NUREG-1125
Volume 24

A Compilation of
Reports of

The Advisory Committee on Reactor Safeguards

2002 Annual

U. S. Nuclear Regulatory Commission

June 2003

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ABSTRACT

This compilation contains 39 ACRS reports submitted to the U. S. Nuclear Regulatory Commission (NRC), or to the NRC Executive Director for Operations, during calendar year 2002. All reports have been made available to the public through the NRC Public Document Room, the U. S. Library of Congress, and the Internet at http://www.nrc.gov/reading-rm/doc-collections. The reports are organized in chronological order.

PREFACE

The enclosed reports, issued during calendar year 2002, 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
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17	Calendar Year 1995
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19	Calendar Year 1997
20	Calendar Year 1998
21	Calendar Year 1999
22	Calendar Year 2000
23	Calendar Year 2001

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

February 12, 2002

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: PROPOSED AMENDMENT TO 10 CFR 50.55a TO INCORPORATE BY REFERENCE ASME BOILER AND PRESSURE VESSEL AND OM CODE CASES

During the 489th meeting of the Advisory Committee on Reactor Safeguards, February 7-8, 2002, the Committee considered the proposed amendment to 10 CFR 50.55a, "Codes and Standards," and decided not to review it. The Committee has no objection to issuing this proposed amendment for public comment. The Committee would like the opportunity to review the draft final rule after reconciliation of public comments.

Reference:
Memorandum dated January 30, 2002, from Jon R. Johnson, Office of Nuclear Reactor Regulation, to John T. Larkins, Executive Director, Advisory Committee on Reactor Safeguards, Subject: Proposed Rule: Incorporation by Reference of ASME BPV and OM Code Cases, 10 CFR 50.55a.

cc: A. Vietti-Cook, SECY
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H. Tovmassian, NRR

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

February 13, 2002

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

SUBJECT: THE REVISED REACTOR OVERSIGHT PROCESS

Dear Dr. Travers:

Your letter of January 10, 2002, provided the staff's responses and planned actions related to the report from the Advisory Committee on Reactor Safeguards (ACRS) dated October 12, 2001. In that report, we provided the results of our review of the revised Reactor Oversight Process (ROP). In general, we concur with the staff's responses to our concerns. However, we continue to believe that some of the threshold values for risk-based performance indicators (PIs) are not meaningful. It is important that the thresholds adequately reflect the levels at which NRC will take action and the urgency with which this action will be taken. Some of the current thresholds do not do this. Also, further discussion is needed regarding the assessment of concurrent findings. Finally, as requested in the SRM dated December 20, 2001, we need to discuss performance deficiencies and apparent conflicts and discrepancies between elements of the ROP which are risk-informed (e.g., significance determination process) and those that are performance-based (e.g., PIs).

We look forward to working with the staff to assist in further development of the ROP.

Sincerely,

George E. Apostolakis
Chairman

References:

1. Letter dated January 10, 2002, from William D. Travers, Executive Director for Operations, NRC, to George E. Apostolakis, Chairman, ACRS, Subject: The Revised Reactor Oversight Process.
2. Letter dated October 12, 2001, from George E. Apostolakis, Chairman, ACRS, to Richard A. Meserve, Chairman, NRC, Subject: The Revised Reactor Oversight Process.

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

February 14, 2002

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

Dear Chairman Meserve:

SUBJECT: REVIEW AND EVALUATION OF THE NUCLEAR REGULATORY COMMISSION'S SAFETY RESEARCH PROGRAM

During the 489th meeting of the Advisory Committee on Reactor Safeguards (ACRS), February 7-8, 2002, and during our retreat meeting on January 24-26, 2002, we discussed the Nuclear Regulatory Commission's Safety Research Program. We met on November 8, 2001, with representatives of the NRC's Office of Nuclear Regulatory Research (RES) to discuss this matter. We also had the benefit of the documents referenced.

In April 2001, the ACRS completed a comprehensive and detailed review and evaluation of the NRC's Safety Research Program, as documented in NUREG-1635, Vol. 4. Favorable comments were made concerning most RES programs. We recommended, however, that some RES programs be brought to closure. We also identified potential future research needs in the following areas:

- New Power Plants and a Revised Regulatory Structure
- Risk Implications of License Renewal and Power Uprates
- Decision-Making Methods

Since we issued our report, RES has made a number of adjustments to its programs to address the Committee's recommendations. In addition, RES has increased its attention to safeguards and security in response to the September 11, 2001, events. Beyond those changes, however, the bulk of the RES program has not changed significantly enough to warrant a comprehensive report. Therefore, the Committee has decided not to issue a detailed report in 2002.

In lieu of such a report, we have reviewed and evaluated the RES responses to the Committee's recommendations. We plan to follow RES and industry programs related to future reactor designs, which will be a major focus of our 2003 research report.

RES Responses to ACRS Recommendations for Program Closures

In NUREG-1635, Vol. 4, we recommended termination of research activities in a number of areas: (1) the control room design review guidance; (2) the Organization for Economic Cooperation and Development (OECD) lower head failure research program; (3) the common-cause failure (CCF) program; and (4) the program, "A Technique for Human Event Analysis" (ATHEANA).

RES has agreed with the first two recommendations and has modified its 2002 research budget accordingly.

With respect to the CCF program, RES agrees that there has been a decreasing trend in the occurrence rate of CCF events. Therefore, RES does not plan any further development of the methodology. RES intends, however, to continue participating in the International Common-Cause Failure Data Exchange Program and CCF data collection from operating experience. We view these actions to be appropriate as they are focused on maintaining and updating significant databases.

Regarding ATHEANA, we noted that important elements (such as a safety-conscious work environment) were missing from the identification of error-forcing contexts, and that ATHEANA did not have a model for the relationship between error-forcing contexts and the probability of unsafe acts. RES plans to continue to implement improvements in ATHEANA throughout Fiscal Year 2002 and to continue to apply ATHEANA to a number of problems such as pressurized thermal shock, steam generator tube rupture, fire, and cable aging. We look forward to reviewing the results. Following these activities, RES plans to sunset the ATHEANA program. RES has also provided us with a research program plan in the area of Human Reliability Analysis. We will review this plan in the near future.

Future Research Initiatives Suggested by the ACRS

In NUREG-1635, Vol. 4, we recommended that research activities be initiated in three areas: (1) to assess the risk implications of license renewal and power uprates; (2) to develop a revised regulatory structure for new power plants; and (3) to explore the use of formal decision-making methods to support regulatory decisions. RES has since initiated a study to evaluate the risk implications of license renewal and power uprates. The other two areas of recommended research are discussed below.

Research Needs to Support Licensing of Future Plants

The agency may soon receive licensing applications that involve reactor designs radically different from those currently in service. RES will play a critical role in preparing the agency to meet the challenges of licensing such new reactor designs. Consequently, RES needs to develop the technical bases that will facilitate effective and efficient licensing reviews of future plants. RES also needs to develop and adapt the analytical tools that would allow independent analysis of plant safety. On June 4-5, 2001, ACRS sponsored a workshop on regulatory challenges for future reactor designs in order to identify associated regulatory and policy issues. A list of regulatory challenges developed by the workshop can be found in NUREG/CP-0175, "Proceedings of the Advisory Committee on Reactor Safeguards Workshop on Future Reactors."

A significant question confronting the agency regarding licensing of new reactor designs is: should the NRC develop a new licensing approach? And, if so, what should be the characteristics of this new approach. As we stated in NUREG-1635, Vol. 4, this question needs to be addressed on an urgent basis because the development of a new design-independent licensing approach will take time. In that document we also stated the desirability of a new approach for risk-informed, design-independent regulatory framework and identified a number of attributes of this framework. To support such an approach, the staff needs to define the full spectrum of regulatory objectives expressed in terms of risk acceptance criteria. New risk metrics, for example frequency-consequence curves, would have to be developed for designs for which core-damage frequency (CDF) and large early release frequency (LERF) may be inappropriate. Such an approach would place expectations on probabilistic risk assessment (PRA) quality and scope for designs that lack the extensive experience base that exists for "standard" light-water reactors.

Applying the current regulatory process to the extent possible for new reactor designs, with only those essential adjustments required to deal with the differences in technology, may represent a viable option. Even in this case, however, a new design-independent, risk-informed regulatory framework could greatly benefit the required adaptation and the development of design-basis accidents. This approach would benefit from significant interaction with reactor vendors and would resemble the original approach to the licensing of the current generation of water reactors, where regulation did not precede but evolved with the development and implementation of reactor technology.

Regardless of the licensing approach that is selected, the agency needs to revisit existing criteria and guidelines that may not be appropriate for the characteristics of the new reactor concepts being proposed. Some of the more important questions needing to be answered are as follows:

- Do we need alternate risk acceptance criteria for the new designs (e.g., frequency-consequence curves)?
- How will multiple units on a site affect the risk acceptance criteria?
- How are uncertainties to be treated in the licensing process (e.g., confidence level, safety margins, defense-in-depth)?
- How will the adequacy of confinement be assessed?
- How will design-basis accidents be identified?
- What will represent acceptable emergency planning requirements?
- How will the scope, quality, and acceptability of PRAs for radically new designs, and codes for thermal-hydraulic, neutronic, and safety assessment, be evaluated?
- What role can "licensing by test" play in the regulatory process?
- Should the manufacturing process of reactor fuels for Pebble Bed Modular Reactor (PBMR) and Gas Turbine-Modular Helium Reactor (GT-MHR) be part of the licensing basis and subject to NRC regulation?

The most pressing issue related to AP1000 certification is what confirmatory research is needed to evaluate the adequacy of the AP600 separate effects and integral test database for application to AP1000.

Some of the new designs may also challenge current defense-in-depth precepts. For example, the traditional balance between prevention and mitigation may not be offered by new designs that rely heavily on fuel integrity during accidents rather than mitigating systems. Uncertainty criteria to allow setting appropriate limits on defense-in-depth requirements may need to be developed.

Finally, the agency needs to determine what independent capabilities and technical databases it must have to assess the safety implications of new technologies; to conduct selected independent verification, analysis, and testing; and to license the new designs. This will require an assessment of necessary fuel and thermal-hydraulic codes, PRA methods, severe accidents and source term codes, etc. Materials under the operating conditions proposed by new designs could also present new challenges that may require significant study. Early interaction with advanced reactor designers is essential for identifying the need for data, models, and analytical tools.

RES is developing a plan to identify the necessary research activities for new reactor designs. The Committee will review this plan.

Use of Formal Decision-Making Methods to Support Regulatory Decisions

In NUREG-1635, Vol. 4, we observed that the decision-making processes used in the regulatory framework process often appear overly subjective and recommended that the staff initiate a research program to investigate how best to use formal decision-making methods to make regulatory decisions more objective and transparent and, thus, more defensible. In our report on the Revised Reactor Oversight Process, dated October 12, 2001, we observed that formal decision analysis could be helpful in making the action matrix and the selection of thresholds for the performance indicators more objective and scrutable. In informal communications to us, RES has recognized the merit of developing formal approaches to support the agency's decision-making processes but has not initiated any work in this area.

Sincerely

George E. Apostolakis
Chairman

References:

1. U.S. Nuclear Regulatory Commission, NUREG-1635, Vol. 4, "Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program," A Report to the USNRC by the Advisory Committee on Reactor Safeguards, April 2001.

2. Memorandum dated July 20, 2001, from William D. Travers, Executive Director for Operations, NRC, to NRC Commissioners, Subject: Response to SRM-M010510B Briefing on Office of Nuclear Regulatory Research (RES) Programs and Performance.
3. U. S. Nuclear Regulatory Commission, NRUEG/CP-0175, "Proceedings of the Advisory Committee on Reactor Safeguards Workshop on Future Reactors," June 4-5, 2001, dated December 2001.
4. Letter dated October 12, 2001, from George E. Apostolakis, ACRS Chairman, to Richard A. Meserve, Chairman, NRC, Subject: The Revised Reactor Oversight Process.

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

February 14, 2002

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

Dear Dr. Travers:

SUBJECT: REEVALUATION OF THE TECHNICAL BASIS FOR THE PRESSURIZED THERMAL SHOCK RULE

During the 489th meeting of the Advisory Committee on Reactor Safeguards, February 7-8, 2002, we reviewed the methodology and initial results of the Pressurized Thermal Shock (PTS) Technical Basis Reevaluation Project. Our Subcommittee on Materials and Metallurgy also reviewed this matter on January 15-16, 2002. During our reviews, we had the benefit of discussions with representatives of the NRC staff. We also had the benefit of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

1. The PTS Reevaluation Project is extensive and appears to be technically sound.

2. The preliminary results of the analysis of the Oconee Unit 1 reactor pressure vessel indicate that when the current PTS screening criterion is reached, the frequency of throughwall cracking of the vessel would be approximately two orders of magnitude below the acceptance criteria for vessel failure given in Regulatory Guide (RG) 1.154. If the ongoing work demonstrates that such results are characteristic of the fleet of pressurized water reactors (PWRs), then the current PTS screening criterion may be overly conservative.

3. When the factors that have large impacts on the failure frequency of the reactor vessel have been identified, they should be scrutinized appropriately.

BACKGROUND

The PTS Rule, 10 CFR 50.61, was established as an adequate protection rule in 1985 in response to a longstanding design-basis issue concerning the integrity of irradiation embrittled PWR pressure vessels during scenarios in which there is a thermal transient in conjunction with the maintenance of system pressure. The rule specifies numerical values of an end-of-life material toughness parameter (RT_{PTS}). Licensees are required to demonstrate that the material toughness (RT_{NDT}) in their pressure vessels is less than the PTS screening criterion, which depends on the orientation of the crack. The analyses that defined the screening criterion included a number of assumptions that may make the criterion overly conservative. The staff is now reevaluating the degree of conservatism in the technical basis for the screening criterion in the Rule and the associated RG 1.154 acceptance criteria.

Elements of the reevaluation include: (1) a probabilistic risk assessment (PRA) to identify the event sequences that could lead to PTS and then estimate their frequencies; (2) thermal-hydraulic calculations of the pressure, temperature, and heat transfer coefficient in the coolant adjacent to the pressure vessel wall following the various event sequences; and (3) probabilistic fracture mechanics (PFM) estimates of the probabilities of initiating, propagating, and arresting a crack in the pressure vessel for the sets of plant operational and thermal-hydraulic conditions identified in the previous elements. The PFM estimates are calculated using the Fracture Analysis of Vessels - Oak Ridge (FAVOR) code, which is based on earlier Oak Ridge National Laboratory codes; these, in turn, had their foundation in fracture experiments on prototypical pressure vessels started in the 1970s. The current version of the FAVOR code (v01.0) incorporates the probabilistic aspects of the inputs, such as, PRA analysis of operational scenarios and thermal hydraulic, material, and stress conditions, with the output being a calculated distribution of the frequency of throughwall cracking of the vessel. The PTS Reevaluation Project involves the application of this integrated analytical process to four PWRs that reflect a range of designs: Oconee Unit 1, Beaver Valley Unit 1, Palisades, and Calvert Cliffs Unit 1.

In this letter, we comment on the technical progress to date. We do not comment on issues such as external events, containment integrity, and source terms, which are pertinent to potential changes to the throughwall cracking frequency criteria given in RG 1.154 or the PTS screening criterion. These topics will be examined in the future.

DISCUSSION

The PTS Reevaluation Project involves integration of tasks involving PRA, thermal-hydraulics, and PFM including an integrated, quantitative treatment of uncertainty. Overall, the analytical logic and the approach to the physical reality of the technical basis appear to be sound.

The staff has committed to provide us with additional information concerning: how the dynamic events associated with a main steamline break will affect the assumed responses of the operators and the plant; the variance narrowing associated with histogram sampling; and the sensitivity of results to changes in reactor operating power and fuel burnup.

An important aspect of this reevaluation is providing explicit credit for mitigative actions by the operators. The Oconee Unit 1 analysis indicates that some of these actions may have a large impact on the vessel failure frequency. The probabilities of operator failure are evaluated by assessing the relevant performance shaping factors and employing expert judgment. Due to the potential significance of these actions, detailed scrutiny of these probability estimates, including sensitivity studies, alternative human reliability analysis models, and independent peer reviews, should be performed.

There appear to be other factors, such as the spatial and size distribution of flaws, that have a significant impact on the results but have a relatively weak empirical basis. Like the modeling of human error probabilities, these factors should also receive appropriate scrutiny. Prior to completing this Project, it is important to document the validation bases of the relevant codes and databases. We look forward to reviewing further progress.

Sincerely,

George E. Apostolakis
Chairman

References:

1. Kirk, M., NRC, and Williams, P., ORNL, "Recommended Method to Account for Uncertainty in the Fracture Toughness Characterization Used to Re-Evaluate the Pressurized Thermal Shock (PTS) Screening Criterion," revised draft dated October 3, 2001 (Draft Predecisional).
2. Williams, P. T. and Dickson, T. L., ORNL, NUREG/CR-xxxx, ORNL/TM-2001-xx, "Fracture Analysis of Vessels - Oak Ridge FAVOR, v01.0, Computer Code: Theory and Implementation of Algorithms, Methods, and Correlations," revised draft dated October 15, 2001 (Draft Predecisional).
3. Dickson, T. L. and Williams, P.T., ORNL, NUREG/CR-xxxx, ORNL/TM-2001-55, "Fracture Analysis of Vessels - Oak Ridge FAVOR, v01.0: Computer Code: User's Guide," revised draft dated October 15, 2001 (Draft Predecisional).
4. SECY-01-0185, "Status Report - Reevaluation of the Technical Basis for the Pressurized Thermal Shock Rule (10 CFR 50.61)," dated October 5, 2001.
5. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.154, "Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors," issued January 1987.

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

March 13, 2002

OFFICE OF
ACRS/ACNW

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: FINAL RULE ON DECOMMISSIONING TRUST PROVISIONS

During the 490th meeting of the Advisory Committee on Reactor Safeguards, March 7-9, 2002, the Committee considered the draft final rule on decommissioning trust provisions. The Committee decided not to review the draft final rule and has no objection to publish it in the Federal Register.

In addition, this matter was discussed with the Chairman of the Advisory Committee on Nuclear Waste, who also decided not to review the subject draft final rule.

Reference:
Draft SECY-paper from William D. Travers, Executive Director for Operations, to the Commissioners, Subject: Final Rule on Decommissioning Trust Provisions, transmitted February 13, 2002.

cc: A. Vietti-Cook, SECY
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

March 14, 2002

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

Dear Chairman Meserve:

SUBJECT: CORE POWER UPRATE FOR ARKANSAS NUCLEAR ONE, UNIT 2

During the 490th meeting of the Advisory Committee on Reactor Safeguards, March 7-9, 2002, we completed our review of the Entergy Operations, Inc. (Entergy) application for a power uprate of 7.5 percent for Arkansas Nuclear One – Unit 2 (ANO-2), and the related NRC staff's Safety Evaluation Report (SER). Our Subcommittee on Thermal-Hydraulic Phenomena also reviewed this matter on February 13, 2002. During our review, we had discussions with representatives of the Applicant and the NRC staff, and we also had the benefit of the documents referenced.

Conclusions and Recommendations

1. The Entergy application for a power level increase from 2815 MWt to 3026 MWt for ANO-2 should be approved.

2. The process used by the staff and the Applicant was comprehensive enough to identify the important issues associated with pressurized water reactor (PWR) power uprates. The process would be greatly improved by the availability of a standard review plan to guide both staff and the Applicant.

3. The process used by the Applicant to perform the Reload Safety Analysis appears to be appropriate. Because this is the first large power uprate for a PWR, the staff should review the Reload Safety Analysis for the transitional core reloads to ensure that the plant will operate in compliance with the regulations.

Discussion

In 1997, the staff performed a comprehensive review of an application for a PWR power uprate involving the Joseph M. Farley nuclear power plant. The Farley plant Licensee used the guidance in WCAP-10263, "A Review Plan for Uprating the Licensed Power of a PWR Power Plant," to prepare its application. This guidance has not been formally reviewed and approved. ANO-2 is a Combustion Engineering reactor, not a Westinghouse reactor like Farley. We believe, however, that there is enough similarity between the ANO-2 plant and the Westinghouse plants to justify the use of WCAP-10263 and the Farley plant SER as templates and guidelines. The Applicant also used General Electric Topical Report NEDC-31897P-A, "Generic Guidelines for BWR Extended Power Uprates," and SECY 97-042, Section 3, "Power Uprate Review Process," to support and substantiate its analyses.

Although we believe that the approach used by Entergy and the staff is sufficiently comprehensive to identify the important PWR power uprate issues, the process would be greatly improved by the availability of a better template such as a standard review plan.

It is difficult to perform a major power uprate in a PWR unless significant modifications are made to the plant. In a PWR, the power is limited by the amount of heat exchange surface. ANO-2 installed larger replacement steam generators that can accommodate the higher thermal power, but, these larger steam generators impose greater accident loads on the containment. The increased energy release during a potential steamline break accident required an increase in the containment building design pressure rating from 54 psig to 59 psig. Instead of modifying the containment building, the Applicant reanalyzed the strength of the containment – considering additional tendons that had not been credited in the original analysis. The containment pressure capability was demonstrated by conducting a pressure test at 68 psig. We conclude that the Applicant's analyses of containment loads and demonstration of the design capability of the containment structure are adequate.

Entergy does not propose to alter the basic thermal-hydraulic design of the reactor core, but will change the neutronic design to provide more core power flattening.

For the uprated power plant, the licensee will use a different code for the analysis of the large-break LOCA. This code has previously been reviewed by the staff. It includes a revised reflood heat transfer coefficient correlation, derived from the FLECHT data, and other code improvements to the Appendix K ECCS evaluation model. The model predicts a peak cladding temperature approximately 150°F less than the previous evaluation model.

Because of the significant changes to the physical plant and to the analytical models used to analyze the plant under accident conditions, the staff should review the transition reload safety analyses for this plant to ensure that the Applicant properly incorporated plant design changes and parameters that describe the characteristics of the transition reload.

The Applicant has scheduled many modifications to the balance of plant to accommodate the increased power output and the additional component duty that will result from an increase in rated power. These involve changes to the Main Unit Turbine/Generator, the Main Unit Condenser, and accessories and associated supporting systems. We did not find significant safety issues associated with the planned modifications.

The uprated power level leads to an increase of reactor head temperature and thereby will increase the susceptibility of the Control Rod Drive Mechanism (CRDM) nozzles to cracking. ANO-2 is a "cold head" plant. There is some bypass flow directed to the reactor head region which lowers the reactor head temperature and reduces susceptibility to cracking of CRDM nozzles. This plant was ranked as an "intermediate plant" using Electric Power Research Institute Materials Reliability Program Reports 44 and 48 and will remain an "intermediate plant." Appropriate management of the issues involved in reactor vessel CRDM weld and nozzle cracking is under active consideration by the staff and the nuclear industry. The resolution of this problem will not be affected by the power uprate.

The ANO-2 reactor vessel has a very large margin to the pressurized thermal shock and upper-shelf energy limits and, thus, the neutron fluence and thermal conditions for the upgraded power level will have little effect.

The ANO-2 application for power uprate was not submitted as a "risk informed" application. However, the Applicant did supply risk information, which the staff examined. The Applicant's evaluation of the increase in Core Damage Frequency (CDF) and Large Early Release Frequency (LERF) indicates that these changes can be classified under the guidelines of Regulatory Guide 1.174 as a "small change" for CDF and as a "very small change" for LERF.

Based on our review of the ANO-2 power uprate application and the associated NRC staff's SER, we believe that the requested power level increase for ANO-2 should be approved.

Additional comments by ACRS Member George E. Apostolakis are provided below.

Sincerely,

George E. Apostolakis
Chairman

Additional Comments by ACRS Member George E. Apostolakis

I appreciate the fact that the power uprate requests are not risk informed. Even though estimates of ΔCDF and ΔLERF are provided, the decision of whether to approve the requested uprate is based primarily on conservative "deterministic" calculations.

An important input to the estimation of ΔCDF and ΔLERF is the change in human error probabilities (HEPs). This change is due to shorter available times for operator action that the power uprate generates.

The licensee and the staff did a commendable job in identifying operator actions that could be affected by the power uprate.

I do object, however, to the HEP quantitative estimates that are provided. I do not believe that there are any credible HEP models that are sufficiently sensitive to the calculated reductions in available time to be able to yield believable HEP estimates. For example, Table 8.1 of the SER lists the following human failure event: "Failure to re-energize 2A1/2A2 from ST2 (SBLOCA or SGTR)." The pre-uprate available time was 42 minutes and the estimated HEP was 0.19. The post-uprate available time was estimated to be 39 minutes and the new HEP was 0.29.

I do not believe these results. I do not think that the model that will discriminate between 42 and 39 minutes has been developed yet. The licensee states that these estimates are produced using several EPRI reports. These reports have not been approved by the NRC and are not widely accepted by the technical community. The staff is careful to state (Section 8.1.4) that "... the licensee's human reliability analysis application is consistent with the identified methodologies...." While this may be a true statement, it really does not say anything about the methodologies themselves.

I do not know whether the staff's conclusion that the HEP values reasonably reflect the reductions in times available for operator action is true. I suspect it is, but I do not have a credible model that will convince me that it is true.

I do not think that the staff should accept results that are produced from methodologies that are neither approved by the NRC, nor widely accepted.

References:

1. Memorandum dated December 19, 2000, from Entergy Operations, Inc., to U.S. NRC, Subject: Arkansas Nuclear One-Unit 2 Application for License Amendment to Increase Authorized Power Level.
2. Memorandum dated January 22, 2002, from Amarjit Singh, ACRS, to ACRS Members, transmitting memorandum dated January 18, 2002, from J. A. Zwolinski, Office of Nuclear Reactor Regulation, NRC, to John T. Larkins, ACRS, transmitting Arkansas Nuclear One, Unit No. 2 - Draft Safety Evaluation for Extended Power Uprate (Predecisional).
3. Letter dated March 1, 2002, from Sherri R. Cotton, Entergy Operations, Inc., to Nuclear Regulatory Commission, Subject: ANO Unit 2, Follow-up Questions Resulting from the ACRS Subcommittee's Review of ANO-2's Proposed Power Uprate, dated March 1, 2002.
4. Memorandum dated February 7, 2002, from Paul Boehnert, ACRS, to ACRS Members, Subject: ACRS Review of ANO Unit 2 Core Power Uprate Request - Additional Background Material.

5. Letter dated February 7, 2002, from Glenn R. Ashley, Entergy Operations, Inc., to USNRC, Subject: ANO Unit 2, Response to Request for Additional Information on Vessel Head Penetration Nozzles Regarding the ANO-2 Power Uprate License Application.
6. Letter dated February 7, 2002, from Glenn R. Ashley, Entergy Operations, Inc., to USNRC, Subject: ANO Unit 2, Comments Regarding the Draft NRC Safety Evaluation for the Proposed ANO-2 Power Uprate.
7. Memorandums from Entergy Operations, Inc., Response to Requests for Additional Information Regarding the ANO-2 Power Uprate License Application, dated December 20 (contains proprietary material), November 16 (contains proprietary material), November 16, November 9, October 31(contains proprietary material), October 30, October 17, October 1, and September 14, 2001.
8. Memorandum dated January 31, 2002, from Entergy Operations, Inc., to Nuclear Regulatory Commission, Subject: Arkansas Nuclear One Unit 2 Response to Follow-up Request for Additional Information Concerning SGTR and MHA Dose Assessment Calculations Supporting ANO-2 Power Uprate.
9. Entergy Operations, Inc., Memorandums, Response to Requests for Additional Information Regarding the ANO-2 Power Uprate License Application, dated May 30, June 20, June 26, June 26, June 28, July 3 (contains proprietary material), July 24, July 24, August 7, August 13, August 21, August 23 (contains proprietary material), August 30, 2001.
10. Letter dated September 29, 2000, from Thomas W. Alexion, Office of Nuclear Reactor Regulation, NRC, to Craig G. Anderson, Entergy Operations, Inc., Subject: ANO Unit 2, Issuance of Amendment Re: Technical Specification Changes and Unreviewed Safety Question Resolution Related to Applicable Limits and Setpoints for Steam Generator Replacement.
11. Letter dated November 13, 2000, from T. Alexion, Office of Nuclear Reactor Regulation, NRC, to Craig G. Anderson, Entergy Operations, Inc., Subject: ANO Unit 2 Issuance of Amendment Re: Technical Specification Changes and Unreviewed Safety Question Resolution Related to Containment Building Design Pressure Increase to 59 PSIG.
12. GE Nuclear Energy Licensing Topical Report, "Generic Guidelines for General Electric Boiling Water Reactor Power Uprate," NEDC-31897P-A, dated May 1992.
13. U. S. Nuclear Regulatory Commission, Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," July 1998.

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

March 14, 2002

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

SUBJECT: CORE POWER UPRATE FOR CLINTON POWER STATION, UNIT 1

Dear Chairman Meserve:

During the 490th meeting of the Advisory Committee on Reactor Safeguards, March 7-9, 2002, we completed our review of the AmerGen Energy Company (AmerGen) license amendment request for an increase in core thermal power for the Clinton Power Station, Unit 1. Our subcommittee on Thermal-Hydraulic Phenomena reviewed this matter during a meeting held on February 13-14, 2002. During our review, we had discussions with representatives of the applicant and the NRC staff. We also had the benefit of the documents referenced.

Conclusions and Recommendations

1. The proposed constant-pressure power uprate of 20% for the Clinton Power Station, Unit 1, should be approved.

2. The staff has been conducting extensive reviews of codes, inputs, and methods for analysis of design-basis accidents at the uprated plant. These reviews make acceptable the exceptions taken by the licensee to the approved power uprate methodologies for such analyses.

3. The AmerGen program to monitor piping expected to suffer from significant flow-assisted corrosion at the uprated flow conditions should be rigorously conducted. The importance of this program should be communicated to NRC staff inspecting the uprated Clinton Power Station.

Discussion

AmerGen, the licensee for the Clinton Power Station, Unit 1, has applied for a 20% power uprate that will take this boiling-water reactor (BWR/6) in a Mark III containment from a licensed power of 2894 MWt to 3473 MWt. The power uprate is to be done in steps of 7 and 13%. Although the power uprate is substantial, the unit will still be operating within the power range of other BWR/6 nuclear steam supply systems. As part of the power uprate, the licensee will incorporate fuel assemblies of a new design into the core.

To a significant extent, the licensee has followed the methodologies defined in the Extended Power Uprate Licensing Topical Reports (ELTR 1 and ELTR 2). These methodologies have been approved by the staff and have been used for the power uprates at the Duane Arnold, Quad Cities, and Dresden plants. This power uprate is, however, a constant-pressure power uprate, and the staff is in the process of reviewing the generic methodology for such an uprate. Consequently, the licensee has taken exceptions to the ELTR1 and ELTR2 methodologies for their specific situations.

The licensee proposes to provide a summary report on design-basis accident analyses as part of its core reload submission, rather than as part of the power uprate application. The staff has not been reviewing reload analyses routinely. For the power uprate at Clinton, the staff is conducting extensive reviews and audits of codes, inputs, and methods used for the accident analysis. These reviews include onsite audits and interviews with analysts. Based on these reviews, the staff has accepted the licensee's proposed deviations from the approved methodologies. We have been quite impressed by the reviews being done by the staff and agree that the exceptions taken by the licensee to the ELTR1 and ELTR2 methodologies are acceptable.

The constant-pressure power uprate produces higher steam and feedwater flows in the plant. The higher flows in the steamlines carrying scavenging steam to the high-pressure feedwater heaters are predicted to increase the flow-assisted corrosion in these lines to as much as 0.070 inches per year. The licensee is persuaded that the predictions of the flow-assisted corrosion rates in these lines with 0.500-inch thick walls are conservative, but acknowledges that the corrosion in these lines will be accelerated by the power uprate.

There has been an unfortunate history within the U.S. nuclear industry of pipe ruptures in nonsafety systems because of flow-assisted corrosion. These ruptures have had safety consequences even when they have occurred in lines that are usually found not to have great risk significance. It is important, then, that the licensee's program for monitoring flow-assisted corrosion in steam and feedwater lines be rigorously conducted. It is also important that the staff reviewing the power uprate application have a good process that communicates the importance of the monitoring program to the staff who inspect the uprated plant.

The licensee proposes not to conduct the large transient tests called for in the current version of the General Electric extended power uprate methodology. The staff has accepted this proposal and feels confident that analysis methods are adequate to predict plant performance. We have not found a value for these tests that are commensurate with costs and risks and, therefore, support the position not to conduct the large-transient tests. The modifications to the plant proposed by the licensee do not involve changes to the "recirculation runback system."

Dr. F. Peter Ford did not participate in the Committee's deliberations regarding this matter.

Sincerely,

George E. Apostolakis
Chairman

References:
1. Memorandum dated January 29, 2002, from John A. Zwolinski, Office of Nuclear Reactor Regulation, NRC, Subject: Draft Safety Evaluation for Clinton Power Station Extended Power Uprate.
2. AmerGen Memorandums dated, June 18, November 30, November 29, December 5, November 21, October 17, September 7, September 28, October 31, December 6, October 23, November 8, October 26, November 20, 2001, January 16, 2002, Response to Requests for Additional Information Supporting License Amendment Requests to Permit Uprated Power Operation, Dresden Nuclear Power Station and Quad Cities Nuclear Power Station.
3. U. S. Nuclear Regulatory Commission Generic Letter 89-08 dated May 2, 1989, "Erosion Corrosion Induced Pipe Wall Thinning."
4. U. S. Nuclear Regulatory Commission Bulletin 87-01, dated July 9, 1987, "Thinning of Pipe Walls in Nuclear Power Plants."
5. U. S. Nuclear Regulatory Commission Augmented Inspection Reports 50-280/86-42 and 50-281/86-42, dated February 10, 1987.
6. GE Nuclear Energy, Topical Report, NEDC-32424P-A, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate" (ELTR-1), February 1999 (Proprietary).
7. GE Nuclear Energy, Topical Report, NEDC-32523P-A, "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate" (ELTR-2), February 2000 (Proprietary).
8. GE Nuclear Energy, Topical Report, NEDC-32523P-A, Supp 1, Volume 1, "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate - Supplement 1, Volume I," February 1999, and Volume II, April 1999 (ELTR-2) (Proprietary).
9. GE Nuclear Energy Topical Report, NEDC-33004P, Revision 1, "Constant Pressure Power Uprate," July 26, 2001 (Proprietary)

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

March 14, 2002

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

Dear Dr. Travers:

SUBJECT: CONFIRMATORY RESEARCH PROGRAM ON HIGH-BURNUP FUEL

In recent months, the Advisory Committee on Reactor Safeguards (ACRS) has reviewed the proposed power uprates for a variety of boiling water reactors (BWRs) and, recently, the pressurized water reactor (PWR) at Arkansas Nuclear One, Unit 2 (ANO 2). In the course of those reviews, we repeatedly asked the Nuclear Regulatory Commission (NRC) staff whether it thought that nuclear fuel has sufficient integrity for duty under uprated power conditions, especially when taken to elevated levels of burnup (up to 62 GWd/t). The staff has argued (e.g., Reference 1) that the fuel does have sufficient integrity basing its confidence on engineering judgment, and noting that a research program had been instituted to confirm that judgment. We believe this judgment requires a firmer technical basis, in fact, some existing data appear to contradict the staff's judgment regarding fuel that has been exposed to burnups in excess of 55 GWd/t (Reference 2).

We now learn that NRC's Office of Nuclear Reactor Regulation (NRR) has withdrawn its support for the confirmatory research on high-burnup fuel (memorandum from S. J. Collins, NRR, to A. C. Thadani, RES, dated January 31, 2002, entitled, "Update of Active NRR Requests for Assistance"). This decision means that NRR is willing to claim fuel used in PWRs is capable of sustaining energy inputs of up to the regulatory limit of 280 cal/g. There is experimental evidence that high-burnup fuel cladding can be ruptured and fuel dispersed with energy inputs much lower than the regulatory limit. Scant evidence is available to show that high-burnup fuel in BWRs can survive energy inputs produced by power oscillations of an anticipated transient without scram (ATWS) event, even if this event is arrested.

We believe that the licensing office's assertion that the confirmatory research on high burnup issues is no longer relevant adversely impacts developing a strong technical basis for these matters and on gaining public confidence. We would appreciate your reviewing this matter and providing us with the rationale behind this decision.

ACRS Member William J. Shack did not participate in the Committee's deliberations on this matter.

Sincerely,

George E. Apostolakis
Chairman

References:

1. Responses from questions at ACRS Subcommittee meeting on February 13, 2002, regarding ANO-2 extended power uprate, NRR Action/Follow-up Items, transmitted March 1, 2002, in e-mail from Thomas Alexion (Internal Use Only).
2. Memorandum dated July 6, 1998, from L. Joseph Callan, Executive Director for Operations, to Commission, Subject: Agency Program Plan for High Burnup Fuel.
3. R.O. Meyer, R.K.McCardell, H.M. Chung, D.J.Diamond, and H.H. Scott, "A Regulatory Assessment of Test Data for Reactivity-Initiated Accidents," Nuclear Safety, Volume 37, Number 4, 1996, pages 271-288 and references therein.

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

March 14, 2002

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

Dear Chairman Meserve:

SUBJECT: PHASE 2 PRE-APPLICATION REVIEW FOR AP1000 PASSIVE PLANT DESIGN

During the 490[th] meeting of the Advisory Committee on Reactor Safeguards (ACRS), March 7-9, 2002, we completed our evaluation of the Phase 2 pre-application review of the Westinghouse AP1000 passive plant design, conducted by the NRC staff. This matter was also reviewed during joint meetings of our Subcommittees on Thermal-Hydraulic Phenomena and Future Plant Designs on February 13-15, 2002, and a meeting of the Subcommittee on Thermal-Hydraulic Phenomena on March 15, 2001. During our review, we had discussions with representatives of the Westinghouse Electric Company and the NRC Staff. We also had the benefit of the documents referenced.

Conclusions and Recommendations

1. The staff has made a competent and thorough review of the Phase 2 issues.

2. We agree that the proposal by Westinghouse to use Design Acceptance Criteria (DAC) for the piping design should be approved.

3. The staff's positions on the other pre-application review issues should also be approved.

4. The Office of Nuclear Regulatory Research (RES) should further investigate acceptable ranges of ratios of Pi-groups for use in scaling.

5. The ad hoc introduction of compensating processes to tune codes to the integral test data should be discouraged.

Discussion

The NRC staff and Westinghouse have agreed to a three-phased approach to the AP1000 standard plant design review. Phase 1, which was to identify the key review issues, was completed previously and resulted in the identification of four key issues:

1. Acceptability of the proposed use of DAC for particular parts of the design review.

2. Acceptability of certain exemptions that Westinghouse intends to request.

3. Applicability of the AP600 test program to the AP1000 design.

4. Applicability of the AP600 analyses codes to the AP1000 design.

The purpose of the Phase 2 review was for the staff to develop positions on these four key issues. These positions are discussed below.

Proposed Use of DAC

The Commission has determined that the level of detail in a design certification application must be sufficient to enable the Commission to judge the applicant's proposed means of ensuring that construction conforms to the design and to reach a final conclusion on all safety questions associated with the design.

The staff has interpreted this policy to mean that the certification application must be complete, with two exceptions:

- items for which the technology is rapidly changing and may be significantly different at the combined operating license (COL) stage.

- items for which the level of detail cannot be provided at the time of certification review (or for which the as-procured and as-built characteristics are needed).

For these exceptions, DAC are required of the applicant. Some precedents for DAC satisfying these criteria were established with the certifications of the Advanced Boiling Water Reactor (ABWR) and System 80+ designs. For these, the staff accepted DAC for the instrumentation and control (I&C) and for the control room design, both of which were deemed to satisfy one or more of the above criteria.

In addition to these two areas for which precedents have been established, Westinghouse has proposed DAC for the AP1000 piping design.

The staff has concluded that the DAC approach should be approved for I&C and control room portions of the design based on the two criteria above and that the DAC on piping design should be approved based on the similarity of AP1000 to AP600 designs, for which the certification included sufficient piping design detail.

While we have some sympathy with this view by the staff and agree that the piping DAC should be approved, we believe the piping DAC could have been approved without invoking the similarity to the AP600 design. Our view is that, as long as sufficient detail is available to permit resolution of safety questions, the degree of detail that an applicant wishes to provide at the certification phase is a business decision. We believe the use of DAC for the piping design fits this characterization.

Exemptions

Westinghouse is requesting exemptions from the regulations in three areas:

(a.) Section 50.34 (f)(2)(iv) requires a "safety parameter display console that will display to operators a minimum set of parameters defining the safety status ... displaying a full range of important plant parameters ..., and capable of indicating when process limits are being approached or exceeded."

(b.) Section 50.62(c)(1) requires that equipment be available to ensure the automatic startup of the auxiliary feedwater system under ATWS conditions.

(c.) GDC 17 of 10CFR50 Appendix A requires two physically independent offsite power sources.

The staff agrees with the Westinghouse positions that: Item (a) will be part of the DAC for control room design; the underlying purpose of Item (b) is satisfied because AP1000 does not have (or need) an auxiliary feedwater system as the emergency core cooling system (ECCS) requirement is met by the passive residual heat removal (PRHR) system automatic initiation under ATWS; and that the underlying purpose of Item (c) is satisfied because, with the passive ECCS, AP1000 does not need offsite power to make its safety case. We also agree with these positions.

Applicability of AP600 Standard Plant Design Analysis Codes and Test Program

To address the applicability of the AP600 codes and test program, Westinghouse prepared a new AP1000 phenomena identification and ranking table (PIRT) and conducted new scaling assessments for both the codes and the tests. The AP1000 PIRT resulted in the same high- and medium-ranked phenomena as were found for the AP600, and it was noted that the AP1000 design did not entail any important new phenomena. In addition, the scaling analyses indicated that the Pi-groups identified as being important and which were to be substantially matched in the integral test program were still in the acceptable range when compared to their values for the full-scale AP1000 design. Thus, Westinghouse maintains that these results demonstrate that the AP600 test database used to validate the analysis codes is applicable to AP1000 and that the codes should be approved for use in evaluating the safety status of AP1000 design.

The staff conducted independent top-down and bottom-up scaling assessments and made audit calculations using RELAP5 for a postulated 2-inch diameter break in the cold leg and for a postulated double-ended direct vessel injection (DVI) line rupture. The staff found that, with some noted exceptions, the experimental data produced by the AP600 separate effects and integral effects test programs are appropriate for verification of the processes expected in an AP1000 plant, and the analysis codes validated for the AP600 standard plant design are applicable to the AP1000 design.

The most significant of the exceptions is that the tests are not considered sufficient to validate the entrainment model used in the NOTRUMP code for the upper plenum regions and for the hot-leg exit through the automatic depressurization system (ADS-4) depressurization valve.

Westinghouse claims that the scaling test data and analyses are sufficient to ensure that the core remains covered and that the entrainment is a self-limiting process that decreases as the core water level decreases. Westinghouse also claims that the period during which the entrainment is important in affecting the water level is so short that entrainment is not safety significant. We think such a case can be made during the certification review and, if so, additional tests would not be necessary.

Nonetheless, the staff's position has merit in that it will be necessary to better predict the entrainment behavior before judgments can be made regarding its safety significance. We believe phenomena that are ranked high or medium in importance should be properly treated in the models partly because unanticipated applications could invalidate the "non-safety-important" judgment. We remain concerned that the codes do not properly model entrainment because inapplicable maps are being used to characterize the flow regimes. The use of inapplicable maps could impact the results of the codes in unanticipated ways. Thus, we are convinced that the technical basis codes need better modeling with respect to entrainment and flow regime maps.

Other Considerations

In the scaling assessments, Westinghouse and the staff used the criterion that Pi-group ratios having values between 0.5 and 2.0 represent acceptable scaling. While this range is intuitively pleasing as an indication that the tests sufficiently match the phenomena in AP1000, we have not seen any technical justification for this criterion. Thus, we believe that RES should initiate a study with the objective of establishing a technically based approach for use in determining the significance of any general Pi-group. We think this would involve sensitivity analyses on the Pi-group in the non-dimensional scaling models. The sensitivity of the results to individual Pi-group ratios could guide the selection of acceptance ranges that might be different for different Pi-groups. Although we do not believe that this work is needed for AP1000 certification, this issue is likely to arise with certification of future reactor designs and such a study could tie down this loose end of the code, scaling, applicability, and uncertainty (CSAU) process.

There are two instances in which Westinghouse proposes to adjust its models to provide a better fit to integral data by introducing compensating processes. In one instance, the NOTRUMP code does not model the momentum flux terms in the conservation of momentum equations dealing with effects of area and density changes. This deficiency in the code impacts its ability to calculate pressurizer drainage and reactor vessel downcomer level. To compensate for this code deficiency in the AP600 certification, Westinghouse imposed a reduction in the in-containment refueling water storage tank (IRWST) level – thus reducing the driving force which would conservatively compensate for the effects that would have resulted from having the correct momentum equations. For the AP1000, instead of this same "fix," Westinghouse proposes to use an increased flow resistance penalty that would make the code calculations fit the APEX facility data for a 2-inch small-break loss-of-coolant accident (SBLOCA).

In another instance, Westinghouse concluded that the NOTRUMP PRHR model does not model the thermal plume in the IRWST. The model will over predict the outside surface heat transfer rate for the heat exchanger when the tube flow velocity exceeds 1.5 ft/sec for any significant period of time. If this situation arises in the analyses, Westinghouse proposes to account for the non-conservative calculation by an ad hoc reduction of the predicted heat exchanger performance.

These temporary fixes should provide conservative results to support the certification of AP1000 design. Nevertheless, we view both of these as instances of purposeful introduction of compensating errors in the codes rather than improving the models. We consider it bad practice to allow these errors to persist in the codes and believe that the actual physics should be properly represented in the long term.

Sincerely,

George E. Apostolakis
Chairman

References:

1. Memorandum dated February 4, 2002, transmitting draft SECY Paper, undated, Subject: Use of Design Acceptance Criteria and Exemptions for the AP1000 Standard Plant Design (Predecisional), and draft SECY Paper, undated, Subject: Applicability of AP600 Standard Plant Design Analysis Codes and Test Program to the AP1000 Standard Plant Design (Predecisional).
2. Memorandum dated June 21, 2000, from John T. Larkins, ACRS, to William D. Travers, Executive Director for Operations, NRC, Subject: AP1000 Pre-Application Review.

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

March 19, 2002

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

Dear Dr. Travers:

SUBJECT: PROPOSED RULEMAKING AND ASSOCIATED GUIDANCE FOR RISK-INFORMING THE SPECIAL TREATMENT REQUIREMENTS OF 10 CFR PART 50 (OPTION 2)

During the 490th meeting of the Advisory Committee on Reactor Safeguards, March 7-9, 2002, we met with representatives of the NRC staff and the Nuclear Energy Institute (NEI) to discuss the proposed rulemaking and associated guidance for risk-informing the special treatment requirements of 10 CFR Part 50 (Option 2). We discussed the staff's draft rule language for 10 CFR 50.69 and proposed industry guidance in NEI 00-04, Revision B, "Option 2 Implementation Guideline." Our Subcommittee on Reliability and Probabilistic Risk Assessment discussed these matters during meetings on December 4, 2001, and February 22, 2002. We also had the benefit of the documents referenced. This report focuses primarily on the proposed industry guidance in NEI 00-04, Revision B.

Conclusion and Recommendations

1. The criteria used by the Integrated Decision-making Panel (IDP) for categorizing structures, systems, and components (SSCs) should be made explicit and should include consideration of risk metrics that supplement core damage frequency (CDF) and large early release frequency (LERF), such as late containment failure and inadvertent release of radioactive material.

2. Categorization of SSCs performed with a more complete set of risk metrics may allow the elimination of additional treatment requirements for components in the risk-informed safety class 3 (RISC-3) category (safety related, low safety significant).

3. The rigor in the treatment of uncertainties in probabilistic risk assessment (PRA) results should be made consistent with the current capabilities of PRA software and data. When simplified methods are used, comparison with more rigorous analyses should be available to demonstrate the adequacy of these methods.

Discussion

The overall categorization process described in NEI 00-04, Revision B, relies heavily on the judgments of the IDP. The Panel's decision concerning the assignment of an SSC to a risk-informed safety class is based on a variety of qualitative and quantitative inputs. The quantitative inputs are produced by a PRA, if available. A large majority of SSCs are categorized without the benefit of quantitative inputs from a PRA. Two major elements of the categorization process are the risk-informed decision criteria and the processes used by the IDP in making judgments.

In our report dated October 12, 1999, we commented extensively on the decision-making process and the need for guidance and training in conducting expert-panel sessions. Our comments on the processes described in the then-proposed Appendix T to 10 CFR Part 50 remain valid and are a continuing concern. This report focuses on additional issues that warrant attention in the revision of NEI 00-04 to support the proposed 10 CFR 50.69 rulemaking.

The traditional criteria for evaluating risk significance use the metrics CDF and LERF. The initial screening of SSCs for which PRA results are available is carried out by using importance measures that are based on these two metrics. We believe that the probability of late containment failure should be added to CDF and LERF to provide a more complete characterization of risk.

In categorizing SSCs for which PRA results are unavailable, qualitative considerations serve as the primary basis for decisionmaking. Even when PRA results are available, the risk-informed approach requires that the IDP consider qualitative inputs based on defense in depth and safety margins, as articulated by the principles in Regulatory Guide 1.174. NEI 00-04, Revision B, provides very little guidance to assist the Panel in making these qualitative assessments. Explicit criteria should be developed for the qualitative categorization of SSCs and the decision-making process needs to be scrutable with results that can be documented. Guidance to accomplish this should be included in NEI 00-04.

The qualitative considerations used by the IDP should include defense in depth and the traditional graded approach in which relatively frequent events are intended to not fail any of the barriers to the release of radioactivity, but relatively infrequent events are allowed some fuel damage provided that the resulting release is limited by the requirements of 10 CFR Part 100. Specific guidance to the IDP could include requirements for the Panel to determine whether (1) the SSC supports a system that acts as a barrier to fission product release during severe accidents; (2) the SSC is relied upon in the emergency operating procedures or the severe accident management guidelines; and (3) failure of the SSC will result in the inadvertent release of radioactive material even in the absence of severe accident conditions.

If any of the above conditions are true, the IDP should consider including such SSCs in RISC-1 (safety related, safety significant) or RISC-2 (non-safety related, safety significant) category. The IDP could justify its conclusions in the risk categorization by demonstrating that one of the following conditions are met:

- Relaxing the requirements will have minimal impact on the failure rate increase.
- Showing that adequate data are available to demonstrate that failure modes that prevent the SSC from fulfilling its function are unlikely to occur.
- Such failure modes can be detected in a timely manner.

The choice of appropriate treatment for RISC-3 has been a difficult issue for staff and industry. We believe that much of this difficulty has arisen because the staff recognizes that risk concerns cannot be completely addressed by CDF and LERF and is, therefore, reluctant to relax some special treatment requirements. By explicitly addressing all risk concerns in the categorization process, as discussed above, it may be easier to obtain agreement that components assigned to RISC-3 do not require any treatment beyond "commercial practice."

We note that materials degradation is not directly assessed in NEI 00-04, Revision B. We believe that aging phenomena and the management of degradation must be considered in the IDP deliberations concerning affected SSCs and passive system components.

The use of risk information in regulatory decisionmaking is relatively new. Some within the NRC, the industry, and the public view this evolution with skepticism. The NRC Strategic Plan has established increasing public confidence as a performance goal. The use of rigorous methods to produce risk information is essential to achieving this goal.[1] In many instances, simplified methods can yield satisfactory results. It should be demonstrated, however, that these simplified methods yield results that are consistent with those provided by more rigorous methods and that their limitations are well understood.

In our reports dated October 12, 1999 and February 11, 2000, we commented extensively on the limitations of importance measures. The requirement to use sensitivity studies to determine ΔCDF and ΔLERF provides evidence that NEI 00-04, Revision B, recognizes the major limitation of importance measures, namely, their inability to determine the change in risk associated with a group of components. We

[1]In his speech to the Regulatory Information Conference on March 5, 2002, Commissioner Diaz stated: "This is the year 2002, almost 30 years after WASH-1400, and it is time that all licensees have a quality Level 2 PRA so they can effectively utilize our regulatory processes."

believe that the IDP would benefit from an explicit identification and discussion of this and other limitations that have been identified in the literature (References 8 and 9).

NEI 00-04, Revision B, shies away from providing guidance or encouragement for licensees to perform uncertainty analyses and relies heavily on sensitivity studies that are substitutes for uncertainty analyses. Modern PRA tools make it relatively routine to perform a genuine uncertainty analysis, i.e., one that propagates the uncertainties in failure rates, and such analysis should be performed where possible.

The argument has been made that using mean values for the failure rates in performing the PRA and the screening is "good enough." We agree that, in the majority of cases, this argument may be true provided that mean values are indeed used, although relatively few investigations are available in the literature (References 8 and 11) to substantiate this claim. We object to the practice of taking arbitrary "point" values of the parameters and declaring them as mean values. Such practices do not contribute to the credibility of the categorization process.

One of the most significant limitations of importance measures is that they measure the impact of individual SSCs on risk, and, consequently, they cannot be used directly to estimate changes in risk for a group of SSCs. This limitation is recognized in NEI 00-04, Revision B, and additional sensitivity studies are suggested to attempt to assess the impact of changing treatment requirements on a group of components. In NEI 00-04, Revision B, it is suggested that the failure rates of RISC-3 SSCs be increased by factors ranging from 2 to 5 to evaluate changes in CDF and LERF. The current justification for this choice of values is weak, and a better justification is needed, especially since these factors are smaller than the factor of 10 used in the South Texas Project multiple exemption request. A distinction between parameter and model uncertainties would be very useful in this case.

We look forward to reviewing the draft final rule language and associated guidance as more progress is made.

Sincerely,

George E. Apostolakis
Chairman

References:

1. U.S. Nuclear Regulatory Commission, Draft Rule Language to amend Title 10 of the *Code of Federal Regulations* (10 CFR) by adding Section 50.69, "Risk-Informed Treatment of Structures, Systems, and Components," dated November 19, 2001.

2. Nuclear Energy Institute, NEI 00-04, Draft Revision B, "Option 2 Implementation Guideline," May 2001.
3. Memorandum dated January 24, 2002, from Michael T. Markley, ACRS staff, to Cynthia Carpenter, Office of Nuclear Reactor Regulation, NRC, Subject: Questions on NEI 00-04, "Option 2 Implementation Guideline."
4. Letter dated February 8, 2002, from Cynthia A. Carpenter, Office of Nuclear Reactor Regulation, NRC, to Anthony R. Pietrangelo, NEI, Subject: NRC Staff Review of Draft Revision B of NEI 00-04, "Option 2 Implementation Guideline."
5. Report dated October 12, 1999, from Dana A. Powers, Chairman, ACRS, to Greta Joy Dicus, Chairman, NRC, Subject: Proposed Plans for Developing Risk-Informed Revisions to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."
6. Report dated February 11, 2000, from Dana A. Powers, Chairman, ACRS, to Richard A. Meserve, Chairman, NRC, Subject: Importance Measures Derived from Probabilistic Risk Assessments.
7. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," July 1998.
8. M.C. Cheok, G.W. Parry, and R.R. Sherry, "Use of Importance Measures in Risk-Informed Regulatory Applications," *Reliability Engineering and System Safety*, 60, 213-226, 1998.
9. W.E. Vesely, "Reservations on 'ASME Risk-Based Inservice Inspection and Testing: An Outlook to the Future,'" *Risk Analysis*, 18, 423-425, 1998.
10. U.S. Nuclear Regulatory Commission, NUREG-1150, Vols. 1-3, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," December 1990
11. M. Modarres and M. Agarwal, "Consideration of Probabilistic Uncertainty in Risk-Based Importance Ranking," Proceedings of the International Topical Meeting on Probabilistic Safety Assessment, PSA '96, *Moving Toward Risk-Based Regulation*, Park City, Utah, September 29-October 3, 1996, 230-236, American Nuclear Society.
12. N.J. Diaz, "When...Large is Small and Small is Large," Remarks at the U.S. Nuclear Regulatory Commission, 2002 Regulatory Information Conference, March 5-7, 2002.

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

April 16, 2002

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: DRAFT REGULATORY GUIDE DG-1118 (PROPOSED REVISION 1 TO REGULATORY GUIDE 1.53), "APPLICATION OF THE SINGLE-FAILURE CRITERION TO SAFETY SYSTEMS"

During the 491st meeting of the Advisory Committee on Reactor Safeguards, April 11-12, 2002, the Committee considered the draft Regulatory Guide DG-1118. The Committee plans to review the draft final version of this guide after reconciliation of public comments. The Committee has no objection to issuing this guide for public comment.

Reference:
Memorandum dated January 23, 2002, from Michael A. Mayfield, Office of Nuclear Regulatory Research, to John T. Larkins, Advisory Committee on Reactor Safeguards, and Joseph A. Murphy, Chairman, Committee to Review Generic Requirements, Subject: Draft Regulatory Guide DG-1118 (Proposed Revision 1 to Regulatory Guide 1.53), "Application of the Single-Failure Criterion to Safety Systems."

cc: A. Vietti-Cook, SECY
J. Craig, OEDO
I. Schoenfeld, OEDO
A. Thadani, RES
M. E. Mayfield, RES
S. Aggarwal, RES
E. Hackett, RES
N. Chokshi, RES
J. Strosnider, NRR
J. Moore, OGC

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

April 17, 2002

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

Dear Chairman Meserve:

SUBJECT: GE NUCLEAR ENERGY LICENSING TOPICAL REPORT, NEDC-33004P, "CONSTANT PRESSURE POWER UPRATE" (REVISION 1)

During the 491st meeting of the Advisory Committee on Reactor Safeguards, April 11–12, 2002, we completed our review of General Electric's application for approval of GE Nuclear Energy Licensing Topical Report, NEDC-33004P, "Constant Pressure Power Uprate," Revision 1, and the related draft safety evaluation performed by the NRC staff. Our Subcommittee on Thermal-Hydraulic Phenomena reviewed this matter during meetings held on January 16-18 and March 6, 2002. During our review, we had discussions with representatives of GE Nuclear Energy and with the NRC staff. We also had the benefit of the documents referenced.

Recommendation

The constant-pressure power uprate methodology should be approved for application to BWR power increases of up to 20 percent of the original licensed thermal power.

Discussion

The GE constant-pressure power uprate methodology represents an innovative approach to BWR power uprates. The methodology in this licensing topical report simplifies the analytical work that the licensee must do to justify a power uprate and minimizes changes to the reactor plant. Most physical changes for power uprates are installed in the balance of plant to accommodate the increased steam and feedwater flows that will occur from the increased power rating. We agree with the staff's determination that the constant-pressure power uprate methodology should be approved for BWR power increases of up to 20 percent.

Although the plant reload analyses used for the uprates are based on methodology that has been reviewed and approved by the staff, we support the staff's continuing effort to audit them. We also encourage staff audits of the application of reload analysis methods to transitional reloads for plants undergoing substantial power uprates.

ACRS Members F. Peter Ford and Victor H. Ransom did not participate in the Committee's deliberations on this matter.

Sincerely,

George E. Apostolakis
Chairman

References:

1. Memorandum dated July 26, 2001, to U. S. Nuclear Regulatory Commission, from J. F. Klapproth, GE Nuclear Energy, transmitting GE Proprietary Licensing Topical Report, "Constant Pressure Power Uprate," Revision 1 (Proprietary).
2. Memorandum dated February 27, 2002, to John T. Larkins, ACRS, from John A. Zwolinski, Office of Nuclear Reactor Regulation, NRC, transmitting Revised Draft Safety Evaluation for GE Constant Pressure Power Uprate Licensing Topical Report (Predecisional).
3. GE Nuclear Energy, Topical Report, NEDC-32424P-A, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate," February 1999 (Proprietary).
4. GE Nuclear Energy, Topical Report, NEDC-32523P-A, "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate," February 2000 (Proprietary).
5. GE Nuclear Energy, Topical Report, NEDC-32523P-A, Supp 1, Volume 1, "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate - Supplement 1, Volume I," February 1999, and Volume II, April 1999 (Proprietary).

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

April 17, 2002

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

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

During the 491st meeting of the Advisory Committee on Reactor Safeguards, April 11-12, 2002, the Committee considered the staff's initiative to revise NUREG-BR-0058, "Regulatory Guidelines of the U.S. Nuclear Regulatory Commission," to clarify the treatment of individual requirements in regulatory analysis. The Committee has decided not to hold a briefing on the preliminary proposed criteria at this time. The Committee plans to review the incorporation of the proposed criteria into NUREG-BR-0058 prior to being issued for public comment.

Reference:
Memorandum dated April 4, 2002, from Cynthia Carpenter, Office of Nuclear Reactor Regulation, NRC, to John T. Larkins, Advisory Committee on Reactor Safeguards, Subject: Criteria for the Treatment of Individual Requirements in a Regulatory Analysis.

cc: A. Vietti-Cook, SECY
J. Craig, OEDO
I. Schoenfeld, OEDO
S. Collins, NRR
C. Grimes, NRR
D. Allison, NRR
C. Prichard, NMSS
A. Thadani, RES

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

April 19, 2002

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

Dear Chairman Meserve:

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE LICENSE RENEWAL APPLICATION FOR THE TURKEY POINT NUCLEAR PLANT, UNITS 3 AND 4

During the 491st meeting of the Advisory Committee on Reactor Safeguards, April 11-12, 2002, we completed our review of Florida Power and Light Company's (FPL's) license renewal application for the Turkey Point Nuclear Plant, Units 3 and 4, and the NRC staff's final safety evaluation report (SER) on the application. Our review included a plant visit and two meetings of our Plant License Renewal Subcommittee, one of which was conducted on March 13, 2002, in Florida City, Florida. During our review, we had the benefit of discussions with representatives of the NRC staff and FPL. In addition, we discussed written comments on Turkey Point from a member of the public. Our subcommittee also heard oral statements from a member of the public during the meeting in Florida City. We had the benefit of the documents referenced.

Recommendation and Conclusion

1. The FPL application for renewal of the operating licenses for Turkey Point, Units 3 and 4, should be approved.

2. The programs instituted to manage aging-related degradation are appropriate and provide reasonable assurance that Turkey Point, Units 3 and 4, can be operated in accordance with their licensing bases for the period of extended operation without undue risk to the health and safety of the public.

Background and Discussion

This report fulfills the requirement of 10 CFR 54.25 that the ACRS review and report on license renewal applications. FPL requested renewal of the operating licenses for Turkey Point, Units 3 and 4, for a period of 20 years beyond the current license terms, which expire on July 19, 2012 (Unit 3), and April 10, 2013 (Unit 4). The final SER documents the results of the staff's review of information submitted by FPL, including commitments that were necessary to resolve open items identified by the staff in the draft SER. The staff reviewed the completeness of the identification of structures, systems, and components (SSCs) subject to aging management review; the integrated plant assessment process; the applicant's identification of the possible aging mechanisms associated with passive, long-lived components; and the adequacy of the aging management programs. The staff also conducted four site inspections to verify the adequacy of the implementation of the methodology described in the application.

We met with the applicant and the staff on September 25 and October 5, 2001, to review the draft SER. We did not identify any new issues to be addressed by the staff and applicant other than the four open items already identified by the staff. The number of open items was small because the applicant implemented lessons learned from the previous license renewal applications and followed the guidance in Nuclear Energy Institute (NEI) Report 95-10, "Industry Guidelines for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule." This approach facilitated the review process.

The process implemented by the applicant to identify SSCs that are within the scope of license renewal has been effective. During our review we questioned why certain SSCs were not included in scope, and in all cases the applicant provided appropriate justification for the exclusion. Among these SSCs were the startup transformers that connect the plant to the offsite power source, which typically provides the alternate AC power source during a station blackout (SBO) event. The applicant argued that Turkey Point does not rely on restoration of offsite power to recover from an SBO event. Instead, it relies on the installed capability to cross-connect the emergency diesel generators (EDGs) from one unit to the other. During an SBO event, each of the four EDGs on site is capable of carrying all essential loads of both units. Sufficient diesel fuel is maintained on site to provide the required long-term alternate power source. During our visit to the site, the applicant used the plant simulator to demonstrate its ability to cross-connect the EDGs from the control room. This capability was used during Hurricane Andrew. On this basis, we concur with the applicant that the EDGs provide an effective alternate power source during an SBO event. Subsequently, the staff has determined, however, that components connecting the units to the offsite power source, including the startup transformers, are needed to fulfil the requirements of the SBO Rule. Therefore, they are part of the licensing basis and must be included in the scope of license renewal. The applicant has agreed to meet this requirement.

The applicant has performed a comprehensive aging management review of SSCs that are within the scope of license renewal. The applicant identified aging effects using many data sources, including previously submitted license renewal applications, Babcock & Wilcox license renewal generic information, industry operating experience, Turkey Point operating experience, the draft Generic Aging Lessons Learned report, and Westinghouse Owners Group (WOG) topical reports. As the first Westinghouse-designed reactor being considered for license renewal, Turkey Point participated in a WOG program that developed a series of generic topical reports to demonstrate that the aging effects of reactor coolant system components could be adequately managed throughout the period of extended operation. The WOG submitted four topical reports for NRC staff review and approval. The topical reports contain generic license renewal evaluations of pressurizers (WCAP-14574), Class 1 piping and associated pressure boundary components (WCAP-14575), reactor internals (WCAP-14577), and reactor coolant system supports (WCAP-14422).

The applicant did not incorporate these reports by reference in the Turkey Point license renewal application because the staff had not approved these reports at the time the application was submitted to the NRC. These reports were subsequently approved by the staff. In its application, the applicant addresses the applicability of these reports to Turkey Point SSCs to facilitate the staff review. We have reviewed these topical reports and found that, when supplemented by the Turkey Point plant-specific responses to the staff's open issues on the topical reports, they effectively support the Turkey Point license renewal application.

Appendix B of the application describes the 16 existing programs and the 7 new programs that FPL has implemented to manage aging effects during the period of extended operation. The resolution of staff questions and SER open items has resulted in additional commitments, including a program to deal with the adverse localized effects of heat on medium and low-voltage nonenvironmentally qualified (EQ) cables, connections, and electrical/instrumentation and control penetrations in containment, as well as an expanded number of piping segments to be managed to address the potential interaction of Class II piping with safety systems.

Unlike previous applicants, FPL has not proposed an aging management program for non-EQ medium-voltage cables that are exposed to significant moisture. The applicant stated that these cables are designed with lead sheath to prevent failure from moisture ingress. The applicant presented information, including significant industry operating experience, that indicates that this type of jacket provides an impermeable barrier. Based on this information, we agree with the applicant and the staff that no aging management program is needed for non-EQ medium-voltage cables that are subjected to significant moisture.

The Turkey Point application identifies cracking of the control rod drive mechanism (CRDM) penetration nozzles as an aging effect to be managed. Appendix B of the application describes the aging management program, "Reactor Vessel Head Alloy 600 Penetration Inspection Program (RVHPIP)," instituted to deal with this aging degradation mechanism. This program identifies primary water stress corrosion cracking (PWSCC) of Alloy 600 nozzles as the aging effect of concern and ties programmatic elements, such as the frequency of inspections, to the results of plant-specific and sister plant inspection findings. In response to an SER open item, the applicant has committed to continue its participation in the Electric Power Research Institute (EPRI) and NEI programs for managing PWSCC in Alloy 600 reactor vessel head penetration nozzles during the period of extended operation, and has made the NEI program and EPRI Materials Reliability Program (MRP) an integral part of the RVHPIP. This ensures that, as the industry gains more experience with this degradation mechanism, the applicant will update the RVHPIP to reflect the new information. Over the past 6 months, the applicant has performed inspections of upper heads of both units. No leakage of the CRDM penetration nozzles was identified.

A member of the public provided us with written comments expressing his concerns with the continued operation of Turkey Point. His concerns included potential voids in containment walls, the ability of Turkey Point to withstand Category 5 hurricanes, and the vulnerability of the site to external threats. Some of these concerns were echoed by another member of the public during the Subcommittee meeting on March 13, 2002 in Florida City. Based on information provided by the staff and the applicant during our meeting, we conclude that the issue of voids in containment walls has been appropriately resolved at Turkey Point. With regard to concerns about storm surges, the Individual Plant Examination of External Events for Turkey Point identifies such surges as small contributors to total risk. However, the staff should document its position on this issue. The staff is generically addressing concerns with external threats.

The staff has performed a comprehensive review of the FPL application. The applicant and the staff have identified plausible aging effects associated with passive, long-lived components. Adequate programs have been established to manage the effects of aging so that Turkey Point, Units 3 and 4, can be operated in accordance with their current licensing bases for the period of extended operation, without undue risk to the health and safety of the public.

Sincerely,

George E. Apostolakis
Chairman

References:

1. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the License Renewal of Turkey Point Nuclear Plant, Units 3 and 4," February 2002.
2. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the License Renewal of Turkey Point Nuclear Plant, Units 3 and 4," September 2001.
3. Nuclear Energy Institute Report 95-10, Revision 1, "Industry Guidelines for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule," January 2000.
4. Westinghouse Owners Group Topical Report, WCAP-14574, "License Renewal Evaluation: Aging Management Evaluation for Pressurizers," July 1996.
5. Westinghouse Owners Group Topical Report, WCAP-14575, "License Renewal Evaluation: Aging Management Evaluation for Class 1 Piping and Associated Boundary Components," August 1996.
6. Westinghouse Owners Group Topical Report, WCAP-14577, Revision 1, "License Renewal Evaluation: Aging Management for Reactor Internals," dated October 9, 2000.
7. Westinghouse Owners Group Topical Report, WCAP-14422, Revision 2, "License Renewal Evaluation: Aging Management for Reactor Coolant System Supports," February 1997.
8. Letter dated February 16, 2002, from Mark P. Oncavage, a public citizen, to Noel Dudley, Senior Staff Engineer, ACRS, transmitting safety concerns regarding the continued operation of Turkey Point through the license renewal period.
9. U. S. Nuclear Regulatory Commission, NUREG-1742, "Perspectives Gained from the Individual Plant Examination of External Events (IPEEE) Program," draft report for public comment, April 2001.

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

April 19, 2002

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

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

During the 491ST meeting of the Advisory Committee on Reactor Safeguards, April 11-12, 2002, the Committee considered the draft final amendment to 10 CFR 50.55a, "Codes and Standards," and decided not to review it. The Committee has no objection to issuing the final amendment for industry use.

Reference:
Memorandum dated April 4, 2002, from J. Strosnider, NRR, to John T. Larkins, Executive Director, ACRS, Subject: Final Amendment to 10 CFR 50.55a, "Codes and Standards."

cc: A. Vietti-Cook, SECY
J. Craig, OEDO
I. Schoenfeld, OEDO
S. Collins, NRR
J. Strosnider, NRR
G. Imbro, NRR

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

May 7, 2002

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: PROPOSED NRC GENERIC LETTER 2002-XX: CONTROL ROOM ENVELOPE HABITABILITY

During the 492nd meeting of the Advisory Committee on Reactor Safeguards, May 2-3, 2002, the Committee considered the proposed NRC Generic Letter 2002-XX: Control Room Envelope Habitability. The Committee has no objection to issuing this Generic Letter for public comment.

The Committee would like the opportunity to review the draft final version of this Generic Letter, subsequent to the staff's resolution of public comments.

Reference:
U. S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, NRC Generic Letter 2022-XX: Control Room Envelope Habitability, March 28, 2002.

cc: A. Vietti-Cook, SECY
J. Craig, EDO
I. Schoenfeld, EDO
S. Collins, NRR
B. Sheron, NRR
D. Matthews, NRR
G. Holahan, NRR
M. Johnson, NRR
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

May 8, 2002

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

SUBJECT: PHEBUS-FP PROGRAM

Dear Chairman Meserve:

During the 492nd meeting of the Advisory Committee on Reactor Safeguards, May 2-3, 2002, we met with representatives of the Institut de Radioprotection et de Sûreté Nucleaire (IRSN) to discuss the PHEBUS-FP experimental program and plans for the PHEBUS-2K and PHEBUS-LOCA programs.

Observations

1. The PHEBUS-FP program is an outstanding example of an international cooperative research program that is yielding valuable data for validating severe accident analysis computer codes.

2. The proposed follow-on programs, PHEBUS-2K and PHEBUS-LOCA, promise to provide data pertinent to issues being, and will be, confronted by the NRC. High burnup fuel behavior under design basis accident conditions, fission product release and degradation of high burnup and MOX fuel, and effects of air ingression on core degradation and fission product release will be addressed in these programs.

3. Participation in these follow-on programs will yield important data not otherwise obtainable, but will require a commitment to long-term research efforts.

Discussion

The PHEBUS-FP program is an international cooperative research program to develop experimental data for validating computer codes used for severe reactor accident analysis. The experimental work is done at the Cadarache Centre in France. Partners in this program include the European Union, Canada, Japan, South Korea, Switzerland, and the United States.

The PHEBUS-FP experiments simulate the major aspects of a severe accident, beginning with the degradation of irradiated reactor fuel, release of fission products, transport of fission products through a simulated reactor coolant system, and injection of these fission products into a model of a reactor containment. Fission product behavior within the containment is examined over a period of about five days. This examination includes study of both aerosol behavior and the chemistry of radioactive iodine.

The experiments in the PHEBUS-FP program are providing data that are valuable for validating and refining computer codes used for reactor accident analysis. Data from the tests have been used to refine models of core degradation and fuel relocation, hydrogen production, and fission product speciation. The data indicate needs for refining models of aerosol deposition within the reactor coolant system and models of the aqueous and gaseous chemistry of iodine within the reactor containment.

The five large-scale tests of the PHEBUS-FP program are supported by numerous separate effects tests and extensive test analyses from a number of perspectives by the international community participating in this program. One of the tests has been designated as an International Standard Problem for benchmarking computer codes used for severe accident analyses, including the MELCOR code developed by the NRC's Office of Nuclear Regulatory Research .

The PHEBUS-FP program is an example of effective international cooperation. Partners contribute both separate effects test results and analyses to aid in the interpretation of the integrated test results. These contributions have been organized into Interpretation Circles that intensively examine individual aspects of the integral phenomenological tests. Results of these examinations are reported to a Scientific Analysis Working Group that makes recommendations to a Steering Committee concerning work needed and plans for tests.

The investigators are now considering follow-ons to the PHEBUS-FP experiments. As in the United States, European operators are under pressure to improve the efficiency of nuclear power plants. They are trying to exploit margins that have existed in the past. Best-estimate, rather than conservative, safety models are becoming more widely used. These models were developed based on data obtained with fuel at modest levels of burnup. A program of in-pile tests of design basis accident phenomena with higher burnup fuel, PHEBUS-LOCA, is now being developed as a follow-on to the PHEBUS-FP program.

A follow-on program, PHEBUS-2K, to examine severe accident phenomena and accident mitigation phenomena is also being developed. This program will examine the degradation of high burnup fuel, degradation of and fission product release from MOX fuel, and the effects of air ingression on core degradation and fission product release. These test results would be pertinent to many issues the NRC is and will be confronting. Experimental investigations of air ingression, for example, will be pertinent to issues of fuel transportation safety, spent fuel pool safety, as well as reactor accident analyses.

There is now an experienced team of researchers at the Cadarache Centre. It is likely that this team could carry out any follow-on cooperative research programs at the PHEBUS facility successfully.

Sincerely,

George E. Apostolakis
Chairman

May 10, 2002

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

Dear Chairman Meserve:

SUBJECT: CORE POWER UPRATE FOR THE BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2

During the 492nd meeting of the Advisory Committee on Reactor Safeguards, May 2-3, 2002, we completed our review of the Carolina Power and Light Company (CP&L) license amendment request for an increase in core thermal power for the Brunswick Steam Electric Plant (BSEP), Units 1 and 2. Our Subcommittee on Thermal-Hydraulic Phenomena also reviewed this matter during a meeting held on April 23, 2002. During our review, we had discussions with representatives of the applicant and the NRC staff. We also had the benefit of the documents referenced.

Recommendation

The CP&L application for an increase in core thermal power from 2558 MWt to 2923 MWt (14.3%) for the Brunswick Steam Electric Plant, Units 1 and 2, should be approved.

Discussion

The BSEP reactors are BWR/4 Mark 1 units, originally licensed to a power level of 2436 MWt. A 5% power uprate to 2558 MWt was approved by the NRC in 1996.

The requested power uprate is similar to those already approved for the Duane Arnold Energy Center, Dresden Nuclear Power Station, and Quad Cities Nuclear Power Station. The CP&L application follows the General Electric (GE) Nuclear Energy and NRC-approved ELTR1 and ELTR2 extended power uprate (EPU) licensing topical report framework, with a few exceptions that are consistent with those previously granted to other applicants and described in GE topical report NEDC-33004P, "Constant Pressure Power Uprate" (CPPU). In our letter of April 17, 2002, we recommended approval of the CPPU topical report for application to BWR power increases of up to 20% of the original licensed thermal power.

CP&L has committed to modify the standby liquid control system (SLCS) in which the boron solution is sufficiently enriched with Boron-10. This modification will allow the shutdown capability to be met in the event of an anticipated transient without scram (ATWS) with the use of only one of the two available SLCS pumps. The licensee calculates that this modification will reduce the plant's internal events core damage frequency (CDF) and large early release frequency (LERF) by 9% and 28%, respectively. Without the use of enriched boron, the ATWS risk increases slightly due to shortened times for operator decisions. Because of the significant safety benefit that is obtained by offsetting the most significant risk increase associated with EPU, we agree that this modification to the SLCS should be implemented.

The staff has determined that the application meets all of the requirements of the regulations, uses approved codes, and follows the required procedures. As in the case of previous staff evaluations of EPUs, these determinations could have benefitted by including the results of independent computations and detailed checks of calculations to support the staff's review and audits of the procedures and conclusions described by the applicant.

We encourage the staff to continue to pay close attention to the details of core reload analyses at Brunswick and other BWR EPU plants. This is particularly important with regard to the ways that core thermal success criteria will continue to be met as more sophisticated fuel design and reload management techniques are implemented. The staff should assess the need for more detailed thermal-hydraulic models of the core, replacing the current "averaging" approaches, to complement present neutronic analyses that model the wide variations in fuel composition and power level throughout the core.

This review demonstrates an inherent problem in the "two-tier" regulatory system. The application for the EPU was not risk-informed, yet a PRA was submitted. This creates a situation in which the PRA is not seriously reviewed, although it is part of the record. Also, the uncertainties in human reliability analysis are significant, but there is no mention of them. The applicant used human reliability models that have not been reviewed by the staff. The staff acknowledges that large uncertainties are present and that the models have not been reviewed. However, the staff concludes that insights regarding the relative importance of operator actions can be gained. In addition, the potential increases in the change in core damage frequency (ΔCDF), that could arise if the PRA were capable of modeling the effect of margin reductions on risk, are not included.

One can claim that the actual value of ΔCDF is not very relevant because the basis for the decision is not risk-informed. Yet, by not raising concerns about the quality of these numbers, the staff implies some degree of acceptance. Maintaining public confidence is a goal of the Commission, which is not served by tacit acceptance of unreviewed models.

PRA quality is essential for risk-informing the regulations. Improvements in PRA quality, such as inclusion of the effects of margin reductions on risk and improving human reliability models, may be discouraged as long as important decisions such as granting power uprates are made by "accepting" PRAs without criticism because the application is not risk-informed.

Drs. F. Peter Ford and Victor H. Ransom did not participate in the Committee's deliberations regarding this matter.

Additional comments by ACRS Member Thomas S. Kress and ACRS Member George E. Apostolakis are presented below.

Sincerely,

George E. Apostolakis
Chairman

<u>Additional Comments by ACRS Member Thomas S. Kress</u>

I agree with the Committee's position on the way the PRA results are used in evaluating non-risk-informed submittals for changes to the licensing basis. I have an additional related concern that the concepts in Regulatory Guide 1.174 are not being properly implemented in the guidance on how to view these submittals in a risk-informed manner.

For example, the Brunswick PRA submittal reports a LERF value of 4.27×10^{-6}/yr and a ΔLERF of about 2×10^{-7}/yr as a result of the power uprate, not including the SLCS modifications. The claim is that these values place this change to the licensing basis into Region II of the Regulatory Guide 1.174 acceptance guideline, which would permit this proposed power uprate.

There are a number of things wrong with this view of Regulatory Guide 1.174.

1. The PRA did not include fire, seismic, or shutdown conditions. If included, these are likely to increase the assessed LERF value by a factor of 2.

2. There are two units on the site. As LERF is a site criterion that is a surrogate for the Commission's prompt fatality safety goal, then the LERF value for each unit must be added together to constitute the appropriate Regulatory Guide 1.174 site LERF. This increases the LERF by a factor of 2.

3. The LERF value submitted was a "point estimate" It can be guessed that the actual mean can be at least a factor of 2 greater than this.

4. The site LERF acceptance value is supposed to be a surrogate for the Commission's prompt fatality safety goal. The power uprate, to a first approximation, will increase the fission product inventory by 15% and, if the dose/consequence model were linear, this would increase the prompt fatalities by 15%. To account for this, the calculated LERF for comparison with the acceptance criteria in Regulatory Guide 1.174 should be increased by 15%.

If these missing conditions were included, the most appropriate site LERF at Brunswick should be about:

$$(2)(4.27 \times 10^{-6})\ (3)\ (1.15) = 3.0 \times 10^{-5}/\text{yr}$$

The assessed ΔLERF of 2×10^{-7}/yr must also be doubled because there are two units on the site; therefore, site ΔLERF = 4.0×10^{-7}/yr.

These two values (LERF = 3.0×10^{-5}/yr and ΔLERF = 4.0×10^{-7}/yr) place Brunswick squarely into Region I of the Regulatory Guide 1.174 acceptance guidelines, which is supposed to put into question the presumption that adequate protection is preserved.

It can be claimed that the modification to SLCS in combination with the power uprate actually results in a decrease in risk and therefore ought to be automatically acceptable. We must be careful with this approach to dealing with aggregate changes. Clearly, the modifications to SLCS that result in a risk decrease are acceptable. However, this change at Brunswick does not by itself decrease the LERF enough to take Brunswick out of the Region I. Therefore, even with this modification, the plant is still in the Region for which increase in risk (due to the power uprates) should not be allowed without additional justification.

While I am convinced that a proper Level 3 risk analysis for the Brunswick site would justify approving the power uprate request, I am disturbed by the staff's cavalier use of the risk-informed decisionmaking process.

Additional Comments by ACRS Member George E. Apostolakis

Dr. Kress' comments are an excellent example of what happens when the staff does not subject the submitted risk information to serious review because the application is not risk-informed. They also demonstrate how conclusions can be affected when the PRA is incomplete and/or poorly done.

References:

1. Memorandum dated March 29, 2002, from John A. Zwolinski, Office of Nuclear Reactor Regulation, NRC, to John T. Larkins, ACRS, Subject: Brunswick Steam Electric Plant, Units 1 and 2 - Draft Safety Evaluation for Proposed Extended Power Uprate License Amendment (Predecisional).
2. Memorandum dated August 9, 2001, from John S. Keenan, CP&L, to U.S. NRC, Subject: Brunswick Steam Electric Plant, Units Nos. 1 and 2, Request for License Amendments Extended Power Uprate (Proprietary).
3. CP&L Memorandums dated March 22, March 20, March 14 (proprietary), March 12 (proprietary), March 7, March 5, March 4 (proprietary), February 25 (3), February 21 (2), February 14, February 13, February 4, February 1, and January 24 (proprietary), 2002, December 20, December 17 (2), December 4, December 1, November 30, November 28 (proprietary), November 7, November 1, and October 17, 2001, regarding Brunswick Steam Electric Plant, Unit Nos. 1 and 2, Extended Power Uprate, responses to request for additional information from NRC.
4. Letter dated May 1, 2002, from Edward T. O'Neil, CP&L, to George E. Apostolakis, ACRS, Subject: Brunswick Steam Electric Plant, Unit Nos. 1 and 2, Response to Request for Information - Extended Power Uprate (Proprietary).
5. GE Nuclear Energy, Topical Report, NEDC-32424P-A, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate" (ELTR-1), February 1999 (Proprietary).

5. GE Nuclear Energy, Topical Report, NEDC-32523P-A, "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate" (ELTR-2), February 2000 (Proprietary).
6. GE Nuclear Energy, Topical Report, NEDC-32523P-A, Supp 1, Volume 1, "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate - Supplement 1, Volume I," February 1999, and Volume II, April 1999 (ELTR-2) (Proprietary).
7. GE Nuclear Energy Topical Report, NEDC-33004P, Revision 1, "Constant Pressure Power Uprate," July 26, 2001 (Proprietary)

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

June 17, 2002

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

Dear Chairman Meserve:

SUBJECT: POLICY ISSUES RELATED TO ADVANCED REACTOR LICENSING

During the 493rd meeting of the Advisory Committee on Reactor Safeguards (ACRS), June 6-8, 2002, we were briefed by representatives of the NRC's Office of Nuclear Regulatory Research (RES) on issues that have potential policy implications for advanced reactor licensing, and the plans for seeking the Commission's guidance for resolving these issues. We also had the benefit of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

(1) The RES staff has identified appropriate policy issues and posed questions that must be addressed to resolve them.

(2) The existing agency positions on some of these policy issues should be reevaluated because of new perspectives on risk-informed regulation and defense in depth, as well as the new reactor designs that may be proposed.

(3) The need for greater specificity in the application of defense in depth should be made a separate overarching issue.

DISCUSSION

The issues identified by the staff fall into the following five areas:

- event selection and safety classification
- fuel performance and qualification
- source term
- containment versus confinement
- emergency evacuation

We note that in order to resolve these issues, the role of PRA and high-level risk acceptance criteria are essential in the design approval process.

The staff also identified two overarching policy issues:

(1) how to implement the Commission's "expectation" that advanced reactors will provide enhanced margins of safety

(2) what should be the relationship between the NRC's safety requirements and international safety requirements

We recommend that the need for greater specificity in the application of defense in depth should be singled out of the first overarching issue and made a separate and distinct overarching issue. With respect to the second overarching issue, we agree that it would be highly desirable to understand the bases for the international safety requirements. Nonetheless, we note that it would not be unreasonable for different countries to have different safety standards on a cost/benefit basis.

The identification and resolution of these policy issues is important to the process of licensing advanced reactors. The existing agency positions on some of these policy issues should be reevaluated because of new perspectives on risk-informed regulation and defense in depth, as well as the new reactor designs that may be proposed. Much work remains to be done, and we plan to maintain continuing interactions with the staff on possible approaches and options for resolving these policy issues.

Sincerely,

George E. Apostolakis
Chairman

References:

1. Information Paper (Draft Predecisional) dated May 23, 2002, from William D. Travers, Executive Director for Operations, NRC, for the Commissioners, Subject: Plan for Resolving Policy Issues Resulting from Technical Considerations Related to Advanced Reactor Licensing.
2. U.S. Nuclear Regulatory Commission, NUREG-1226, "Development and Utilization of the NRC Policy Statement on the Regulation of Advanced Nuclear Power Plants," dated June 1988.

UNITED STATES
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WASHINGTON, D.C. 20555-0001

June 17, 2002

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

Dear Chairman Meserve:

SUBJECT: PROPOSED REVISION TO 10 CFR 50.48 ENDORSING NFPA-805, "PERFORMANCE-BASED STANDARD FOR FIRE PROTECTION FOR LIGHT WATER REACTOR ELECTRIC GENERATING PLANTS"

During the 493rd meeting of the Advisory Committee on Reactor Safeguards (ACRS), June 6-8, 2002, we reviewed the proposed revision to 10 CFR 50.48 to endorse the National Fire Protection Association (NFPA) standard 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," as a voluntary, alternative set of risk-informed, performance-based fire protection requirements for light water reactors. During our review, we had the benefit of discussions with representatives of the NRC staff and the Nuclear Energy Institute (NEI). We also had benefit of the documents referenced.

RECOMMENDATION

The NRC staff should proceed with issuing the proposed rule for public comment, consistent with the rulemaking plan schedule.

DISCUSSION

The current fire protection requirements for nuclear power plants are deterministic. As such, they are designed to establish an engineering margin for fire protection by ensuring the post-fire survival of at least one set of safety systems that can be used to take the plant safely to hot and cold shutdown. These requirements were developed before the NRC staff or the industry had the benefit of probabilistic risk assessments (PRAs) for fires, or a significant body of commercial reactor operating experience. Consequently, the current requirements are prescriptive and, due to their inflexibility, may create an unnecessary regulatory burden. Today, it is possible to better quantify the probabilities of fire-initiated events, and to integrate fire analysis results to assess the overall safety impact of fire events.

Members of the NRC staff participated in the development of this standard. The staff has concluded that, with certain exceptions, NFPA-805 can serve as a risk-informed, performance-based, voluntary alternative to the fire protection requirements of 10 CFR 50.48(b) and Appendix R to 10 CFR Part 50. We have examined the exceptions to NFPA-805 standard in the proposed rule and found them appropriate.

The NRC staff has proposed a revision to 10 CFR 50.48 that seeks to establish a fire protection rule that is better oriented toward reactor safety in that it allows a risk-informed, performance-based option. The proposed rule endorses NFPA-805 and would allow licensees the flexibility to use alternative approaches to meet the fire safety objectives. A risk-informed rule will enable licensees and the staff to focus their resources on the most risk-significant fire protection equipment and activities, and may also reduce the need for exemptions.

The NRC staff is planning a four-step process to implement the risk-informed option for fire protection. The first step is to modify the rule to enable licensees to utilize NFPA-805 standard as an option. The second step involves the development of implementation guidance for the rule by NEI, and possible endorsement of the NEI guidance by the staff in a regulatory guide. The third step is the development of inspection guidance, followed by inspector training as the fourth and final step.

During our 459th meeting, February 3-6, 1999, we reviewed a draft version of the NFPA-805 standard. Our report of February 18, 1999, was critical of the draft, concluding that "The draft Standard is not, however, a distinct, risk-informed, performance-based alternative to these existing fire protection requirements." We believe that the NFPA and the NRC staff have been responsive to our comments.

The staff considers the proposed rule to be "an essential first step in integrating risk insights and the advances in fire science that have occurred since issuance of Appendix R over twenty years ago." We concur with the staff and conclude that it is appropriate to issue the proposed rule for public comment at this time.

While we are encouraged by the progress toward risk-informing the existing fire protection requirements, we offer a cautionary note. The real value of this work accrues when licensees voluntarily adopt the new standard and begin to revise their fire protection programs. The implementation guidance, including the approved techniques for performing fire PRAs and fire modeling, must require methods and models commensurate with the levels of risk, while being careful to not create unnecessary barriers to the use of the Standard.

Sincerely,

George E. Apostolakis
Chairman

References:

1. NFPA-805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," 2001 Edition, National Fire Protection Association.
2. Memorandum dated May 8, 2002, from Gary M. Holahan, Office of Nuclear Reactor Regulation, NRC, to Karen D. Cyr, et.al., Subject: Concurrence on Part 50 Proposed Rulemaking Package: Light Water Reactor Adoption of Risk-Informed, Performance-Based Fire Protection Requirements (NFPA-805).

3. Report dated February 18, 1999, from Dana A. Powers, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: NFPA-805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants."

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

June 17, 2002

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

Dear Dr. Travers:

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

During the 493rd meeting of the Advisory Committee on Reactor Safeguards, June 6-8, 2002, we reviewed the technical assessment of Generic Safety Issue (GSI)-168, "Environmental Qualification of Low-Voltage Instrumentation and Control Cables," proposed by the Office of Nuclear Regulatory Research (RES). During this review, we had the benefit of discussions with representatives of the RES staff. We also had the benefit of the documents referenced.

RECOMMENDATIONS AND CONCLUSIONS

We recommend that:

- A discussion of the treatment of the instrumentation and control (I&C) cables during the license renewal term be included in the generic communication recommended by RES.
- The staff encourage the industry to perform further developmental work on techniques for monitoring I&C cable condition.

We agree with the staff's conclusions that:

- The current equipment qualification (EQ) process for low-voltage I&C cables is adequate for the duration of the current license term of 40 years.
- Knowledge of the conservatism in the operating environment, as compared to the qualification environment, coupled with observation of the condition of the cables can be used to extend the qualified life of the cables.
- A combination of condition monitoring techniques is needed since no single technique is effective to detect degradation of I&C cables.

- Test results and other pertinent information should be disseminated to the nuclear industry through a generic communication.

DISCUSSION

GSI-168 considers the EQ of low-voltage I&C cables. These cables are particularly important, since their failure could result in misleading information being presented to operators.

Originally, 43 sub-issues were identified concerning the operability of I&C cables during a loss-of-coolant accident. All but six were resolved by researching previous literature on the subject. The RES technical assessment evaluates the results of research and testing performed to resolve the six remaining issues. We agree with RES that, although some I&C cables failed during testing, the current EQ process is adequate for the current license term of 40 years. This conclusion is based on the implementation of licensee programs to demonstrate that there is sufficient margin in environmental conditions in which the I&C cables operate and on implementation of a monitoring program for these cables.

Walkdowns to look for visible signs of anomalies attributable to cable aging coupled with monitoring of operating environment have proven to be useful. The staff should encourage the industry to perform additional developmental work on techniques for monitoring I&C cable condition.

The RES assessment suggests that industry implementation of a monitoring program will result in a small reduction in core damage frequency. We support the RES recommendation that a generic communication be issued to the industry to notify them of these results. The generic communication should include a discussion of the treatment of the I&C cables during the license renewal term.

We would like to review the proposed resolution of GSI-168.

Additional comments by ACRS Members Dana A. Powers, F. Peter Ford, Victor H. Ransom, Stephen L. Rosen, and John D. Sieber are provided below.

Sincerely,

George E. Apostolakis
Chairman

<u>Additional Comments by ACRS Members Dana A. Powers, F. Peter Ford, Victor H. Ransom, Stephen L. Rosen, and John D. Sieber</u>

The staff has recommended a resolution of cable integrity issues for one class of design-basis accidents, loss-of-coolant accidents. For these accidents, temperature and radiation loads are of dominant concern. Other design-basis accidents, such as main steamline breaks, can impose other loads on cables such as large amplitude vibrations and bending. The staff has not investigated the effects of these other loads on the integrity of aged cables adequately. What the staff has done is adequate to resolve the six, open, sub-issues of GSI-168. The staff should consider additional examinations of cable integrity as part of its ongoing work on mechanical loads and vibrations associated with main steamline breaks and other design-basis accidents.

References:

1. Memorandum dated May 6, 2002, from Michael E. Mayfield, Division of Engineering Technology, Office of Nuclear Regulatory Research, to John T. Larkins, ACRS, Subject: Proposed Technical Assessment of Generic Safety Issue (GSI) 168, "Environmental Qualification of Low-Voltage Instrumentation and Control (I&C) Cables."
2. U. S. Nuclear Regulatory Commission, NUREG-6704, "Assessment of Environmental Qualification Practices and Condition Monitoring Techniques for Low-Voltage Electrical Cables," February 2001.

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

June 17, 2002

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

Dear Dr. Travers:

SUBJECT: RECOMMENDATIONS PROPOSED BY THE OFFICE OF NUCLEAR REGULATORY RESEARCH FOR RESOLVING GENERIC SAFETY ISSUE-189, "SUSCEPTIBILITY OF ICE CONDENSER AND MARK III CONTAINMENTS TO EARLY FAILURE FROM HYDROGEN COMBUSTION DURING A SEVERE ACCIDENT"

During the 493rd meeting of the Advisory Committee on Reactor Safeguards, June 6-8, 2002, we reviewed the recommendations proposed by the Office of Nuclear Regulatory Research (RES) to resolve Generic Safety Issue (GSI)-189, "Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident." During this review, we had the benefit of discussions with the NRC staff, a representative of the Union of Concerned Scientists, members of the public, and a representative of the Tennessee Valley Authority. We also had the benefit of the documents referenced.

RECOMMENDATION

RES should complete its additional analyses to quantify the uncertainties prior to providing the technical assessment results to the Office of Nuclear Reactor Regulation (NRR), and NRR should factor the uncertainties into the final resolution of GSI-189.

DISCUSSION

GSI-189 was proposed in response to SECY-00-0198, "Status Report on Study of Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50 (Option 3) and Recommendations on Risk-Informed Changes to 10 CFR 50.44 (Combustible Gas Control)." In SECY-00-0198, the staff recommended that safety enhancements that have the potential to pass the backfit test be assessed for mandatory application through the generic issue process.

During severe accidents, ice condenser and pressure-suppression Mark III containments condense steam and concentrate hydrogen to the extent that they would become vulnerable to a hydrogen detonation. In 1980, these plant types were retrofitted with powered igniters and air return fans[1] to provide controlled burning of the hydrogen over the time period of production to limit the concentration and preclude a hydrogen detonation. During a station blackout (SBO) event, however, alternating current (AC) power to the igniters and fans would not be available. The issue, therefore, is whether it would be feasible and cost-beneficial to provide backup AC power supplies to the igniters and/or the air return fans.

RES conducted an analysis to provide technical input to NRR to support a regulatory analysis for potential backup power options that could be used to resolve this GSI. It consists of a cost/benefit analysis following the appropriate regulatory analysis guidelines.

The scope of the study included the following four options.

1. A pre-staged dedicated diesel generator to provide backup AC power only to the igniters.

2. A pre-staged dedicated diesel generator to provide backup AC power to both the igniters and the air return fans.

3. A low-cost "off-the-shelf" portable diesel generator to provide backup AC power only to the igniters.

4. Use of passive autocatalytic recombiners for hydrogen control in lieu of igniters and/or air return fans.

A fifth option of a low-cost "off-the-shelf" portable diesel generator to provide backup AC power to both the igniters and the fans was considered to be impractical because the required power was deemed to be too large for a portable diesel.

RES performed analyses by using the MELCOR and CONTAIN computer codes to assess the change in the conditional probability of containment failure with and without the availability of AC power. The MELCOR analysis was also used to assess whether the use of igniters alone (without the air return fans) would be sufficient to prevent a hydrogen detonation.

On the basis of its analyses, RES concluded that providing backup power to igniters alone would be sufficient to preclude a hydrogen detonation, and only the low-cost option (Option 3) passed the regulatory analysis cost-benefit criterion.

We believe that these results are highly uncertain, with regard to both the costs and benefits and the judgment that igniters alone would preclude a hydrogen detonation. RES is continuing its technical analysis to better quantify the uncertainties that affect these judgments. We expect that the resulting uncertainty determination will include assessment of the uncertainty related to the use of a control volume code (MELCOR) to determine detailed hydrogen concentration distributions as well as general model uncertainties. As recognized by the regulatory analysis guidelines, the ultimate resolution of this issue should consider these uncertainties. We recognize that the computed cost-benefit ratio based on point values indicates that Option 2, above, does not pass the backfit screening. However, this cost-benefit

[1] Air return fans are a feature of ice condenser plants only.

ratio is close to being acceptable. When the uncertainties are factored into the assessment, the analysis could yield a different conclusion.

We would like to review the results of the additional analyses and the proposed RES recommendation to NRR for resolving GSI-189.

ACRS member Victor H. Ransom did not participate in the Committee's deliberations regarding this matter.

Sincerely,

George E. Apostolakis
Chairman

References:

1. Memorandum dated May 13, 2002, from Farouk Eltawila, Office of Nuclear Regulatory Research, to John T. Larkins, ACRS, Subject: RES Proposed Recommendation for Resolving Generic Safety Issue 189: "Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident."
2. Information Systems Laboratories, Inc. report entitled, "Backup Power for PWRs with Ice Condenser Containments and for BWRs with Mark III Containments under SBO Conditions: Impact Assessment," dated May 1, 2002
3. Brookhaven National Laboratory draft letter report entitled, "Benefit Cost Analysis of Enhancing Combustible Gas Control Availability at Ice Condenser and Mark III Containment Plants," dated April 25, 2002.
4. Draft report entitled, "Hydrogen Control Calculations for the Sequoyah Plant Station Blackout Scenario," April 2002
5. NUREG/CR-5586, "Mitigation of Direct Containment Heating and Hydrogen Combustion Events in Ice Condenser Plants," October 1990.

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

June 20, 2002

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

SUBJECT: VESSEL HEAD PENETRATIONS AND VESSEL HEAD DEGRADATION

Dear Chairman Meserve:

During the 493rd meeting of the Advisory Committee on Reactor Safeguards, June 6-8, 2002, we heard presentations by and held discussions with representatives of the Electric Power Research Institute Materials Reliability Program (EPRI/MRP), First Energy Nuclear Operating Company (FENOC), and the NRC staff regarding cracking and leaking observed in pressurized water reactor (PWR) Alloy 600 reactor pressure vessel (RPV) head penetrations, including control rod drive mechanism (CRDM) nozzles, and the degradation observed at Davis-Besse Nuclear Power Station. This matter was also discussed during a meeting of the Materials and Metallurgy and the Plant Operations Subcommittees on June 5, 2002. During our reviews, we had the benefit of the documents referenced.

This report addresses technical issues associated with vessel head penetrations (VHP) cracking and degradation. We have excluded here issues of safety culture and the adequacy of the Reactor Oversight Process, which the Davis-Besse incident raises.

CONCLUSIONS AND RECOMMENDATIONS

1. The draft "Vessel Head Penetration Nozzle Cracking Action Plan," developed by the Office of Nuclear Reactor Regulation (NRR) is sufficiently comprehensive to allow the short- and long-term management of cracking issues associated with Alloy 600.

2. The approach proposed by industry to manage cracking incidents in VHP assemblies through the use of various inspection methods is reasonable in principle, and is in line with NRC's goal to move toward risk-informed regulation. Prior to issuance of another generic communication, certain questions regarding the specific inspection techniques and frequencies, now the subject of ongoing discussions between the staff and industry, should be resolved.

3. We agree with the staff's conclusion that there are no plants with conditions similar to those that led to the degradation at Davis-Besse. This conclusion is based on the initial responses to Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity," dated March 18, 2002, and on

interactions with licensees, resident inspectors, regional staff, and other information provided to the staff.

4. In order to define the inspection frequencies, corrosion rates in low-alloy steel adjacent to vessel head penetrations should be determined.

Background

Presentations on cracking in VHP assemblies were made by the staff and industry at subcommittee and full Committee meetings in July and November 2001, and again in April 2002 on the low-alloy steel corrosion observed at Davis Besse in April 2002. Following the meeting in July 2001, we issued a letter supporting the issuance of Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles." That letter included several technical questions associated with, for instance, the adequacy and qualification of visual inspection processes and the qualification of stress corrosion data bases that would be used to define inspection periodicities. In June 2002, presentations were made by the staff and industry on data relevant to these issues.

Discussion

The staff has developed a draft VHP Nozzles Cracking Action Plan, which addresses short- and long-term regulatory issues. The short-term actions relate, for example, to reviewing the responses to Bulletin 2001-01, addressing policy matters related to management of cracking through continued inspections for leakage, and dealing with plant-specific issues. The long-term actions relate to the criteria and regulatory tools for nozzle inspection requirements and considerable efforts to develop the technical basis to support the regulatory approach for managing this issue. This approach includes flaw evaluation criteria, crack growth rate evaluations, nondestructive examination, probabilistic fracture mechanics, and risk assessment. The MRP is performing a considerable amount of complementary work and engaging in a healthy communication with the staff.

A persistent question raised in all of the ACRS meetings relates to the completeness of cracking prediction methods, which must account for the combined effects of materials, environment, and stress parameters on crack initiation and propagation. All of these effects are being addressed in the draft NRR action plan and the ongoing MRP Alloy 600 project. Thus, the effect of environment (primarily temperature), stress (intensity), and the range of material conditions are accounted for in deriving the probabilistic fracture mechanics basis for defining inspection frequencies. There is, however, another method, based on time and temperature, that was used by the staff and industry in 2001 and 2002 to rank various plants for inspection prioritization. If this method continues to be used as a management tool, then it should be upgraded to cover not only operating time and temperature, but also material effects. These more complete algorithms have been used in France to manage CRDM cracking.

The draft action plan focuses on the evaluation of the cracking kinetics of Alloys 600 and 182, the materials currently used in the construction of the VHP assemblies. This focus is appropriate for managing the current problem. However, it is foreseen that many plants will choose to replace their pressure vessel heads with new heads equipped with VHP assemblies using Alloys 690 and 152. These alloys have performed well in laboratory tests, replacement steam generators tubes, and VHP assemblies in France. However, there is an insufficient information base on Alloys 690 and 152 to achieve the same technical management objectives set forth in the current action plan for Alloys 600 and 182. Thus, it would be appropriate for the industry to initiate programs that will quantify improvements in stress corrosion resistance in VHP assemblies and determine the impact that this has on inspection methods and frequencies for Alloys 690 and 152.

The industry's proposed inspection plan for VHP assemblies indicates a choice of inspection techniques and frequencies of inspection for specific plants based on the impact of cracking on the risk of rod ejection. This plan has a sound technical foundation, and is consistent with the staff's objective of managing cracking incidents through adequate and timely inspection and with a sound risk-informed basis. However, the current focus of the industry's plan is limited to circumferential cracking, whereas, in addition to circumferential cracking, the staff's concern is throughwall cracking and RPV head material degradation. The industry's proposal is the subject of intensive discussions. Topics of discussion include inspection techniques (visual versus 100% volumetric), frequency of inspections, code requirements concerning leakage and depth of crack, and maintenance of the defense-in-depth principle.

Based on the initial responses to Bulletin 2002-01, the staff concluded that there are no plants with conditions similar to those that led to the degradation at Davis-Besse. This conclusion was based on visual inspections of the RPV head for boric acid deposits, interactions with licensees, resident inspectors, regional staff, and other information provided to the staff. It was agreed among staff and industry, however, that this inspection technique, though adequate for detecting gross degradation, is not capable of sizing any pressure vessel corrosion. Thus, there is a need to develop an inspection strategy (i.e., inspection technique and frequency) that is appropriate for this type of corrosion degradation and then factor it into the current proposed industry inspection plan which is centered on the CRDM cracking. Part of this upgraded inspection strategy must be based upon the kinetics of low-alloy steel corrosion in the annulus between the CRDM tube and the pressure vessel head. Several scenarios have been hypothesized that could lead to high corrosion rates with limiting conjoint criteria that would suggest that high corrosion rates in this location (circa 1 inch/year) would not be observed frequently. The plant design and operating conditions that control corrosion in this location is not now known. Therefore, there is an urgent need to confirm these hypotheses experimentally.

The staff and industry are working to resolve these problems, and we would like to be kept informed as the work progresses.

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

Sincerely,

George E. Apostolakis
Chairman

References

1. Draft Memorandum from Brian Sheron, Office of Nuclear Reactor Regulation, NRC, to Samuel J. Collins, Office of Nuclear Reactor Regulation, NRC, Subject: Vessel Head Penetration Nozzles Cracking Action Plan, received March 29, 2002.
2. NRC RES-MRP Alloy 600 Meeting Slides (Inspection Plan and Inspection Plan Writeup), May 22, 2002,
3. U. S. Nuclear Regulatory Commission Bulletin 2001-01: Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles, August 3, 2001.
4. U. S. Nuclear Regulatory Commission Bulletin 2002-01: Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity, March 18, 2002
5. Letter dated May 21, 2002, from H. Bergendahl, First Energy Nuclear Operating Company, to J. E. Dyer, NRC Region III, Subject: Transmittal of Davis-Besse Nuclear Power Station, Unit 1 Return to Service Plan.
6. Letter dated May 15, 2002, from H. Bergendahl, First Energy Nuclear Operating Company, to J. E. Dyer, NRC Region III, Subject: Supplemental Information in Response to NRC Question Number 24 on the Preliminary Probable Cause Summary Report Dated March 22, 2002.
7. Letter dated April 18, 2002, from H. Bergendahl, First Energy Nuclear Operating Company, to J. E. Dyer, NRC Region III, Subject: Confirmatory Action Letter Response - Root Cause Analysis Report.
8. Letter dated May 3, 2002, from J. E. Dyer, Administrator, Region III, to H. Bergendahl, First Energy Nuclear Operating Company, Subject: Davis-Besse Nuclear Power Station NRC Augmented Inspection Team- Degradation of the Reactor Pressure Vessel Head - Report No 50-346/02-03(DRS).
9. NRC Information Notice 2002-13: Possible Indicators of Ongoing Reactor Pressure Vessel Head Degradation, April 4, 2002.

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

July 16, 2002

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: DRAFT REGULATORY GUIDE DG-1119 (PROPOSED REVISION 1 TO REGULATORY GUIDE 1.180), "GUIDELINES FOR EVALUATING ELECTROMAGNETIC AND RADIO-FREQUENCY INTERFERENCE IN SAFETY-RELATED INSTRUMENTATION AND CONTROL SYSTEMS"

During the 494th meeting of the Advisory Committee on Reactor Safeguards, July 10-12, 2002, the Committee considered draft Regulatory Guide DG-1119 and decided not to review it at this time. The Committee has no objection to issuing this Guide for public comment. The Committee plans to review the draft final version of DG-1119 after reconciliation of public comments.

Reference:
Memorandum dated July 5, 2002, from Mike Mayfield, Office of Nuclear Regulatory Research, to John T. Larkins, Executive Director, ACRS, Subject: ACRS Review of Draft Regulatory Guide DG-1119, Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems (Proposed Revision 1 to Regulatory Guide 1.180).

cc: A. Vietti-Cook, SECY
J. Craig, OEDO
J. Schoenfeld, OEDO
A. Thadani, RES
M. Mayfield, RES
C. Antonescu, RES

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

July 18, 2002

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

Dear Dr. Travers:

SUBJECT: DRAFT ADVANCED REACTOR RESEARCH PLAN

During the 494th meeting of the Advisory Committee on Reactor Safeguards, July 10-12, 2002, and a meeting of our Subcommittee on Future Plant Designs on July 8, 2002, we were briefed by representatives of the NRC's Office of Nuclear Regulatory Research (RES) on the subject Plan. We also had the benefit of the document referenced.

The draft Advanced Reactor Research Plan appears to us to be a very competent effort by the staff. It is comprehensive and reflects a high level of understanding of the issues, existing state of the art, and past and ongoing research results and activities. We commend the RES staff on its effort to date. The Plan is not yet complete in the sense that it does not establish resources, schedules, and milestones. Nevertheless, we believe that addressing the research needs already identified in the Plan is very important.

COMMENTS

1. We agree that research on High Temperature Gas Cooled Reactors (HTGRs) should continue. However, given the current uncertain status of the Pebble Bed Modular Reactor (PBMR), the research for HTGRs should focus on generic issues and the Gas Turbine-Modular Helium Reactor (GT-MHR) concept.

2. We consider the development of fission product release models for TRISO fuels to be the key research need for the gas-cooled reactor concepts. All the current models for fission product release in the MELCOR computer code are empirical and based on data obtained from light water reactor (LWR) fuel at burnup levels less than 45 GWd/t. To extend these models to HTGRs will require research on fission product release from highly irradiated HTGR fuel. Even the form of the empirical models (diffusive in nature) may not be appropriate to TRISO fuel for which the release of fission products is primarily related to the failure rate of the coatings, which is not well-described by a diffusive-like correlation.

3. A viable research plan can be developed in the absence of a well-defined framework for risk-informed regulations. However, such a framework can help prioritize the research and is important for other reasons. The work on the framework should be given higher priority.

4. Plans should be developed for experiments to investigate degradation and fission product release characteristics of the advanced LWR's core with very high-burnup fuel [particularly International Reactor Innovative and Secure (IRIS) design].

5. A risk-informed approach for selecting design-basis events and choosing acceptance criteria for the new designs needs to be developed.

6. The use of Phenomena Identification and Ranking Table (PIRT) is an essential ingredient of the Plan and should be developed early in the process. Because we have doubts that a "super-PIRT" that encompasses the entire program would be effective, the PIRTs should be focused on specific research areas.

7. Consideration should be given to research to determine whether the buildup and characteristics of radioactivity in the coolant system during the operating phase of the HTGRs could be used to infer whether the as-installed fuel quality meets the required (licensing-basis) quality.

8. The Plan should include an element to maintain cognizance of the international near-term deployment and GEN IV concepts, with anticipation that research eventually may be needed to address issues associated with technology concepts that are significantly different than those of the Plan's focus.

9. If in-vessel retention via external flooding of the reactor vessel is anticipated as an accident management strategy for AP1000 (and perhaps IRIS), we believe this reopens the need for additional consideration of fuel coolant interactions (steam explosions). The state of the art for fuel coolant interactions is not yet sufficiently advanced to predict the occurrence and energetics of steam explosions.

10. Because there is a general need for large-scale integral testing of new concepts, the staff should evaluate the utility of the proposed concept of "licensing by test."

Additional comments by ACRS Members Dana A. Powers, Stephen L. Rosen, and Graham B. Wallis are provided below.

Sincerely

George E. Apostolakis
Chairman

Additional Comments by ACRS Members Dana A. Powers, Stephen L. Rosen, and Graham B. Wallis

Design-basis accidents are prominent features of the regulatory process for existing reactors. The design-basis accident concept, which originated in the 1950s, was an important element of reactor safety analysis in an era when comprehensive, integrated analyses involving wide ranges of accident initiators and the possibility of multiple systems failures were not practical undertakings. It can be argued that design-basis accidents have served the safety regulation of the current generation of nuclear power plants well. It must also be acknowledged that the accident at Three Mile Island revealed deficiencies of the design-basis accident concept. Design-basis accidents divert safety focus toward stylized accidents that, by definition, have exceptionally low probabilities at the expense of ensuring plants have capabilities of coping with more likely events.

The conduct of comprehensive, integrated plant analyses is now well-developed and, indeed, such analyses are essential features of the regulatory process for advanced reactors. These analyses supplant the need for design-basis accidents in the regulatory process for advanced reactors. Specialized attention to a few, low probability accidents does not add to plant safety if integrated, comprehensive accident analyses are done well. Design-basis accidents do create unnecessary burdens for both licensees and regulators. Design-basis accidents, then, should not be considered in the Advanced Reactor Research Plan.

<u>Reference:</u>

U. S. Nuclear Regulatory Commission, Advanced Reactor Research Plan (Draft), Revision 1, Office of Nuclear Regulatory Research, June 2002.

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

July 18, 2002

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

Dear Dr. Travers:

SUBJECT: RISK METRICS AND CRITERIA FOR REEVALUATING THE TECHNICAL BASIS OF THE PRESSURIZED THERMAL SHOCK RULE

During the 494th meeting of the Advisory Committee on Reactor Safeguards, July 10-12, 2002, we met with representatives of the NRC staff to discuss the status of the staff's work to identify risk metrics and criteria that can be used for reevaluating the technical basis of the pressurized thermal shock (PTS) rule. During our review, we had the benefit of the documents referenced.

We were previously briefed by the staff on the methodology and initial results of the PTS reevaluation project during our meeting on February 7-8, 2002, and we issued a letter dated February 14, 2002.

OBSERVATION

The proposed options for PTS acceptance criteria do not properly reflect the potential impact of air-oxidation source term on risk.

Discussion

The NRC staff has proposed the following three options for quantitative acceptance criteria for reactor vessel failure frequency.

A reactor vessel failure frequency of 5×10^{-6}/year, which is the same as the current PTS acceptance criteria provided in Regulatory Guide (RG) 1.154.

A reactor vessel failure frequency of 1×10^{-5}/year based on consideration of the core damage frequency (CDF) provided in RG 1.174 and the Option 3 framework for risk-informing 10 CFR Part 50.

A reactor vessel failure frequency of 1×10^{-6}/year based on consideration of the RG 1.174 large early release frequency (LERF) that is a surrogate for the prompt fatality safety goal and on the Option 3 framework for risk-informing 10 CFR Part 50.

Because of the potentially severe challenge to containment integrity posed by reactor vessel failure resulting from PTS sequences, we believe that a risk-informed acceptance criterion for reactor vessel failure frequency should be based on considerations of LERF and not on CDF. However, the current LERF surrogate goal in RG 1.174 is not a proper starting point for developing an acceptance criterion because the source terms used to develop the current goal do not reflect the air-oxidation phenomena that would be a likely outcome of a PTS event.

There is currently no commonly accepted source term for air-oxidation events. However, we suggest that the "SST1" source term in NUREG/CR-2239 and the resulting calculated consequences at each site be extrapolated to assess the consequences of a postulated range of air-oxidation-induced source terms that would include significant releases of ruthenium, cerium, and actinides. Given such a source term, an acceptance criterion for the frequency of vessel failure from PTS events could be developed directly from the prompt fatality safety goal with due consideration of uncertainties and defense-in-depth.

If the consideration of an air-oxidation source term is too daunting and subject to unacceptable uncertainty, it may be necessary to fall back on a frequency-based approach to identify criteria that would provide assurance that reactor vessel failure from PTS events is very unlikely. The choice of such criteria is a value judgment that should reflect consideration of the Safety Goals and uncertainties.

We believe it is likely that qualitative consideration of the likelihood of containment failure along with the potential consequences of an air-oxidation source term will lead to an acceptance criterion for reactor vessel failure frequency that would be substantially smaller than any of those currently proposed by the staff.

Sincerely,

George E. Apostolakis
Chairman

References:

1. SECY-02-0092, Memorandum dated May 30, 2002, for the Commissioners, from William D. Travers, Executive Director for Operations, NRC, Subject: Status Report: Risk Metrics and Criteria for Pressurized Thermal Shock
2. U.S. Nuclear Regulatory Commission, NUREG/CR-2239, "Technical Guidance for Siting Criteria Development," December 1982.
3. U. S. Nuclear Regulatory Commission, Regulatory Guide 1.154, "Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors," January 1987.
4. U. S. Nuclear Regulatory Commission, Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," July 1998.
5. Letter dated February 14, 2002, from George E. Apostolakis, Chairman, ACRS, to William D. Travers, Executive Director for Operations, NRC, Subject: Reevaluation of the Technical Basis for the Pressurized Thermal Shock Rule.

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

July 23, 2002

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

Dear Chairman Meserve:

SUBJECT: DRAFT FINAL REVISION 1 TO REGULATORY GUIDE 1.174 AND TO CHAPTER 19 OF THE STANDARD REVIEW PLAN

During the 494th meeting of the Advisory Committee on Reactor Safeguards, July 10-12, 2002, we reviewed the draft final Revision 1 to Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," and to the Standard Review Plan (SRP), Chapter 19, "Use of Probabilistic Risk Assessment in Plant-Specific, Risk-Informed Decisionmaking: General Guidance." During this review, we had the benefit of discussions with representatives of the NRC staff. We also had the benefit of the documents referenced.

RECOMMENDATIONS

1. Revision 1 to RG 1.174 and the associated SRP Chapter 19 should not be issued until more substantive changes are made.

2. Both RG 1.174 and SRP Chapter 19 should emphasize that all sources of risk from internal and external initiators during low-power and shutdown (LPSD), as well as full-power, operations must be included in the risk assessment. If bounding estimates of the risk contribution from plant modes not rigorously analyzed are used, justification of the estimates should be provided.

3. RG 1.174 and SRP Chapter 19 should state that changes to the licensing basis will, in general, require probabilistic risk assessments (PRAs) that conform at least to Category 2 of the American Society of Mechanical Engineers (ASME) standard [and the comparable category of the future American Nuclear Society (ANS) standards for external events and LPSD operations] and a Grade 3 of the industry peer review process.

DISCUSSION

The publication of RG 1.174 and associated SRP Chapter 19 in 1998 was a major milestone in the NRC initiative to risk inform the regulations. RG 1.174 introduced the concept of an integrated decisionmaking process that had as inputs risk information, considerations of defense in depth, and sufficient safety margins. The Guide also defined acceptable ranges of values for the possible increases in core damage frequency (CDF) and large early release frequency (LERF) that could result from proposed changes in the licensing basis.

The approach to the use of risk information established in RG 1.174 was consistent with the philosophy of a risk-informed, rather than risk-based, regulatory system. As such, the Guide did not determine acceptance in terms of strict numerical values for ΔCDF and ΔLERF and did not specify how to integrate various inputs to the decisionmaking process. Also, the Guide stated that the scope, level of detail, and technical acceptability of the PRA should be commensurate with the application.

Because the explicit use of risk information from PRAs for regulatory purposes was relatively new in the late 1990s, the staff recognized in SRP Chapter 19 that licensees probably would not have a PRA that analyzed all significant plant modes and accident initiators. It was also recognized that the quality of available PRAs varied widely. Because risk information was to be utilized in an integrated decisionmaking process, it was accepted that such incomplete PRAs could still provide useful insights into the risk impact of proposed licensing changes, thus leading to more effective regulatory decisions.

Although this approach has been successful in the development of some risk-informed licensing changes, such as risk-informed inspection programs, it had the unfortunate consequence that it did not encourage the development of full-scope PRAs for all operational modes nor the use of rigorous methods in developing risk information.

Incomplete and less-than-rigorous PRAs undermine the credibility of the entire risk-informed regulatory process. We believe that the slow progress of risk-informed initiatives can be attributed, in part, to this lack of credibility.

The proposed revisions make no substantive changes to the existing RG 1.174, and publication of these revisions may send the wrong message that incomplete PRAs are acceptable for a broad range of risk-informed changes to the licensing basis. Therefore, these revisions should not be issued.

Revision to RG 1.174 and the associated SRP Chapter 19 should reflect the progress of the last five years, as exemplified by PRA standards that provide new consensus guidance for a high-quality PRA and the industry peer review process. In view of these developments, the revised guidance should specify that a licensee who wishes to take advantage of risk information produce such information using methods consistent with rigorous consensus standards and include all significant sources of risk from all plant modes. Such a provision would bolster the credibility of risk-informed decisions and reduce the staff and licensee effort required to assess risk implications of licensing changes.

The PRAs should include rigorous uncertainty analyses, at least for parameter uncertainties. This would allow more focused attention on those sources of uncertainty that are much more difficult to address, such as model uncertainty and incompleteness. When approximations (e.g., bounding estimates of the risk due to plant modes not rigorously analyzed) or approximate methods are used, they should be justified. Sensitivity analyses should be used judiciously and only after an uncertainty analysis has been performed. We look forward to working with the staff on these important issues.

Additional comments by ACRS Members Dana A. Powers and John D. Sieber are provided below.

Sincerely,

George E. Apostolakis
Chairman

Additional Comments by ACRS Members Dana A. Powers and John D. Sieber

Our colleagues have the laudable desire to encourage improvements in the scope and depth of probabilistic risk assessments being utilized by licensees especially when they seek risk-informed changes to licensing bases. Our colleagues are, however, seeking improvements with such demands for rigor and stringency, regardless of need, that they may alienate some applicants. Such demands will increase the burdens associated with RG 1.174 and SRP Chapter 19 for both applicants and the NRC staff.

References:
1. SECY-02-0070, memorandum dated April 24, 2002, for the Commissioners, from William D. Travers, Executive Director for Operations, NRC, Subject: Publication of Revisions 1 to Regulatory Guide 1.174 and SRP Chapter 19 and Notice of a Staff Plan for Endorsing Consensus Probabilistic Risk Assessment Standards and Industry Peer Review Programs, with attachments:
   - Regulatory Guide 1.174, Revision 1, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," April 2002.
   - Standard Review Plan, NUREG-0800, Chapter 19, Revision 1, "Use of Probabilistic Risk Assessment in Plant-Specific, Risk-Informed Decisionmaking: General Guidance," April 2002.
2. Letter dated June 14, 2000, from G. M. Eisenberg, ASME International, to M. Markley, ACRS staff, transmitting Draft 12 of Proposed ASME Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications, dated May 30, 2000.
3. Letter dated January 24, 2001, from Shawn M. Coyne-Nalbach, American Nuclear Society, to M. T. Markley, ACRS staff, transmitting Draft BSR/ANS-58.21, "External Events PRA Methodology Standard," dated December 25, 2000.

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

September 17, 2002

MEMORANDUM TO: Williams D. Travers
Executive Director for Operations

FROM: John T. Larkins
Executive Director, ACRS/ACNW

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

During the 495th meeting of the Advisory Committee on Reactor Safeguards, September 12-14, 2002, the Committee considered the Draft Regulatory Guide DG-1122, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities." The Committee plans to review the draft final version of DG-1122 after reconciliation of public comments. The Committee agrees with the staff's proposal to issue DG-1122 for public comment.

Reference:
U. S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Draft Regulatory Guide DG-1122, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," September 2002.

cc: A. Vietti-Cook, SECY
J. Craig, OEDO
I. Schoenfeld, OEDO
S. Collins, NRR
M. Crutchley, NRR
A. Thadani, RES
M. Cunningham, RES
S. Newberry, RES
C. Carpenter, RES
M. Drouin, RES

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

September 18, 2002

MEMORANDUM TO: Williams D. Travers
Executive Director for Operations

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

SUBJECT: PROPOSED RULEMAKING ON 10 CFR 50.69, DRAFT REGULATORY GUIDE DG-1121, and NEI 00-04

During the 495th meeting of the Advisory Committee on Reactor Safeguards, September 12-14, 2002, the Committee reviewed (1) proposed rulemaking on 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors;" (2) Draft Regulatory Guide DG-1121, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance;" and (3) the Nuclear Energy Institute document NEI 00-04 (Draft Revision C), "10 CFR 50.69 SSC [Structures, Systems, and Components] Categorization Guideline."

The ACRS agrees with the staff's proposal to issue the proposed 10 CFR 50.69 and DG-1121, which endorses NEI 00-04, for public comment. The Committee plans to continue to meet with the staff to discuss these documents as further progress is made.

References:
1. U.S. Nuclear Regulatory Commission, Draft Federal Register Notice, 10 CFR Part 50, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors," dated August 12, 2002 (Predecisional).
2. U.S. Nuclear Regulatory Commission, Draft Regulatory Guide DG-1121, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance," August 2002 (Predecisional).
3. U.S. Nuclear Regulatory Commission, Draft Regulatory Analysis for Proposed 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors," dated July 15, 2002 (Predecisional).
4. NEI 00-04 (Draft-Revision C), "10 CFR 50.69 SSC Categorization Guideline," June 2002.

cc: A. Vietti-Cook SECY
J. Craig, OEDO
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

September 24, 2002

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

Dear Dr. Travers:

SUBJECT: HUMAN FACTORS AND HUMAN RELIABILITY ANALYSIS RESEARCH PLANS

During the 495th meeting of the Advisory Committee on Reactor Safeguards, September 12-14, 2002, we discussed plans for research in the areas of Human Factors and Human Reliability Analysis with the staff of the Office of Nuclear Regulatory Research (RES). Our Subcommittee on Human Factors had explored these research plans during its meeting with the RES staff on September 10, 2002. We also had the benefit of the referenced documents.

CONCLUSION

RES research programs on Human Factors and Human Reliability Analysis are well directed toward meeting agency needs. These programs can be further refined by considering the following recommendations.

RECOMMENDATIONS

- The Human Reliability Analysis Program needs to articulate its long-term vision of the technology necessary to the agency. This vision should include the availability of a well-validated model for quantifying individual and team error rates.

- The past focus on overt, individual errors of omission is being augmented to include latent human errors and needs to be expanded to address explicitly team interactions both in the control room and elsewhere in the plant.

- Human Factors and Human Reliability Analysis research programs should be expanded to search for leading indicators of degradation in human performance, both at the individual and group levels.

- The NRC should consider development of a control room simulator devoted to support research on human factors and human reliability.

DISCUSSION

Consideration of human factors and the quantification of the reliability of human performance arise frequently in the safety analysis of nuclear power plants especially in this era in which risk quantification plays an important role in the regulatory process. It is likely that human factor and human reliability analysis will remain important issues even for advanced reactors that emphasize passively safe designs.

RES research programs on Human Factors and Human Reliability Analysis consist of a mix of applications of technologies to issues of rulemaking, licensing, and licensee monitoring as well as further development of these technologies. Important applications indicative of the ubiquity of Human Factors and Human Reliability Analysis include:

- worker fatigue,
- control room staffing at existing and advanced nuclear power plants,
- pressurized thermal shock,
- steam generator tube rupture,
- fire protection, and
- dry cask storage.

Our discussions of the Human Factors and Human Reliability Analysis research programs did not focus on the applications of the technologies developed in these programs. Our attentions were, instead, on the further development of the technologies. In both the Human Factors and Human Reliability Analysis programs, the emphasis now is on the collection and analysis of data to validate tools being provided to the agency. These programs are also involved in the generation of guidance for the use of the tools and guidance to support the inspections and reviews of submittals by licensees and applicants. These are valuable undertakings by the research programs.

Recent work in the Human Factors research program has shown that latent errors (errors made in the past, but not discovered until a plant event occurs) may be more risk significant than classically considered human errors of omission made in response to an event. The research program is now investigating latent errors further and how such errors may be treated in probabilistic risk assessments (PRAs).

The expansion of PRAs to treat latent errors should be accompanied by further expansions in the treatment of human performance at nuclear power plants. For example, quantitative treatment of team performance is needed. An analysis of team performance throughout the plant (e.g., maintenance and engineering teams) would supplement the more traditional emphasis on performance in the control room. Human performance analysis should include individual and team performance in the context of the entire plant organization.

Much attention is being given to the use of data collected in reactor simulators for the development and validation of models used in human reliability analysis. Such data may be quite valuable for these purposes. Indeed, it is not apparent that these data have been thoroughly mined. We note especially the possibility that data collected by licensees and vendors in the development of symptom-oriented procedures and reflected in plant-specific PRAs could be of use for model development and validation.

On the other hand, simulator data must be viewed skeptically as the basis for validating models of operator performance in a plant control room. For example, it is not evident that simulator performance of a cohesive team reflects the performance in the control room when one or more members of the team has been replaced because of sickness or vacation. This may occur up to one-third of the time. The NRC may need an explicit element of its research program to qualify simulator data for use in validating human reliability model validation. In pursuing this research, it should be recognized that in the simulator environment, the operator's concern with the potential negative consequences of operator actions is slight.

Simulator data now available from licensees address primarily human performance during design-basis events. Human performance during severe reactor accidents may be more pertinent to nuclear power plant risk assessments. One way for the RES research program to address this need for pertinent data is to have a simulator devoted to research. A highly flexible research simulator would be especially useful as human factors and human reliability at advanced nuclear power plants are explored by the research programs.

The NRC has developed its Reactor Oversight Program with the untested hypothesis that degradation of human performance will be detected in a timely way by degradation in plant performance indicators. We remain concerned that this hypothesis may be in error. Even if the hypothesis is accurate, the indication of degraded human performance will not be a leading indicator. The degraded human performance may be detected when the degradation has become significant. The NRC needs research to investigate the hypothesis concerning the detectability of degradation of human performance. The NRC may want to follow a program undertaken by the Electric Power Research Institute to find leading indicators of human performance degradation.

Human reliability analysis has not been a static field. Initial efforts to quantify the reliability of human performance focused on the time available for effective human action. With improved understanding, additional variables affecting human performance were identified, including stress, workload, training, quality of procedures, system feedback, and the man/machine interface. There has been a proliferation of models or methods to account for various subsets of these additional variables. There has not been a disciplined effort to review these models critically and develop a well-validated model that takes a comprehensive view of factors affecting human reliability. The NRC's research program should focus on the continuing development of a consistent, comprehensive methodology for the quantification of human performance reliability.

In conclusion, we found that the Human Factors and Human Reliability Analysis research programs to be well developed and providing important inputs to the regulatory processes. We look forward to continuing discussions with the staff on these programs as results are obtained.

Sincerely,

George E. Apostolakis
Chairman

References:

1. U.S. Nuclear Regulatory Commission, NUREG/IA-0137, International Agreement Report, "A Study of Control Room Staffing Levels for Advanced Reactors," November 2000.
2. U.S. Nuclear Regulatory Commission, NUREG/CR-6691 (BNL-NUREG-52600), "The Effects of Alarm Display, Processing, and Availability on Crew Performance," November 2000.
3. U.S. Nuclear Regulatory Commission, Policy Issue (Information) SECY-01-0196, Memo dated November 1, 2001, for the Commissioners, from William D. Travers, Executive Director for Operations, NRC, Subject: Status of the NRC Program on Human Performance in Nuclear Power Plant Safety.
4. U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, N. Siu, E. Thornbury, and M. Cunningham, "The NRC Human Reliability Analysis [HRA] Research Program," paper given at OECD/NEA/CSNI Workshop on HRA, May 7-9, 2001, in Rockville Maryland.

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

October 1, 2002

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

Dear Dr. Travers:

SUBJECT: DRAFT REGULATORY GUIDE DG-1120 AND STANDARD REVIEW PLAN SECTION 15.0.2 CONCERNING NRC REVIEWS OF TRANSIENT AND ACCIDENT ANALYSIS METHODS

During the 495th meeting of the Advisory Committee on Reactor Safeguards, September 12-14, 2002, we met with representatives of the NRC staff to discuss Draft Regulatory Guide DG-1120 (DG-1120), "Transient and Accident Analysis Methods," and draft final Standard Review Plan Section 15.0.2 (SRP 15.0.2), "Review of Transient and Accident Analysis Methods." Our Subcommittee on Thermal-Hydraulic Phenomena also reviewed these documents during a meeting on July 17, 2002. We also had the benefit of the documents referenced.

RECOMMENDATION

The Draft DG-1120 and SRP 15.0.2 should be issued for public comment after the minor differences between Section 5 of the Regulatory Guide and Section 6 of the Standard Review Plan Section have been reconciled.

DISCUSSION

The NRC staff has developed a Draft DG-1120 and SRP 15.0.2 to document a set of general principles and specific expectations applicable to both the form and content of applicants' code submittals, and the staff's review of those submittals. The staff undertook this effort in response to concerns identified by the NRC (Maine Yankee Lessons Learned Report) and the ACRS (review of the AP600 passive plant design).

Our Thermal-Hydraulic Phenomena Subcommittee held meetings with the NRC staff to discuss the status of its work in December 1998, November 1999, and April 2000. At that time, the Subcommittee concluded that, although the SRP 15.0.2 was ready to be issued for public comment, the accompanying Draft Regulatory Guide, then identified as DG-1096, needed substantial improvement. We reviewed revisions of both documents during our May 2000 meeting, and the documents were later issued for public comment. Subsequent to closure of the public comment period, the staff held a workshop with representatives of the nuclear industry. Based on concerns expressed by industry representatives pertaining to regulatory burden, the staff decided to make revisions to the Regulatory Guide. DG-1120 is the current revised version of DG-1096.

The major public comments concerned the degree to which the process described in DG-1096 applied to small changes in approved analysis methods. It was suggested that, for such changes, the extent and scope of the submission could be appropriately abridged.

In response, the staff has added a new Section 5 to the Regulatory Guide (now identified as DG-1120), describing a graded approach which specifies the extent to which the full Evaluation Model Development and Assessment Process may be reduced for a specific application. We agree with the proposed graded approach defined in the revised regulatory guide.

The rest of DG-1120 is substantially unchanged from the document that we previously reviewed and supported. We see no need to alter it. Several thermal-hydraulic codes are currently under review or will shortly be reviewed by the staff. The DG-1120 will be a useful reference document for applicants, the staff, and the ACRS. We look forward to its expeditious publication and implementation.

The Draft SRP 15.0.2 has also been modified. It is somewhat inconsistent with DG-1120. We have discussed these inconsistencies with the staff and they have agreed to reconcile these documents.

ACRS Member Graham M. Leitch did not participate in the Committee's deliberations on this matter.

Sincerely,

George E. Apostolakis
Chairman

References:

1. U.S. Nuclear Regulatory Commission, Draft Regulatory Guide DG-1120, "Transient and Accident Analysis Methods," dated June 2002.
2. U. S. Nuclear Regulatory Commission Draft Standard Review Plan Section 15.0.2, "Review of Transient and Accident Analysis Methods," December 2000.
3. Memorandum dated May 31, 2002, from Gary Holahan, Office of Nuclear Reactor Regulation, NRC, to Farouk Eltawila, Office of Nuclear Regulatory Research, NRC, Subject: Office of Nuclear Reactor Regulation Comments on Revisions to DG-1096 and Draft SRP Section 15.0.2.
4. Letter dated March 22, 1999, from Dana A. Powers, Chairman, ACRS, to William D. Travers, Executive Director for Operations, NRC, Subject: Lessons Learned from the ACRS Review of the AP600 design.
5. Letter dated October 7, 1996, from Shirley Ann Jackson, Chairman, NRC, to Honorable Angus King, Governor of Maine, transmitting U.S. Nuclear Regulatory Commission Report, "Independent Safety Assessment of Maine Yankee Atomic Power Company," October 1996.

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

October 17, 2002

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

Dear Chairman Meserve:

SUBJECT: CONFIRMATORY RESEARCH PROGRAM ON HIGH-BURNUP FUEL

During the 496th meeting of the Advisory Committee on Reactor Safeguards, October 10-12, 2002, we met with representatives of the NRC's Office of Nuclear Regulatory Research (RES) to discuss their confirmatory research program on high-burnup fuel, as well as research they do to support safety regulation of dry cask storage of spent fuel including high-burnup fuel. We also met with representatives of the NRC's Office of Nuclear Reactor Regulation to discuss their plans to review an EPRI topical report on the response of high-burnup fuel to reactivity insertion events. Our Subcommittee on Reactor Fuels met on October 9, 2002, to review these topics in detail and to discuss with representatives of EPRI their work to define fuel failure criteria and coolability criteria for high-burnup fuel exposed to reactivity transients. We also had the benefit of the referenced documents.

CONCLUSIONS

- RES has a well-organized and leveraged program of confirmatory research on the behavior of high-burnup fuel under the conditions of reactivity insertion events in pressurized water reactors, design-basis loss-of-coolant accidents (LOCAs), and anticipated transients without scram in boiling water reactors. RES has also undertaken research on creep of high-burnup fuel cladding to support safety regulation of dry cask fuel storage.

- A consensus has emerged that the energy input that will rupture fuel cladding in a reactivity insertion event is much less than that implied by the criteria in existing regulatory guides and decreases with increasing fuel burnup at least above 40 GWd/t.

- RES is nearing resolution of the issues of reactivity insertion events in high-burnup fuel and has initiated experimental investigations of high-burnup fuel under conditions of design-basis LOCAs. We remain concerned that the time-temperature conditions used in the study of high-burnup fuel during design-basis LOCAs may not reveal phenomena unique to high-burnup fuel.

DISCUSSION

There are economic and societal incentives to use nuclear fuel to higher levels of burnup. Burnup levels now approved exceed the data bases underlying the models that are used to predict fuel behavior under upset and design-basis accident conditions. French and Japanese tests of high-burnup fuel have shown cladding failure and even fuel dispersal during reactivity insertions at energy levels substantially below the criteria found in Regulatory Guide 1.77.

RES has undertaken a research program to confirm that the current limit on fuel burnup (62 GWd/t) ensures adequate protection of the public health and safety. A research program of experimental and analytic research involving the collaboration of NRC, EPRI, and numerous foreign partners has been organized. Risk-informed methods have been used to select issues of high-burnup fuel to investigate. The program addresses high-burnup fuel behavior under conditions of design-basis LOCAs and boiling water reactor anticipated transients without scram, as well as reactivity insertion events in pressurized water reactors.

RES has upgraded the fuel behavior computer code (FRAPTRAN) and neutron transport code (PARCS) available for regulatory analysis of high-burnup fuel. It has also completed detailed phenomena identification and ranking studies for high-burnup fuel under a variety of conditions. In addition, RES has participated with its foreign partners in the continued experimental study of reactivity transients in high-burnup fuel.

Analyses of data on high-burnup fuel behavior during reactivity transients have progressed in many quarters, including within the RES program and independently by EPRI. It is now broadly *accepted that the energy input necessary to fail fuel in a reactivity transient is much less than* the criterion in Regulatory Guide 1.77. At least for burnups greater than 40 GWd/t, the energy needed to fail fuel decreases with increasing fuel burnup. This sensitivity to sudden energy inputs is thought to be attributable to embrittlement of the fuel cladding. RES is also showing with realistic analyses that design-basis reactivity transients do not produce energy inputs of the magnitude and speed necessary to fail cladding embrittled at burnups less than 62 GWd/t.

RES anticipates that with the aid of 2 or 3 additional inpile tests in France's CABRI reactor and tests at elevated temperatures in Japan's NSRR, it will be able to quantitatively characterize the degradation of the capacity of fuel to sustain sudden energy inputs with increasing fuel burnup. RES is pursuing both empirical and mechanistic pathways to develop this characterization.

Controversy still exists within the reactor fuel community on whether distinct burnup-dependent criteria should be developed for fuel cladding rupture and for the energy input sufficient to cause loss of coolable configuration of the fuel. RES currently supports a single criterion for fuel failure that would also be conservative for coolability, whereas EPRI has proposed the continued use of separate criteria.

Recently, RES initiated out-of-pile tests of individual fuel rod segments under conditions of design-basis LOCAs. The objective is to replicate with high-burnup fuel the investigations of fresh cladding behavior that were the bases for the so-called "embrittlement" criteria specified in 10 CFR 50.46 and Appendix K. These tests involve monotonic heatup of fuel rods to a limiting temperature (2200°F) and monotonic cooling and quenching. We remain concerned that other safety-significant phenomena, such as spallation of pre-existing oxide from the cladding, may be important for high-burnup fuel and may be revealed only when more complicated, and realistic, time-temperature conditions are used in the tests. The single rod segment tests will not reveal features of high-burnup fuel behavior that arise when multiple rods are present.

The behavior of high-burnup fuel during anticipated transients without scram has become a topic of particular interest as power uprates have shortened the time available for operators to respond to the transients in boiling water reactors. Analyses done to date show that high energy inputs can occur during power oscillations in these transients, but the power inputs occur too slowly to produce intense pellet-clad mechanical interactions that threaten cladding integrity. This analytic finding appears to be substantiated by a recent test in Japan's NSRR.

We conclude that the confirmatory research program for high-burnup fuel is well-designed and is making good progress in light of the challenges of inpile and out-of-pile tests with fuel irradiated to high levels of burnup. We remain supportive of this program.

We recognize that this confirmatory research program is not addressing the risk consequences of taking fuel to high levels of burnup. These consequences will be examined in planned studies of high-burnup fuel in beyond design-basis accident conditions. We look forward to hearing about RES plans to explore this important aspect of high-burnup fuel in nuclear power plants.

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

Sincerely,

George E. Apostolakis
Chairman

References:

1. Memorandum dated July 6, 1998, from L. Joseph Callen, Executive Director for Operations, NRC, to the Commissioners, Subject: Agency Program Plan for High Burnup Fuel.
2. Nuclear Energy Institute, EPRI Report 1002865, "Topical Report on Reactivity Initiated Accidents: Bases for RIA Fuel Rod Failures and Core Coolability Criteria," June 12, 2002.
3. U. S. Nuclear Regulatory Commission, NUREG/CR-6742, "Phenomenon Identification and Ranking Tables (PIRTs) for Rod Ejection Accidents in Pressurized Water Reactors Containing High Burnup Fuel," August 2001.
4. U. S. Nuclear Regulatory Commission, NUREG/CR-6743, "Phenomenon Identification and Ranking Tables (PIRTs) for Power Oscillations Without Scram in Boiling Water Reactors Containing High Burnup Fuel," August 2001.
5. U. S. Nuclear Regulatory Commission, NUREG/CR-6744, "Phenomenon Identification and Ranking Tables (PIRTs) for Loss-of-Coolant Accidents in Pressurized and Boiling Water Reactors Containing High Burnup Fuel," August 2001.

6. U. S. Nuclear Regulatory Commission, NUREG/CR-6739, Vol. I, "FRAPTRAN: A Computer Code for the Transient Analysis of Oxide Fuel Rods," August 2001.
7. U. S. Atomic Energy Commission, Regulatory Guide 1.77, "Assumptions Used for Evaluating A Control Rod Ejection Accident for Pressurized Water Reactors," May 1974.
8. Letter dated March 14, 2002, from George E. Apostolakis, ACRS Chairman, to William D. Travers, Executive Director for Operations, NRC, Subject: Confirmatory Research Program on High-Burnup Fuel.

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

October 17, 2002

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

Dear Chairman Meserve:

SUBJECT: DRAFT REPORT "GUIDANCE FOR PERFORMANCE-BASED REGULATION"

During the 496th meeting, October 10-12, 2002, the Advisory Committee on Reactor Safeguards reviewed the draft report "Guidance for Performance-Based Regulation," dated August 2002. During this review, we had the benefit of discussions with representatives of the NRC staff and of the documents referenced.

In its Strategic Plan, the Commission states that it is a strategy of the NRC to employ less-prescriptive, performance-based regulatory approaches to maintain safety and reduce regulatory burden. This draft report is intended to provide guidance for developing performance-based options in regulatory decisionmaking, such as changes to regulatory guides.

We agree with the staff's proposal to publish this guidance as a NUREG/BR report. This guidance will be useful to the staff in developing performance-based alternatives to existing regulations. Before issuing the new guidance, however, the staff should provide more extensive discussion of safety margins and performance parameters.

The draft guidance declares a safety margin to be adequate, even when the performance objective is not met, provided that corrective action can be taken to avoid a serious condition. This is an essential attribute of performance-based regulation, and the implementers of the proposed guidance will face the issue of selecting the "serious condition." In the reactor arena, one can envision, for example, using the cornerstones of the reactor oversight process or core damage to define the serious conditions. This selection significantly affects the choice of the performance parameters. A discussion of the possible options in the three arenas of reactors, nuclear materials, and nuclear waste would be useful.

The selection of performance parameters and objective criteria for satisfactory performance intimately relates to the ease with which it can be demonstrated that the criteria have been met. There may be uncertainties in the estimation of the performance parameters that complicate this comparison. An acknowledgment of this fact, with some examples, would be useful.

We plan to continue to meet with the staff to discuss further progress on this important and difficult issue.

Sincerely,

George E. Apostolakis
Chairman

References:
1. U.S. Nuclear Regulatory Commission, Draft NUREG-BR-XXXX, "Guidance for Performance-Based Regulation," August 2002.
2. U.S. Nuclear Regulatory Commission, NUREG-1614, Vol. 2, Part 1, "Strategic Plan," Fiscal Year 2000-Fiscal Year 2005.
3. U.S. Nuclear Regulatory Commission, SECY-01-0205, memorandum dated November 16, 2001, from William D. Travers, Executive Director for Operations, NRC, for the Commissioners, Subject: Status Report on Performance-Based Approaches to Regulation.
4. Letter dated September 8, 2000, from Dana A. Powers, Chairman ACRS, to William D. Travers, Executive Director for Operations, Subject: Proposed High-Level Guidelines for Performance-Based Activities.

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

November 13, 2002

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

Dear Chairman Meserve:

SUBJECT: RECOMMENDATIONS PROPOSED BY THE OFFICE OF NUCLEAR REGULATORY RESEARCH FOR RESOLVING GENERIC SAFETY ISSUE-189, "SUSCEPTIBILITY OF ICE CONDENSER AND MARK III CONTAINMENTS TO EARLY FAILURE FROM HYDROGEN COMBUSTION DURING A SEVERE ACCIDENT"

During the 497th meeting of the Advisory Committee on Reactor Safeguards (ACRS), November 7-9, 2002, we reviewed the recommendations proposed by the Office of Nuclear Regulatory Research (RES) to resolve Generic Safety Issue (GSI)-189, "Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident." During this review, we had the benefit of discussions with the NRC staff and their contractors, as well as representatives from Duke Energy Corporation. This matter was also discussed during a meeting of the Thermal Hydraulic Phenomena and the Reliability and Probabilistic Risk Assessment Subcommittees on November 5, 2002. We also had the benefit of the documents referenced.

RECOMMENDATIONS

1. Features to resolve GSI-189 should be incorporated into affected plants through plant-specific severe accident mitigation guidelines (SAMGs).

2. The NRC staff should develop guidance on how uncertainties are to be evaluated and considered in regulatory analysis decisions.

DISCUSSION

To reduce the potential for containment failure as a result of detonation of hydrogen generated during severe accidents, ice condenser and Mark III containments are equipped with distributed igniters and air return fans that prevent stratification and enhance the condensing effectiveness of the ice compartment. For station blackout (SBO) events, neither preferred AC nor backup AC power provided by the emergency diesel generators would be available for the igniters and air return fans. Therefore, a potential resolution of this GSI includes the possible addition of a backup diesel generator to power either the igniters or a combination of igniters and air return fans. The addition of passive recombiners is also a consideration.

The RES study reevaluated the role of air return fans on ice condenser containment performance by updating the MELCOR code scoping calculations for the Sequoyah Nuclear Plant and reviewing previous evaluations. As a result of that study, RES concluded that (1) flow conditions inside the ice condenser region are not conducive to producing a transition to detonation, and (2) the operation of the fans merely shifts the burning of the hydrogen more towards the lower containment compartments. Consequently, the staff has concluded that air return fans are not needed to avoid detonation and that the igniters alone are sufficient. We accept this conclusion and the staff's further conclusion that the cost-benefit analysis need only consider backup power for the igniters or the addition of passive recombiners.

The RES staff has conducted a regulatory analysis. At the suggestion of the ACRS, this analysis included consideration of the associated uncertainties. The work scope, however, was not sufficient to conduct a full uncertainty analysis. The uncertainty information utilized for the benefits was estimated from the existing probabilistic risk assessments (PRAs) of relevant plant types.

The "point estimates" of the cost-benefit analysis indicated that only backup diesels for ice condenser plants would pass the benefit minus cost (B-C)[1] test and that the Mark III containments would fail the B-C test (although not by much).

Considering the uncertainties associated with both the costs and the benefits, the B-C for both containment types range from negative to positive, with a substantial amount of the uncertainty distribution on the positive side. The regulatory analysis guidelines should be implemented using the mean value of B-C, and this mean value should be determined in a technically defensible manner. Although the uncertainty estimates in the report were developed in a somewhat ad hoc manner, we believe that they are adequate for this analysis. The NRC staff should develop guidance on the appropriate estimation and treatment of uncertainties.

Although the cost-benefit conclusions were not decisive, considerations of defense-in-depth and public confidence have led the RES staff to conclude that further action by the Office of Nuclear Reactor Regulation is warranted for both containment types. We agree with this assessment, but feel that the justification is not sufficient to support the issuance of an order or a rule. Features to resolve GSI-189 should be incorporated into the appropriate plant-specific SAMGs to allow flexibility for licensees to consider plant-specific options.

Sincerely,

George E. Apostolakis
Chairman

[1]Commonly known as "Net Present Worth" or "Net Present Value."

References:
1. Memorandum dated October 11, 2002, from Farouk Eltawila, Office of Nuclear Regulatory Research, to John T. Larkins, Advisory Committee on Reactor Safeguards, Subject: RES Proposed Recommendation for Resolving Generic Safety Issue 189: "Susceptibility of Ice Condenser and MARK III Containments to Early Failure from Hydrogen Combustion During a Severe Accident."
2. Letter dated June 17, 2002, from George E. Apostolakis, Chairman, ACRS, to William D. Travers, Executive Director for Operations, NRC, Subject: Recommendations Proposed by the Office of Nuclear Regulatory Research for Resolving Generic Safety Issue-189, "Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident."

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

November 19, 2002

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: DRAFT REGULATORY GUIDE DG-1123, "VERIFICATION, VALIDATION, REVIEWS, AND AUDITS FOR DIGITAL COMPUTER SOFTWARE USED IN SAFETY SYSTEMS OF NUCLEAR POWER PLANTS"

During the 497th meeting of the Advisory Committee on Reactor Safeguards, November 7-9, 2002, the Committee considered Draft Regulatory Guide DG-1123 and decided to review the final version of DG-1123 after reconciliation of public comments. The Committee agrees with the staff's proposal to issue DG-1123 for public comment.

cc: A. Vietti-Cook, SECY
J. Craig, OEDO
I. Schoenfeld, OEDO
A. Thadani, RES
M. E. Mayfield, RES
S. Arndt, RES
E. Lee, RES

Reference:
Memorandum dated October 8, 2002, from Michael E. Mayfield, Office of Nuclear Regulatory Research, to John T. Larkins, ACRS, Subject: Request to Defer the ACRS Review of Draft Regulatory Guide DG-1123.

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

December 13, 2002

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

Dear Chairman Meserve:

SUBJECT: DRAFT COMMISSION PAPER ON POLICY ISSUES RELATED TO NON-LIGHT-WATER REACTOR DESIGNS

During the 498th meeting of the Advisory Committee on Reactor Safeguards, December 5-7, 2002, we met with representatives of the NRC's Office of Nuclear Regulatory Research (RES) to discuss the subject draft Commission paper. We also had the benefit of the referenced documents.

The RES staff has identified seven technical issues with policy implications that need resolution prior to certification reviews of advanced non-light-water reactor designs. The staff has also provided options for resolving those issues and has recommended the preferred options. We agree with the staff's recommended preferred options for each of the seven issues.

We commend the staff on its thoughtful study and look forward to further interactions on this subject.

Sincerely,

George E. Apostolakis
Chairman

References:

1. Draft Predecisional SECY paper, undated, from William D. Travers, Executive Director for Operations, NRC, to the Commission, Subject: Policy Issues Related to Licensing Non-Light-Water Reactor Designs.
2. U. S. Nuclear Regulatory Commission, NUREG-1226, "Development and Utilization of the NRC Policy Statement on the Regulation of Advanced Nuclear Power Plants."

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

December 18, 2002

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

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE LICENSE RENEWAL APPLICATIONS FOR THE NORTH ANNA POWER STATION UNITS 1 AND 2 AND THE SURRY POWER STATION UNITS 1 AND 2

Dear Chairman Meserve:

During the 498th meeting of the Advisory Committee on Reactor Safeguards, December 5-7, 2002, we completed our review of the License Renewal Application for North Anna Power Station (NAS) Units 1 and 2, the Surry Power Station (SPS) Units 1 and 2, and the final Safety Evaluation Report (SER) prepared by the staff of the U. S. Nuclear Regulatory Commission (NRC). Our review included a meeting of our Plant License Renewal Subcommittee on July 9, 2002. During our review, we had the benefit of discussions with representatives of the NRC staff and Virginia Electric and Power Company (Dominion). We also had the benefit of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

1. The Dominion application for renewal of the operating licenses for NAS Units 1 and 2 and SPS Units 1 and 2 should be approved.

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

BACKGROUND AND DISCUSSION

This report fulfills the requirement of 10 CFR 54.25 which states that the ACRS should review and report on license renewal applications. Dominion requested renewal of the operating licenses for NAS Units 1 and 2 and SPS Units 1 and 2 for a period of 20 years beyond the current license terms, which expire on April 1, 2018 (NAS Unit 1); August 21, 2020 (NAS Unit 2); May 25, 2012 (SPS Unit 1); and January 29, 2013 (SPS Unit 2). The final SER, issued on November 5, 2002, documents the results of the staff's review of information submitted by Dominion, including commitments that were necessary to resolve the open items identified by the staff in the initial SER. This review of the application was conducted concurrently for two stations with a total of four units. Given the similarity of the units and the formatting of the application, which clearly highlighted the few differences, the concurrent review did not present any unusual difficulties.

The staff reviewed the completeness of the identification of structures, systems, and components (SSCs) subject to aging management; the integrated plant assessment process; the applicant's identification of the possible aging mechanisms associated with passive, long-lived components; and the adequacy of the aging management programs. The staff also conducted three inspections. First, a 1-week inspection was performed to assess the applicant's scoping and screening methodology. Next a 1-week inspection was conducted at each facility to assess plant material condition and aging management programs. Lastly, an inspection was performed to close open items resulting from the earlier inspections.

The staff provided the Committee with details of the scope and results of its inspections of material condition at both plants. We agree with the staff's assessment that there are no issues that would preclude renewal of the operating license for NAS Units 1 and 2 and SPS Units 1 and 2.

On the basis of our review of the final SER, we agree that all open items and confirmatory items have been appropriately closed. We also discussed several items that were raised at the Subcommittee meeting on July 9, 2002, and found that the staff and the applicant have satisfactorily addressed each item.

The processes implemented by the applicant to identify SSCs that are within the scope of license renewal were effective. As with several previous applicants, the staff engaged in considerable discussion with the applicant regarding the portion of the offsite power system to be included within the scope of license renewal. After reviewing the information provided by the applicant, we agree that appropriate portions of the offsite power system are included in scope. During our review, we questioned why certain other SSCs were not included within the scope and, in all cases, the applicant provided appropriate justification for exclusion.

The applicant has performed a comprehensive aging management review of SSCs that are within the scope of license renewal. There are 19 existing aging management programs and four new programs.

The applicant has satisfactorily responded to NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity," dated March 18, 2002. Further, the applicant has committed to replace all four reactor vessel heads. The replacement of the NAS Unit 2 head is currently in progress.

The applicant used the guidance specified in Westinghouse Owners Group reports for reactor coolant system piping, pressurizer, and reactor internals. The staff reviewed and approved the use of these reports with certain stipulations. Each stipulation was sufficiently addressed in the staff's review.

We questioned the method by which reactor coolant piping is to be inspected in light of the failure of the initial volumetric inservice inspection to detect vessel nozzle cracking at V.C. Summer. Although continued improvement in the inspection methodology is warranted, the staff considers current methods adequate to detect primary water stress corrosion cracking. This is a generic issue and we remain concerned with the effectiveness of inspection techniques. Dominion has committed to employ best industry practices as they are developed.

Dominion has also committed to conduct a one-time inspection of a representative sample of buried piping. Opportunistic inspections of in-scope buried piping will be performed when the piping is uncovered during other maintenance activities. If significant degradation is identified, the results will be entered into the licensee's corrective action program and the inspection will be expanded. If no opportunity presents itself by the end of the current license period, excavations will be made to inspect the piping.

The applicant's erosion/corrosion program is of particular interest in light of the previous carbon steel piping failures at SPS. Dominion uses the CHECWORKS program to identify locations to be monitored and trend erosion/corrosion rates. The program appears to be effective in managing erosion/corrosion.

Certain medium-voltage cables exposed to moisture for long periods of time fail due to a phenomenon called "water treeing." To preclude this failure, the applicant has committed to a program that will control water in manholes and underground ducts associated with energized power cables. The Cable Monitoring Activities Program for non-environmentally qualified cable has been enhanced to ensure that if degraded cable is identified, the cable environment, including the potential for moisture shall be evaluated and appropriate corrective actions initiated through the corrective action program.

During the discussion of time-limited aging analyses, we expressed a concern that the applicant had not submitted its evaluations of the reactor vessel margins for pressurized thermal shock and upper shelf energy. The staff had accepted the applicant's position that these values were acceptable without performing an independent evaluation. Subsequently, the staff obtained this information from the applicant and the staff performed an independent evaluation. Although in some cases the margins are small, we agree with the staff's position that margin does exist. We believe that in the future such critical parameters should be reviewed by the staff. The staff agreed to require that these data be provided with future license renewal applications.

In several situations, Dominion and other applicants have committed to actions based on future technology development. In Dominion's case, two examples are (1) the method for inspecting reactor coolant piping, and (2) the method for testing of medium-voltage cables exposed to moisture. The NRC staff needs to continue to keep abreast of these developing technologies and review and approve methodologies at the appropriate time.

License renewal applications include a number of activities and commitments, for example one-time inspections, that will not be accomplished until near the end of the current license period. There is a large amount of inspection activity that needs to be conducted at that time period. The staff is aware of this future work load and is working on a plan to properly manage this significant effort.

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

Sincerely,

George E. Apostolakis

George E. Apostolakis
Chairman

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

December 20, 2002

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

Dear Chairman Meserve:

SUBJECT: FRAMATOME ANP S-RELAP5 REALISTIC LARGE-BREAK LOSS-OF-COOLANT ACCIDENT CODE

During the 498th meeting of the Advisory Committee on Reactor Safeguards, December 5-7, 2002, we met with representatives of Framatome ANP and the NRC staff to review the Framatome ANP S-RELAP5 realistic large-break loss-of-coolant accident (LOCA) methodology and its application to pressurized-water reactor (PWR) accident analyses. Our Subcommittee on Thermal-Hydraulic Phenomena reviewed this matter during meetings held on January 16-18 and November 12-14, 2002. We also had the benefit of the documents referenced.

RECOMMENDATIONS

1. The S-RELAP5 code should be approved for application to realistic large-break LOCA analyses.

2. The staff should confirm that zirconium oxide spallation during a LOCA is not a significant phenomenon that needs to be modeled in realistic codes.

3. The staff should continue to accept the treatment of the break size as a statistical variable.

4. Future submittals of this code should include:

   - Improved documentation that can be more readily understood by technically knowledgeable reviewers
   - Assessment of the sensitivity of code predictions to terms in the momentum equations
   - Comprehensive nodalization studies

5. The staff should investigate ways to facilitate updating of the computer platforms on which approved codes can be run.

6. The staff should make independent audit calculations as part of the assessment of vendor codes. This will be facilitated when the TRAC-M code becomes operational.

DISCUSSION

Framatome ANP has developed a realistic or "best estimate" version of its large-break LOCA code, S-RELAP5. The code is based on the MOD2 and MOD3 versions of the NRC RELAP5 code. A realistic version of the code employs analytical models that more accurately describe the physics and reduces the need for conservative assumptions. Use of a realistic code requires an estimate of the uncertainty in the calculated results, as specified by a 1988 revision to the Emergency Core Cooling System Rule. Framatome has elected to follow the basic approach specified in the Code Scaling, Applicability, and Uncertainty (CSAU) Methodology and has chosen to employ the non-parametric order statistics methodology.

As part of its analysis of uncertainties, Framatome performed a comprehensive sensitivity analysis of the influence of parameters in the code. Those that proved to be important were included in a probabilistic analysis to determine with 95 percent confidence that the predicted Peak Cladding Temperature (PCT) would lie within the 95th percentile of possible values. Framatome also showed that the other evaluation criteria, the degree of nodal clad oxidation and total maximum clad oxidation, would be satisfied with high probability. In performing this analysis, Framatome treated the break size as a statistical variable. We consider this novel approach appropriate because the "worst" break size is itself dependent on the particular choice of all of the other statistical parameters in the analysis.

The S-RELAP5 code is developed from the RELAP5 code, which has a history of thirty years of evolution and application. The staff has accepted the basis of the code and has performed an extensive review of its technical details, such as correlations for fluid mechanical phenomena and heat transfer. In some instances, the staff examined and evaluated parts of the source code itself. The staff also made independent assessments of the code using data from the 2D/3D LOCA test program and performed parametric studies of the effect of wall drag coefficients on the predicted PCT in a PWR. The staff also compared the code predictions against selected FLECHT-SEASET data. Their assessment of the code confirmed that it is acceptable for calculations of PCT following a large-break LOCA.

Although we support the staff's assessment of S-RELAP5 for the large-break LOCA scenario, we continue to have difficulty understanding some features of the code from the documentation provided. While experience shows that the code works effectively, its theoretical basis and functional implementation should be made clear. Framatome has reassured us that this will be improved in later editions of the documentation, which we expect to review in association with future submittals. In particular, the development of the momentum equations is unclear and incomplete. Applications of these equations to nodes that are not components of one-dimensional ducts introduce significant distortion of the actual physics, which is reflected in approximations to several terms in the equations.

The success of the code in predicting integral system test results indicates that, for the purposes of large-break LOCA analyses, the PCT is insensitive to these approximations. However, this may not be true for other applications or other evaluation criteria. One way to assess whether the momentum equation in its present form is adequate is to apply multipliers, or correction factors, to the various terms, such as the inertia and momentum flux terms, to reflect the uncertainties that are known to exist in the physical modeling. Sensitivity studies can then be performed to see if these correction factors matter. If they do, then more precise models, or an appropriate statistical treatment of these uncertainties, may be required for some purposes.

The results obtained from codes are known to depend on the particular nodalization that is employed. For this reason, the shapes, numbers, and physical modeling of nodes are treated as an experimental process until an arrangement is found that satisfactorily predicts a chosen database. The nodalization is then frozen for application to a nuclear system. This prevents assessment of possible scaling effects that might be nodalization dependent. It is contrary to practice in other areas of computational fluid dynamics where nodalization of an actual system is systematically varied until adequate convergence is demonstrated. The present strategy is based on arguments of computational complexity and requirements for computer time that should no longer be valid. It would be more convincing to both the technically informed public and the users of the code if this convergence were to be more explicitly demonstrated. The staff should require such demonstrations in the future.

In the move to reduce conservatisms, it is important that the bounding of omitted, but pertinent, phenomena not be lost. In examining realistic LOCA models, the staff should ensure that the bounding of omitted phenomena has been retained, or that the previously omitted phenomena are now included in the analysis. An example that was revealed during our review of this application is the effect of oxide spallation and thermal shock on the kinetics of clad oxidation during a LOCA. Conservative Baker-Just oxidation kinetics (in some imperfect way) bound these unmodeled processes, whereas more realistic Cathcart-Pawel kinetics may not, particularly for high burnup fuel. The staff needs to confirm that the evaluation criteria are still met when these additional processes are evaluated, along with more realistic descriptions of the reaction kinetics.

We were surprised to learn that a major impediment to more extensive and thorough testing of the code is the difficulty of transferring it to up-to-date computer systems. The source of this difficulty is the quality assurance requirement for licensing codes imposed by Appendix B to Title 10, Part 50, of the *Code of Federal Regulations*. The staff should investigate ways in which this transfer can be facilitated.

The NRC has been developing the TRAC-M code by synthesizing and improving the capabilities of existing codes. The staff needs its own computer code in order to perform an independent audit of results obtained from proprietary vendor codes. TRAC-M was not used in this capacity to assess S-RELAP5 because it is not yet fully operational. There is a pressing need for TRAC-M to be made available, and for the staff to acquire experience with it, so that it can be routinely used for such purposes.

Sincerely,

George E. Apostolakis
Chairman

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4. Report from Virgil E. Schrock, ACRS Consultant, dated November 19, 2002, regarding Consultant Report on Subcommittee Meeting, November 12-14, 2002.
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6. Memorandum dated October 12, 2002, from Fred J. Moody, ACRS Consultant, "Several Questions Pertaining to the Use of S-RELAP5."
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December 20, 2002

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

Dear Dr. Travers:

SUBJECT: DRAFT FINAL AMERICAN NUCLEAR SOCIETY EXTERNAL EVENTS PROBABILISTIC RISK ASSESSMENT METHODOLOGY STANDARD

During the 498th meeting of the Advisory Committee on Reactor Safeguards, December 5-7, 2002, we met with Dr. R. Budnitz, Chairman of the American Nuclear Society (ANS) External Events Working Group, to discuss the draft final ANS External Events Probabilistic Risk Assessment (PRA) Methodology Standard (BSR/ANS 58.21).

CONCLUSIONS

- The draft final ANS External Events PRA Methodology Standard adds to the standards available to assist in preparing PRAs for nuclear power plants.
- The Standard defines requirements for three capability categories of external events PRAs that differ in terms of their level of resolution, conservatism, and use of site-specific data.
- The Standard does not address the issue of seismically induced fires. ANS is currently working on a standard for fire PRA. The interface between the fire PRA and external events PRA will need attention.

DISCUSSION

The NRC is moving toward greater use of risk information in the regulation of nuclear plants, and that move is creating growing demands for high-quality PRAs. Specifically, these demands are for risk assessments dealing with internal events, external events, events initiated by fire, and events during low-power and shutdown operations. The usual methods for ensuring quality for engineering analyses such as risk assessments are checking compliance with recognized standards and peer reviews.

The American Society of Mechanical Engineers (ASME) has already developed a Standard for internal events. A draft standard for external events has recently been prepared by ANS and is currently undergoing the usual approval processes established by ANS and the American National Standards Institute (ANSI). An essential part of the standard is peer review.

The ANS Standard is consistent with the ASME Standard for internal events. Indeed, the ANS Standard for external events presumes the availability of a risk assessment of internal events. The ANS Standard defines requirements for three PRA "capability categories" that differ in terms of their level of resolution, conservatism, and use of plant-specific data.

Requirements for each of the capability categories are quite similar and even identical in many cases. Consequently, the requirements require interpretation to ascertain the capability category. Little guidance is provided on how to interpret these requirements in terms of PRA capability. Standards for PRAs take the form of guidance and not prescriptive analytic methods typical of other engineering standards. It is unlikely, therefore, that completely reproducible peer review evaluations of the capability categories could be derived from the ANS Standard or any other PRA standard.

The ANS Standard addresses seismic events, high-wind events, and external-flooding events. Much of the Standard is devoted to seismic events which is appropriate. The Standard does not address seismically induced fires. Such fires could be significant risk contributors and must be considered in risk assessments needed to support risk-informed regulation concerning external events. ANS is currently working on a standard for fire PRA. The interface between the fire PRA and external events PRA will need attention.

The ANS Standard includes materials describing and setting requirements for the seismic margins method of plant analysis. It is unclear why this non-probabilistic method is addressed in a PRA standard. On the other hand, the material on the seismic margins method does not detract from the material on probabilistic methods.

We congratulate the authors on taking a good step toward developing an External Events PRA Methodology Standard and especially for their attention to the treatment of uncertainty.

Sincerely,

George E. Apostolakis
Chairman

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

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG-1125, Volume 24

2. TITLE AND SUBTITLE

A Compilation of Reports of the Advisory Committee on Reactor Safeguards: 2002 Annual

3. DATE REPORT PUBLISHED

MONTH	YEAR
June	2003

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

6. TYPE OF REPORT

Compilation

7. PERIOD COVERED *(Inclusive Dates)*

Jan. thru Dec. 2002

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)*

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

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

13. AVAILABILITY STATEMENT

Unlimited

14. SECURITY CLASSIFICATION

(This Page)
Unclassified

(This Report)
Unclassified

15. NUMBER OF PAGES

16. PRICE

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, DC 20555-0001

OFFICIAL BUSINESS
PENALTY FOR PRIVATE USE, $300