatism in the calculated F_{en} values while simplified calculations will yield more conservative results. The calculated F_{en} values are then used to adjust ASME fatigue usage to account for environmental effects.

Another issue addressed by the staff in RG 1.207 is the non-conservatism of the current ASME stainless steel air design curve. Recent evaluations of stainless steel and nickel alloy fatigue test data demonstrate that the ASME air design curve is non-conservative in the mid-to-high cycle fatigue range. NUREG/CR-6909 provides a new stainless steel air design curve and a comprehensive technical basis for the new curve. RG 1.207 states that the F_{en} values defined for stainless steel in NUREG/CR-6909 should be used in conjunction with the new stainless steel air design curve when evaluating the fatigue usage of ASME Class 1 components.

In addition, the staff and ANL evaluated the incorporation of the F_{en} approach methodology in fatigue analyses for Ni-Cr-Fe alloys (e.g., Alloy 600 and 690) and welds. The staff concluded that the new fatigue design curve proposed for stainless steels also adequately represented the fatigue behavior of these alloys.

NUREG/CR-6909 contains evaluations of the margins of the ASME design curves. In conducting these evaluations, ANL researchers reviewed the literature to assess the factors (excluding environment) necessary to account for the effects of various uncertainties and differences between actual components and laboratory test specimens. The researchers also performed statistical analyses using Monte Carlo simulations to develop fatigue design curves. The staff has concluded that this approach is acceptable because the fatigue design curves are based on crack initiation, rather than component failure, and thus provide adequate margin. We concur with the staff's conclusion.

Key comments on RG 1.207 received from industry and the ASME were that the existing ASME design curves and methodology are adequate, that there is no need for a new regulatory guide, that the new guide will require more detailed and costly analyses in the design of new plants, and that the use of the new guide will also result in the need for an excessive number of snubbers and pipe whip restraints. These comments and the associated staff's responses were discussed with representatives of the ASME, AREVA, and the NRC staff during our Subcommittee and full Committee meetings. We are satisfied that these comments have been properly addressed by the staff. Therefore, RG 1.207 should be issued as final.

ASME representatives expressed their intent to continue their efforts to prepare a Code Case that would treat the reactor coolant environmental effects on fatigue and meet the objectives of RG 1.207. If a Code Case is developed that is acceptable to the staff, it would also provide an acceptable alternative to RG 1.207 for addressing the environmental effects on fatigue. The staff should interact with ASME in the development of this Code Case, as appropriate.

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

Additional comments by ACRS Member Otto L. Maynard are provided below.

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

Additional comments by ACRS Member Otto L. Maynard

I agree with my colleagues that the draft final Regulatory Guide (RG) 1.207 represents a significant improvement in the understanding and quantification of the influence of reactor coolant environments on the fatigue properties of certain materials. However, comments from the public and affected stakeholders have not been properly addressed. Stakeholders have argued that the implementation of RG 1.207 will require increased expenditures with questionable safety benefit. The ASME and other stakeholders have further argued that the existing codes and standards provide sufficient conservatism to account for fatigue-related issues. The NRC staff has not identified events or significant fatigue-related issues that would have been prevented by the implementation of the provisions of RG 1.207.

RG 1.207 should not be issued until the staff has demonstrated that the improvement in safety from implementation of the new requirement is sufficient to justify the increased cost to licensees of new reactors.

References:

1. Memorandum from Farouk Eltawila, Director, Division of Risk Assessment and Special Projects, Office of Nuclear Regulatory Research, to John T. Larkins, Executive Director, Advisory Committee on Reactor Safeguards/Advisory Committee on Nuclear Waste, "Transmittal of Final Regulatory Guide 1.207, 'Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of of the Light-Water Reactor Environment for New Reactors,'" dated November 29, 2006
2. Draft final Regulatory Guide 1.207 (DG-1144), "Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors" (Received November 16, 2006)
3. Draft final NUREG/CR-6909, Rev. 1, "Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials" (Received November 16, 2006)
4. Draft Staff Response to Public Comments on DG-1144 and on draft NUREG/CR-6909 (Received November 16, 2006)
5. Redline-strikeout comparison between DG-1144 and draft final Regulatory Guide 1.207 (Received November 16, 2006)

NRC FORM 335
(9-2004)
NRCMD 3.7

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER
(Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG-1125, Volume 28

2. TITLE AND SUBTITLE

A Compilation of Reports of The Advisory Committee on Reactor Safeguards 2006 Annual

3. DATE REPORT PUBLISHED

MONTH	YEAR
June	2007

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

6. TYPE OF REPORT

7. PERIOD COVERED *(Inclusive Dates)*

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Same as above

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

This compilation contains 57 Advisory Committee on Reactor Safeguards reports submitted to the U. S. Nuclear Regulatory Commission (NRC), or to the NRC Executive Director for Operations, during calendar year 2006. All reports have been made available to the public through the NRC Public Document Room, the U. S. Library of Congress, and the Internet at http://www.nrc.gov/reading-rm/doc-collections.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Early Site Permit
Extended Power Uprate
Generic Letter
Generic Safety Issue
License Renewal
NUREG
PWR Sump Performance
Regulatory Guide
Safety Evaluation Report
Safety Research
Standard Review Plan
Risk Informed
Simplified Boiling Water Reactor

13. AVAILABILITY STATEMENT
unlimited

14. SECURITY CLASSIFICATION
(This Page)
unclassified
(This Report)
unclassified

15. NUMBER OF PAGES

16. PRICE

Printed
on recycled
paper

Federal Recycling Program