NUREG-1125
Volume 27

A Compilation of
Reports of

The Advisory Committee on Reactor Safeguards

2005 Annual

U. S. Nuclear Regulatory Commission

June 2006

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 27

A Compilation of
Reports of

The Advisory Committee on Reactor Safeguards

2005 Annual

U. S. Nuclear Regulatory
Commission

June 2006

ABSTRACT

This compilation contains 64 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 2005. 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.

<u>PREFACE</u>

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

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

<u>ACRS MEMBERSHIP (2005)</u>

CHAIRMAN:	Dr. Graham B. Wallis, Professor Emeritus Dartmouth College
VICE CHAIRMAN:	Dr. William J. Shack Argonne National Laboratory
MEMBERS:	Dr. George E. Apostolakis, Professor Massachusetts Institute of Technology
	Dr. Mario V. Bonaca, Retired Northeast Utilities
	Dr. Richard S. Denning Battelle Memorial Institute and Ohio State University
	Dr. F. Peter Ford, Retired General Electric Research and Development Center (Term ended 03/05)
	Dr. Thomas S. Kress, Retired Oak Ridge National Laboratory
	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 (Term ended 6/05)
	Mr. John D. Sieber, Retired Duquesne Light Company

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

Page

TABLE OF CONTENTS

TABLE OF CONTENTS

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

February 11, 2005

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: DRAFT REGULATORY GUIDE, DG-1137, "GUIDELINES FOR LIGHTNING PROTECTION FOR NUCLEAR POWER PLANTS"

During the 519th meeting of the Advisory Committee on Reactor Safeguards held on February 10-11, 2005, the Committee considered DG-1137, "Guidelines for Lightning Protection for Nuclear Power Plants." The Committee will consider reviewing the draft final version of this guide after reconciliation of public comments. The Committee has no objection to the staff's proposal to issue DG-1137 for public comment.

Reference:
Memorandum dated November 30, 2004, from Michael E. Mayfield, Director, Division of Engineering Technology, RES, to John T. Larkins, Executive Director, ACRS, Subject: Request to Defer the ACRS Review of Draft Guide, DG-1137, "Guidelines for Lightning Protection for Nuclear Power Plants."

cc: A. Vietti-Cook, SECY
W. Dean, OEDO
J. Dixon-Herrity, OEDO
J. Dyer, NRR
J. Craig, RES
M. Mayfield, RES
A. Levin, RES
C. Antonescu, RES

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

February 24, 2005

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

SUBJECT: WATERFORD STEAM ELECTRIC STATION, UNIT 3 - EXTENDED POWER UPRATE

Dear Chairman Diaz:

During the 519th meeting of the Advisory Committee on Reactor Safeguards, February 10-11, 2005, we met with representatives of the NRC staff and Entergy to review the utility's license amendment request for an increase in core thermal power for the Waterford Steam Electric Station, Unit 3. Our Subcommittee on Thermal-Hydraulic Phenomena also reviewed this matter during its meeting on January 26, 2005. During our review, we had the benefit of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

1. The application by Entergy for an 8% extended power uprate (EPU) at Waterford 3 should be approved, subject to (1) the staff's approval of the alternate source term (AST) application and (2) documentation of the resolution of the boron precipitation issue during long-term cooling for Waterford 3 by the submittal of the analysis details and their acceptance in the staff's safety evaluation (SE).

2. We agree with the staff that the requirement for large-transient testing should be waived for this application.

3. The staff should review the generic potential for boron concentration and precipitation to interfere with core cooling following a loss-of-coolant accident (LOCA).

DISCUSSION

Waterford 3 was originally licensed on March 16, 1985, at 3390 MWt. The current licensed power level of 3441 MWt includes a 1.5% measurement uncertainty uprate authorized on March 29, 2002. The present uprate request would raise the power 8% above the current level to 3716 MWt. The licensee plans to make all the changes during one outage and implement the uprate early in 2005, as soon as approval is received.

The Waterford uprate application follows a methodology similar to the one for the Arkansas Nuclear One, Unit 2, uprate, approved on April 24, 2002. This is the first application for which the staff has used the new uprate review standard (RS-001). The staff's review has been comprehensive and the rationale for the staff's decisions is clear in the SE.

The power uprate will be achieved by small changes in the hot and cold leg temperatures and in the circulating flow rate in the primary circuit. The operating pressure will not be changed. There will be an increase in the steam and feedwater flow rate on the secondary side. The number of fuel assemblies to be replaced at each refueling will increase roughly in proportion to the power uprate.

There will also be some modifications to balance-of-plant equipment. For example, the high-pressure turbine will be upgraded and higher capacity installed in the generator, switchgear, and main transformers, as needed. Condenser tubes will be stiffened to accommodate the higher steam flow.

To meet the control room habitability requirements, the licensee needs approval of an AST. The AST application is under review with scheduled completion by March, 2005. Although the Committee did not review the AST application, the staff anticipates a successful review.

In response to the licensee's request, the staff proposes to waive the requirement for large-transient testing at the new power level. The licensee will carry out a testing program for each of the planned modifications. Interactions among the modifications have been investigated through analytical modeling. The licensee argues that an integral large-transient test will not provide significant additional information. The staff believes that the proposed test program and previous operating experience will meet the objectives of confirming the functionality of equipment, codes and models, and emergency operating procedures. The potential value of a large-transient test is insufficient to justify imposing a trip event on the plant and the electrical grid.

Because of the increased steam flow associated with the power uprate, we sought evidence that the steam dryers would operate successfully. The licensee provided a detailed description of the construction and operation of these dryers. The flow rates and operating conditions expected after the power uprate are within the range previously tested, and the dynamic loads are lower than have been experienced, without untoward occurrences, at Palo Verde, where the dryer units and their supports are substantially the same as at Waterford. There is therefore a reasonable expectation that the dryers will operate successfully following the proposed power uprate.

The matter of boron concentration during long-term cooling was discussed during the Subcommittee meeting. The staff and licensee positions had not yet been resolved. We have since heard presentations from both the staff and the licensee.

The licensee and the staff have demonstrated by conservative analyses that there exists, at Waterford, a significant margin to the boron solubility limit. The final resolution of this issue needs to be documented in a revision to the application and in the SE.

These analyses provided assurance that long-term cooling can be successfully achieved at Waterford. However, there may be generic issues, not specific to power uprates, that are related to the precipitation of boric acid and its effects on long-term core cooling. Although the BACCHUS test results suggest that mixing may occur between the core and lower plenum and reduce the boron concentration in the core, there is no quantification of the mechanisms nor an assessment of how applicable the results are to the general case.

In discussing the boron precipitation issue, we also became aware that there is not a good technical basis for evaluating the properties of a boron-water mixture, together with chemicals added from the containment sump, when the concentration is close to the solubility limit. As this mixture boils, the solute may accumulate at the surface of bubbles and significantly change hydraulic properties such as the drift flux and foamability.

Our discussions also revealed that there is not a good understanding of the deposition of boron on the overheated portions of the fuel rods, which are predicted to be exposed for up to 45 minutes during some small-break LOCAs. Splashes and droplets of borated water may be deposited on the exposed fuel rods and spacer grids and the water will evaporate, leaving boric acid deposits that will decompose at the prevailing temperature to form dry boric oxide. We encourage the staff to establish a basis for a quantitative assessment of these phenomena as it considers the potential for boron concentration and precipitation to interfere with core cooling following a LOCA.

Additional comments by ACRS Members Stephen L. Rosen and F. Peter Ford are provided below.

Sincerely

Graham B. Wallis

Graham B. Wallis
Chairman

Additional Comments by ACRS Members Stephen L. Rosen and F. Peter Ford:

The licensee has argued that while integral large-transient tests (main turbine trip and generator breaker opening at 100% power) are safe, they are unnecessary. The licensee relies on computer modeling and previous operating experience at 100% (92% EPU) conditions to justify elimination of these tests.

Since integral tests of a plant's response to transient initiators can reveal otherwise undetected flaws, these tests should be conducted to confirm that plant modifications made to support the upgrades have been installed as designed and function properly in an integrated manner to bring the plant to safe and stable conditions. We are not convinced by the licensee's arguments and the staff's conclusion that integral tests are not necessary. An initial startup testing program limited to 92% of full power would not have been adequate. Similarly, we believe that approval of the EPU application should be conditioned on the successful completion of integral large-transient tests (main turbine trip and generator breaker opening at 100% power) shortly after reaching EPU conditions.

References:

1. Memorandum from Herbert N. Berkow to Ralph Caruso, dated January 10, 2005, "Waterford Steam Electric Station, Unit 3 (Waterford 3) - Draft Safety Evaluation (Version 2) for the Proposed Extended Power Uprate
2. Amendment Request NPF-38-249, Extended Power Uprate Waterford Steam Electric Station, Unit 3 Docket No. 50-382 License No. NPF-38, dated November 13, 2003
3. Supplement to Amendment Request NPF-38-249, Extended Power Uprate Waterford Steam Electric Station, Unit 3 Docket No. 50-382 License No. NPF-38, dated January 29, 2004

References (continued)

4. Supplement to Amendment Request NPF-38-249, Extended Power Uprate Waterford Steam Electric Station, Unit 3 Docket No. 50-382 License No. NPF-38, dated March 4, 2004
5. Supplement to Amendment Request NPF-38-249, Extended Power Uprate Waterford Steam Electric Station, Unit 3 Docket No. 50-382 License No. NPF-38, dated April 15, 2004
6. Supplement to Amendment Request NPF-38-249, Extended Power Uprate Waterford Steam Electric Station, Unit 3 Docket No. 50-382 License No. NPF-38, dated May 7, 2004
7. Supplement to Amendment Request NPF-38-249, Extended Power Uprate Waterford Steam Electric Station, Unit 3 Docket No. 50-382 License No. NPF-38, dated May 12, 2004
8. Supplement to Amendment Request NPF-38-249, Extended Power Uprate Waterford Steam Electric Station, Unit 3 Docket No. 50-382 License No. NPF-38, dated May 13, 2004
9. Supplement to Amendment Request NPF-38-249, Extended Power Uprate Waterford Steam Electric Station, Unit 3 Docket No. 50-382 License No. NPF-38, dated May 21, 2004
10. Supplement to Amendment Request NPF-38-249, Extended Power Uprate Waterford Steam Electric Station, Unit 3 Docket No. 50-382 License No. NPF-38, dated May 26, 2004
11. Supplement to Amendment Request NPF-38-249, Extended Power Uprate Waterford Steam Electric Station, Unit 3 Docket No. 50-382 License No. NPF-38, dated July 14, 2004
12. Supplement to Amendment Request NPF-38-249, Extended Power Uprate Waterford Steam Electric Station, Unit 3 Docket No. 50-382 License No. NPF-38, dated July 15, 2004
13. Supplement to Amendment Request NPF-38-249, Extended Power Uprate Waterford Steam Electric Station, Unit 3 Docket No. 50-382 License No. NPF-38, dated July 28, 2004
14. Supplement to Amendment Request NPF-38-249, Extended Power Uprate Waterford Steam Electric Station, Unit 3 Docket No. 50-382 License No. NPF-38, dated August 10, 2004
15. Supplement to Amendment Request NPF-38-249, Extended Power Uprate Waterford Steam Electric Station, Unit 3 Docket No. 50-382 License No. NPF-38, dated August 19, 2004
16. Supplement to Amendment Request NPF-38-249, Extended Power Uprate Waterford Steam Electric Station, Unit 3 Docket No. 50-382 License No. NPF-38, dated August 25, 2004

References (continued)

17. Supplement to Amendment Request NPF-38-249, Extended Power Uprate Waterford Steam Electric Station, Unit 3 Docket No. 50-382 License No. NPF-38, dated September 1, 2004
18. Supplement to Amendment Request NPF-38-249, Extended Power Uprate Waterford Steam Electric Station, Unit 3 Docket No. 50-382 License No. NPF-38, dated September 14, 2004
19. Supplement to Amendment Request NPF-38-249, Extended Power Uprate Waterford Steam Electric Station, Unit 3 Docket No. 50-382 License No. NPF-38, dated October 8, 2004
20. Supplement to Amendment Request NPF-38-249, Extended Power Uprate Waterford Steam Electric Station, Unit 3 Docket No. 50-382 License No. NPF-38, dated October 8, 2004
21. Supplement to Amendment Request NPF-38-249, Extended Power Uprate Waterford Steam Electric Station, Unit 3 Docket No. 50-382 License No. NPF-38, dated October 13, 2004
22. Supplement to Amendment Request NPF-38-249, Extended Power Uprate Waterford Steam Electric Station, Unit 3 Docket No. 50-382 License No. NPF-38, dated October 18, 2004
23. Supplement to Amendment Request NPF-38-249, Extended Power Uprate Waterford Steam Electric Station, Unit 3 Docket No. 50-382 License No. NPF-38, dated October 19, 2004
24. Supplement to Amendment Request NPF-38-249, Extended Power Uprate Waterford Steam Electric Station, Unit 3 Docket No. 50-382 License No. NPF-38, dated October 21, 2004
25. Supplement to Amendment Request NPF-38-249, Extended Power Uprate Waterford Steam Electric Station, Unit 3 Docket No. 50-382 License No. NPF-38, dated October 29, 2004
26. Supplement to Amendment Request NPF-38-249, Extended Power Uprate Waterford Steam Electric Station, Unit 3 Docket No. 50-382 License No. NPF-38, dated October 29, 2004
27. Supplement to Amendment Request NPF-38-249, Extended Power Uprate Waterford Steam Electric Station, Unit 3 Docket No. 50-382 License No. NPF-38, dated November 4, 2004
28. Supplement to Amendment Request NPF-38-249, Extended Power Uprate Waterford Steam Electric Station, Unit 3 Docket No. 50-382 License No. NPF-38, dated November 8, 2004
29. Supplement to Amendment Request NPF-38-249, Extended Power Uprate Waterford Steam Electric Station, Unit 3 Docket No. 50-382 License No. NPF-38, dated November 16, 2004
30. Supplement to Amendment Request NPF-38-249, Extended Power Uprate Waterford Steam Electric Station, Unit 3 Docket No. 50-382 License No. NPF-38, dated November 19, 2004

References (continued)

31. Review and Evaluation of MHI BACCHUS PWR Vessel Mixing Tests, WCAP-16317-P, November 2004 (Proprietary)
32. Memorandum from Herbert Berkow to Ralph Caruso, "Waterford Steam Electric Station, Unit 3 (Waterford 3) - Prevention of Boric Acid Precipitation in a Post-LOCA Long Term Cooling Mode for the Proposed Extended Power Uprate", February 1, 2005 (Proprietary)
33. J. Tuunanen, H. Tuomisto, and P. Raussi, "Experimental and Analytical Studies of Boric Acid Concentrations in a VVER-440 Reactor during the Long-term Cooling Period of Loss-of-Coolant Accidents", Nuclear Engineering and Design 148 (1994) 217-231

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

February 24, 2005

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

SUBJECT: REVIEW OF THE FINAL SAFETY EVALUATION REPORT FOR THE MIXED OXIDE FUEL FABRICATION FACILITY CONSTRUCTION AUTHORIZATION REQUEST

Dear Chairman Diaz:

During the 519th meeting of the Advisory Committee on Reactor Safeguards (ACRS), February 10-11, 2005, we met with representatives of the NRC staff and a representative of the Union of Concerned Scientists to discuss the Final Safety Evaluation Report for the Mixed Oxide (MOX) Fuel Fabrication Facility (MF^3) Construction Authorization Request submitted by Duke Cogema Stone & Webster (DCS) on February 28, 2001. Our review focused on safety issues and did not include questions of materials accountability and control or physical protection issues. We were joined in our reviews of this Final Safety Evaluation Report by members of the Advisory Committee on Nuclear Waste. This matter was also discussed with representatives of DCS and the NRC staff during the 504th and 507th meetings of the ACRS on July 9-11 and November 6-8, 2003, respectively, and at meetings held by the Reactor Fuels subcommittee on November 16, 2001, April 10, 2002, April 21, 2003, and December 15-16, 2004, as the Department of Energy's design requirements for the facility evolved. We also had the benefit of the documents referenced.

CONCLUSION AND RECOMMENDATION

The information from DCS on the safety of construction, maintenance, and operation of the Mixed Oxide Fuel Fabrication Facility at the Savannah River Site provides sufficient assurance to proceed with construction and an integrated safety analysis. The Final Safety Evaluation Report for the Mixed Oxide Fuel Fabrication Facility Construction Authorization Request should be issued.

BACKGROUND

The Mixed Oxide Fuel Fabrication Facility, MF^3, is to manufacture mixed oxide (plutonium dioxide - uranium dioxide) reactor fuel for use in the Catawba and McGuire commercial nuclear power reactors. The facility is being developed as part of the national strategy to dispose of excess weapons-grade plutonium (predominantly ^{239}Pu) by using this plutonium for power production. MF^3 will be located on the Department of Energy's Savannah River Site near Augusta, Georgia, and Aiken, South Carolina.

The MF^3 will receive slightly contaminated, weapons-grade plutonium dioxide and other feeds from a Department of Energy facility to be built on the Savannah River Site. At MF^3, this plutonium dioxide will be dissolved, purified by solvent extraction, precipitated, converted back to plutonium dioxide, and blended to yield a solid solution with uranium dioxide. The solid solution will be further blended with uranium dioxide and formed into reactor fuel pellets and eventually zirconium-clad fuel rods and assemblies. The facility design for these operations is patterned after a facility operated for many years in France to perform similar activities with reactor-grade plutonium dioxide. Contaminated wastes produced by MF^3 will be returned to the Department of Energy at the Savannah River Site.

Prior to construction of MF^3, the applicant, DCS, must obtain NRC approval (10 CFR Part 70.23(b)). The approval process for the facility involves two major steps. The applicant is now engaged in the first step which is to yield a construction permit. The applicant will later have to request a license to possess and utilize special nuclear materials. For this first step, the applicant is required to submit:

- a description of the facility site
- a description and safety assessment of the design bases of the principal structures, systems, and components of the facility
- a description of the provisions for protection against natural phenomena
- a description of the quality assurance program to be applied to the design, fabrication, construction, testing, and operation of the facility

The safety assessment of the design bases provides the rationale for the selection of functions or values and demonstrates that the design bases will provide reasonable assurance that the facility can withstand natural phenomena and the consequences of possible accidents.

A detailed quantitative analysis of the facility safety in an Integrated Safety Analysis (ISA) is not required in this first step of the approval process.

FACILITY LOCATION

MF^3 will be located on the well characterized Savannah River Site. Many other Department of Energy nuclear facilities are located on this site and perform functions similar to the functions of MF^3. Seismic hazards and other natural phenomena at the site have been extensively studied. The applicant has incorporated the current understanding into the MF^3 site characterization, and the Final Safety Evaluation Report provides a thorough, competent review of this material.

An important feature of the Savannah River Site is its isolation from what is ordinarily considered the public. However, the boundary for MF^3 is designated to be coincident with the controlled area boundary of the facility, not with the boundary of the Savannah River Site. The workforce at the Savannah River Site, but not associated with MF^3 is, therefore, considered part of the public, and thus there is no longer a large separation between the public and nuclear facilities on the Savannah River Site. The applicable regulations (10 CFR 70.22(i)) require emergency plans only if the maximum dose to a member of the public is expected to exceed 1-rem. The applicant expects to demonstrate that the 1-rem exposure limit will not be exceeded as a result of an accident at MF^3 and does not intend to prepare an emergency plan that requires offsite response capability and preplanning for actions to protect members of the public. Alternatively, the applicant will establish an evacuation plan for the facility and a protocol with the Department of Energy to integrate this plan with current emergency response plans at the Savannah River Site. The applicant concludes that no special controls or principal structures, systems, and components are required to protect workers at MF^3 beyond those that have been identified for control of radioactive and chemical material releases. The staff accepts this conclusion at this stage of the approval process and will examine the details of the protocol in the second stage of the approval process.

We agree with the staff that examination of the details of the emergency response plans can be deferred to the second stage. We do identify some areas of concern that should be addressed at that time. The distance from a point of release of radioactive or hazardous chemical material from MF^3 in the event of an accident to a point outside the boundary of the MF^3 controlled area is small. Should there be an accident at MF^3, it is imperative that emergency actions to protect all personnel be undertaken quickly and effectively. A Memorandum of Understanding with the Department of Energy at Savannah River is necessary but not sufficient to assure this prompt response will occur. The applicant should also develop an emergency response plan for the protection of workers. The plan should include pre-planning of emergency actions, the development of emergency procedures, training of personnel, clearly defined management responsibilities, and clearly defined lines of communication.

SAFETY ASSESSMENT

The technological bases for the MF^3 are well known. We concur with the staff's conclusions that:

- The design bases of the principal structures, systems, and components of MF^3 provide reasonable assurance of protection against natural phenomena and operational accidents.
- DCS has adequately addressed baseline design criteria.
- The proposed facility design is based on defense-in-depth practices.

Though there is some potential for the release of hazardous chemicals, the dominant hazard posed by the facility is the dispersal of plutonium or other radioactive elements. The facility is designed with nested zones so that leakage is always inward. Minor releases of radioactive materials are filtered through double High Efficiency Particulate Absorbers (HEPA filters). The principal mechanisms for substantial dispersal of radioactive materials are criticality events, explosions, and fires. The applicant has addressed the issues of criticality safety in about 80 criticality control units following well-established standards including the requirements of 10 CFR Part 70 and the ANSI/ANS-8 standard. The staff has done an exhaustive review of the applicant's submission and has done independent analysis of specific technical items. We concur with the staff's conclusion that the applicant has established an adequate organization to deal with a nuclear criticality safety program, has developed appropriate design features for the facility, and has the means to assure nuclear criticality safety.

Fire hazards at the facility are significant. There are several thousand kilograms of hydrocarbons (nominally kerosene) which are used as solvents in the solvent extraction processes. Hydrogen is used for stoichiometry control in the mixed oxide fuel fabrication. Reactive materials such as hydroxylamine nitrate are used in the process chemistry. Notably, the processes could produce inadvertently the explosive "red oil" in its acid recovery and evaporation processes.

Red oil, which is not really an oil and may not be red, is a poorly understood hydrolysis product that has caused damage at plutonium purification facilities operated by the Department of Energy and by others in the world. MF^3 can produce red oil in both the "open" and "closed" geometries used at the facility. There is insufficient knowledge to allow red oil and its reactions to be modeled theoretically to determine conditions that avoid explosive reactions. The applicant is undertaking a research program to assess the kinetics of red oil formation and reaction, but significant results are unlikely to be available before design decisions must be made. Therefore, the applicant must rely on the empirical experience with red oil formation and combustion. For the open geometries, DCS has adopted safety standards developed by the Department of Energy. Similar standards are not available for the closed systems. The applicant claims that sufficiently large vents and provision for quenching can be used to control temperatures below 125 °C, which will prevent runaway reactions. The applicant's technical bases for these conclusions are not clear to us. By the second stage in the approval process, the staff needs to develop adequate confidence that the applicant's control strategy for closed systems can indeed prevent runaway reactions under reasonably conceivable transient conditions.

A similar situation arises in connection with the applicant's plans for dealing with autocatalytic decomposition of hydroxylamine nitrate, though there is a much better understanding of the detailed kinetics of the pertinent reactions. In cases without nitrogen oxides, the applicant proposes to control temperature and concentrations. The detailed bases for the limits and the staff verification of the associated margins need to be better elucidated. In cases with nitrogen oxides, the applicant has proposed limitations on the concentrations. A large exhaust path is provided to prevent overpressurization. The staff has accepted these design provisions as "Reasonably and Generally Accepted Good Engineering Practices." A more quantitative evaluation of the engineering margins should be provided in the second stage of the approval process.

Fires in moderation-controlled spaces of plutonium facilities have long been a major concern at reactor fuel fabrication facilities. The use of water to suppress fires may initiate a criticality event. Operating experience has shown that fires suppressed by alternative agents (sometimes called "clean agents"), but not cooled, can reignite when air is readmitted. In the second stage of the approval process for MF^3, the applicant should demonstrate that in moderation-controlled spaces with limited amounts of combustible materials, post-fire cooling by conduction and thermal radiation is sufficient to prevent re-ignition. For moderation-controlled spaces with large amounts of combustible materials, the applicant should demonstrate that post-fire cooling can be achieved under adverse conditions. Manually-controlled systems using limited water quantities sprayed from installed nozzles should be considered during the ISA phase.

WASTE HANDLING

MF^3 will return waste to the Department of Energy. The facility to receive this waste at the Savannah River site has not been designed, nor have the waste acceptance criteria been established. This raises the possibility that additional unit operations will have to be added to MF^3. Perhaps of more importance, the possibility of unplanned interruptions in waste receipt by the Department of Energy needs to be considered in the integrated safety analysis of the MF^3 design. It will be necessary to conduct operations at MF^3 in a way that assures there is always sufficient waste storage capacity to bring the facility to a safe configuration in the event that waste receipt is interrupted. A protracted hiatus in waste receipt would raise issues of waste aging within MF^3. Experience has shown chemical evolutions brought on by evaporation, radiolysis, and other chemical processes can lead to the formation of hazardous chemicals or conditions in wastes awaiting transport to the Department of Energy. Measures to mitigate any hazards posed by aging wastes need to be addressed in the safety analyses for the final stage of the authorization process for MF^3 for timeframes of short, intermediate, and long duration.

In conclusion, the NRC staff has prepared a wide-ranging, technically competent Final Safety Evaluation Report on the construction authorization request for MF^3. This Final Safety Evaluation Report should be issued.

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

REFERENCES

1. Draft Final Safety Evaluation Report on the Construction Authorization Request for the Mixed Oxide Fuel Fabrication Facility at the Savannah River Site, South Carolina, December 2004 (as revised February 2, 2005).
2. Draft Safety Evaluation Report on the Construction Authorization Request for the Mixed Oxide Fuel Fabrication Facility at the Savannah River Site, South Carolina, Revision 1, April 2003.
3. U.S. Nuclear Regulatory Commission, Document control Desk, from Peter Hastings, Duke Cogema Stone & Webster, Subject: Mixed Oxide Fuel Fabrication Facility Construction Authorization Request Revised, Duke Cogema Stone & Webster, October 31, 2002.
4. U.S. Nuclear Regulatory Commission, Document Control Desk, from Peter Hastings, Duke Cogema Stone & Webster, Subject: Mixed Oxide Fuel Fabrication Facility Construction Authorization Request, Duke Cogema Stone & Webster, February 28, 2001.

References (continued)

5. Letter to Mr. Peter Hastings, Duke Cogema Stone & Webster, from Andrew Persinko, NRC Subject: June 2003 Monthly Open Item Status Report, July 8, 2003.
6. DOE-STD-1022-94, Natural Phenomena Hazards Characterization Criteria, U.S. Department of Energy, Washington, DC, 1995.
7. Draft Environmental Impact Statement , U.S. Nuclear Regulatory Commission, February 28, 2003.
8. U.S. Nuclear Regulatory Commission, Document Control Desk, from Peter Hastings, Duke Cogema Stone & Webster, Subject: Update to Mixed Oxide Fuel Fabrication Facility Environmental Report Revisions 1 & 2, December 10, 2002.

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

March 8, 2005

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

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

During the 520th meeting of the Advisory Committee on Reactor Safeguards held on March 3-5, 2005, the Committee considered the proposed Generic Letter 2005-XX, "Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power." The Committee plans to review the draft final version of this Generic Letter after reconciliation of public comments. The Committee has no objection to the staff's proposal to issue the proposed Generic Letter for public comment.

Reference:
Memorandum dated March 2, 2005, from Michael E. Mayfield, Director, Division of Engineering, NRR, to John T. Larkins, Executive Director, ACRS, Subject: Request for Review and Endorsement by the Advisory Committee on Reactor Safeguards (ACRS) Regarding the Proposed Draft Generic Letter 2005-XX, "Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power."

cc: A. Vietti-Cook, SECY
W. Dean, OEDO
J. Dixon-Herrity, OEDO
J. Dyer, NRR
M. Mayfield, NRR
M. Crutchley, NRR
J. Lamb, NRR
J. Lazevnick, NRR
A. Markley, NRR
A. Levin, RES

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

March 11, 2005

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

SUBJECT: REVISED DRAFT NUREG REPORT "ESTIMATING LOSS-OF-COOLANT ACCIDENT (LOCA) FREQUENCIES THROUGH THE ELICITATION PROCESS"

Dear Chairman Diaz:

During the 520th meeting of the Advisory Committee on Reactor Safeguards, March 3-5, 2005, we reviewed the revised draft NUREG Report, "Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process," (Reference 1). We reviewed a previous draft of this report (Reference 2) during the 518th meeting, December 2-4, 2004, and issued a report on December 10, 2004 (Reference 3). During these reviews, we had the benefit of discussions with the NRC staff and of the documents referenced.

RECOMMENDATION

The revised draft NUREG Report should be issued for public comment.

DISCUSSION

In our report dated December 10, 2004 (Reference 3), we recommended that the November 4, 2004 version of the draft NUREG Report be revised prior to being issued for public comment. We also commented that the Executive Summary should contain the composite distribution the analysts believe represents the expert community's current state of knowledge regarding loss-of-coolant accident (LOCA) frequencies. Below, we comment further on the appropriate choice of a composite distribution.

There are numerous ways in which individual expert opinions can be adjusted for potential biases and aggregated to produce a composite distribution that represents the group's judgment. The NUREG Report acknowledges this fact and presents several sensitivity analyses that provide insights into the numerical impact on the results of alternative assumptions and methods.

In our earlier report, we noted that the aggregation method chosen by the staff is at variance with the method described in NUREG-1150 (Reference 4) and NUREG/CR-6372 (Reference 5), i.e., taking the arithmetic average of the probability distributions of the experts. The staff has now produced composite distributions using the method in NUREG-1150 and NUREG/CR-6372 and called this method "mixture distribution aggregation."

The aggregation method may have a significant impact on the final results. The method the authors of the draft NUREG Report favor is the "geometric averaging" of the expert percentiles with some adjustment for potential overconfidence on the part of some experts. The composite distribution that the staff reports as best representing the expert consensus is the result of this geometric averaging. This distribution is less conservative than the composite distribution produced using the mixture distribution aggregation used in NUREG-1150 and NUREG/CR-6372.

The purpose of eliciting expert opinions is to provide input to the decisionmaking process, which in the present case is the selection of the transition break size in risk-informing 50.46. Ideally, the decisionmakers would be provided a probability distribution of the frequencies of the various LOCA categories that would reflect the current state of the art. As recognized in the draft NUREG Report, there is no consensus regarding the preferred method for processing individual expert opinions, and different methods may lead to significantly different results. In addition, the authors of the draft NUREG Report state that the study has limitations with respect to the scenarios and mechanisms considered.

One way of treating these issues is to select a bounding value for the break size, i.e., one that is larger than the break sizes from all the sensitivity analyses at a frequency of 10^{-5} per year. If a break size that is not bounding is selected, then the appropriateness of this selection would have to be justified with suitable rationale.

The revised NUREG Report should be issued for public review and comment. We would like to review the draft final version of the NUREG report after resolution of public comments.

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References:

1. Letter dated February 17, 2005, from Michael E. Mayfield, Director, Division of Engineering Technology, RES, to John T. Larkins, Executive Director, RES, Subject: Transmittal of Revised Draft NUREG Report, "Estimating Loss-of-Coolant (LOCA) Frequencies Through the Elicitation Process" and Associated Appendices (Pre-Decisional).
2. Memorandum dated November 4, 2004, from Michael E. Mayfield, Director, Division of Engineering Technology, RES, to John T. Larkins, Executive Director, ACRS, Subject: Transmittal of Draft NUREG on Passive System LOCA Frequency Development for use in Risk-Informed Revision of 10 CFR 50.46, Appendix K to Part 50, and GDC and Appendices (Pre-Decisional).

References (continued)

3. Letter dated December 10, 2004, from, Mario V. Bonaca, Chairman, ACRS, to Luis A. Reyes, EDO, NRC, Subject: Estimating Loss-of-Coolant Accident Frequencies Through the Elicitation Process.
4. U.S. Nuclear Regulatory Commission, *Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants*, Report NUREG-1150, 1990.
5. R.J. Budnitz, G. Apostolakis, D.M. Boore, L.S. Cluff, K.J. Coppersmith, C.A. Cornell, and P.A. Morris, *Recommendations for Probabilistic Seismic Hazard Analysis: Guidance and Use of Experts*, Report NUREG/CR-6372, 1997.

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

March 11, 2005

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

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

Dear Mr. Reyes:

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

CONCLUSIONS

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

DISCUSSION

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

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

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

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

- Nature of the Proposed Site

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

- Population in the Vicinity of the Site

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

- Geology and Seismicity of the Site

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

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

- Meteorology

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

- Potential Radiological Source Terms

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

- Emergency Plans

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

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

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References:

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

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

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

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

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

March 11, 2005

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

SUBJECT: PRESSURIZED THERMAL SHOCK (PTS) REEVALUATION PROJECT: TECHNICAL BASIS FOR REVISION OF THE PTS SCREENING CRITERION IN THE PTS RULE

Dear Mr. Reyes:

During the 520th meeting of the Advisory Committee on Reactor Safeguards, March 3-5, 2005, we continued our review of the technical basis for revision of the pressurized thermal shock (PTS) screening criterion in the PTS rule (10 CFR 50.61). This matter was also reviewed during a joint meeting of our Thermal-Hydraulic Phenomena, Materials and Metallurgy, and Reliability and Probabilistic Risk Assessment (PRA) Subcommittees on November 30-December 1, 2004, and at our meeting of December 2, 2004. 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 a comprehensive technical basis for analyzing the susceptibility of reactor pressure vessels to PTS and to support rulemaking to revise the current PTS Rule (10 CFR 50.61).

2. The external peer review of the technical work was valuable, and the staff response to the criticisms and questions raised by the peer review panel has strengthened the technical basis.

3. The documentation for the project is not yet final, but significant progress has been made. One of the documents, NUREG-1809, "Thermal-Hydraulic Evaluation of Pressurized Thermal Shock," should be substantially revised.

DISCUSSION

The PTS Rule, 10 CFR 50.61, was established in 1985 to ensure the integrity of irradiation-embrittled reactor pressure vessels during overcooling events. 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 (3) probabilistic fracture mechanics analyses to determine the probability of failure under the induced thermal and pressure stresses on the embrittled vessel.

The Reevaluation Project included 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 initiation and propagation, and (3) the characterization of the fracture toughness of the vessel materials.

A substantial experimental program to establish the thermal-hydraulic parameters was undertaken at the APEX facility at Oregon State University to supplement integral test data from Upper Plenum Test Facility (UPTF), Loss of Fluid Test (LOFT), Multiloop Integral System Test (MIST), and Rig of Safety Assessment (ROSA) facilities.

More realistic distributions for flaw density and geometry were developed based on detailed examination of welds and materials from vessels of cancelled plants and an elicitation of expert opinion. The accuracy and rigor of the probabilistic fracture mechanics code FAVOR, which is used in these analyses, has been improved. Much of this work is directly applicable to other situations involving embrittled pressure vessels such as providing 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).

The documentation includes a comprehensive summary report, NUREG-1806, that contains the comments of the peer reviewers and the staff responses to these comments. It also includes six reports on the thermal-hydraulic analyses; eight reports on the PRA studies; and eight reports on the probabilistic fracture mechanics code and analyses, the characterization of the fracture toughness of embrittled pressure vessels, and the characterization of crack distributions in pressure vessel materials and welds.

The focus of the current meeting was on NUREG-1809, "*Thermal Hydraulic Evaluation of Pressurized Thermal Shock,*" which provides a comparison of RELAP5 calculations with results from scaled, integral facility tests and some separate effects tests. The results from the test facilities span a wide range of flow conditions intended to be representative of those that occur in the vessel downcomer during a PTS event. These comparisons were used to develop quantitative estimates of uncertainties in the predicted values of the downcomer pressure and fluid temperature. These uncertainties are small compared to those that arise from the uncertainties in the boundary conditions for the scenarios such as break location, decay heat level, high-pressure injection (HPI) temperature, operator control of HPI, etc. Although the data that can be used to validate a model for the heat transfer coefficient are more limited than those available for validating downcomer pressure and fluid temperature, the results from the integral test facilities suggest that the models used for the heat transfer coefficient in the baseline PTS analyses were reasonably accurate and perhaps slightly conservative.

NUREG-1809 should be substantially revised and we would like to review the final revision. The PTS Reevaluation Project has required a substantial commitment of resources by the Agency. Good documentation is important to preserve the technical basis for revision of the PTS screening criterion.

We commend the staff for outstanding technical work in this multidisciplinary study, which provides a sound technical basis for the development of a revised 10 CFR 50.61.

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References:

1. EricksonKirk, M., et al., "Technical Basis for Revision of Pressurized Thermal Shock (PTS) Screening Limit in the PTS Rule (10 CFR 50.61): Summary Report," NUREG-1806, Draft for Peer Review Panel and ACRS Review, November 2, 2004.

2. Bessette, D. E., W. Arcieri, R. Beaton, and D. Fletcher, "Thermal Hydraulic Evaluation of Pressurized Thermal Shock," NUREG-1809, Draft, February 2005.

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

March 14, 2005

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

SUBJECT: PROPOSED RULEMAKING TO MODIFY 10 CFR 50.46, "RISK-INFORMED CHANGES TO LOSS-OF-COOLANT ACCIDENT TECHNICAL REQUIREMENTS"

Dear Chairman Diaz:

During the 520th meeting of the Advisory Committee on Reactor Safeguards on March 3-5, 2005, we reviewed the proposed rule for a voluntary alternative to 10 CFR 50.46, "Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements," (Reference 1). We also reviewed a draft version of a proposed rule (Reference 2) during the 518th meeting on December 2-4, 2004 and issued a letter on December 17, 2004 (Reference 3). During these reviews, we had the benefit of discussions with the NRC staff, the Nuclear Energy Institute, Westinghouse Owners Group and members of the public. We also had the benefit of the documents referenced.

RECOMMENDATION

The proposed rule for risk-informing 10 CFR 50.46 should be released for public comment.

DISCUSSION

The current proposed rule is consistent with the first two recommendations of our December 17, 2004 letter (Reference 3). It contains requirements intended to provide reasonable assurance of a coolable core geometry for breaks up to the double-ended guillotine break of the largest pipe in the reactor coolant system and permits operation only in configurations for which such capability has been demonstrated. The transition break size in the current version of the rule is equivalent to a single-ended rupture of the largest pipe attached to the reactor coolant system rather than the double-ended rupture in the earlier version.

The staff agrees with our recommendation that a better quantitative understanding of the possible risk benefits of a smaller transition break size is needed before finalizing the selection of the transition break size. The staff is attempting to identify areas where quantification of potential benefits might be meaningful. We have also heard a presentation from the industry on efforts to develop quantified estimates of the safety benefits associated with a smaller transition break size. These estimates are expected to be available during the rule comment period.

One of the changes in the proposed rule from the one that we reviewed in December is the omission of a quantitative criterion for the likelihood of late containment failure. We continue to believe that this should be considered in determining changes in risk due to changes in the licensing basis. We accept, however, that this is not an issue unique to changes in the licensing basis made possible by a risk-informed 10 CFR 50.46, and should be dealt with in the more general context of a revision to Regulatory Guide (RG) 1.174.

The proposed rule is an enabling rule. A licensee who wishes to make changes to its facility, technical specifications, or procedures based on the new rule will need to submit an application for a license amendment to allow such changes. The process of evaluating the risk due to such changes is critical to risk-informing 10 CFR 50.46. Since 1998, the NRC has been evaluating the acceptability of risk-informed changes to the licensing basis using RG 1.174. The guidance and acceptance criteria in RG 1.174 are intended to ensure that any increases in risk associated with changes to the licensing basis are small and that sufficient defense in depth and safety margins are maintained to address uncertainties.

The staff argues that it is necessary to include some of the high-level guidance of RG 1.174 in the proposed rule, and a new regulatory guide would be developed to provide additional guidance. The language in the draft proposed rule and in the statement of considerations is consistent with RG 1.174 (including the bundling of changes in risk due to unrelated changes in the licensing basis). It is not clear why the process of accepting the changes to the licensing basis that will be possible due to changes in 10 CFR 50.46 should be specified in the rule itself when it is already in RG 1.174, which is currently in use for evaluating risk-informed changes to the licensing basis. As part of the public comment process, input should be sought on the need to incorporate in the rule requirements for the acceptability of changes to the licensing basis and to develop a new regulatory guide for evaluating such changes.

The proposed rule contains provisions intended to ensure that plants that adopt a risk-informed 10 CFR 50.46 will still have a capability to mitigate loss-of-coolant accidents beyond the transition break size and permits operation only in configurations for which such capability has been demonstrated. However, the rule provides only high-level requirements for the analytical methods needed to demonstrate such capability and the statement of considerations just outlines a possible approach. The staff is developing a regulatory guide to provide more detailed guidance on acceptable methods for such analyses. The development of this regulatory guide is critical to the success of a risk-informed 10 CFR 50.46. We look forward to interacting with the staff on the development of this guide and discussing the draft final rule after resolution of public comments.

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References:

1. Memorandum dated December 2, 2004, from Catherine Haney, Program Director, Policy and Rulemaking Program, NRR, to various members NRR, Subject: Office Concurrence on Proposed Rule - Risk Informed Changes to Loss-of-Coolant Accident Technical Requirements (Pre-Decisional).

2. Letter dated February 14, 2005, from Catherine Haney, Program Director, NRR, to multiple addresses, NRR, Subject: Office Concurrence on Proposed Rule - Risk Informed Changes to Loss-of-Coolant Accident Technical Requirements (TAC #MB8397) 2004 (Pre-Decisional).

3. Letter dated December 17, 2004, from, Mario V. Bonaca, Chairman, ACRS, to Luis A. Reyes, EDO, NRC, Subject: Risk-Informing 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors."

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

April 8, 2005

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

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

During the 521st meeting of the Advisory Committee on Reactor Safeguards, April 7-8, 2005, the Committee considered the draft NUREG, "Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications." The Committee plans to review this document after reconciliation of public comments. The Committee has no objection to the staff's proposal to issue the draft NUREG for public comment.

Reference:
Memorandum dated April 1, 2005, from Charles E. Ader, Director, Division of Risk Analyses and Applications, RES, to John T. Larkins, Executive Director, ACRS, Subject: Draft NUREG entitled, "Verification & Validation of Selected Fire Models for Nuclear Power Plant Applications."

cc: A. Vietti-Cook, SECY
W. Dean, OEDO
J. Dixon-Herrity, OEDO
C. Ader, RES
A. Levin, RES
M. Salley, RES
M. Crutchley, NRR

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

April 8, 2005

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: PROPOSED REVISION TO REGULATORY GUIDE 3.71, "NUCLEAR CRITICALITY SAFETY STANDARDS FOR FUELS AND MATERIALS FACILITIES"

During the 521st meeting of the Advisory Committee on Reactor Safeguards, April 7-8, 2005, the Committee considered the draft Regulatory Guide DG-3023, "Nuclear Criticality Safety Standards for Fuels and Material Facilities." The Committee plans to review this document after reconciliation of public comments. The Committee has no objection to the staff's proposal to issue the draft Regulatory Guide for public comment.

Reference:

Memorandum dated March 25, 2005, from Jack R. Strosnider, Director, Office of Nuclear Material Safety and Safeguards, to John T. Larkins, Executive Director, ACRS, "Request to Waive Advisory Committee on Reactor Safeguards and Advisory Committee on Nuclear Waste Review of Proposed Revision to Regulatory Guide 3.71, 'Nuclear Criticality Safety Standards for Fuels and Materials Facilities'"

cc: A. Vietti-Cook, SECY
W. Dean, OEDO
J. Dixon-Herrity, OEDO
J. Strosnider, NMSS
H. Felsher, NMSS

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

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

April 8, 2005

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: PROPOSED REVISION TO MANAGEMENT DIRECTIVE 6.4, "GENERIC ISSUES PROGRAM"

During the 521st meeting of the Advisory Committee on Reactor Safeguards, April 7-8, 2005, the Committee considered the proposed revision to Management Directive (MD) 6.4, "Generic Issues Program." The Committee decided not to review this document. The Committee has no objection to the staff's proposal to issue the revised MD 6.4.

Reference:
Memorandum dated February 23, 2005, from Carl J. Paperiello, Director, RES, to multiple-addressees, Subject: Second Draft Revision of Management Directive 6.4, "Generic Issues Program."

cc: A. Vietti-Cook, SECY
W. Dean, OEDO
J. Dixon-Herrity, OEDO
C. Paperiello, RES
F. Eltawila, RES
H. VanderMolen, RES
A. Levin, RES
M. Crutchley, NRR

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

April 14, 2005

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

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

Dear Chairman Diaz:

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

CONCLUSIONS AND RECOMMENDATIONS

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

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

BACKGROUND AND DISCUSSION

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

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

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

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

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

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

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

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

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

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

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

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

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

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References

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

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

May 6, 2005

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: DRAFT REGULATORY GUIDE DG-8029 (PROPOSED REVISION 2 OF REGULATORY GUIDE 8.7), "INSTRUCTIONS FOR RECORDING AND REPORTING OCCUPATIONAL RADIATION DOSE DATA"

During the 522nd meeting of the Advisory Committee on Reactor Safeguards, May 5-6, 2005, the Committee considered draft Regulatory Guide DG-8029, "Instructions for Recording and Reporting Occupational Radiation Dose Data," and decided not to review this guide. The Committee has no objection to the staff's proposal to issue the draft Regulatory Guide for public comment.

<u>Reference</u>:
Memorandum dated April 29, 2005, from Farouk Eltawila, Director, Division of Systems Analysis and Regulatory Effectiveness, RES, to John T. Larkins, Executive Director, ACRS, Subject: Request to Defer ACRS Review of Draft Regulatory Guide 8.7, "Instructions for Recording and Reporting Occupational Radiation Dose Data."

cc: A. Vietti-Cook, SECY
W. Dean, OEDO
J. Dixon-Herrity, OEDO
F. Eltawila, RES
R. Assa, RES
S. Burrows, RES

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

May 13, 2005

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

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

Dear Chairman Diaz:

During the 522nd meeting of the Advisory Committee on Reactor Safeguards, May 5-6, 2005, we completed our review of the license renewal application for Arkansas Nuclear One, Unit 2 (ANO-2), and the associated final Safety Evaluation Report (SER) prepared by the NRC staff. Our Plant License Renewal Subcommittee also reviewed this matter during a meeting on December 1, 2004. During our review, we had the benefit of discussions with representatives of the NRC staff and Entergy Operations, Inc. (Entergy). We also had the benefit of the documents referenced. This report fulfills the requirements of 10 CFR 54.25 that the ACRS review and report on all license renewal applications.

CONCLUSION AND RECOMMENDATION

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

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

BACKGROUND AND DISCUSSION

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

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

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

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

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

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

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

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

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

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

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

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

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

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

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

May 13, 2005

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

SUBJECT: GUIDANCE FOR ASSESSING EXEMPTION REQUESTS FROM NUCLEAR POWER PLANT LICENSED OPERATOR STAFFING REQUIREMENTS

Dear Mr. Reyes:

During the 522nd meeting of the Advisory Committee on Reactor Safeguards, May 5-6, 2005, we reviewed the proposed revisions to NUREG-0800, "Standard Review Plan" (SRP) Chapter 13.0, "Conduct of Operations," Section 13.1.2 - 13.1.3, "Operating Organization," and the associated supporting document, NUREG-1791, "Guidance for Assessing Exemption Requests from the Nuclear Power Plant Licensed Operating Staffing Requirements Specified in 10 CFR 50.54(m)." During our review, we had the benefit of discussions with representatives of the NRC staff and of the document referenced.

CONCLUSIONS AND RECOMMENDATIONS

1. The revision to SRP Section 13.1.2 -13.1.3 should be issued.

2. Sections 10.1.3.2 and 10.3.3 of NUREG-1791 should be revised to emphasize the importance of objective measures to evaluate the safety implications of staffing schemes. The development of objective criteria for using simulation data in the evaluation should be explored.

3. NUREG-1791 will provide useful guidance for the staff, but it should be modified as recommended above. It will also provide guidance to applicants seeking exemptions to 10 CFR 50.54(m).

DISCUSSION

The introduction of advanced reactor designs and the increased use of advanced automation technologies in existing nuclear power plants will likely change the roles, responsibilities, composition, and size of the crews required to control plant operations.

Current requirements for control room staffing are primarily given in 10 CFR 50.54(m). They are based on the concept of operation for existing light-water reactors that may no longer apply to upgraded control rooms or future reactors. Therefore, applicants for an operating license for an advanced reactor and current licensees who have implemented significant changes to existing control rooms may submit applications for exemptions from current staffing regulations. To prepare for this eventuality, the staff has drafted a revision to the SRP Section 13.1.2-13.1.3 that refers staff reviewers to NUREG-1791.

NUREG-1791 describes a process for reviewing and determining the acceptability of exemption requests, including review of the:

- concept of operations,
- operational conditions,
- operating experience,
- functional requirements and function allocation,
- task analysis,
- job definitions,
- staffing plan,
- additional data and analyses, and
- staffing plan validation.

Useful checklists and references support the guidance in NUREG-1791. We note the omission of NUREG/CR-6838, "Technical Basis for Regulatory Guidance for Assessing Exemption Requests from the Nuclear Power Plant Licensed Operator Staffing Requirements Specified in 10 CFR 50.54(m)," from this set of references. The staff stated that this document provides the technical basis for the guidance in NUREG-1791.

The evaluation criteria applied at each stage of the review are qualitative and subjective. Ideally, the reviewer would have quantitative measures of the safety of the plant with the proposed staffing level. Such measures are not within the current capability of probabilistic risk assessment techniques. As a practical alternative, control room simulators could be used to objectively assess the relative ability of different staffing schemes to respond to a spectrum of operating, off-normal, design-basis-accident, and beyond-design-basis conditions. The value of control room simulation has been clearly demonstrated, for example, in the validation of emergency operating procedures.

Full-scope simulators may not be available for new plant designs when an applicant applies for an exemption. In this case, analytic simulators or other simulation techniques may be used as alternatives. Section 10.1.3.2 of NUREG-1791 discusses human-in-the-loop simulation techniques but stresses the difficulties of simulator validation without recognizing the benefits. This section should be revised to emphasize the importance of objective measures to evaluate the safety implications of staffing schemes. Similarly, the development of objective criteria for using simulation data in the evaluation should be explored for possible inclusion in Section 10.3.3.

Revisions to SRP Sections 13.1.2 and 13.1.3 should be issued. NUREG-1791 will provide useful guidance for the staff, but it should be modified as recommended above. It will also provide guidance to applicants seeking exemptions to 10 CFR 50.54(m).

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References:
Memorandum to J. Larkins, Executive Director, ACRS, from B. Boger, Director, Division of Inspection Program Management, Subject: Request for Advisory Committee on Reactor Safeguards Review of Standard Review Plan Chapter 13.0, Sections 13.1.2-13.1.3, "Operating Organization" Revision and Supporting Documents dated April 4, 2005.

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

June 3, 2005

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: PROPOSED BULLETIN ENTITLED "EMERGENCY PREPAREDNESS AND RESPONSE ACTIONS FOR SECURITY-BASED EVENTS"

During the 523rd meeting of the Advisory Committee on Reactor Safeguards, June 1-3, 2005, the Committee considered the proposed bulletin entitled "Emergency Preparedness and Response Actions for Security-Based Events." The Committee decided to review the staff's reconciliation of the licensees' responses. The Committee has no objection to the staff's proposal to issue this bulletin.

Reference:
Memorandum dated May 20, 2005, from R. William Borchardt, Deputy Director, Office of Nuclear Regulation, to Sher Bahadur, Chairman, Committee to Review Generic Requirements, Subject: Request for Committee to Review Generic Requirements (CRGR) Review and Endorsement of the Proposed Bulletin Entitled, "Emergency Preparedness and Response Actions for Security-based Events"

cc: A. Vietti-Cook, SECY
W. Dean, OEDO
J. Dixon-Herrity, OEDO
J. Dyer, NRR
R. Zimmerman, NSIR
D. Pickett, NRR
G. Casto, NSIR
M. Norris, NSIR

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

June 7, 2005

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: DRAFT FINAL REGULATORY GUIDES ON ASME CODE CASES

During the 523rd meeting of the Advisory Committee on Reactor Safeguards, June 1-3, 2005, the Committee considered the following draft final Regulatory Guides that list code cases published by the American Society of Mechanical Engineers (ASME):

- Regulatory Guide 1.84, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III," Revision 33

- Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," Revision 14

- Regulatory Guide 1.193, "ASME Code Cases Not Approved for Use," Revision 1

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

References:
Memorandum dated April 22, 2005, from Carl J. Paperiello, Director, Office of Nuclear Regulatory Research, to John T. Larkins, Executive Director, ACRS, Subject: ACRS Review of Final Regulatory Guides: ASME Code Cases

Memorandum dated June 3, 2005, from Carl J. Paperiello, Director, Office of Nuclear Regulatory Research, to James E. Dyer, Director, Office of Nuclear Reactor Regulation, Subject: Correction to NRC Staff Position on ASME Code Case N-586

cc: A. Vietti-Cook, SECY
W. Dean, OEDO
J. Dixon-Herrity, OEDO
C. Paperiello, RES
R. Barrett, RES
W. Norris, RES
R. Assa, RES

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

June 9, 2005

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

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

Dear Mr. Reyes:

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

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

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

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

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

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

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References

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

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

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

June 10, 2005

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

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

Dear Chairman Diaz:

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

CONCLUSIONS AND RECOMMENDATIONS

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

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

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

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

DISCUSSION

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

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

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

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

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

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

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

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

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

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

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

REFERENCES:

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

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

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

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

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

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

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

June 10, 2005

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

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

Dear Mr. Reyes:

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

CONCLUSIONS AND RECOMMENDATIONS

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

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

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

DISCUSSION

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

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

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

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

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

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

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

REFERENCES

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

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

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

June 14, 2005

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

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

Dear Mr. Reyes:

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

CONCLUSION AND RECOMMENDATION

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

DISCUSSION

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

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

- Nature of the Proposed Site

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

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

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

- Population in the Vicinity of the Site

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

- Geology and Seismicity of the Site

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

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

- Meteorology

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

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

- Emergency Plans

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

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

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References:

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

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

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

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

June 14, 2005

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

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

Dear Mr. Reyes:

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

CONCLUSIONS AND RECOMMENDATIONS

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

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

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

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

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

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

BACKGROUND

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

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

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

DISCUSSION

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

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

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

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

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

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

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

We look forward to reviewing the revised Regulatory Guide.

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

---

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

REFERENCES

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

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

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

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

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

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

July 7, 2005

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: PROPOSED RULEMAKING TO AMEND 10 CFR PARTS 19 AND 20: COLLECTION, REPORTING, AND LABELING REQUIREMENTS, AND CLARIFICATION OF DOSE DETERMINATION METHODOLOGY

During the 524th meeting of the Advisory Committee on Reactor Safeguards, July 6-8, 2005, the Committee considered a proposed rule to amend the requirements for radiation exposure related to the collection, reporting, and labeling of information and the dose determination methodology in 10 CFR Parts 19 and 20. The Committee decided not to review it. The Committee has no objection to the staff's proposal to issue this proposed rule for public comment.

Reference:
Memorandum dated July 6, 2005, from Catherine Haney, Program Director, Policy and Rulemaking Program, Office of Nuclear Reactor Regulation, to John T. Larkins, Executive Director, Advisory Committee on Reactor Safeguards, Subject: Proposed Rulemaking to Amend 10 CFR Parts 19 and 20: Collection, Reporting, And Labeling Requirements, and Clarification of Dose Determination Methodology

cc: A. Vietti-Cook, SECY
W. Dean, OEDO
J. Dixon-Herrity, OEDO
D. Matthews, NRR
M. Crutchley, NRR
C. Haney, NRR
S. Schneider, NRR

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

July 7, 2005

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

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

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

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

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

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

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

July 7, 2005

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: PROPOSED GENERIC LETTER 2005-XX, "INACCESSIBLE OR UNDERGROUND CABLE FAILURES THAT DISABLE ACCIDENT MITIGATION SYSTEMS"

During the 524th meeting of the Advisory Committee on Reactor Safeguards, July 6-8, 2005, the Committee considered the proposed Generic Letter 2005-XX, "Inaccessible or Underground Cable Failures that Disable Accident Mitigation Systems." The Committee plans to review the draft final version of this Generic Letter after reconciliation of public comments. The Committee has no objection to the staff's proposal to issue the proposed Generic Letter for public comment.

Reference:
Memorandum dated June 21, 2005 from Michael E. Mayfield, Director, Division of Engineering, NRR, to John T. Larkins, Executive Director, ACRS, Subject: Request to Advisory Committee on Reactor Safeguards (ACRS) for Review of Draft Generic Letter 2005-XX, "Inaccessible or Underground Cable Failures that Disable Accident Mitigation Systems"

cc: A. Vietti-Cook, SECY
W. Dean, OEDO
J. Dixon-Herrity, OEDO
M. Mayfield, NRR
M. Crutchley, NRR
J. Calvo, NRR
R. Jenkins, NRR
T. Koshy, NRR

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

July 7, 2005

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: PROPOSED REVISION TO STANDARD REVIEW PLAN SECTION 6.5.2, "CONTAINMENT SPRAY AS A FISSION PRODUCT CLEANUP SYSTEM"

During the 524th meeting of the Advisory Committee on Reactor Safeguards, July 6-8, 2005, the Committee considered draft Revision 3 to Section 6.5.2, "Containment Spray as a Fission Product Cleanup System," of NUREG-0800, "Standard Review Plan." The Committee decided not to review it. The Committee has no objection to the staff's proposal to issue this revision.

Reference:
Memorandum dated June 9, 2005, from Michael E. Mayfield, Director, Division of Engineering, Office of Nuclear Reactor Regulation, to John T. Larkins, Executive Director, Advisory Committee on Reactor Safeguards, Subject: Review of Proposed Revision to Standard Review Plan Section 6.5.2, "Containment Spray as a Fission Product Cleanup System"

cc: A. Vietti-Cook, SECY
W. Dean, OEDO
J. Dixon-Herrity, OEDO
M. Mayfield, NRR
M. Crutchley, NRR
S. Koenick, NRR
K. Parczewski, NRR

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

July 8, 2005

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: PROPOSED REGULATORY GUIDE (DG)-1128, "CRITERIA FOR ACCIDENT MONITORING INSTRUMENTATION FOR NUCLEAR POWER PLANTS" (REVISION 4 OF REGULATORY GUIDE 1.97)

During the 524th meeting of the Advisory Committee on Reactor Safeguards, July 6-8, 2005, the Committee considered the proposed Regulatory Guide DG-1128, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants." The Committee plans to review the draft final version of this Regulatory Guide after reconciliation of public comments. The Committee has no objection to the staff's proposal to issue the proposed Regulatory Guide DG-1128 for public comment.

Reference:
Memorandum dated June 30, 2005, from Richard J. Barrett, Director for the Division of Engineering Technology, RES, to John T. Larkins, Executive Director, ACRS, Subject: Proposed Regulatory Guide (DG)-1128, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants" (Revision 4 of Regulatory Guide 1.97), (ADAMS Accession No. ML051730634).

cc: A. Vietti-Cook, SECY
W. Dean, OEDO
J. Dixon-Herrity, OEDO
R. Assa, RES
R. Barrett, RES
M. Evans, RES
G. Tartal, RES

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

July 15, 2005

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

SUBJECT: ASSESSMENT OF THE QUALITY OF THE NRC RESEARCH PROJECTS

Dear Chairman Diaz:

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

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

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

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

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

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

- RES submits to us a list of research projects that are candidates for review because they have reached sufficient maturity that meaningful technical review can be conducted.

- We select a maximum of four projects for detailed review in the fiscal year.

- A panel of three ACRS members is established to assess the quality of each research project.

- The panel follows the guidance developed by the ACRS full Committee in conducting the technical review. This guidance is discussed further below.

- Each panel assesses the quality of the assigned research project and presents an oral and a written report to the ACRS full Committee for review. This review is to ensure uniformity in the evaluations by the various panels.

- The Committee revises these reports and provides them promptly to the cognizant research manager, as appropriate.

- The Committee submits an annual summary report to the RES Director.

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

- Results meet the objectives

- Documentation of research results and methods is adequate

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

- Justification of major assumptions (12%)

- Soundness of technical approach and results (52%)

- Uncertainties and sensitivities addressed (11%)

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

- Clarity of presentation (16%)

- Identification of major assumptions (9%)

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

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

- Effects of chemical reactions on head loss in debris beds that may block sump screens
- Experimental studies of loss-of-coolant accident generated debris accumulation and head loss on sump screens
- Improvements to the MACCS computer code, plume model adequacy

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

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

- Standardized Plant Analysis Risk (SPAR) model development program
- Thermal-hydraulic experiments at the Pennsylvania State University
- Steam generator tube integrity research being performed at the Argonne National Laboratory

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

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References:

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

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

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

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

July 15, 2005

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

Dear Mr. Reyes:

SUBJECT: PROPOSED REVISION 2 TO REGULATORY GUIDE 1.152, "CRITERIA FOR USE OF COMPUTERS IN SAFETY SYSTEMS OF NUCLEAR POWER PLANTS"

During the 524th meeting of the Advisory Committee on Reactor Safeguards on July 6-8, 2005, we reviewed the proposed Revision 2 to Regulatory Guide 1.152. During this review, we had the benefit of discussions with the NRC staff and of the documents referenced.

RECOMMENDATION

Revision 2 to Regulatory Guide 1.152 should be issued.

DISCUSSION

The current Regulatory Guide 1.152, Revision 1, endorses IEEE Std. 7-4.3.2-1993, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations." This standard has been updated as IEEE Std. 7-4.3.2-2003 to address advances in computer technology. The proposed Revision 2 to Regulatory Guide 1.152 endorses the updated IEEE standard (excluding its annexes) and provides guidance on cyber security, which is not addressed in the IEEE standard.

During the public comment period, the staff received 20 comment letters. We agree with the staff's resolution of these comments. Several commenters objected to the inclusion of guidance on cyber security in the regulatory guide and argued that the staff should wait until the industry or technical societies develop relevant guidance. The staff responded that they believe such guidance is several years away. Due to the increasing use of digital systems at nuclear power plants, the staff has chosen to address cyber security without further delay. The staff is actively involved in industry activities in this area and plans to revise the regulatory guide when an industry standard becomes available. We agree with the staff's position on this matter.

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References:

1. Memorandum from Richard J. Barrett, Director, Division of Engineering Technology, Office of Nuclear Regulatory Research, to John T. Larkins, Executive Director, Advisory Committee on Reactor Safeguards, "Regulatory Guide 1.152, Revision 2, "Criteria for Use of Computers in Safety Systems of Nuclear Power Plants," May 31, 2005. (ADAMS #ML051290100)
2. IEEE Power Engineering Society, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations," IEEE Std. 7-4.3.2-2003, December 19, 2003.

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

July 18, 2005

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

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

Dear Chairman Diaz:

During the 524th meeting of the Advisory Committee on Reactor Safeguards, July 6-8, 2005, we completed our review of the license renewal application for the Donald C. Cook Nuclear Plant (CNP), Units 1 and 2, and the final Safety Evaluation Report (SER) prepared by the NRC staff. Our Plant License Renewal Subcommittee also reviewed this matter during a meeting on February 9, 2005. During these reviews, we had the benefit of discussions with representatives of the NRC staff and Indiana Michigan Power Company, the applicant. We also had the benefit of the documents referenced. This report fulfills the requirements of 10 CFR 54.25 that the ACRS review and report on all license renewal applications.

CONCLUSION AND RECOMMENDATION

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

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

BACKGROUND AND DISCUSSION

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

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

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

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

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

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

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

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

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

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

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

Sincerely

Graham B. Wallis

Graham B. Wallis
Chairman

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

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

July 18, 2005

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

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

Dear Chairman Diaz:

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

CONCLUSIONS

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

DISCUSSION

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

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

- Nature of the Proposed Site

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

- Population in the Vicinity of the Site

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

- Geology and Seismicity of the Site

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

- Meteorology

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

- Potential Radionuclide Releases

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

- Emergency Plans

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

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

- *evaluations of the site rest on an assumption that can be justified only after a site* permit has been issued,

- a physical attribute exists for the site that is not acceptable for the design of systems, structures and components important to safety, or

- evaluations can be completed only after some future act has taken place.

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

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

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

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

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

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

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

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

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

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

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

July 18, 2005

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

SUBJECT: DRAFT FINAL REGULATORY GUIDE DG-1137, "GUIDELINES FOR LIGHTNING PROTECTION OF NUCLEAR POWER PLANTS"

Dear Mr. Reyes:

During the 524th meeting of the Advisory Committee on Reactor Safeguards, July 6-8, 2005, we met with the NRC staff and its contractor to discuss the draft final Regulatory Guide DG-1137, "Guidelines for Lightning Protection of Nuclear Power Plants," and NUREG/CR-6866, "Technical Basis for Regulatory Guidance on Lightning Protection in Nuclear Power Plants." We also had the benefit of the documents referenced.

Recommendation

The Regulatory Guide should be issued.

Discussion

This draft Regulatory Guide provides guidance for designing and installing lightning protection systems (LPSs) to ensure that electrical transients resulting from lightning phenomena do not render safety-related systems inoperable or cause spurious operation of such systems. It is intended for new plants but can also be used by existing plants.

The guidance covers protection of the power plant and relevant ancillary facilities, as well as the testing and maintenance of LPSs. It does not cover testing and design practices specifically intended to protect safety-related I&C systems against the secondary effects of lightning discharges, i.e., low-level power surges and electromagnetic and radio-frequency interference (EMI/RFI). These practices are covered in Regulatory Guide 1.180, "Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems."

Draft Regulatory Guide DG-1137 endorses four standards issued by the Institute of Electrical and Electronics Engineers (IEEE) that are intended to provide comprehensive lightning protection guidance for nuclear power plants:

- IEEE Std. 665-1995 (reaffirmed 2001), "IEEE Guide for Generating Station Grounding"
- IEEE Std. 666-1991 (reaffirmed 1996),"IEEE Design Guide for Electrical Power Service Systems for Generating Stations"
- IEEE Std. 1050-1996, "IEEE Guide for Instrumentation and Control Equipment Grounding in Generating Stations
- IEEE Std.C62.23-1995 (reaffirmed 2001), "IEEE Application Guide for Surge Protection of Electric Generating Plants."

The staff endorses these IEEE standards in their entirety except for an error in IEEE Std. 665-1995 (reaffirmed 2001). These standards contain numerous references to other secondary standards. Accordingly, the staff's endorsement of the four standards includes the applicable portions of the secondary standards.

We agree with the staff that DG-1137 and the endorsed IEEE standards provide adequate guidance for use by the industry in designing and installing LPSs at nuclear power plants. The Regulatory Guide should be issued.

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References:
1. Memorandum from R. Barrett, RES, to J. Larkins, ACRS, dated June 24, 2005, Subject: Request ACRS to Review Final Guide, DG-1137, "Guidelines for Lightning Protection of Nuclear Power Plants" (ADAMS Accession No. ML051780237)
2. Draft Regulatory Guide DG-1137, "Guidelines for Lightning Protection of Nuclear Power Plants," May 2005 (ADAMS Accession No. ML051780242)
3. NUREG/CR-6866, "Technical Basis for Regulatory Guidance on Lightning Protection in Nuclear Power Plants," May 2005 (ADAMS Accession No. ML051780247)
4. Letter from T. Groblewski, Progress Energy, to the Rules and Directives Branch, dated April 15, 2005, regarding comments to DG-1137 (ADAMS Accession No. ML051110419)
5. Letter from F. Mashburn, Tennessee Valley Authority, to the Rules and Directives Branch, dated April 20, 2005, regarding comments to DG-1137 (ADAMS Accession No. ML051160217)
6. Staff responses to public comments (ADAMS Accession No. ML051780249)

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

July 25, 2005

MEMORANDUM TO: Luis A. Reyes
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"

The Advisory Committee on Reactor Safeguards (ACRS) members considered the draft final amendment to 10 CFR 50.55a, which incorporates by reference Regulatory Guide 1.84, Revision 33, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III," and Regulatory Guide 1.147, Revision 14, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1." They decided not to review this draft final amendment to 10 CFR 50.55a.

Reference:
Memorandum dated May 9, 2005, from Catherine Haney, Program Director, Policy and Rulemaking Program, Division of Regulatory Improvement Programs, NRR, to John T. Larkins, Executive Director, Advisory Committee on Reactor Safeguards, Subject: Final Amendment to 10 CFR 50.55a, "Codes and Standards".

cc: A. Vietti-Cook, SECY
W. Dean, OEDO
J. Dixon-Herrity, OEDO
J. Lyons, NRR
M. Crutchley, NRR
H. Tovmassian, NRR
S. Coffin, NRR
C. Haney, NRR

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

July 25, 2005

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

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

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

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

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

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

September 14, 2005

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: DRAFT FINAL REGULATORY GUIDE 3.71, "NUCLEAR CRITICALITY SAFETY STANDARDS FOR FUELS AND MATERIALS FACILITIES"

During the 525th meeting of the Advisory Committee on Reactor Safeguards, September 8-10, 2005, the Committee considered the proposed Revision 1 to Regulatory Guide (RG) 3.71, "Nuclear Criticality Safety Standards for Fuels and Material Facilities," which the staff issued for public comment in May 2005 as Draft Regulatory Guide DG-3023. The Committee has decided not to review the revised RG, and has no objection to the staff's proposal to issue it.

References:
Memorandum dated August 19, 2005, from Jack R. Strosnider, Director, Office of Nuclear Material Safety and Safeguards, to John T. Larkins, Executive Director, ACRS, Subject: Regulatory Guide 3.71, "Nuclear Criticality Safety Standards for Fuels and Materials Facilities"

cc: A. Vietti-Cook, SECY
W. Dean, OEDO
J. Dixon-Herrity, OEDO
B. Purnell, NMSS
J. Strosnider, NMSS
S. Jones, NMSS

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

September 20, 2005

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

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

Dear Mr. Reyes:

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

CONCLUSIONS AND RECOMMENDATIONS

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

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

DISCUSSION

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

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

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

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

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

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

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

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

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References:

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

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

September 20, 2005

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: ACRS REVIEW OF THE NORTH ANNA EARLY SITE PERMIT APPLICATION - FINAL SAFETY EVALUATION REPORT CHANGED PAGES PRIOR TO PUBLISHING AS A NUREG

During the 525th meeting of the Advisory Committee on Reactor Safeguards, September 8-10, 2005, the Committee considered the changes reflected in Revision 5 of Dominion Nuclear North Anna, LLC, application for an early site permit. The changes included depicting the selected horizontal and vertical operating-basis earthquake and safe-shutdown earthquake spectra. The staff has evaluated the new seismic information and concluded that the previous analysis was conservative. The Committee agrees with the staff that the changes to the final Safety Evaluation Report issued on June 16, 2005 are limited to minor corrections of factual inaccuracies and changes to the structure of the document. The Committee decided that additional review of this document prior to issuance is not necessary.

References:

1. Memorandum dated August 24, 2005 from William D. Beckner, Program Director, Division of Regulatory Improvement Programs, NRR, to John T. Larkins, Executive Director, ACRS, Subject: ACRS Review of the North Anna Early Site Permit Application - Final Safety Evaluation Report Changed Pages Prior to Publishing As a NUREG.

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

cc: A. Vietti-Cook, SECY
W. Dean, OEDO
J. Dixon-Herrity, OEDO
W. Beckner, NRR
L. Dudes, NRR
B. Sosa, NRR
M. Crutchely, NRR

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

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

September 20, 2005

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: PROPOSED RULE: LICENSES, CERTIFICATIONS, AND APPROVALS FOR NUCLEAR POWER PLANTS, 10 CFR PART 52 AND CONFORMING AMENDMENTS TO PARTS 1, 2, 10, 19, 20, 21, 25, 26, 50, 51, 54, 55, 72, 73, 95, 140, 170, and 171

During the 525th meeting of the Advisory Committee on Reactor Safeguards, September 8-10, 2005, the Committee considered the proposed rule: Licenses, Certifications, and Approvals for Nuclear Power Plants, 10 CFR Part 52 and Conforming Amendments to Parts 1, 2, 10, 19, 20, 21, 25, 26, 50, 51, 54, 55, 72, 73, 95, 140, 170, and 171. The Committee plans to review the draft final version of the proposed rule after reconciliation of public comments. The Committee has no objection to the staff's proposal to issue the proposed rule for public comment.

Reference:
Memorandum dated August 9, 2005 from Eileen M. McKenna, Acting Program Director, Division of Regulatory Improvement Programs, NRR, to John T. Larkins, Executive Director, ACRS, Subject: Request to Advisory Committee on Reactor Safeguards to defer its formal review of the "Proposed Rule: Licenses, Certifications, and Approvals for Nuclear Power Plants, 10 CFR Part 52 and Conforming Amendments to Parts 1, 2, 10, 19, 20, 21, 25, 26, 50, 51, 54, 55, 72, 73, 95, 140, 170, and 171."

cc: A. Vietti-Cook, SECY
W. Dean, OEDO
J. Dixon-Herrity, OEDO
E. McKenna, NRR
M. Crutchley, NRR
H. Tovmassian, NRR

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

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

September 21, 2005

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

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

Dear Chairman Diaz:

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

- What shall be the minimum level of safety that new plants need to meet to achieve enhanced safety?
- How shall the risk from multiple reactors at a single site be accounted for?

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

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

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

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

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

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

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

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

6. What should these criteria be?

7. How should compliance with these criteria be demonstrated?

DISCUSSION

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Question 6. What should these criteria be?

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

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

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

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

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

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

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

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

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

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

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

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

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

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

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

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

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

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

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

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

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

<u>References:</u>

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

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

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

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

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

September 22, 2005

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

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

Dear Chairman Diaz:

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

CONCLUSION AND RECOMMENDATION

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

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

BACKGROUND AND DISCUSSION

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

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

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

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

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

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

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

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

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

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

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

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

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

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References:

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

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

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

September 22, 2005

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

SUBJECT: DRAFT FINAL REVISIONS TO GENERIC LICENSE RENEWAL GUIDANCE DOCUMENTS

Dear Chairman Diaz:

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

CONCLUSIONS AND RECOMMENDATIONS

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

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

DISCUSSION

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

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

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

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

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

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

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

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

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

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

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

September 22, 2005

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

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

Dear Mr. Reyes:

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

RECOMMENDATION

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

DISCUSSION

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

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

Nature of the Site

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

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

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

Weather

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

Seismicity

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

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

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

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

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

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

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

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

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

September 22, 2005

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: PROPOSED REVISION TO STANDARD REVIEW PLAN SECTION 6.2.1.3 "MASS AND ENERGY RELEASE ANALYSIS FOR POSTULATED LOSS-OF-COOLANT ACCIDENTS"

During the 525th meeting of the Advisory Committee on Reactor Safeguards, September 8-10, 2005, the Committee considered the proposed revision to NUREG-800, Standard Review Plan (SRP), Section 6.2.1.3, "Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents." The Committee decided not to review this revised SRP Section. The Committee has no objection to the staff's proposal to issue the proposed revised SRP Section for public comment.

References:
Memorandum dated July 14, 2005, from James E. Lyons, Director, Division of Systems Safety and Analysis, NRR, to John T. Larkins, Executive Director, ACRS, Subject: Request to Waive Advisory Committee on Reactor Safeguards (ACRS) Review of Proposed Revision to Standard Review Plan Section 6.2.1.3 "Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents"

cc: A. Vietti-Cook, SECY
W. Dean, OEDO
J. Dixon-Herrity, OEDO
E. Throm, NRR
M. Crutchley, NRR
J. Dyer, NRR
J. Lyons, NRR

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

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

September 23, 2005

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

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

Dear Mr. Reyes:

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

RECOMMENDATIONS

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

DISCUSSION

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

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

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

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

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

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

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

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

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

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

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

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

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

September 23, 2005

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

Michael Snodderly for

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

SUBJECT: PROPOSED DRAFT REGULATORY GUIDE DG-8028, "CONTROL OF ACCESS TO HIGH AND VERY HIGH RADIATION AREAS IN NUCLEAR POWER PLANTS"

During the 525th meeting of the Advisory Committee on Reactor Safeguards, September 8-10, 2005, the Committee considered the proposed draft Regulatory Guide DG-8028, "Control of Access to High and Very High Radiation Areas in Nuclear Power Plants." The Committee plans to review the draft Regulatory Guide DG-8028 after reconciliation of public comments. The Committee has no objection to the staff's proposal to issue the proposed draft Regulatory Guide DG-8028 for public comment.

References:
Memorandum dated July 20, 2005, from Farouk Eltawila, Director for the Division of Systems Analysis and Regulatory Effectiveness, RES, to John T. Larkins, Executive Director, ACRS, Subject: Request for ACRS Review of Proposed Draft Regulatory Guide DG-8028, "Control of Access to High and Very High Radiation Areas in Nuclear Power Plants," (ADAMS Accession No. ML052100152).

Draft Regulatory Guide DG-8028, "Control of Access to High and Very High Radiation Areas in Nuclear Power Plants," (ADAMS Accession No. ML051670289).

cc: A. Vietti-Cook, SECY
W. Dean, OEDO
J. Dixon-Herrity, OEDO
D. Matthews, NRR
S. Coffin, NRR
D. Diec, NRR
S. Weerakkody, NRR
M. Crutchley, NRR

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

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

October 11, 2005

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: PROPOSED RULE ON SAFETY/SECURITY INTERFACE

During the 526th meeting of the Advisory Committee on Reactor Safeguards, October 6-8, 2005, the Committee considered the proposed rule (10 CFR 73.55) on the safety/security interface. The Committee plans to review the draft final version of this rule after reconciliation of public comments. The Committee has no objection to the staff's proposal to issue the proposed rule for public comment.

Reference:
Memorandum dated August 18, 2005, from Eileen M. McKenna, Acting Program Director, Policy and Rulemaking Program, Office of Nuclear Reactor Regulation, to John T. Larkins, Executive Director, Advisory Committee on Reactor Safeguards, Subject: Request to Defer ACRS Review of Draft Proposed Rule on Safety/Security Interface.

cc: A. Vietti-Cook, SECY
W. Dean, OEDO
J. Dixon-Herrity, OEDO
D. Matthews, NRR
E. McKenna, NRR
J. Birmingham, NRR
G. Tracy, NSIR
T. Meek, NRR

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

October 13, 2005

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: DRAFT REGULATORY GUIDE DG-1120 (DG-1120), "TRANSIENT AND ACCIDENT ANALYSIS METHODS," AND STANDARD REVIEW PLAN CHAPTER 15 SECTION 15.0.2 (SRP 15.0.2), "REVIEW OF TRANSIENT AND ACCIDENT METHODS"

During the 526th meeting of the Advisory Committee on Reactor Safeguards, October 6-7, 2005, the Committee considered the latest version of the draft Regulatory Guide DG-1120, "Transient and Accident Analysis Methods," and Standard Review Plan Chapter 15 Section 15.0.2, "Review of Transient and Accident Methods," which were provided to the Committee for information. The Committee has decided to review these documents.

References:
Memorandum dated September 13, 2005, from Farouk Eltawila, Director, Division of Systems Analysis and Regulatory Effectiveness, Office of Nuclear Regulatory Research, to John T. Larkins, Executive Director, ACRS, Subject: "Publication of Draft Regulatory Guide (DG-1120), 'Transient and Accident Analysis Methods,' and Standard Review Plan Chapter 15 Section 15.0.2 (SRP 15.0.2, 'Review of Transients [sic] and Accident Methods'"

cc: A. Vietti-Cook, SECY
W. Dean, OEDO
J. Dixon-Herrity, OEDO
F. Eltawila, RES
S. Marshall, RES
C. Paperillo, RES
J. Dyer, NRR
T. Meek, NRR
R. Landry, NRR
F. Akstulewica, NRR
J. Wermiel, NRR

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

October 18, 2005

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

SUBJECT: PROPOSED RECOMMENDATION FOR RESOLVING GENERIC SAFETY ISSUE 80, "PIPE BREAK EFFECTS ON CONTROL ROD DRIVE HYDRAULIC LINES IN THE DRYWELLS OF BWR MARK I AND II CONTAINMENTS"

Dear Mr. Reyes:

During the 526th meeting of the Advisory Committee on Reactor Safeguards, October 6-7, 2005, we reviewed the recommendation proposed by the Office of Nuclear Regulatory Research (RES) for resolving Generic Safety Issue (GSI) - 80, "Pipe Break Effects on Control Rod Drive Hydraulic Lines in the Drywells of BWR Mark I and II Containments." During our review, we had the benefit of discussions with representatives of the NRC staff and the documents referenced.

RECOMMENDATION

We agree with the RES recommendation that GSI-80 be closed without any changes to the regulations or guidance.

BACKGROUND AND DISCUSSION

Damage to control rod drive (CRD) hydraulic lines by mechanical impact as a result of a loss-of-coolant accident (LOCA) was raised as an issue by the ACRS in 1978 during operating license reviews of some boiling water reactors (BWRs). Such damage could prevent control rod insertion, creating the potential for recriticality when the core is reflooded.

The failure to insert a significant number of control rods could pose two separate safety problems. First, when the core is reflooded by cold emergency core cooling water, the reactor will undergo a cold-water reactivity transient if the core is not sufficiently subcritical. The portions of the core where control rods failed to insert can return to a significant power level and may even overshoot to power levels in excess of the full-power limits of the fuel. Second, the residual heat removal system is sized to remove the core decay heat. If fission heat is added to decay heat, the heat removal capacity of the system may not be able to remove this additional heat load. The suppression pool would overheat and could lead to coolant boil-off, containment failure, and core melt.

The staff addressed this issue several times over the years and initially concluded that the frequency of this scenario was sufficiently low that the staff assigned a low priority to the final resolution of GSI-80. During subsequent site visits conducted by the staff associated with GSI-156.6.1, "Pipe Break Effects on Systems and Components," new piping configurations were discovered that were not considered in the original evaluation of GSI-80. In its periodic review of low-priority GSIs conducted in March 1998, the staff concluded that the priority for resolving GSI-80 should be increased. Consequently, RES initiated an additional study to determine the safety significance of the issue.

Control rod insertion would not be prevented by breakage or crimping of the CRD accumulator piping or breakage of the scram discharge piping. RES performed an assessment of the probability of a high-energy pipe break that would result in the crimping and complete closure of one or more of the scram discharge pipes and would prevent CRD discharge flow and rod insertion. RES found that the estimated frequency of high-energy pipe breaks that can impact the CRD discharge piping is low. Even if such an event occurs, analysis using the ANSYS code indicates that it would result in bending and breaking of the CRD piping rather than crimping the piping closed. Pipe-to-pipe impact testing confirmed these results. Consequently, the staff concluded that the contribution of this accident to the core damage frequency is very low.

We agree with the RES recommendation to close GSI-80 without any changes to existing regulations or regulatory guidance.

Sincerely,

William J. Shack
Acting Chairman

References:

1. Memorandum from F. Eltawila, RES, to J. Larkins, ACRS, dated August 11, 2005, Subject: Proposed Closure of Generic Safety Issue 80, "Pipe Break Effects on Control Rod Drive Hydraulic Lines in the Drywells of BWR Mark I and II Containments," (ADAMS Accession No. ML052230517)

2. NUREG-0933, "A Prioritization of Generic Safety Issues," 1984

3. NUREG/CR-6395, "Enhanced Prioritization of Generic Safety Issue 156.6.1: 'Pipe Break Effects on Systems and Components Inside Containment,'" U.S. Nuclear Regulatory Commission, November 1999 (ADAMS Accession No. ML003732008)

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

October 19, 2005

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

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

Dear Mr. Reyes:

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

CONCLUSIONS AND RECOMMENDATIONS

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

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

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

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

BACKGROUND

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

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

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

DISCUSSION

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

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

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

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

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

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

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

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

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

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

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

Sincerely,

William J. Shack
Acting Chairman

References:

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

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

November 4, 2005

Dr. Carl J. Paperiello
Director
Office of Nuclear Regulatory Research
Washington, D.C. 20555-0001

SUBJECT: ACRS ASSESSMENT OF THE QUALITY OF SELECTED NRC RESEARCH PROJECTS - FY 2005

Dear Dr. Paperiello:

Enclosed is our report on the quality assessment of the following research projects:

- Station Blackout Risk Evaluation for Nuclear Power Plants performed as a part of Standardized Plant Analysis Risk (SPAR) Models Development Program
  - This project was found to be more than satisfactory. The results meet the research objectives.
- Steam Generator Tube Integrity Program at the Argonne National Laboratory
  - This project was found to be satisfactory. The results meet the research objectives.
- Analysis of Rod Bundle Heat Transfer Facility Two-Phase Interface Drag Experiments at the Penn State University
  - This project marginally satisfied the research objectives. The Committee identified significant deficiencies.

These projects were selected from a list of candidate projects suggested by the Office of Nuclear Regulatory Research.

A fourth research project on reactor containment performance being conducted at Sandia National Laboratories will be evaluated later, once a particularly pivotal report on the research becomes available.

We anticipate receiving your list of candidate projects for review during the next 12 months.

Sincerely,

William J. Shack
Acting Chairman

Enclosure: As stated

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

ACRS Assessment of the Quality of Selected NRC Research Projects

October 2005

U.S. Nuclear Regulatory Commission
Advisory Committee on Reactor Safeguards
Washington, DC 20555-0001

ABOUT THE ACRS

The Advisory Committee on Reactor Safeguards (ACRS) was established as a statutory Committee of the Atomic Energy Commission (AEC) by a 1957 amendment to the *Atomic Energy Act* of 1954. The functions of the Committee are described in Sections 29 and 182b of the Act. The *Energy Reorganization Act* of 1974 transferred the AEC's licensing functions to the U.S. Nuclear Regulatory Commission (NRC), and the Committee has continued serving the same advisory role to the NRC.

The ACRS provides independent reviews of, and advice on, the safety of proposed or existing NRC-licensed reactor facilities and the adequacy of proposed safety standards. The ACRS reviews power reactor and fuel cycle facility license applications for which the NRC is responsible, as well as the safety-significant NRC regulations and guidance related to these facilities. On its own initiative, the ACRS may review certain generic matters or safety-significant nuclear facility items. The Committee also advises the Commission on safety-significant policy issues, and performs other duties as the Commission may request. Upon request from the U.S. Department of Energy (DOE), the ACRS provides advice on U.S. Naval reactor designs and hazards associated with the DOE's nuclear activities and facilities. In addition, upon request, the ACRS provides technical advice to the Defense Nuclear Facilities Safety Board.

ACRS operations are governed by the *Federal Advisory Committee Act* (FACA), which is implemented through NRC regulations at Title 10, Part 7, of the *Code of Federal Regulations* (10 CFR Part 7). ACRS operational practices encourage the public, industry, State and local governments, and other stakeholders to express their views on regulatory matters.

MEMBERS OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

Dr. George E. Apostolakis, Professor of Nuclear Engineering, Professor of Engineering Systems, Massachusetts Institute of Technology, Cambridge, Massachusetts

Dr. Mario V. Bonaca, Retired Director, Nuclear Engineering Department, Northeast Utilities, Connecticut

Dr. Richard S. Denning, Senior Research Leader, Battelle Memorial Institute, and Adjunct Professor, the Ohio State University, Columbus, Ohio

Dr. Thomas S. Kress, Retired Head of Applied Systems Technology Section, Oak Ridge National Laboratory, Oak Ridge, Tennessee

Dr. Dana A. Powers, Senior Scientist, Sandia National Laboratories, Albuquerque, New Mexico

Dr. Victor H. Ransom, Professor Emeritus, Purdue School of Nuclear Engineering, West Lafayette, Indiana

Dr. William J. Shack,(Vice-Chairman), Associate Director, Energy Technology Division, Argonne National Laboratory, Argonne, Illinois

Mr. John D. Sieber, (Member-at-Large), Retired Senior Vice-President, Nuclear Power Division, Duquesne Light Company, Pittsburgh, Pennsylvania

Dr. Graham B. Wallis, (Chairman), Sherman Fairchild Professor Emeritus, Thayer School of Engineering, Dartmouth College, Hanover, New Hampshire

ACKNOWLEDGMENT

The Committee would like to acknowledge the contributions of Dr. Hossein Nourbakhsh and Mr. Sam Duraiswamy of the ACRS Staff to the development of this assessment.

ABSTRACT

In this report, the Advisory Committee on Reactor Safeguards (ACRS) presents the results of its assessment of the quality of selected research projects sponsored by the Office of Nuclear Regulatory Research(RES) of the NRC. An analytic/deliberative methodology was adopted by the Committee to guide its review of research projects. The methods of multi-attribute utility theory were utilized to structure the objectives of the review and develop numerical scales for rating the project with respect to each objective. The results of the evaluations of the quality of the three research projects are summarized as follows:

- Station Blackout Risk Evaluation for Nuclear Power Plants performed as a part of Standardized Plant Analysis Risk (SPAR) Models Development Program

  - This project was found to be more than satisfactory. The results meet the research objectives.

- Steam Generator Tube Integrity Program at the Argonne National Laboratory

  - This project was found to be satisfactory. The results meet the research objectives.

- Analysis of Rod Bundle Heat Transfer Facility Two-Phase Interface Drag Experiments at the Penn State University

  - This project marginally satisfied the research objectives. The Committee identified significant deficiencies.

CONTENTS

FIGURES

TABLES

ACRONYMS

Acronym	Definition
ACRS	Advisory Committee on Reactor Safeguards
AEC	Atomic Energy Commission
ANL	Argonne National Laboratory
B&W	Babcock & Wilcox
BWR	Boiling Water Reactor
CFR	Code of Federal Regulations
DOE	Department of Energy
EDG	Emergency Diesel Generator
FACA	Federal Advisory Committee Act
FY	Fiscal Year
GPRA	Government Performance and Results Act
LOCA	Loss-of-Coolant Accident
MAUT	Multi-Attribute Utility Theory
NRC	Nuclear Regulatory Commission
PRA	Probabilistic Risk assessment
PWR	Pressurized Water Reactor
RES	Office of Nuclear Regulatory Research
SBO	Station Blackout
SG	Steam Generator
SPAR	Standardized Plant Analysis Risk

1 INTRODUCTION

The Nuclear Regulatory Commission (NRC) maintains a safety research program to ensure that the agency's regulations have sound technical bases. The research effort is needed to support regulatory activities and agency initiatives while maintaining an infrastructure of expertise, facilities, analytical tools, and data to support regulatory decisions.

The Office of Nuclear Regulatory Research (RES) is required to have an independent evaluation of the effectiveness (quality) and utility of its research programs. This evaluation is required by the NRC Strategic Plan that was developed as mandated by the Government Performance and Results Act (GPRA). The Advisory Committee on Reactor Safeguards (ACRS) has agreed to assist RES by performing independent assessments of the quality of selected research projects. Quality assessment of individual research projects constitutes a new undertaking for the Committee; one that is quite different in scope and depth in comparison to the ACRS biennial review of the overall NRC research activities. During fiscal year (FY) 2004, the ACRS conducted a trial review of the quality of selected research projects [Ref. 1]. Based on the outcome of this trial review , the Committee has established the following review process:

- RES submits to the ACRS a list of candidate research projects for review because they have reached sufficient maturity that meaningful technical review can be conducted.
- The ACRS selects no more than four projects for detailed review during the fiscal year.
- A panel of three ACRS members is established to assess the quality of each research project.
- The panel follows the guidance developed by the ACRS full Committee in conducting the technical review. This guidance is discussed further below.
- Each panel assesses the quality of the assigned research project and presents an oral and a written report to the ACRS full Committee for review. This review is to ensure uniformity in the evaluations by the various panels.
- The Committee revises these reports, as needed, and provides them to the cognizant research manager, as appropriate.
- The Committee submits an annual summary report to the RES Director.

An analytic/deliberative decisionmaking framework was adopted for evaluating the quality of NRC research projects. The definition of quality research adopted by the Committee includes two major characteristics:

- Results meet the objectives
- The results and methods are adequately documented

Within the first characteristic, ACRS considered the following general attributes in evaluating the NRC research projects:

- Soundness of technical approach and results
  - Has execution of the work used available expertise in appropriate disciplines?

- Justification of major assumptions
  - Have assumptions key to the technical approach and the results been tested or otherwise justified?

- Treatment of uncertainties/sensitivities
  - Have significant uncertainties been characterized?
  - Have important sensitivities been identified?

Within the general category of documentation, the projects were evaluated in terms of following measures:

- Clarity of presentation

- Identification of major assumptions

In this report, the ACRS presents the results of its assessment of the quality of the research projects associated with:

- Station Blackout Risk Evaluation for Nuclear Power Plants performed as a part of Standardized Plant Analysis Risk (SPAR) Models Development Program

- Steam Generator Tube Integrity Program at the Argonne National Laboratory

- Analysis of Rod Bundle Heat Transfer Facility Two-Phase Interface Drag experiments at the Penn State University

These projects were selected from a list of candidate projects suggested by RES.

A fourth research project on reactor containment performance being conducted at Sandia National Laboratories will be evaluated during FY-2006, once a particularly pivotal report on this research becomes available.

The methodology for developing the quantitative metrics (numerical grades) for evaluating the quality of NRC research projects is presented in Section 2 of this report. The results of assessment and ratings for the selected projects are discussed in Section 3.

2 METHODOLOGY FOR EVALUATING THE QUALITY OF RESEARCH PROJECTS

To guide its review of research projects, the ACRS has adopted an analytic/deliberative methodology [Ref. 2 and 3]. The analytical part utilizes methods of multi-attribute utility theory (MAUT) [Ref. 4 and 5] to structure the objectives of the review and develop numerical scales for rating the project with respect to each objective. The objectives were developed in a hierarchical manner (in the form of a "value tree"), and weights reflecting their relative importance were developed. The value tree and the relative weights developed by the full Committee are shown in Figure 1.

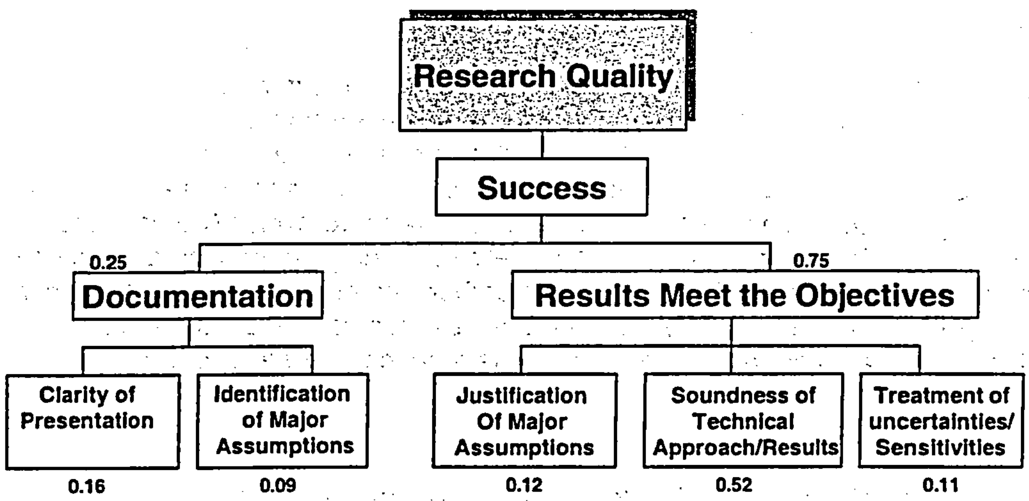

Figure 1 The value tree used for evaluating the quality of research projects

The quality of projects is evaluated in terms of the degree to which the results meet the objectives of the research and of the adequacy of the documentation of the research. It is the consensus of the ACRS that meeting the objectives of the research should have a weight of 0.75 in the overall evaluation of the research project. Adequacy of the documentation was assigned a weight of 0.25. Within these two broad categories, research projects were evaluated in terms of subsidiary "performance measures":

- justification of major assumptions (weight: 0.12)
- soundness of the technical approach and reliability of results (weight: 0.52)
- treatment of uncertainties and characterization of sensitivities (weight: 0.11)

Documentation of the research was evaluated in terms of the following performance measures:

- clarity of presentation (weight: 0.16)
- identification of major assumptions (weight: 0.09)

To evaluate how well the research project performed with respect to each performance measure, constructed scales were developed as shown in Table 1. The starting point is a rating of 5, Satisfactory (professional work that satisfies the research objectives). Often in evaluations of this nature, a grade that is less than excellent is interpreted as pejorative. In this ACRS evaluation, a grade of 5 should be interpreted literally as satisfactory. Although innovation and excellent work are to be encouraged, the ACRS realizes that time and cost place constraints on innovation. Furthermore, research projects are constrained by the work scope that has been agreed upon. The score was, then, increased or decreased according to the attributes shown in the table. The overall score of the project was produced by multiplying each score by the corresponding weight of the performance measure and adding all the weighted scores.

The value tree, weights, and constructed scales were the result of extensive deliberations of the whole ACRS. As discussed in Section 1, a panel of three ACRS members was formed to review each selected research project. Each member of the review panel independently evaluated the project in terms of the performance measures shown in the value tree. The panel deliberated the assigned scores and developed a consensus score, which was not necessarily the arithmetic average of individual scores. The panel's consensus score was discussed by the full Committee and adjusted in response to ACRS members' comments. The final consensus scores were multiplied by the appropriate weights, the weighted scores of all the categories were summed, and an overall score for the project was produced. A set of comments justifying the ratings was also produced.

Table 1. Constructed Scales for the Performance Measures

SCORE	LABEL	INTERPRETATION
10	Outstanding	Creative and uniformly excellent
8	Excellent	Important elements of innovation or insight
5	Satisfactory	Professional work that satisfies research objectives
3	Marginal	Some deficiencies identified; marginally satisfies research objectives
0	Unacceptable	Results do not satisfy the objectives or are not reliable

3. RESULTS OF QUALITY ASSESSMENT

3.1 STATION BLACKOUT RISK EVALUATION FOR NUCLEAR POWER PLANTS PERFORMED AS A PART OF SPAR MODELS DEVELOPMENT PROGRAM

In 1988, the NRC issued the Station Blackout Rule, 10 CFR 50.63, and the associated Regulatory Guide 1.155 establishing requirements and guidance to ensure decay heat removal for the period following loss-of-offsite power. Subsequent Probabilistic risk assessments (PRAs) indicated that compliance with these regulatory documents resulted in appropriately small core damage frequencies for station blackout (SBO) scenarios. On August 14, 2003, a widespread grid-related loss-of-offsite power event resulted in the controlled shut down of nine nuclear power plants. The NRC initiated a program to reevaluate the frequencies and durations of loss-of-offsite power, as well as the SBO risk contribution. The results of this study are documented in Reference 6. This report that the Committee reviewed is an update of previous reports analyzing the risk from loss-of-offsite power and subsequent SBO events in all operating U.S. power plants.

The SPAR models were used to evaluate the core damage frequency from internal events only for each plant during power operation. A number of enhancements to the SPAR models had to be made for this evaluation. The reliability estimates for diesel generators were also updated using recent data. Updated data were also collected for turbine-driven pumps, high-pressure core spray motor-driven pumps, and diesel-driven pumps. For the pressurized water reactors (PWRs), pump-seal failure models were selected based on the most recent developments.

The scope of this quality review is limited to the above report rather than a broader assessment of the quality of the updated SPAR models requested by RES. The Committee judged that it would have been overly ambitious to undertake such an evaluation in a single step and within the time constraints of the present review. The ACRS decided to have its Reliability and Probabilistic Risk Assessment Subcommittee perform a much broader review of the SPAR models during the upcoming year. Thus, in evaluating this report, the Committee has not considered the validity of the SPAR models that form the basis for the study.

GENERAL OBSERVATIONS

This report is an excellent example of the value of the SPAR Models Development Program and of the contribution that RES can make to the understanding of the safety of operating plants. The independent capability to evaluate risk issues across the population of operating plants has great value. By utilizing the same model and assumptions for all types of reactors in the fleet, the staff has been able to reach several conclusions regarding the effects of plant-specific design features on the risk from SBO. The availability of these models allows for periodic reevaluation of issues and trends associated with, for example, the effect of deregulation on grid reliability, and the effect of online maintenance on SBO.

The consensus scores for this project are shown in Table 2. This project was found to be more than satisfactory with a number of elements of excellence present. Comments and conclusions within the evaluation categories are:

Table 2. Summary Results of ACRS Assessment of the Quality of the Project on Station Blackout Risk Evaluation for Nuclear Power Plants

Performance Measures	Consensus Scores	Weights	Weighted Scores
Clarity of presentation	7.0	0.16	1.12
Identification of major assumptions	7.0	0.09	0.63
Justification of major assumptions	6.33	0.12	0.76
Soundness of technical approach/results	6.66	0.52	3.46
Treatment of uncertainties/sensitivities	6.0	0.11	0.66
		Overall Score:	**6.63**

Documentation

- Clarity of presentation **(Consensus score = 7.0)**

  The report is clearly written and well organized. It provides a good description of prior work and describes in detail the logic utilized in the selection of databases and assumptions. It presents the results in the context of previous evaluations, provides good explanation of changes, and discusses important trends and insights.

- Identification of major assumptions **(Consensus score = 7.0)**

  Assumptions are clearly stated, and the report does a good job of explaining the logic behind these assumptions.

Results Meet Objectives

- Justification of major assumptions **(Consensus score = 6.33)**

  Major assumptions are generally well justified, for example the use of industry-average data rather than plant-specific data for component unreliability, train test and maintenance outage probabilities, and initiating event frequencies.

In some instances, a full explanation is not provided. For example, no argument is provided for not modifying the Babcock & Wilcox (B&W) seal leakage model, except that there is no pending submittal to the NRC. From that statement, the reader is left with no insights regarding the quality of the B&W seal leakage model. Another example is the choice of a factor of two in the emergency diesel generator (EDG) performance sensitivity study. It is not clear why a factor of two was chosen.

- Soundness of technical approach and results **(Consensus score = 6.66)**

  There is nothing novel about the approach (this is not a criticism). The event trees are borrowed from those that had been developed previously.

  The use of industry-wide data to place all nuclear power plants on a common basis helped in determining the relative effectiveness of general features of electric power systems and backup safe shutdown modes in reducing the risk from SBO.

- Treatment of uncertainties and characterization of sensitivities **(Consensus score = 6.0)**

  The report includes the results of an uncertainty analysis and of a sensitivity study.

  The sensitivity results are point estimates, i.e., no uncertainty analyses were performed for the sensitivity cases. It is this last point that generated discussion among the panel members. What does a "sensitivity analysis" mean in the probabilistic world? In traditional engineering analysis where all the calculations were done on a "point estimate" basis, a sensitivity study usually means to vary, more or less arbitrarily, various parameters and evaluate their impact on the final answer. In probabilistic analyses, this approach must be reconsidered. Possible variability in parameter values should be included in the uncertainty distributions of these parameters. The focus should be on the assumptions and parameters that drive the results. An example is the use of the risk achievement worth to identify events that may have a significant impact on the core damage frequency calculated in a PRA. The ACRS acknowledges that this issue should be discussed in a broader context with the staff and that, perhaps, it would be unfair to judge the authors of this report harshly on an issue that has not been widely debated.

3.2 STEAM GENERATOR TUBE INTEGRITY PROGRAM AT THE ARGONNE NATIONAL LABORATORY[1]

The overall objective of the steam generator tube integrity research program is to provide experimental data and predictive correlations and models needed to permit the NRC staff to independently evaluate the integrity of steam generator tubes as plants age and degradation proceeds, new forms of degradation appear, and as new defect-specific management schemes are implemented. This program builds upon the results of NRC steam generator tube integrity and inspection research conducted since 1977.

The objectives of the specific project (task 3, Research on Tube Integrity and Integrity Predictions) selected for quality assessment were to:

- Determine if the flow stress of MA Nickel Alloy 600 tube material exhibits dependence on the stress rate or the strain rate (i.e.: the rate of internal pressurization).
- Determine the relationship between crack or ligament size (width, depth, and length), orientation, geometry, morphology, and number of ligaments and the tube leak rate and burst pressure.
- Confirm the validation of the tube leak rate correlation model and its relevance to choked two-phase flow expected at operating temperatures and pressures, including the relative uncertainties involved under various conditions.
- Compare laboratory leak rate and burst pressure models with the results of tests of samples of defective steam generator tubes removed from a decommissioned steam generator from McGuire Nuclear Plant.

These studies were conducted at the Argonne National Laboratory. The results of studies that the ACRS reviewed were documented in References 7 and 8.

The consensus scores for this project are shown in Table 3. This project was found to be satisfactory. The results meet the research objectives. Comments and conclusions within the evaluation categories are:

Documentation

- Clarity of presentation **(Consensus score = 4.7).**

  The manuscripts documenting the results of this project [Ref. 7 and 8] are exceptionally informal. These documents read like laboratory reports prepared by technicians and sent to professional staff to be used in the preparation of a more

[1]Dr. William J. Shack, ACRS member, did not participate in the Committee's deliberations regarding this matter.

Table 3 Summary Results of the ACRS Assessment of the Quality of the Project on Steam Generator Tube Integrity

Performance Measures	Consensus Scores	Weights	Weighted Scores
Clarity of presentation	4.7	0.16	0.75
Identification of major assumptions	4.7	0.09	0.42
Justification of major assumptions	4.7	0.12	0.56
Soundness of technical approach/results	5.0	0.52	2.6
Treatment of uncertainties/sensitivities	4.3	0.11	0.47
		Overall Score:	**4.8**

The reports are inadequate for the archival documentation of expensive tests. Experimental methods are mentioned in casual ways with no effort, even by reference, to show that these methods are adequate or produce reliable, reproducible results.

Calibration and qualification of instruments are not discussed at all.

Theoretical models and even data analysis methods are mentioned without reference.

Figures showing data and correlations are exceptionally difficult to interpret since minimal legends and labeling are employed despite the figures being quite "busy." For the leak rate studies (page 34 of Ref. 7),except for specimen SLG900, no results are provided. The discussion on page 44 is not clear when correlating L/D ratios and choked flow.

A reader who does not routinely examine reports from this laboratory and is not intimately familiar with the equipment and methods of the laboratory will have difficulty in understanding the documentation. (Only after reading Ref 8 did one come to understand that the unlabeled scale in some photos in Ref. 7 was an inch scale and not a centimeter scale despite all the text on lengths referring to millimeters!) In the end, one can understand the points the authors are trying to make in Ref. 7, but with difficulty. Clarity of presentation is not of high quality, but adequate to understood the work.

It is dubious that the experimental results could ever be used directly in a regulatory process involving licensees. The qualification of methods and calibration of instruments simply will not be acceptable for such direct use.

- Identification of major assumptions **(Consensus score = 4.7)**

  The major assumptions employed are not separately and explicitly stated but some of these assumptions are embedded in the text. In a complex report such as this, it is an acceptable and appropriate practice to state assumptions in the context of the issues where they are used or evaluated and rejected.

  As noted above, identification and justification of assumptions are difficult to evaluate. There is not a coherent effort to do this in the document largely because it is not evident that results have any applicability. It is not evident that the results for the notched specimens discussed in the document will be used to infer the behavior of real cracks in tubes under accident conditions.

  The investigators have done a better job in identifying factors that will affect the experimental results and including their sensitivities in test programs.

  The documentation does not provide adequate justification for sensitivities that are included nor does it include discussions concerning the sensitivities of other factors that has not been considered.

  The document fails completely to address uncertainties in measurements or to provide adequate descriptions of parametric uncertainties in reporting results of fitting the data to correlations. Presumably, if needed, these uncertainties as well as uncertainties in measurements could be extracted. Therefore, only a modest reduction in the score has been imposed.

Results Meet Objectives

- Justification of major assumptions **(Consensus score = 4.7)**

  Certain assumptions are implicit in the statement of scope. However, the work plan and scope were designed so that the major assumptions would be tested experimentally to verify the validity of these assumptions. An example was the assumption that flow stress is virtually independent of the rate at which stress and strain are applied to the specimen. This assumption had its origins in earlier test work performed by others prior to the in-depth study undertaken in this project. ANL could not confirm the validity of this assumption and undertook an effort to determine why a rate effect was observed in their tests and not in the earlier tests. Other examples of implicit assumptions involved issues such as ligament linkage and its relationship to both leakage and burst pressure, the quantification of choke flow leakage through cracks with two-phase flow, and the existence of a correlation between leakage and crack growth. The investigators did not make an explicit effort to identify and justify these assumptions.

In connection with the development of failure 'maps', it is asserted that the complex ligament geometries of real cracks can be idealized as either solely axial, solely circumferential, or radial. The report does not include any discussion on how close those assumptions are to reality. As noted above, an assumption about application of correlation developed for two cracks being applicable to configurations with four and six cracks is neither articulated nor justified.

In some cases, the assumed level of familiarity with previous work limits the discussion to the extent that the bases for assumptions are not clear. For example, in the predictions of ligament rupture against the McGuire tests, the ligament rupture pressure of each test was predicted by the equivalent rectangular crack methods. There is no explanation of why this is the appropriate model. An explanation would be worthwhile given that the benchmark is only partially successful. The abstract states that this is the "latest correlation." But some additional explanation would have contributed to a better understanding.

Much of the work on main steamline break effects on damaged tubes (Ref.8) relies on analytical simulation with TRAC-M and RELAP-5 codes. The ability of these codes to model appropriately pressure drops in complex geometries such as those of steam generator tube bundles and tube support plates has been questioned. The report does not discuss this issue. There are good comparisons of results from the two codes and finite-element analysis results, but applicability of these models is an important issue that deserves some discussion.

- Soundness of technical approach and results **(Consensus score = 5.0)**

The scope of work was thorough in identifying the major steps and the technical approach to be used by the investigators. The investigators used sound scientific and engineering methods to conduct these investigations. In addition, it is clear that the investigators followed up on anomalies and results that differed from prior assumptions to gain insights into the phenomenon that they were investigating. These new insights were factored into the analytical models under development to the extent that they could be, and uncertainties were estimated for data that had a range of numerical results. The investigators stated that the models provided conservative predictions.

Though quibbles abound in the review of the technical approach, no flaws were identified that would detract from the value of the results in any major way. On the other hand, the technical approaches adopted in the following four efforts were not inspired, so no bases for higher scores were identified either.

Pressurization rate effects

The first reported task was the confirmation of claims that rupture of flawed tubes is dependent on the rate of pressurization. The approach undertaken was to test a variety of flawed tubes similar to those used by investigators making the claim of a pressurization rate effect. The testing was, however, done in a consistent fashion unlike the testing done by those making the claims.

Testing was done at pressurization rates that varied from quasi-static to greater than 69 MPa/s. This range included, apparently, the pressurization rate used by those making the claims of a pressurization rate effect. Whether it includes prototypic pressurization rates is not stated, but it appears likely that it did. Tests were done at enough pressurization rates that it should be possible to infer by interpolation results for any pressurization rate likely to be of practical interest. This appears to be a technically sound and defensible approach.

In addition, tests are planned on cracks that were formed by a stress corrosion cracking process. The results of these tests will be presumably used to relate the results of tests with machined flaws to more realistic cracks. Again, this seems a prudent and reasonable approach.

Development of failure maps

To prepare failure maps, the authors have correlated data on the ligament ruptures of two types of flaws in tubes. A simple polynomial model has been used for correlation and it does not seem to have been selected based on some theoretical considerations. Details of the procedure for fitting the data to correlations are not spelled out to any extent. It is apparent that the polynomial is a very approximate description of the data and the parametric values must be changed for different crack lengths. Fitting apparently neglected the uncertainties in the data. Had these uncertainties been recognized, it might have been possible to use simpler correlation expressions. A similar polynomial correlation was developed for rupture pressure for the case of two cracks separated by a circumferential ligament. It appears that the data used for correlation may have come from room temperature tests, but documentation is not definitive on this point, and salient references have not yet been retrieved.

The correlations were then used to develop maps of crack length versus ligament width showing behavior for various pressure differences and crack geometries assuming 80 and 90% through-wall cracks. This approach is common and technically sound for maps involving two cracks separated by an axial or a radial ligament, provided that the correlations developed from test data are applicable at the assumed 300°C.

Maps were also prepared for cases with four and six cracks. There seems to be no demonstration that the correlations of ligament rupture and tube rupture obtained for two cracks are applicable to cases with four or more cracks. To be sure, there is an extrapolation taking place here that is not especially well highlighted in the documentation. Nevertheless, one must concede that if this extrapolation is palatable, the approach adopted in preparing the maps is a widely accepted one. Use of the maps, on the other hand, would demand a great deal more than is attempted in this limited effort. A reader would benefit from some comparison of the map predictions to data for the multiple crack cases.

Leak Rate Studies

The leak rate studies were undertaken to determine the limits of applicability with respect to the through-wall crack length and crack tightness of the simple orifice model for predicting leak rates of cracked tubes. The effort undertaken focused on conditions that will lead to "flashing" of the coolant within the crack. Crack length divided by the hydraulic diameter of the crack was used as the metric for cracks in tubes used in the tests. This is acceptable because realistic cracks are used in the test program. Analysis of the results was supplemented by data from the literature concerning flow through better instrumented slits in plates. The technical approach appears to be adequate to the task.

Results obtained in the effort only address conditions for subcooling in the range of 50-60°C . Such a subcooling range corresponds to cold leg conditions. A plausibility argument is advanced that "conservative" results will be predicted for hot leg conditions that are more appropriate for issues associated with steam generator tube leakage. Thus, results only marginally meet the objective if the objective is to find limits of applicability of the orifice model for conditions where it is likely to be of interest to apply.

Rupture and Leak Rate Predictions for McGuire Steam Generator Tubes

The technical approach for this effort involved acquisition of flawed tubes form the McGuire plant and characterization of the flaws first by nondestructive examianation methods and later by fractography. The tubes were then tested for leakage in a facility that is presumably well established and well described in some other publications. Unfortunately, no references were provided to validate this presumption. No description of the method for measuring leak rates was provided. Presumably, a well established method exists and the authors could have informed the reader about this method by means of a reference. Though poorly documented, the technical approach appears sound.

- Treatment of uncertainties and characterization of sensitivities **(Consensus score = 4.3)**

The comparison of predictive models of leak rate and rupture as applied to actual tubes removed from a retired McGuire steam generator with leakage and burst test data of these tubes showed reasonable agreement. In the discussion, explanations were provided as to why the predictive models differed from the actual test results. A range of uncertainty and the degree of conservatism between the models and observed results were estimated, in order to establish the degree of usefulness of the correlations developed. Because of the complex nature of stress corrosion cracks, predictive uncertainty exists and has been estimated and factored into the resulting conclusions.

The investigators do a rather good job in developing their experimental projects in considering sensitivities such as sensitivity to the number of cracks, ligament sizes, crack orientation and the like. The investigators have not estimated uncertainties associated with any measured value that they report. Where they have fit data to a parametric correlation, they have failed to cite any uncertainties in the parametric values and certainly have not reported covariance matrices for models involving more than two parameters. They do not report on the uncertainties of predictions derived from correlations. Episodically, the authors report linear correlation coefficients that are essentially useless in the interpretation of the quality of a fit of a parameterized equation to data without a great deal more information about the fitting results.

The adequacy of the investigators' treatments of sensitivities in the development of their research efforts is acknowledged. Neglect of uncertainties in reports of measurements is the basis for reduction of the score in this category.

3.3 ANALYSIS OF ROD BUNDLE HEAT TRANSFER FACILITY TWO-PHASE INTERFACE DRAG EXPERIMENTS AT THE PENN STATE UNIVERSITY

The objective of a task at the Penn State University was to analyze data that had been collected in the Rod Bundle Heat Transfer Facility in order to gain insights to be used in the development and validation of the TRACE computer code. The specific set of data was collected to examine level swell under reflood conditions. The rod bundle in the experimental setup simulates a PWR fuel assembly with spacer grids, as in the standard 17x17 Westinghouse array. The experimental bundle involves a 7x7 array of full length, electrically heated fuel pins. The principal data collected in the experiments were the pressure drop along the length of the pins with varied reflood flow rate, power level, and inlet subcooling. Other properties of the flow, such as void fraction, interfacial drag force, and the product of interfacial area and friction factor, were determined by inference from a simplified model of energy conservation.

The data are said to be "more detailed" than previous data, but no comparisons are made to illustrate why, or to show consistency (or otherwise) with previous work.

The review is based on the only report [Ref. 9] that was provided to the Committee of results from the test program. It is entitled "Analysis of Rod Bundle Heat Transfer Test Facility Two-Phase Interfacial Drag Experiments." It has no number and is believed to be a draft. The title of the report is somewhat misleading, since there were no measurements of interfacial drag. The only parameter measured, apart from those defining the boundary conditions of the experiment, such as flow rate, power supplied etc., was the pressure drop over several lengths of a rod bundle.

The broader experimental program, which represents a substantial undertaking, with extensive measurement of parameters such as temperature, droplet size, and velocity, was not part of this review.

The Committee also had the benefit of an earlier report describing the test facility and of the RES Thermal-hydraulics Research Plan, dated March 1, 2005. RES provided a memo dated June 6, 2005 entitled, "Usage of Data from the Rod Bundle Heat Transfer Test".

The consensus scores for the project are shown in Table 4. This project marginally satisfied the research objectives. The Committee identified important deficiencies. Comments and conclusions within the evaluation categories are:

Documentation

- Clarity of presentation **(Consensus score = 4.33)**

  The report is readable and it is reasonably clear on what was done. However, the objectives of the work are not clearly stated.

Table 4 Summary Results of the ACRS Assessment of the Quality of the Project on Analysis of Rod Bundle Heat Transfer Facility Two-Phase Interface Drag Experiments at the Penn State University

Performance Measures	Consensus Scores	Weights	Weighted Scores
Clarity of presentation	4.33	0.16	0.69
Identification of major assumptions	4.0	0.09	0.36
Justification of major assumptions	3.33	0.12	0.40
Soundness of technical approach/results	3.33	0.52	1.73
Treatment of uncertainties/sensitivities	0.66	0.11	0.07
		Overall Score:	**3.25**

Figures are mostly clear but some lack essential details. Descriptions of the location of pressure taps are inconsistent.

The report requires substantial manipulation of pressure drop data to infer void fraction, interfacial drag force, and the product of interfacial area and friction factor but the main report does not explain how these properties are obtained. The reader has to study the appendices to determine the assumptions and theory applied.

- Identification of major assumptions **(Consensus score = 4.0)**

"Correction" of data is described but insufficiently to provide understanding of how spacers were treated, or why certain flow regimes were used to predict terms needed to convert from pressure drop to void fraction. These assumptions prejudice the eventual use for TRACE development, since they are in parallel to the comparisons with TRACE. It would be better to have TRACE predict the raw data.

The assumption that the pressure drop does not influence fluid properties appears to be used but is not identified.

The assumption that the only source of vapor generation is the addition of heat ignores the significant effect of flashing that is not identified.

The assumption that "the total pressure drop is small" is incorrect. Since the pressure drop along the bundle can be substantial (almost 6psi), specification of a single "pressure" (e.g. 20psia) for each experiment is inadequate without identifying clearly where it is measured.

Results Meet Objectives

- Justification of major assumptions **(Consensus score = 3.33)**

  Several inappropriate flow regimes are used.

  The energy balance is erroneous, omitting an important "flashing" term, leading to inaccurate prediction of quality.

  Property changes along the bundle due to pressure drop are ignored, though they are influenced by pressure and temperature changes.

  The effect of spacers on the flow pattern, pressure drop, and void fraction is not explained. In "correcting" the pressure drop measurements to compute a void fraction, some justification is provided for the friction pressure drop correction, but none for the acceleration pressure drop correction.

- Soundness of technical approach and results **(Consensus score = 3.33)**

  It is doubtful if the results are useful for TRACE development. There is no discussion of models currently in TRACE or direct comparison with these models.

  The presentation and reduction of data contain errors and there is no investigation of the effects of assumptions.

  Several of the comparisons with theory are inappropriate. There is no critical examination of features of the data, such as large fluctuations in the pressure drop data and the apparent lack of steady state in some tests.

  Since the intent of the report is to derive interfacial drag, there should be more information on how this was done, the sources of error, the effect of parameters, the effect of spacers, etc. Only one example is given, and it appears to have a basic flaw, since the large spikes of extreme values that are predicted indicate the flow to be close to homogeneous, which is inconsistent with evidence provided by the void fraction results.

- Treatment of uncertainties and characterization of sensitivities **(Consensus score = 0.66)**

  There is no treatment or discussion of uncertainties.

4. REFERENCES

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

2. National Research Council, *Understanding Risk: Informing Decisions in a Democratic Society*. National Academy Press, Washington, DC, 1996.

3. Apostolakis, G.E., and Pickett, S.E., "Deliberation: Integrating Analytical Results into Environmental Decisions Involving Multiple Stakeholders," *Risk Analysis*, 18:621-634, 1998.

4. Clemen, R., *Making Hard Decisions*, 2nd Edition, Duxbury Press, Belmont, CA, 1995.

5. Keeney, R.L., and H. Raiffa, *Decisions with Multiple Objectives: Preferences and Value Tradeoffs*, Wiley, New York, 1976.

6. Eide, S. A., T.E. Wierman, and D.M. Rasmuson, " Station Blackout Risk evaluation for Nuclear Power Plants," Draft, NUREG/CR, INEEL/EXT-04-02525, Idaho National Engineering and Environmental Laboratory, January 2005.

7. Majumdar, S., K. Kasza, S. Bakhtiari, J. Oras, J. Franklin, and C. Vulyak, Jr., "Pressurization Rate Effect on Flawed Tube Rupture, Failure Maps for Complex Multiple Cracks, Validation of Leak Rate Correlation Model and Rupture and Leak Rate Predictions for McGuire Steam Generator Tubes," Draft, NUREG/CR-xxxx, Argonne National Laboratory, April 2004.

8. Majumdar, S., K. Kasza, J. Oras, J. Franklin and C. Vulyak, Jr., "Sensitivity Studies of Failure of Steam Generator Tubes during Main Steam Line Break and Other Secondary Side Depressurization Events," Draft, NUREG/CR-xxxx, Argonne National Laboratory , April 2004.

9. Hochreiter, L. E., F. B. Cheung, T. F. Lin, and D.J. Miller, "Analysis of Rod Bundle Heat Transfer Test Facility Two-Phase Interfacial Drag Experiments," The Pennsylvania State University, June 2005.

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

November 8, 2005

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: PROPOSED REVISION TO STANDARD REVIEW PLAN SECTION 12.5, "OPERATIONAL RADIATION PROTECTION PROGRAM"

During the 527th meeting of the Advisory Committee on Reactor Safeguards, November 3-5, 2005, the Committee considered the proposed revision to Standard Review Plan (SRP) Section 12.5, "Operational Radiation Protection Program." The Committee decided not to review the proposed revision to SRP Section 12.5 and has no objection to the staff's proposal to issue it for public comment.

Reference:
Memorandum dated October 18, 2005, from Bruce Boger, Director for the Division of Inspection Program Management, NRR, to John T. Larkins, Executive Director, ACRS, Subject: Request to Waive Advisory Committee on Reactor Safeguards (ACRS) Review of Proposed Revision to Standard Review Plan Section 12.5, "Operational Radiation Protection Program."

cc: A. Vietti-Cook, SECY
W. Dean, OEDO
J. Dixon-Herrity, OEDO
J. Dyer, NRR
B. Boger, NRR
T. Quay, NRR
R. Pedersen, NRR
T. Meeks, NRR

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

November 9, 2005

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: DRAFT FINAL GENERIC LETTER 2005-XX, "STEAM GENERATOR TUBE INTEGRITY AND ASSOCIATED TECHNICAL SPECIFICATIONS"

During the 527th meeting of the Advisory Committee on Reactor Safeguards, November 3-5, 2005, the Committee considered the draft final Generic Letter 2005-XX, "Steam Generator Tube Integrity and Associated Technical Specifications." The Committee plans to review the staff's evaluation of licensee responses to this Generic Letter. The Committee has no objection to the staff's proposal to issue the Generic Letter.

Reference:
Memorandum dated August 11, 2005 from Michael E. Mayfield, Director, Division of Engineering, NRR, to John T. Larkins, Executive Director, ACRS, Subject: Proposed Draft Generic Letter 2005-XX, "Steam Generator Tube Integrity and Associated Technical Specifications."

cc: A. Vietti-Cook, SECY
W. Dean, OEDO
J. Dixon-Herrity, OEDO
J. Dyer, NRR
B. Sheron, NRR
T. Meek, NRR
W. Bateman, NRR
A. Hiser, NRR
K. Karwoski, NRR
M. Banerjee, NRR

[illegible]

[illegible]

[illegible]

[illegible]

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

November 9, 2005

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

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

During the 527th meeting of the Advisory Committee on Reactor Safeguards, November 3-5, 2005, the Committee considered the proposed revision to NUREG-800, Standard Review Plan, Section 14.2.1, "Generic Guidelines for Extended Power Uprate Testing Programs," which was forwarded to the Committee for information. The Committee has decided to review this document.

Reference:
Memorandum dated October 26, 2005, from R. William Borchardt, Deputy Director, Office of Nuclear Reactor Regulation, to Sher Bahadur, Chairman, Committee to Review Generic Requirements, "Request to Waive the Committee to Review Generic Requirements Review of Standard Review Plan Section 14.2.1."

cc: A. Vietti-Cook, SECY
W. Dean, OEDO
J. Dixon-Herrity, OEDO
P. Prescott, NRR
T. Meek, NRR
J. Dyer, NRR
R. Borchardt, NRR
S. Bahadur, RES

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

November 15, 2005

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

for

SUBJECT: FINAL RULE - AP1000 DESIGN CERTIFICATION

During the 527th meeting of the Advisory Committee on Reactor Safeguards, November 3-5, 2005, the Committee considered the final AP1000 design certification rule. The Committee decided not to review the final rule and has no objection to the staff's proposal to issue it.

Reference:
Memorandum dated November 2, 2005, from David B. Mathews, Director, Division of New Reactor Licensing, Office of Nuclear Reactor Regulation to John T. Larkins, Executive Director, Advisory Committee on Reactor Safeguards, Subject: Final Rule - AP1000 Design Certification.

cc: A. Vietti-Cook, SECY
W. Dean, OEDO
J. Dixon-Herrity, OEDO
D. Mathews, NRR
W. Beckner, NRR
L. Dudes, NRR
J. Wilson, NRR
L. Quinones-Navarro, NRR
R. Assa, RES

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

November 18, 2005

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

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

Dear Chairman Diaz:

During the 527th meeting of the Advisory Committee on Reactor Safeguards, November 3-5, 2005, we completed our review of the license renewal application for the Point Beach Nuclear Plant (PBNP) Units 1 and 2, and the final Safety Evaluation Report (SER) prepared by the NRC staff. We issued an interim report on the safety aspects of this application and the draft SER on June 9, 2005. Our Plant License Renewal Subcommittee also reviewed this matter during a meeting on May 31, 2005. During these reviews, we had the benefit of discussions with representatives of the NRC staff and the Nuclear Management Company, LLC (NMC). We also had the benefit of the documents referenced. This report fulfills the requirements of 10 CFR 54.25 that the ACRS review and report on all license renewal applications.

RECOMMENDATIONS

1. With the inclusion of the conditions in Recommendation 2, the NMC application for license renewal of PBNP Units 1 and 2 should be approved.

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

BACKGROUND AND DISCUSSION

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

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

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

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

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

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

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

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

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

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

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

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

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

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

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

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

November 18, 2005

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

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

Dear Chairman Diaz:

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

CONCLUSIONS AND RECOMMENDATIONS

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

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

DISCUSSION

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

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

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

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

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

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

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

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

<u>References</u>:

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

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

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

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

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

November 18, 2005

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

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

Dear Mr. Reyes:

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

RECOMMENDATION

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

BACKGROUND AND DISCUSSION

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

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

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

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

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

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

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

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

November 21, 2005

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

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

Dear Mr. Reyes:

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

CONCLUSION

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

RECOMMENDATIONS

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

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

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

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

BACKGROUND

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

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

DISCUSSION

The draft plan divides the research into six areas:

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

The proposed research areas are comprehensive.

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

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

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

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

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

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

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References:

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

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

December 16, 2005

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: PROPOSED REVISION TO STANDARD REVIEW PLAN SECTION 17.5, "QUALITY ASSURANCE PROGRAM DESCRIPTION DESIGN CERTIFICATION, EARLY SITE PERMIT AND NEW LICENSE APPLICANTS"

During the 528th meeting of the Advisory Committee on Reactor Safeguards, December 7-10, 2005, the Committee considered the proposed revision to Standard Review Plan (SRP) Section 17.5, "Quality Assurance Program Description Design Certification, Early Site Permit and New License Applicants." The Committee has no objection to the staff's proposal to issue this proposed revision to SRP Section 17.5 for public comments. The Committee would like the opportunity to review the draft final revision after reconciliation of public comments.

Reference:
Memorandum dated November 23, 2005, from Michael E. Mayfield, Director, Division of Engineering, Office of Nuclear Reactor Regulation, to John T. Larkins, Executive Director, Advisory Committee on Reactor Safeguards, Subject: Request to Defer Advisory Committee on Reactor Safeguards Review of Proposed Revision to Standard Review Plan Section 17.5, "Quality Assurance Program Description Design Certification, Early Site Permit and New License Applicants."

cc: A. Vietti-Cook, SECY
W. Dean, OEDO
J. Dixon-Herrity, OEDO
M. Mayfield, NRR
P. Prescott, NRR
T. Meeks, NRR
R. Assa, RES

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

December 16, 2005

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: PROPOSED REVISION TO REGULATORY GUIDE 1.76, "DESIGN BASIS TORNADO AND TORNADO MISSILES FOR NUCLEAR POWER PLANTS," AND STANDARD REVIEW PLAN, SECTIONS 2.3.1, "REGIONAL CLIMATOLOGY," AND 3.5.1.4, 'MISSILES GENERATED BY TORNADO AND EXTREME WINDS"

During the 528th meeting of the Advisory Committee on Reactor Safeguards, December 7-10, 2005, the Committee considered the proposed revision to Regulatory Guide 1.76, and Standard Review Plan Sections 2.3.1 and 3.5.1.4. The Committee has no objection to the staff's proposal to issue the proposed revisions for public comments. The Committee would like the opportunity to review the draft final revision after reconciliation of public comments.

Reference:
Memorandum dated December 6, 2005, from James E. Lyons, Director, Division of Risk Assessment, Office of Nuclear Reactor Regulation, to John T. Larkins, Executive Director, Advisory Committee on Reactor Safeguards, Subject: Request to Defer Review of Proposed Revision to Regulatory Guide 1.76, "Design Basis Tornado and Tornado Missiles for Nuclear Power Plants," and Standard Review Plan, Sections 2.3.1 "Regional Climatology", and 3.5.1.4 "Missiles Generated by Tornado and Extreme Winds."

cc: A. Vietti-Cook, SECY
W. Dean, OEDO
J. Dixon-Herrity, OEDO
J. Lyons, NRR
R. Harvey, NRR
T. Meeks, NRR
R. Assa, RES

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

December 16, 2005

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

for

SUBJECT: DRAFT REGULATORY GUIDE DG-1120, "TRANSIENT AND ACCIDENT ANALYSIS METHODS," AND STANDARD REVIEW PLAN CHAPTER 15 SECTION 15.0.2, "REVIEW OF TRANSIENT AND ACCIDENT METHODS"

During the 526th meeting of the Advisory Committee on Reactor Safeguards, October 6-7, 2005, the Committee considered the latest version of the draft Regulatory Guide DG-1120, "Transient and Accident Analysis Methods," and Standard Review Plan Chapter 15 Section 15.0.2, "Review of Transient and Accident Methods," which were provided to the Committee for information. The Committee decided at that time to review these documents. During the 528th meeting, December 7-10, 2005, the Committee reconsidered this decision, and has decided that it does not wish to review these documents.

References:

1. Memorandum dated September 13, 2005, from Farouk Eltawila, Director, Division of Systems Analysis and Regulatory Effectiveness, Office of Nuclear Regulatory Research, to John T. Larkins, Executive Director, ACRS, Subject: Publication of Draft Regulatory Guide DG-1120 (DG-1120), "Transient and Accident Analysis Methods," and Standard Review Plan Chapter 15 Section 15.0.2 (SRP 15.0.2), "Review of Transients and Accident Methods."

2. Memorandum dated October 13, 2005, from John T. Larkins, Executive Director, ACRS, to Luis A. Reyes, Executive Director for Operations, Subject: Draft Regulatory Guide DG-1120 (DG-1120), "Transient and Accident Analysis Methods," and Standard Review Plan Chapter 15 Section 15.0.2 (SRP 15.0.2), "Review of Transient and Accident Methods."

cc: A. Vietti-Cook, SECY
W. Dean, OEDO
J. Dixon-Herrity, OEDO
F. Eltawila, RES
S. Marshall, RES
C. Paperiello, RES
J. Dyer, NRR
T. Meek, NRR
R. Landry, NRR
F. Akstulewicz, NRR
J. Wermiel, NRR

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

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

December 21, 2005

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

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

Dear Mr. Reyes:

During the 528th meeting of the Advisory Committee on Reactor Safeguards, December 7-10, 2005, we reviewed the draft final Generic Letter 2005-XX, "Impact of Potentially Degraded Hemyc/MT Fire Barrier Materials on Compliance with Approved Fire Protection Programs." During our review, we had the benefit of discussions with representatives of the NRC staff and the documents referenced.

RECOMMENDATION

The Generic Letter should be issued.

BACKGROUND AND DISCUSSION

A fundamental principle of a nuclear power plant fire protection program is the protection of at least one train of equipment required to safely shut down the plant. If a fire area contains components from more than one train of safe shutdown equipment, a means must be provided to ensure that a fire within that area would not disable both safe shutdown pathways. One acceptable approach is to provide a fire detection system, an automatic fire suppression system, and an 1-hour fire barrier to separate the trains. In the absence of a fire suppression system, adequate protection can be provided by a 3-hour fire barrier. A number of plants have used the Hemyc and MT electrical raceway fire barrier systems for this purpose. These fire barrier systems consist of Kaowool ceramic fiber insulation with a covering of Refrasil or Siltemp fire-resistant fabric.

The staff raised questions about the fire resistance capability of these systems during fire protection inspections conducted at nuclear plants in 1999 and 2000. The questions led to independent confirmatory testing by the NRC of Hemyc and MT materials at the Omega Point testing laboratory in March and April 2005. Tests were performed on assemblies of components protected by fire barriers. The results failed to meet acceptance criteria for all of the configurations. Shrinkage of the Refrasil covering resulted in gaps and tears in the blankets, allowing localized direct access of the cable to the fire environment. For the 1-hour barriers, failure typically occurred at approximately 30 minutes, and for the 3-hour barriers, failure typically occurred at approximately 120 minutes.

Similar issues with fire barrier materials have been identified in the past. The 1989 test results of the Thermo-lag fire barrier system indicated that it could not satisfy testing standards. The NRC issued a number of generic communications on this subject. These documents provide a precedent for the activities that are currently being undertaken to understand the scope of the Hemyc and MT issue and ensure that appropriate corrective actions are undertaken.

As a consequence of the test results, the staff developed a generic letter that requests the licensees to report whether Hemyc or MT fire barrier materials are installed and relied on for safe shutdown purposes. Also, the generic letter asks licensees to provide a description of existing programmatic controls to ensure that other types of fire barriers will be assessed for potential degradation and resulting adverse effects. Licensees that have installed Hemyc or MT fire barrier materials must describe the extent of installation, compliance with 10 CFR 50.48 in light of recent test findings, compensatory measures that have been implemented, and a general description of the plan and schedule for corrective actions.

The generic letter also states that affected licensees should provide confirmation by December 1, 2007 that their fire protection programs are in compliance with applicable regulatory requirements. They should also provide a summary of the evaluation used to determine the adequacy of the fire protection program in the presence of potentially degraded Hemyc or MT fire barriers, including the results of any supporting tests performed.

The results of the NRC's independent, confirmatory tests indicate that Hemyc and MT fire barrier systems may not provide adequate protection. The generic letter should be issued to seek information from the licensees. We look forward to a briefing by the staff after they have reviewed the responses submitted by the licensees.

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References:

1. Memorandum from J. Lyons, NRR, to J. Larkins, ACRS, dated November 8, 2005, Subject: Proposed Generic Letter 2005-XX, "Impact of Potentially Degraded Hemyc/MT Fire Barrier Materials on Compliance with Approved Fire Protection Programs" (ADAMS Accession No. ML053110527)
2. Memorandum from J. Larkins, ACRS, to L. Reyes, EDO, dated July 7, 2005, Subject: Proposed Generic Letter 2005-XX, "Impact of Potentially Degraded Hemyc/MT Fire Barrier Materials on Compliance with Approved Fire Protection Programs" (ADAMS Accession No. ML051940496)
3. Memorandum from D. Lew, RES, to J. Hannon, NRR, dated March 28, 2005, Subject: Preliminary Pass/Fail Test Results for Hemyc 1-Hour Rated Electrical Raceway Fire Barrier Systems (ADAMS Accession No. ML050880176)
4. Information Notice 2005-07, "Results of Hemyc Electrical Raceway Fire Barrier System Full Scale Testing," dated April 1, 2005 (ADAMS Accession No. ML050890089)
5. Regulatory Issue Summary 2005-07, "Compensatory Measures to Satisfy the Fire Protection Program Requirements," dated April 19, 2005 (ADAMS Accession No. ML042360547)

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

December 23, 2005

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

SUBJECT: EARLY SITE PERMIT APPLICATION FOR THE GRAND GULF SITE AND THE ASSOCIATED FINAL SAFETY EVALUATION REPORT

Dear Mr. Reyes:

During the 528th meeting of the Advisory Committee on Reactor Safeguards, December 7-10, 2005, we met with representatives of the NRC staff and System Energy Resources, Inc. (SERI), the applicant for an early site permit (ESP) for the Grand Gulf site, and discussed the application and the NRC staff's final Safety Evaluation Report (FSER). We provided an interim report on this application and the draft Safety Evaluation Report on June 14, 2005. We reviewed this application to fulfill the requirement of 10 CFR 52.23 that the ACRS report on those portions of an ESP application that concern safety. We also had the benefit of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

- The NRC staff has written a very readable and comprehensive Safety Evaluation Report. The three permit conditions the staff proposes for the early site permit and the 26 action items for the combined license phase are appropriate.

- This Safety Evaluation Report should be issued once the staff has made more explicit its analyses of the hazards posed to the proposed site by explosions in transportation accidents on the Mississippi River.

- The staff needs to provide additional guidance to applicants concerning the discussion in an application of "Major Features" of the emergency planning for a proposed site.

DISCUSSION

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

The SERI application for an early site permit does not specify a particular power plant technology for the new reactor or reactor modules to be placed on the site. The early site permit application, instead, uses a "plant parameter envelope" of power plant characteristics that is intended to bound the reactor technology that could eventually be selected.

- Nature of the Proposed Site

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

The nearest major airport is at Jackson, Mississippi, about 65 miles from the proposed site. The staff has determined that the air traffic corridors near the site pose no undue risk. There is a highway 4½ miles from the site. The principal ground transportation hazard is thought to be the delivery of hydrogen to the site for use in the currently operating boiling water reactor. The staff has found that the delivery and storage of this hydrogen would pose no undue risk to the proposed new power plant site.

The most important transportation route near the site is the Mississippi River. The nearest bank of this river is about 1.1 miles from the proposed site. Explosions and releases of toxic gases and vapors could pose threats to the proposed site. The staff and the applicant have agreed to defer consideration of the threats posed by the accidental releases of toxic vapors and gases until a specific plant for the site has been chosen and the habitability of the control room can be evaluated.

The staff has concluded that the detonation of 5000 tons TNT-equivalent bounds the explosion threat to the proposed site. According to staff-approved methods of analysis, such a detonation would require a standoff distance of about 2.1 miles from the facility. The staff concludes, however, that because the site is located behind a 65-foot bluff, the 1.1 mile standoff is adequate. The technical basis for this conclusion needs to be made clear in the Safety Evaluation Report prior to its issuance. This clarification should include a description of the reliability of the calculational method adopted by the staff.

The staff has concluded also that the detonation bounds the explosive hazard posed by vapor explosions such as might occur in the release of liquefied natural gas during a transportation accident on the river. The technical basis for this conclusion should also be made clear in the Safety Evaluation Report. The clarification should include a discussion of whether the staff used the TNT-equivalent method to analyze vapor explosions and the conservatisms associated with such an approximation if it was adopted.

- Population in the Vicinity of the Site

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

- Geology and Seismicity of the Site

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

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

- Meteorology

Vigorous storms such as hurricanes and tornados are the principal weather threats to a reactor located on the proposed site. SERI and the staff have used historical information to characterize these and other weather features of the site. In our review of the Safety Evaluation Report, we examined the applicability of hurricane frequency data on the prediction of future storm activity. There is evidence that storm activity is increasing in the Gulf of Mexico due to known weather cycles. The staff and the applicant have used historical data over a sufficient period to capture data from previous weather cycles. We find no definitive evidence that storm intensities in excess of the bounds established by the applicant and accepted by the staff will develop. These bounds may not be especially conservative. Representatives of SERI informed us that inland wind gusts produced by the recent hurricane Katrina at the latitude of the proposed site were somewhat less than 92 mph which can be compared to the 96 mph maximum three-second wind gust adopted for the site characterization. The staff has stated that should future weather evidence indicate site characteristics accepted in the Safety Evaluation Report are not adequate, these characteristics will be amended as needed.

The proposed site is located on a bluff about 65 feet above the normal river level. Land on the opposite bank of the river is more easily flooded than the proposed site. Consequently, major river flooding is not a threat to the site. Local, onsite flooding will have to be addressed if the permit is granted and a decision is made to construct a power plant on the site.

- Emergency Plans

The applicant has elected to submit for review just the "major features" of emergency planning for the proposed site, as is allowed by the regulations. The staff has concluded that these major features are largely adequate. The applicant has stated that the remaining information would be submitted with a combined license application. The applicant and the staff encountered challenges in defining the limitations that should exist on descriptions of major features of emergency planning, especially for a site where reactors currently exist. These challenges could be avoided in the future by providing additional guidance to the applicants.

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

References:

1. U.S. Nuclear Regulatory Commission, Final Safety Evaluation Report, "Safety Evaluation of Early Site Permit Application in the Matter of System Energy Resources, Inc., a Subsidiary of Entergy Corporation, for the Grand Gulf Early Site Permit Site," October 21, 2005.

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

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

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

January 4, 2006

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

SUBJECT: VERMONT YANKEE EXTENDED POWER UPRATE

Dear Chairman Diaz:

During the 528th meeting of the Advisory Committee on Reactor Safeguards, December 7-9, 2005, we discussed the Vermont Yankee Extended Power Uprate (EPU) Application. As part of this review, our Subcommittee on Power Uprates held a meeting on November 15 -16, 2005 in Brattleboro, Vermont to receive input from the public, the applicant, and the staff. A second Subcommittee meeting was held in Rockville, Maryland on November 29 - 30, 2005. During our review, we had the benefit of discussions with the staff, the public, and Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc. (Entergy), the licensee. We also had the benefit of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

1. The Entergy application for the extended power uprate at the Vermont Yankee Nuclear Power Station (VY) should be approved.

2. The change in the licensing basis associated with the requested containment overpressure credit should be approved.

3. Load rejection and main steam isolation valve closure transient tests are not warranted. The planned transient testing program adequately addresses the performance of the modified systems.

4. The times available to perform critical operator actions remain adequate under EPU conditions.

5. The margin added to the safety limit minimum critical power ratio (SLMCPR) is an appropriate interim measure until General Electric (GE) obtains additional data to complete the validation of nuclear analysis methods.

6. The monitoring that will be performed during the ascension to uprate power provides adequate assurance that, if resonant vibrational modes are induced in the steam dryer, they will be identified prior to component failure.

7. An enhanced, focused engineering inspection was performed. An additional expanded inspection is not warranted.

8. The review standard for extended power uprates (RS-001) provides a structured process for the review of applications for extended power uprates. Its continued use and improvement are encouraged.

BACKGROUND

Vermont Yankee Nuclear Power Station (VY) is a boiling-water reactor of the BWR/4 design with a Mark-1 containment. Entergy has applied for an extended power uprate of approximately 20% from the current maximum authorized power level of 1593 MWt to 1912 MWt. The application is similar to other uprates that have been approved within the last five years at Duane Arnold, Dresden Units 2 and 3, Quad Cities Units 1 and 2, and Brunswick Units 1 and 2. In Constant Pressure Power Uprates (CPPU), except for steam and feedwater flow rates, plant operating conditions are essentially unchanged from the pre-EPU values. The extra power is generated largely by flattening the power distribution across the core, and the fuel design safety limits are met at the proposed extended power uprate conditions.

DISCUSSION

When a large-break design-basis loss-of-coolant accident (LOCA) and anticipated transient without scram (ATWS) were analyzed at VY at the proposed EPU level using current design basis assumptions and methodologies, the available net positive suction head (NPSH) was found to be insufficient to avoid cavitation of the low pressure coolant injection (LPCI) and core spray pumps. The need for increased NPSH occurs because at the higher power level the suppression pool heats up more in both of these scenarios than at the currently licensed power level. In the calculations performed to support VY's existing operating license, containment pressure was assumed to be atmospheric when computing the available NPSH.

In its application, Entergy requests changing its licensing basis methodology to grant credit for containment accident pressure in determining available NPSH for emergency core cooling pumps for these LOCA and ATWS scenarios. Using conservative methods and a containment leak rate consistent with its technical specifications, Entergy has determined a conservative lower bound for the time-dependent pressure in containment that would result from these scenarios under EPU conditions. The incremental pressure credits that are requested for these two scenarios are less than these computed pressures. For the LOCA scenario, the maximum containment pressure credit is 6 psi, and the total time for which some overpressure credit is required is 56 hours. For the ATWS scenario, the corresponding values are 2 psi and 1 hour.

The ACRS has historically opposed a general granting of containment overpressure credit. In determining whether such credit should be granted, one aspect to be considered is whether practical alternatives exist, such as the replacement of pumps with those with less restrictive NPSH requirements. If no practical alternatives are available, important considerations include (1) the length of time for which containment pressure credit is required and (2) the margin between the magnitude of the pressure increment that is being granted and the expected minimum containment pressure. Another consideration is the nature of the containment design and whether it provides a positive indication of integrity, prior to the event, as is the case in subatmospheric and inerted designs.

Because of the plant configuration, extent of modifications required, and worker dose that would be involved, we conclude that there are no practical design modifications that would preclude the need to consider the request for containment overpressure credit. VY has an inerted containment. There is, then, a low likelihood of significant pre-existing containment leakage. For the ATWS scenario, the magnitude of pressure required to show adequate NPSH is small compared to the accident pressure, and the time during which the overpressure credit is required is short. For the LOCA scenario, although the duration for which the containment overpressure credit is required is comparatively long, the overpressure credit requested is smaller than what is conservatively predicted to be available.

Under the EPU conditions at VY, the general design requirements regarding single failures in design-basis accidents do not prevent granting of the overpressure credit for the LOCA scenario of concern. The worst single failure that was identified by the licensee involves loss of one train of heat removal from the suppression pool. Conservative, bounding calculations show that the containment overpressures during this scenario are higher than needed to provide sufficient NPSH. Allowing no credit for containment overpressure is equivalent to assuming an additional failure that causes loss of the overpressure. Thus, for all scenarios involving only a single failure, sufficient NPSH is available to ensure that pump cavitation damage is avoided. To maintain defense-in-depth, however, it has been staff practice to require the assumption that containment overpressure is not available in assessing the potential for pump damage.

In evaluating Entergy's request for containment overpressure credit, the staff included in its decisionmaking process more realistic analyses to determine whether containment overpressure would be needed at the proposed EPU power level to prevent pump cavitation in actual accident scenarios. The staff also considered the results of probabilistic analyses to assess the risk significance of scenarios in which containment overpressure is lost.

Design-basis accidents are typically analyzed using conservative methodologies and input assumptions to ensure safety in spite of uncertainties in input and methodology. An alternative approach is to use realistic analyses with a more complete and explicit consideration of uncertainties. Such a methodology has not yet been fully developed for analysis of the need for containment overpressure credit. The staff and the licensee have instead performed sensitivity analyses to determine the effect of relaxing some of the conservative assumptions. More realistic values were used for a number of input parameters to determine the associated reduction in the predicted temperature of the suppression pool, which is the major parameter in determining whether overpressure credit is necessary. The staff concluded that, on a more realistic but still conservative basis, the temperature of the suppression pool would not become high enough in the LOCA scenario to require a credit for containment overpressure.

Independent risk analyses were performed by the staff and the licensee to determine the potential risk significance of granting credit for containment overpressure. These analyses included the conservative assumption that the emergency core cooling system (ECCS) success criteria would not be met whenever containment overpressure is lost and design-basis analyses would suggest that overpressure credit was needed, although the licensee's sensitivity studies indicated that peak suppression pool temperature would probably not be high enough that containment overpressure credit would be required. The results of the analyses indicate that the overall risk associated with the EPU is small and that the change in risk resulting from allowing the requested containment overpressure credit is also small.

Although we concur with the staff's conclusion to grant credit for containment overpressure, we would have preferred to see the assessment performed and presented in a more coherent manner, with a more complete and rigorous consideration of uncertainties. The staff is developing additional guidance to be used in the consideration of overpressure credit in the future. We look forward to reviewing their proposed approach.

The staff performed an expanded engineering inspection of VY. Such an inspection was requested by the Public Service Board of the State of Vermont. The inspection focused on safety-significant components and operator actions. It was performed under the direction of the NRC Office of Nuclear Reactor Regulation (NRR) and included regional inspectors and contractors who had no recent oversight responsibilities for VY. There were eight findings, but they were of low safety significance. A number of members of the public asked for a more extensive inspection, similar to that performed at the Maine Yankee plant. Based on the results of the inspection that was performed and the performance of VY as determined by the Reactor Oversight Process, such an extensive inspection is not warranted.

Hardware and operational changes are required for the power uprate. In order to achieve the proposed EPU power level, all three feedwater pumps must operate, rather than the two pumps currently required. If one of these pumps fails, the plant will undergo an automatic runback of power so that the two remaining pumps will be sufficient. A new signal has been added to trip a feedwater pump in the event of a condensate pump trip. A concern has been raised about the potential for loss of all feed pumps due to low suction pressure as a result of a condensate pump trip. Consequently, Entergy has agreed to perform a trip of a condensate pump to demonstrate that it will not cause loss of all feedwater. This will also test the integrated response of control systems associated with recirculation flow runback, feedwater level control, and reactor pressure control.

Entergy does not plan to undertake large transient tests, such as a main steam isolation valve closure that would result in a reactor trip. Such tests would not directly address confirmation of the performance of systems changed to support EPU. The ACRS concurs with the staff's assessment that the large transient tests are not warranted.

Only minor changes have been made in the emergency operating procedures to accommodate EPU modifications. One of the impacts of the power uprate is a reduction in available response time for operator actions. The operators respond in essentially the same manner as for the current operating conditions but, in some cases, have less time to take an action. A systematic assessment has been made by Entergy of the maximum time available for critical operator actions. The VY simulator has been modified to represent the EPU condition and operators have been trained for EPU conditions. The simulator exercises have demonstrated the ability of the operators to respond correctly within the required time period.

The reactor operating domain is defined so that: (1) the core will not be operated in an unstable regime, (2) the minimum critical power ratio is low enough to prevent dryout of the fuel pins, and (3) the linear heat generation rate is low enough to assure the integrity of fuel cladding during steady and transient conditions. The boundaries of this operating domain are based on neutronic and thermal-hydraulic calculations performed by GE. The computer codes that are used in these analyses have been reviewed and approved by the staff.

In reviewing the application of these methods to EPU uprates, the staff determined that the operation of the fuel extends into a region where the expected void fraction within the fuel bundle is greater than that for which the codes have been validated. To demonstrate the ability of the code to predict isotopic concentrations in this regime, GE has committed to performing gamma scans on the fuel design that is being used in the power uprate. In the interim, Entergy has undertaken an "Alternative Approach" in which it has performed an uncertainty analysis for the model predictions and, as a result, has added an additional margin of 0.02 to the SLMCPR. We concur with the staff's assessment that the addition of such a margin is an appropriate interim measure. The review of the adequacy of the GE computer codes is a generic activity that is being undertaken by the staff. We will have an opportunity to review the staff's assessment of these codes in more detail when we consider the MELLLA+ topical report in 2006.

Higher steam and feedwater flow rates at EPU conditions may lead to an increase in flow accelerated corrosion for some components. The evidence indicates that current flow accelerated corrosion rates at VY are low. Many of the components that would most likely be affected use chromium- molybdenum alloy materials that are resistant to flow accelerated corrosion, and Entergy has committed to an inspection program that will provide reasonable assurance that degradation will be detected prior to reaching an unsafe condition.

Increased flow rates also have the potential to induce vibrations that could lead to failure of components. Because of the previous experience at Quad Cities, the steam dryer has been the primary focus of attention. A number of cracks have been found in inspections of the VY steam dryer. Two cracks found near the lifting lugs were attributed to the initial fabrication of the steam dryer. These cracks have been ground out and repaired. The other cracks that have been found appear to be superficial and were deemed to be the result of intergranular stress corrosion, not flow-induced vibration. Stiffeners have been added to the dryer to provide additional strength and also to raise its natural frequencies.

Entergy has performed hydrodynamic, acoustic and structural resonance analyses to assess the potential for stimulation of a resonant mode of the dryer. These analyses indicate that there is margin between the magnitude of the potential stresses imposed on the steam dryer and the level at which fatigue failure would occur. However, the state of validation of these methods is poor.

To provide further assurance of the integrity of the dryer, additional strain gages have been added to the steam lines at VY. Experiments performed in a scale-model system by GE indicate that acoustic signals initiated in the region of the steam dryer can be correlated with signals measured by strain gages on the steam lines. A similar correlation has been observed at Quad Cities Unit 2 where both the steam dryer and steam lines have been instrumented.

Entergy has developed a program for power ascension involving holds at a number of power levels. The steam line strain gages will be monitored at the various power levels. Any anomalies will lead to a reduction in power until the issue is resolved. Entergy has also committed to inspections of the steam dryers in the next three outages following the uprate. The additional monitoring, the power ascension program, and the inspections provide confidence that, if excessive excitation does occur in the steam dryer, it will be identified before substantial damage is incurred.

Power uprates are not submitted as risk-informed license applications. Nevertheless, licensees have submitted assessments of risk associated with the extended power uprates and the staff includes consideration of this risk information in its decisionmaking process. The purpose of the staff's risk review as stated in RS-001 is to "determine if there are any issues that would potentially rebut the presumption of adequate protection provided by the licensee meeting the deterministic requirements and regulations." The staff has reviewed Entergy's assessment of risk at the proposed EPU conditions and compared the VY probabilistic risk assessment (PRA) results with the staff's SPAR model results for this plant. The values of core damage frequency (CDF) and large early release frequency (LERF) are low and provide substantial margin to values that raise questions of adequate levels of safety. As we noted previously, the staff also used risk insights in their independent determination of the acceptability of the potential for pump cavitation during long-term core cooling in LOCA and ATWS scenarios.

This was the second application by the staff of RS-001 in the review of an EPU proposed upgrade. RS-001 provides a structured approach to the review.

Sincerely,

Graham B. Wallis

Graham B. Wallis
Chairman

Additional Comments by ACRS Members Richard S. Denning, Thomas S. Kress, Victor H. Ransom, and Graham B. Wallis

Considering all the evidence, including precedents set at other similar plants, we agreed with our colleagues to approve the proposed 20% EPU for VY.

It seems unlikely that there will be a problem with adequate NPSH of the core spray and residual heat removal (RHR) pumps at Vermont Yankee, with a 20% power uprate. However, we were asked to make a professional judgment that would have been more straightforward if the information supplied to us had been more complete. We suspect that more information already exists that could be reorganized, supplemented as needed, and presented logically to provide a more convincing case in the following way, which would set a better precedent for future applications:

1. Derive sufficient detail of the probability distribution for containment pressure following large LOCA and ATWS sequences, based on realistic analysis of the physical phenomena and the attendant uncertainties.

2. Derive sufficient detail of the probability distribution for suppression pool temperature following these events, based on realistic analysis of the physical phenomena and the attendant uncertainties.

3. Combine the results of steps 1 and 2 with realistic and uncertainty analyses of other phenomena influencing NPSH to derive the probability of successful operation of RHR and core spray pumps. This may provide adequate evidence for a conclusion to be reached, if it can be shown that only a small containment overpressure is likely to be needed for a short time, if at all, and it has a high probability of being available. If further evidence is required, these results can be incorporated into the PRA to derive the realistic contribution, if any, to total plant risk due to insufficient NPSH.

Both Entergy and the staff have shown that relaxing a few of the many conservatisms and using realistic values (for example, of the initial temperature of the suppression pool) removes the need for additional NPSH. Such arguments are insufficiently conclusive. The reason is that when one gives up an element of conservatism, without replacing it by a less stringent assumption that is still demonstrably conservative, there is a finite probability that values of the derived parameter will not bound all possibilities. The proper way to relax the many conservative assumptions is to make (some of) them realistic with the inclusion of uncertainty. This will lead to a probability distribution (or more precisely some aspects of it, such as the 95/95 confidence level) for an output such as pool temperature.

From the analyses that we have seen in presentations by Entergy and by the staff, it appears likely that the realistic contribution to risk from inadequate RHR and core spray pump NPSH will prove to be very small, even essentially zero, for the case of the proposed power uprate at VY, but this could be better demonstrated in a manner which is both physically and logically consistent. The probabilities associated with the governing physical phenomena may be regarded as more secure than some other inputs to the usual PRA assessment. Conclusions based on them may help to convince those who doubt if conventional risk-based arguments alone should allow the relaxation of defense-in-depth that is achieved by the independence of cladding and containment barriers to radioactivity release. In particular, if it can be shown that the probability of needing containment overpressure is sufficiently small, the independence of these barriers would effectively be preserved.

REFERENCES:

1. Memorandum from Ledyard B. Marsh to John Larkins, "Vermont Yankee Nuclear Power Station - Draft Safety Evaluation for the Proposed Extended Power Uprate (TAC No. MC0761)", October 21, 2005

2. Letter from Wayne Lanning to Jay Thayer, "Vermont Yankee Nuclear Power Station, NRC Inspection Report 05000271/2004008", December 2, 2004

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

NRC FORM 335 (9-2004) NRCMD 3.7

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG-1125, Volume 27

2. TITLE AND SUBTITLE

A Compilation of Reports of the Advisory Committee on Reactor Safeguards: 2005 Annual

3. DATE REPORT PUBLISHED

MONTH	YEAR
June	2006

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

6. TYPE OF REPORT

Compilation

7. PERIOD COVERED *(Inclusive Dates)*

Jan. thru Dec. 2005

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 64 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 2005. 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
Digital Instrumentation and Control
Human Factors
License Renewal
Nuclear Reactors
Nuclear Reactor Safety
PWR Sump Performance
Reactor Operations
Risk-Informed
Safety Engineering
Safety Research

13. AVAILABILITY STATEMENT

Unlimited

14. SECURITY CLASSIFICATION

(This Page) Unclassified

(This Report) Unclassified

15. NUMBER OF PAGES

16. PRICE

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, DC 20555-0001

OFFICIAL BUSINESS

Printed on recycled paper

Federal Recycling Program