NUREG-1125
Volume 25

A Compilation of
Reports of

The Advisory Committee on Reactor Safeguards

2003 Annual

U. S. Nuclear Regulatory Commission

April 2004

AVAILABILITY OF REFERENCE MATERIALS IN NRC PUBLICATIONS

NRC Reference Material

As of November 1999, you may electronically access NUREG-series publications and other NRC records at NRC's Public Electronic Reading Room at http://www.nrc.gov/reading-rm.html. Publicly released records include, to name a few, NUREG-series publications; *Federal Register* notices; applicant, licensee, and vendor documents and correspondence; NRC correspondence and internal memoranda; bulletins and information notices; inspection and investigative reports; licensee event reports; and Commission papers and their attachments.

NRC publications in the NUREG series, NRC regulations, and *Title 10, Energy*, in the Code of *Federal Regulations* may also be purchased from one of these two sources.

1. The Superintendent of Documents
   U.S. Government Printing Office
   Mail Stop SSOP
   Washington, DC 20402–0001
   Internet: bookstore.gpo.gov
   Telephone: 202-512-1800
   Fax: 202-512-2250
2. The National Technical Information Service
   Springfield, VA 22161–0002
   www.ntis.gov
   1–800–553–6847 or, locally, 703–605–6000

A single copy of each NRC draft report for comment is available free, to the extent of supply, upon written request as follows:

Address: Office of the Chief Information Officer,
Reproduction and Distribution
Services Section
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
E-mail: DISTRIBUTION@nrc.gov
Facsimile: 301–415–2289

Some publications in the NUREG series that are posted at NRC's Web site address http://www.nrc.gov/reading-rm/doc-collections/nuregs are updated periodically and may differ from the last printed version. Although references to material found on a Web site bear the date the material was accessed, the material available on the date cited may subsequently be removed from the site.

Non-NRC Reference Material

Documents available from public and special technical libraries include all open literature items, such as books, journal articles, and transactions, *Federal Register* notices, Federal and State legislation, and congressional reports. Such documents as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings may be purchased from their sponsoring organization.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at—

The NRC Technical Library
Two White Flint North
11545 Rockville Pike
Rockville, MD 20852–2738

These standards are available in the library for reference use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from—

American National Standards Institute
11 West 42nd Street
New York, NY 10036–8002
www.ansi.org
212–642–4900

Legally binding regulatory requirements are stated only in laws; NRC regulations; licenses, including technical specifications; or orders, not in NUREG-series publications. The views expressed in contractor-prepared publications in this series are not necessarily those of the NRC.

The NUREG series comprises (1) technical and administrative reports and books prepared by the staff (NUREG–XXXX) or agency contractors (NUREG/CR–XXXX), (2) proceedings of conferences (NUREG/CP–XXXX), (3) reports resulting from international agreements (NUREG/IA–XXXX), (4) brochures (NUREG/BR–XXXX), and (5) compilations of legal decisions and orders of the Commission and Atomic and Safety Licensing Boards and of Directors' decisions under Section 2.206 of NRC's regulations (NUREG–0750).

NUREG-1125
Volume 25

A Compilation of
Reports of

The Advisory Committee on Reactor Safeguards

2003 Annual

U. S. Nuclear Regulatory
Commission

April 2004

ABSTRACT

This compilation contains 47 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 2003. In addition, a report to the Commission on the NRC Safety Research Program, NUREG-1635, Volume 5, is included by reference only. 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.

PREFACE

The enclosed reports, issued during calendar year 2003, 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

ACRS MEMBERSHIP (2003)

CHAIRMAN:	Dr. Mario V. Bonaca, Retired Northeast Utilities
VICE CHAIRMAN:	Dr. Graham B. Wallis, Professor Emeritus Dartmouth College
MEMBERS:	Dr. George E. Apostolakis, Professor Massachusetts Institute of Technology
	Dr. F. Peter Ford, Retired General Electric Research and Development Center
	Dr. Thomas S. Kress, Retired Oak Ridge National Laboratory
	Mr. Graham M. Leitch, Retired PECO Energy, and Maine Yankee Atomic Power Company
	Dr. Dana A. Powers Sandia National Laboratories
	Dr. Victor H. Ransom, Professor Emeritus Purdue School of Nuclear Engineering
	Mr. Stephen L. Rosen, Retired STP Nuclear Operating Company at South Texas Project
	Mr. John D. Sieber, Retired Duquesne Light Company
	Dr. William J. Shack Argonne National Laboratory

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

Page

TABLE OF CONTENTS

Page

TABLE OF CONTENTS

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

February 14, 2003

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

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE LICENSE RENEWAL APPLICATION FOR THE MCGUIRE NUCLEAR STATION UNITS 1 AND 2 AND THE CATAWBA NUCLEAR STATION UNITS 1 AND 2

Dear Chairman Meserve:

During the 499th meeting of the Advisory Committee on Reactor Safeguards on February 6–8, 2003, we completed our review of the License Renewal Application (LRA) for the McGuire Nuclear Station Units 1 and 2 (McGuire) and the Catawba Nuclear Station Units 1 and 2 (Catawba), and the related final safety evaluation report (SER) prepared by the NRC staff. Our review included a meeting of our Plant License Renewal Subcommittee on October 8, 2002. During our review, we had the benefit of discussions with representatives of the NRC staff and Duke Energy Corporation (Duke). We also had the benefit of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

1. The Duke application for renewal of the operating licenses for McGuire Units 1 and 2 and Catawba Units 1 and 2 should be approved.

2. The programs instituted to manage aging-related degradation are appropriate and provide reasonable assurance that McGuire Units 1 and 2 and Catawba Units 1 and 2 can be operated in accordance with their current licensing bases for the period of extended operation without undue risk to the health and safety of the public.

BACKGROUND AND DISCUSSION

This report fulfills the requirement of 10 CFR 54.25, which states that the ACRS should review and report on all license renewal applications. McGuire Units 1 and 2 and Catawba Units 1 and 2 are 3,411- MWt, four-loop Westinghouse pressurized-water reactors (PWRs) in ice condenser containments. In its application, Duke requested that the NRC renew the operating licenses for all four units beyond their current license terms, which expire on June 12, 2021 (McGuire Unit 1); March 3, 2023 (McGuire Unit 2); December 6, 2024 (Catawba Unit 1); and February 24, 2026 (Catawba Unit 2). At the time of the application, only McGuire Unit 1 met the requirements of 10 CFR 54.17(c), which prohibits an applicant from submitting an application for license renewal earlier than 20 years before the expiration of its current operating license. Duke requested an exemption from this requirement, which the NRC staff granted based on the similarities of the four units and the efficiency of a single application.

The final SER documents the results of the staff's review of information submitted by Duke, including commitments that were necessary to resolve open items identified by the staff in the initial SER. In particular, the staff reviewed the completeness of the applicant's identification of structures, systems, and components (SSCs) that are subject to aging management; the integrated plant assessment process; the applicant's identification of the possible aging mechanisms associated with passive, long-lived components; and the adequacy of the applicant's aging management programs. The staff also conducted several inspections at Duke's engineering offices and at the McGuire and Catawba sites to verify the adequacy of the methodology described in the application and its implementation.

During our Plant License Renewal Subcommittee meeting on October 8, 2002, the lead NRC license renewal inspector for Region II provided an overview of the NRC's inspection process. This process, which is well-structured and effective, is becoming increasingly important as license renewal applications become less detailed. As a result, as in other recent applications, the review of the McGuire and Catawba LRA required a substantial number of requests for additional information and depended heavily on review of plant drawings at the sites.

On the basis of our review of the final SER, we agree with the staff's conclusion that all open and confirmatory items have been closed appropriately, and there are no issues that preclude renewal of the operating licenses for McGuire Units 1 and 2 and Catawba Units 1 and 2.

The process implemented by the applicant to identify SSCs that are within the scope of license renewal was effective. However, in the initial SER the staff identified a number of SSCs that should have been in the scope of license renewal but were excluded by Duke's interpretation of license renewal requirements. Among those SSCs were fan and damper housings, building sealants, electrical equipment connecting the units to the offsite power source for recovery from station blackout (SBO), and jockey pumps and manual fire suppression equipment in potential fire exposure areas. The inclusion of fan and damper housings, building sealants, and SBO equipment has been disputed in previous license renewal applications.

For fan and damper housings, Duke initially took the position that loss of pressure retention or structural integrity function would be evidenced by functional failure, as is a failure of the active components of dampers and fans. By contrast, the staff views the passive components of these assemblies as being within the scope of license renewal, just like pump casings, which are explicitly called for in 10 CFR 54.21. We agree that the explicit example provided in the rule supports the staff's interpretation. With regard to jockey pumps, the staff determined that these components are relied upon to meet the requirements of 10 CFR 50.48, "Fire Protection." We concur with the staff's determination. Duke agreed to close these open items by bringing all of the identified SSCs into the scope of license renewal.

During our review, we questioned why certain other SSCs were not included within the scope and, in all cases, the applicant provided appropriate justification for exclusion. We conclude that the applicant and the staff have appropriately identified all SSCs that are within the scope of license renewal.

The applicant performed a comprehensive aging management review of SSCs that are within the scope of license renewal. Appendix B to the LRA describes 51 aging management programs for license renewal, which include existing, enhanced, and new programs. In addition, the resolution of staff questions and SER open items has resulted in further commitments, including the implementation of a one-time inspection of the condenser circulating water system expansion joints at Catawba to characterize potential degradation, one-time VT-1 inspection of the pressurizer spray head, one-time inspection of the internal surfaces of the auxiliary feedwater system carbon steel piping components, and an inspection program for non-environmentally qualified neutron flux instrumentation circuits. The SER lists 21 such committed actions to be implemented by the applicant.

The McGuire and Catawba LRA includes a new aging management program, the Alloy 600 Aging Management Review. This program is intended to identify Alloy 600/690, 82/182, and 52/152 locations; to rank susceptibility to primary water stress corrosion cracking (PWSCC); and to verify that nickel-based alloy locations are adequately inspected by the Inservice Inspection Program, the Control Rod Drive Mechanism and other Vessel Head Penetration (VHP) programs, the Reactor Vessel Internals Program, and the Steam Generator Integrity Program. This review will provide general oversight and management of cracking due to PWSCC. We applaud this initiative to provide comprehensive oversight of activities to manage PWSCC. Given the current challenge created by PWSCC, we encourage Duke to implement this program soon, in the current license term, rather than waiting for the end of the initial license terms of the four units.

With regard to reactor vessel penetration nozzle cracking and head wastage issues, Duke has committed to incorporate the future industry resolution of these issues into the VHP Nozzle Program and the Alloy 600 Management Review Program. This provides reasonable assurance that the effects of aging associated with the VHP Nozzle Program and the Alloy 600 Review Program will be adequately managed so that the intended function(s) will be maintained in a manner that is consistent with the current licensing basis throughout the period of extended operation.

Duke is the first utility to seek license renewal for plants that use ice condensers in the containment to absorb thermal energy in the event of a loss-of-coolant-accident or a steamline break. Duke has developed a new program to manage aging degradation of ice baskets and ice condenser components at McGuire and Catawba. We agree with the staff's conclusion that the proposed program is adequate to identify and manage aging effects during the period of extended operation.

Duke identified those components of the McGuire and Catawba plants that are supported by time-limited aging analyses and provided sufficient data to demonstrate that the components have sufficient margin to operate properly for the period of extended operation.

As noted in previous applications, LRAs include a substantial number of activities and commitments that will not be accomplished until near the end of the current license period. Consequently, the NRC staff will need to conduct a substantial amount of inspection activity just before the plants enter the extended period of operation. The staff is aware of this future workload and has issued Inspection Procedure 71003, "Post-Approval Site Inspection for License Renewal," to manage this significant effort. Given the large number of power plants that will approach the license renewal term at approximately the same time, this nationwide inspection effort is likely to impose a major demand for staff resources.

The staff has performed an outstanding review of the Duke application. The applicant and the staff have identified plausible aging effects associated with passive, long-lived components. The applicant has also established adequate programs to manage the effects of aging so that McGuire Units 1 and 2 and Catawba Units 1 and 2 can be operated in accordance with their current licensing bases for the period of extended operation without undue risk to the health and safety of the public.

Sincerely,

Mario V. Bonaca

Mario V. Bonaca
Chairman

References:
1. Letter dated June 13, 2001, from M. S. Tuckman, Duke Energy Corporation, to U. S. Nuclear Regulatory Commission, transmitting Application to Renew the Operating Licenses of McGuire Nuclear Station, Units 1 and 2 and Catawba Nuclear Station, Units 1 and 2.
2. U.S. Nuclear Regulatory Commission, NUREG-XXX, "Safety Evaluation Report Related to the License Renewal of McGuire Nuclear Station, Units 1 and 2 and Catawba Nuclear Station, Units 1 and 2," January 2003.
3. U.S. Nuclear Regulatory Commission, NRC Inspection Procedure 71003, "Post-Approval Site Inspection for License Renewal," December 9, 2002.

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

February 14, 2003

Dr. William D. Travers
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: DRAFT FINAL REGULATORY GUIDE DG-1077, "GUIDELINES FOR ENVIRONMENTAL QUALIFICATION OF MICROPROCESSOR-BASED EQUIPMENT IMPORTANT TO SAFETY IN NUCLEAR POWER PLANTS"

Dear Dr. Travers:

During the 499th meeting of the Advisory Committee on Reactor Safeguards on February 6-8, 2003, we met with representatives of the NRC's Office of Nuclear Regulatory Research to discuss the Draft Final Regulatory Guide DG-1077, "Guidelines for Environmental Qualification of Microprocessor-Based Equipment Important to Safety in Nuclear Power Plants." We also had the benefit of the documents referenced.

RECOMMENDATION

DG-1077 provides appropriate guidance for environmental qualification of microprocessor-based equipment and should be issued.

DISCUSSION

DG-1077 endorses the use of the Institute of Electrical and Electronics Engineers Standard 323-1983 (as reaffirmed in 1996) or the International Electrotechnical Commission Standard 60780, with some enhancements and exceptions, as well as RG 1.89 and RG 1.180. The staff used NUREG/CR-6741 as the basis for the guidelines for environmental qualification of microprocessor-based equipment.

Both NUREG/CR-6741 and DG-1077 recognize the important differences in the degradation and failure modes associated with microprocessor-based equipment and provide specific guidance to properly resolve issues in the environmental qualification of this equipment for use in nuclear power plants. The guidance recognizes the international market for replacement, and new instrumentation and controls, and appropriately incorporates international standards.

Sincerely,

Mario V. Bonaca

Mario V. Bonaca
Chairman

References:

1. U. S. Nuclear Regulatory Commission, DG-1077, "Guidelines for Environmental Qualification of Microprocessor-based Equipment Important to Safety in Nuclear Power Plants," December 2001.
2. Institute of Electrical and Electronics Engineers, IEEE 323-1983, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations," September 30, 1983.
3. International Electrotechnical Commission, IEC 60780, "Nuclear Power Plants - Electrical Equipment of the Safety System - Qualifications," Second Edition, 1998.
4. U. S. Nuclear Regulatory Commission, RG 1.89, "Environmental Qualification of Certain Electrical Equipment Important to Safety for Nuclear Power Plants," June 1984.
5. U. S. Nuclear Regulatory Commission, RG 1.180, "Guidelines for Evaluating Electromagnetic Radiofrequency Interference in Safety-Related Instrument and Control Systems," January 2000.
6. U. S. Nuclear Regulatory Commission, NUREG/CR-6741, "Application of Microprocessor-Based Equipment in Nuclear Power Plants - Technical Basis for Qualification Methodology," January 2003.

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

February 20, 2003

Dr. William D. Travers
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: PROPOSED RESOLUTION OF GENERIC SAFETY ISSUE-191, "ASSESSMENT OF DEBRIS ACCUMULATION ON PWR SUMP PERFORMANCE"

Dear Dr. Travers:

During the 499th meeting of the Advisory Committee on Reactor Safeguards, February 6-8, 2003, we reviewed the proposed NRC Generic Letter 2003-XX, "Potential Impact of Debris Blockage on Emergency Recirculation During Design-Basis Accidents at Pressurized-Water Reactors," and Draft Regulatory Guide DG-1107, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," associated with the resolution of Generic Safety Issue (GSI)-191. Our Subcommittee on Thermal-Hydraulic Phenomena also reviewed this matter during its meeting on February 4, 2003. During this review, we had the benefit of discussions with representatives of the NRC staff. We also had the benefit of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

1. We agree with the staff's proposal to issue the proposed Generic Letter for public comment.

2. We agree with the staff's proposal to issue Draft Regulatory Guide DG-1107 provided that the accompanying request for public comments incorporates our recommendation 3 and the associated discussion.

3. The staff should evaluate the possibility that strainers may prove to be so susceptible to debris blockage that alternative solutions may be required to ensure long-term cooling. The staff should invite public comments on this matter.

4. The "acceptable methods" discussed in DG-1107 should be peer reviewed after technical reports from the Office of Nuclear Regulatory Research (RES) contractors become available.

DISCUSSION

In our letter of September 14, 2001, we agreed with the staff that potential issues associated with the performance of containment sumps in pressurized water reactors had been identified. We stated that, if plant-specific analyses were required, guidance for performing these analyses should be developed by the staff. We also indicated our desire to review the proposed final disposition of this issue. The staff developed the proposed Generic Letter and associated draft Regulatory Guide DG-1107 for resolving GSI-191.

The proposed Generic Letter will serve the purpose of initiating the process of gathering plant-specific information and requiring licensees to develop plans for resolving potential issues. This is an appropriate first step toward resolving GSI-191. The schedule for responding to the Generic Letter is realistic and should be maintained in order to reach an expeditious resolution. The ability of licensees to respond may depend significantly on the availability and acceptability of guidance being prepared by the Nuclear Energy Institute (NEI), which is expected to be published in September 2003.

DG-1107 describes the technical issues that require resolution in order for plants to ensure that sump recirculation will function adequately following a loss-of-coolant-accident (LOCA). These issues include potential sources of debris, debris generation and transport, and screen blockage. These phenomena are influenced by many details of the location and size of the LOCA, the forms of insulation on neighboring piping and vessels, and the numerous flow paths by which the debris can reach the sump. A workable approach to predicting these phenomena requires scientific understanding combined with suitable engineering models that adequately describe selected key parameters and their relationships. DG-1107 correctly anticipates the possible need to make conservative assumptions because of high degrees of uncertainty associated with these processes.

DG-1107 describes "acceptable methods" for predicting debris sources and generation, transport, accumulation, and loss of net positive suction head. These methods are being studied at the Los Alamos National Laboratory and the University of New Mexico. Our Thermal-Hydraulic Phenomena Subcommittee heard presentations summarizing parts of this work during its meeting on February 4, 2003. We have not yet received the final report that describes the suggested design methods and their implementation. Therefore, we are uncertain as to whether these methods are sufficiently mature to be included in the final regulatory guide. The "acceptable methods" require peer review when the supporting documents are available. The staff will also need to develop a technical basis for assessing the acceptability of the methods proposed in the forthcoming NEI guidance document.

We anticipate that the "acceptable methods" will require further review, and possible revision, before publication of the final Regulatory Guide. However, these future activities should not impede the release of the draft Regulatory Guide DG-1107 for public comment.

The staff also needs to consider the possibility that strainers may prove to be so susceptible to debris blockage that alternative solutions may be required to ensure long term cooling. This might involve, for example, changes in the types of insulation used within containment or implementing diverse means of providing long-term cooling.

We look forward to reviewing the final draft of the Generic Letter and DG-1107 after resolution of public comments.

Sincerely,

Mario V. Bonaca

Mario V. Bonaca
Chairman

References:

1. U. S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Generic Letter 2003-XX, "Potential Impact of Debris Blockage on Emergency Recirculation During Design-Basis Accidents at Pressurized-Water Reactors," transmitted January 15, 2003.
2. U. S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Draft Regulatory Guide DG-1107, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," transmitted December 2002.
3. Letter dated September 14, 2001, from George E. Apostolakis, ACRS, to William D. Travers, EDO, Subject: Generic Safety Issue-191, "Assessment of Debris Accumulation on PWR Sump Performance."

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

February 21, 2003

Dr. William D. Travers
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: PRESSURIZED THERMAL SHOCK (PTS) REEVALUATION PROJECT: TECHNICAL BASES FOR POTENTIAL REVISION TO PTS SCREENING CRITERIA

Dear Dr. Travers:

During the 499th meeting of the Advisory Committee on Reactor Safeguards, February 6-8, 2003, we reviewed a draft report that the NRC's Office of Nuclear Regulatory Research (RES) staff has prepared to document its work to develop technical bases for revising the pressurized thermal shock screening criteria in the PTS rule (10 CFR 50.61). Our Subcommittee on Materials and Metallurgy also reviewed this matter on February 5, 2003. During our review, we had the benefit of discussions with representatives of the NRC staff and the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

1. The PTS Reevaluation Project has developed comprehensive technical bases for analyzing the susceptibility of reactor pressure vessels to PTS and to support rulemaking to revise the current PTS Rule 10 CFR 50.61. Plant-specific studies show that the current PTS screening criteria are very conservative for the given plants. This work may also provide a basis for reducing unnecessary conservatism in current regulation on operational limits on pressure vessel heatup and cooldown (Appendix G to 10 CFR Part 50).

2. The draft technical bases summary report needs substantial revision to describe more clearly the basic phenomena, issues, approaches, and conclusions. Topical reports on some important technical tasks have not yet been completed.

3. We support plans for an external peer review of the technical work.

DISCUSSION

The PTS Rule 10 CFR 50.61 was established to ensure the integrity of irradiation-embrittled reactor pressure vessels. Reactor pressure vessel steels undergo a transition from highly ductile behavior at high temperatures to brittle behavior at low temperatures. This change in behavior occurs abruptly over a narrow range of temperatures, and a temperature RT_{NDT} can be defined to characterize the transition in fracture behavior. Under irradiation, the transition temperature RT_{NDT} increases, making the vessel susceptible to brittle fracture at higher temperatures.

Estimation of the frequency of vessel failure requires (1) identification of sequences that could lead to rapid cooling of the vessel and estimation of their frequencies of occurrence; (2) determination of the pressure, temperature, and heat transfer coefficient adjacent to the embrittled portion of the vessel for each of the event sequences and use of these to determine the thermal stress on the vessel and the fracture toughness of the vessel material; and (3) probabilistic fracture mechanics analyses to determine the probability of failure under the induced thermal and pressure stresses on the embrittled vessel.

The studies conducted by the PTS Reevaluation Project to assess the frequency of vessel failure are much more comprehensive than those done in the early 1980s. These recent studies include systematic consideration of uncertainties in (1) the frequency of initiating events for PTS scenarios, (2) the thermal-hydraulic conditions that provide the driving forces for crack propagation and initiation, and (3) the assessment of the fracture toughness of the vessel materials. Substantial work has also been done to develop more realistic distributions for flaw density and geometry and improve the accuracy and rigor of the probabilistic fracture mechanics code, FAVOR, which is used in these analyses.

The results from detailed plant-specific studies of Oconee Nuclear Station, Unit 1; Palisades Plant; and Beaver Valley Power Station, Unit 1, show that the current PTS screening criteria are very conservative for these plants. Two of these plants are among the most susceptible to irradiation embrittlement in the reactor fleet. Moreover, the staff has presented good arguments as to why these results can be considered representative of the entire fleet of pressurized water reactors. The staff also currently has additional studies under way to further confirm the generic applicability of these results.

The distributions of the predicted vessel failure frequency are very broad. There are about three orders of magnitude between the 5th and 95th percentiles of the failure frequency. The distributions are also highly skewed, so that the mean and 95th percentiles are virtually identical. At embrittlement levels corresponding to the current screening criterion, the mean frequency of vessel failure is about 1×10^{-8}/year. This is a factor of about 500 lower than the current acceptance level. For plant lifetimes of 60-80 years, the predicted mean vessel failure frequencies will range from 5×10^{-10}/year to 5×10^{-8}/year.

Based on current estimates, 10 plants will be within 20°F of the current screening criteria at the end of their original 40-year licenses. Because the transition temperature increases about 1°F per year of operation, revision of the current PTS screening criteria could significantly impact the licensees decisions regarding whether to pursue license renewal for these plants.

The staff has concurred with our recommendation in our report of July 18, 2002, that a risk-informed acceptance criterion for vessel failure frequency should be based on considerations of large early-release frequency and not on core damage frequency. The scoping studies presented by the staff suggest that it is likely that the performance of containment systems after vessel failure will be adequate to ensure that a vessel failure frequency criterion of 1×10^{-6}/year will be adequate to ensure that the risk due to PTS is acceptably low. These studies also provide an approach for developing a risk-informed failure frequency criterion. Nevertheless, further consideration of the possibility of late containment failure may be needed and should be pursued if rulemaking is undertaken.

The documentation of the technical bases is currently inadequate and incomplete. Topical reports on some important technical tasks have not yet been completed. For example, no referenceable reports are available on the experiments and analyses that were performed to assess the potential for strong temperature gradients in the downcomer region near the beltline region that would invalidate the one-dimensional treatment of the thermal boundary conditions used in the probabilistic fracture mechanics analyses. Similarly, no referenceable reports are available on the studies undertaken at the University of Maryland that were used to develop a method to address thermal-hydraulic uncertainties, or to document the methods and approaches used for the probabilistic risk assessments used to determine the frequency of PTS events. A meaningful peer review cannot be performed without more complete documentation.

The draft technical bases summary document needs substantial revision to describe more clearly the basic phenomena, issues, approaches, and conclusions. Because this study synthesizes technical information from several engineering disciplines, it is important to explain how these disciplines interact and how the synthesis influences the conclusions. For example, the staff has identified a wide range of changes that reduce conservatism in the analyses. These include changes in the crack distribution model, finer binning of thermal-hydraulic sequences, removal of conservative bias in the toughness model, and crediting of operator actions. The staff also identified changes that increase the failure frequency, such as inclusion of medium and large-break loss-of-coolant accidents and errors of commission. The staff has shown that it has a good understanding of the relative importance of these various factors in producing the change in the predicted frequency of vessel failure. The staff has also made a systematic attempt to assess the impact of uncertainties. A clear explanation of which factors have the largest impact on the change in the predicted frequency of failure would focus attention on understanding those uncertainties that have the greatest impact.

The staff also needs to revise the discussion of the treatment of uncertainties. Although the studies have attempted to distinguish between aleatory and epistemic uncertainties and the FAVOR code implements methods that account for the different ways they impact the failure frequencies, the current document does not always make clear that the epistemic and aleatory uncertainties were correctly handled.

We commend the staff for an outstanding multidisciplinary study and look forward to reviewing the staff's final reports.

Sincerely,

Mario V. Bonaca

Mario V. Bonaca
Chairman

References:
1. Memorandum dated December 31, 2002, from Ashok C. Thadani, Office of Nuclear Regulatory Research, NRC, to Samuel J. Collins, Office of Nuclear Reactor Regulation, NRC, transmitting Draft NUREG-????, "Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Criteria in the PTS Rule (10CFR50.61)," December 2002.
2. Letter dated July 18, 2002, from George Apostolakis, ACRS Chairman, to William D. Travers, Executive Director for Operations, NRC, Subject: Risk Metrics and Criteria for Reevaluating the Technical Basis of the Pressurized Thermal Shock Rule.

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

February 24, 2003

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: PROPOSED AMENDMENTS TO 10 CFR PART 50, APPENDIX E, PARAGRAPHS IV.B AND IV.F.2

During the 499th meeting of the Advisory Committee on Reactor Safeguards, February 6-8, 2003, the Committee considered the proposed amendments to 10 CFR Part 50, Appendix E, relating to (1) NRC approval of changes to Emergency Action Levels and (2) exercise requirements for co-located licensees, and decided not to review these amendments. The Committee agrees with the staff's proposal to issue these amendments for public comment.

cc: A. Vietti-Cook, SECY
J. Craig, OEDO
I. Schoenfeld, OEDO
B. Boger, NRR
T. Quay, NRR
M. Jamgochian, NRR

Reference:
Memorandum dated January 23, 2003, from Bruce Boger, Office of Nuclear Reactor Regulation, to John T. Larkins, ACRS, Subject: Request to Defer ACRS Review of a proposed Rulemaking to Revise 10 CFR Part 50, Appendix E.

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

March 11, 2003

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: DRAFT REGULATORY GUIDE DG-1079, "CRITERIA FOR POWER SYSTEMS FOR NUCLEAR POWER PLANTS" (PROPOSED REVISION 3 TO REGULATORY GUIDE 1.32)

During the 500th meeting of the Advisory Committee on Reactor Safeguards on March 6-8, 2003, the Committee considered the Draft Regulatory Guide DG-1079, "Criteria for Power Systems for Nuclear Power Plants," and decided to review it after reconciliation of public comments. The Committee agrees with the staff's proposal to issue the draft Regulatory Guide for public comment.

Reference:
Memorandum dated February 5, 2003, from Michael Mayfield,Office of Nuclear Regulatory Research, to John Larkins, Executive Director, ACRS, Subject: Draft Regulatory Guide DG-1079, "Criteria for Power Systems for Nuclear Power Plants," (Proposed Revision 3 to Regulatory Guide 1.32)

cc: A. Vietti-Cook, SECY
J. Craig, OEDO
I. Schoenfeld, OEDO
A. Thadani, RES
M. Mayfield, RES
C. Ader, RES
S. Aggarwal, RES

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

March 12, 2003

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

Dear Chairman Meserve:

SUBJECT: DRAFT REVIEW STANDARD, RS-002: "PROCESSING APPLICATIONS FOR EARLY SITE PERMITS"

During the 500th meeting of the Advisory Committee on Reactor Safeguards, March 6-8, 2003, we met with representatives of the NRC's Office of Nuclear Reactor Regulation (NRR) to discuss the staff's draft Review Standard for processing applications for early site permits (ESPs). We also had the benefit of the document referenced.

CONCLUSIONS

The draft ESP Review Standard is appropriate for review of early site permit applications and will accommodate the industry's proposed use of the Plant Parameter Envelope (PPE) concept.

DISCUSSION

The staff has modified the appropriate sections of the Standard Review Plan (SRP) to make use of existing guidance to the extent possible. The modifications generally consist of elimination of the contents of the SRP that are not applicable to ESP and revisions to bring the SRP up to date. In general, references to plant layouts or design details are deleted and replaced with a statement of the form: "... [specify these details for] a nuclear power plant or plants of specified type that might be constructed on the proposed site to the extent this information is available." Some review issues that require knowledge of items that are design-specific, such as source terms, will be accommodated by bounding values specified in the PPE portion of the application and confirmed at the Combined License (COL) stage. For already approved sites with existing plants, most of the review areas called for by the standard will have already been sufficiently addressed. The applicant will merely need to verify, compile, and docket these review areas.

Although the sections of the current SRP that deal with siting issues require a specific design, the proposed ESP standard recognizes that by specifying parameters such as distance to the exclusion area boundary, source term characteristics, and relative concentration (χ/Q) values in the PPE, it will be possible to demonstrate that a plant that fits within the PPE can be safely located on the site. The PPE can also accommodate the need to assess incremental environmental impact to ensure that it is acceptable. We believe that granting ESP based on the guidance of this proposed review standard will assure adequate protection and acceptable environmental impact when a plant is built on the approved site.

Sincerely,

Mario V. Bonaca
Chairman

Reference:

U.S. Nuclear Regulatory Commission Review Standard (Draft) RS-002, "Processing Applications for Early Site Permits," Draft for Interim Use and Public Comment, December 23, 2002.

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

March 13, 2003

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

SUBJECT: REACTOR OVERSIGHT PROCESS

Dear Chairman Meserve:

The Advisory Committee on Reactor Safeguards (ACRS) and its Plant Operations Subcommittee have had a number of interactions with the U.S. Nuclear Regulatory Commission (NRC) staff on the Reactor Oversight Process (ROP). In reports dated October 12, 2001, and February 13, 2002, the ACRS raised several issues that included:

- the appropriateness of the threshold values for the yellow-red performance indicator (PI) levels, and
- inconsistencies between the performance assessment and the significance determination process (SDP).

The ACRS met with the staff at its 500th meeting on March 6, 2003, to discuss these issues. At the conclusion of this meeting, it was evident that there are still significant disagreements between the staff and the Committee. This report, then, is intended to clarify the ACRS views on this matter and to serve as a basis for further discussion.

The ACRS views on the ROP are as follows:

1. The purpose of the ROP is to assess safety performance so that the agency can take appropriate action.

2. The ROP is risk-informed because it focuses on performance areas and indicators that affect safety.

3. It is incorrect to base thresholds for PIs on risk metrics such as ΔCDF (changes in core damage frequency) and ΔLERF (changes in large, early release frequency).

4. The thresholds separating all the performance levels (colors) should be performance-based and determined by expert judgement similar to the selection of the current green/white thresholds.

5. The principal role for the SDP is to assign risk characterization to inspection findings – not to be an evaluation of performance.

6. PIs are needed for the cross-cutting issues and their development should be pursued by the staff.

7. The Action Matrix should reflect the complementary results of the performance assessment and the SDP.

8. Lack of parity among thresholds may result in suboptimal allocation of NRC and licensee resources.

DISCUSSION

Our view is that the purpose of the ROP is to assess changes in performance, not changes in risk. We believe that the ROP is risk-informed because it focuses attention on performance areas that are known to be cornerstones of safety. As we have noted previously, however, it is misleading to assess the importance of changes even in a risk-informed PI in terms of ΔCDF.

Clearly, degraded performance can translate into an increase in the risk posed by a given plant. However, a realistic estimate of ΔCDF cannot be determined from changes in a single isolated parameter with the assumption that all other factors that can affect CDF remain constant. Thus, the selection of thresholds based on ΔCDF, as was done for the "number-of-scrams" PI, is misleading with respect to indicating the extent of degraded performance. Our view is that such thresholds should be selected on a performance basis and chosen through expert judgment and not be based on such risk considerations.

The SDP process should continue to evaluate the risk significance of events and findings. This information complements the performance assessment findings from the PIs. The two sets of information are complementary, and it is appropriate that both be addressed in the Action Matrix.

We continue to doubt the validity of the assumption that degraded performance in the cross-cutting areas will be revealed by the current PIs and inspections. Efforts to develop new PIs should be focused on licensees' corrective action programs, human performance, and safety conscious work environment.

The staff and the Committee agree that the significance of the thresholds for the various PIs should be examined. In addition to improving the coherence of the Action Matrix, parity in significance will yield another benefit. NRC and licensee resources are naturally biased toward performance areas that are rated other than green. If the thresholds are chosen inappropriately, then resources may be misallocated.

Sincerely,

Mario V. Bonaca

Mario V. Bonaca
Chairman

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

March 13, 2003

Dr. William D. Travers
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Dear Dr. Travers:

SUBJECT: DRAFT FINAL REGULATORY GUIDE DG-1119, "GUIDELINES FOR EVALUATING ELECTROMAGNETIC AND RADIO-FREQUENCY INTERFERENCE IN SAFETY-RELATED INSTRUMENTATION AND CONTROL SYSTEMS"

During the 500th meeting of the Advisory Committee on Reactor Safeguards on March 6-8, 2003, we met with representatives of the NRC's Office of Nuclear Regulatory Research (RES) to discuss draft final Regulatory Guide DG-1119, "Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems." We also had the benefit of the documents referenced.

RECOMMENDATION

DG-1119 provides appropriate guidance for evaluating the effects of electromagnetic and radio-frequency (EMI/RFI) interference on safety-related instrumentation and control (I&C) systems and should be issued. Public comments have been appropriately dispositioned by the staff.

DISCUSSION

Regulatory Guide 1.180, "Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems," which DG-1119 will replace, was issued in January 2000. Since the issuance of Regulatory Guide 1.180, there have been significant changes in the electromagnetic and radio-frequency envelopes in which these instruments and controls operate. The revised regulatory guide provides guidance to address these changes. These revisions include endorsing Military Standard (MIL-STD) 461E and the International Electrotechnical Commission (IEC) 61000 series of EMI/RFI test methods, extending the guidance to cover signal line testing, incorporating frequency ranges where portable communication devices are experiencing an increase in use, and relaxing the operating envelopes (test levels) when experience and confirmatory research warrants.

RES performed two studies to develop the technical bases for guidance on EMI/RFI and power surge withstand capability. RES also documented the research and current knowledge base for electromagnetic compatibility testing along interconnecting signal lines, which had been an open item in the original version of Regulatory Guide 1.180. This issue is now addressed in the current draft final guide, DG-1119. The recommendations on test criteria, test methods, and operating envelopes were significantly influenced by existing military standards used by the Department of Defense.

Sincerely,

Mario V. Bonaca
Chairman

References:

1. U.S. Nuclear Regulatory Commission, Regulatory Guide DG-1119, "Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems," February 2003.
2. U. S. Nuclear Regulatory Commission, NUREG/CR-5609, "Electromagnetic Compatibility Testing for Conducted Susceptibility Along Interconnecting Signal Lines," dated June 2002.
3. U. S. Nuclear Regulatory Commission, NUREG/CR-XXXX, ORNL/TM-2001/140, "Comparison of U.S. Military and International Electromagnetic Compatibility Guidance," June 2002.

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

March 14, 2003

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

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE LICENSE RENEWAL APPLICATION FOR THE PEACH BOTTOM ATOMIC POWER STATION UNITS 2 AND 3

Dear Chairman Meserve:

During the 500th meeting of the Advisory Committee on Reactor Safeguards, March 6-8, 2003, we completed our review of the license renewal application for the Peach Bottom Atomic Power Station Units 2 and 3 and the final Safety Evaluation Report (SER) prepared by the NRC staff. Our Plant License Renewal Subcommittee also reviewed this matter during a meeting on October 30, 2002. During our review, 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.

RECOMMENDATIONS AND CONCLUSIONS

1. The Exelon application for renewal of the operating licenses for Peach Bottom Atomic Power Station Units 2 and 3 should be approved.

2. The programs instituted by the applicant to manage age-related degradation are appropriate and provide reasonable assurance that Peach Bottom Atomic Power Station Units 2 and 3 can be operated in accordance with their current licensing bases for the period of extended life without undue risk to the health and safety of the public.

3. The scram at Peach Bottom Unit 2 that occurred on December 21, 2002, highlighted a number of weaknesses in the current corrective action and preventive maintenance programs. We expect that ongoing corrective actions committed by the licensee will resolve these weaknesses.

BACKGROUND AND DISCUSSION

This report fulfills the requirement of 10 CFR 54.25 which states that the ACRS review and report on license renewal applications. Peach Bottom Units 2 and 3 are General Electric boiling water reactors (BWRs) Type 4, with Mark I containments. Exelon requested renewal of their operating licenses for 20 years beyond the current license terms, which expire on August 8, 2013 for Unit 2 and July 2, 2014 for Unit 3. Peach Bottom Unit 1 is on the same site as Units 2 and 3. It is permanently shutdown and in SAFSTOR condition. There are no systems shared between Unit 1 and Units 2 and 3.

The final SER documents the staff's review of the information submitted by Exelon, including commitments that were necessary to resolve open items identified by the staff in the initial SER. Peach Bottom is the second BWR plant to seek license renewal and the first to use a system-based approach to identify structures, systems, and components (SSCs) that should be included in the scope of license renewal. The staff reviewed the completeness of the applicant's identification of SSCs that are subject to aging management; the integrated plant assessment process; the identification of the possible aging mechanisms associated with passive, long-lived components; and the adequacy of the aging management programs. The staff also conducted several inspections at Exelon's engineering offices and the Peach Bottom site to verify the adequacy of the methodology described in the application and its implementation.

During our Plant License Renewal Subcommittee meeting on October 30, 2002, the staff presented a well-structured and effective overview of its inspections. As in other applications, the review of the Peach Bottom license renewal application required a substantial number of requests for additional information (RAIs) and depended heavily on review of plant drawings at the site.

On the basis of our review of the final SER, we agree with the staff's conclusion that all open items and confirmatory items have been appropriately closed, and there are no issues that would preclude renewal of the operating licenses for Peach Bottom Units 2 and 3. We also concur with all four license conditions requiring the applicant to take certain actions before beginning the period of extended operation.

The process implemented by the applicant to identify SCCs that are within the scope of license renewal has been effective. The applicant included portions of nonsafety-related systems in the scope of license renewal if their failure could impact in-scope safety-related systems. When a system met this criterion, the entire system, passing through seismic Class I structures, was considered in scope. Portions of these systems that run through non-seismic structures were evaluated by walkdowns and were added to the scope as appropriate. An example of such a system is the service water system that could spray liquid on the safety systems.

Certain nonsafety systems have portions that perform a safety function, and the applicant realigned these portions to be included as part of the in-scope safety system. For example, a nonsafety-related system such as chilled water or instrument air that penetrates the containment has been realigned to be considered in scope as a part of the containment pressure retaining function. The in-scope portions of the realigned system typically include the first valve outside and inside containment and all of the piping in between.

Peach Bottom is located on the Susquehanna River on a large pond created by the Conowingo Dam (also owned by Exelon). Peach Bottom relies on the pond for operation of the units, but does not depend on the pond for emergency service water. It does depend, however, on power from Conowingo for station blackout (SBO) via a submerged electrical cable. Consequently, Conowingo is in scope for SBO considerations. The license for the Conowingo Dam will expire before the extended license period for the Peach Bottom Plant and is expected to be renewed. Should this not occur, other provisions for SBO will be required.

Open items have been closed by bringing all identified SSCs into scope. During our review, we questioned why certain other SSCs were not included in scope and, in all cases, the applicant provided appropriate justification for their exclusion. We conclude that the applicant and the staff have appropriately identified all SSCs that are within the scope of license renewal.

The applicant also performed a comprehensive aging management review of all SSCs that are within the scope of license renewal. The application describes 34 aging management programs for license renewal, which include existing, augmented, and new programs.

The applicant has proposed to inspect only the refueling water storage tank and infer from that inspection the condition of the condensate storage tank. Since these storage tanks are similar in construction, are exposed to similar water chemistry, and are located in similar environments, we agree with the staff that this is an acceptable approach.

Peach Bottom Units 2 and 3 have toroidal suppression pools and there was discussion regarding the material condition of the coating and steel. The applicant satisfactorily described inspections conducted to date to ensure the quality of material condition of the coating and steel and also described plans for future inspections.

There was a concern that the applicant did not appear to have an aging management program for the buried portions of the standby gas treatment system (SGTS) ductwork. The applicant stated that the ductwork was either hot and/or insulated and no aging management program was required. During the third license renewal inspection at Peach Bottom, the inspectors visually examined accessible exterior and interior surfaces of the SGTS and found no age-related degradation. Based on the results of this inspection, the staff agreed with the applicant.

Peach Bottom has had a history of cable failure due to moisture intrusion in 4Kv and 13Kv service. Many cables have been replaced with moisture-resistant cables. In recent NRC inspections, water intrusion was evident in certain manholes and seems to be an ongoing problem. Consequently, the applicant committed to a program to manage the aging of inaccessible medium-voltage cables. This aging management program provides reasonable assurance that the intended functions of the systems and components will be maintained consistent with the current licensing basis during the period of extended operation.

With regard to the inspection of reactor vessel internals, the applicant has committed to the programs prescribed in 15 BWR Vessel and Internals Project (BWRVIP) reports. These programs have all been approved by the NRC staff for 60 year plant life except those described in BWRVIP-78, BWR Integrated Surveillance Program, and BWRVIP-86, BWR Integrated Surveillance Program Implementation Plan, which are approved only for 40 year plant life. The staff is currently reviewing these BWRVIP reports for 60 years. The applicant has agreed to a license condition to notify the NRC, before entering the period of extended operation, of its decision to implement either the staff-approved integrated surveillance program (ISP) or a staff-approved plant-specific ISP. Also, the staff has not yet approved BWRVIP-76, "BWR Core Shroud Inspection and Flaw Evaluation Guidelines." Because the staff's review is not complete, the applicant has agreed to another license condition to notify the NRC of its decision to implement either the staff-approved core shroud inspection and evaluation guidelines program, or a staff-approved plant-specific program.

Exelon has also identified those components at Peach Bottom that are supported by time-limited aging analyses (TLAAs). These TLAAs show that the components analyzed have sufficient margin to operate for the period of extended life.

Peach Bottom Unit 2 experienced a scram on December 21, 2002. This event highlighted a number of weaknesses in the current corrective action and preventive maintenance programs. We expect that ongoing corrective actions committed by the licensee will resolve these weaknesses. During inspections, the staff should assess the effectiveness as well as the adequacy of implementation of these programs.

The applicant and the staff have identified plausible aging effects associated with passive, long-lived components. Adequate programs have been established to manage the effects of aging so that Peach Bottom Units 2 and 3 can be operated in accordance with their current licensing bases for the period of extended life without undue risk to the health and safety of the public.

Sincerely,

Mario V. Bonaca

Mario V. Bonaca
Chairman

References:

1. Letter dated July 2, 2001, from J. A. Benjamin, Exelon Generation Company, LLC, to U. S. Nuclear Regulatory Commission, transmitting Application to Renew the Operating Licenses of Peach Bottom Atomic Power Station Units 2 and 3
2. U.S. Nuclear Regulatory Commission, NUREG-XXX, "Safety Evaluation Report Related to the License Renewal of Peach Bottom Atomic Power Station, Units 2 and 3" February, 2003.

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

April 15, 2003

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: CLOSEOUT OF GENERIC SAFETY ISSUE 168, "ENVIRONMENTAL QUALIFICATION OF LOW-VOLTAGE INSTRUMENTATION AND CONTROL CABLES"

During the 501st meeting of the Advisory Committee on Reactor Safeguards, April 10-12, 2003, the Committee considered the staff's proposed closeout of Generic Safety Issue 168, "Environmental Qualification of Low-voltage Instrumentation and Control Cables" and decided not to perform any further review of this subject. The Committee agrees with the staff's plans to issue the proposed generic communication.

Reference:
Memorandum dated March 7, 2003, from Jose Calvo, Office of Nuclear Reactor Regulation, to John T. Larkins, Executive Director, ACRS, Subject: Closeout of Generic Safety Issues 168, "Environmental Qualification of Low-voltage Instrumentation and Control Cables"

cc: A. Vietti-Cook, SECY
I. Schoenfeld, OEDO
S. Collins, NRR
W. Bateman, NRR
J. Calva, NRR
T. Koshy, NRR

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

April 15, 2003

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: DRAFT REGULATORY GUIDE DG-1101, "SITE INVESTIGATIONS FOR FOUNDATIONS OF NUCLEAR POWER PLANTS" (DRAFT FINAL REVISION 2 TO REGULATORY GUIDE 1.132)

During the 501st meeting of the Advisory Committee on Reactor Safeguards, April 10-12, 2003, the Committee considered the final draft Regulatory Guide DG-1101, "Site Investigations for Foundations of Nuclear Power Plants" and decided not to review this document. The Committee agrees that the staff should continue with its process for issuing the final draft DG-1101.

Reference:
Memorandum dated April 10, 2003, from Michael Mayfield, to John T. Larkins, Executive Director, ACRS, Subject: Draft Regulatory Guide DG-1101 (Draft Final Revision 2 to Regulatory Guide 1.132), "Site Investigations for Foundations of Nuclear Power Plants"

cc: A. Vietti-Cook, SECY
I. Schoenfeld, OEDO
A. Thadani, RES
M. Mayfield, RES
A. Hsia, RES
Y. Li, RES

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

April 17, 2003

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

SUBJECT: PROPOSED NRC GENERIC LETTER 2003-XX: CONTROL ROOM HABITABILITY

Dear Chairman Diaz:

During the 501st meeting of the Advisory Committee on Reactor Safeguards, April 10-12, 2003, we discussed a proposed NRC Generic Letter on control room habitability with representatives of the NRC staff and representatives of the Nuclear Energy Institute (NEI) and its Control Room Habitability Task Force. Our discussions were facilitated by the documents referenced in this report.

RECOMMENDATIONS AND CONCLUSIONS

- The proposed NRC Generic Letter 2003-XX: Control Room Habitability should be issued.
- The NRC should consider using the Human Factors Research Program to develop quantitative information on potential performance degradation when control rooms are contaminated with smoke or operators are wearing special protective equipment.

DISCUSSION

Testing at 30 nuclear power plants has shown that unfiltered inleakage to control rooms in the great majority of cases exceeds, often substantially, the inleakage assumed in plant safety analyses. The tested installations have had to repair facilities and revise analyses to comply with their licensing basis.

In light of evidence from tests done to date, the staff believes many other licensees will find actual inleakage into their control rooms substantially higher than assumed in safety analyses. The staff has prepared a Generic Letter to alert licensees to the test findings and to request licensees to demonstrate that the control rooms at their facilities comply with the current licensing and design bases. The staff has prepared this Generic Letter following consultation with the nuclear industry and public meetings in each of the NRC's Regions. We believe the Generic Letter should be issued.

The Generic Letter requests that licensees provide the requested information within 180 days or request within 60 days an extended time for response. It is likely that many licensees will find it necessary to conduct tests to verify assumptions concerning control room inleakage. Such testing often takes about two weeks to conduct. Because testing resources are limited, scheduling of tests may make it necessary for licensees to delay responses well beyond the 180 days specified in the Generic Letter.

A further complication licensees may encounter in responding to the requests in the Generic Letter is the evolving nature of control room habitability guidance available to licensees. The staff has developed four draft regulatory guides pertinent to control room habitability. We have not reviewed these draft regulatory guides in detail. We do understand that some of the guides endorse portions of guidance in the NEI document, NEI 99-03 Rev. 0. In March 2003, NEI has issued a substantially revised version, NEI 99-03 Rev. 1. We encourage plans by staff and NEI to hold workshops to clarify the guidance that the licensees can adopt to respond to requests in the Generic Letter.

Guidance developed by the staff and by NEI has addressed appropriate operator responses should control rooms be contaminated by smoke, hazardous chemicals, or radioactive materials. There is, however, very little data on the potential degradation of operator performance within control rooms in the event of contamination or when operators are forced to wear protective equipment, such as self-contained breathing apparatus. The NRC should consider using the Human Factors Research Program to develop quantitative information on the potential performance degradation to facilitate staff reviews of licensees' plans and proposals in these areas.

We would appreciate a further briefing by the staff on the control room habitability issues once licensees have responded to the requests of the Generic Letter and staff has reviewed and analyzed the responses.

Sincerely yours,

Mario V. Bonaca

Mario V. Bonaca
Chairman

References:
1. U.S. Nuclear Regulatory Commission, NRC Generic Letter 2003-XX: "Control Room Habitability," March 2003.
2. U.S. Nuclear Regulatory Commission, Regulatory Guide DG-1111, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," Revision 1, December 2002.
3. U.S. Nuclear Regulatory Commission, Regulatory Guide DG-1113, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors," December 2002.

4. U.S. Nuclear Regulatory Commission, Regulatory Guide DG-1114, "Control Room Habitability at Light-Water Nuclear Power Reactors," February 2003.

5. U.S. Nuclear Regulatory Commission, Regulatory Guide DG-1115, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," March 2003.

6. Letter dated March 11, 2003, from Alexander Marion, NEI, to F. Mark Reinhart, NRC, transmitting NEI 99-03, Revision 1, "Control Room Habitability Guidance."

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

April 21, 2003

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

SUBJECT: DRAFT FINAL RISK-INFORMED REVISION TO 10 CFR 50.44, "COMBUSTIBLE GAS CONTROL IN CONTAINMENT"

Dear Chairman Diaz:

During the 501st meeting of the Advisory Committee on Reactor Safeguards, April 10-12, 2003, we reviewed a draft final rulemaking package for a risk-informed revision to Title 10, Section 50.44, of the *Code of Federal Regulations* (10 CFR 50.44), "Combustible Gas Control in Containment." During our review, we had the benefit of discussions with representatives of the NRC staff. We also had the benefit of the documents referenced.

Recommendation

The Commission should approve the proposed rule for a risk-informed revision to 10 CFR 50.44.

Discussion

In a Staff Requirements Memorandum (SRM) dated February 3, 2000, the Commission approved proceeding with the plan for risk-informing the technical requirements of 10 CFR Part 50. Section 50.44 was selected as a trial case for risk-informing 10 CFR Part 50. In Attachment 2 to SECY-00-0198, dated September 14, 2000, the staff assessed the risk-significance of combustible gas control for the various types of containments. In our report dated September 13, 2000, we concluded that this work provided the basis for developing a risk-informed revision to 10 CFR 50.44 that could provide a safety benefit while reducing unnecessary burden for licensees. We therefore recommended that the Commission should direct the staff to proceed with rulemaking.

In our letter dated December 12, 2001, we concluded that the proposed rule would provide effective and efficient regulation to deal with combustible gases in containments. We requested an opportunity to review the proposed final rule after reconciliation of public comments.

The staff provided us with its draft final rule language, including reconciliation of public comments, and associated documents on March 14, 2003. The draft final rule retains requirements for (i) hydrogen control systems for Mark III and ice condenser containments,

(ii) inerting Mark I and Mark II containments, and (iii) ensuring a mixed atmosphere in the containment. It also retains the requirement to monitor hydrogen in the containment atmosphere for all containment designs, but it no longer classifies monitors as safety-related components. The draft final rule also codifies the existing regulatory practice of monitoring oxygen concentrations in containments with inerted atmospheres. In addition, it relocates the current requirements for high point vents to a new section identified as 10 CFR 50.46a. The draft final rule eliminates the current design-basis loss-of-coolant accident hydrogen release and requirements for hydrogen recombiners and purge systems to mitigate such a release. It also deletes the requirement prohibiting licensees from venting the reactor coolant system if it could "aggravate" a challenge to containment.

The draft final rule will provide effective and efficient regulation to deal with combustible gases in containments and should be approved.

Sincerely,

Mario V. Bonaca
Chairman

References:

1. Letter dated March 14, 2003, from Christopher I. Grimes, Office of Nuclear Reactor Regulation, NRC, to John T. Larkins, Executive Director, ACRS, Subject: Request for Review of Final Part 50 Rulemaking on Risk-Informed Revision of Combustible Gas Control (Predecisional).
2. Memorandum dated February 3, 2000, from Annette Vietti-Cook, Secretary of NRC, to William D. Travers, Executive Director for Operations, NRC, Subject: Staff Requirements - SECY-99-264 - Proposed Staff Plan for Risk-Informing Technical Requirements in 10 CFR Part 50.
3. SECY-00-0198, Memorandum for the Commissioners, from William D. Travers, Executive Director for Operations, NRC, dated September 14, 2000, Subject: Status Report on Study of Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50 (Option 3) and Recommendations on Risk-Informed Changes to 10 CFR 50.44 (Combustible Gas Control).
4. Letter dated December 12, 2001, from George E. Apostolakis, Chairman, ACRS, to William D. Travers, Executive Director for Operations, NRC, Subject: Proposed Rulemaking for Risk-Informed Revisions to 10 CFR 50.44, "Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors."
5. Letter dated September 13, 2000, from Dana A. Powers, Chairman, ACRS, to Richard A. Meserve, Chairman, NRC, Subject: Proposed Risk-Informed Revisions to 10 CFR 50.44, "Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors."

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

April 21, 2003

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

Dear Chairman Diaz:

SUBJECT: PROPOSED RESOLUTION OF PUBLIC COMMENTS ON DRAFT REGULATORY GUIDE DG-1122, "AN APPROACH FOR DETERMINING THE TECHNICAL ADEQUACY OF PROBABILISTIC RISK ASSESSMENT RESULTS FOR RISK-INFORMED ACTIVITIES"

During the 501st meeting of the Advisory Committee on Reactor Safeguards, April 10-12, 2003, we met with representatives of the NRC staff and the Nuclear Energy Institute (NEI) to discuss the NRC staff's proposed resolution of public comments received in regard to Draft Regulatory Guide (DG)-1122, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities." We also had the benefit of the documents referenced.

Recommendations

1. The draft final Regulatory Guide should include definitions of the terms "dominant," "important," "key," and "significant."

2. The peer review of the probabilistic risk assessments (PRAs) should include an assessment of the uncertainties and the validity of key assumptions.

3. The draft final Regulatory Guide should include guidance on how to perform sensitivity and uncertainty analyses.

4. To ensure consistency, the draft final Regulatory Guide should prescribe a minimum list of topics to be included in the peer review.

5. The staff needs to clarify how the Capability Categories are consistent with the provision in the Regulatory Guide that the event probabilities reflect the actual operating history and experience of the plant as well as applicable generic experience.

6. The staff should provide guidance on acceptable qualitative characterization of risk contributions not calculated in limited-scope PRAs.

Discussion

Ever since the Commission started its initiative to risk-inform the regulations, the quality of risk information that is input to the integrated decisionmaking process has been a subject of debate. To help the staff evaluate the quality of submitted PRAs in a timely manner, the American Society of Mechanical Engineers (ASME) has issued a standard for PRAs for "internal" accident initiators and the industry has developed a peer review process. DG-1122 and the associated Standard Review Plan (SRP) Chapter 19.1 document the regulatory position regarding these efforts.

DG-1122 provides guidance to licensees in four areas:

- A minimal set of functional requirements of a technically acceptable PRA.
- NRC position on consensus PRA standards and industry PRA program documents.
- Demonstration that the PRA (*in toto* or specific parts) used in regulatory applications is of sufficient technical adequacy.
- Documentation that the PRA (*in toto* or specific parts) used in regulatory applications is of sufficient technical adequacy.

The staff has received a large number of comments from ASME and the industry, most of which have been resolved. The true test of the usefulness of this Regulatory Guide is to subject it to pilot applications. We believe that several issues must be resolved before issuing a draft final Regulatory Guide for trial use so that better insights can be obtained.

ASME and NEI disagree with three staff positions. These positions deal with the definition of terms such as "dominant" sequences or events, the assessment by the peer reviewers of key assumptions, and the minimum list of topics that the peer review process should include.

The ASME standard provides an ambiguous definition of "dominant" and uses the term interchangeably with "significant" and "key." This term is critical to the application of the standard because it determines whether certain requirements are imposed and it is part of the definitions of the Capability Categories. ASME and the industry disagree with the staff's proposal to test a quantitative definition of the term.

As stated above, the purpose of the standard and the peer review process is to assist the staff in determining the quality of risk information used in particular regulatory applications. The staff's review of licensee applications will be eased if there is common understanding of key concepts. We believe that clear definitions of the terms "dominant," "important," "key," and "significant" should be included in the draft final Regulatory Guide before issuing it for trial use.

PRAs rely on numerous assumptions that are often critical to the validity of the results. Although the ASME standard requires that the key assumptions be identified, it does not require the peer reviewers to assess the validity of these assumptions. We agree with the staff that such an assessment should be required.

The ASME standard provides a list of PRA "suggestions" that the reviewers should consider in their review. These are not intended to be either a minimum or a comprehensive list of requirements. The staff argues that these suggestions should, in fact, be requirements; otherwise consistency in the reviews cannot be ensured. We agree.

In our report dated July 23, 2002, we recommended that proposed Revision 1 to Regulatory Guide 1.174 and SRP Chapter 19 state that changes to the licensing basis would, in general, require PRAs that conformed at least to Category II of the ASME standard and a Grade 3 of the industry peer review process.

While DG-1122 does not explicitly state that PRAs should conform at least to Category II of the ASME standard, it does state that the PRA model represent the as-built and as-operated plant, and that the event probabilities reflect the actual operating history and experience of the plant and applicable generic experience. It is not clear how this can be consistent with Category I of the ASME standard. The staff needs to clarify how the Capability Categories are consistent with these requirements. Similar clarification regarding the grades of the peer review process specified in NEI 00-02 should be made.

DG-1122 correctly states that understanding the relevant uncertainties is an essential element of risk characterization. A systematic treatment should include rigorous analyses for parametric uncertainties, sensitivity studies to identify the important epistemic uncertainties, and quantification of the latter. In a risk-informed environment, the proper role of sensitivity studies is to identify what is important to the results, not to replace uncertainty analyses. The staff should include guidance in the draft final Regulatory Guide regarding sensitivity and uncertainty analyses.

DG-1122 states that, for many applications that involve total plant risk, the risk characterization should account for all operating states and initiating events either quantitatively or qualitatively. More guidance is needed on this subject.

We would like to review the draft final version of DG-1122 before issuing our letter on its trial use.

Sincerely,

Mario V. Bonaca

Mario V. Bonaca
Chairman

References:

1. Letter dated April 4, 2003, from Scott F. Newberry, Nuclear Regulatory Research, NRC, to John T. Larkins, Executive Director, ACRS, Subject: ACRS Review of Draft Guide-1122, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," and the Associated Standard Review Plan Chapter 19.1.

2. U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Draft Regulatory Guide DG-1122, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," November 2002.
3. American Society of Mechanical Engineers, ASME RA-S-2002, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," dated April 5, 2002.
4. Nuclear Energy Institute, NEI-00-02, "Probabilistic Risk Assessment Peer Review Process Guidance," Revision A3, dated March 20, 2002.
5. Letter dated July 23, 2002, from G. E. Apostolakis, Chairman, ACRS, to Richard A. Meserve, Chairman, NRC, Subject: Draft Final Revision 1 to Regulatory Guide 1.174 and to Chapter 19 of the Standard Review Plan.
6. Letter dated April 8, 2003, from Dr. Sidney A. Bernsen, Chairman, ASME Committee on Nuclear Risk Management, to Dr. Mario V. Bonaca, Chairman, ACRS regarding reconciliation of ASME PRA Standard with DG-1122.

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

April 22, 2003

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: DRAFT FINAL AMENDMENT TO 10 CFR 50.55a, "CODES AND STANDARDS"

During the 501st meeting of the Advisory Committee on Reactor Safeguards, April 10-12, 2003, the Committee considered the draft final amendment to 10 CFR 50.55a to incorporate by reference the following Regulatory Guides that list code cases published by the American Society of Mechanical Engineers (ASME) and approved by the NRC staff:

- Regulatory Guide 1.84, Revision 32 (DG-1090), "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III"
- Regulatory Guide 1.147, Revision 13 (DG-1091), "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1"
- Regulatory Guide 1.192 (DG-1089), "Operation and Maintenance Code Case Acceptability, ASME OM Code"

In addition, the Committee considered Regulatory Guide 1.193 (DG-1112), "ASME Code Cases Not Approved for Use," which is not referenced in the draft final amendment to 10 CFR 50.55a.

The Committee decided not to review these documents and agrees with the staff's proposal to issue these documents.

References:
1. Letter dated March 20, 2003, from Christopher I. Grimes, NRR, to John T. Larkins, ACRS, transmitting the final amendment to 10 CFR 50.55a.
2. Memorandum dated March 20, 2003, from Michael E. Mayfield, RES, to John T. Larkins, ACRS, transmitting the following Regulatory Guides:
    - Regulatory Guide 1.84, Revision 32 (DG-1090), "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III."
    - Regulatory Guide 1.147, Revision 13 (DG-1091), "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1."

- Regulatory Guide 1.192, Revision 0 (DG-1089), "Operation and Maintenance Code Case Acceptability, ASME OM Code."
- Regulatory Guide 1.193, Revision 0 (DG-1112), "ASME Code Cases Not Approved for Use."

cc: A. Vietti-Cook, SECY
W. Dean, OEDO
I. Schoenfeld, OEDO
A. Thadani, RES
M. Mayfield, RES
W. Norris, RES
S. Collins, NRR
C. Grimes, NRR
A. Tovmassian, NRR

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

April 29, 2003

Dr. William D. Travers
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: NUREG-CR-6813, "ISSUES AND RECOMMENDATIONS FOR ADVANCEMENT OF PRA TECHNOLOGY IN RISK-INFORMED DECISION MAKING"

Dear Dr. Travers:

We have undertaken an effort to assess the agency's needs for improved Probabilistic Risk Assessment (PRA) technology to risk inform its regulations. As part of this effort, we commissioned Karl N. Fleming of Technology Insights to prepare the attached report on issues whose resolution would increase the use of risk information in regulatory decisions. This report is based on the author's extensive experience as a practitioner and a participant in the development of the American Society of Mechanical Engineers (ASME) Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications. In addition, Mr. Fleming conducted interviews with PRA practitioners and decision makers from NRC staff and selected industry representatives. This report has been published as NUREG/CR-6813, "Issues and Recommendations for Advancement of PRA Technology in Risk-Informed Decision Making."

Based on the information gathered during the interviews, his reviews of a number of risk-informed initiatives, and the experience in performing and reviewing PRAs, the author identified a set of recurrent issues that arise in the use of PRAs for risk-informed decision making. Obviously any such list only represents a "snapshot" at a particular time, since many of the issues are being addressed in ongoing activities such as the standards development, the industry peer-review process, the NRC coherence program, and the development of Draft Regulatory Guide DG-1122, "Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities."

The attached report groups the identified issues into the following general categories:

- Use of limited-scope PRAs in risk-informed applications submitted in accordance with Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," to quantify full-scope metrics
- Lack of completeness within the specified scope
- Model-to-plant fidelity issues
- Lack of, or inadequate, treatment of uncertainties
- Quantification issues (e.g., error due to cut-set truncation)
- Multi-unit site modeling issues
- Lack of treatment of aging effects
- Issues with the use and interpretation of risk metrics
- Lack of coherence between probabilistic and deterministic safety approaches

The interviews conducted by the author identified completeness as the most important area. Issues associated with completeness include the following notable examples:

- Lack of criteria for and consistency in evaluating the impact of missing elements in scope on the application of RG-1.174
- Lack of acknowledgment or consideration of limitations in the PRAs used in submittals
- Inadequate justification and documentation for screening events from a PRA
- Lack of incorporation of operating experience in PRAs
- Inadequate treatment of common-cause failures
- Lack of detailed review by plant personnel to ensure fidelity with plant systems, operator actions, etc.

The author also makes the observation that while valid technical arguments can be made to justify the exclusion of portions of a full-scope PRA model for some applications, resources must be continually expended by both the NRC and its licensees to determine the validity of decisions that are based on an incomplete model. The author further notes that at some point it becomes reasonable to ask whether these burdens are comparable to the effort needed to develop a full-scope PRA.

We believe that this report will serve as a useful resource in the agency's ongoing effort to risk inform its regulations.

Sincerely,

Mario V. Bonaca

Mario V. Bonaca
Chairman

Attachment: NUREG/CR-6813, "Issues and Recommendations for Advancement of PRA Technology in Risk-Informed Decision Making," April 2003.

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

May 16, 2003

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

SUBJECT: IMPROVEMENT OF THE QUALITY OF RISK INFORMATION FOR REGULATORY DECISIONMAKING

Dear Chairman Diaz:

In a March 31, 2003, Staff Requirements Memorandum (SRM) on risk-informed changes to 10 CFR 50.46, the Commission stated that "the PRA should be a level 2 internal- and external-initiating event all mode PRA, which has been subjected to a peer review process and submitted to and endorsed by the NRC." Similarly, in an SRM dated March 28, 2003, the Commission directed the staff to "ask for specific comment in the Statements of Consideration on whether NRC should amend 50.69(c)(1)(i) to require a comprehensive high quality PRA. For example, this PRA should be a level 2 internal- and external-initiating event all mode PRA, which has been subjected to a peer review process and submitted to and endorsed by the NRC."

In this report, we focus on several aspects of Probabilistic Risk Assessment (PRA) methodology and practice that need to be addressed to achieve such comprehensive high-quality PRAs. We limit our discussion to the PRA methodology needed for the calculation of core damage frequency (CDF) and the estimation of large early release frequency (LERF) consistent with Regulatory Guide (RG) 1.174 and do not address issues unique to Level 2 PRA. We have had the benefit of the results of a study performed for us by K.N. Fleming of Technology Insights (Reference 1), as well as of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

1. Completeness of risk information requires that PRAs address low-power and shutdown (LPSD) modes and "external" events, such as fires and earthquakes, in addition to power operations.

2. Guidance should be developed on how licensees and peer-review teams should consider operating experience in order to improve PRA completeness.

3. The assessment of uncertainties should address model uncertainties. Guidance for the quantitative evaluation of model uncertainties should be developed.

DISCUSSION

Reference 1 presents the results of about 20 interviews with members of the NRC staff and selected representatives of the nuclear industry. The NRC staff members included senior management and staff from the Office of Nuclear Regulatory Research (RES) and the Office of Nuclear Reactor Regulation (NRR). The subject of the interviews was risk-informed decisionmaking.

The study found that most staff interviewees believe that the reluctance of the industry to improve the scope and quality of the PRAs is a major impediment to the advancement of risk-informed regulation. The areas of difficulty include both the use of limited-scope PRAs and the lack of completeness within a specified scope. Even for risk contributors that were treated, incompleteness of treatment was cited as an issue.

A further observation of Reference 1 is that, while valid technical arguments can be made to justify limited-scope PRA model for some applications, resources must be expended by both the licensee and the NRC to determine the validity of decisions that are based on an incomplete model. It is reasonable to ask whether these burdens are comparable to the effort needed to develop a full-scope PRA.

Our review of safety evaluations of licensee risk-informed submittals has revealed that the staff does include consideration of all modes of operation as well as "external" events. When the licensees submit incomplete PRAs (e.g., missing the LPSD part) or use bounding analyses, typically for some external events, the staff has to account for the missing PRA elements subjectively, as allowed by the "integrated decisionmaking process" of RG 1.174 (Reference 2).

These subjective evaluations do not necessarily lead to conservative decisions. Reference 1 points out that, when bounding analyses are used for external events, some risk contributors may not be identified. For example, there are some risk-significant sequences that involve combinations of failures from fires and other events independent of the fire, i.e., a fire may disable one train of a safety system and another train may be unavailable due to other causes. It is unlikely that a bounding analysis for fires would identify such sequences.

Certain risk-informed applications, e.g., risk informing the special treatment requirements require the use of importance measures (e.g., Fussell-Vesely and Risk Achievement Worth). These are global measures of risk that are strongly affected by the scope and quality of the PRA. As stated in our report dated February 11, 2000 (Reference 3), incomplete assessments of risk contributions from LPSD operations, fires, and human performance distort the importance measures, undermining confidence in the risk categorization of structures, systems, and components (SSCs).

All-mode PRAs permit the risk characterization of SSCs that are used only in shutdown or low-power modes, such as components of residual heat removal systems. In addition, all-mode PRAs facilitate cycle risk optimization. For example, by comparing the risk contributions of diesel generator maintenance during shutdown and during operation, plants with internal events PRAs and LPSD PRAs have shown that on-line diesel generator maintenance reduces overall cycle risk, even though it may slightly increase risk during power operation.

In addition to the PRA scope, completeness also refers to the set of accident sequences within scope. Reference 1 notes that, in general, PRAs do not make use of experience gained over the years in identifying sequences that should be analyzed. In addition, operating experience should be reviewed.

As noted in our report dated October 11, 2000 (Reference 4), RES has been issuing reports that contain evaluations of actual plant performance in terms of initiating-event frequencies and reliabilities of critical plant systems, as well as comparisons with corresponding data used in PRAs. Augmented Inspection Team reports provide detailed evaluations of major incidents. The Accident Sequence Precursor (ASP) program identifies significant accident sequences that actually have occurred and draws relevant conclusions. Generic Safety Issues (GSIs) are an additional source of information that should be considered in upgrading PRAs.

Unfortunately, this wealth of useful information does not appear to be widely used by PRA practitioners. Reference 1 suggests that as many as 20% of events evaluated by the ASP program involve initiating events and accident sequences not modeled in existing PRAs. Although PRAs use the statistical information from past experience in the estimation of failure rates, the sequences of events that actually have occurred are not generally utilized. The reasonableness of PRA results is often judged by comparing them with the results of other PRAs for similar plants. Although such comparisons are useful, we believe that analyses of operating experience such as the RES reports should be utilized to a greater extent. The staff should prepare guidance to the licensees and peer-review teams to make sure that PRAs benefit from this experience.[1]

The Reactor Safety Study (Reference 5) developed probability distributions for parameters such as failure rates and initiating-event frequencies. This precedent, combined with the fact that parameter uncertainties are easier to deal with than model uncertainties, has led to the unfortunate, yet widely held, belief that uncertainty analysis is synonymous with parameter uncertainty evaluation. In addition, it has been found that the principal PRA results are fairly insensitive to parameter uncertainties,[2] thus leading to the belief that quantifying such uncertainties is an unnecessary burden.

However, models that are included in the PRAs can be important sources of uncertainty. For example, there are several models for human performance during accidents that are based on different assumptions and analytical approaches. Human reliability experts have not yet reached consensus on what assumptions are appropriate. Using only one of these models yields results whose uncertainties are unknown, since the use of another model could yield different results. Yet this model uncertainty is rarely considered.

The Ispra Research Center of the European Union organized a benchmark exercise in which

---

[1] We note that in the SRM dated March 28, 2003, the Commission directs that "relevant operational experience should be evaluated in an ongoing manner with the aim of reducing the uncertainty in assessing the effect of treatment on reliability and common-cause failures."

[2] A notable exception is the case of significant correlations between broad epistemic distributions (Reference 6). These have had an impact on the frequency of interfacing-system loss-of-coolant accidents (Reference 7).

15 teams from 11 countries used a number of human reliability analysis (HRA) models available at the time to estimate the probability of the crew not responding correctly to a transient (Reference 8). The results produced by the teams using the same HRA model differed by orders of magnitude. The results produced by a single team using a number of HRA models also differed by orders of magnitude. Although these results are fairly old now, we believe that they are still representative of the model uncertainties present in HRA.

Several other examples of the impact of model uncertainties are presented in Reference 9. In one PRA, the dominant model uncertainties resulted from the reactor coolant pump (RCP) seal loss-of-coolant accident (LOCA) timing and operator recovery possibilities. In another, they were due to the RCP seal LOCA timing again and the heating, ventilation, and air conditioning (HVAC) success criteria. The authors stated that, in all cases, the CDF was affected significantly by these uncertainties.

The staff has recognized that model uncertainty must be addressed by decisionmakers. Draft Regulatory Guide DG-1122 (Reference 10) includes the following statement in its description of the technical elements of a PRA: "The sensitivity of the model results to model boundary conditions and other key assumptions is evaluated using sensitivity analyses to look at key assumptions both individually and in logical combinations." RG 1.174 states that uncertainties due to incompleteness and model assumptions should be evaluated.

Most licensees have not included a systematic treatment of uncertainties in their PRAs. A systematic treatment would include analyses of parametric uncertainties, sensitivity studies to identify the important model uncertainties, and quantification of the latter.

Tools for performing analyses of parametric uncertainties are readily available and are included in most of the widely used PRA software. The disciplined use of sensitivity studies to address model uncertainties is not as well understood. Developing guidance for quantifying model uncertainty is not infeasible. Such an effort would build on past practice and the literature. For example, NUREG-1150 (Reference 11) quantified the probabilities of alternative assumptions in severe accident assessments by eliciting expert opinions. Since NUREG-1150, other methods have been developed that are not as resource intensive (References 9 and 12). Furthermore, RES has sponsored a workshop in which a number of ideas and methods for handling model uncertainties have been proposed and debated (Reference 13).

More guidance regarding sensitivity and uncertainty analyses would contribute greatly to confidence in risk-informed regulatory decisionmaking. Such guidance should include a clear discussion of the roles of sensitivity and uncertainty analyses, as well as practical procedures for performing these analyses. It should address not only how uncertainties should be treated in the PRA, but, also, how they impact decisionmaking with examples to show the pitfalls if uncertainties are inadequately addressed.

Sincerely,

Mario V. Bonaca
Chairman

<u>References</u>:

1. U. S. Nuclear Regulatory Commission, NUREG/CR-6813, "Issues and Recommendations for Advancement of PRA Technology in Risk Informed Decision Making," Technology Insights, April 2003.
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 Current Licensing Basis," June 1998.
3. Report dated February 11, 2000, from Dana A. Powers, Chairman, ACRS, to Richard A. Meserve, Chairman, U. S. Nuclear Regulatory Commission, Subject: Importance Measures Derived from Probabilistic Risk Assessments.
4. Report dated October 11, 2000, from Dana A. Powers, Chairman, ACRS, to Richard A. Meserve, Chairman, U. S. Nuclear Regulatory Commission, Subject: Union of Concerned Scientists Report, "Nuclear Plant Risk Studies: Failing the Grade."
5. U. S. Nuclear Regulatory Commission, WASH-1400, "Reactor Safety Study, An Assessment of Accident Risks in U. S. Nuclear Power Plants," (NUREG-75/014), October 1975.
6. G. Apostolakis and S. Kaplan, "Pitfalls in Risk Calculations," *Reliability Engineering*, Vol. 2, pp. 135-145, 1981.
7. Pickard, Lowe, and Garrick, Inc., Westinghouse Electric Corporation, Fauske & Associates, Inc., "Zion Probabilistic Safety Study," prepared for Commonwealth Edison Company, Chicago, 1981.
8. A. Poucet, "The European Benchmark Exercise on Human Reliability Analysis," Presented at the American Nuclear Society International Topical Meeting on Probability, Reliability, and Safety Assessment, PSA '89, Pittsburgh, PA, April 2-7, 1989.
9. D. Bley, S. Kaplan, and D. Johnson, "The Strengths and Limitations of PSA: Where We Stand," *Reliability Engineering and System Safety,* Vol. 38, pp. 3-26, 1992.
10. U. S. Nuclear Regulatory Commission, Draft Regulatory Guide DG-1122, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," November 2002.
11. U. S. Nuclear Regulatory Commission, NUREG-1150, "Severe Accident Risks: An Assessment for Five US Nuclear Power Plants," December 1990.
12. U. S. Nuclear Regulatory Commission, NUREG/CR-6372, Volumes 1 and 2, "Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and Use of Experts," April 1997.
13. U. S. Nuclear Regulatory Commission, NUREG/CP-0138, "Proceedings of Workshop on Model Uncertainty: Its Characterization and Quantification," Annapolis, MD, October 20-22, 1993, Editors: A. Mosleh, N. Siu, C. Smidts, and C. Lui, 1994.

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

May 16, 2003

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

SUBJECT: VESSEL HEAD PENETRATION CRACKING AND REACTOR PRESSURE VESSEL DEGRADATION

Dear Chairman Diaz:

During the 502nd meeting of the Advisory Committee on Reactor Safeguards, May 8-9, 2003, we met with representatives of the NRC staff regarding pressurized water reactor (PWR) vessel head penetration (VHP) cracking and reactor pressure vessel degradation. This matter was discussed with members of the EPRI Materials Reliability Program (MRP) at the 500th ACRS meeting, March 6-8, 2003, and with the MRP and NRC staff during a joint Materials and Metallurgy and Plant Operations Subcommittee meeting, April 22-23, 2003. During our reviews we had the benefit of the documents referenced.

This topic was addressed in our previous reports dated July 23, 2001, and June 20, 2002. This report expands on technical concerns raised in these previous reports.

CONCLUSIONS AND RECOMMENDATIONS

(1) The action plans, developed to address the recommendations of the Lessons Learned Task Force (LLTF), define the work needed to provide a sound technical basis for assessing industry's development of a proactive life management methodology for materials degradation in PWR vessel head penetrations.

(2) The LLTF action plans need to be augmented in some areas:

(a) Cracking prediction algorithms that address pressure vessel penetrations other than those in the vessel head
(b) Flaw Evaluation Guidelines for vessel head penetrations
(c) Qualification criteria for vessel head penetration inspection techniques
(d) Other degradation modes for high-chromium nickel-base alloys

(3) Although we support cooperation with other organizations in collecting the required data, the staff must analyze the data independently.

DISCUSSION

The NRC issued a series of Bulletins (2001-01, 2002-01, 2002-02) and finally an Order (EA-03-009) in February 2003 to deal with the various materials degradation phenomena that have been observed in PWR VHPs. The Order mandated interim inspection requirements (technique, location, and frequency) that would be operative until revised inspection requirements could be defined in 10 CFR 50.55a. These actions were based on engineering judgment informed by available data.

The EPRI MRP is developing a proactive life management methodology for the various degradation modes. The program involves: (a) identification of potential degradation modes, (b) development of inspection techniques, (c) specification of inspection intervals, and (d) a safety assessment. The NRC needs to develop the capability to evaluate this methodology. The LLTF action plans lay the groundwork for such a capability in the areas of stress corrosion cracking, boric acid corrosion, barrier integrity, and inspection.

There are several technical challenges that are not fully addressed in the current LLTF action plans.

The metric "Effective Degradation Years" used by the industry and NRC for prioritizing inspections of VHPs is based solely on operating temperature and time. As we have pointed out in previous reports, the prioritization algorithm is incomplete because it does not take into account stress and material parameters. However, this algorithm is adequate for prioritizing VHP inspections for the near future because the material and stress conditions in this particular configuration seem sufficiently similar.

Different prioritization algorithms will be needed for other penetrations (such as the pressure vessel bottom head or pressurizer) where markedly different residual stress profiles are expected. Given the potential cracking event in the bottom head at South Texas Project Unit 1, prioritization algorithms for these other penetrations should be developed now.

Management of boric acid corrosion of low-alloy steel in the VHP subassembly using the inspection schedule required by the Order should be adequate to detect the cracking which is the precursor to the boric acid corrosion. However, it remains a concern that corrosion rates on the order of one inch per year in the low-alloy steel at Davis-Besse were unpredicted. This lack of prediction capability could be of concern if the inspection methodology failed to detect a crack just before the crack penetrated to the annulus between the control rod drive mechanism (CRDM) tube and the pressure vessel. Thus, a specific objective of the LLTF action plans should be the development of a predictive capability for boric acid corrosion under the specific system conditions relevant to the VHP geometry and operating conditions. In order to efficiently resolve this issue, there should be adequate attention to the fundamental aspects of this degradation phenomenon.

The recently revised Flaw Evaluation Guidelines issued by the NRC for disposition of cracks in vessel head subassemblies are acceptable, but there are concerns regarding the details, which will need to be addressed. For instance, (a) there is no guidance about the residual stress profile that is needed in the calculation of stress intensity, and (b) there is no justification given for the choice of the (75th percentile 50% confidence) curve fit of the crack propagation rate vs. stress intensity data for Alloy 600 as the crack disposition relationship (rather than the "95/50" curve used in the earlier guideline), and the impact this has on the uncertainty in predicted crack depths at the end of an inspection period.

The industry will be changing their materials of construction for vessel head penetration to more "stress corrosion resistant" alloys (Alloys 690,152, and 52). There is evidence, largely from abroad, that such resistance, originally seen in the laboratory, is experienced in plant operation. However, there are insufficient stress corrosion data to enable the NRC to analyze quantitatively the improvement in resistance to cracking in VHPs utilizing these new alloys. Until these data are available there should be no relaxation in the inspection requirements for new reactor vessel heads imposed by the current Order.

The use of the Flaw Evaluation Guidelines will require determination of the size of cracks in the VHP subassembly as a function of the crack location and orientation. It is not clear from the industry presentations at the subcommittee meeting that the various inspection techniques can provide adequate crack sizing capability (i.e., resolution, repeatability, probability of detection). The LLTF action plans objectives state that revised inspection guidelines will be developed following examination of VHP inspection results and evaluation of current methodologies for determining leakage probability, non-destructive testing, etc. This is a crucial area in the control of VHP head degradation.

The LLTF action plans do not include an assessment of other modes of degradation in the high-chromium nickel-base alloys such as Alloys 182 and 82, and the replacement Alloys 690, 152, and 52. For instance, the fracture toughness of these alloys can be lowered under specific conditions of temperature and exposure, and this known phenomenon might be of significance during cooling accident situations and in the definition of flaw acceptance criteria. Furthermore, the weld alloys, such as Alloy 52, have a known propensity to crack during welding fabrication. The NRC should be in a position to analyze these scenarios.

As in many of the nuclear-related fields, there has been an attrition over the past decade in the experimental and analytical capabilities needed to resolve the above challenges in a timely manner. Thus, it is appropriate that industry and NRC have cooperative programs to collect data. It is important to emphasize that the NRC must develop and retain its own independent analytical capability.

Dr. William J. Shack did not participate in the deliberations on this matter.

Additional comments by ACRS members Dana A. Powers and Thomas S. Kress are presented below.

Sincerely,

Mario V. Bonaca
Chairman

<u>Additional Comments by ACRS Members Dana A. Powers and Thomas S. Kress</u>

Our colleagues have noted in this report that the assurance of the integrity of pressure boundaries in nuclear power plants will rely on inspection methods for the foreseeable future. Current technologies for inspection of reactor pressure boundaries have very limited capabilities. Though we do not at all impugn the efforts by EPRI and commercial firms to optimize these technologies, the truth is that these methods are cumbersome to apply, have low probabilities of detecting flaws and cracks, do not provide adequate characterizations of the sizes and orientations of cracks and flaws, and do not provide indications of the rates of crack growth. There are great needs for innovations in technologies for more convenient inspection of pressure boundaries, higher probabilities of detection, better characterization of flaws and cracks and indications of crack growth. These needs for better technology extend beyond the nuclear community into many if not most industrial areas. The NRC should join with others to solicit and stimulate the Government and the private sector to innovate more useful methods for the inspection of metal structures.

<u>References</u>:

1. Letter dated April 11, 2003, from Richard Barrett, Office of Nuclear Reactor Regulation, NRC, to Alex Marion, Nuclear Energy Institute, Subject: Flaw Evaluation Guidelines.
2. U.S. Nuclear Regulatory Commission, Subject: Davis-Besse Reactor Vessel Head Degradation Lessons-Learned Task Force Report, September 30, 2002.
3. U.S. Nuclear Regulatory Commission Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," August 3, 2001.
4. U.S. Nuclear Regulatory Commission Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity," March 18, 2002.
5. U.S. Nuclear Regulatory Commission Bulletin 2002-02, "Reactor Pressure Vessel Head and Vessel Head Penetration Nozzle Inspection Program," August 9, 2002.
6. U.S. Nuclear Regulatory Commission Order EA-03-009, "Issuance of Order Establishing Interim Inspection Requirements for Reactor Pressure Vessel Heads at Pressurized Water Reactors," February 11, 2003.

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

May 16, 2003

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

Subject: DRAFT FINAL REGULATORY GUIDE 1.178 AND STANDARD REVIEW PLAN SECTION 3.9.8 FOR RISK-INFORMED INSERVICE INSPECTION OF PIPING

Dear Chairman Diaz:

During the 502nd meeting of the Advisory Committee on Reactor Safeguards, May 8-9, 2003, we met with representatives of the NRC staff to discuss the draft final Regulatory Guide (RG) 1.178, "An Approach for Plant Specific Risk-Informed Decisionmaking for Inservice Inspection of Piping," and the associated Standard Review Plan (SRP) Section 3.9.8, "Standard Review Plan for the Review of Risk-Informed Inservice Inspection Applications." We also had the benefit of the documents referenced.

RECOMMENDATIONS

1. The draft final RG 1.178 and associated SRP Section 3.9.8 should be issued.

2. The staff should consider undertaking a study in which EPRI , Westinghouse Owners Group (WOG), and French methodologies are applied to the same piping system and the resulting inspection plans are compared to gain a better understanding of the impact of the different approaches.

DISCUSSION

RG 1.178 and the associated SRP Section 3.9.8 were issued for trial use in September 1998. In our report of June 12, 1998, we concluded that a risk-informed inservice inspection (RI-ISI) program would result in reductions in the risk from piping failures, occupational radiation exposures, and associated inspection costs and that RG 1.178 provided general guidance for developing RI-ISI programs. The detailed methodologies needed for the development of such programs are provided in topical reports prepared by EPRI and WOG.

Based on the staff's experience during the trial use period, the staff is now preparing to issue a final revised version of RG 1.178 and SRP Section 3.9.8. Most of the changes in RG 1.178 are editorial.

The most important substantive changes are additional documentation requirements in RG 1.178 and reviewer directions in SRP Section 3.9.8 to ensure that the probabilistic risk assessment used to support the submittal is of adequate quality. The revised guide states that the licensee's submittal should discuss the measures taken to ensure quality and to address any limitations of the analysis that are expected to impact conclusions about the acceptability of proposed changes. If a peer review were performed, the submittal should discuss the resolution of the findings of the review. We support the staff's decision to require such documentation.

Although the staff has the general impression that the EPRI methodology gives somewhat more conservative results than the WOG methodology, no systematic comparison of the results of the two methodologies has been made by staff or industry. The two methodologies take different approaches to risk categorization of piping segments and different approaches to the assessment of pipe failure frequency. The EPRI methodology uses absolute values of conditional core damage probability. The WOG methodology uses Fussell-Vesely and Risk Achievement Worth importance measures. From our discussions, we understand that most of the international nuclear community is adopting the EPRI and WOG methodologies, with one exception. A third methodology has been developed in France. The staff should consider a study comparing the results from the application of the three methodologies to the same piping system. Such a comparison could give useful insights into the process of risk categorization. Also, high confidence in the effectiveness of RI-ISI programs will become increasingly important when considering risk-informed approaches to 10 CFR 50.46.

Sincerely,

Mario V. Bonaca
Chairman

References:

1. Letter dated April 25, 2003, from Scott F. Newberry, Office of Nuclear Regulatory Research, to John T. Larkins, Executive Director, ACRS, Subject: ACRS Review of Draft Revised Regulatory Guide 1.178, "An Approach for Plant Specific Risk-Informed Decisionmaking for Inservice Inspection of Piping," and the Associated Standard Review Plan Chapter 3.9.8" (Predecisional).
2. Report dated June 12, 1998, from R. L. Seale, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Proposed Final Standard Review Plan Section 3.9.8 and Regulatory Guide 1.178 for Risk-Informed Inservice Inspection of Piping.
3. Electric Power Research Institute, "Risk-Informed Inservice Inspection Evaluation Procedure," EPRI TR-106706, June 1996.

4. Westinghouse Energy Systems, WCAP-14572, Revision 1, "Westinghouse Owners Group Application of Risk Informed Methods to Piping Inservice Inspection Topical Report," October 1997.
5. Westinghouse Energy Systems, WCAP-14572, Revision 1, Supplement 1, "Westinghouse Structural Reliability and Risk Assessment (SRRA) Model for Piping Risk-Informed Inservice Inspection," October 1997.
6. European Commission - DG Environment, Study Contract: B4-3040/99/23123/MAR/C2, Final Report of Task 1 of RIBA Project, "Review of Existing Risk-Informed Methodologies," December 2001.

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

June 13, 2003

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: REVISION 4 TO REGULATORY GUIDE 1.101, "EMERGENCY PLANNING AND PREPAREDNESS FOR NUCLEAR POWER REACTORS"

During the 503rd meeting of the Advisory Committee on Reactor Safeguards, June 12-13, 2003, the Committee considered Revision 4 to Regulatory Guide 1.101, "Emergency Planning and Preparedness for Nuclear Power Reactors," and decided not to review this document. The Committee agrees that the staff should continue with its process for issuing Revision 4 to Regulatory Guide 1.101.

Reference:
Memorandum dated May 14, 2003, from Bruce A. Boger, NRR, to John T. Larkins, ACRS, Subject: Review of Proposed Final Revision 4 to Regulatory Guide 1.101, "Emergency Planning and Preparedness for Nuclear Power Reactors"

cc: A. Vietti-Cook, SECY
I. Schoenfeld, OEDO
S. Collins, NRR
B. Boger, NRR
T. Blount, NRR
P. Wen, NRR

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

June 24, 2003

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

SUBJECT: UPDATE TO LICENSE RENEWAL GUIDANCE DOCUMENTS: RESPONSE TO STAFF REQUIREMENTS MEMORANDUM DATED JULY 17, 2002

Dear Chairman Diaz:

In a Staff Requirements Memorandum (SRM) dated July 17, 2002, the Commission stated that, "The ACRS should consider providing a recommendation as to how license renewal guidance documentation should be updated to reflect supporting information, particularly with regard to time-limited aging analyses that should, as a minimum, be included in license renewal applications to maximize the efficiency of the review process and minimize requests for additional information."

The staff has been developing Interim Staff Guidances (ISGs) on various license renewal issues based on the insights gained from its review of several license renewal applications (LRAs). To date, the staff has developed 16 such ISGs in coordination with NEI, except the one on Standardized Format for License Renewal Applications, which was developed by NEI and approved by the staff. In developing our recommendations, we have taken into account these ISGs and other staff initiatives associated with enhancing the license renewal process. In addition to addressing the issue raised in the SRM, we also include recommendations to be considered in updating the license renewal guidance documents and enhancing the license renewal process.

We met with representatives of the NRC staff and NEI on June 13, 2003, to discuss the ISG process and several specific ISGs. Our Subcommittee on License Renewal met with representatives of NEI on June 11, 2003, to obtain their views on the Standardized Format for License Renewal Applications. We also had the benefit of the documents referenced.

RECOMMENDATIONS

1. We agree with the guidance provided in ISGs 1 - 16. The ISG process is a major step toward improving the efficiency of the review process and reducing the number of requests for additional information (RAIs). The staff should continue to provide guidance on emerging license renewal issues through the ISG process and incorporate

such guidance into the future revisions of the generic liense renewal guidance documents.

2. Proposed ISG 16, "Time-Limited Aging Analyses Supporting Information for License Renewal Applications," was developed in response to our concern that some of the LRAs do not include sufficient information on time-limited aging analyses (TLAAs). This ISG is particularly responsive to the SRM, in that it directly addresses the supporting information on TLAAs that needs to be included in LRAs. ISG 16 should be finalized and issued for use by the applicants.

3. The Generic Aging Lessons Learned (GALL) Report specifies limits for sulfate ion concentrations in below-grade water to avoid decrepitation of concrete. The staff should consider whether similar limits and guidance are needed for phosphate ion concentration.

DISCUSSION

In the SRM, the Commission asked that we consider ways to maximize the efficiency of the license renewal review process and minimize the number of RAIs.

In some areas, the staff has found it necessary to submit similar RAIs to several applicants. This indicates that the guidance may be inadequate in these areas. The staff has, therefore, undertaken an effort to prepare ISGs to further define or clarify these areas. The intention is to incorporate these ISGs into future revisions of the guidance documents. The ISG process will improve the efficiency of the license renewal process and reduce the number of RAIs. The staff should continue with the ISG process to provide guidance on emerging license renewal issues.

To date, in coordination with NEI the staff has developed 16 ISGs to address various license renewal and process issues. Of these, proposed ISG 16 is developed in response to the concern expressed in our report of December 18, 2002, on the LRA for the North Anna and Surry Nuclear Power Stations. In that report, we stated that the applicant had not submitted its evaluations of the reactor vessel margins for pressurized thermal shock and upper shelf energy, and that such critical parameters should be included in future LRAs. This ISG also deals with the issue raised in the SRM with regard to supporting information on TLAAs that should be included in the LRAs. This has been a troublesome area in that lack of specifics in the application has necessitated a number of RAIs. The staff should finalize ISG 16 and issue it for use by the industry in preparing future LRAs.

In advance of completion of ISGs, we would expect applicants to be aware of the staff's RAIs on previous LRAs and address them, as appropriate, before submitting their applications. Such a practice would reduce the number of RAIs. We are beginning to see this occurring in more recent applications.

We are currently reviewing the LRA for the Ft. Calhoun Station Unit 1, which is the first application to be entirely based on the generic license renewal guidance documents. We see a moderately reduced number of RAIs and a more streamlined application. We expect further efficiencies as the staff gains more experience in reviewing LRAs prepared in accordance with these documents.

We believe that the efficiency of the license renewal process will greatly improve as a result of incorporating the ISGs into the guidance documents, reviewing RAIs on previous applications, and preparing LRAs in accordance with the guidance documents and the recently issued Standardized Format for License Renewal Applications.

The GALL Report specifies limits for sulfate ion concentrations in below-grade water to avoid concrete decrepitation. Such decrepitation occurs when ionic reactions convert calcium hydroxide to a more voluminous species such as calcium sulfate hydrate. Reactions with phosphate ion could lead to similar degradation. Conversion to the very stable species hydroxyapatite ($Ca_5 (PO_4)_3 OH$) is of particular concern. The phosphate ion concentrations necessary to cause conversions to hydroxyapatite are not specified in the literature, but can be estimated from known aqueous thermochemistry. These estimates suggest that relatively low concentrations of phosphate could cause decrepitation of concrete. These estimates are based on thermodynamic considerations and could be conservative if the kinetics of the reactions are slow. Still, the potential for decrepitation by phosphate ions indicated by the thermodynamics should be addressed by the staff.

Between approval of the LRA and entering the period of extended operation, the staff has a substantial inspection workload to ensure that the licensees appropriately implement the commitments made during the review process. The staff has made an effort to identify this workload in Inspection Procedure 71003. Many licensees begin to implement these commitments soon after approval of their extended licenses. The staff needs to anticipate the resultant workload.

There are several cases in which licensees have committed to perform activities in accordance with technologies and methodologies that are still under development. Relevant examples include (1) a method for identifying incipient cable failure due to moisture treeing and (2) improved methodologies for inservice inspection methodologies of reactor coolant piping, with the sensitivity to detect flaws such as those identified at the Virgil C. Summer Nuclear Station only after they led to leakage. The staff should continue to keep abreast of these developing methodologies, evaluate them, and conduct inspections to ensure that licensees are complying with their commitments.

Current performance is of little value in predicting licensee performance many years in the future. Nevertheless, a review of the current findings of the reactor oversight process (ROP) for a given plant may yield some insights about the areas of licensee strengths and areas for future improvement and may help focus future inspection activities in areas critical to the success of license renewal (e.g., corrective action and preventative maintenance programs).

In response to our request, the staff is now providing the current status of the ROP findings, as well as a broad assessment of the current material condition of the plant, during our review of each LRA.

We believe that the actions already taken or in progress, and those additional actions described here will improve the efficiency of the license renewal process and reduce the number of RAIs.

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

Sincerely,

Mario V. Bonaca

Mario V. Bonaca
Chairman

References:

1. Staff Requirements Memorandum, dated July 17, 2002, from Anette L. Vietti-Cook, Secretary of the Commission, to John T. Larkins, ACRS, Subject: Meeting with ACRS on July 10, 2002.
2. Memorandum dated May 21, 2003, from P. T. Kuo, Office of Nuclear Reactor Regulation, to John T. Larkins, ACRS, transmitting the following Interim Staff Guidances (ISGs):
   - ISG-01, GALL Report presenting one acceptable way to manage aging effects for license renewal
   - ISG-02, Scoping of equipment relied on to meet the requirements of the station blackout (SBO) rule for license renewal
   - ISG-03, Aging management program of concrete
   - ISG-04, Aging management of fire protection system for license renewal
   - ISG-05, Identification and treatment of electrical fuse holders for license renewal
   - ISG-06, Identification and treatment of housing for active components for license renewal
   - ISG-07, Scoping of fire protection equipment for license renewal
   - ISG-08, Updating the improved license renewal guidance documents-ISG process
   - ISG-09, Identification and treatment of structures, systems, and components which meet 10 CFR 54.4(a)(2)
   - ISG-10, Standardized format for license renewal applications
   - ISG-11, Aging management of environmental fatigue for carbon/low alloy steel
   - ISG-12, Operating experience with cracking of Class 1 small bore piping
   - ISG-13, Management of loss of preload on reactor vessel internals bolting using the loose parts monitoring system
   - ISG-14, Operating experience with cracking on bolting

- ISG-15, Revision to generic aging lessons learned aging management program (AMP) XI.E2
- ISG-16, Time-limited aging analyses supporting information for license renewal applications

3. NRC Inspection Manual, Inspection Procedure 71003, "Post-Approval Site Inspection for License Renewal Program Applicability," dated December 9, 2002.
4. Report dated December 18, 2002, from George E. Apostolakis, ACRS Chairman, to Richard A. Meserve, NRC Chairman, Subject: Report on the Safety Aspects of the License Renewal Applications for the North Anna Power Station Units 1 and 2 and Surry Power Station Units 1 and 2.
5. U. S. Nuclear Regulatory Commission, NUREG-1801, Vol. 1, "Generic Aging Lessons Learned (GALL) Report," dated March 1, 2001.
6. A. J. Bard, R. Parsons, and J. Jordan, Standard Potentials in Aqueous Solution, Marcel Dekker Publishing Company, 1985.

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

July 15, 2003

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: DRAFT FINAL REGULATORY GUIDE DG-1105, "PROCEDURES AND CRITERIA FOR ASSESSING SEISMIC SOIL LIQUEFACTION AT NUCLEAR POWER PLANT SITES"

During the 504th meeting of the Advisory Committee on Reactor Safeguards, July 9-11, 2003, the Committee considered the draft final Regulatory Guide DG-1105, "Procedures and Criteria for Assessing Seismic Soil Liquefaction at Nuclear Power Plant Sites," and decided not to review it. The Committee has referred this Regulatory Guide to the Advisory Committee on Nuclear Waste for possible consideration.

Reference:
Memorandum dated July 2, 2003, from Michael E. Mayfield, Director, Division of Engineering Technology, RES, to John T. Larkins, Executive Director, ACRS, Subject: Request for ACRS Review of Draft Final of DG-1105, "Procedures and Criteria for Assessing Seismic Soil Liquefaction at Nuclear Power Plant Sites"

cc: A. Vietti-Cook, SECY
I. Schoenfeld, OEDO
A. Thadani, RES
M. Mayfield, RES
Y. Li, RES
J. Garrick, ACNW, w/ref.
H. Larson, ACNW, w/ref.

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

July 15, 2003

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: REVISION TO SECTION 9.5.1, "FIRE PROTECTION PROGRAM," OF THE STANDARD REVIEW PLAN

During the 504th meeting of the Advisory Committee on Reactor Safeguards, July 9-11, 2003, the Committee considered a revision to Section 9.5.1, "Fire Protection Program," of the Standard Review Plan, NUREG-0800 and its associated Branch Technical Position. The NRC staff advises us that the revision was based on information gathered during the development of Regulatory Guide 1.189, "Fire Protection for Operating Nuclear Power Plants." The Committee has decided not to review the subject document.

Reference:
Memorandum dated May 13, 2003, from R. William Borchardt, NRR, to John T. Larkins, Executive Director, ACRS, Subject: Development of Revised Fire Protection Standard Review Plan and Branch Technical Position

cc: A. Vietti-Cook, SECY
I. Schoenfeld, OEDO
S. Collins, NRR
W. Borchardt, NRR
J. Hannon, NRR
D. Frumkin, NRR

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

July 16, 2003

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

SUBJECT: SAFETY CULTURE

Dear Chairman Diaz:

During the 503rd meeting of the Advisory Committee on Reactor Safeguards, June 12-13, 2003, we met with representatives of the public, the industry, and the NRC staff (References 1, a-l) to discuss the collective understanding and attributes of safety culture at nuclear power plants. We also had the benefit of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

1. The existing regulations provide an appropriate framework for monitoring the impact of licensee safety culture on performance.

2. The NRC should periodically self-assess its safety climate.

DISCUSSION

The concept of safety culture encompasses a broad spectrum of characteristics that include personnel attitudes, the control of work activities, and organizational structures. Although safety culture means different things to different people, a working definition of the term has been provided by the International Nuclear Safety Advisory Group (INSAG) of the International Atomic Energy Agency (IAEA) (Reference 2). In its view, safety culture is "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." In a Policy Statement on the Conduct of Nuclear Power Plant Operations (Reference 3), the Commission proposed a similar definition.

Although there are alternative definitions of safety culture, there is general agreement on the important attributes of safety culture. These include a questioning attitude, conservative decisionmaking, attention to detail, personal accountability, adherence to procedures, as well as the management traits and processes, such as leadership, conservative operating philosophy, effective training, and effective corrective and preventive action, that reinforce these attributes of the workforce.

Although we are unaware of any quantitative relationship between the characteristics of safety culture and safety performance, there is evidence from nonnuclear power plant applications that safety attitudes and safety performance are positively correlated (Reference 4). It is clearly the judgment of many people in many industries that safety attitudes have enormous impact on safety performance. The Institute of Nuclear Power Operations (INPO), for example, routinely evaluates attributes of safety culture at operating plants. In its policy statement, the Commission stated its conviction that "the working environment provided for the conduct of operations at nuclear power facilities has a direct relationship to safety." We agree that safety culture is important to safety performance.

The mission of a regulatory agency is to ensure good safety performance. Because safety culture is important to such performance, the question arises as to what is the proper role of the regulator with respect to safety culture. The Commission's policy statement makes it clear that it is the responsibility of utility management to establish and maintain "a professional working environment with a focus on safety." The Commission noted, however, that this policy statement should not be construed as limiting NRC authority to take action on matters affecting the safe operation of the plants.

The current regulations do address several important attributes of safety culture, albeit at a fairly high level. Appendix B to 10 CFR Part 50 requires the licensees to establish a quality assurance program. Quality assurance means "all those planned and systematic actions necessary to provide adequate confidence that a structure, system, or component will perform satisfactorily in service." Criterion XVI of Appendix B, "Corrective Actions," states: "Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected."

Conditions that will promote quality as envisaged in Appendix B include adherence to procedures and an effective corrective action program. These are attributes of safety culture. Furthermore, a questioning attitude, conservative decisionmaking, personal accountability, and attention to detail are essential elements of an effective corrective action program. Again, these are elements of safety culture.

A sampling of letters from the NRC regional offices to plant managers shows that the staff does focus considerable attention on aspects of safety culture. Findings such as "plant personnel focused on replacement rather than understanding causes of wear" and "industry experience was not incorporated so as to minimize wear" could be said to reflect two aspects of safety culture that are commonly cited, namely, a "questioning attitude" of personnel and the plant's "organizational learning." It is important to note that these findings are not the results of an evaluation of questioning attitude in general or the effectiveness of the organizational learning processes of the licensees using tools from the social sciences, such as questionnaires. These findings are based on observations related to specific incidents; i.e., they are based on actual licensee performance.

The Reactor Oversight Process (ROP) identifies three "cross-cutting" issues (Reference 5): Human Performance, Safety-Conscious Work Environment, and Problem Identification and Resolution (PI&R). All three are strongly affected by safety culture. The examples of findings given to utility managers that we cited above resulted from inspections carried out under the ROP.

The NRC Inspection Manual appears to provide adequate guidance to ensure that licensees are detecting and correcting problems. Inspection Procedure 71152, Identification and Resolution of Problems, requires that every 2 years the inspectors select a sample of conditions adverse to quality that the licensee has processed through its corrective action program. The purpose is to focus on problem identification, resolution, and the effectiveness of corrective actions.

Appendix 1 to this inspection procedure lists a number of questions that are intended to help the inspectors assess whether there are impediments to the establishment of a safety-conscious work environment. These should not be construed as being formal interviews. Appendix 1 states: "It is not intended that these questions be asked verbatim, but rather that they form the basis for gathering insights regarding whether there are impediments to the formation of a safety-conscious work environment."

We conclude that the regulatory framework for monitoring aspects of safety culture is largely in place. This framework is appropriately performance based. Agency actions resulting from performance findings are appropriately based on their risk significance according to the action matrix of the ROP. Broader evaluations of safety culture, such as management emphasis on safety and personnel attitudes, belong to the industry. At our June 2003 meeting, we were pleased to learn from industry representatives that there is a great deal of activity on understanding what a good safety culture is and improving tools for evaluating it.

The catalyst for the renewed industry-wide interest in the issue of safety culture and its impact on human performance was, of course, the recent incident at the Davis-Besse nuclear power plant. The NRC staff's Lessons-Learned Task Force (LLTF) concluded that (Reference 6):

- the NRC failed to adequately review, assess, and followup on relevant operating experience, and
- the NRC failed to integrate known or available information into its assessments of Davis-Besse's safety performance.

The LLTF has made numerous recommendations regarding the improvement of the NRC's processes. Some of these are directly related to safety culture. For example, recommendation 3.3.1(1) addresses the issue of "maintaining a questioning attitude in the conduct of inspection activities." We agree with this recommendation. However, we believe that the agency's safety culture is fundamentally sound. The NRC is focused on safety, and safety issues receive the attention warranted by their significance. At this point, it is useful to distinguish between the concepts of safety culture and safety climate. Safety culture refers to the enduring fundamental values of an organization. Safety climate is a temporal state, a snapshot in time of conditions that may influence safety culture attributes.[1] Safety climate is subject to change and can vary throughout the organization.

[1]In testimony before the Commission on June 25, 1998, the Director of the survey used by the NRC Office of Inspector General to assess the agency's safety culture, said, "We needed to make sure we had an overview of culture; namely, shared values and beliefs, practices, and policies, but we also needed to get a valid snapshot of the most urgent or acute issues facing the agency currently. That more has to do with the climate or the 'now' of a particular organization."

The agency is already assessing its programs and policies, e.g., by assessing the effectiveness of various regulations. We believe that it would be useful for the NRC to undertake a self-assessment of its current safety climate. This evaluation should include aspects of safety culture such as conservative decisionmaking, willingness to raise and report issues, and questioning attitude in the presence of inconclusive evidence.

It is important to place the current emphasis on safety culture in perspective. The industry and NRC staff have mature programs to monitor reliability at the active equipment level. The reliability of passive equipment is monitored through inservice inspection and testing programs. Human reliability is monitored through simulator testing programs for control room crews. Awareness of safety culture adds to understanding and management of the deeper causes that shape human performance.

Sincerely,

Mario V. Bonaca
Chairman

References:
1. Presentations at ACRS Workshop on Safety Culture, June 12-13, 2003:
   a. Ashok Thadani, Director, Office of Nuclear Regulatory Research, NRC, *Safety Culture*, June 12, 2003.
   b. Chuck Dugger, Vice President, Nuclear Energy Institute, *Collective Understanding of Safety Culture*, June 12, 2003.
   c. Thomas E. Murley, Nuclear Energy Agency, *Early Signs of Deteriorating Safety Performance*, June 12, 2003.
   d. Howard Whitcomb, III, Esq, *Comments on Collective Understanding of Safety Culture,* and William N. Keisler, Nuclear Maintenance Integration Consultants, *Organization Half-Life, The Un-Monitored Disintegration in Reactor and Public Safety*, June 12, 2003.
   e. David Collins, Engineering Analyst, Dominion Nuclear Connecticut, *Managing Safety Culture*, June 12, 2003.
   f. Alan Price, Vice President, Dominion Nuclear Connecticut, *Safety Culture*, June 12, 2003.
   g. David Trimble, Clare Goodman, Lisamarie Jarriel, and J.J. Persensky, NRC, *Attributes of Safety Culture*, June 12, 2003.
   h. George Felgate, Director, Analysis Division, Institute of Nuclear Power Operations, *Safety Culture Attributes*, June 12, 2003.
   i. Lew Myers, Chief Operating Officer, First Energy Nuclear Operating Company, *Organizational Safety Culture*, June 12, 2003.
   j. Jack Grobe, Chairman, Davis-Besse Oversight Panel and Geoff Wright, Inspection Team Leader, NRC, *Management and Human Performance Inspection at Davis-Besse*, June 12, 2003.
   k. William O'Connor, Chairman of the Board, Utility Service Alliance and Vice President, Nuclear Generation, Detroit Edison, *Nuclear Safety Culture Assessment*, June 12, 2003.

1. Sonja B. Haber, Human Performance Analysis Corporation, *Attributes of Safety Culture*, June 12, 2003.
2. International Nuclear Safety Advisory Group, *Safety Culture*, Safety Series No. 75-INSAG-4, International Atomic Energy Agency, Vienna, 1991.
3. U.S. Nuclear Regulatory Commission, "Policy Statement on the Conduct of Nuclear Power Plant Operations," *Federal Register*, 54FR 3424, January 24, 1989.
4. U. S. Nuclear Regulatory Commission, NUREG-1756, *Safety Culture: A Survey of the State of the Art*, Prepared by Advisory Committee on Reactor Safeguards, J. N. Sorensen, Senior Fellow, ACRS (see also *Reliability Engineering & System Safety*, vol. 76, pp. 189-204, 2002).
5. U. S. Nuclear Regulatory Commission, *Recommendations for Reactor Oversight Process Improvements*, SECY-99-007, 1999.
6. U. S. Nuclear Regulatory Commission, *Degradation of the Davis-Besse Nuclear Power Station Reactor Pressure Vessel Head. Lessons-Learned Report*, September 30, 2002.

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

July 17, 2003

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

SUBJECT: PROPOSED CRITERIA FOR THE TREATMENT OF INDIVIDUAL REQUIREMENTS IN A REGULATORY ANALYSIS

Dear Chairman Diaz:

During the 504th meeting of the Advisory Committee on Reactor Safeguards, July 9-11, 2003, we discussed with representatives of the NRC staff proposed criteria for the treatment of individual requirements in a regulatory analysis. We had the benefit of the document referenced.

CONCLUSION

The proposed criteria are responsive to the Commission's Staff Requirements Memorandum (SRM) dated December 31, 2001.

DISCUSSION

In the SRM dated December 31, 2001, the Commission directed the staff to ". . . provide the Commission with recommendations for revising existing guidance in order to implement a disciplined, meaningful, and scrutable methodology for evaluating the value-impact of any new requirements that could be added by a risk-informed alternative rule." The concern is that aggregating or "bundling" different requirements in a single regulatory analysis could potentially mask the inclusion of an inappropriate individual requirement. To address this concern, the staff has developed proposed criteria for the treatment of individual requirements in a regulatory analysis. The staff plans to incorporate the final criteria into NUREG/BR-0058, "Regulatory Analysis Guidelines of the U. S. Nuclear Regulatory Commission." We believe the proposed criteria are appropriate and responsive to the Commission's direction.

Sincerely,

Mario V. Bonaca
Chairman

Reference:
1. Federal Register, Volume 68, No. 75, dated April 18, 2003, pages 19162-19166, Subject: Regulatory Analysis Guidelines: Proposed Criteria for the Treatment of Individual Requirements in a Regulatory Analysis.

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

September 15, 2003

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: DRAFT FINAL REVISION 1 OF REGULATORY GUIDE 1.138, "LABORATORY INVESTIGATIONS OF SOILS AND ROCKS FOR ENGINEERING ANALYSIS AND DESIGN OF NUCLEAR POWER PLANTS" (DRAFT WAS ISSUED AS DG-1109)

During the 505th meeting of the Advisory Committee on Reactor Safeguards, September 10-13, 2003, the Committee considered the draft final Revision 1of Regulatory Guide 1.138, "Laboratory Investigations of Soils and Rocks for Engineering Analysis and Design of Nuclear Power Plants" and decided not to review it. The Committee agrees with the staff's proposal to issue this Regulatory Guide for industry use.

Reference:
Memorandum dated September 5, 2003, from Michael E. Mayfield, RES to John T. Larkins, ACRS, Subject: Request for ACRS Review of Draft Final Revision 1 of Regulatory Guide 1.138, "Laboratory Investigations of Soils and Rocks for Engineering Analysis and Design of Nuclear Power Plants."

cc: A. Vietti-Cook, SECY
I. Schoenfeld, OEDO
A.Thadani, RES
M. Mayfield, RES
A. Hsia, RES
Y. Li, RES

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

September 15, 2003

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: DRAFT FINAL REGULATORY GUIDE DG-1099, "ANCHORING COMPONENTS AND STRUCTURAL SUPPORTS IN CONCRETE"

During the 505th meeting of the Advisory Committee on Reactor Safeguards, September 10-13, 2003, the Committee considered the draft final regulatory Guide DG-1099, "Anchoring Components and Structural Supports in Concrete," and decided not to review it. The Committee agrees with the staff's proposal to issue this Regulatory Guide for industry use.

Reference:
Memorandum dated September 2, 2003, from Michael E. Mayfield, RES to John T. Larkins, ACRS, Subject: Request for ACRS Review of Draft Final Regulatory Guide, DG-1099, "Anchoring Components and Structural Supports in Concrete."

cc: A. Vietti-Cook, SECY
W. Dean, OEDO
I. Schoenfeld, OEDO
A. Thadani, RES
W. Borchardt, NRR
M. Mayfield, RES
H. Graves, RES

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

September 17, 2003

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 ST. LUCIE NUCLEAR PLANT UNITS 1 AND 2

Dear Chairman Diaz:

During the 505th meeting of the Advisory Committee on Reactor Safeguards on September 10-13, 2003, we completed our review of the License Renewal Application (LRA) for the St. Lucie Nuclear Plant Units 1 and 2, and the related final Safety Evaluation Report (SER) prepared by the NRC staff. Our Plant License Renewal Subcommittee reviewed this LRA and the staff's initial SER during a meeting on April 9, 2003. During our review, we had the benefit of discussions with representatives of the NRC staff and Florida Power and Light Company (FPL or the applicant). We also had the benefit of the documents referenced.

CONCLUSION AND RECOMMENDATION

1. The programs instituted by FPL to manage age-related degradation are appropriate and provide reasonable assurance that St. Lucie Units 1 and 2 can be operated in accordance with their current licensing bases for the period of extended operation without undue risk to the health and safety of the public.

2. The FPL application for renewal of the operating licenses for St. Lucie Units 1 and 2 should be approved.

BACKGROUND AND DISCUSSION

This report fulfills the requirement of 10 CFR 54.25, which states that the ACRS should review and report on all license renewal applications. St. Lucie Units 1 and 2 are 2700 MWt Combustion Engineering-designed pressurized water reactors in large dry containments. In its application, FPL requested renewal of the operating licenses for St. Lucie Units 1 and 2 for 20 years beyond the current license term, which expires on March 1, 2016 for Unit 1 and April 6, 2023 for Unit 2. St. Lucie Unit 1 was licensed approximately 7 years before St. Lucie Unit 2. During these 7 years, significant events occurred at operating nuclear plants, including the Three Mile Island Unit 2 event and the Browns Ferry Fire event. The lessons learned from these events resulted in design differences between St. Lucie Unit 1 and Unit 2, which are appropriately reflected in the LRA.

The final SER documents the results of the staff's review of the information submitted by the applicant, including commitments that were necessary to resolve open items identified by the staff in the initial SER. In particular, the staff reviewed the completeness of the applicant's identification of structures, systems, and components (SSCs) that are subject to aging management; the integrated plant assessment process; the applicant's identification of the possible aging mechanisms associated with passive, long-lived components; and the adequacy of the applicant's aging management programs.

The staff also conducted several inspections at St. Lucie, including an audit of the adequacy of the scoping and screening methodology and its implementation to ensure that SSCs within the scope of license renewal have been appropriately identified; an inspection of the aging management programs to confirm that existing programs are functioning well and to examine the applicant's plans for establishing new and enhanced aging management programs; and a walkdown of plant systems to assess how the systems are being maintained.

On the basis of our review of the final SER, LRA, and the inspection report, we conclude that the process implemented by the applicant to identify SSCs that are within the scope of license renewal was effective, the applicant performed a comprehensive aging management review of such SSCs, and the staff and the applicant appropriately identified all SSCs that are within the scope of license renewal. The applicant stated that it plans to implement 70 to 80% of the commitments for license renewal prior to the issuance of the renewed licenses. We agree with the staff's conclusion that all open and confirmatory items have been closed appropriately and there are no issues that preclude renewal of the operating licenses for St. Lucie Units 1 and 2.

The groundwater at the St. Lucie site is characterized by high concentrations of chlorides and sulfates that create an aggressive environment for concrete structures. The applicant has committed to enhance those elements of the St. Lucie's Systems and Structures Monitoring Program that deal with inspections of accessible and inaccessible concrete structures. This Program will be enhanced to include specific provisions consistent with industry standards and inspection guidelines for monitoring concrete structures. The monitoring plan for inaccessible concrete structures includes inferring material conditions of inaccessible structures from inspection of accessible structures exposed to groundwater and opportunistic inspections of below-grade concrete. The applicant stated that during construction, concrete of sufficient quality was used to inhibit degradation of concrete and protect the embedded reinforcing steel. No concrete degradation has been found during opportunistic inspections of inaccessible concrete structures performed in 1997 and 2002. Based on this information, we agree with the staff that the enhancements proposed by the applicant provide reasonable assurance that the integrity of concrete structures at St. Lucie will be adequately monitored during the period of extended operation.

St. Lucie's Alloy 600 Inspection Program includes provisions and commitments for inspecting reactor pressure vessel (RPV) head penetration nozzles. The applicant has performed visual and ultrasonic inspections of the RPV heads of both units, and no evidence of leakage has been identified. An axial flaw was identified and repaired in two control element drive mechanism penetrations of Unit 2. The applicant has ordered replacement heads for both units. The applicant will continue to participate in the industry program for assessing and managing primary water stress corrosion cracking (PWSCC) in Alloy 600 RPV head penetration nozzles, and has committed to perform inspections as recommended by this program. Based on the applicant's responses to related NRC bulletins and its commitment to participate in the industry's program for assessing and managing PWSCC of the RPV head penetration nozzles, there is reasonable assurance that the integrity of St. Lucie Units 1 and 2 RPV heads will be adequately monitored and maintained.

The applicant identified those components at St. Lucie Units 1 and 2 that are supported by time-limited aging analyses (TLAAs) and provided data to demonstrate that the components have sufficient margin to operate properly during the period of extended operation.

Two of the TLAAs are unique to St. Lucie because they qualify repairs of long-lived passive components for the period of extended operation. The first addresses the repairs that took place at St. Lucie Unit 1 to deal with damage identified in 1983 in the core support barrel (CSB) and thermal shield assemblies. The thermal shield was permanently removed. Four lugs were found to have separated from the CSB and through-wall cracks were found adjacent to the lug areas. These cracks were arrested with crack-arrestor holes that were sealed by inserting expandable plugs. The repairs were qualified for the remaining life of the plant and have been repeatedly inspected and found to be effective. In order to qualify these repairs for 60-years life, the fatigue analysis of the CSB middle cylinder and the acceptance criterion for the expandable-plugs preload based on irradiation-induced stress relaxation had to be repeated to cover 60-years of operation. The staff performed a thorough review of this TLAA and found it acceptable. The work presented by the applicant and the staff, and the inservice inspections to which the CSB will continue to be subjected provide reasonable assurance that the integrity of the CSB will be adequately monitored and maintained during the period of extended operation.

The second TLAA involves the 1994 half-nozzle repair of four leaking pressurizer instrument nozzles at Unit 2 and the 2001 half-nozzle repair of one leaking hot leg instrument nozzle at Unit 1. These repairs need to be qualified for the extended period of operation. The staff's review of the supporting analyses, which includes a request for relief from certain requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, is still under way. The applicant has committed that if the acceptability of the half-nozzle design cannot be demonstrated for the period of extended operation, then this TLAA will be dispositioned by other means, possibly including appropriate nozzle replacement to comply with ASME Code replacement criteria. This commitment ensures that these repairs will be adequately qualified for the period of extended operation.

The applicant and the staff have identified plausible aging effects associated with passive, long-lived components. Adequate programs have been established to manage the effects of aging so that St. Lucie Units 1 and 2 can be operated in accordance with their current licensing bases for the period of extended operation without undue risk to the health and safety of the public.

Sincerely,

Mario V. Bonaca
Chairman

References:

1. U.S. Nuclear Regulatory Commission, NUREG -xxxx, "Safety Evaluation Report Related to the License Renewal of St. Lucie Nuclear Plant, Units 1 and 2," July 2003.
2. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report with Open Items Related to the License Renewal of St. Lucie Nuclear Plant, Units 1 and 2," February 2003.
3. Letter dated November 29, 2001 from J. A. Stall, Florida Power and Light Company, to U.S. Nuclear Regulatory Commission, transmitting Application to Renew the Operating Licenses of St. Lucie Nuclear Plant, Units 1 and 2.
4. U. S. Nuclear Regulatory Commission, Region II Inspection Report No. 50-335/03-03, 50-389/03-03.

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

September 22, 2003

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

SUBJECT: 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)

Dear Chairman Diaz:

During the 505th meeting of the Advisory Committee on Reactor Safeguards, September 10-13, 2003, we met with representatives of the NRC staff and the Nuclear Energy Institute to discuss the draft final Regulatory Guide (RG) x.xxx on An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment (PRA) results for Risk-Informed Activities (formerly DG-1122). We also had the benefit of the documents referenced.

Recommendations and Conclusion

1. The draft final RG should be issued for trial use with an appropriate sample of pilot plants.

2. We agree with the staff's decision to develop a separate regulatory guide on how to perform sensitivity and uncertainty analyses.

3. Inadequate PRA scope and quality may significantly affect regulatory decisionmaking.

Discussion

In our April 21, 2003 report, we made several recommendations for improving DG-1122. In his June 4, 2003 response, the Executive Director for Operations (EDO) agreed with all of our recommendations except the inclusion of guidance on how to perform sensitivity and uncertainty analyses. The staff argues that the American Society of Mechanical Engineers (ASME) standard for PRA already requires such analyses and that it would be more appropriate to discuss methods for performing them in a separate regulatory guide. We were told by the staff that this guide may be available for our review in early 2004. We look forward to reviewing it.

We agree with the staff and industry that the draft final RG should be issued for trial use. During our meeting with the staff, we made several suggestions for improving some of the language of the guide, in particular the definition of the term "significant." The staff should consider those suggestions before issuing this guide.

In SECY-03-0122, the staff states that an industry peer review group used the ASME PRA standard as the basis for evaluating a plant-specific PRA. Members of that group commented that the standard had "raised the bar" with respect to PRA quality, although they did not necessarily believe that this was inappropriate. We have also heard in the past that our reports that address PRA quality "ratchet up" the PRA requirements. We believe that it is important to make our position clear.

Our recommendations for the improvement of PRA scope and quality are not intended to "raise the bar" capriciously, but are always focused on the impact of such improvements on the integrated decisionmaking process that utilizes risk information. For example, in our report dated May 16, 2003, we recommended that the assessment of uncertainties should include model uncertainties. Such uncertainties may be very large in some cases and may affect the PRA results and insights in a way that could impact the relevant decisionmaking processes. If these uncertainties are not addressed explicitly, their magnitude and potential impact may not be fully appreciated and, thus, the decisionmaking process may not be truly risk informed.

Although our recommendations for PRA improvements are always motivated by our desire to have robust regulatory decisions, we note that enhanced confidence in PRA quality contributes to the agency's performance goal of increasing public confidence in NRC regulatory processes.

Sincerely,

Mario V. Bonaca
Chairman

References:

1. Memorandum dated September 10, 2003, from Scott Newberry, Nuclear Regulatory Research, NRC, to John T. Larkins, Executive Director, ACRS, Subject: ACRS Review of Regulatory Guide, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," and the associated Standard Review Plan Chapter 19.1.
2. Memorandum dated July 18, 2003, for the Commissioners from William D. Travers, Executive Director for Operations, SECY-03-0122, Policy Issue Information, Status Report on Draft Regulatory Guide, DG-1122, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," and Draft Standard Review Plan Chapter 19.1, "Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities."
3. Letter dated June 4, 2003, from William D. Travers, Executive Director for Operations, NRC, to Mario V. Bonaca, Chairman, ACRS, Subject: Proposed Resolution of Public Comments on Draft Regulatory Guide (DG)-1122, "An Approach for Determining the

Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities."

4. American Society of Mechanical Engineers, ASME RA-S-2002, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," dated April 5, 2002.
5. Report dated May 16, 2003, from Mario V. Bonaca, Chairman, ACRS, to Nils J. Diaz, Chairman, NRC, Subject: Improvement of the Quality of Risk Information for Regulatory Decisionmaking.
6. Report dated April 21, 2003, from Mario V. Bonaca, Chairman, ACRS, to Nils J. Diaz, Chairman, NRC, Subject: Proposed Resolution of Public Comments on Draft Regulatory Guide DG-1122, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities."

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

September 22, 2003

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

SUBJECT: DRAFT FINAL REVISION 1 TO REGULATORY GUIDE 1.53, "APPLICATION OF THE SINGLE-FAILURE CRITERION TO SAFETY SYSTEMS"

Dear Chairman Diaz:

During the 505th meeting of the Advisory Committee on Reactor Safeguards, September 10-13, 2003, we met with representatives of the NRC staff and the Institute of Electrical and Electronics Engineers, Inc. (IEEE) to discuss the draft final Revision 1 to Regulatory Guide (RG) 1.53, "Application of the Single-Failure Criterion to Safety Systems." We also had the benefit of the documents referenced.

Recommendation

Revision 1 to RG 1.53, "Application of the Single-Failure Criterion to Safety Systems," should be issued.

Discussion

In June 1973, the NRC issued revision 0 to RG 1.53, "Application of the Single-Failure Criterion to Safety Systems," which describes acceptable methods for complying with the NRC's regulations for meeting the single-failure criterion in the electrical power, instrumentation, and control portions of nuclear power generating station safety systems. Revision 0 conditionally endorses IEEE Std 379-1972, "IEEE Trial-Use Guide for the Application of the Single-Failure Criterion to Nuclear Power Generating Station Protection Systems."

The NRC staff has never updated RG 1.53 as had been planned at the time of issuance. The IEEE revised and published new editions of Std 379 in 1977, 1988, 1994, and 2000. These later editions clarified and strengthened the procedure for a single-failure analysis and provided additional guidance to address single-failure analysis in designs that use digital computers. These editions also provided guidance for applying the single-failure criterion to shared systems on using a probabilistic assessment to determine whether certain failures and events can be excluded from a single-failure analysis.

Given the outdated guidance in RG 1.53, licensees have been using various editions of IEEE Std 379 when making modifications to their plants. Revision 1 to RG 1.53 endorses IEEE Std 379-2000, "IEEE Standard Application of the Single-Failure Criterion to Nuclear

Power Generating Station Safety Systems," which provides methods acceptable to the NRC staff for satisfying the NRC's regulations with respect to the application of the single-failure criterion to the electrical power, instrumentation, and control portions of nuclear power plant safety systems. Revision 1 to RG 1.53 should be issued.

Sincerely,

Mario V. Bonaca
Chairman

References:

(1) Memorandum dated July 11, 2003, from Ashok Thadani, Director, Office of Nuclear Regulatory Research, to John T. Larkins, Executive Director, ACRS, Subject: Revision 1 of Regulatory Guide 1.53, "Application of the Single-Failure Criterion to Safety Systems."

(2) U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Draft Regulatory Guide DG-1118, "Application of the Single-Failure Criterion to Safety Systems," May 2002.

(3) The Institute of Electrical and Electronics Engineers, Inc., IEEE Std 379-2000 (Revision of IEEE Std 379-1994), "IEEE Standard Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems," September 21, 2000.

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

September 24, 2003

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

SUBJECT: DRAFT FINAL REVIEW STANDARD FOR EXTENDED POWER UPRATES, RS-001

Dear Chairman Diaz:

During the 505th meeting of the Advisory Committee on Reactor Safeguards, September 10-13, 2003, we met with representatives of the NRC staff to discuss the draft final Review Standard for Extended Power Uprates, RS-001, that was prepared as indicated in SECY-02-0106. We also had the benefit of the documents referenced.

RECOMMENDATION AND CONCLUSION

1. The Review Standard should be released for use in review of future applications for extended power uprates.

2. We commend the staff for the development of an excellent review standard.

DISCUSSION

Power uprates have been of three general magnitudes: (1) measurement uncertainty recapture of 1 to 2 percent, (2) stretch uprates up to about 7 percent, and (3) extended power uprates up to 20 percent. This Review Standard is intended only for use in review of extended power uprate applications. The staff has assigned uprate reviews a high priority and considers them to be among the most significant current licensing actions. We agree with this assessment and reiterate our view that a Review Standard is essential for maintaining efficiency and thoroughness of the review process. In addition, the Review Standard can facilitate the transfer of knowledge from one generation of reviewers to the next through lessons learned, critiques, feedback, and future updates.

In several letters related to uprate applications, we recommended that the staff develop a Standard Review Plan for uprate reviews. These recommendations arose from our concerns about: (1) the potential for synergistic effects when uprates are combined with other plant licensing actions, (2) potential safety margin reductions, and (3) the adequacy of agency uprate review procedures. The staff documented a plan for uprate reviews in SECY-02-0106 dated June 14, 2002. In this document, the staff committed to prepare a review standard that would include: (1) a clear definition of the review scope, (2) references to existing review criteria, and (3) template BWR and PWR safety evaluations.

During our review, we identified two concerns. First, there was considerable variation from section to section in the requirements for independent calculations. Some sections even went so far as to state that independent calculations were not expected. This concern was resolved in the final standard by establishing guidance for when independent calculations are appropriate. Our second concern was that the criteria for integral system transient testing were vague. We agree with the final staff position that integral system transient testing should be performed unless licensees can provide an adequate justification for not performing them.

We have expressed a concern about synergistic or compounding effects of uprates with other regulatory actions. While such effects are difficult to identify explicitly, the application of the Review Standard will help call attention to such effects. This is particularly true for areas with materials concerns where flow accelerated corrosion, fluid structure interaction, fatigue, and stress corrosion cracking can interact and shorten component life.

Sincerely,

Mario V. Bonaca
Chairman

References:

1. Memorandum dated August 1, 2003, from Ledyard B. Marsh, Office of Nuclear Reactor Regulation, NRC, to John T. Larkins, ACRS/ACNW, transmitting Review Standard RS-001, "Review Standard for Extended Power Uprates," with public comments, ACRS Comments, and SRP Sections.
2. Memorandum dated July 9, 2001, from William D. Travers, Executive Director for Operations, NRC, for the Commissioners, SECY-01-0124, Subject: Power Uprate Application Reviews.
3. Memorandum dated December 20, 2001, from Annette L. Vietti-Cook, Secretary of the Commission, to John T. Larkins, ACRS, Subject: Staff Requirements - Meeting with ACRS December 5, 2001.
4. Memorandum dated June 14, 2002, from William D. Travers, Executive Director for Operations, NRC, for the Commissioners, SECY-02-0106, Policy Issue Information, Subject: Review of ACRS Recommendation for the Staff to Develop a Standard Review Plan for Power Uprate Reviews.

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

September 24, 2003

Dr. William D. Travers
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: PROPOSED RECOMMENDATIONS FOR RESOLVING GENERIC ISSUE 186, "POTENTIAL RISK AND CONSEQUENCES OF HEAVY LOAD DROPS IN NUCLEAR POWER PLANTS"

Dear Dr. Travers:

During the 505th meeting of the Advisory Committee on Reactor Safeguards, September 11-13, 2003, we met with representatives of the NRC staff to discuss the proposed recommendations for resolving Generic Issue 186, "Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants." We also had the benefit of the documents referenced.

CONCLUSIONS

(1) We concur with the staff's conclusion that regulatory action is warranted to reduce the number and potential severity of load drop events. While these events do not pose a high nuclear plant safety risk, they do raise significant concerns regarding worker safety.

(2) We concur with the following recommendations developed by the staff:

   (a) Evaluate the capability of rigging components and materials to withstand rigging errors.
   (b) Endorse American Society of Mechanical Engineers (ASME) NOG-1, "Rules for Construction of Overhead and Gantry Cranes" for Type 1 cranes.
   (c) Reemphasize the need to follow and enforce NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" Phase 1 guidelines and continue to assess implementation of heavy load controls in safety-significant applications through the Reactor Oversight Process.
   (d) Evaluate the need to establish standardized calculation methodologies for heavy load drops.

DISCUSSION

NUREG-1774, "Survey of Crane Operating Experience at U.S. Nuclear Power Plants from 1968 through 2002," summarizes the number, type, and severity of load drop events that continue to occur at operating plants. It also documents that human error and rigging deficiencies below the hook account for many of the observed load drop events. In addition, the report concludes that licensees could have reduced the frequency of crane operating events attributable to human error if they had focused appropriate attention on the crane operating practices described in NUREG-0612.

The NRC staff examined several of the more serious crane load drop events for possible inclusion in the Accident Sequence Precursor (ASP) Program, but none of those events exceeded the ASP screening risk threshold of 1×10^{-5} per reactor year. However, we are concerned that worker fatalities have occurred and we conclude that the proposed, measured regulatory attention is appropriate.

We concur with the staff's recommendation to endorse ASME NOG-1 for single-failure-proof cranes. This will clarify the requirements for the construction or upgrade of cranes to the single-failure-proof crane category, which is referred to in NUREG-0612.

The staff also found that load drop calculational methodologies, assumptions, and predicted consequences vary greatly from licensee to licensee. Accurate load drop analysis is essential to determine transport height and load path restrictions. Therefore, the staff recommends evaluating the need to establish standardized load drop calculation methodologies.

We would like to review the proposed resolution of Generic Issue 186.

Sincerely,

Mario V. Bonaca
Chairman

References:

1. Letter dated August 14, 2003, from Farouk Eltawlia, Office of Nuclear Regulatory Research, NRC, to Dr. John T. Larkins, Executive Director, ACRS, Subject: Proposed Regulation and Guidance Development Recommendations for Generic Issue 186, "Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants."
2. U.S. Nuclear Regulatory Commission, NUREG-1774, "A Survey of Crane Operating Experience of U.S. Nuclear Power Plants from 1968 through 2002," July 2003.
3. U.S. Nuclear Regulatory Commission, Generic Letter 85-11, "Completion of Phase II of Control of Heavy Loads at Nuclear Power Plants," NUREG 0612, June 28, 1985.
4. U. S. Nuclear Regulatory Commission, NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," July 1980.

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

September 30, 2003

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

SUBJECT: DRAFT FINAL REVISION 3 TO REGULATORY GUIDE 1.82, "WATER SOURCES FOR LONG-TERM RECIRCULATION COOLING FOLLOWING A LOSS-OF-COOLANT ACCIDENT"

Dear Chairman Diaz:

During the 505th meeting of the Advisory Committee on Reactor Safeguards, September 10-13, 2003, we met with representatives of the NRC staff to discuss the draft final Revision 3 to Regulatory Guide 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident" (Ref. 1). Our Subcommittee on Thermal-Hydraulic Phenomena also reviewed this matter during its meeting on August 19, 2003. We previously provided a letter, dated February 20, 2003, concerning an earlier draft of this guidance. Regulatory Guide 1.82 (RG 1.82) is being revised to enhance the debris blockage evaluation guidance for pressurized water reactors. We also had the benefit of the documents referenced.

Recommendations

1. Draft final Revision 3 to RG 1.82 should be issued in order to facilitate licensee response and the resolution of technical issues. In addition, the staff should carefully review implementing guidance being developed by the Nuclear Energy Institute (NEI) because of the issues identified, the complex phenomena involved, and the need for more accurate plant-specific assessments.

2. The knowledge base report (Ref. 2) is a compendium of research results relevant to the problem, but it is confusing and it cannot be used directly as guidance for the analysis of sump blockage. Acceptable methods should be developed for use in satisfying the functional requirements described in RG 1.82.

3. An adequate technical basis should be developed to resolve the issues related to chemical reactions.

4. The staff should consider the possibility that the uncertainties associated with the calculational methodology may be so large, or that strainers may prove to be so susceptible to debris blockage, that alternative solutions may be required to ensure long-term cooling. This might involve, for example, changing the types of insulation used within containment or implementing diverse means of providing long-term cooling.

5. The staff should investigate a risk-informed approach to sump screen blockage.

Conclusions

- The technical basis for analyzing the phenomena described in RG 1.82 is not mature, the available information is inconsistent, and the knowledge base is evolving. Therefore, it is likely that the licensees' responses will be disparate and difficult to evaluate unless more consistent guidance is developed.

- The zone of influence (ZOI) models need revision and resolution of inconsistencies.

- Neither RG 1.82 nor the knowledge base report (Ref. 2) gives adequate consideration to chemical reactions.

Discussion

The sump screen blockage issue has a long history, dating back to the 1979 unresolved safety issue (USI) A-43. More stringent requirements have been developed as incidents or new knowledge revealed a need. These are reflected in various Bulletins, Generic Letters, and earlier revisions to RG 1.82. The case of boiling water reactors (BWRs) was revisited after the resolution of USI A-43 in 1985 because of several events, such as the one at the Swedish Barsebäck Nuclear Plant, Unit 2, in 1992, which demonstrated that larger quantities of fibrous debris could reach the strainers than had been predicted by models and analysis methods developed for the resolution of USI A-43 (Ref. 2). The BWR issue was resolved by installing large-capacity strainers in response to Bulletins 93-02 and 93-03. The strainers were designed on the basis of a BWR Owners Group report, NEDO-32686, "Utility Resolution Guidance for ECCS Suction Strainer Blockage," November 1996, which was approved by the staff.

The results of recent parametric study (Ref. 3) of 69 pressurized water reactors (PWRs) revealed that following a large-break LOCA, sump screen blockage was very likely in 53 of them. The same report stated that preliminary findings suggest that two-phase jets with a stagnation pressure of 1400 psia can inflict significant damage at distances much farther away than those measured in either USI A-43 studies or BWR air-jet impact tests program. Recent research has led to the discovery that very thin beds of fibrous insulation of the order of 1/8 inch thickness, in combination with particulates, can effectively block a sump screen. A risk study that supported the parametric study suggested an increase in the total core damage frequency (CDF) of an order of magnitude or more (Ref. 2). These studies were qualified with the caveat that many features of the problem are plant specific and, therefore, must be evaluated at that level. There appears to be sufficient evidence that new NRC guidance is necessary and appropriate action by PWR licensees may be needed.

Revision 3 to RG 1.82 describes the functional performance requirements for water sources that support long-term cooling. It also describes the main phenomena that are to be considered in the analysis of the performance of these sources, although it makes only general reference to chemical phenomena that may be important. Revision 3 to RG 1.82 should be issued in order to facilitate licensee response and the resolution of technical issues.

NEI is developing an implementing guidance document for licensees. Because of the many phenomena involved, and the significant plant-dependent nature of their manifestation, the staff will have to carefully review the NEI guidance and may need to perform confirmatory research.

While the revised RG 1.82 provides an extensive description of the phenomena of interest, it has little to say about the methods to be used for analyzing such phenomena. The major source of information on possible approaches has been the knowledge base report (Ref. 2) prepared recently by the Los Alamos National Laboratory. While this report comprises a compendium of research results obtained over several decades, these results are sometimes inconsistent and some have been superseded by recent work. The report does not clearly identify which results are valid, does not resolve apparent inconsistencies in the various studies, does not present a synthesis of validated methodologies that can be applied to actual plants, and provides little perspective to guide the user in the choice of appropriate quantitative methods.

For example, the production of debris is considered to occur in a ZOI. This is a useful concept, but for practical purposes, quantitative methods for describing the ZOI are necessary and Reference 2 provides several conflicting approaches. On page 3-25 it states that in a conical jet the centerline stagnation pressure is essentially constant at a distance of about 5-7 pipe diameters, at approximately 2 ± 1 bars. Figure 3-17 shows stagnation pressures between 3.5 and 5.5 bars in the same region. Both of these results originate from methods developed to resolve USI A-43, which were found to underestimate the Barsebäck damage. Results of recent studies show a pressure of about 11 bars in this same region. Page 3-6 states that the ZOI associated with prototypic two-phase (steam-water) jets is larger than the ZOI indicated by air jet simulated tests. Combining this with the statement on page 38 of NUREG/CR-6762, Volume I that single-phase air jets inflict significant damage to fibrous insulation types at a distance of 60 pipe diameters, one would conclude that the zone of influence is much greater than indicated in Figure 3.17. If licensees were to use such disparate information, we would anticipate the same variability in application of methods that was apparent in the BWR submittals.

During our meetings, the staff stated that the ZOI could comprise a large fraction of the entire containment. This does not seem consistent with the rather small ZOI shown in Figure 3-18 of the knowledge base report (Ref. 2). This figure is based on a set of spheres with the same volume as the zones shown in Figure 3-17, which is claimed to be a conical jet model originating from the work (Ref. 4) on jet loads reported by the Sandia National Laboratories (SNL) in NUREG/CR-2913, Rev. 4. The figure does not appear in the SNL report, but is actually Figure 3.25 of NUREG-0897, "Containment Emergency Sump Performance: Technical Findings Related to Unresolved Safety Issue A-43," 1985 (Ref.5). Use of this figure for estimating loads on containment structures appears to be a result of a misapplication of the SNL work, which considered the impact of a two-phase jet, issuing from a round break of diameter (D), on a large flat target perpendicular to the axis of the jet and a distance (L) away. The pressure distribution was computed on the target, as a function of radial distance, (R), from the axis. The stagnation pressure on the axis was lower than the original stagnation pressure of the jet because a shock wave occurred before impact on the target. This shock wave was the only mechanism of energy dissipation. Figure 3.17 in the technical basis report (Ref. 2) was constructed from contours of constant pressure on the target as (L) was varied.

This approach to computing a ZOI has two major errors. The first is the use of pressure distribution on a flat target to characterize the pressure felt by an object (such as a pipe) inserted into the same flow field when the target is there. The pressure falls away from the stagnation point on the target because of the large velocity of the fluid along the plate. However, if a pipe were placed on or near the plate at some radius, the fluid coming to rest at the stagnation point on this pipe would achieve a high pressure, comparable with the stagnation pressure at the axis of the target, as it was brought to rest. Moreover, the fluid that is diverted by the plate and disperses to the sides over a cylindrically-shaped area still has a very high velocity. For example, Figures 4.10 to 4.14 of the SNL report (Ref. 4) show that, in this example with L/D =2, at a radius of 5 diameters, the fluid flowing along the plate has a speed of about 2500 ft/sec while the fluid flowing along the plate from which the jet issued has a speed of about 3500 ft/sec. This latter fluid has not suffered a shock and has lost none of its energy. The result is a disc-shaped jet with an area that is 80 times the area of the original jet issuing radially into the surrounding space. Should the part of the jet that has not passed through a shock strike an object, the pressure load, according to the SNL model, would only be mitigated by whatever shock wave occurred in front of that object. Should the jet be focused by passing between suitable structures, it could conceivably recover most of its original stagnation pressure of 150 bars. The point is that even if there is a flat target in front of the jet the loads on other structures are not determined solely by the pressure distribution on that target.

The second misuse of the SNL work is to interpret the contours of static pressure on the target plate as being representative of the stagnation pressure distribution in a jet when the plate is not there. The reduction in radial static pressure over the plate is determined by the radial velocity which is not the same as in a jet in the absence of a target. Moreover, the stagnation pressure distribution in the jet is what is needed to determine the maximum pressure on structures, not the static pressure, and it is uniform until the flow passes through a shock wave. In fact, with the assumptions of the original SNL model, the stagnation pressure is uniform everywhere, to any distance, until a shock wave is passed through by the fluid. To assess the pressure exerted on an object, one would have to compute the flow field for a free jet and evaluate the strength of the shock wave ahead of that object when placed in this field. In practice, in a real containment, there will be shock reflections from multiple objects, redirection of the flow, and possible refocusing of the energy.

Given these concerns, the NRC staff should reevaluate the basis for establishing a ZOI. That basis should be quantitatively related to actual damage observed in plants and in experiments designed to assess the actual damage observed in various flow fields. These events and experiments have been reported (Ref. 2) but have not been used to develop validated practical prediction methods.

Another concern is the lack of consideration given to chemical effects, in both RG 1.82 and the knowledge base report. A hot, acidic, borated, two-phase jet has the potential to react chemically with paints, coatings, insulation, and other materials, particularly those incorporating aluminum and zinc. When the hot, borated water drains to the pool, it is dosed with alkaline material to create a high pH in the pool. In the presence of zinc, this is known to lead to the production of zinc hydroxide with concomitant evolution of hydrogen. Results of some preliminary experiments performed by LANL indicate that several other precipitates may be formed, some of which have a gel-like or sticky consistency that could exacerbate the potential for screen blockage.

In addition, hydrogen evolution in the pool is likely to affect the settling of materials that are heavier than water. A zinc particle, for example, will sink in pure water; however, if a reaction produces hydrogen bubbles that stick to the surface of the zinc particle, the particle may become buoyant and rise to the surface, probably eventually sinking again as the bubbles are released, with the cycle repeating. Similarly, a sediment of fibrous debris could be rendered buoyant by gas bubbles released within it.

The chemical kinetics of the reactions of concern may be too slow to influence sump blockage. However, this needs to be shown by definitive analysis and testing. Moreover to the extent possible, such testing should be performed under the conditions expected in an actual plant.

RG 1.82 gives passing reference to chemistry in Sections 1.3.2.6 and 2.3.1.8, which state that debris created by the resulting containment environment (thermal and chemical) should be considered in the analysis. However, in response to a public comment, the staff acknowledged that there are no NRC-published references pertinent to consideration of these chemical reactions. While RG 1.82 discusses effects of buoyancy on debris transport, it does not mention buoyancy induced by the release of gas by chemical reactions.

The knowledge base report describes many experiments, most of which were conducted under laboratory conditions, designed to investigate the transport of debris. These are useful sources of information; however, the report presents many qualifications of these results, particularly in view of the variety of phenomena involved in an actual plant. For example, one area of concern is the potential for debris to block flow paths to the sump before reaching the pool; these paths are numerous and vary significantly from plant to plant.

Knowledge about the head loss to be expected on sump screens is evolving, with recognition that the combination of fibrous and particulate materials can produce unusual effects. Again, this knowledge base needs to be consolidated into a form that is less susceptible to misinterpretation by readers. For instance, page 7-6 of the knowledge based report (Ref. 2) states that the NUREG/CR-6224 correlation will need considerable modification, whereas page 7-29 appears to endorse the same correlation with the statement that its predictions were within $\pm$ 25% of the test data.

There is also a need to synthesize this information into practical methods of prediction. The forthcoming NEI guidance should help in this regard.

As we discussed in our letter dated February 20, 2003, there is a possibility that the assessment of the blockage of the sump strainer may be subject to such large uncertainties as to be intractable, and alternative solutions may be required to ensure long-term cooling. These might involve, for example, using active sump screen systems, changing the types of insulation used within containment, or implementing diverse means of providing long-term cooling, including using additional water sources to extend the injection phase. Section 1.1.4 of Revision 3 to RG 1.82 discusses the use of active sump screen systems, but these may be only one of several possible alternatives that should be considered to ensure long-term cooling.

PWR sump blockage is an issue for which the design-basis accident approach may lead to unnecessary conservatism. A risk-informed approach may be appropriate in which the design-basis requirement to maintain effective long-term recirculation cooling would be retained, but risk information would be used to establish an acceptable approach to comply with the requirements.

The quantification of the sump blockage issue is an excellent example of where risk information can be applied to design-basis accident issues to the benefit of the public and the licensees. The staff should explore the feasibility of a risk-informed approach to sump screen blockage.

Sincerely,

Mario V. Bonaca

Mario V. Bonaca
Chairman

References:
1. U.S. Nuclear Regulatory Commission, Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident, Draft Regulatory Guide 1.82, Revision 3, August 2003
2. Rao, D.V., et al., Knowledge Base for the Effect of Debris on Pressurized Water Reactor Emergency Core Cooling Sump Performance, NUREG/CR-6808, LA-UR-03-0880, Los Alamos National Laboratory, February 2003
3. Rao, D.C., B.C. Letellier, C. Shaffer, S. Ashbauch, and L.S. Bartlein, GSI-191 Technical Assessment: Parametric Evaluations for Pressurized Water Reactor Recirculation Sump Performance, NUREG/CR-6762, Vol. 1, LA-UR-01-4083, Los Alamos National Laboratory, August 2002
4. Weigand, G., et al., Two Phase Jet Loads, NUREG/CR-2913 Rev. 4, Sandia National Laboratories, January, 1983
5. U.S. Nuclear Regulatory Commission, Containment Emergency Sump Performance, NUREG-0897, Rev 1, Nuclear Regulatory Commission, October 1985

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

October 7, 2003

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: DRAFT REGULATORY GUIDE DG-1129, "CRITERIA FOR INDEPENDENCE OF ELECTRICAL SAFETY SYSTEMS" (REVISION 3 TO REGULATORY GUIDE 1.75)

During the 506th meeting of the Advisory Committee on Reactor Safeguards, October 1-4, 2003, the Committee considered the draft Regulatory Guide DG-1129, "Criteria for Independence of Electrical Safety Systems," and decided to review it after reconciliation of public comments. The Committee agrees with the staff's proposal to issue this Guide for public comment.

Reference:
Memorandum dated October 1, 2003, from Michael E. Mayfield, RES, to John T. Larkins, ACRS, Subject: Draft Regulatory Guide, DG-1129, "Criteria for Independence of Electrical Safety Systems" (Revision 3 to Regulatory Guide 1.75).

cc: A. Vietti-Cook, SECY
W. Dean, OEDO
I. Schoenfeld, OEDO
A. Thadani, RES
J. Dyer, NRR
M. Mayfield, RES
S. K. Aggarwal, RES
J. Mitchell, RES

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

October 8, 2003

Dr. William D. Travers
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: DRAFT FINAL REGULATORY GUIDE 1.168, REVISION 1, "VERIFICATION, VALIDATION, REVIEWS, AND AUDITS FOR DIGITAL COMPUTER SOFTWARE USED IN SAFETY SYSTEMS OF NUCLEAR POWER PLANTS"

Dear Dr. Travers:

During the 506th meeting of the Advisory Committee on Reactor Safeguards, October 1-3, 2003, we met with representatives of the NRC staff to discuss draft final Regulatory Guide 1.168, Revision 1, "Verification, Validation, Reviews, and Audits for Digital Computer Software Used in Safety Systems of Nuclear Power Plants." We also had the benefit of the documents referenced.

RECOMMENDATION

Revision 1 to Regulatory Guide 1.168 should be issued final.

DISCUSSION

Revision 1 to Regulatory Guide 1.168 addresses quality assurance for digital computer software in safety-related systems of nuclear power plants. This Guide endorses the Institute of Electrical and Electronics Engineers (IEEE) Standards 1012-1998 and 1028-1997 with minor exceptions. IEEE Standard 1012 addresses Verification and Validation, and IEEE Standard 1028 addresses Reviews and Audits. We agree with the staff that updating Regulatory Guide 1.168 by endorsing the new version of the IEEE Standards will simplify the staff's review process and enable licensees and applicants to develop a unified coherent means of meeting the requirements of 10 CFR Part 50.

The proposed revision to Regulatory Guide 1.168 was issued for public comment in January 2003 as Draft Regulatory Guide DG-1123. The comments received from members of the public have been appropriately resolved by the staff. We agree with the staff's position that

Regulatory Guide 1.168, Revision 1, has neither backfit nor substantive policy implications. This Guide should be issued.

Sincerely,

Mario V. Bonaca
Chairman

References:
1. Memorandum dated August 15, 2003, from Michael Mayfield, Office of Nuclear Regulatory Research, to John T. Larkins, ACRS, Subject: Request for ACRS review of Draft Final Revision 1 of DG-1123, "Verification, Validation, Reviews, and Audits for Digital Computer Software Used in Safety Systems of Nuclear Power Plants."
2. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.168, Revision 1, "Verification, Validation, Reviews, and Audits for Digital Computer Software Used in Safety Systems of Nuclear Power Plants," August 2003.
3. Institute of Electronics and Electrical Engineers (IEEE) Standard-1012-1998, "IEEE Standard for Software Verification and Validation."
4. Institute of Electronics and Electrical Engineers (IEEE) Standard-1028-1997, "IEEE Standard for Software Reviews."

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

October 9, 2003

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 FORT CALHOUN STATION, UNIT 1

Dear Chairman Diaz:

During the 506th meeting of the Advisory Committee on Reactor Safeguards on October 1-3, 2003, we completed our review of the License Renewal Application (LRA) for the Fort Calhoun Station, Unit 1 and the related final Safety Evaluation Report (SER) prepared by the NRC staff. Our Plant License Renewal Subcommittee reviewed this application and the staff's initial SER during a meeting on June 11, 2003. During our review, we had the benefit of discussions with representatives of the NRC staff and Omaha Public Power District (OPPD or the applicant). We also had the benefit of the documents referenced.

CONCLUSION AND RECOMMENDATION

1) The programs instituted by OPPD to manage age-related degradation are appropriate and provide reasonable assurance that Fort Calhoun can be operated in accordance with its current licensing basis for the period of extended operation without undue risk to the health and safety of the public.

2) The OPPD application for renewal of the operating license for Fort Calhoun should be approved.

BACKGROUND AND DISCUSSION

This report fulfills the requirement of 10 CFR 54.25, which states that the ACRS should review and report on all license renewal applications. Fort Calhoun is a single unit, 1500 MWt Combustion Engineering pressurized water reactor. In its application, OPPD requested renewal of the operating license for Fort Calhoun for 20 years beyond the current license term, which expires August 9, 2013. The Fort Calhoun LRA is the first to be prepared in accordance with the Generic Aging Lessons Learned report.

The Fort Calhoun final SER documents the results of the staff's review of the information submitted by the applicant, including commitments that were necessary to resolve open items identified by the staff in the initial SER. In particular, the staff reviewed the completeness of the applicant's identification of structures, systems, and components (SSCs) that are subject to aging management; the integrated plant assessment process; the applicant's identification of the possible aging mechanisms associated with passive, long-lived components; and the adequacy of the applicant's aging management programs.

The staff also conducted inspections at Fort Calhoun, including an audit of the adequacy of the scoping and screening methodology and its implementation to ensure that SSCs within the scope of license renewal have been appropriately identified; an inspection of the aging management programs to confirm that existing programs are functioning well and to examine the applicant's plans for establishing new and enhanced aging management programs; and a walkdown of plant systems to assess how the systems are being maintained.

On the basis of our review of the final SER, the LRA, and the inspection reports, we conclude that the process implemented by the applicant to identify SSCs that are within the scope of license renewal was effective, the applicant performed a comprehensive aging management review of such SSCs, and the staff and the applicant appropriately identified all SSCs that are within the scope of license renewal. We agree with the staff's conclusion that all open and confirmatory items have been closed appropriately and there are no issues that preclude renewal of the operating license for Fort Calhoun.

Buckling of the containment liner plate has occurred in a small localized area. The applicant has analyzed this condition and concluded that this buckling does not affect the functionality of the containment liner plate. We agree with the staff that this issue is not an unanalyzed age-related issue.

The Fort Calhoun Alloy 600 Inspection Program includes provisions and commitments for inspecting reactor pressure vessel (RPV) head penetration nozzles. The applicant has performed bare metal visual inspection of the RPV head and found no evidence of leakage. The applicant intends to replace the RPV head, pressurizer, and steam generators in 2006. The applicant will continue to participate in the industry program for assessing and managing primary water stress corrosion cracking (PWSCC) of Alloy 600 and Alloy 82/182 welds, and has committed to perform inspections as recommended by this program. Based on the applicant's responses to related NRC bulletins and its commitment to participate in the industry's program for assessing and managing PWSCC of the RPV head penetration nozzles, there is reasonable assurance that the integrity of the RPV head will be adequately monitored and maintained.

Between 1988 and 1990, the Fort Calhoun Thermal Shield Monitoring Program identified loosening of the positioning pins for the thermal shield. During the 1992 refueling outage, seven lower and four upper pins were replaced. These actions reduced vibrations back to normal levels, and no abnormal vibration has been detected since 1992. In order to manage loss of preload of the positioning pins during the period of extended operation, the applicant has included the existing Thermal Shield Monitoring Program in the Reactor Vessel Internal Inspection Program. Based on the past success of the Thermal Shield Monitoring Program in detecting loss of preload, the applicant has not supplemented this Program with a loose parts monitoring program. We agree with the applicant and the staff that a loose parts monitoring program for thermal shield bolting is not required because the Thermal Shield Monitoring Program has been shown to be capable of early identification of loss of preload so as to preclude potential damage to the RPV internals.

The applicant and the staff have identified plausible aging effects associated with passive, long-lived components. Adequate programs have been established to manage the effects of aging so that Fort Calhoun 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.

Sincerely,

Mario V. Bonaca
Chairman

References:
1. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the License Renewal of Fort Calhoun Station, Unit 1," September 2003.
2. Letters dated January 9, 2002 and April 5, 2002, from W. G. Gates, Omaha Public Power District to U.S. Nuclear Regulatory Commission, transmitting the Application to Renew the Operating License of Fort Calhoun Station, Unit 1.
3. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report with Open Items Related to the License Renewal of Fort Calhoun Station, Unit 1," April 2003.
4. NRC Inspection Report 50-285/02-07, "Scoping and Screening," dated December 20, 2002.
5. NRC Inspection Report 50-285/03-07, "Aging Management Program Review," dated March 20, 2003.
6. Fort Calhoun Station, Unit 1 License Renewal Audit Report, dated April 9, 2003.

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

November 12, 2003

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: DRAFT FINAL REVISION TO 10 CFR PART 50, "FINANCIAL INFORMATION REQUIREMENTS FOR APPLICATIONS TO RENEW OR EXTEND THE TERM OF AN OPERATING LICENSE FOR A POWER REACTOR"

During the 507th meeting of the Advisory Committee on Reactor Safeguards, November 5-7, 2003, the Committee considered the subject rulemaking and decided not to review it. The Committee does not have any objection to the staff's proposal to issue the final rule.

Reference:
Memorandum dated October 3, 2003, from David B. Matthews, Office of Nuclear Reactor Regulation, to John T. Larkins, Advisory Committee on Reactor Safeguards, Subject: Request for Review of Final Rule, 10 CFR Part 50, "Financial Information Requirements for Applications to Extend or Renew the Term of an Operating License for a Power Reactor."

cc: A. Vietti-Cook, SECY
I. Schoenfeld, OEDO
D. B. Matthews, NRR
C. Haney, NRR
G. Mencinsky, NRR
M. Crutchley, NRR

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

November 13, 2003

Dr. William D. Travers
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Dear Dr. Travers:

SUBJECT: ACRS REVIEW OF ROUTINE UPDATES TO 10 CFR 50.55a, "CODES AND STANDARDS"

During the 507th meeting of the Advisory Committee on Reactor Safeguards, November 5-7, 2003, we considered a request from the Office of Nuclear Reactor Regulation (NRR) with regard to the timing of our review of routine updates to 10 CFR 50.55a. Specifically, NRR requests that in the future ACRS defer its review of routine updates to 10 CFR 50.55a until after public comments have been resolved. Our understanding is that routine updates are those that do not involve significant issues. We agree with NRR's request. We would like to be notified of routine updates to 10 CFR 50.55a at the draft proposed stage.

We also considered a proposed update to 10 CFR 50.55a, which incorporates by reference the 2001 Edition and the 2002 and 2003 Addenda of Division 1 of Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code. The Committee decided not to review it.

Sincerely,

Mario V. Bonaca
Chairman

Reference:
Memorandum dated October 27, 2003, from R. William Borchardt, Deputy Director, NRR, to John T. Larkins, Executive Director, ACRS, Subject: Advisory Committee on Reactor Safeguards Review of Routine Title 10 of the Code of Federal Regulations (10 CFR) 50.55a Rulemakings

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

November 17, 2003

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

Dear Chairman Diaz:

SUBJECT: PROPOSED RESOLUTION OF GENERIC SAFETY ISSUE-189, "SUSCEPTIBILITY OF ICE CONDENSER AND MARK III CONTAINMENTS TO EARLY FAILURE FROM HYDROGEN COMBUSTION DURING A SEVERE ACCIDENT"

During the 507th meeting of the Advisory Committee on Reactor Safeguards, November 5-7, 2003, we reviewed the recommendation proposed by the Office of Nuclear Reactor Regulation (NRR) to resolve Generic Safety Issue (GSI)-189, "Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident." During this review, we had the benefit of discussions with representatives of the NRC staff, the BWR Owners' Group, Duke Energy, and the Union of Concerned Scientists. This matter was previously discussed with the Office of Nuclear Regulatory Research (RES) during our June and November 2002 meetings. We also had the benefit of the documents referenced.

Conclusions and Recommendations

NRR should proceed with rulemaking to require a backup power supply to the hydrogen igniters for PWR Ice Condenser and BWR Mark III plants. The requirement should be for a pre-staged small generator with installed cables, conduit, panels, and breakers, or an equivalent diverse power supply.

Discussion

In our June and November 2002 meetings, the staff had communicated that further action was justified on a defense-in-depth basis and that the preliminary recommended options were for either a small portable generator and cabling or a pre-staged small generator with installed cables, conduit, panels, and breakers. In our report of November 13, 2002, we agreed with the staff that backup power for the hydrogen igniters as a safety enhancement was justified on a defense-in-depth basis, and we suggested that NRR investigate the viability of implementing backup power requirements through plant-specific severe accident management guidelines (SAMGs).

In subsequent public meetings, licensees stated that implementing backup power requirements through SAMGs is not a viable option because power to the igniters will be needed sooner than could be provided by this option, and that the effort to use portable generators could be a significant distraction from more critical actions required of the operators.

We still agree with the staff's assessment that backup power is an appropriate defense-in-depth safety enhancement and, in light of the industry's assessment of the viability of portable generators, we conclude that the appropriate option is to require a pre-staged small generator with installed cables, conduit, panels and breakers, or an equivalent diverse power supply. We agree with an industry view that the rulemaking should be accompanied by guidance that specifies the design requirements.

Sincerely,

Mario V. Bonaca

Mario V. Bonaca
Chairman

<u>References</u>:

1. Memorandum dated September 30, 2003, from Suzanne C. Black, Division Director, NRR, to John T. Larkins, Executive Director, ACRS, Subject: Background Information for Presentation Regarding Generic Safety Issue-189, Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident.
2. Letter dated October 23, 2003, from Kenneth S. Putnam, Chairman, BWR Owners' Group, to Document Control Desk, U.S. Nuclear Regulatory Commission, Subject: BWR Owners' Group Position on Issues Identified In Generic Safety Issue-189 and the Benefits and Cost of the Identified Alternatives To Resolving GSI-189 Concerns.

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

November 17, 2003

Dr. William D. Travers
Executive Director for Operations
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: DRAFT FINAL REVISION 3 OF REGULATORY GUIDE 1.32, "CRITERIA FOR POWER SYSTEMS FOR NUCLEAR POWER PLANTS"

Dear Dr. Travers:

During the 507th meeting of the Advisory Committee on Reactor Safeguards, November 5-7, 2003, we met with representatives of the NRC staff to discuss the draft final Revision 3 of Regulatory Guide 1.32, "Criteria for Power Systems for Nuclear Power Plants." We also had the benefit of the documents referenced.

RECOMMENDATION

Revision 3 of Regulatory Guide 1.32 should be issued.

DISCUSSION

Revision 3 of Regulatory Guide 1.32 addresses an acceptable method for licensees to satisfy Criteria 17 and 18 of 10 CFR Part 50, Appendix A when designing, modifying, operating, testing, and documenting Class 1E power systems for nuclear power plants. This Guide endorses Institute of Electrical and Electronics Engineers (IEEE) Standard 308-2001, "Criteria for Class 1E Power Systems for Nuclear Power Generating Stations," with one exception. IEEE 308-2001 states that shared Class 1E (safety-related) power systems are permissible in multi-unit stations provided certain rigorous conditions for sharing are met. The staff took exception to sharing dc power systems because Regulatory Guide 1.81, "Shared Emergency and Shutdown Electric Systems for Multi-Unit Nuclear Power Plants," states that dc power systems in multi-unit nuclear power plants should not be shared. We agree with the staff's exception to IEEE Standard 308-2001.

Proposed Revision 3 of Regulatory Guide 1.32 was issued for public comment in May 2003 as Draft Regulatory Guide DG-1079. One comment was received and resolved satisfactorily by the staff. Revision 3 of Regulatory Guide 1.32 should be issued.

Sincerely,

Mario V. Bonaca

Mario V. Bonaca
ACRS Chairman

References:

1. Memorandum dated October 1, 2003, from Michael E. Mayfield, Office of Nuclear Regulatory Research, NRC, to Dr. John T. Larkins, Executive Director, ACRS, transmitting Regulatory Guide 1.32, Revision 3, "Criteria for Power Systems for Nuclear Power Plants," September 2003.
2. Institute of Electrical and Electronics Engineers (IEEE) Standard 308-2001, "Criteria for Class 1E Power Systems for Nuclear Power Generating Stations," published March 1, 2002.

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

November 18, 2003

Dr. William D. Travers
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Dr. Travers:

SUBJECT: REGULATORY EFFECTIVENESS OF UNRESOLVED SAFETY ISSUE A-45, "SHUTDOWN DECAY HEAT REMOVAL REQUIREMENTS"

During the 507th meeting of the Advisory Committee on Reactor Safeguards, November 5-7, 2003, we reviewed NUREG/CR-6832, "Regulatory Effectiveness of Unresolved Safety Issue (USI) A-45, 'Shutdown Decay Heat Removal Requirements'." During our review, we had the benefit of discussions with representatives of the NRC staff. We also had the benefit of the documents referenced.

Conclusions and Recommendations

1. The assessment of the actions taken in the resolution of USI A-45 suggests that in most cases the associated risk is consistent with the NRC safety goals and defense-in-depth expectations. At 11 pressurized water reactors (PWRs) the risks associated with the loss of decay heat removal (DHR) are not consistent with the staff's expectations.

2. The staff should not continue to rely on the results of the Individual Plant Examinations (IPEs) to assess the effectiveness of NRC regulations. Either more access to current licensee risk information must be obtained or further efforts to upgrade the Standardized Plant Analysis Risk (SPAR) models should be pursued.

3. Assessment of the effectiveness of NRC regulations is important and should be continued.

Discussion

Failure of DHR systems can be a significant contributor to core damage frequency (CDF) and the frequency of large releases of radioactive material. In March 1981, the NRC designated "Shutdown Decay Heat Removal Requirements" as USI A-45. The staff concluded that risks due to loss of DHR could be "unduly" high for some plants. However, DHR vulnerabilities are very plant specific and detailed plant-specific analyses were needed to resolve this issue. Rather than develop a separate program to analyze DHR vulnerabilities, the staff decided to include these analyses in the IPE program.

The staff expectations for adequate resolution of USI A-45 were expressed in risk terms. The goal was to ensure that the contribution of DHR events to CDF was not unduly large compared to the safety goal for CDF. In NUREG-1289, the staff proposed a classification scheme for susceptibility to DHR events, as shown in the table below.

Category	Classification of Level 1 DHR Vulnerability	Criterion(/RY)
C1	Frequency of core damage due to failures of DHR function acceptably small, or reducible to an acceptable level by simple improvements	Less than 3.0E-05
C2	DHR performance characteristics intermediate between Categories 1 and 3	Less than 3.0E-04 but greater than 3.0E-05
C3	Frequency of core damage so large that prompt action to reduce the probability of core damage to an acceptable level is necessary	Greater than 3.0E-04

The assessment report, NUREG/CR-6832, uses IPE results to classify operating plants according to this classification scheme. No reactors were found to be in Category C3. All boiling water reactors and the majority of PWRs were found to be in Category C1. Eleven PWRs, however, were found to be in Category C2.

Based on the results of this assessment, the staff concludes that a significant reduction in the risk associated with the loss of DHR was achieved as a result of plant changes from the implementation of regulatory initiatives such as USI A-44, "Station Blackout," USI A-46, "Seismic Qualification of Equipment in Operating Plants," Generic Issue 124, "Auxiliary Feedwater System Reliability," and Generic Letter 89-16, "Installation of a Hardened Wetwell Vent," as well as from modifications made during the development of the plant-specific IPEs and Individual Plant Examination of External Events.

The assessment report, however, does not discuss whether any follow-up actions are planned for the 11 PWRs in Category 2. Are additional plant-specific actions appropriate for these plants? Would more sophisticated analyses show that the estimates based on the IPEs are overly conservative? Is it possible to make independent assessments of these plants with SPAR models?

Additional analyses were also performed to evaluate the effect of the feed-and-bleed capability on reducing plant CDF. The change in CDF ranged from 2.20E-05/RY to 8.60E-05/RY for the four plant models examined. The results confirm that the feed-and-bleed capability is very important in many PWRs in assuring adequate response to DHR events. However, use of feed-and-bleed is clearly a last resort. In addition, there is a limited time window in which the decision to use feed-and-bleed must be made. During our meeting, the staff could not provide assurance that the feed-and-bleed capabilities had been adequately evaluated. However, the staff did state that it planned to do some additional analyses of feed-and-bleed to ensure that realistic success criteria have been used in the SPAR models. This is helpful, but does not directly address the problem of ensuring that the licensees' evaluations have been realistic.

The assessment was based on results from the IPEs. The IPEs were primarily intended to assess severe accident vulnerabilities, and do not necessarily provide realistic estimates of CDF even for internal events. Better risk information is needed to more realistically assess the effectiveness of the regulations. One possibility is that the staff must have more access to licensee PRAs. There may be problems using this information for regulatory purposes since the staff has not performed a comprehensive review of all of the licensee PRAs, although virtually all licensee PRAs have now been through an industry peer review. However, a similar criticism can be made about the IPEs which were not subjected to complete review. Another possibility is further upgrading the capability of the SPAR models. Some (about 20) of these models have recently been upgraded for use in developing an integrated safety indicator. These models have been benchmarked against licensee models, and we have been informed that either the results are now in good agreement or the staff understands the reasons for the differences and has decided to use its own models. Upgrading the remainder of the SPAR models would give the staff an independent capability to assess the effectiveness of current and proposed regulations, and to improve the significance determination process.

Sincerely,

Mario V. Bonaca
Chairman

References:
1. U. S. Nuclear Regulatory Commission, NUREG/CR-6832, "Regulatory Effectiveness of Unresolved Safety Issue (USI) A-45, 'Shutdown Decay Heat Removal Requirements'," August 2003.
2. U. S. Nuclear Regulatory Commission, NUREG-1289, "Regulatory and Backfit Analysis: Unresolved Safety Issue A-45, Shutdown Decay Heat Removal Requirements," November 1988.

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

December 9, 2003

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: PROPOSED REVISIONS TO REGULATORY GUIDES

During the 508th meeting of the Advisory Committee on Reactor Safeguards, December 3-5, 2003, the Committee considered proposed revisions to the following Regulatory Guides and decided not to review them:

(1) Revision 33 to Regulatory Guide 1.84, (DG-1124), "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III";

(2) Revision 14 to Regulatory Guide 1.147, (DG-1125), "Inservice Inspection Code Case Acceptability, ASME Section XI";

(3) Revision 1 to Regulatory Guide 1.193, (DG-1126), "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 may review the draft final version of these Regulatory Guides following the reconciliation of public comments.

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

Reference:
Memorandum dated November 23, 2003, from Michael E. Mayfield, RES to John T. Larkins, ACRS, Subject: Review of Draft ASME Code Case Regulatory Guides.

cc: A. Vietti-Cook, SECY
W. Dean, OEDO
I. Schoenfeld, OEDO
A. Thadani, RES
M. Mayfield, RES
J. Mitchell, RES
W. Norris, RES
J. Dyer, NRR
D. Matthews, NRR
M. G. Crutchley, NRR
H. Tovmassian, NRR

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

December 10, 2003

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: PROPOSED RULE: FITNESS FOR DUTY PROGRAMS, 10 CFR PART 26

During the 508th meeting of the Advisory Committee on Reactor Safeguards, December 3-5, 2003, the Committee considered the proposed rule, "Fitness for Duty Programs, 10 CFR Part 26". The Committee has no objection to the staff's proposal to issue this rule for public comment. The Committee would like the opportunity to review the draft final rule after reconciliation of public comments.

Reference:
Memorandum dated November 17, 2003, from Catherine Haney, Program Director, Office of Nuclear Reactor Regulation, to John T. Larkins, Executive Director, Advisory Committee on Reactor Safeguards, Subject: Proposed Rule: Fitness for Duty Programs, 10 CFR Part 26.

cc: A. Vietti-Cook, SECY
W. Dean, OEDO
I. Schoenfeld, OEDO
M. Crutchley, NRR
C. Haney, NRR
R. Karas, NRR
G. West, NSIR
J. Mitchell, RES

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

December 12, 2003

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

Dear Chairman Diaz:

SUBJECT: DRAFT FINAL RULE REVISING 10 CFR 50.48, "FIRE PROTECTION," TO PERMIT LICENSEES TO VOLUNTARILY ADOPT FIRE PROTECTION REQUIREMENTS CONTAINED IN NATIONAL FIRE PROTECTION ASSOCIATION STANDARD 805 (NFPA 805)

During the 508th meeting of the Advisory Committee on Reactor Safeguards, December 3-5, 2003, we reviewed the draft final rule amending 10 CFR 50.48 to permit existing reactor licensees to voluntarily adopt fire protection requirements contained in National Fire Protection Association Standard 805 (NFPA 805), "Performance-Based Standard for Fire Protection for Light-Water Reactor Electric Generating Plants," 2001 Edition, as an alternative to the existing deterministic fire protection requirements. We had the benefit of the referenced documents and discussions with the NRC staff and representatives of the Nuclear Energy Institute (NEI) during two previous Fire Protection Subcommittee meetings held June 4, 2002 and September 9, 2003.

CONCLUSIONS AND RECOMMENDATIONS

1. The final rule amending 10 CFR 50.48 to permit licensees to voluntarily adopt fire protection requirements contained in NFPA 805 should be issued.

2. We agree that the staff should continue to work cooperatively with the industry to develop detailed guidance for the implementation of a risk-informed, performance-based fire protection program in accordance with NFPA 805.

DISCUSSION

Current fire protection requirements for nuclear power plants are deterministic. They are designed to ensure the post-fire survival of at least one set of safety systems that can be used to take the plant to cold shutdown. The requirements were developed before the NRC staff or the industry had the benefit of probabilistic risk assessments (PRAs) for fires and recent advances in fire modeling. Consequently, the current requirements are prescriptive and, due to their inflexibility, may create an unnecessary regulatory burden.

In SECY-98-058, "Development of a Risk-Informed, Performance-Based Regulation for Fire Protection at Nuclear Power Plants," dated March 26, 1998, the staff proposed to work with the NFPA and the industry to develop a risk-informed, performance-based consensus standard for fire protection at nuclear plants that could be used as an alternative to the current deterministic fire protection requirements. The Commission approved the proposal, and the staff began cooperative participation in the development of NFPA 805.

On January 13, 2001, the NFPA Standards Council approved NFPA 805, which specifies the minimum fire protection requirements for existing light-water nuclear power plants during all phases of plant operations, including shutdown and decommissioning. The standard describes a method for the use of risk-informed, performance-based approaches and fundamental fire protection design elements for establishing adequate fire protection procedures, systems, and features.

The staff believes that the methodology in NFPA 805, with certain exceptions noted in the proposed rule language, is an acceptable approach for satisfying existing fire protection requirements. The staff has proposed to incorporate NFPA 805 by reference into 10 CFR 50.48 as a voluntary alternative to existing requirements.

According to the staff projections, the implementation of a performance-based alternative would result in a reduction in future regulatory interactions associated with requests for license exemptions and deviations related to fire protection changes. It would also allow licensees and the staff to focus their attention and resources on the most risk-significant fire protection equipment and activities through more flexible, efficient, and rational processes. The staff should monitor inspection resources and expertise to ensure that appropriate inspection guidance and training are in place to support the effective inspection of the different approaches to fire protection (Appendix R, Branch Technical Position 9.5-1, License Condition and NFPA 805).

Since NFPA 805 primarily addresses technical issues and does not provide a framework or guidance pertaining to the regulatory process for plants choosing to adopt NFPA 805, NEI has volunteered to develop an implementing guide. NEI 04-01, "Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program Under 10 CFR 50.48(c)," is being developed and is expected to provide direction and clarification for plants choosing to adopt NFPA 805. The guide would further provide supplemental technical guidance and methods for demonstrating compliance with fire protection requirements.

The Committee agrees that the staff should continue to work with the industry to develop implementation guidance that includes instructions on transitioning to and administering a fire protection program consistent with NFPA 805 and that does not create unnecessary barriers to the use of the standard.

Sincerely,

Mario V. Bonaca

Mario V. Bonaca
Chairman

References:

1. Draft Federal Register Notice, Subject: Final Rule, Voluntary Fire Protection Requirements for Light Water Reactors; Adoption of NFPA 805 as a Risk-Informed, Performance-Based Alternative.
2. National Fire Protection Association (NFPA) Standard 805, "Performance-Based Standard for Fire Protection for Light-Water Reactor Electric Generating Plants," 2001 Edition.

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

December 12, 2003

Dr. William D. Travers
Executive Director for Operations
US Nuclear Regulatory Commission
Washington, DC 20555-0001

Dear Dr. Travers:

SUBJECT: DRAFT 10 CFR PART 52 CONSTRUCTION INSPECTION PROGRAM FRAMEWORK DOCUMENT

During the 508th meeting of the Advisory Committee on Reactor Safeguards on December 3-5, 2003, we met with representatives of the NRC staff to discuss the draft 10 CFR Part 52 Construction Inspection Program Framework Document. We also had the benefit of the document referenced.

CONCLUSIONS AND RECOMMENDATIONS

1. The framework document provides a good basis for the development of appropriate inspection manual chapters for the certification and licensing of new plants.

2. We commend the staff for developing the "sign-as-you-go" (SAYGO) and the Construction Inspection Program Information Management System (CIPIMS) concepts. These should help make the inspection process more efficient and effective.

3. We agree with the staff that the use of statistical sampling to limit the number of required inspections, tests, analyses, and acceptance criteria (ITAAC) inspections will be valid in only a few areas.

4. We recommend that the number of ITAACs that are subjected to minimal inspection be small.

DISCUSSION

The staff has developed this draft framework document to provide guidance on revising construction inspection manual chapters and inspection procedures to support the 10 CFR Part 52 licensing process. The framework document meets this objective and is well written and organized. The staff has done a commendable job of outlining the needs and the required processes.

The guidance includes a SAYGO phased verification process, which will document conclusions on individual ITAACs as they are completed. It also includes an electronic information tracking and scheduling system, the CIPIMS, to track all inspection findings, conclusions, and unresolved items. The combination of the two concepts should make the inspection process more efficient and effective. We commend the staff for including these innovative concepts in the program.

The staff has concluded that it will have insufficient resources and time to inspect all ITAACs in detail. Consequently, the staff proposes implementing a statistical sampling process to limit the number of inspections required to determine that all ITAACs have been satisfied to the desired level of confidence. The staff has noted that such a statistical sampling method will be valid only for limited ITAAC areas, but has not yet identified them. We agree that a statistical sampling inspection process will be valid only for areas where the ITAAC is related to a large number of nominally homogeneous items such as welds and certain repetitive components. We look forward to reviewing the final disposition of this concept.

Some ITAACs will not have received any NRC inspection directly related to that ITAAC or to a similar one. For such ITAACs, the staff will review the documentation associated with the licensee's declaration that the ITAAC has been satisfactorily completed, which is required for all ITAACs. We recommend that the number of ITAACs that are subjected to such minimal inspection be small.

"Negative SAYGO ITAAC Conclusions" are also discussed. Such a negative conclusion would reflect a decision that the staff could not make a positive interim ITAAC conclusion on a selected construction activity. In addition, such a conclusion would indicate that deficiencies in the construction activity were not addressed by the corrective action program. Licensees would be expected to identify specific corrective actions. The staff should also require the licensee to identify and correct the weakness in its corrective action program that led to the observed deficiency. The licensee should also be required to examine the root cause of the corrective action program weakness for generic implications for other Part 52 activities.

Sincerely,

Mario V. Bonaca

Mario V. Bonaca
Chairman

<u>Reference</u>:

U.S. Nuclear Regulatory Commission, Draft 10 CFR Part 52, "Construction Inspection Program Framework Document," May 2003.

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

December 12, 2003

Dr. William D. Travers
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Dear Dr. Travers:

SUBJECT: DRAFT NUREG-0800, STANDARD REVIEW PLAN (SRP), CHAPTER 18.0, HUMAN FACTORS ENGINEERING

During the 508th meeting of the Advisory Committee on Reactor Safeguards on December 3-5, 2003, we met with representatives of the NRC staff to discuss the updates to Chapter 18.0 of the SRP and the documents referenced in that chapter: NUREG-0711, Rev. 2, "Human Factors Engineering Program Review Model," for guidance on the review process; NUREG-1764, "Guidance for the Review of Changes to Operator Actions," for criteria tailored to plant modifications and license amendment requests involving credited operator actions; and NUREG-0700, Rev. 2 "Human-System Interface Design Review Guidelines," for guidance concerning human-system interfaces. Our Subcommittee on Human Factors reviewed this matter during a meeting held on December 2, 2003. We also had the benefit of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

1. The update to Chapter 18.0 of the SRP and the documents referenced in that chapter properly incorporate needed changes that facilitate anticipated reviews and clarify the human factors engineering review process.

2. The staff has developed in NUREG-1764 an innovative use of risk importance measures to screen licensee submissions for human factors review and to guide the depth and detail of these reviews. This significant development holds the promise of more effective use of staff resources and improved plant safety.

DISCUSSION

The staff has completed an update to Chapter 18.0 of the SRP and its associated documents (NUREG-0700, NUREG-0711, and NUREG-1764). This chapter of the SRP provides the framework for the conduct of human factors engineering reviews. This update is needed to support reviews of advanced reactors and digital upgrades to existing control rooms. Changes have been made to move review process guidelines from NUREG-0700 to NUREG-0711. In addition, the formats of the documents have been made consistent. NUREG-1764 is a new document that provides risk guidelines to enable a graded approach to determining the level of human factors review by the staff. These documents incorporate needed changes to facilitate anticipated reviews and to clarify the human factors engineering review process.

The staff has developed a method using risk importance measures to screen licensee submissions for human factors review. The method is applicable to new actions such as the substitution of manual activities for automated actions, changes in components affecting human performance, and changes in the environment for human performance. The screening method provides guidance on the level of detail for the human factors review merited by the submission.

The screening method is an innovative use of risk in the human factors arena and is being tested by NRR. We look forward to seeing how this testing progresses. The use of risk information in human factors reviews holds the promise of more efficient use of NRC resources to focus on issues of greatest risk significance and reduce the extent of staff reviews of human actions if these actions can be shown to have limited or no risk significance. It may well lead to improved plant safety.

Sincerely,

Mario V. Bonaca
Chairman

References:

1. ACRS report dated July 23, 2002, to Chairman Richard A. Meserve, Subject: Draft Final Revision 1 to Regulatory Guide 1.174 and to Chapter 19 of the Standard Review Plan.
2. ACRS letter dated September 24, 2002, to Dr. William D. Travers, Executive Director for Operations, Subject: Human Factors and Human Reliability Analysis Research Plans.
3. ACRS letter dated November 13, 1995, to Mr. James M. Taylor, Executive Director for Operations, Subject: NUREG-0700, Revision 1, "Human-System Interface Design Review Guidance."
4. U.S. Nuclear Regulatory Commission Standard Review Plan, NUREG-0800, Chapter 18.0, "Human Factors Engineering," Draft Revision 2, December 2003.
5. U.S. Nuclear Regulatory Commission, NUREG-0700, Revision 2, "Human-System Interface Design Review Guidelines," May 2002.
6. U.S. Nuclear Regulatory Commission, NUREG-0711, Revision 2, "Human Factors Engineering Program Review Model."
7. U.S. Nuclear Regulatory Commission, NUREG-1764, "Guidance for the Review of Changes to Human Actions," Final Report.

NRC FORM 335
(2-89)
NRCM 1102,
3201, 3202

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 25

2. TITLE AND SUBTITLE

A Compilation of Reports of the Advisory Committee on Reactor Safeguards: 2003 Annual

3. DATE REPORT PUBLISHED

MONTH	YEAR
April	2004

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

6. TYPE OF REPORT

Compilation

7. PERIOD COVERED *(Inclusive Dates)*

Jan. thru Dec. 2003

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 47 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 2003. In addition, a report to the Commission on the NRC Safety Research Program, NUREG-1635, Volume 5, is included by reference only. 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.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Core Power Uprate
Human Factors
License Renewal
Nuclear Reactors
Nuclear Reactor Safety
Reactor Operations
Safety Engineering
Safety Research

13. AVAILABILITY STATEMENT

Unlimited

14. SECURITY CLASSIFICATION

(This Page) Unclassified

(This Report) Unclassified

15. NUMBER OF PAGES

16. PRICE

NRC FORM 335 (2-89)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, DC 20555-0001

OFFICIAL BUSINESS