NUREG-1125
Volume 26

A Compilation of Reports of

The Advisory Committee on Reactor Safeguards

2004 Annual

U. S. Nuclear Regulatory Commission

June 2005

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ABSTRACT

This compilation contains 47 Advisory Committee on Reactor Safeguards reports submitted to the U. S. Nuclear Regulatory Commission (NRC), or to the NRC Executive Director for Operations, during calendar year 2004. In addition, a report to the Commission on the NRC Safety Research Program, NUREG-1635, Volume 6, is included by reference only. All reports have been made available to the public through the NRC Public Document Room, the U. S. Library of Congress, and the Internet at http://www.nrc.gov/reading-rm/doc-collections. The reports are organized in chronological order.

PREFACE

The enclosed reports, issued during calendar year 2004, contain the recommendations and comments of the U. S. NRC's Advisory Committee on Reactor Safeguards on various regulatory matters. NUREG-1125 is published annually. Previous issues are as follows:

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

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Dr. William J. Shack
Argonne National Laboratory

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

February 9, 2004

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: ERRATA SHEET FOR REGULATORY GUIDES

The Advisory Committee on Nuclear Waste and the Advisory Committee on Reactor Safeguards have considered the minor corrections as noted on the errata sheet to the following Regulatory Guides and decided not to review them:

(1) Regulatory Guide 1.184, "Decommissioning of Nuclear Power Reactors"

(2) Regulatory Guide 1.185, "Standard Format and Content for Post-Shutdown Decommissioning Activities Report"

The Committees have no objection to the staff's proposal to issue these errata sheets for publication and recognizes there is no public comment period contemplated for these errata, since four public meetings were held in 2000 in conjunction with development of the 2002 NUREG revision.

Reference:
Memorandum dated January 12, 2004, from Daniel M. Gillen, NMSS to John T. Larkins, ACRS, Subject: Request for Approval to Publish Errata Sheet for Regulatory Guides 1.184 and 1.185.

cc: A. Vietti-Cook, SECY
W. Dean, OEDO
I. Schoenfeld, OEDO
D. Weaver, OEDO
A. Thadani, RES
A. Beranek, RES
J. Mitchell, RES
M. Virgilio, NMSS
J. Greeves, NNSS
D. Gillen, NMSS
T. Smith, NMSS
P. Justice, NMSS
M. Crutchley, NRR

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

February 12, 2004

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

Dear Dr. Travers:

SUBJECT: DRAFT SAFETY EVALUATION REPORT FOR THE ECONOMIC AND SIMPLIFIED BOILING WATER REACTOR (ESBWR) PRE-APPLICATION REVIEW

During the 509th meeting of the Advisory Committee on Reactor Safeguards February 5-6, 2004, we met with representatives of the NRC staff and General Electric Nuclear Energy (GENE) to discuss the draft Safety Evaluation Report for the ESBWR pre-application review (Ref. 1). Our Subcommittee on Thermal-Hydraulic Phenomena reviewed this matter during its meeting on January 14-15, 2004. We also had the benefit of the documents referenced.

CONCLUSION

We agree with the staff's decision to approve the TRACG code for use in analysis of the ESBWR in response to a LOCA scenario.

OBSERVATIONS

The SER is well-written and provides summaries of specific features of TRACG followed by conclusions regarding their acceptability.

Compared with previous SERs that we have reviewed, the staff has improved its explanation of why some features of the code are acceptable, and some are not. However, the amount of substantive technical information contained in the SER is limited. Presentations by GENE and the staff of additional quantitative evidence, and our own review of many supporting proprietary documents were necessary to provide the Committee with additional supporting evidence for the staff's conclusions.

Some of the supporting evidence is proprietary, and it is difficult to provide this evidence in the SER. This is unfortunate from the point of view of public confidence. Although we have determined that the staff's conclusions are appropriate, an outside observer would have insufficient evidence to appreciate why they were made. A clear exposition of the process, criteria, and evidence by which the staff reached its decisions would improve the transparency of the agency's decisionmaking process. We were pleased to see that both the staff and GENE were able to present some of this evidence in the open meeting of the ACRS on February 5, 2004. We hope that the staff can continue to work with the industry to identify similar ways to include more technical supporting information in SERs without compromising sensitive intellectual property.

The staff's conclusion of acceptability is a judgement. The quality of this judgement depends on the skill, experience, and diligence of the staff. It would help in future decisionmaking if clear criteria for acceptability were established and articulated. These might include, for example, a measure of when the experimental evidence is sufficient to establish uncertainties in the relevant parameters in the code and in the resulting measures of success, with a prescribed level of confidence. We note that this issue has been addressed partially by a regulatory guide issued in 1989 (Ref. 2) that contains criteria for assessing the performance of best-estimate LOCA computer codes. However, two additional general guidance documents that build on Ref. 2 have been available for over 3 years in draft form as a draft regulatory guide and an associated Standard Review Plan section. We are disappointed to learn that they still have not been completed. (Refs. 3 and 4).

The staff's ability to examine the source code, run TRACG on its own computers, challenge the assumptions, run sensitivity analyses, and compare its predictions with those of the NRC's TRACE code played a large role in convincing us that approval should be given for the use of TRACG to analyze the ESBWR. We believe that these practices should continue.

GENE's extensive and successful comparisons of TRACG predictions with data from a number of experiments at different scales were additional evidence that helped to give us confidence in the decision even though some simplifying assumptions in the code were not fully justified a priori.

FURTHER USE OF TRACG FOR ANALYZING THE ESBWR

The only decision that is being made at this time is to approve the future use of TRACG in the design certification process for the analysis of LOCA scenarios for the ESBWR. When TRACG is used for design certification purposes, attention should be given to the following:

1. The arguments for "conservative" assumptions in the condensation processes in the suppression pool and in the Passive Containment Cooling System and Isolation Condenser performance should be made more explicit and quantified, or demonstrated to be bounding.

2. The assumptions about various mixing processes and noncondensable hideout in the drywell should be made more specific and quantified in such a way that it is clear that the full range of possibilities is covered.

3. Assumptions about operator actions should be justified and shown to cover all relevant actions that can reasonably be expected.

4. The analysis of uncertainty by adding up the effects of two-sigma variations around the base values is not a satisfactory approach, although margins are so large that we are confident that regulatory requirements can be met. Some measure of confidence in the predicted uncertainties in key parameters, such as water level above the core, should be established. There is perhaps a potential for simplifying this process if a convincing bounding analysis can be developed for one or more of these parameters to demonstrate that large margins are available between the calculated values and the associated regulatory limits.

5. Scaling should be addressed comprehensively, including an evaluation of the sources of error. The staff described a scaling approach that appears to do this (Ref. 5). We look forward to seeing how the staff applies this approach when it considers other aspects of the ESBWR during the design certification review.

6. The effects of entrance length on the flow regimes in the "chimney" should be quantified.

7. The predicted leak rate of the vacuum breakers is much lower than experience with current vacuum breakers would suggest. The staff needs to obtain adequate assurance that the predictions of leakage for this new vacuum breaker design are realistic and are based on a sufficient range of test results simulating likely operational history and conditions.

8. The explanations of level tracking and its robustness should be made more complete, including confirmation calculations to show that the correct hydrostatic head is predicted under all important conditions and that spurious "levels" or other features do not emerge.

9. The TRACG containment calculations sometimes showed significant deviations from corresponding predictions using the NRC's CONTAIN code. It is not sufficient to argue that these deviations are "conservative." The differences are large enough to require explanation in terms of physical phenomena and the development of quantitative estimates of the magnitudes in order to verify that the hypothesized explanations are correct.

We appreciate the professional, forthright, and responsive interactions that we have had with both GENE and the staff during this evaluation.

Dr. Peter Ford did not participate in the formulation of this letter.

Sincerely,

Mario V. Bonaca
Chairman

<u>References</u>:

1. Memorandum dated December 11, 2003, from James E. Lyons, NRR to John T. Larkins, ACRS, Subject: Draft Safety Evaluation for the ESBWR Pre-Application Review
2. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance," May 1989
3. U.S. Nuclear Regulatory Commission, Draft Regulatory Guide DG-1120, "Transient and Accident Analysis Methods," December 2002
4. U.S. Nuclear Regulatory Commission, Standard Review Plan Section 15.0.2, NUREG-0800, "Review of Transient and Accident Analysis Methods," January 2003
5. M. DiMarzo, "<u>A Simplified Model of the BWR Depressurization Transient</u>," Nuclear Engineering and Design, 205 (2001), pgs. 107-114, July 28, 2000

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

February 26, 2004

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

SUBJECT: REVIEW AND EVALUATION OF THE NRC SAFETY RESEARCH PROGRAM

Dear Chairman Diaz:

Attached is an advance copy of the 2004 ACRS report entitled, "Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program." This report presents the Committee's observations and recommendations concerning the NRC Safety Research Program. The final report will be published as NUREG-1635, Vol. 6.

This report focuses on that portion of the NRC research program dealing with the safety of existing nuclear reactors and advanced light water reactor designs, AP1000 and ESBWR, submitted for certification. In its review of the NRC research activities, the Committee considered the programmatic justification for the research as well as the technical approach and progress of the work. This review attempts to identify research crucial to the NRC mission. This review also attempts to identify research activities that have made valuable contributions to the agency mission in the past, but now have reached the point where additional research is not needed for efficient and effective safety regulation. The report does not address research on the vulnerability of existing nuclear power plants to acts of sabotage and terrorism.

As agreed to by the Commission, the ACRS will provide its next report to the Commission on the overall NRC Safety Research Program in March 2006.

Sincerely,

Mario V. Bonaca
Chairman

Attachment: As stated

[Final Version Attached]

<Attachment Included by Reference Only>

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

March 17, 2004

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

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE LICENSE RENEWAL APPLICATION FOR THE VIRGIL C. SUMMER NUCLEAR STATION

Dear Chairman Diaz:

During the 510th meeting of the Advisory Committee on Reactor Safeguards, March 3-6, 2004, we completed our review of the License Renewal Application (LRA) for the Virgil C. Summer Nuclear Station and the related Safety Evaluation Report (SER) prepared by the staff of the U. S. Nuclear Regulatory Commission (NRC). Our Plant License Renewal Subcommittee reviewed both the LRA and the staff's draft SER during a meeting on December 3, 2003. During these reviews, we had the benefit of discussions with the NRC staff and representatives of South Carolina Electric and Gas Company (SCE&G), the applicant. We also had the benefit of the documents referenced.

CONCLUSION AND RECOMMENDATION

1. The programs instituted and committed to by SCE&G to manage age-related degradation are appropriate and provide reasonable assurance that the Virgil C. Summer Nuclear Station can be operated in accordance with its current licensing basis for the period of extended operation without undue risk to the health and safety of the public.

2. The SCE&G application for renewal of the operating license for Virgil C. Summer Nuclear Station should be approved.

BACKGROUND AND DISCUSSION

This report fulfills the requirements of 10 CFR 54.25, which states that the ACRS should review and report on all license renewal applications. SCE&G prepared its application in accordance with NUREG-1801, "The Generic Aging Lessons Learned (GALL) Report." In that LRA, SCE&G requested renewal of the operating license for the V.C. Summer Nuclear Station for a period of 20 years beyond the current license term, which expires on August 6, 2022. The plant is a single unit, Westinghouse-designed, three-loop, pressurized-water reactor rated at 2,900 megawatts-thermal (MWt) with replacement steam generators that were installed in 1994.

The staff's initial SER did not include any open or confirmatory items, so the staff was able to expedite issuance of the final SER. Consequently, to accommodate the staff's accelerated schedule, we advanced our final review of this matter by 2 months.

The final SER documents the results of the staff's review of the information submitted by the applicant and identified during onsite NRC inspections and audits. In particular, the staff reviewed the completeness of the applicant's identification of structures, systems, and components that are within the scope of license renewal, the integrated plant assessment process, the applicant's identification of the plausible aging mechanisms associated with passive long-lived components, the adequacy of the applicant's aging management programs (AMPs), and the identification and assessment of time limited aging analyses (TLAAs) that required review.

During our review, we also discussed the effectiveness of existing programs that the applicant has established to deal with significant equipment degradation issues identified by operating experience.

In 2001, for example, the applicant identified a through-wall crack in the "A" hot leg to vessel nozzle weld. The root cause of that crack was high residual stresses resulting from weld repairs performed during plant construction. To address the problem, the applicant installed a new spool piece to replace the segment containing the defective weld. The applicant also inspected the "B" and "C" hot leg to vessel nozzle welds using ultrasonic methods, supplemented by eddy current testing. A recordable indication was detected in the "B" nozzle weld. To address the problem, the applicant applied a mechanical stress improvement process to the "B" and the "C" hot leg nozzle welds. This process introduced compressive stresses on the inside of the pipe which has inhibited flaw propagation.

Since earlier ultrasonic testing failed to identify the "A" hot leg to vessel nozzle weld defect before it propagated completely through the pipe wall, we questioned the effectiveness of the applicant's Alloy 600 AMP for managing primary water stress corrosion cracking (PWSCC) in ASME Class 1 dissimilar welds (e.g., Alloy 82/182 welds). The applicant stated that it continues to take advantage of improvements in ultrasonic testing methods and is now using the latest ultrasonic technology. Furthermore, the applicant has committed to incorporate emerging regulatory requirements and industry recommendations into its Alloy 600 program prior to the period of extended operation. We found the applicant's commitment acceptable.

The applicant has also conducted inspections of the upper and lower reactor vessel heads in accordance with the current NRC Bulletins and Orders. These inspections have not revealed any visible indications of leakage on the upper head. The applicant plans to perform bare metal inspections during the upcoming refueling outage. The plant is ranked as a "low-susceptibility" plant with regard to vessel head penetration cracking phenomena and the applicant currently has no plans to replace the upper head. Inspections of the reactor vessel bottom head revealed traces of boron. The applicant believes that the source of this boron is the aforementioned leak in the "A" hot leg to vessel nozzle weld. Consequently, the applicant cleaned the lower head to establish a fresh surface baseline for future inspections.

The ultimate heat sink for the plant is a lake created by a series of safety grade dikes and dams which are included in the scope of license renewal. The Service Water Pump House, which takes suction from this lake, experienced major settlement at the time of original construction. However, recent data indicate that no further settlement is occurring. This structure is also within the scope of license renewal. The Service Water Structures Survey Monitoring Program monitors the pump house and the intake structure for movement, cracking, settlement, and structural degradation. We concur with the staff's assessment that the attributes of this plant-specific program are appropriate and sufficient.

The SER describes the groundwater at the plant site as mildly acidic, with a pH slightly below 5.5. Therefore, the groundwater is considered aggressive in the SER, even though measured chloride and sulfate concentrations are extremely low. Although recent analyses of the groundwater performed by the applicant at five new wells indicate that the groundwater pH may actually be above 5.8, the applicant has committed to enhance the existing plant programs and procedures that manage potential aging effects on concrete structures. Therefore, the staff has reasonable assurance that the applicant can effectively maintain the concrete plant structures throughout the period of extended operation. The applicant also asserted that the inspection of a nearby 70-year-old hydroelectric plant with similar concrete exposed to similar groundwater has revealed no signs of degradation.

During its review of the LRA, the staff evaluated 42 aging management programs which include 26 existing programs and 16 new programs. Several of the new programs are not yet developed. As with other applicants, we encouraged SCE&G to establish a schedule for the implementation of these commitments well ahead of the beginning of the license renewal period, so as not to place an unreasonable demand on applicant and NRC resources.

We concur with the staff's assessment that the applicant has appropriately evaluated the plant TLAAs. For metal fatigue, three limiting reactor coolant system components were identified that could potentially exceed the design basis fatigue cumulative usage factor during the period of extended operation. Specifically, those components are the normal and alternate charging nozzles and the pressurizer surge line nozzle. The applicant has committed to track thermal transients on these three nozzles and will perform additional evaluations of these components prior to the period of extended operation. Any component with a projected cumulative usage factor exceeding the established limits will be either re-evaluated or replaced prior to entering the license renewal period. Additional independent calculations performed by the staff have confirmed that the plant reactor vessel is qualified to operate until the end of the license renewal period without exceeding established reactor vessel neutron embrittlement limits.

On the basis of our review of the LRA, the final SER, and the NRC's inspection and audit reports, we agree that there are no issues specifically related to the matters described in 10 CFR 54.29(a)(1) and (a)(2) which preclude renewal of the plant's operating license. The programs instituted and committed to by SCE&G to manage age-related degradation are appropriate and provide reasonable assurance that the plant can be operated in accordance with its current licensing basis for the period of extended operation without undue risk to the health and safety of the public. The SCE&G application for renewal of the operating license for the V.C. Summer Nuclear Station should be approved.

Sincerely,

Mario V. Bonaca

Mario V. Bonaca
Chairman

References:

1. U. S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the License Renewal of the Virgil C. Summer Nuclear Power Station," January 2004.
2. South Carolina Electric and Gas Company, "License Renewal Application for Virgil C. Summer Nuclear Power Station," August 6, 2002.
3. U. S. Nuclear Regulatory Commission, "Draft Safety Evaluation Report Related to the License Renewal of the Virgil C. Summer Nuclear Power Station," October 2003.
4. NRC Inspection Report 50-395/03-007, Scoping and Screening, June 13, 2003.
5. NRC Inspection Report 50-395/03-008, Aging Management Review, September 29, 2003.
6. U. S. Nuclear Regulatory Commission, "Aging Management Program Audit Report," October 9, 2003.

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

March 17, 2004

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

SUBJECT: ACRS REVIEWS OF THE WESTINGHOUSE ELECTRIC COMPANY APPLICATION FOR CERTIFICATION OF THE AP1000 PLANT DESIGN-INTERIM LETTER

Dear Dr. Travers:

During the 510th meeting of the Advisory Committee on Reactor Safeguards, March 3-6, 2004, we met with representatives of the NRC staff and Westinghouse Electric Company to discuss the status of the open items identified in the staff's draft safety evaluation report (DSER) as well as issues previously raised by the ACRS. During our review, we discussed this matter during our Full Committee meetings on November 7, 2002, April 11, 2003, and October 1, 2003. In addition, our Reliability and Probabilistic Risk Assessment Subcommittee held a meeting on January 23-24, 2003; our Thermal-Hydraulic Phenomena Subcommittee held meetings on March 19-20, 2003, July 16-17, 2003, and February 10-11, 2004; and our Future Plant Designs Subcommittee held a meeting on July 17-18, 2003 to discuss the technical aspects of the AP1000 design. Our reviews have not addressed security matters and their impact on the. AP1000 design. We also had the benefit of the documents referenced.

There are several areas in which we have comments related to the certification of the AP1000 reactor design. These are listed below by subject matter.

Draft SER and Design-Basis Compliance

The NRC staff is conducting a thorough design-basis review. We will continue to monitor the progress of the staff's review of the following issues.

Issue 1 – Automatic Depressurization System (ADS)-4 Squib Valve Function: The most important safety function in the AP-1000 design is the automatic depressurization of the primary system. We have had discussions on the performance characteristics of the ADS-4 Squib valves. We agree with the staff that inspections, test, analyses, and acceptance criteria (ITAAC) should be used to assure that the combined license (COL) inspection and testing program verifies that these valves meet the design-basis specifications.

Issue 2 – Assurance of Long-Term Cooling (Strainer Blockage): The AP1000 appears to incorporate a robust design to prevent sump screen blockage. The design utilizes screen areas slightly larger than those of current pressurized water reactors (PWRs); locates the screens higher off the floor with a flow guard overhead; uses deeper water levels; uses much lower recirculation flow rates and consequent lower flow velocities approaching and entering the screen; and uses reflective insulation and high density non-safety coatings. Since the issue of ensuring long-term cooling is still under regulatory discussion, we recommend that the AP1000 design for this be the subject of ITAAC to ensure that it complies with the generic regulatory resolution of this issue.

Codes and Validation Testing

We believe that the database used to validate the Westinghouse suite of codes for design-basis assessment of the AP600 design has generally been shown to apply adequately for AP1000 design. We previously identified issues related to liquid entrainment in the upper plenum and the ADS-4 takeoff line. These have been addressed to our satisfaction.

During the early phase of the limiting small-break loss-of-coolant accident (SBLOCA), namely the double-ended direct vessel injection (DEDVI) break, the Westinghouse NOTRUMP code underpredicts the core average void fraction compared to APEX test results. The reasons for this difference between code predictions and test results are understood. The staff has concluded that 10 CFR Part 50, Appendix K acceptance criteria are conservatively met and that the erroneous predictions do not significantly propagate into the long-term cooling phase. We agree with the staff's assessment that the AP1000 meets the Appendix K requirements.

Neither the Westinghouse code NOTRUMP nor the staff's RELAP5 code proved adequate for accurately modeling certain important phenomena, such as liquid entrainment from the core, deentrainment and entrainment in the upper plenum, entrainment into the ADS-4 line and pressure drop in the ADS-4 line. There were also disagreements between the two codes in the prediction of the level swell and the collapsed liquid level preceding the time of in-containment refueling water storage tank (IRWST) injection.

The approach taken by Westinghouse to resolve these issues was to perform code predictions with bounding assumptions and to make confirmatory hand calculations showing that in all cases the core would be adequately cooled. The staff performed independent calculations and sensitivity studies to help guide their assessment of these predictions.

While these approaches have resolved these technical concerns for AP1000 design, they illustrate a need for awareness of situations where additional work may be needed in addition to a set of code predictions. We commend the staff for its thoroughness in pursuing these matters and resolving them.

Issue 3 – Code Deficiencies: When deficiencies such as these are identified in codes, they should lead to the consideration of research programs to correct the weaknesses and avoid resorting to a patchwork of ad hoc methods.

Issue 4 – Range of Pi-Group Values: We have yet to be shown a sufficient technical justification that a range of 0.5 to 2.0 for various scaling Pi-groups represents general adequate scaling. We, therefore, recommend that the staff undertake confirmatory research on pertinent scaling issues for relating test facilities to prototypic systems and verify that the Pi group range of 0.5 to 2.0 is appropriate.

Probabilistic Risk Assessment (PRA)

We have judged that the AP1000 PRA quality is sufficient for design iteration and certification purposes.

Materials

Several items of concern relating to materials degradation were identified during our reviews. These ranged from quality assurance (QA) criteria for Alloy 52/152 weldments, to fracture toughness of high chromium nickel-base alloys under specific operating conditions, to the stress corrosion resistance of some alloys currently regarded as immune to such failure. The applicant believes the best alloys have been selected for these applications based on currently available information. Ongoing and future studies may suggest material and environmental changes that will be addressed at the COL stage.

Severe Accidents

The ACRS and the staff have questioned the technical justification for the aerosol removal coefficient (lambda) for containment. This issue has been addressed by Westinghouse using the STARNAUA code for the limiting sequence. We understand that the staff is using the MELCOR code to calculate the time dependence. We look forward to reviewing the staff's analysis.

Issue 5 – In-Vessel Retention/Fuel-Coolant Interactions (FCI): The assessment of in-vessel retention has not included exothermic intermetallic reactions which have been shown by some prototypic experiments to be important. If these factors are properly accounted for, the associated energetics of any resulting ex-vessel steam explosions are likely to be greater than has been currently evaluated. We would like to review the FCI models used and see additional justification that the initial conditions related to intermetallic reactions will not give rise to an energetic FCI that could fail containment.

Issue 6 – Organic Iodine Production: The acidification of containment water as a result of radiolysis of organic material could give rise to significant airborne fission product iodine in gaseous organic form. We need to review how Westinghouse and the staff have dealt with this potential.

Issue 7 – There is experimental evidence that a free-standing steel containment can fail in a catastrophic manner when its failure pressure is exceeded. Such a failure mode can lead to very rapid depressurization and, potentially, to resuspension of fission products that have been previously deposited or settled out. While the surrounding concrete structure of the AP1000 design may impede such a catastrophic depressurization, we would, nevertheless, like to see a sensitivity study on the fission product source term to assess the potential maximum effect on the risk of latent fatalities as compared to the Safety Goal.

We look forward to reviewing the final draft SER and the resolution of any open items before we conclude our review of the AP1000 design.

Sincerely,

Mario V. Bonaca

Mario V. Bonaca
Chairman

References:

1. Letter from George E. Apostolakis, Chairman, to Richard A. Meserve, Chairman, NRC, Subject: Phase 2 Pre-Application Review for AP1000 Passive Plant Design, March 14, 2002.
2. Letter from James E. Lyons, NRR, to W.E. Cummins, Westinghouse, Subject: Applicability of AP600 Standard Plant Design Analysis Codes, Test Program and Exemptions to the AP1000 Standard Plant Design, March 25, 2002.
3. U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, "Draft Safety Evaluation Report, Related to Certification of the AP1000 Standard Design," June 16, 2003.
4. Westinghouse AP1000 Design Control Document (DCD), APP-GW-GL-700, Tier 2 Information, June 2003.
5. Memorandum from William D. Travers, Executive Director for Operations, to the Commissioners, SECY-03-0113, Subject: Semiannual Update of the Status of New Reactor Licensing Activities, July 7, 2003.

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

March 18, 2004

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

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE LICENSE RENEWAL APPLICATION FOR THE H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT 2

Dear Chairman Diaz:

During the 510th meeting of the Advisory Committee on Reactor Safeguards, March 3-6, 2004, we completed our review of the License Renewal Application (LRA) for the H. B. Robinson Steam Electric Plant, Unit 2, known as Robinson Nuclear Plant, and the related final Safety Evaluation Report (SER) prepared by the NRC staff. Our Plant License Renewal Subcommittee reviewed this application and the staff's initial SER during a meeting on September 30, 2003. During these reviews, we had the benefit of discussions with representatives of the NRC staff and the Carolina Power and Light Company (CP&L). We also had the benefit of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

1. The programs instituted and committed to by CP&L to manage age-related degradation are appropriate and provide reasonable assurance that the Robinson Nuclear Plant can be operated in accordance with its current licensing basis for the period of extended operation without undue risk to the health and safety of the public.

2. The CP&L application for renewal of the operating license for Robinson Nuclear Plant should be approved.

BACKGROUND AND DISCUSSION

This report fulfills the requirements of 10 CFR 54.25, which states that the ACRS should review and report on all license renewal applications. In its application, CP&L requested renewal of the operating license for the Robinson Nuclear Plant for 20 years beyond the current license term, which expires on July 31, 2010. Robinson Nuclear Plant is a Westinghouse-designed, three-loop, pressurized-water reactor rated at 2,339 megawatts-thermal (MWt) with replacement steam generators installed in 1984. It is located adjacent to Unit 1 of the H.B. Robinson Steam Electric Plant, a coal fired steam power plant. The LRA was prepared in accordance with NUREG 1801, The Generic Aging Lessons Learned Report.

The Robinson Nuclear Plant final SER documents the results of the staff's review of the information submitted by the applicant, including commitments that were necessary to resolve open and confirmatory items identified by the staff in the initial SER and those identified during onsite NRC inspections and audits. In particular, the staff reviewed the completeness of the applicant's identification of structures, systems, and components that are within the scope of license renewal, the integrated plant assessment process, the identification of the plausible aging mechanisms associated with passive long-lived components, the adequacy of the aging management programs, and the identification and assessment of time limited aging analyses (TLAAs) requiring review.

Several design features that are unique to Robinson Nuclear Plant, such as grouted tendons, containment liner insulation, and some shared systems with a fossil unit, were identified. All shared systems are included in the scope of the LRA.

Robinson Nuclear Plant site has aggressive ground water due to a low pH. The applicant has committed to inspect the dam spillway and the intake structures every 10-years and will also perform opportunistic inspections of inaccessible concrete structures.

The pressurizer spray head is not in scope and, given its importance for cooldown, we questioned its omission. The applicant responded that the accident-basis analysis for plant operation does not include pressurizer spray so its exclusion is permissible. The applicant further stated that degradation of the nozzle would be noticed during normal operation.

The applicant stated that the plant has 37 existing aging management programs, of which 27 have been enhanced, and 10 new programs have been added. Several of these programs have yet to be developed and they will require NRC approval. As with other applicants, we encouraged CP&L to establish a schedule for implementing these commitments well ahead of the beginning of the license renewal period so as not to place an unreasonable demand on both the applicant and NRC resources. CP&L has committed to have 18 of these programs in place by mid 2004.

Time limited aging analyses were performed by the applicant to evaluate reactor vessel neutron embrittlement, metal fatigue for certain components, environmental qualification, grouted concrete containment tendon prestress, boraflex aging, and foundation pile corrosion. All these issues have been resolved satisfactorily. In the case of reactor vessel neutron embrittlement, the staff performed independent calculations and found the applicant's analysis acceptable.

We agree with the staff's conclusion that all open and confirmatory items have been closed appropriately. We conclude that on the basis of our review of the final SER, the LRA, and the NRC inspection and audit reports, there are no issues, specifically related to the matters described in 10 CFR 54.29(a)(1) and (a)(2), that preclude renewal of the operating license for the plant. The programs instituted and committed to by CP&L to manage age-related degradation are appropriate and provide reasonable assurance that the plant can be operated in accordance with its current licensing basis for the period of extended operation without undue risk to the health and safety of the public. The CP&L application for renewal of the operating license for the Robinson Nuclear Plant should be approved.

Sincerely,

Mario V. Bonaca

Mario V. Bonaca
Chairman

References:

1. U.S. Nuclear Regulatory Commission, "Final Safety Evaluation Report Related to H. B. Robinson Steam Electric Plant, Unit 2," January 2004.
2. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report with open items Related to the License Renewal of the H. B. Robinson Steam Electric Plant, Unit 2," August 2003.
3. Letter from J. W. Moyer, Carolina Power and Light Company, to the U.S. Nuclear Regulatory Commission, Subject: Application for Renewal of Operating License, H. B. Robinson Steam Electric Plant, Unit 2, June 14, 2002.
4. NRC Inspection Report 50-261/03-08, H.B. Robinson Steam Electric Plant, May 8, 2003.
5. NRC Inspection Report 50-261/03-09, H.B. Robinson Steam Electric Plant, July 31, 2003.

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

March 18, 2004

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

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

Dear Dr. Travers:

Thank you for your letter of December 22, 2003, regarding the staff's response to our September 30, 2003 letter to Chairman Diaz concerning Draft Final Revision 3 to Regulatory Guide (RG) 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident." We have reviewed your response and would like to meet with you or your representatives to discuss a number of issues:

- The extent of the technical challenges faced by both the industry and the NRC
- The limitations of the present knowledge base, and plans to address these limitations
- The maturity of the technical content of the RG
- The technical basis for the RG, including particularly the role of the knowledge base described in NUREG/CR-6808, "Knowledge Base for the Effect of Debris on Pressurized Water Reactor Emergency Core Cooling Sump Performance"
- The adequacy of the guidance that the Nuclear Energy Institute (NEI) provided in the draft guidance document that is currently under review by the NRC staff
- The extent to which the staff should use risk information in resolving the issue of water sources for long-term recirculation cooling following a loss-of-coolant accident
- The possible need for alternative solutions to the issue of sump strainer clogging, should the uncertainties associated with the calculational methodology be large, or the strainers prove to be unduly susceptible to debris blockage

- The staff's specific plans to resolve the technical issues associated with Generic Safety Issue 191 - "Assessment of Debris Accumulation of PWR Sump Pump Performance"

Sincerely,

Mario V. Bonaca
Chairman

References:

1. U.S. Nuclear Regulatory Commission, Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident, Draft Regulatory Guide 1.82, Revision 3, August 2003.
2. Rao, D.V., et al., Knowledge Base for the Effect of Debris on Pressurized Water Reactor Emergency Core Cooling Sump Performance, NUREG/CR-6808, LA-UR-03-880, Los Alamos National Laboratory, February 2003.

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

April 20, 2004

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

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

During the 511th meeting of the Advisory Committee on Reactor Safeguards, April 15 - 17, 2004, the Committee considered the draft final amendments to 10 CFR Part 50, Appendix E Relating to Emergency Action Levels and Exercise Requirements for Co-Located Licensees and decided not to review these amendments. The Committee agrees with the staff's proposal to publish the final amendments to the rule for industry use.

Reference:
Memorandum dated April 13, 2004, from David B. Matthews, NRR to John T. Larkins, ACRS, Subject: ACRS Review of Final Amendments to 10 CFR 50, Appendix E Relating to (1) Emergency Action Levels, Paragraph IV.B and (2) Exercise Requirements for Co-Located Licensees, Paragraph IV.F.2.

cc: A. Vietti-Cook, SECY
W. Dean, OEDO
D. Weaver, OEDO
A Thadani, RES
J. Dyer, NRR
D. Matthews, NRR
M. Jamgochian, NRR

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

April 22, 2004

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

SUBJECT: OPTIONS AND RECOMMENDATIONS FOR POLICY ISSUES RELATED TO LICENSING NON-LIGHT WATER REACTOR DESIGNS

Dear Chairman Diaz:

During the 511th meeting of the Advisory Committee on Reactor Safeguards, April 15-17, 2004, we met with representatives of the NRC staff to discuss proposed options and recommendations for policy issues related to licensing non-light water reactor (LWR) designs. We also had the benefit of the documents referenced.

OBSERVATIONS AND RECOMMENDATIONS

1. The Quantitative Health Objectives (QHOs) apply to the site as a whole. The sum of the contributions from each reactor on the site to acute and latent fatalities should be bounded by the QHOs.

2. The Committee has not reached consensus on the approach that should be taken to determine the core damage frequency (CDF) goal. Two views are presented in the discussion below.

DISCUSSION

In a June 26, 2003 Staff Requirements Memorandum (SRM), the Commission provided direction to the staff on two policy issues related to licensing non-LWR designs:

- "The staff should provide further details on the options for, and associated impacts of, requiring that modular reactor designs account for the integrated risk posed by multiple reactors."

- "...The staff should pursue the development of functional performance standards [for non-LWR containment functional performance] and then submit options and recommendations to the Commission."

In our meeting with the staff, we discussed the staff's response to this SRM. The development of functional performance standards is part of the development of a technology-neutral regulatory system. Therefore, in this report, we comment only on the issue of integrated risk. The staff has indicated that it plans to recommend to the Commission an option to treat integrated risk in assessing modular reactor designs as follows:

- "Taking into consideration the integrated effect of risk when assessing accident prevention for modular reactor designs, independent of reactor power level, and

- Taking into consideration the integrated effect of risk when assessing accident mitigation for modular reactor designs in a fashion that allows for consideration of reactor power level."

We agree with the staff that the QHOs are intended to protect the population around a site from the risk due to all reactors and spent fuel storage facilities on that site. Meeting the site QHOs will depend on the site's total number of plants and their design and power levels and the resulting risk will be the sum of risks from all contributors.

The issue of integrated risk has ramifications for the technology-neutral regulatory framework that is being developed by the staff. We understand that the preliminary proposal for this framework is to develop frequency-consequence (F-C) risk goals as a surrogate for the QHOs and to also deal with higher frequency incidents. Once again, an "achieved" F-C value would be a site characteristic and would depend not only on meteorology and population but on the design, number of units, and power levels at the site. For comparison with any goal, the composite F-C risk goals for the site will be required. The staff should keep this in mind in developing the technology-neutral framework.

With respect to the accident prevention (CDF) goal, we understand the staff's option to mean that, for any new modular plant, the CDF goal (e.g., 10^{-4} per reactor year) will be divided by the number of modules to arrive at a goal for each module.

The Committee has two differing views on the approach to determining a CDF goal for modular plants.

One view supports the staff's recommendation. A core damage accident is a very undesirable event even if it involves no adverse health consequences. Consequently, it is unacceptable that one site has a CDF of 10^{-5} per reactor year while another site with 20 modular reactors has a CDF of 20×10^{-5} per reactor year. The risk from and the likelihood of a core damage accident at all sites cannot be precisely equal. However, there is the expectation that they be comparable. The staff's recommendation addresses this expectation.

The alternative view is that CDF is an accident prevention goal and its value should be the same for each reactor at every site. A CDF goal should depend on the total number of reactors nationwide (not the number on a site). Requiring each module to have a CDF value given by the overall CDF goal divided by the number of modules introduces a new Safety Goal concept, a site CDF. Such a concept was never intended to be part of the Safety Goals.

The intent of a CDF goal has always been twofold: (1) to limit the chances of having an accident anywhere in the country over the projected lifetime of the plants, and (2) to serve as a defense-in-depth measure that balances accident prevention and mitigation for any given design. The extension of this concept to a site CDF goal is going far beyond the original intent.

Although the concept of a site CDF goal has some merit, it is fraught with so many troublesome issues that some ACRS members believe it is untenable. Some of these issues are:

- It introduces a new safety goal that likely will supercede the latent fatality Safety Goal.
- It tends to undermine the early site permit concept in that it could severely limit the number of acceptable sites.
- It will redefine the intended appropriate balance between prevention and mitigation for a given design.
- It is at cross purposes with the certification process which certifies a reactor design independent of a site.
- This concept, if adopted, would tend to lead to a lack of regulatory coherence and stability.
- Licensees desiring to use a specific site must choose a reactor design to accommodate the eventual projected maximum number of modules irrespective of the fact that these modules will be added much later in time and actually may never get to the maximum. A licensee would select a large (non-modular) plant that has a relatively high CDF to ensure meeting the site CDF goal rather than having a modular design that has to render each module to such a low CDF that the determination via PRA is suspect and difficult to achieve.
- With a universal CDF limit that is site independent, the population around any site would already have an acceptably low chance of seeing an accident at that site. There is no compelling reason to further limit the CDF value for the site.

Sincerely,

Mario V. Bonaca

Mario V. Bonaca

References:
1. Draft SECY Paper (PREDECISIONAL) from William D. Travers, EDO, to the Commissioners, Subject: Response to the June 26, 2003, Staff Requirements Memorandum on Policy Issues Related to Licensing Non-Light Water Reactor Designs, April 2004.
2. Staff Requirements Memorandum (SRM) from Annette L. Vietti-Cook, Secretary, to William D. Travers, EDO, SECY-03-0047, Subject: Policy Issues Related to Licensing Non-Light-Water Reactor Designs, dated June 26, 2004.
3. Report dated December 13, 2002, from George E. Apostolakis, Chairman, ACRS, to Richard A. Meserve, Chairman, NRC, Subject: Draft Commission Paper on Policy Issues Related to Non-Light-Water Reactor Designs.

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

April 23, 2004

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

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE LICENSE RENEWAL APPLICATION FOR THE R. E. GINNA NUCLEAR POWER PLANT

Dear Chairman Diaz:

During the 511th meeting of the Advisory Committee on Reactor Safeguards, April 15 -17, 2004, we completed our review of the License Renewal Application (LRA) for the R. E. Ginna Nuclear Power Plant and the related final Safety Evaluation Report (SER) prepared by the NRC staff. Our Plant License Renewal Subcommittee reviewed both the LRA and the staff's SER with Open Items during a meeting on November 4, 2003. During these reviews, we had the benefit of discussions with representatives of the NRC staff and the Rochester Gas and Electric Company (RG&E). We also had the benefit of the documents referenced.

CONCLUSION AND RECOMMENDATION

1. The programs instituted and committed to by RG&E to manage age-related degradation are appropriate and provide reasonable assurance that the R. E. Ginna Nuclear Power Plant can be operated in accordance with its current licensing basis for the period of extended operation without undue risk to the health and safety of the public.

2. The RG&E application for renewal of the operating license for R. E. Ginna Nuclear Power Plant should be approved.

BACKGROUND AND DISCUSSION

This report fulfills the requirements of 10 CFR 54.25, which states that the ACRS should review and report on all license renewal applications. The Ginna plant is a single unit, Westinghouse-designed, two-loop, pressurized-water reactor (PWR) rated at 1520 megawatts-thermal (MWt). RG&E prepared its application in accordance with NUREG-1801, "The Generic Aging Lessons Learned (GALL) Report." In that LRA, RG&E requested renewal of the operating license for the plant for a period of 20 years beyond the current license term, which expires on September 18, 2009.

The final SER documents the results of the staff's review of the information submitted by the applicant or identified during the several inspections conducted at the plant site. In particular, the staff reviewed the completeness of the applicant's identification of structures, systems, and components that are within the scope of license renewal; the integrated plant assessment process; the applicant's identification of the plausible aging mechanisms associated with passive long lived components; the adequacy of the applicant's aging management programs; and the identification and assessment of time limited aging analyses (TLAAs) requiring review.

The Ginna plant is the oldest PWR currently in operation in the U.S. The plant was constructed prior to the establishment of 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants." The plant was therefore subjected to the Systematic Evaluation Program (SEP) to determine if possible changes in the plant and licensing commitments would be required. The applicant indicated that since Ginna was an SEP plant, RG&E made early use of risk insights to evaluate cost beneficial plant modifications. During the scoping and screening process, unique plant-specific design features were identified, such as a standby auxiliary feedwater system and grouted containment tendons. Significant improvements have been made to the plant, including replacement of the steam generators in 1996 and the reactor vessel head in 2003.

The applicant has developed 33 aging management programs to manage aging degradation at Ginna during the period of extended operation. Of these, 20 are consistent with GALL, 10 are consistent with GALL with minor exceptions, and 3 are plant-specific programs for periodic surveillance and preventive maintenance, thimble tube inspection, and systems monitoring. The applicant has indicated that 31 license renewal commitments have been incorporated into the Ginna Corrective Action Tracking System. Less than half of these commitments have been completed. The remaining activities are scheduled for completion prior to the period of extended operation. We encouraged RG&E to establish a schedule for implementing these commitments well ahead of the beginning of the license renewal period so as not to place an unreasonable demand on both the applicant and NRC resources.

During our review, we also discussed the effectiveness of existing programs that the applicant has established to deal with significant equipment degradation issues identified by operating experience.

Ginna has conducted inspections of upper and lower reactor vessel heads in accordance with the current NRC bulletins and orders. No leakage was identified on the upper head. Nonetheless, the head was replaced in 2003. The new head has control rod drive mechanism penetrations fabricated using Alloy 690 TT and J-groove welds utilizing Alloy 52. The reactor vessel insulation is designed to allow full bare metal visual inspections of the upper and lower heads. The lower head penetrations were visually inspected in 2003 and there were no indications of leakage. These penetrations are expected to have low susceptibility to primary-water stress-corrosion-cracking due to the relatively low reactor operating temperature and the low residual stresses. The lower head was also cleaned to facilitate future inspections.

In response to NRC Bulletin 2003-01, Ginna inspected the containment sump screens and identified a gap in the "B" sump screen that could have allowed some debris to bypass the screen. The gap was repaired. The applicant plans to assess debris generation and transport following the Nuclear Energy Institute guidelines and, if necessary, make appropriate changes to the facility to satisfy NRC requirements.

We concur with the staff that TLAAs have been evaluated appropriately by the applicant. Independent calculations performed by the staff confirm that the Ginna reactor vessel will be able to operate throughout the period of extended operation with adequate margin to reactor vessel neutron embrittlement limits. For metal fatigue, no components have projected cumulative usage factors that exceed the design basis limits for the period of extended operation. All other TLAAs show that components evaluated will maintain acceptable margin to respective limits for the period of extended operation.

On the basis of our review of the LRA, the final SER, and the NRC's inspection and audit reports, we agree with the staff that there are no issues, specifically related to the matters described in 10 CFR 54.29(a)(1) and (a)(2), preclude renewal of the plant's operating license. The programs instituted and committed to by RG&E to manage age-related degradation are appropriate and provide reasonable assurance that the plant can be operated in accordance with its current licensing basis for the period of extended operation without undue risk to the health and safety of the public. The RG&E application for renewal of the operating license for the R. E. Ginna Nuclear Power Plant should be approved.

Sincerely,

Mario V. Bonaca
Chairman

References:

1. U. S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the License Renewal of the R. E. Ginna Nuclear Power Plant," February 2004.
2. Rochester Gas and Electric Company, "License Renewal Application for R. E. Ginna Nuclear Power Plant," August 6, 2002.
3. U. S. Nuclear Regulatory Commission, "Safety Evaluation Report with Open Items Related to the License Renewal of the R. E. Ginna Nuclear Power Plant," October 9, 2003.
4. NRC Inspection Report 50-244/03-010, Scoping and Screening Methodology, August 22, 2003.
5. NRC Inspection Report 50-244/03-008, Aging Management Program, December 2, 2003.
6. U. S. Nuclear Regulatory Commission, "Aging Management Program Audit Report," September 8, 2003.

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

April 26, 2004

Mr. Ashok C. Thadani
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: PROPOSED APPROACH TO ASSESS THE QUALITY OF NRC RESEARCH PROJECTS

Dear Mr. Thadani:

We are pleased to assist the Office of Nuclear Regulatory Research (RES) management by providing assessments of the quality of selected research projects. Quality assessment of individual research projects as requested by RES will constitute a new undertaking for us; one that is quite different in scope and depth in comparison to our biennial review of the NRC research activities. Because this activity is new to us, a proposed strategy is provided below for conducting the reviews. This strategy is to be tried during FY 2004 and refined in FY 2005. Elements of the strategy are:

- The RES management will provide a list of eight candidate projects in the beginning of each fiscal year. Projects on this list should address a range of topics and scope, and may have a range of maturities up to and including completed projects. No project should be included on the list that does not have a documented work plan.

- We will select four of the listed projects for review at the rate of one project per quarter. For the remainder of FY 2004, only two projects will be reviewed.

- A panel of nominally three ACRS members will be formed for each of the projects selected for review. This panel will consist typically of an experienced chairman, a member with special expertise in the general area of the research program, and one other ACRS member.

- The panel will meet with the project staff only after all relevant documents have been assembled and the panel has had the opportunity to examine these documents. The amount of documentation will depend on the maturity of the project, but should include at least a project plan.

- The panel will conduct its detailed review of a project, prepare a report and defend its assessments of the project before the full Committee. The defense of the assessments by the panel before the full Committee is intended to ensure consistency among the reviews of the various research projects.

- The panel report, amended as mandated by the full ACRS, will be made available to RES as soon as possible.

- o At the end of the fiscal year, we will issue a summary report to the RES Director.

The panel will consider the following questions in its review of the research projects:

- Are objectives of the research project adequately defined and quantified where appropriate?

- Has an adequate technical approach been defined and pursued?
  - Has planning of the work taken advantage of applicable technology established in the literature?
  - Has planning or execution of the work used available expertise in appropriate disciplines?
  - Have assumptions key to the technical approach and results been identified and tested or otherwise justified?
  - Have interfaces among elements of the project been identified and the interface requirements controlled?
  - Have technically defensible alternatives been considered and compared in trade studies of appropriate formality?
  - Has risk information been used appropriately?

- Have adequate quality controls been applied to the work?
  - Has peer review been used effectively?

- Do the results of the research satisfy objectives of the program?
  - Have significant uncertainties been characterized?
  - Have important sensitivities been identified?

- Has documentation of research results been appropriate and done adequately?

Sincerely,

Mario V. Bonaca

Mario V. Bonaca
Chairman

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

April 27, 2004

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

SUBJECT: DRAFT PLAN FOR IMPLEMENTATION OF THE COMMISSION'S PHASED APPROACH TO PRA QUALITY

Dear Chairman Diaz:

During the 511th meeting of the Advisory Committee on Reactor Safeguards on April 15-17, 2004, we reviewed the NRC staff's draft plan for implementing the Commission's phased approach to Probabilistic Risk Assessment (PRA) Quality. Our Subcommittee on Reliability and PRA reviewed this matter during a meeting held on March 25, 2004. During these reviews, we had the benefit of discussions with the NRC staff and the Nuclear Energy Institute. We also had the benefit of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

1. The NRC staff has developed a practical strategy that would encourage the development of guidance documents necessary to implement the Commission's phased approach to PRA quality.

2. The staff review of an application using a PRA with a scope greater than that for which guidance documents exist should not be given low priority.

3. Proactive licensees should not be discouraged from pushing the boundaries of the state of the practice by the prospect of low priority staff review. Licensees should be encouraged to address in their application the relevant technical issues cited in the December 18, 2003 Staff Requirements Memorandum (SRM) (Reference 1). The staff should give high priority to these reviews.

4. Development of guidance on how to perform sensitivity and uncertainty analyses should receive a higher priority in the draft plan.

5. The staff should be prepared to develop its own guidance documents if industry consensus standards are not developed in a timely manner to meet the Commission's schedule for achieving Phase 3.

DISCUSSION

In the December 18, 2003 SRM, the Commission approved the implementation of a phased approach to achieving an appropriate quality for PRAs used in support of NRC's risk-informed regulatory decisionmaking. Also, the Commission directed the staff to develop an action plan that would define a practical strategy for implementing this phased approach to PRA quality. In response to the Commission's direction, the NRC staff has developed a draft action plan (Reference 2).

The SRM recognizes that PRA quality cannot be separated from the regulatory decision the PRA is intended to support. As Regulatory Guide (RG) 1.174 (Reference 3) states and the SRM confirms, PRA quality requirements must be commensurate with the specific application. We have also emphasized the importance of focusing on the decision itself in our letters (References 4 and 5). The quality of the decision will be higher if the PRA provides all the risk insights that are relevant to that decision. We believe it is more appropriate to refer to the technical adequacy of the PRA for a specific regulatory decision rather than its quality.

The adequacy of a PRA is determined by three elements: scope (internal and external initiating events, full power and low-power and shutdown modes), level of detail, and technical adequacy. The SRM distinguishes between a "baseline PRA" and "risk-informed decisionmaking elements." The baseline PRA is independent of the application while the risk-informed decisionmaking elements are unique to the application. The term "PRA quality" (or, better, "adequacy") refers to the baseline PRA.

In the SRM, the phases are differentiated by the availability of guidance (industry consensus standards, industry guidance documents, and regulatory guides). Conformance with this guidance accompanied by peer and NRC staff reviews are expected to ensure technical adequacy. The staff is directed to give low priority or even return non-conforming applications. The goal is to reach Phase 3 by December 31, 2008. In this phase, it is envisioned that the licensees will have baseline PRAs that will conform to existing guidance documents and will be of sufficient depth to support all anticipated applications. Phase 4, in which the PRAs will be using state-of-the-art methods, will be considered after Phase 3 has been completed and is not part of the proposed plan.

Many guidance documents, especially consensus standards, prescribe what should be done rather than how. The technical adequacy of a PRA, however, depends very much on how various assessments are done. The argument against describing acceptable methods is that doing so will stifle the development of innovative methods. In some cases, such as the assessment of common-cause failures, the acceptable methods emerge slowly as practitioners begin to converge to their use. Although we acknowledge that guidance documents cannot, and should not, be overly prescriptive, and that the evaluation of technical adequacy will have to rely on the judgment of peer reviewers, we believe that sufficient guidance should be given so that these reviewers will be aware of what the agency expects in the area of technical adequacy.

It is recognized that we are currently somewhere between Phase 1 and Phase 2. A question that is raised is whether the staff should give low priority to its reviews of applications that use PRAs with a scope greater than that for which guidance exists. The staff recommends that review of these applications should be given low priority. Although we acknowledge the staff's concern that such reviews will be resource intensive, we believe that proactive licensees should not be discouraged from pushing the boundaries of the state of the practice. In fact, the development of guidance documents at a later time will rely heavily on the work of these licensees.

The Commission has directed the staff to include in its plan "the resolution of technical issues, such as model uncertainty, treatment of seismic and other external events, and human performance issues for each application and phase." During our review of Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," in September 2003, the staff told us that guidance regarding model uncertainties and the conduct of uncertainty and sensitivity studies might be available for our review in early 2004. The draft plan lists a projected completion date of December 30, 2004, for this guidance (a NUREG report).

Development of guidance on how to perform sensitivity and uncertainty analyses should receive a higher priority in the draft plan. Additionally, there are regulatory decisions that must be made before the projected completion date of December 30, 2004. These decisions may be sensitive to the technical issues cited in the SRM. We understand that the industry has its own projects on these matters (Reference 6). We believe that the staff review of the products of these projects, when submitted to the NRC, should receive high priority.

In a letter to the EDO (Reference 7), representatives of the American Nuclear Society (ANS) and American Society of Mechanical Engineers (ASME) express concern that "the schedule defined in the SRM seems rather ambitious." The plan should provide the staff sufficient time to develop its own guidance, if consensus standards are not produced in a timely manner to achieve Phase 3 by December 31, 2008, as the SRM directs.

Sincerely,

Mario V. Bonaca

Mario V. Bonaca
Chairman

References:

1. Staff Requirements Memorandum from Annette Vietti-Cook, Secretary, to Chairman Diaz, Subject: COMNJD-03-0002 - Stabilizing the PRA Quality Expectations and Requirements, dated December 18, 2003.
2. Letter from Gareth Parry, NRR, to Michael R. Johnson and Suzanne Black, Division of Systems Safety and Analysis, NRR, Subject: Draft Plan for Implementation of the Commission's Phased Approach to PRA quality, March 15, 2004.
3. 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," November 2002.
4. Letter from Mario V. Bonaca, Chairman, ACRS, to Nils J. Diaz, Chairman, NRC, Subject: Improvement of the Quality of Risk Information for Regulatory Decisionmaking, May 16, 2003.
5. Letter from Mario V. Bonaca, Chairman, ACRS, to Nils J. Diaz, Chairman, NRC, Subject: Draft Final Regulatory Guide x.xxx, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," (Formerly DG-1122), September 22, 2003.
6. Letter from Anthony R. Pietrangelo, Nuclear Energy Institute, to Mike Tschiltze, NRR, Subject: Provides an Industry Perspective on the Commission Paper and NRC Staff Draft Implementation Plan Regarding PRA Quality and Scope, April 8, 2004.
7. Memorandum from James F. Mallay, Chair, ANS Standards Board, and C. Wesley Rowley, ASME Vice President, Nuclear Codes and Standards, to William D. Travers, EDO, NRC, Subject: Comments on the Commission-Approved Implementation of a Phased Approach to Develop PRAs to Adequately Support Risk-Informed Applications and Regulatory Decision-Making, March 15, 2004.

April 27, 2004

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

SUBJECT: SECY-04-0037, "ISSUES RELATED TO PROPOSED RULEMAKING TO RISK-INFORM REQUIREMENTS RELATED TO LARGE BREAK LOSS-OF-COOLANT ACCIDENT (LOCA) BREAK SIZE AND PLANS FOR RULEMAKING ON LOCA WITH COINCIDENT LOSS-OF-OFFSITE POWER"

Dear Chairman Diaz:

During the 511th meeting of the Advisory Committee on Reactor Safeguards on April 15-17, 2004, we reviewed SECY-04-0037, "Issues Related to Proposed Rulemaking to Risk-Inform Requirements Related to Large Break Loss-of-Coolant Accident (LOCA) Break Size and Plans for Rulemaking on LOCA With Coincident Loss-of-Offsite Power." Our Subcommittee on Regulatory Policies and Practices reviewed this matter during a meeting on April 1, 2004. During these reviews, we had the benefit of discussions with the NRC staff and the Nuclear Energy Institute. We also had the benefit of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

1. The risk-informed revision to 10 CFR 50.46 should permit a wide range of applications of the new break size as long as it can be demonstrated that the resulting changes in risk are small and adequate defense-in-depth is maintained.

2. The process and criteria in Regulatory Guide (RG) 1.174 are appropriate for evaluating the acceptability of changes proposed under a revised rule. However, explicit criteria to ensure mitigative capability for breaks beyond the new maximum break size and to limit the risk associated with late containment failure should be developed as part of the revised rule to ensure that sufficient defense-in-depth is maintained as plant changes are made.

3. We concur with the recommendation of the staff that the appropriate metric for the design basis maximum break size is the direct LOCA initiating event frequency.

4. Additional criteria and guidance are not needed for tracking cumulative risk due to the changes resulting from a risk informed 10 CFR 50.46.

5. The results of the expert elicitation for the frequency of LOCA events are yet to be finalized and peer-reviewed, but the process employed and the qualifications of the panel members appear to be well suited to the problem. The results should help provide a technical basis for the selection of the new maximum break size.

DISCUSSION

The double-ended guillotine break (DEGB) of the largest pipe in the system has always been recognized as an unlikely event. It was intended to be a surrogate accident that bounded the consequences of a wide spectrum of reactor accidents. Probabilistic risk assessments (PRAs) for existing plants show that the defense-in-depth provided by the emergency core cooling system (ECCS) capability and robust containment designs developed to deal with this accident have resulted in plants with low core damage frequency (CDF) and low risk to the public. However, experience and PRA have also shown that the focus on such large, highly unlikely breaks can have detrimental effects on safety. The demands on equipment resulting from the need to demonstrate the equipment's capability to deal with the DEGB can reduce the equipment's reliability and capability to function during the much more likely small and medium LOCAs. Improved understanding of the likelihood of various initiating events and the responses of reactor systems to those events suggests that a risk-informed approach to dealing with large break LOCAs could result in greater operational flexibility with little increase or even decreases in risk.

In a Staff Requirements Memorandum (SRM), dated March 31, 2003 (Reference 2), the Commission directed the staff to complete the technical basis supporting the redefinition of the maximum design-basis break size and to provide proposed rule changes to the Commission. In its evaluation of the SRM, the staff identified a number of policy and technical issues that it felt needed to be resolved to ensure that the new rulemaking for maximum break size redefinition does not result in any unintended consequences. The staff discusses these issues in SECY-04-0037.

Because the consequences of 10 CFR 50.46 in the regulatory system are pervasive, the staff believes the Commission needs to provide additional guidance on the scope of changes to be permitted under a new rule. The staff distinguishes between a "narrow" scope and a "broad" scope rule change. In a "narrow" scope rule change, specific areas of application would be identified, similar to the current use of leak-before-break to restrict the sizes of breaks considered in determining dynamic effects. An example would be to permit the use of the redefined maximum break size in determining the start times for emergency diesels. A "broad" scope rule would permit a wide range of applications of the new break size as long as it could be demonstrated that the resulting changes in risk are small and adequate defense-in-depth is maintained. We believe that the revised rule should support a broad scope of applications.

It may be possible to deal with some applications generically in the revised rule, but in most cases applications of the new rule will be developed by licensees and will require plant-specific demonstrations that the resulting changes in risk are acceptable. RG 1.174 provides a process for determining the acceptability of changes in risk associated with changes in the licensing basis. In SECY-04-0037, the staff's preliminary conclusion is that the numerical criteria listed in RG 1.174 for defining acceptable changes to a plant's licensing basis are not stringent enough to use for modifying the fundamental building blocks and protections provided in the current regulations. We disagree. The uncertainties may be different in different situations, but if a certain change in risk is acceptable in terms of a change to a licensing basis, we see no reason why there should be a different level of acceptable risk for a modification of a rule, even one as fundamental as 10 CFR 50.46.

The number and kind of changes that will be possible for a licensee to make under the new rule will depend strongly on the scope and technical detail of the licensee's PRA. What is important is a convincing demonstration that the resulting changes are indeed small enough to meet the RG 1.174 criteria. If a limited scope PRA is used, contributions to the ΔCDF and the total CDF and corresponding large, early release frequency (LERF) quantities from the omitted portions of the PRA and the associated uncertainties must still be conservatively estimated and demonstrated to be consistent with the RG 1.174 criteria.

The expert elicitation and other evaluations of the likelihood of large pipe breaks demonstrate that the frequency of such failures due to normal loads and conventional modes of degradation is quite low. It is much more difficult to quantify the potential for such failures due to abnormal loads, security issues, and human errors. Thus a capability to mitigate breaks beyond the new maximum break up to the DEGB of the largest pipe needs to be maintained. In the March 31, 2003 SRM, the Commission directed the staff not to permit changes in ECCS coolant flow rates or reduce containment capabilities. However, the degree of defense-in-depth provided by these systems may change as plants make changes in response to the new rule. We believe that the staff should be directed to develop criteria and guidance to quantify the capability to mitigate DEGB beyond the new maximum break size and thus ensure that sufficient defense-in-depth is maintained. One possibility is a criterion for the conditional probability of core damage given a DEGB beyond the new maximum break size, but other approaches are possible. Calculations of the conditional probability could be performed using the realistic approaches taken in PRAs to assess core damage rather than the conservative approach taken in Appendix K to assess core damage. Some degree of core damage could be permitted to occur, but coolability would be maintained and rapid failure of the vessel precluded. It may also be necessary to develop guidance to ensure the functionality of equipment that may no longer be required under design basis conditions, but would be needed to mitigate a beyond design basis break.

RG 1.174 includes consideration of the risks associated with late containment failures, but it does not provide any explicit criteria for evaluating such risks. Such a criterion was developed in the Framework for Risk-Informed Changes to the Technical Requirements of 10 CFR 50 document (Reference 3) where it was proposed that the conditional probability of a large late release (i.e., one that does not contribute to LERF, but occurs within approximately 24 hours of the onset of core damage) be limited to 10^{-1} or less. This criterion or a suitable alternative should also be considered when considering changes associated with a revised rule.

One of the important technical issues raised by the staff in SECY-04-0037 is the choice of the appropriate metric to determine the design basis LOCA maximum break size. The staff argues that a metric based on the expected frequency of pipe breaks is more direct than one based on the impact of LOCAs on CDF and LERF. Also, the staff argues that most licensees will be following a phased approach in upgrading their PRAs and any definitions based on CDF and LERF could result in maximum break sizes that vary simply because of changes in PRA methods. We concur with the staff's conclusion that break frequency is the best metric. The rationalist approach to defense-in-depth considers the frequency of an initiating event as a basic criterion in assessing the confidence that must be provided for the response to the initiating event.

The staff proposes to identify large break LOCA sizes applicable to various categories of plants if possible. If not, the staff would specify a plant-specific implementation process necessary to determine the appropriate plant-specific break size. We believe that it is possible and desirable to make generic definitions applicable to categories of plants.

As a consequence of the redefinition of the maximum break size, licensees may propose plant changes that will result in increases in risk. The RG 1.174 process will ensure that the change in risk associated with any specific change in the licensing basis will be small, but there is certainly a possibility that a significant number of changes will be proposed because of the change to 10 CFR 50.46. The staff proposes to determine the information that needs to be tracked for individual changes authorized by the rule over the life of the plant and to develop analysis guidelines for cumulative risk estimates that can be compared to applicable risk criteria. We believe that the limitations implied by the RG 1.174 criteria, the inclusion of specific defense-in-depth criteria for mitigation of beyond design basis breaks, and an explicit criterion for late containment failure will limit changes in cumulative risk to acceptable levels. RG 1.174 provides sufficient guidance and criteria to track and control cumulative risk, and additional requirements are not necessary.

The elicitation process to determine degradation related LOCA frequencies was well structured and the expert panel has an appropriate range of expertise (Reference 4). Although the results are still under final review, we expect that they will be confirmed by the planned peer review and will provide a technical basis for the selection of a maximum break size in terms of the frequency of the initiating event.

There are important policy and technical issues to be considered in the development of a risk-informed 10 CFR 50.46. We look forward to interacting with the staff as it pursues this effort after receiving further guidance from the Commission.

Sincerely,

Mario V. Bonaca

Mario V. Bonaca
Chairman

References:
1. Memorandum from William D. Travers, EDO, to the Commissioners, SECY-04-0037, Subject: Issues Related to Proposed Rulemaking to Risk-Inform Requirements Related to LBLOCA Break Size and Plans for Rulemaking on LOCA With Coincident Loss-of-Offsite Power, March 3, 2004.
2. Staff Requirements Memorandum (SRM) from Annette L. Vietti-Cook, Secretary, to William D. Travers, EDO, Subject: SECY-02-057 - Update to SECY-01-0133, "Fourth Status Report on Study of Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50 (Option 3) and Recommendations on Risk-Informed Changes to 10 CFR 50.46 (ECCS Acceptance Criteria)," dated March 31, 2003.
3. Draft, Revision 2, "Framework for Risk-Informed Changes to the Technical Requirements of 10 CFR 50," August 2000.
4. Memorandum from Michael E. Mayfield, RES, to John T. Larkins, Executive Director, ACRS, Subject: Forwarding of Commission Paper on "Loss-of-Coolant Accident (LOCA) Break Frequencies for the Option III Risk-Informed Reevaluation of 10 CFR 50.46, Appendix K to 10 CFR Part 50, and General Design Criteria (GDC) 35," and its Corresponding Attachment (Pre-Decisional For Internal ACRS Use Only), March 29, 2004.

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

May 7, 2004

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

SUBJECT: USE OF MIXED OXIDE LEAD TEST ASSEMBLIES AT THE CATAWBA NUCLEAR STATION

Dear Dr. Travers:

During the 512th meeting of the Advisory Committee on Reactor Safeguards, May 5-8, 2004, we met with representatives of Duke Power, the Union of Concerned Scientists, the Nuclear Energy Institute, and the NRC staff to discuss Duke Power's application to irradiate four mixed oxide fuel lead test assemblies using weapons-grade plutonium dioxide in the core of one of the reactors at the Catawba Nuclear Station. Our Subcommittee on Reactor Fuels reviewed this matter during its meeting on April 21, 2004. We had the benefit of the document referenced.

The application and the staff review of the application considered the irradiation of the four mixed oxide lead test assemblies in a core of 193 assemblies with no other lead test assemblies present in the core. Irradiation of the lead test assemblies is intended to provide data needed to support the development and the review of a future application to make more extensive use of mixed oxide fuel at the Catawba or the McGuire Nuclear Stations. We conclude that, under the restricted circumstances considered in both the Duke Power application and the NRC staff's safety evaluation, the four mixed oxide lead test assemblies in nonlimiting core locations that do not contain control rods can be irradiated in either of the Catawba reactor cores with no undue risk to the public health and safety.

Sincerely,

Mario V. Bonaca
Chairman

<u>Reference</u>:
Letter from R. Martin to H. Barron, "Safety Evaluation For Proposed Amendments to The Facility Operating License And Technical Specifications to Allow Insertion of Mixed Oxide Fuel Lead Assemblies," April 5, 2004.

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

May 13, 2004

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

SUBJECT: GOOD PRACTICES FOR IMPLEMENTING HUMAN RELIABILITY ANALYSIS

Dear Dr. Travers:

During the 512th meeting of the Advisory Committee on Reactor Safeguards, May 5-8, 2004, we reviewed Draft Letter Report (JCN W6994), "Good Practices for Implementing Human Reliability Analysis (HRA)," dated April 6, 2004. Our Subcommittees on Reliability and Probabilistic Risk Assessment (PRA) and on Human Factors also reviewed this matter in detail on April 22, 2004. During these reviews, we had the benefit of discussions with representatives of the NRC staff and their contractors. We also had the benefit of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

1. Draft Letter Report, "Good Practices for Implementing Human Reliability Analysis (HRA)," should be issued for public comment.
2. Draft Letter Report, "Good Practices for Implementing Human Reliability Analysis (HRA)," should be peer-reviewed by domestic and international experts.

DISCUSSION

Central to the Commission's policy on a phased approach to PRA quality is the availability of guidance documents. HRA is an important element of PRA. While there is general consensus that the Human Reliability Handbook (NUREG/CR-1278) provides reasonable models for evaluating human performance during routine activities such as maintenance, there is no agreement among HRA experts on how to model human performance during accident conditions. Since the guidance provided by available documents such as the American Society of Mechanical Engineers Standard and the Nuclear Energy Institute PRA Peer Review Process Guidance (NEI-00-02) is at a high level, there is a need to develop more detailed guidance. The draft letter report is intended to fulfill this need.

The report provides a set of good practices that HRA analysts should follow regardless of the particular model that they use. This is an important first step toward achieving consensus within the HRA community regarding the quantification of human reliability. The report is based on staff and contractor experience and is ready for public comment.

Developing a set of good practices for assessing human reliability during accidents is particularly challenging. Several models based on different assumptions have been proposed by domestic and international experts. We believe that the report will benefit from a formal peer review by these experts. Their participation in the development of the report will provide the additional, and very important, benefit of contributing to its acceptance by the international community. Therefore, the staff should organize such a review.

We were disappointed that organizational issues did not receive the attention they deserve. Experts analyzing operating events have concluded that these issues frequently are significant performance shaping factors. While we acknowledge that the state-of-the-practice on these issues is not mature, omitting them in the report ignores an important determinant of human performance.

We look forward to reviewing the report after the public comment period and the peer review.

Sincerely,

Mario V. Bonaca

Mario V. Bonaca
Chairman

References:

1. "Good Practices for Implementing Human Reliability Analysis (HRA)," Draft Letter Report (JCN W6994), April 6, 2004.
2. J. Forester, D. Bley, S. Cooper, E. Lois, N. Siu, A. Kolaczkowski, and J. Wreathall, "Expert Elicitation Approach for Performing ATHEANA Quantification," *Reliability Engineering and System Safety* 83 (2004) 207-220.
3. "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," ASME RA-S-2002, American Society of Mechanical Engineers, April 5, 2002.
4. "Probabilistic Risk Assessment Peer Review Process Guidance," NEI 00-02, Revision A3, Nuclear Energy Institute, March 2000.
5. A.D. Swain and H.E. Guttmann, "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications - Final Report," NUREG/CR-1278, SAND 80-0200, Sandia National Laboratories, August 1983.

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

May 21, 2004

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

SUBJECT: RESOLUTION OF CERTAIN ITEMS IDENTIFIED BY THE ACRS IN NUREG-1740, "VOLTAGE-BASED ALTERNATIVE REPAIR CRITERIA"

Dear Dr. Travers:

During the 512th meeting of the Advisory Committee on Reactor Safeguards, May 5-8, 2004, we completed our review of the progress made by the staff in resolving various steam generator (SG) tube integrity issues highlighted in our document NUREG-1740, "Voltage-Based Alternative Repair Criteria." We also heard presentations by and held discussions with the staff and its contractors regarding this matter during our 509th meeting, February 5-7, 2004. In addition, our Subcommittees on Materials and Metallurgy and on Thermal-Hydraulic Phenomena met with the representatives of the staff and their contractors on February 3-4, 2004, to review this matter in detail. We had the benefit of the documents referenced.

BACKGROUND

The Steam Generator Action Plan (SGAP) to resolve SG tube integrity issues includes the following items:

- Item 3.1: Investigate the effects of depressurization during a main steamline break (MSLB) on SG tube integrity. The staff proposes to close this item.
- Item 3.2: Complete investigation of jet penetration of adjacent tubes. The staff proposes to close this item.
- Item 3.3: Develop experimental information on source term attenuation on the secondary side of steam generators (ARTIST tests).
- Item 3.4: Develop a better understanding of SG tube behavior under severe accident conditions. The staff proposes to close subtasks 3.4 a, b.1, c, e, and g.
- Item 3.5: Develop improved methods of assessing risk associated with SG tubes under accident conditions. The staff proposes to close subtasks 3.5a and b.
- Item 3.6: Assess the technical basis for improving the probability of crack detection in SG tubes. The staff proposes to close this item.
- Item 3.7: Assess the need for better leakage correlations as a function of voltage for 7/8" SG tubes. The staff proposes to close this item.

- Item 3.8: Monitor the predictions of flaw growth for systematic deviations from expectations. The staff proposes to close this item.
- Item 3.9: Assess the need for a more technically defensible treatment of radionuclide release to be used in safety analyses of design-basis events. The staff proposes to close this item.
- Item 3.10: Develop a better mechanistic understanding of tube cracking processes.
- Item 3.11: Resolve Generic Safety Issue 163, 'Multiple Steam Generator Tube Leakage'.

CONCLUSIONS AND RECOMMENDATIONS

1. The analyses of the effects of depressurization during a MSLB on tube integrity have been completed and item 3.1 is appropriately closed out. However we recommend that, as a confirmatory measure, a review be performed of the U.S. industry SG tube pullout data and the associated extent of tube locking at tube support plates (TSPs) in degraded SGs.

2. The probability of jet impingement damaging adjacent tubes is negligibly small. Item 3.2 should be closed as proposed by the staff.

3. The staff has developed a technically defensible description of the probability of detection of a flawed tube as a function of flaw size. Item 3.6 should be closed. The continued use of the current constant flaw detection probability should be reexamined to have more realism in the evaluation.

4. The existing correlation of leakage with eddy current voltage for 7/8" diameter SG tubes is not accurate enough to be used. We agree with the staff that the choice of a 2 volt limit for the 7/8" diameter SG tubes is conservative with respect to the risk posed. Item 3.7 should be closed. We recommend that qualified data continue to be collected and analyzed in order to develop an improved correlation.

5. The studies of bypass scenarios due to thermally induced SG tube failures are still in progress. The staff should improve the thermal-hydraulic analyses needed for these studies to enable a realistic prediction for the fraction of heat that is transferred to the SG, rather than estimating a value for this fraction based on the 1/7th scale test results. The staff also needs to document the technical basis for its conclusion that the likelihood of cold-leg loop seal clearing is sufficiently low that the countercurrent flow situation is the appropriate model for these scenarios.

6. The staff continues to treat iodine spiking in a conservative, empirical fashion. We recommend that the staff develop a mechanistic understanding of iodine spiking so that analyses reflect current plant operations and the capabilities of modern fuel rods.

7. Item 3.8 (predictions of flaw growth) should not be closed until progress has been made on developing the cracking model under item 3.10.

DISCUSSION

We had extensive discussions with the staff on the differing professional opinion (DPO) concern that the blowdown forces and movement of TSP caused by the SG depressurization following a MSLB could cause cracks to form, grow, and unplug, leading to much higher primary-to-secondary side leakage than was initially assumed by the staff.

The staff presented the results of the tests and finite-element analyses which were performed to evaluate the stability of cracks in the SG tubes subjected to stresses due to the motion of the TSPs. This work supports the staff's contention that the bending stresses in the SG tubes are small compared to the axial stresses, and the effect of bending loads on the leakage from flawed SG tubes is also small.

The staff calculated hydraulic loadings on the internal components of a SG following a MSLB for use in the structural evaluation of the stability of cracks subjected to such loads. The loading calculations were performed using a TRACE three-dimensional model of the SG. The TRACE code's capability to predict an acoustically dominated thermal-hydraulic transient, such as that caused by a MSLB, was verified by comparing TRACE predictions to experimental data.

The tensile stresses in the SG tubes locked at the TSP junctions were also calculated for varying numbers of locked SG tubes. The staff concluded that even if a few percent of the SG tubes are locked at the TSP, the dynamic loads associated with a MSLB will have little impact on the integrity of the SG tubes, unless extensive circumferential cracking is present.

Results of an extensive study on the SG tube pullout forces at a French nuclear plant (Dampierre) strongly support the conclusion that SGs with drilled hole carbon steel TSPs will have enough SG tubes locked at the TSP and that the dynamic loads will be of little concern. The staff recognizes that its conclusions would be strengthened by a more extensive review of industry data, especially from the U.S. pressurized water reactor plants, on the expected number of locked SG tubes in degraded SGs and on the SG tube pullout stress data. However, the staff already requires that licensees who wish to implement an alternate repair criteria that go beyond the Generic Letter (GL) 95-05 limits expand tubes to provide assurance that TSP motion will not uncover flaws. The voltage limits in GL 95-05, the low likelihood of the uncovering of the TSP region at the lower TSPs where cracking is likely, and the low probability of circumferential cracking in the TSP region lead to the conclusion that the staff's present positions are reasonably conservative. However, since the staff's conclusions regarding MSLB-induced leakage is based heavily on the Dampierre data, we believe that a review of U.S. industry data should be considered as a confirmatory measure.

In addressing SGAP item 3.2, the staff developed analytical models and performed erosion tests to determine the effect of fluid jets on thinning of SG tubes. The staff performed tests that provide a conservative simulation of a steamline break to determine the susceptibility of steam generator tubes to erosive damage from impacting steam jets from adjacent flawed tubes. We agree with the staff's conclusion that the probability of damage progression via jet cutting of adjacent SG tubes is low and need not be considered in accident analysis.

Under SGAP item 3.6, the staff presented a detailed analysis of the probability of detection (POD) of SG tube cracks. The staff concluded that the continued use of a POD value of 0.6 was conservative for eddy current voltages in the 1-2 volts range where the observed POD value approaches 0.9. We believe that the analysis of this topic and the supporting data are sufficient to close out this item. However, the decision to continue the use of a POD value of 0.6, rather than using the developed analytical capability that recognizes the entire POD vs. crack depth correlation, should be reexamined with a view to making the evaluation more realistic.

It is apparent that some of the technical issues raised in NUREG-1740 may not be resolved. For instance, an improved correlation between eddy current voltage and leakage for 7/8" diameter SG tubes has not been achieved in SGAP item 3.7. Although there is a qualitative understanding of the reasons for the unacceptable scatter in the relationship, it is apparent that the accumulation of further unqualified data will not improve the correlation factor. We agree with the staff that the choice of a 2 volt limit for the 7/8" diameter SG tubes is conservative with respect to the risk posed. Item 3.7 should be closed. We also believe that in order to get relief on this 2 volt limit, a qualified correlation of burst pressure and leak rate vs. flaw size needs to be developed.

The staff has performed thermal-hydraulic analyses of the core, upper plenum, hot leg, SG plenum, and SG tubes during a severe accident scenario. One output of the thermal-hydraulic analyses is the component temperatures that result from the convective heating. Hot gases from the core rise into the upper plenum and flow toward the SG inlet plenum along the top portion of the hot leg. Cooler gases from the SG return by countercurrent flow along the bottom of the hot leg to the reactor vessel. This may lead to thermally induced failure of the hot leg or the SG tubes.

The analyses of this complex flow process have used an adjusted one-dimensional thermal-hydraulic system code. The code was calibrated with the 1/7th scale SG test data, to determine the flow to the SG and set the boundary conditions for a computational fluid dynamics (CFD) simulation of the mixing of hot and cold gases in the SG inlet plenum.

It is apparent that the amount of heat which goes to the SG depends on the convective flow and heat transfer processes between the reactor core and the SG and it may not be appropriate to assume it as an input variable based on the 1/7th scale test, as was done in the SCDAP/RELAP analysis. Several features of the 1/7th scale tests (such as the method of cooling the SG tubes) may be atypical of the full-scale plants. The CFD work should be extended to include sufficient parts of the upper plenum and core flow process to permit a calculation of the hot-leg entrance conditions. This would permit the mixing process surrounding the cold plume emerging from the hot leg and descending into the reactor pressure vessel to be modeled in much the same way as the hot plume was modeled by the CFD calculation in the SG inlet plenum.

Despite the large uncertainties in the predicted failure times of the SG tubes and other reactor coolant system (RCS) components, the staff assumed that it will be able to determine the relative failure times with sufficient accuracy to permit conclusions about whether operation with flawed SG tubes will have a significant impact on the likelihood of SG bypass accidents. The uncertainty in the calculated thermal response of the primary system components under severe accident conditions may be too large to use such results to determine whether the primary system component or the SG tubing fails first. The primary system conditions (i.e., high-temperature natural circulation) involve phenomena beyond the prediction capability of a one-dimensional thermal-hydraulics system code. In addition, the uncertainties in the associated heat transfer and friction correlations under these conditions are significant. Even when the calculation is augmented with CFD modeling, the uncertainty of the predicted outcome may still be very large. The staff should continue its efforts to conduct uncertainty analyses to determine the probability of containment bypass. The staff also needs to document the technical basis for its conclusion that the likelihood of cold-leg loop seal clearing is sufficiently low that the countercurrent flow situation is the appropriate model for these scenarios.

The staff has not accepted our recommendation to develop a mechanistic understanding of the iodine spiking issue. The staff continues to use a conservative, empirical estimate of iodine spiking for accident consequence analyses. This estimate is based on historical data that may not reflect current practices in plant operations or the capabilities of modern fuels to prevent coolant contamination. We again encourage the staff to take advantage of iodine studies available in the literature and develop a mechanistic understanding of the phenomenon.

In our meetings we did not discuss in detail the progress in certain SGAP tasks such as the modeling of time-dependent degradation (e.g., stress corrosion cracking) of tubes (item 3.10) and the development of source term information in the ARTIST program (item 3.3).

We look forward to continuing discussions with the staff on this important project.

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

Sincerely,

Mario V. Bonaca

Mario V. Bonaca
Chairman

References:

1. Memorandum from William D. Travers, Executive Director for Operations, NRC, to John T. Larkins, Executive Director, ACRS, Subject: Differing Professional Opinion on Steam Generator Tube Integrity Issues, July 20, 2000.
2. ACRS Report dated September 11, 2000, from Dana A. Powers, Chairman, ACRS Ad Hoc Subcommittee, to William D. Travers, Executive Director for Operations, NRC, Subject: Differing Professional Opinion on Steam Generator Tube Integrity.
3. ACRS Report dated February 1, 2001, from Dana A. Powers, Chairman, ACRS Ad Hoc Subcommittee, to William D. Travers, Executive Director for Operations, NRC, Subject: Differing Professional Opinion on Steam Generator Tube Integrity.
4. U.S. Nuclear Regulatory Commission, NUREG-1740,"Voltage-Based Alternative Repair Criteria," Advisory Committee on Reactor Safeguards, March 2001.
5. ACRS Report dated May 15, 1995, from T. S. Kress, Chairman, ACRS, to James M. Taylor, Executive Director for Operations, NRC, Subject: Proposed Final Generic Letter 95-05, "Voltage-Based repair Criteria for Westinghouse Steam Generator Tubes."
6. ACRS Report dated October 18, 2001, from George E. Apostolakis, Chairman, ACRS, to Richard A. Meserve, Chairman, NRC, Subject: NRC Action Plan to Address the Differing Professional Opinion Issues on Steam Generator Tube Integrity.
7. The Committee also reviewed the following documents that the staff provided to ACRS in support of the February 2004 Briefings:

SGAP Item No.	Document Title/Date	Public/Non-Public
3.1.a and b	Memo to Mayfield from Eltawila, 12/30/02, "Calculation of Steam Generator Tube Support Plate and Tube Loads Following a MSLB or FWLB Using TRAC-M;"	Non-Public
	Att: RES Report SMSAB-02-05, 9/02	Non-Public
3.1.d thru h	Memo from Mayfield to Strosnider, 12/26/02, "Closure of Steam Generator Action Plan Items 3.1d) to h);"	Non-Public
	Att: NUREG/CR-XXXX, "Sensitivity Studies of Failure of Steam Generator Tubes During Main Steam Line Break and Other Secondary Side Depressurization Events"	Non-Public (Proprietary)
3.1.i	Memo from Mayfield to Strosnider, 7/14/03, "Closure of Steam Generator Action Plan Item 3.1i);	Non-Public
	Att: Technical Letter Report: Tests and Analysis Of Failure of Degraded Tubes Under Internal Pressure and Bending Loading"	Non-Public
3.2	Memo from Mayfield to Strosnider, 7/9/02, "Closure of Steam Generator Action Plan Items 3.2 and 3.6;"	Public
	Att: NUREG/CR-6756, "Analysis of Potential for Jet-Impingement Erosion from Leaking Steam Generator Tubes during Severe Accidents;"	Public
	Att: NUREG/CR-6774, "Validation of Failure And Leak-Rate Correlation for Stress Corrosion Cracks in Steam Generator Tubes"	Public
3.4.a	Memo from King to Zimmerman, 9/28/01, "Completion of Subtask Milestone in Steam Generator Action Plan;"	Public
	Attached report: ISL-NSAD-NRC-01-004	Non-Public
3.4.c	ISL-NSAD-TR-02-03, "Tube-to-Tube Temperature Variations During the Station Blackout Event" (Draft), 8/02	Non-Public

SGAP Item No.	Document Title/Date	Public/Non-Public
3.4.e.1	Memo from Rosenthal to Barrett, Wermiel, Banerjee, A Completion of Preliminary Milestone from Steam Generator Action Plan, 8/31/01	Non-Public
	Att: NUREG-1781, "CFD Analysis of 1/7th Scale Steam Generator Inlet Plenum Mixing During a PWR Degraded Core Accident," 10/03	Public
3.4.e.2	Draft Report, "CFD Prediction of Full-Scale Steam Generator Inlet Plenum Mixing for the Evaluation of Scale Effect," 3/02	Non-Public
3.4.e.3	Memo from Eltawila to Holahan, "Preliminary Results from SGAP Item 3.4.e.3 Related to the CDF Evaluation of Inlet Plenum Mixing," 2/25/03	Non-Public
3.5.a	Memo from Cunningham to Chokshi and Rosenthal, "Transmission of a Proposed Framework for Analysis of Severe Accident Induced Steam Generator Tube Ruptures," 4/1/02	Non-Public
3.5.b	Memo from Newberry to Strosnider, "Closure of Steam Generator Action Plan Items 3.5(b) and 3.5(c), "Severe Accident Induced-Steam Generator Tube Rupture (SAI-SGTR) Methodology Report," 6/30/03 (Note: Contrary to this memo item 3.5.c is not yet closed)	Public
	Draft Report, "Methodology for Assessing Severe Accident-Induced Steam Generator Tube Rupture," 6/03	Public
3.6	Memo from Mayfield to Strosnider, 7/9/02 "Closure of Steam Generator Action Plan Items 3.2 and 3.6;"	Public
	Att: NUREG/CR-6785, "Evaluation of Eddy Current Reliability from Steam Generator Mock-Up Round-Robin," 9/02	Public
3.7	Memo from Barrett to Sheron and Borchardt, "Steam Generator Action Plan - Completion of Item Number 3.7 (TAC No. MB7216)," 4/25/03;	Public
	Letter from Bateman to Marion (NEI), "Exclusion of French Data from the Steam Generator Degradation Specific Management Database," 10/8/02	Public

SGAP Item No.	Document Title/Date	Public/Non-Public
3.8	Memo from Strosnider to Sheron and Borchardt, "Steam Generator Action Plan - Completion of Item Number 3.8 (TAC No. MB0258)," 1/3/02	Public
3.9	Draft Writeup, "ASGAP Item 3.9 - Iodine Spiking"	Non-Public

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

June 3, 2004

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: DRAFT REGULATORY GUIDE, DG-1130, "CRITERIA FOR USE OF COMPUTERS IN SAFETY SYSTEMS OF NUCLEAR POWER PLANTS," (PROPOSED REVISION 2 TO REGULATORY GUIDE 1.152)

During the 513th meeting of the Advisory Committee on Reactor Safeguards on June 2-4, 2004, the Committee considered the Draft Regulatory Guide DG-1130, "Criteria for Use of Computers in Safety Systems in Nuclear Power Plants." The Committee agrees with the staff's proposal to issue the draft Regulatory Guide for public comment.

Reference:

Memorandum dated May 24, 2004, from Michael Mayfield, Office of Nuclear Regulatory Research, to John T. Larkins, Executive Director, ACRS, Subject: Draft Regulatory Guide DG-1130, "Criteria for Use of Computers in Safety Systems of Nuclear Power Plants," (Proposed Revision 2 to Regulatory Guide 1.152).

cc: A. Vietti-Cook, SECY
W. Dean, OEDO
D. Weaver, OEDO
J. Craig, RES
M. Mayfield, RES
S. Aggarwal, RES

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

June 7, 2004

MEMORANDUM TO: Bruce A. Boger, Director
Division of Inspection Program Management
Office of Nuclear Reactor Regulation

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

SUBJECT: DEFERRAL OF ACRS REVIEW OF DRAFT SRP CHAPTER 13.0 "CONDUCT OF OPERATION," SECTION 13.1.2 - 13.1.3, "OPERATING ORGANIZATION" REVISION AND SUPPORTING DOCUMENTS UNTIL AFTER PUBLIC COMMENT

On April 15, 2004, you wrote a memorandum that requested that the Advisory Committee on Reactor Safeguards (ACRS) consider deferral of its review of the subject sections of the standard review plan, and the associated draft final NUREG document, until after public comments had been requested and received by the staff.

The Committee considered your request at its 513th meeting, and it agrees with your recommendation that its review can be deferred until after the public comments have been received and addressed by the staff.

cc: ACRS Members
ACRS Staff
J. Bongarra (NRR)
A. Szabo (RES)

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

June 9, 2004

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: PROPOSED REVISIONS TO STANDARD REVIEW PLAN SECTIONS 5.2.3, 5.3.1, AND 5.3.3

During the 513th meeting of the Advisory Committee on Reactor Safeguards on June 2-4, 2004, the Committee discussed proposed revisions to Standard Review Plan Sections 5.2.3, "Reactor Coolant Pressure Boundary Materials," 5.3.1, "Reactor Vessel Materials," and 5.3.3, "Reactor Vessel Integrity." The Committee decided not to review these proposed revisions.

Reference:

Memorandum dated May 10, 2004, from Richard J. Barrett, Office of Nuclear Reactor Regulation to John T. Larkins, Executive Director, ACRS, Subject: Draft Revisions to SRP Sections 5.2.3, 5.3.1, and 5.3.3.

cc: A. Vietti-Cook, SECY
W. Dean, OEDO
D. Weaver, OEDO
D. Matthews, NRR
K. Wickman, NRR
R. Barrett, NRR
M. Gamberoni, NRR

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

June 9, 2004

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

SUBJECT: DIGITAL INSTRUMENTATION AND CONTROLS RESEARCH PROGRAM

Dear Mr. Reyes:

During the 513th meeting of the Advisory Committee on Reactor Safeguards on June 2-4, 2004, we reviewed the staff's research activities on risk assessment of digital instrumentation and controls (I&C) systems, which are part of the Digital I&C Research Program. Our Subcommittees on Reliability and Probabilistic Risk Assessment and on Plant Operations reviewed this matter during a meeting held on March 26, 2004. During these reviews, we had the benefit of discussions with the NRC staff and its contractors. We also had the benefit of the documents referenced.

CONCLUSIONS

We support the effort of the Digital I&C Research Program to develop more quantitative measures of digital system reliability.

DISCUSSION

As nuclear power plants move toward increased use of digital technology, new challenges are created due to analog technology obsolescence and the functional advantages of digital technology. The use of this technology may introduce new failure modes into plant systems. These must be understood and modeled for two major reasons that affect regulatory decisions:

1. Probabilistic risk assessment (PRA), the principal analytical tool supporting the Commission's risk-informed initiatives, must be modified to include models for the failure modes that digital software may introduce.

2. The current regulatory review of digital systems is based largely on controlling the process for software development without any assessment regarding failure modes or reliability.

The goals of the Digital I&C Research Program are to:

- Gain an understanding of how digital systems fail and how likely it is that they will fail in use.
- Develop methods and tools for including digital system models into PRA.
- Develop guidance for regulatory applications involving digital system reliability.

This program will provide additional information on digital I&C failures, digital failure assessment methods and system models, digital reliability assessment methods, and integration in PRAs. This information should be included in the staff's reviews. It is evident that no single type of information will be sufficient for regulatory decision-making. This program is pursuing multiple approaches. We strongly support the goals of this program.

Sincerely,

Mario V. Bonaca
Chairman

Additional Comments by ACRS Member George E. Apostolakis

I agree with the Committee's conclusion. I offer two recommendations for the staff's consideration as it moves forward with this very important program:

A. The databases containing software-induced failures of technological systems should be reviewed and conclusions should be drawn regarding failure modes and their frequency of occurrence.

B. Available methods for the identification of failure modes and the assessment of the reliability of systems that are software driven should be reviewed critically. Their domains of validity should be determined by examining their assumptions and comparing them with the insights gained from the database review.

While I agree with the program's goals, I would like to see convincing evidence that funded projects will answer important questions that arise in the study of digital systems. These questions ultimately lead to fundamental issues related to the proper treatment of software "failures."

The literature on digital software (References 1-3) indicates that there are two main interpretations of the concept of software safety. The first interpretation views "failure" as a property of the software itself, just as the failure modes of hardware are considered properties of the components. This "software-centric" view is to be contrasted with the second interpretation, the "system-centric" view, which asserts that it is meaningless to talk about the failure of a piece of software in isolation. In this view, the concept of failure becomes meaningful only when the software is considered within a system, in which case one speaks of system failures. This approach is very similar to the modeling of human performance (Reference 3). An unsafe human act is considered meaningful only in the system context within which it occurs, an observation that has led the Office of Nuclear Regulatory Research to the development of the concept of "error-forcing context" (Reference 4).

A natural way to determine which interpretation of the concept of software failure is appropriate (and under what conditions) is to examine the available data that involve software failures. The staff's research program includes a task at Brookhaven National Laboratory that deals with databases. Although the Committee has not reviewed this effort in detail, I gather from the staff's presentation at the subcommittee meeting that this review was not intended to explore the concept of failure discussed above. The Committee was told that software failures have caused serious accidents in other industries and that an examination of licensee event reports has concluded that digital failures are approximately evenly divided among hardware, software, and human-system interface related failures. I recommend that the analysis be expanded to provide insights into which of the above interpretations would be the appropriate one to explain what happened.

There are many models for the evaluation of probabilities of software performance. These models fall naturally into two categories, depending on which interpretation of failure one adopts. The software-centric models borrow heavily from reliability models that have been developed for hardware components, e.g., exponential failure models. The system-centric models propose the expansion of standard system analysis tools, such as fault trees, to include software interactions with the hardware.

These models must be reviewed critically before the staff decides on which approach to adopt. This review should include an evaluation of the fundamental assumptions behind each model and a comparison with the insights gained from the review of the databases. For example, the staff told the Committee that "Markov-type modeling at the processor level appears to be capable of capturing digital design features." While this may, in fact, be a good conclusion, I would like to be convinced by arguments supporting the assumption of a constant rate of transition between "good" and "failed" states and by evidence from actual experience that supports this assertion. What kinds of events occurring in time at a constant rate does this model consider? Are errors in requirements and specifications included? These errors have been found to be the cause of a large number of software errors[1]. How are the interactions of the software with the rest of the system to be modeled? These interactions include potentially unexpected system conditions that exercise the software in a way for which it may not have been designed, thus leading to "wrong" responses. Past research results point to situations of this kind, where one could argue that the software did not fail, but nevertheless was induced to do the "wrong thing" by a design flaw left in the system or the software itself.

I believe that the implementation of the two recommendations that I have offered will provide a strong technical foundation for the achievement of the staff's goals in this program.

REFERENCES

1. National Research Council, *Digital Instrumentation and Control Systems in Nuclear Power Plants: Safety and Reliability Issues,* National Academy Press, Washington, D.C., 1997.

2. N.G. Leveson, *Safeware: System Safety and Computers,* Addison-Wesley, Reading, MA, 1995.

3. C. Garrett, and G. Apostolakis, "Context in the Risk Assessment of Digital Systems," *Risk Analysis,* 19:23-32, 1999.

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[1]Some analyses of NASA failure experience indicate that nearly 75% of failures found in operational software are rooted in requirement errors (Reference 5).

4. S. E. Cooper, A. M. Ramey-Smith, J. Wreathall, G. W. Parry, D. C. Bley, W. J. Luckas, J. H. Taylor and M. T. Barriere, *A Technique for Human Error Analysis (ATHEANA).* NUREG/CR-6350, U.S. Nuclear Regulatory Commission, Washington, DC, 1996.

5. R. R. Lutz, *Targeting Safety-Related Errors during Software Requirements Analysis,* Proceedings of the First ACM SIGSOFT Symposium on the Foundations of Software Engineering, pp. 99-106, 1993.

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

June 15, 2004

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

SUBJECT: DRAFT FINAL 10 CFR 50.69, "RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS, AND COMPONENTS FOR NUCLEAR POWER REACTORS"

Dear Chairman Diaz:

During the 513th meeting of the Advisory Committee on Reactor Safeguards on June 2-4, 2004, we reviewed the draft final 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors," (Reference 1). Our Subcommittee on Reliability and Probabilistic Risk Assessment reviewed this matter during a meeting held on February 19, 2004. During these reviews, we had the benefit of discussions with the NRC staff and the Nuclear Energy Institute (NEI). We also had the benefit of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

1. The final 10 CFR 50.69 should be issued.

2. We agree with the staff that Regulatory Guide (RG) 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance," should be issued for trial use.

DISCUSSION

10 CFR 50.69 has been developed to allow licensees to implement an alternative regulatory framework with respect to "special treatment." Special treatment refers to those requirements that provide increased assurance beyond normal industrial practice. Under this framework, licensees using a risk-informed process for categorizing structures, systems, and components (SSCs) according to their safety significance can remove SSCs of low-safety significance from the scope of certain identified special treatment requirements. Guidance for implementing 10 CFR 50.69 is contained in NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline," which the staff has conditionally endorsed in RG 1.201.

The high-level requirements for categorization and associated treatment of SSCs embodied in the proposed final 50.69 rule are appropriate and the final rule should be issued.

The guidance for implementing 50.69 included in NEI 00-04 and RG 1.201 provides an acceptable approach to the categorization, but some additional experience with the guidance is needed. Therefore, we support the staff's proposal to issue the RG for trial use. Most of the conditions in RG 1.201 on the acceptance of NEI 00-04 can be addressed by relatively minor modifications and clarifications of NEI 00-04. The most substantial difference involves the monitoring of the performance of risk-informed safety class (RISC)-3 components.

In a Staff Requirements Memorandum dated March 28, 2003 (Reference 2), the Commission stated that "Relevant operational experience should be evaluated in an ongoing manner with the aim of reducing the uncertainty in assessing the effect of treatment on reliability and common-cause failures." 10 CFR 50.69(e)(1) requires the feedback of plant operational experience. The revised rule requires a monitoring program, but implementing such a program is a challenging task. RG 1.201 may not provide adequate guidance for implementing this program. Refining the guidance for implementing a monitoring program need not delay initial application of the rule and would benefit from experience gained from the trial use of RG 1.201.

In RG 1.201, the staff proposes the use of a threshold number based on the expected number of failures associated with the reliability values used in the categorization process and the observed number of failures over a period of time. This is not a technically sound approach. Perhaps the methods used in investigating operating experience to identify common-cause failures could offer insights into what should be done (Reference 3).

The NEI 00-04 guidance still addresses uncertainties only through sensitivity studies, limited to the parameters that appear in the probabilistic risk assessment (PRA). An Electric Power Research Institute (EPRI) study (Reference 4) showed that, in the cases studied, a rigorous parameter uncertainty analysis would not lead to a different categorization of SSCs from the one produced using point estimates based on mean values. In one instance, this was not true, i.e., a SSC was categorized as being of low-safety significance on the basis of point estimates, but of high-safety significance when uncertainty distributions were propagated rigorously. The EPRI sensitivity study, however, did agree with the rigorous results. These results are in general agreement with the conclusions regarding parameter uncertainties presented in Reference 5.

Both RG 1.201 and the NEI 00-04 guidance recognize that the conservative assumptions often made in PRAs for external events can result in values of importance parameters that misrepresent safety significance. Although internal events PRAs tend to be more realistic than external events PRAs, they do contain modeling assumptions that may result in incorrect estimates of importance parameters. For example, in the development of the Multiple System Performance Index, comparisons of importance parameters determined from the Simplified Plant Analysis Risk (SPAR) models and plant PRAs showed that values of the importance parameters were affected by model assumptions even though the overall core damage frequency (CDF) from the SPAR model and the plant PRA were in reasonable agreement, and the SPAR models had to be significantly revised to get agreement for the importance parameters. Despite this recognition and experience, the effect of model uncertainties on the importance parameters is not addressed in the current guidance beyond the instruction to treat values of the importance parameters determined from internal and external events PRAs separately and to use the highest value of the importance parameter determined from either PRA.

Even in the case of parametric uncertainties, there is no specific guidance on how the distributions for the parameters that are used for the sensitivity studies are selected. It is not clear whether they are plant specific or generic in nature.

The staff is developing general guidance on the treatment of uncertainties for PRAs. This guidance can perhaps be adapted for use in the categorization process and tested during the trial use period.

The choice of fixed screening values for the importance measures results in different ΔCDF and Δ large, early release frequency (LERF) for different plants. As we stated in our report dated October 12, 1999 (Reference 6): "It is evident that the absolute value of the baseline risk metric is a critical element in these evaluations and that the importance measures contain only relative information with respect to a given risk metric."

"The change in risk depends on this absolute value also, i.e., ΔCDF at two plants with different baseline CDFs, will be different for the same change in the unavailability of a component whose importance measures have the same value at these plants." Reference 5 states that "if we are interested in controlling the change in risk in an absolute sense, it does not make sense to have a universally fixed value of Fussell-Vesely as a criterion for risk significance," and "it is clear that it does not make much sense to define a universal criterion based on Risk Achievement Worth."

Despite these shortcomings in the determination of importance measures, the cross checks in the process provided by the Integrated Decisionmaking Panel and the requirement to compute the overall changes in CDF and LERF make the categorization process robust enough to proceed with trial use of RG 1.201. We would like to review insights gained from the trial use period.

Mr. Stephen L. Rosen did not participate in the development of this report.

Sincerely,

Mario V. Bonaca
Chairman

References:

1. Memorandum dated May 17, 2004, from Catherine Haney, NRR, to John T Larkins, Executive Director, ACRS, Subject: Final Rule - Section 50.69 "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Plants."
2. Staff Requirements Memorandum dated March 28, 2003, from Annette Vietti-Cook, Secretary, to Chairman Diaz, NRC, Subject: Staff Requirements - SECY-02-0176 - Proposed Rulemaking to add new Section 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components."
3. "Guidelines on Modeling Common-Cause Failures in Probabilistic Risk Assessment," NUREG/CR-5485, November 1998.
4. EPRI Technical Report 1008905, "Parametric Uncertainty Impacts on Option 2 Safety Significance Categorization," Final Report, June 2003.
5. 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.
6. 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."

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

July 13, 2004

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

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

During the 514th meeting of the Advisory Committee on Reactor Safeguards, July 7-9, 2004, the Committee considered the draft final revision to 10 CFR 50.55a, which incorporates by reference the 2001 Edition and the 2002 and 2003 Addenda of Division 1 of Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code; the 2001 Edition and the 2002 and 2003 Addenda of Division 1 of Section XI of the ASME BPV Code; and the 2001 Edition and the 2002 and 2003 Addenda of the ASME *Code for Operation and Maintenance of Nuclear Power Plants*. The Committee decided not to review the draft final revision to 10 CFR 50.55a.

Reference:
Memorandum dated June 24, 2004, from R. William Borchardt, Deputy Director, NRR to John T. Larkins, Executive Director, ACRS, Subject: Advisory Committee on Reactor Safeguards Review of Routine Title 10 of the Code of Federal Regulations 50.55a Rulemaking

cc: A. Vietti-Cook, SECY
W. Dean, OEDO
D. Weaver, OEDO
C. Paperiello, RES
A. Levin, RES
W. Norris, RES
J. Dyer, NRR
D. Matthews, NRR
M. G. Crutchley, NRR
S. Tingen, NRR
H. Tovmassian, NRR

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

July 15, 2004

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: DEFERRAL OF ACRS REVIEW OF DRAFT REGULATORY GUIDE, DG-1128, "CRITERIA FOR ACCIDENT MONITORING INSTRUMENTATION FOR NUCLEAR POWER PLANTS," (REVISION 4 TO REGULATORY GUIDE 1.97)

On June 16, 2004, Michael Mayfield wrote a memorandum that requested that the Advisory Committee on Reactor Safeguards consider deferral of its review of the subject draft regulatory guide, until after public comments had been requested and received by the staff.

The Committee considered the subject request at its 514th meeting and it agrees with the staff recommendation that its review can be deferred until after the public comments have been received, and addressed by the staff.

cc: A. Vietti-Cook, SECY
W. Dean, OEDO
D. Weaver, OEDO
C. Paperiello, RES
D. Tifft, RES
M. Mayfield, RES

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

July 19, 2004

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

SUBJECT: PROPOSED DRAFT FINAL GENERIC LETTER ON POTENTIAL IMPACT OF DEBRIS BLOCKAGE ON EMERGENCY RECIRCULATION DURING DESIGN BASIS ACCIDENTS AT PWRs

Dear Mr. Reyes:

During the 514th meeting of the Advisory Committee on Reactor Safeguards on July 7-9, 2004, we reviewed the staff's proposed draft final generic letter (GL) on the potential impact of debris blockage on emergency recirculation during design basis accidents at pressurized water reactors (PWRs). Our Subcommittee on Thermal-Hydraulic Phenomena reviewed this matter during a meeting held on June 22-23, 2004. During these reviews, we had the benefit of discussions with the NRC staff and its contractors, industry representatives, and members of the public. We also had the benefit of the documents referenced.

RECOMMENDATIONS

1. A generic letter should be issued for implementation.

2. The staff should continue confirmatory research in areas where the technical basis of the guidance is uncertain, and on issues such as chemical and downstream effects that are not directly addressed by the guidance proposed by the Nuclear Energy Institute (NEI).

DISCUSSION

In our letter of February 20, 2003, we recommended issuance of the draft GL for public comment in order to initiate the process of gathering plant-specific information and requiring licensees to develop plans for resolving issues associated with potential sump screen blockage following a loss-of-coolant accident (LOCA).

We believe that a final GL should be issued to provide a consistent regulatory basis for action by the nuclear industry to ensure effective long-term reactor core cooling, in light of recent developments in the mechanistic understanding of the important phenomena.

We have not seen a final version of the GL. We understand that the staff is working on the details of the appropriate regulatory process, without changing the intent to resolve the technical issues expeditiously and practically.

The responses of licensees are likely to follow the guidance prepared by NEI and currently under review by the staff. We debated whether it was appropriate to issue a GL prior to completing the final NEI guidance document and the associated Safety Evaluation Report. We have concluded that issuing a GL now will enable licensees to start the process of gathering information, planning activities, and performing preliminary analysis in anticipation of more complete analysis when guidance is available. It is important that the guidance be available in a timely fashion. We understand that it will be presented to our Thermal-Hydraulic Phenomena Subcommittee in August and to the full Committee in September 2004. At that time, the issue of chemical effects on screen blockage, which is not addressed in the NEI guidance, will not be resolved. We expect that the description of the risk-informed options in this guidance will be complete, including a clear description of what is meant by "mitigation capability" for breaks above a risk-justified transition break size. We look forward to discussing these issues with NEI and the staff.

The staff and the industry are also pursuing procedural approaches to make use of the robustness and adaptability of the installed systems to help ensure long-term cooling in the event of diminished recirculation pump capability. Such approaches, as suggested in our report of September 30, 2003, are an important aspect of risk-informing the sump blockage issue. They may significantly reduce the risk associated with sump screen blockage, an effect which should be expressed in terms of quantitative measures so that its effectiveness can be assessed.

Some technical issues raised by us previously have not yet been fully resolved. Research, such as the currently ongoing investigation of chemical effects and studies of downstream effects, is needed for the final resolution of the sump blockage issue. The staff should also initiate any research necessary to confirm that the guidance used by the licensees is adequate. For example, there are still several different models for the "zone of influence." We have questioned the technical basis for these models. Coupling an existing computational fluid dynamic code, such as FLUENT, to the steam-water properties available as a subprogram, would make it possible to model the homogeneous supersonic jets issuing from various break geometries. This work would show the shock wave patterns and mechanisms for energy dissipation and would be very helpful in evaluating the simplified zone of influence models.

Sincerely,

Mario V. Bonaca
Chairman

References:

1. Proposed revised draft Generic Letter provided to ACRS on July 7, 2004 (Predecisional).
2. NRC Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors", June 9, 2003.
3. Proposed Generic Communication: "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors", 69 Fed. Reg 16,980, March 31, 2004.
4. Proposed revised draft Generic Letter provided to ACRS on June 15, 2004 (Predecisional).
5. Public Comment received from the Westinghouse Owners Group, May 27, 2004.
6. Public Comment received from Winston & Strawn (NUBARG), June 1, 2004.
7. Public Comment received from NEI, June 1, 2004.
8. Public Comment received from TVA, May 28, 2004.
9. Public Comment received from FPL, June 1, 2004.
10. Public Comment received from Duke Power, June 1, 2004.
11. Public Comment received from Westinghouse, May 27, 2004.
12. Public Comment received from UCS, May 20, 2004.
13. Public Comment received from Progress Energy, June 1, 2004.
14. Public Comment received from Lanson R. Rogers, March 31, 2004.
15. Public Comment received from Dominion, June 1, 2004.
16. Public Comment received from STARS, June 2, 2004.

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

July 20, 2004

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

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE WESTINGHOUSE ELECTRIC COMPANY APPLICATION FOR CERTIFICATION OF THE AP1000 PASSIVE PLANT DESIGN

Dear Chairman Diaz:

During the 514th meeting of the Advisory Committee on Reactor Safeguards, July 7-9, 2004, we completed our safety review of the Westinghouse Electric Company application for certification of its AP1000 passive plant design. This report is intended to fulfill the requirement of 10 CFR 52.53 that "the ACRS shall report on those portions of the application which concern safety." During our reviews, we had the benefit of discussions with representatives of Westinghouse, its consultants, and the NRC staff. We also had benefit of discussions with a member of the public and of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

The AP1000 design is robust and there is reasonable assurance that it can be built and operated without undue risk to the health and safety of the public.

DISCUSSION

<u>AP1000 Application</u>

On March 28, 2002, Westinghouse tendered its application to the NRC for certification of the AP1000 design. This application was submitted in accordance with Subpart B, "Standard Design Certification," of 10 CFR Part 52, "Early Site Permits; Standard Design Certification; and Combined License for Nuclear Power Plants," and Appendix O, "Standardization of Design: Staff Review of Standard Designs." The application was docketed on June 25, 2002 and assigned Docket No. 52-006.

The application consists of the AP1000 Design Control Document (DCD) and the AP1000 probabilistic risk assessment (PRA) report. The applicant originally submitted the AP1000 DCD on March 28, 2002. The DCD Tier 1 information contains inspection, tests, analyses and acceptance criteria (ITAAC), and Tirer 2 information describes design of the facility. Design certification is sought for the power generation complex, excluding those elements and features considered site specific. All safety-related structures, systems, and components (SSCs) are located on the nuclear island and are to be included in the design certification.

Three aspects of the plant design (instrumentation and control systems, human factors engineering, and some piping) will be completed by the combined license (COL) applicant using the design processes described in the DCD and ITAAC. A fourth aspect related to assurance of long-term recirculation cooling following a LOCA will be confirmed by the COL applicant using NRC guidance that is approved at that time.

The staff issued a draft safety evaluation report (DSER) on June 16, 2003, and an advanced copy of final safety evaluation report on May 25, 2004.

AP1000 Design Description

The AP1000 design is similar in concept to the AP600 design, but provides much higher power levels. To accommodate the higher power [1000 Mwe for AP1000 compared to 600 Mwe for AP600], the following systems and components were increased in size and/or capacity for AP1000 over those of AP600:

- Core length and number of assemblies
- Key NSSS components
  - height of reactor vessel
  - steam generators
  - canned motor reactor coolant pumps
  - pressurizer
- Containment height (volume)
- Capacities of passive safety system components
- Automatic depressurization system (ADS) stage-4 squib valve
- Turbine capacity

As was the case for AP600, the AP1000 design is intended to meet the safety requirements and goals defined for advanced light water reactors with passive safety features specified in the Electric Power Research Institute Utility Requirements document. The plant consists of five principal structures: the nuclear island, the turbine building, the annex building, the diesel generator building, and the radwaste building. The nuclear island includes all safety-related and seismic Category 1 structures and is designed to withstand the effects of natural phenomena and postulated events. It consists of a containment building, a concrete shield building, and an auxiliary building.

The containment building consists of a free-standing, 1¾ inch thick steel containment vessel which has a total free volume of about 2 million cubic feet and a design pressure of 59 psig. The vessel performs the function of limiting the release of radioactivity to the atmosphere for postulated design basis accidents and is part of the passive containment cooling system. In the event of an accident, the passive containment cooling system releases water which runs down the outside surface of the containment vessel to enhance heat removal.

The shield building comprises the structure and annulus area that surrounds the containment vessel. The annulus is configured to complete the passive containment cooling function by providing for natural convection of the outside air up along the containment vessel and out the top.

The auxiliary building is designed to provide protection and separation for the seismic Category 1 mechanical and electrical equipment located outside the containment building. The building also provides protection for safety-related equipment against the consequences of internal and external events. The main control room, Class 1E I&C systems, Class 1E electrical systems, and reactor fuel handling areas are contained in the auxiliary building.

The turbine building houses the main turbine generators and associated fluid and electrical systems. The annex building includes the health physics area, the technical support center, access control, and personnel facilities. The diesel generator building houses two diesel generators and their associated support systems. The radwaste building contains facilities for handling, processing, and storing radioactive waste.

The overall plant arrangement utilizes building configuration and structural designs to minimize the building volumes and quantities of bulk materials (concrete, structural steel, rebar) consistent with design basis safety, operational, maintenance, and structural needs. The plant arrangement provides separation (generally by concrete walls) between safety and non-safety equipment to preclude adverse interactions among them. Separation between redundant safety equipment provides confidence that the safety functions can be performed.

The ITAAC program is intended to ensure that the plant, when built, will conform to the design parameters and assumptions that existed at the time of design certification.

Safety Enhancement Features

The AP1000 design has enhanced safety features similar to those in the AP600 design. These include an improved reactor core design, a large reactor vessel, a large pressurizer, an in-containment refueling water storage tank (IRWST), an automatic depressurization system, a digital microprocessor-based I&C system, hermetically sealed canned motor coolant pumps mounted to the steam generators, and increased battery capacity. The AP1000 design has a defense-in-depth provision for external flooding of the reactor vessel which is intended to provide for in-vessel retention of any accident-induced core melt. The reactor vessel has no bottom head penetrations. This both reduces the potential for a LOCA that would quickly drain the vessel and enhances natural circulation cooling via the cavity flood water.

The AP1000 safety approach is to credit only passive systems to meet all the design basis accident (DBA) requirements with only a one time realignment of valves. Available active pumps, diesels, AC power, cooling water, HVAC, I&C, etc., are not required. The active non-safety-related systems support normal operation and minimize challenges to the passive safety systems. Although these systems are not credited in the safety evaluation case, they provide additional defense-in-depth. The regulatory treatment of nonsafety systems (RTNSS) process was used to impose special requirements on some nonsafety systems to ensure, with high confidence, that they would be available when needed. These systems provide redundancy and diversity that contribute to achieving very low values for core damage frequency (CDF) and large release frequency (LRF).

Probabilistic Risk Assessment

The AP1000 design certification application included a PRA in accordance with regulatory requirements. This PRA was done well and rigorous methods were used. We found that this PRA was acceptable for certification purposes. The mean estimates of the risk metrics are:

CDF:.......... 5 E-07/yr
LRF:........... 6 E-08/yr

These risk metrics are well within the agency's expectations for advanced plants. The fact that the PRA was an integral part of the design process was significant to achieving this estimated low risk.

ACRS Review Approach

The ACRS review activities for the AP1000 design certification are listed in the appendix to this report. These activities should be viewed in concert with all the ACRS review activities conducted for certification of the AP600 design. The AP1000 design is similar in concept to the AP600 design. Consequently, the ACRS's approach to reviewing the certification application for the AP1000 design was to become familiar with the changes from the AP600 design made to accommodate the increased power level compared to AP600 and to assure ourselves that these did not pose any new safety considerations or result in an unacceptable increase in risk. As part of this approach, the new phenomena identification and ranking table (PIRT) was reviewed to see if any new phenomena were identified and if there were any significant changes in rankings of events and phenomena identified for AP600. The new AP1000 scaling analyses were also reviewed to determine what portions of the AP600 test and analysis program were directly applicable to the AP1000 design.

We concluded that most of the previous AP600 review findings were applicable to the AP1000 design. This conclusion greatly enhanced the efficiency of the reviews of the AP1000 safety assessments. We had a number of subcommittee and full- Committee meetings to review the AP1000 as listed in the Appendix. Our reviews did not address security related issues.

During these reviews, we issued three letters identifying issues of concern and areas for which we needed additional discussions. We agreed with the staff's proposed resolutions [Reference 8] of all but two of these issues. In the case of the "in-vessel retention" and "organic iodine production" issues, we developed our own arguments for the resolution. Thus, all ACRS issues have been resolved.

We also discussed the concerns expressed by a member of the public. Most of these are "process" related and are within the purview of the staff. We considered one technical item, the effect of solar heating on the passive containment cooling system's ability to deal with design basis accidents. Westinghouse has assumed a conservative containment cooling water temperature of 120 °F and an air inlet temperature of 115 °F. They have also proposed a 120 °F technical specification limit on the containment cooling water temperature. We find these values to be sufficiently conservative for design basis evaluation.

The ACRS reviewed the NRC staff's safety evaluation and concluded that the staff had done a comprehensive and competent assessment of the compliance of the AP1000 design with the required regulations.

Sincerely,

Mario V. Bonaca

Mario V. Bonaca
Chairman

References: See next page

References:
1. U.S. Nuclear Regulatory Commission, "Draft Safety Evaluation Report Related to Certification of the AP1000 Standard Design" Volumes 1 and 2, dated June 16, 2003.

2. U.S. Nuclear Regulatory Commission, "Advanced Copy of the Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design" Volumes 1 and 2, dated May 25, 2004 (Predecisional Information).

3. Westinghouse Electric Company, AP1000 Design Control Document (DCD), APP-GW-GL-700, Revision 11, dated May 20, 2004.

4. Report dated July 23, 1998, from R. L. Seale, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: The Safety Aspects of the Westinghouse Electric Company Application for Certification of the AP600 Passive Plant Design.

5. Letter dated June 21, 2000, from J. T. Larkins, Executive Director, ACRS, to W. D. Travers, Executive Director for Operations, NRC, Subject: AP1000 Pre-Application Review.

6. Report dated March 14, 2002, from G. E. Apostolakis, Chairman, ACRS, to Richard A. Meserve, Chairman, NRC, Subject: Phase 2 Pre-Application Review for AP1000 Passive Plant Design.

7. Letter dated March 17, 2004, from M. V. Bonaca, Chairman, ACRS, to W. D. Travers, Executive Director for Operations, NRC, Subject: ACRS Reviews of the Westinghouse Electric Company Application for Certification of the AP1000 Plant Design-Interim Letter.

8. Letter dated May 18, 2004, from W. D. Travers, Executive Director for Operations, NRC, to M. V. Bonaca, Chairman, ACRS, Subject: Response to ACRS Interim Letter Regarding the ACRS Reviews of the Westinghouse Electric Company Application for Certification of the AP1000 Plant Design.

9. Letter dated April 30, 2004, from R. P. Vijuk, Manager, Westinghouse Electric Company, to U. S. Nuclear Regulatory Commission, Subject: Westinghouse Responses to ACRS Open Items.

10. Letter dated May 11, 2004, from R. P. Vijuk, Manager, Westinghouse Electric Company, to U.S. Nuclear Regulatory Commission, Subject: Transmittal of Revised Responses to AP1000 DSER Open Items.

11. Letter dated July 30, 2003, from S. G. Sterrett, Assistant Professor, Duke University, to ACRS Subcommittee on Future Plant Designs, Subject: AP1000 Fluid Systems Design & QA Procedures.

12. Letter dated July 31, 2003, from S. G. Sterrett, Assistant Professor, Duke University, to ACRS Subcommittee on Future Plant Designs, Subject: Heat of Solar Radiation and AP1000 Ultimate Heat Sink.

13. Letter dated April 20, 2004, from J. E. Lyons, Program Director, NRC, to S. G. Sterrett, Assistant Professor, Duke University, Subject: Response to Concerns About The AP1000 Design Certification.

14. Letter dated July 8, 2004, from S. G. Sterrett, Assistant Professor, Duke University, to ACRS Members, J. P. Segala (AP1000 Project Manager, NRC), and J. E. Lyons (Program Director, NRC), Subject: NRC Response to Concerns About AP1000 Design Certification.

APPENDIX

CHRONOLOGY OF THE ACRS REVIEW OF THE WESTINGHOUSE APPLICATION FOR THE AP1000 PASSIVE PLANT DESIGN CERTIFICATION

The extensive ACRS review of the AP1000 design and its interactions with representatives of the NRC staff and Westinghouse are discussed in the minutes and transcripts of the following ACRS meetings.

ACRS MEETING/DATES	SUBJECT
475th ACRS Meeting 8/29-9/1/2000	Issues identified during AP1000 pre-application Review (Phase 1)
Thermal-Hydraulic Phenomena 3/15/2001	Westinghouse proposed approach to address AP1000 thermal-hydraulic issues
481th ACRS Meeting 4/5-7/2001	Thermal-Hydraulic issues associated with the AP1000 passive plant design
Thermal-Hydraulic Phenomena 2/13-14/2002	Phase-2 pre-application review; application of AP600 test programs and analyses codes to AP1000 design
Future Plant Designs 2/14-15/2002	Phase-2 pre-application review; use of design acceptance criteria (DAC), and regulatory exemptions for AP1000 design
490th ACRS Meeting 3/7-9/2002	Phase-2 pre-application reviews of the AP1000 Design
497th ACRS Meeting 11/7-9/2002	Westinghouse AP1000 design-review schedule
Reliability and Probabilistic Risk 1/23-24/2003	Reliability of AP1000 design components; ADS-4 squib valve function
499th ACRS Meeting 2/6-8/2003	PRA Subcommittee Chairman report on the PRA for AP1000 design
Thermal-Hydraulic Phenomena 3/19-20/2003	Review of liquid entrainment issue for AP1000 design
501st ACRS Meeting 4/10-12/2003	Thermal-hydraulic Subcommittee Chairman report on thermal-hydraulic issues for AP1000 design
Thermal-Hydraulic Phenomena 7/16-17/2003	Review of sump strainer, boron precipitation, ADS-4 squib valve, and computer codes

Future Plant Designs 7/17-18/2003	Review I&C, man-machine interface, materials, leak-before-break, human factors, and DSER open items
506th ACRS Meeting 10/1-4/2003	Interim review of the AP1000 design, and resolution of open items
Thermal-Hydraulic Phenomena 2/10-11/2004	Thermal-hydraulic issues for AP1000 design, computer codes, and core-level during long-term cooling
510th ACRS Meeting 3/3-6/2004	Interim review of AP1000 design, open items and ACRS concerns
513th ACRS Meeting 6/2-4/2004	NRC staff's response to the ACRS report of March 17, 2004 for the AP1000 design
Future Plant Designs 6/25/2004	Review of NRC staff's response to ACRS concerns, technical issues, any remaining open items, and FSER review
514th ACRS Meeting 7/7-9/2004	Final ACRS review of the AP1000 design

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

September 10, 2004

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: PROPOSED REGULATORY GUIDE 1.75, REVISION 3, "CRITERIA FOR INDEPENDENCE OF ELECTRICAL SAFETY SYSTEMS"

During the 515th meeting of the Advisory Committee on Reactor Safeguards, September 9-11, 2004, the Committee considered Revision 3 to Regulatory Guide 1.75, "Criteria for Independence of Electrical Safety Systems," which incorporates by reference the Institute of Electrical and Electronics Engineering Standard (IEEE Std) 384-1992, "Standard Criteria for Independence of Class 1E Equipment and Circuits," as supplemented. The Committee decided not to review Revision 3 to Regulatory Guide 1.75 and agrees that the staff should continue with its process for issuing the document.

Reference:
Memorandum dated July 29, 2004, from Michael Mayfield, Director, Division of Engineering Technology, RES to John T. Larkins, Executive Director, ACRS, Subject: Proposed Regulatory Guide 1.75, Revision 3, "Criteria for Independence of Electrical Safety Systems."

cc: A. Vietti-Cook, SECY
W. Dean, OEDO
R. Tadesse, OEDO
C. Paperiello, RES
M. Mayfield, RES
N. Chokshi, RES
S. Aggarwal, RES
J. Dyer, NRR
B. Boger, NRR

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

September 14, 2004

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: DRAFT REGULATORY GUIDE, DG-1.XXX, "RISK-INFORMED, PERFORMANCE-BASED FIRE PROTECTION FOR EXISTING LIGHT-WATER NUCLEAR POWER PLANTS"

During the 515th meeting of the Advisory Committee on Reactor Safeguards, September 9-11, 2004, the Committee considered the draft Regulatory Guide DG-1.XXX, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants, and decided not to review it until after public comment." The Committee agrees with the staff's proposal to issue this Regulatory Guide for public comment.

Reference:
Memorandum dated August 11, 2004, from Suzanne C. Black, NRR to John T. Larkins, ACRS, Subject: Draft Regulatory Guide, DG-1.XXX, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants."

cc: A. Vietti-Cook, SECY
W. Dean, OEDO
R. Tedesse, OEDO
J. Dyer, NRR
S. Black, NRR
P. Lain, NRR

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

September 16, 2004

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

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE LICENSE RENEWAL APPLICATION FOR THE DRESDEN 2 AND 3 AND QUAD CITIES 1 AND 2 NUCLEAR POWER STATIONS

Dear Chairman Diaz:

During the 515th meeting of the Advisory Committee on Reactor Safeguards, September 9-11, 2004, we completed our review of the license renewal application for Dresden Units 2 and 3 and Quad Cities Units 1 and 2 Nuclear Power Stations and the final Safety Evaluation Report (SER) prepared by the NRC staff. Our Plant License Renewal Subcommittee also reviewed this matter during a meeting on April 14, 2004. During our review, we had the benefit of discussions with representatives of the NRC staff and Exelon Generation Company, LLC (Exelon). We also had the benefit of the documents referenced.

RECOMMENDATIONS AND CONCLUSION

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

2. The staff should require that, prior to entering the period of extended operation, Exelon conduct an evaluation to ensure that operating experience at extended power uprate (EPU) levels is properly addressed by the aging management programs. The staff should review and approve this evaluation.

3. The steam dryers should be included in the scope of license renewal for Dresden and Quad Cities.

4. The staff should develop guidance to apply Recommendations 2 and 3 to future boiling water reactor (BWR) license renewal applications.

BACKGROUND AND DISCUSSION

This report fulfills the requirement of 10 CFR 54.25, which states that the ACRS review and report on all license renewal applications. Dresden Units 2 and 3 and Quad Cities Units 1 and 2 are General Electric BWR/3 plants, with Mark 1 containments. Exelon requested renewal of their operating licenses for 20 years beyond the current license terms, which expire on December 22, 2009, for Dresden Unit 2 and January 12, 2011, for Dresden Unit 3, and December 14, 2012, for both Quad Cities units. The Dresden units were designed with isolation condensers for core isolation cooling whereas the Quad Cities units have the more typical reactor core isolation cooling system. The Dresden units have a separate shutdown cooling system whereas Quad Cities units have the more typical arrangement in which shutdown cooling is achieved as an operational mode of the residual heat removal system.

Dresden Units 2 and 3 share the site and surrounding areas with Unit 1, a dual-cycle BWR that has been placed in a safe storage condition until Units 2 and 3 are ready for decommissioning. Unit 1 systems, structures, and components (SSCs) which support the operation of Units 2 and 3 are included in the scope of license renewal.

The final SER documents the staff's review of the information submitted by Exelon, including commitments that were necessary to resolve open items identified by the staff in the initial SER. The staff reviewed the completeness of the applicant's identification of SSCs that are within the scope of license renewal; the integrated plant assessment process; the applicant's identification of the plausible aging mechanisms associated with passive, long-lived components; the adequacy of the applicant's aging management programs; and the identification and assessment of time-limited aging analyses (TLAAs) requiring review.

The staff also conducted several inspections at Exelon's engineering offices and at the Dresden and Quad Cities sites to verify the adequacy and implementation of the methodology described in the application. During our Plant License Renewal Subcommittee meeting on April 14, 2004, the staff presented a well structured and effective overview of its inspections and audits.

Dresden and Quad Cities have been approved for an EPU to 2957 MWt. The EPU increased power output by 17% at both Dresden units and 17.8% at both Quad Cities units. Shortly after operating at the EPU level, Quad Cities Unit 2 experienced unexpected high moisture carryover, which was caused by a damaged steam dryer. The applicant repaired the dryer and the unit was returned to service. Shortly thereafter, similar damage was identified in Unit 1 and then again in Unit 2. The applicant has since chosen to operate Quad Cities Units 1 and 2 at pre-EPU power levels until the root cause of the dryer damage can be determined.

Dresden and Quad Cities have limited operating time at the EPU levels and during that period, several problems believed to be associated with the EPU have occurred at Quad Cities. Most notable is the steam dryer damage, but other problems with limit and pressure switches have also occurred. After operation at the higher power level (2957 MWt), but prior to entering the period of extended operation, the staff should require that the applicant conduct an evaluation of operating experience at all four units and at other plants operating at EPU levels, to ensure that this experience has been properly addressed by the aging management programs. The staff should review and approve this evaluation.

Such an evaluation should be required of all license renewal applicants that plan to operate during the license renewal period at EPU levels that are substantially higher than experienced in the bases supporting their license renewal applications. Such an evaluation is consistent with the intent of 10 CFR 54.17(c) that substantial operating experience be reflected in the bases supporting the license renewal application. Current operating experience at EPU levels is limited and an understanding of the impact of operation at such levels on plant components is only now beginning to develop. A recent BWR Owners Group (BWROG) survey of EPU plants has identified component failures, potential decreases in time between failures, unexpected increases in component wear, and a significant number of events, which are all directly attributable to EPUs. It is therefore important that the staff require, after sufficient plant-specific operating experience is available at the EPU level but before entering the period of extended operation, that license renewal applicants ensure applicable operating experience at EPU levels has been properly addressed by aging management programs. The staff should also provide guidance in this area.

During our review, we questioned why certain SSCs were not included in the scope and, in all but one case, the applicant provided appropriate justification for the exclusions. That case was the steam dryers at these plants. Although the dryers are not safety related, structural failure at Quad Cities caused pieces to pass down the main steamlines and some of these pieces were found on the turbine stop valve screens. These pieces passed through the Main Steam Isolation Valves (MSIVs), which are safety related components, and could have caused these valves to fail to operate properly. Since the MSIVs are safety related, this meets the criteria for inclusion of the steam dryers in the scope of license renewal in accordance with 10 CFR 54.4(a)(2). The steam dryers at Dresden and Quad Cities should be included in scope even if an analysis demonstrates that future failures are unlikely. Further, steam dryers for all BWRs seeking license renewal should be within scope based on 10 CFR 54.4(a)(2). The staff should develop guidance on this topic so that the NRC position on this matter is clearly understood by the industry.

The applicant performed a comprehensive aging management review of all SSCs that are within the scope of license renewal. The application describes 48 aging management programs for license renewal, which include existing, augmented, and new programs. As with other applicants, we encouraged Exelon to establish a schedule for implementing license renewal commitments well ahead of the beginning of the license renewal period to preclude placing an unreasonable demand on applicant and NRC resources.

Aging degradation of reactor vessel internals has been an ongoing problem in BWRs. The applicant has committed to programs developed by the BWR Vessel and Internals Project (BWRVIP) to manage these problems. We questioned if the clamping device installed in 1995 to mitigate core shroud cracking had been evaluated for the period of extended operation. The applicant has committed to follow the BWRVIP-76 program, as approved by the staff, to manage aging effects on the core shroud and other hardware. The staff is currently reviewing this program and will evaluate the adequacy of the recommended inspection intervals in light of the accumulating evidence of fluence, strength, and surface effects on cracking of (for example) 316L and the lack of comparable data for XM-19 and X750.

In 1994, Dresden experienced a failure due to corrosion of the carbon steel piping to the instrument air receiver. We questioned whether this corrosion was an ongoing aging effect. The staff responded that this piping has been replaced with stainless steel and that the air receiver was replaced with one having epoxy coating applied to the interior. The moisture separator drain trap was also modified. These modifications should reduce moisture accumulation and corrosion. The applicant has also instituted a periodic blowdown program and no subsequent failures have occurred.

In the license renewal application, Exelon has identified the components that are supported by TLAAs. Their review of the TLAAs shows that the analyzed components have sufficient margin to operate for the period of extended operation.

We reviewed the Reactor Vessel Upper Shelf Energy for limiting plates and welds for all four units. All are acceptable for the period of extended operation, including the most limiting weld at Quad Cities Unit 2, which was found acceptable based on a plant-specific equivalent margins analysis. We are satisfied with the staff's review of the applicant's data and calculations.

The staff is currently reviewing BWRVIP-116, "BWR Integrated Surveillance Program Implementation for License Renewal." The applicant has agreed to a license condition to notify the NRC, before entering the period of extended operation, of their decision to implement either the staff-approved BWRVIP integrated surveillance program (ISP) or a staff-approved plant-specific ISP.

On the basis of our review of the final SER, we agree with the staff's conclusion that all open and confirmatory items have been appropriately closed. We also concur with all three license conditions requiring the applicant to take certain actions before beginning the period of extended operation.

With the inclusion of commitments to perform an evaluation of operating experience at the EPU levels before entering the period of extended operation and to include steam dryers in the scope of license renewal, the application for license renewal for Dresden Units 2 and 3 and Quad Cities Units 1 and 2 should be approved.

Sincerely

Mario V. Bonaca

Mario V. Bonaca
Chairman

References:

1. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the License Renewal of the Dresden Nuclear Power Station, Units 2 and 3 and Quad Cities Nuclear Power Station, Units 1 and 2," July 2004.
2. Exelon Generation Company, "License Renewal Application for Dresden Nuclear Power Station (DNPS), Units 2 and 3, and Quad Cities Nuclear Power Station (QCNPS), Units 1 and 2," January 2003.
3. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report with Open Items Related to the License Renewal of the Dresden Nuclear Power Station, Unit 2 and 3 and Quad Cities Nuclear Power Station, Unit 1 and 2." February 2004.
4. U.S. Nuclear Regulatory Commission Inspection Reports 50-237,249/03-04 and 50-254,265/03-04, NRC License Renewal Scoping/Screening Inspection Report, September 15, 2003.
5. U.S. Nuclear Regulatory Commission Inspection Report 50-237,249/03-10, NRC Aging. Management Program Inspection Report, December 5, 2003.
6. U.S. Nuclear Regulatory Commission Inspection Report 50-254,265/03-14, NRC Aging Management Program Inspection Report, December 5, 2003.
7. Electric Power Research Institute, "BWR Vessel and Internals Project BWR Core Shroud Inspection and Flaw Evaluation Guidelines (BWRVIP-76)," January 2000.
8. Electric Power Research Institute, "BWRVIP-116: BWR Vessel and Internals Project Integrated Surveillance Program (ISP) Implementation for License Renewal," July 2003.

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

October 12, 2004

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

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

During the 516th meeting of the Advisory Committee on Reactor Safeguards held October 7-9, 2004, the Committee considered Draft NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology For Nuclear Power Facilities," for review. The Committee decided not to review the draft document at this time and agrees that the staff should continue with its process for issuing the document for public comment. The Committee would like an opportunity to review the document after reconciliation of public comments.

Reference:
Memorandum dated September 17, 2004, from Charles Ader, Director, Division of Risk Analyses and Applications, RES to John T. Larkins, Executive Director, ACRS, Subject: Draft NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology For Nuclear Power Facilities."

cc: A. Vietti-Cook, SECY
W. Dean, OEDO
R. Tadesse, OEDO
C. Paperiello, RES
C. Ader, RES
J. Dyer, NRR
J. Hyslop, RES

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

October 12, 2004

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: DRAFT FINAL REGULATORY GUIDE DG-1085, "STANDARD FORMAT AND CONTENT OF DECOMMISSIONING COST ESTIMATES FOR NUCLEAR POWER REACTORS," AND NUREG-1713, "STANDARD REVIEW PLAN FOR DECOMMISSIONING COST ESTIMATES FOR NUCLEAR POWER REACTORS"

During the 516th meeting of the Advisory Committee on Reactor Safeguards, October 7-9, 2004, the Committee considered for review draft final Regulatory Guide DG-1085, "Standard Format and Content of Decommissioning Cost Estimates for Nuclear Power Reactors," and draft final NUREG-1713, "Standard Review Plan for Decommissioning Cost Estimates for Nuclear Power Reactors." The Committee decided not to review these documents. As recommended by the Committee, these documents will be forwarded to the Advisory Committee on Nuclear Waste for consideration.

Reference:
Memorandum dated September 24, 2004, from Catherine Haney, Program Director, NRR, to John T. Larkins, Executive Director, ACRS, Subject: Publication of Regulatory Guide DG-1085, "Standard Format and Content of Decommissioning Cost Estimates for Nuclear Power Reactors," and NUREG-1713, "Standard Review Plan for Decommissioning Cost Estimates for Nuclear Power Reactors"

cc: A. Vietti-Cook, SECY
W. Dean, OEDO
R. Tadesse, OEDO
J. Dyer, NRR
C. Haney, NRR
M. G. Crutchley, NRR
C. Pittiglio, NRR
H. Larson, ACNW

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

October 13, 2004

MEMORANDUM TO: Luis. A. Reyes
Executive Director for Operations

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

SUBJECT: SEABROOK STATION, UNIT NO. 1 - ADVISORY COMMITTEE ON REACTOR SAFETY REVIEW OF STRETCH POWER UPRATE AMENDMENT (TAC NO. MC2364)

During the 516th meeting of the Advisory Committee on Reactor Safeguards, October 7-9, 2004, the Committee considered the recommendation made in a memorandum to me dated September 15, 2004, from Ledyard B. Marsh, Office of Nuclear Reactor Regulation, that the Committee not review the stretch power uprate request from the Seabrook Station, Unit No. 1 (Seabrook) licensee. The memorandum characterized the Seabrook power uprate request as essentially a 5.2 percent stretch power uprate with minor modifications. The Committee has decided to accept your staff's recommendation and not review the Seabrook power uprate request.

The Committee does not expect a written response to this memorandum.

cc: A. Vietti-Cook, SECY
W. Dean, OEDO
R. Tadesse, OEDO
J. Dyer, NRR
T. Marsh, NRR
J. Lamb, NRR
T. Harris, NRR
W. Ruland, NRR
L. Raghavan, NRR

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

October 14, 2004

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

SUBJECT: DRAFT NUREG-XXXX, "THE REPORT ON THE INDEPENDENT VERIFICATION OF THE MITIGATING SYSTEMS PERFORMANCE INDEX (MSPI) RESULTS FOR THE PILOT PLANTS"

Dear Chairman Diaz:

During the 516th meeting of the Advisory Committee on Reactor Safeguards (ACRS), October 7-9, 2004, we reviewed draft NUREG-XXXX, "The Report on the Independent Verification of the Mitigating Systems Performance Index (MSPI) Results for the Pilot Plants." During this review, we had the benefit of discussions with the NRC staff. Our Subcommittees on Reliability and Probabilistic Risk Assessment and on Plant Operations reviewed this matter during a meeting held on April 14, 2004. We also had the benefit of the document referenced.

CONCLUSIONS AND RECOMMENDATIONS

The Mitigating System Performance Index is substantially superior to the group of safety system unavailability performance indicators, which it replaces. Draft NUREG-XXXX should be issued, its recommendations should be implemented, and the process for incorporating the MSPI into the Reactor Oversight Process should continue.

DISCUSSION

The Reactor Oversight Process (ROP) was developed to guide the staff in allocating resources to inspection and enforcement activities for power reactor licensees. This allocation of staff resources is based on the use of the ROP Action Matrix, which in turn is developed using risk-informed inspection findings and plant performance indicators (PIs) for the seven cornerstones of licensee performance. Several years ago, the Commission determined that it is desirable to move toward more risk-informed PIs as a way to refine and improve the Action Matrix and the ROP resource allocation process.

Over the last four years, the staff developed a method for calculating a risk-informed system-level MSPI. This MSPI would replace the Safety System Unavailability (SSU) PIs related to the availability of the mitigating and related support systems that were believed to be risk-important in most plants, but not equally so in all plants, depending on the degree of redundancy and other plant design features. The MSPI was based on the concepts of NUREG-1753 and was developed to overcome performance indicator variability caused by plant-specific features and characteristics, thus providing a better measure of mitigating system reliability and availability. The current MSPI formulation is risk-informed and generally plant-specific.

The MSPI addresses known issues with the current indicators by:

- eliminating the use of fault exposure time in the SSU PI
- incorporating an unreliability indicator
- defining unavailability consistently with the Maintenance Rule and the Institute for Nuclear Power Operations/World Association of Nuclear Operators (INPO/WANO) indicators
- incorporating a cooling water support system indicator and eliminating cascading of cooling water support system failures
- incorporating plant-specific thresholds

We commend the staff and industry for developing the MSPIs. These indicators should be implemented promptly.

The industry shared the MSPI development effort with the staff and issued draft NEI 99-02. A pilot program, consisting of data from 20 plants, was conducted in 2002, and evaluated in 2003. During the pilot program, a number of deficiencies were identified and corrected. The subsequent evaluation of the pilot program, described in draft NUREG-XXXX, resulted in six recommendations. These six recommendations address the major technical issues associated with the proposed MSPI formulation and should be implemented. The guidance in the draft Appendix F to NEI 99-02 and the NRC Inspection Manual will need to be modified to incorporate these recommendations.

We have previously provided recommendations on the elements of formulation of the MSPIs. Our recommendations have been incorporated into the current MSPI. The necessary elements required to implement the MSPIs are now fully identified and addressed. We recommend that the necessary documents be revised and issued and the MSPI program be initiated.

Sincerely,

Mario V. Bonaca

Mario V. Bonaca
Chairman

Reference:

Draft Report on the Independent Verification of Mitigating Systems Performance Index Results for the Pilot Plants, NUREG-XXXX, August 2004.

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

October 14, 2004

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

SUBJECT: REVIEW OF ACR-700 PRE-APPLICATION SAFETY ASSESSMENT REPORT

Dear Mr. Reyes:

During the 516th meeting of the Advisory Committee on Reactor Safeguards, October 7-9, 2004, we met with representatives of the NRC staff and Atomic Energy of Canada Limited (AECL) Technologies to discuss the staff's pre-application safety assessment report (PASAR) for the 700 MWe advanced CANDU reactor (ACR-700). We also had the benefit of the document referenced.

CONCLUSION

The staff has done an excellent job on its pre-application review of the "focus topics" for which the staff identified technical, regulatory, and policy issues. We agree with the staff's assessment of these issues. The PASAR will provide excellent guidance for the subsequent certification process.

DISCUSSION

We have used the PASAR to help identify areas for our own review in order to satisfy the statutory requirement of performing a safety review of any design certification. These areas are listed below. The list is provided to alert the applicant and staff to the issues of most interest to ACRS at this time and to help in scheduling our reviews.

1. Thermal Hydraulics: Analysis of design basis accidents (DBAs) and development of acceptance criteria; experimental validation of the CATHENA computer code.

2. Reactor Kinetics: Determination of the coolant void reactivity (CVR) coefficient; uncertainty analysis and acceptance criteria for CVR.

3. PRA: Probabilistic risk assessment quality; risk-informed selection of DBAs; relationship among DBA, limited core damage, and severe accidents; risk acceptance criteria.

4. Severe Accidents: Research on core melt progression, heat transfer, and fission product release and speciation; results of ACR-700 simulations using the MELCOR computer code; the staff's assessment of the MAAP4-CANDU computer code; fuel/coolant interaction experiments, modeling, and accident progression; severe accident source term phenomenology; hydrogen catalytic converters; hydrogen stratification in containment.

5. Long-Term Cooling: Provisions for long-term cooling following a LOCA.

6. Software: Functionality and reliability of software; PRA treatment of software failures.

7. Materials: Materials monitoring methods for complex piping systems; delayed hydride cracking and irradiation-enhanced creep in zirconium alloys; fatigue behavior of zirconium alloys; degradation kinetics; dissimilar metal weld anomalies.

8. Fire Protection: Compliance with U.S. regulatory requirements; fire risk assessment.

9. On-line Refueling: Regulatory requirements and risk implications.

We look forward to continuing to participate in the certification review of the ACR-700.

Sincerely,

Mario V. Bonaca

Mario V. Bonaca
Chairman

Reference:

U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Pre-application Safety Assessment Report Related to the Advanced CANDU Reactor 700 MWe, dated September 2004.

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

October 18, 2004

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

SUBJECT: SAFETY EVALUATION OF THE INDUSTRY GUIDELINES RELATED TO PRESSURIZED WATER REACTOR SUMP PERFORMANCE

Dear Chairman Diaz:

During the 516th meeting of the Advisory Committee on Reactor Safeguards, October 7-9, 2004 we met with representatives of the NRC staff, its contractors, and the industry to review the staff's draft safety evaluation (SE) related to Nuclear Energy Institute (NEI) Guidance Report (proposed document no. NEI 04-07), "Pressurized Water Reactor Sump Performance Evaluation Methodology" (Ref. 1 and 2). The guidance report and the associated SE are intended to describe a methodology that is acceptable to the staff for use by licensees in responding to Generic Letter (GL) 2004-02 (Ref. 3). Our Subcommittee on Thermal-Hydraulic Phenomena reviewed this matter during a meeting on September 22-23, 2004. We also had the benefit of the documents referenced.

Recommendations

1. The SE should not be issued in its present form. Both it and the NEI guidance contain too many technical faults and limitations to provide the basis for a defensible and robust long-term solution to Generic Safety Issue (GSI) 191, "Assessment of Debris Accumulation on Pressurized Water Reactor (PWR) Sump Performance."

2. The faults and limitations in the present technical knowledge base need to be addressed so that acceptable guidance can be developed. The staff should develop sufficient understanding to determine either the uncertainty or the degree of margin resulting from the application of the methodology.

3. If licensees are to be responsible for filling gaps in the analytical and experimental data base, the staff should clearly state the agency's expectations for the necessary quality and acceptability requirements.

4. The risk-informed approach should be extended to treat the entire sequence of phenomena that lead from the break to the end effects on the pump net positive suction pressure (NPSH) and thus the effectiveness of recirculation cooling. This would provide a technical basis for application of the Regulatory Guide (RG) 1.174 process. It will require a quantitative assessment of model uncertainties related to the physical phenomena.

Discussion

GSI-191 is concerned with long-term cooling of the reactor core following a loss-of-coolant accident (LOCA). In the later stages of the accident scenario, water is drawn from the sump and recirculated through the core. The pumps that recirculate this water are protected by screens. There is concern that debris generated by the accident might accumulate on the screens sufficiently to compromise the NPSH of the pumps.

The staff has taken several steps in its efforts to resolve GSI-191.

In Bulletin 2003-01(Ref. 4), which was issued on June 9, 2003, the staff requested that licensees "confirm their compliance with 10 CFR 50.46(b)(5) and other existing applicable regulatory requirements."

Regulatory Guide 1.82, Revision 3, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," was issued in November 2003. As we pointed out in our letter of September 30, 2003 the RG presented many requirements for analyzing the phenomena affecting sump performance, but provided little guidance on how to do the analysis.

In GL.2004-02, dated September 13, 2004, the staff requested licensees "to perform an evaluation of the ECCS and CSS recirculation functions in light of the information provided in this letter and, if appropriate, take additional actions to ensure system function."

The response of licensees to this GL depends on their ability to assess the constituent phenomena and their effect on the pump NPSH. Guidance on how to make these assessments is the subject of the NEI submittal and the staff's SE.

The staff has made an effort, both in its research programs and in its review process, to develop sufficient knowledge and judgment to evaluate the adequacy of these assessments. The staff believes that the NEI methodology, as modified by the staff in the draft SE, provides a conservative basis for evaluating PWR sump blockage. It is our judgment that too many gaps remain for a technically defensible resolution at this time. The variety of technical challenges is large and there are unknowns and gaps in the knowledge base. For example, some basic methods include equations that contain incorrect physical descriptions of the phenomena. There are also questions about the extent of the supporting test data and the data's interpretation, and the guidance on implementing the proposed methods is vague and contains inconsistencies.

Risk-Informed Approach

In our report of September 30, 2003, we recommended that the staff investigate a risk-informed approach to sump screen blockage. Such an approach is presented in Section 6.0 of the NEI guidance report and is essentially endorsed in Section 6.0 of the SE.

Although we welcome the use of risk in this context, it is not clear that the NEI approach, as modified by the SE, will have a significant practical impact. Plants will still have to compute the consequences of a large-break LOCA with respect to debris generation, transport, and deposition. It appears that there will only be a significant difference in what licensees may be required to implement if the requirement for "mitigation" is somehow related to risk, as well as to the assumptions that surround the traditional 10 CFR 50.46 LOCA requirements. Thus, we are recommending that the risk-informed approach be extended.

Besides the complex technical issues that make resolution difficult, the present deterministic requirements of 10 CFR 50.46 make it difficult for licensees to demonstrate compliance. The process of risk analysis was specifically developed to handle complex situations where uncertainties exist. We are recommending that risk information be developed and that the model uncertainties be quantified in the representation of the phenomena described in RG 1.82, Rev. 3. This would provide a technical basis for application of the RG 1.174 process.

Technical Errors

We were surprised to find significant technical errors of a fundamental nature in the analytical knowledge base supporting the guidance.

For example, the zone-of-influence (ZOI) model is based on the ANSI/ANS 1988 standard. There are several inconsistencies and errors in the models described in the standard, as discussed in References 7 and 9. The "impingement pressure" is undefined. The assumed flow pattern does not correspond to observed and computed patterns for supersonic jets. The conditions in a free jet and in a jet impinging on a large target appear to be mixed up. The analysis of the area of an "asymptotic plane" is based on an unrealistic representation of the physics and inappropriate one-dimensional approximations which can be used to calculate a variety of results, spanning a factor of four. In addition, the density at this fictional "asymptotic plane" is evaluated as if the fluid were at rest, whereas in reality it is flowing at a high Mach number.

We also identified significant basic errors in the NUREG/CR-6224 correlation and its use for evaluating head loss. These are described in References 5, 6, and 8.

Incomplete or Confusing Guidance

A "thin bed" is invoked at several places in the NEI guidance report and in the SE. There is no clear definition of what it is, how to predict its occurrence or its effects for different combinations of particulates and fibers, or how to "substantiate no formation" of it.

This is an example of an area where we consider the guidance to be confusing and inadequate. Appendix VIII of the SE addresses the thin bed effect in the context of calcium silicate (CalSil). The definition given is:

> "The thin-bed effect refers to the debris bed condition in a fibrous/particulate bed of debris whereby a relatively high head loss can occur due to a relatively thin layer of debris, by itself or embedded as a stratified layer within other debris, because the bed porosity is dominated by the particulate and the bed porosity approaches that of the corresponding particulate sludge."

This definition is qualitative, but it clarifies some uncertainties in the SE. If this thin layer can occur anywhere in the bed, it might, for example, form on top of a thick (say four inch) layer of fiberglass. In contrast, the guidance described in Appendix V of the SE appears to apply only to uniform beds of debris.

The final phrase in the definition is conjecture since there appears to be no definitive evidence of the cause of the phenomenon. In Los Alamos National Laboratory (LANL) test 6H, which is discussed in Appendix V, there was an anomalous head loss that increased by an order of magnitude over the course of two hours while the flow rate was kept constant. Though the correlation could be made to fit this single data point by adjusting the specific surface area to the unusually high value of 800,000 ft^2/ft^3, this process provides no basis for extrapolation to other conditions.

Although this appendix provides some discussion of what might be the cause of the thin bed effect, it is speculative. Page 9 of Appendix VIII appears to discuss experiments performed in the recent past and not formally reported. It is clearly a work-in-progress and is not sufficiently mature to form the basis of guidance that could have a large impact on all plants containing CalSil insulation. It is also unclear if the effect could occur with certain latent debris or with the debris from destroyed coatings, as well as with CalSil.

Another example of confusing guidance occurs on page vi of the executive summary in the SE, where the staff imposes the following exception:

> "The [NEI guidance report] does not provide guidance for those plants that can substantiate no thin bed effect, which may impact head loss results and limiting break conditions."

It appears that plants with any CalSil insulation could have the thin bed effect and therefore the methodology will predict clogging of their screens. Plants that do not have CalSil cannot use the guidance.

There is some discussion of the thin bed effect on page 69 of the main text of the SE. It is stated in the SE that the head loss model is valid for thicknesses larger than 0.125 inch.

> "Below this value, the bed does not have the required structure to bridge the strainer holes and filter the sludge particles."

This apparently indicates that one does not have to worry about beds with a thickness less than 0.125 inch thick. In other words, 0.125 inch is a criterion for a thick bed, as it defines a minimum thickness. On this basis, thin beds should be acceptable.

At the bottom of page 69, it is stated that calcium silicate can form a bed without supporting fibers, which contradicts the above statement.

On page 70 we find that "sufficient conservatism should be used in estimating the quantities of fibrous debris available to form a thin bed." Does this mean that one should assume that no fibers are required? Since all plants have fibers of some sort, even in latent debris, do they all exhibit the thin bed effect?

It is clear that the qualitative discussions in Appendix VIII of the SE need to be replaced by consistent and unequivocal guidance on how to deal with the possibility that a layer, or layers, in the deposit on a screen may result in a particularly high head loss. This needs to be supported by definitive experiments. Loose use of the term thin bed in the present version of the SE merely confuses an already uncertain situation.

Another example of inappropriate guidance is the staff's agreement with NEI that it is conservative to assume uniform debris accumulation on all types and orientations of screens. This appears to be a step of faith. It does not consider the possibility of a thin bed over part of the screen, or the layering of debris to form a thin bed as one of the strata in the accumulated layers. Anomalous and unexplained phenomena, which may increase the head loss by an order of magnitude, have been observed in tests with CalSil, and may be due to nonuniform effects of this sort.

If a stratified layer can form within a thick bed, such a layer could form somewhere in the strata on the screen and cause high enough head loss to challenge the NPSH in any plant with enough calcium silicate insulation to provide a layer comparable to the layer in LANL's test 6H, namely 0.018 inch. This amounts to about a gallon of CalSil on a 100-square-foot screen. Since the fine particles of calcium silicate are readily transportable to the screen, this would essentially mean that no plant could tolerate use of any amount of this insulation.

Examples of Gaps in the Empirical Knowledge Base

Coatings

The effect on coatings of a two-phase jet issuing from the break is not well understood. While NEI suggests a damage pressure of 1000 psi for qualified coatings, the staff adopts the default conservative requirement that the zone of influence (ZOI) for coatings be a sphere with radius equal to ten times the break diameter. The staff also requires that all unqualified coatings within the containment be assumed to be destroyed and entrained. These requirements appear to be arbitrary assumptions. They are said to be "conservative" but no rationale has been offered to support this claim. It appears that the requirements lead to the prediction of large amounts of particulate matter being generated and transported to the screens.

The nature and effects of coating debris are unknown. There is no guidance on how to compute head loss for coating debris, whether or not a thin bed effect will occur, or whether coatings are truly particulate, or actually flakes. If they are particulate, then supposedly a correlation such as NUREG/CR-6224 (if it were to be corrected) could be used. However, if the coatings are flakes, then a new model would have to be developed to account for their potential to behave like leaves on a street drain, overlapping and obstructing the flow paths in a way that is not described by the usual "specific surface area" models

Debris transport

There are many uncertainties in the modeling of debris transport. For example, one of the staff's assumptions, allowing only 15% of the debris to be held up in inactive pools, is based on model predictions for one specific plant and leads to the conclusion that much of the debris will reach the pool. This conclusion may not apply to other plants. A method to perform plant-specific calculations is sketched in very general terms but its implementation would be difficult.

Head Loss Correlations

While the NUREG/CR-6224 correlation, if it were to be modified to remove the technical errors, might be approved for use, the database does not contain enough data to determine the parameters to use in the correlation under realistic conditions. Licensees are required to develop their own results for latent debris and coatings. For other materials, licensees are required to "ensure that [the correlation] is applicable." Since the database for several materials, such as CalSil, does not cover all plant conditions, licensees may have to carry out new experiments to obtain definitive data. We are recommending that the staff clearly state the agency's expectations for the necessary quality and acceptability requirements for these experiments.

Appendix V of the SE provides additional guidance on the use of the NUREG/CR-6224 correlation. The formulae provided for computing specific surface area are based on the assumption that the debris bed is uniformly mixed, which may not be the case following a LOCA. There are several mechanisms that may create dense layers that contribute to higher head loss than would be computed using uniform mixing, as discussed in Appendix VIII. Table V-5 provides validation ranges of sources for fibrous insulation debris. No assessment is made of applicability to the LOCA context or of the very sparse data from some of the sources. For instance, an applicant who chooses to use the highest value of specific surface area for CalSil deduced in LA-UR-04-1227 (which can lead to an order of magnitude increase in head loss) will be basing all its predictions on a single, isolated, data point obtained at a single flow rate and temperature, with a single thickness, composition, and mode of formation of the bed, none of which may be representative of plant conditions.

Chemical and Downstream Effects

No definite guidance is provided for evaluating either chemical or downstream effects, both of which have may become of major importance as more knowledge is acquired.

The Committee will continue to work with the staff to develop solutions to the sump screen blockage issue.

Sincerely,

Mario V. Bonaca

Mario V. Bonaca
Chairman

Additional Comments from ACRS Members Graham B. Wallis and F. Peter Ford

We agree with the recommendations of our colleagues. The SER and NEI guidance documents contain too many technical faults and limitations to provide the basis for a long-term defensible and robust evaluation of the PWR sump performance. However, in the short term, there may be some practical actions that can be explored. For instance, the staff should encourage licensees to pursue, at an early stage, corrective actions that will be as independent as possible of known model uncertainties. These actions may include, for example; removing material that is known to manifest anomalous or particularly detrimental sump blockage results in tests (and whose removal does not introduce secondary detrimental effects); use of "double-jacketing"; demonstrating that materials such as coatings are proven by testing to be sufficiently robust that conservatism in assessing their vulnerability can be reduced; testing alternative filtering devices, such as debris catchers and active screens to the point where their performance can be realistically characterized empirically, or can be conservatively bounded.

Such actions in this initial phase could be followed by a more complete evaluation of the technical analytical problem (as outlined in this ACRS letter) and the implementation of long-term risk-informed solutions.

Additional comments by ACRS Member Graham B. Wallis.

I agree that practical steps should be explored and long-term solutions developed. However, this responds only to part of the purpose of GL2004-02, which also requests that licensees perform an evaluation in the short term. In the absence of definitive guidance for making this evaluation, the staff needs to develop a success path that will avoid wasteful iterations while guidance is developed. One approach might be to break up the process into a set of phases, each of which is based on the best available methods, results that are clearly evident despite uncertainties, and decisions that can be implemented within a realistic schedule.

To justify actions which may have a major impact on operating plants, the staff needs to do a better job of explaining the rationale for regulatory decisions, particularly when the technical bases and assumptions are questionable.

References

1. U.S. Nuclear Regulatory Commission, Safety Evaluation by the Office of Nuclear Reactor Regulation Related to NRC Generic Letter 2004-02, Nuclear Energy Institute Guidance Report (Proposed Document Number NEI 04-07), "Pressurized Water Reactor Sump Performance Evaluation Methodology", September 2004

2. Nuclear Energy Institute, Guidance Report (Proposed Document Number NEI 04-07), "Pressurized Water Reactor Sump Performance Evaluation Methodology", May 2004

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

4. U.S. Nuclear Regulatory Commission Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors", June 9, 2003

5. Graham B. Wallis, "The NUREG/CR-6224 Head Loss Correlation", September 3, 2004

6. Graham B. Wallis, "Flow Through a Compressible Porous Mat: Analysis of the Data Presented in Series 6 Tests Reported by LANL in LA-UR-1227", September 2, 2004

7. Graham B, Wallis, "The ANSI/ANS Standard 58.2-1988: Two-Phase Jet Model", August 31, 2004

8. Sanjoy Bannerjee, "Review of Head Loss Prediction Across Sump Screens," October 5, 2004

9. Victor Ransom, "Comments on GSI-191 Models for Debris Generation", September 14, 2004

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

October 18, 2004

Mr. David O'Brien
Commissioner
Vermont Department of Public Service
112 State Street
Drawer 20
Montpelier, VT 05620-2601

SUBJECT: VERMONT YANKEE EXTENDED POWER UPRATE REQUEST

Dear Mr. O'Brien:

I am responding to your letter of September 17, 2004, which made several requests regarding the proposed extended power uprate for the Vermont Yankee Nuclear Plant. The Advisory Committee on Reactor Safeguards (ACRS) will be considering the uprate request after the NRC staff has completed its review. The ACRS expects to discuss the application first at a meeting of its Thermal-Hydraulic Phenomenon Subcommittee, and then at a subsequent regular Full Committee meeting.

The Committee's meetings are generally open to the public, and anyone who wishes to make a presentation on a subject under consideration is welcome to participate. Because of the interest in this matter in Vermont, we will ensure that you are aware of the schedule for these meetings so that you can plan to attend and contribute to the discussion. We are also considering your request to hold the Subcommittee meeting at a location close to the plant, so that the public can attend and observe how the Committee operates. The agendas for the Full Committee meetings include a broad range of issues that involve a much wider community of stakeholders, and it is unlikely that we would be able to hold a regular meeting of the Full Committee outside the NRC headquarters.

If you have any questions, please contact the ACRS staff member who is responsible for power uprate reviews, Mr. Ralph Caruso (301-415-8065).

Sincerely,

Mario V. Bonaca

Mario V. Bonaca
Chairman

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

October 22, 2004

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

SUBJECT: PROPOSED RESOLUTION OF GENERIC SAFETY ISSUE 185, "CONTROL OF RECRITICALITY FOLLOWING SMALL-BREAK LOCAs IN PWRs"

Dear Chairman Diaz:

During the 516th meeting of the Advisory Committee on Reactor Safeguards, October 7-9, 2004, we met with representatives of the NRC staff to review the draft NUREG-XXXX, "Avoiding Recriticality From Transport of Diluted Water From Loop Seals to Core During Small Break Loss-of-Coolant Accidents in Pressurized Water Reactors," dated February 2004 (Ref. 1), prepared by the Office of Nuclear Regulatory Research (RES) to provide the technical basis for resolution of Generic Safety Issue (GSI) -185. Our Subcommittee on Thermal-Hydraulic Phenomena reviewed this matter during a meeting on September 23, 2004. We also had the benefit of the documents referenced.

CONCLUSION

We agree with the RES recommendation that GSI-185 be closed without imposition of any new regulatory requirements for all existing Framatome-Babcock & Wilcox (B&W), Westinghouse, and Combustion Engineering Plants.

DISCUSSION

Boron dilution accidents have been historically considered in pressurized water reactor (PWR) safety analyses. A new concern about boron dilution accidents arose in 1995 when Framatome Technologies submitted an analysis suggesting that if a B&W-designed nuclear steam supply system (NSSS) spends some time in a boiler-condenser mode following a small-break loss-of-coolant accident (LOCA), a substantial amount of deborated water may accumulate in the steam generator, loop seal, reactor coolant pump (RCP), cold leg, downcomer, and lower plenum. The resumption of natural circulation or the restart of a RCP were identified as mechanisms for causing deborated water to flow into the core and potentially cause a prompt criticality. Subsequently, in late 1996, Framatome Technologies developed guidance to restrict RCP restart in order to prevent potential fuel damage.

On February 1, 1999, the Office of Nuclear Reactor Regulation (NRR) requested the establishment of a GSI relating to reactivity insertion accidents resulting from boron dilution (Ref. 2). This request was based on 1) new information about reactivity insertion accident experiments on high burn-up fuel indicating that fuel damage would occur at lower energy deposition, and 2) new analyses performed by Framatome Technologies and submitted by the B&W Owners Group (B&WOG) (Ref. 3). The event was deemed to be of potential importance for all PWRs.

In accordance with the procedures described in NUREG-0933, RES performed a prioritization study for this issue (Ref. 4). The prioritization study supported establishing GSI-185. RES subsequently prepared a plan that focused on the major assumptions and limitations of the prioritization study in order to develop a more accurate and realistic, yet conservative, analysis (Ref. 5). The results of this research are documented in draft NUREG-XXXX. The boron dilution accidents of interest are associated with small-break LOCAs, involving break sizes of 0.5 to 2 in. diameter. The coolant loss from these small-break LOCAs is large enough to decrease the system coolant inventory but not large enough to depressurize the system. During such LOCAs, natural circulation ceases at approximately 60% coolant inventory and the system enters a boiler-condenser mode of operation in which steam is generated in the core and flows though the voided hot leg to the steam generator tubes, where condensation will occur. In once-through steam generator systems, all the condensate drains to the lower portion of the steam generator tubes and displaces and mixes with the borated water. Eventually, the steam generator outlet plenum and loop seal could become filled with deborated water and the tubes partially filled. For U-tube steam generator systems the scenario is similar except that only condensate from the downside of the U-tubes displaces and mixes with the borated system coolant in the outlet plenum, RCP, and loop seal.

If natural circulation resumes or if a RCP is restarted, then the deborated mass of coolant would be swept through the cold leg into the reactor vessel downcomer and lower plenum. As the deborated water flows into the core, an increase in reactivity will occur, with the potential for recriticality and for fuel damage if sufficient energy deposition occurs. A key technical issue associated with the evaluation of this accident sequence is the extent of mixing between the deborated coolant and the borated coolant in the reactor vessel as the deborated slug is transported to the core. Both the Framatome Technologies analyses for the B&WOG and the RES prioritization study were based on the conservative assumption of plug flow with no mixing.

RES used two new analytical elements to address the limitations of the previous analyses: (1) a more realistic model for the mixing of deborated water with the borated system coolant and (2) improved methods for calculating the core neutronic behavior as the deborated water enters the core.

The RES mixing model incorporates realistic, yet conservative, quantification of the mixing of the deborated water slug with the borated system coolant as the slug flows through the cold leg, downcomer, and lower plenum and into the core. The mixing model combines the elements of two mixing extremes, namely plug flow (no mixing) and back-mixed flow (ideally mixed). The cold leg loop seal, cold leg, and downcomer are treated as pipe components with plug flow while the steam generator outlet plenum, the RCP, and the reactor vessel lower plenum are modeled as a back-mixed volumes. The mixing model was validated for the components external to the reactor using experimental data from the University of Maryland B&W integral test facility. The model was then validated with regard to the in-vessel mixing using the FLUENT code. FLUENT was used to produce a multi-dimensional computational fluid dynamics model of the downcomer and the lower plenum consisting of 353,690 cells. The FLUENT results were compared to University of Maryland downcomer mixing data and good agreement was obtained. The FLUENT model was then used to predict the core inlet transient concentration for a step change in the cold leg concentration and this result was compared to the RES mixing model prediction. This result demonstrated that the mixing model provided a realistic but conservative prediction of core inlet boron concentration.

The new core neutronic behavior model uses a 30 channel RELAP5 model for thermal hydraulics in the fuel assemblies coupled to the multidimensional PARCS code for neutron kinetic analysis. The core has one-eighth symmetry and the multi-dimensional model used has 29 fuel bundles. The RELAP5 code is used to model flow in the 29 bundles with one channel for the core bypass flow. The PARCS-RELAP5 code was validated by comparing its power predictions to predictions obtained with French and Russian codes for boron dilution reactivity transients. This model is judged to be appropriate and accurate for boron dilution transient simulation.

The RES analyses for resolution of GSI-185 have shown that recriticality for the postulated boron dilution event is not possible in Westinghouse and Combustion Engineering reactor designs for either resumption of natural circulation or restart of a RCP. The volumes of the steam generator outlet plena, loop seals, and RCPs are too small to accumulate sufficient deborated water to cause recriticality of the core. For B&W reactor designs with lowered loop seals, however, the resumption of natural circulation may cause a brief criticality event but the fuel temperatures remain within the normal full-power operating conditions. The inadvertent restart of a RCP in a lowered-loop B&W plant may cause fuel temperatures to rise above design limits, but would result only in limited fuel damage and would not lead to a loss of coolable core geometry.

The probability of this event is believed to be very small for the following reasons: (1) the range of break sizes that will result in boiler-condenser operation is small; (2) the high-pressure injection system must fail; (3) the plant is vulnerable to this event only during the first 20% of a fuel cycle; (4) additional failures or operator actions, such as break isolation, must occur to allow the system to remain in the boiler-condenser mode long enough (approximately one hour) to generate sufficient deborated water and (5) a RCP must be inadvertently restarted. The low probability of this event, coupled with its limited consequences, supports the conclusion that GSI-185 can be considered resolved for B&W lowered-loop plants, as well as for all other operating PWRs.

Sincerely,

Mario V. Bonaca

Mario V. Bonaca
Chairman

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4. Memorandum to Ashok C. Thadani From Farouk Eltawila, Generic Issue No. 185, "Control of Recriticality Following Small-Break LOCAs in PWRs," July 7, 2000.
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

November 2, 2004

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

SUBJECT: REPORT ON "AN OVERVIEW OF DIFFERENCES IN NUCLEAR SAFETY REGULATORY APPROACHES AND REQUIREMENTS BETWEEN UNITED STATES AND OTHER COUNTRIES"

Dear Chairman Diaz:

In an April 28, 2003 Staff Requirements Memorandum (SRM), resulting from the April 11, 2003 meeting with the Advisory Committee on Reactor Safeguards (ACRS), the Commission stated that in the course of reviewing and advising the Commission on reactor issues, the ACRS should explore and consider other international regulatory approaches and inform the Commission where there are significant differences in regulatory approaches and requirements.

To set a baseline for this ongoing obligation, we asked Dr. Nourbakhsh , ACRS Senior Staff Engineer, to prepare an overview of differences in nuclear safety regulatory approaches and requirements in the United States and other countries.

The attached report, which has been reviewed by us, is based on the author's review of a number of documents issued by various international organizations. Despite considerable similarities in the objectives and actual implementation of nuclear safety regulatory approaches, there are differences in specific regulatory requirements. Even within the European Union, efforts to harmonize safety requirements and regulatory practices have been unsuccessful so far.

This report provides an overview of nuclear safety regulatory approaches and discusses differences in specific requirements of current interest:

- Design-basis assessment
- Periodic safety reviews
- Protection against severe accidents
- Risk-Informed regulation and practices
- Materials degradation issues and aging management

We will endeavor to keep the Commission informed of significant differences in regulatory requirements between United States and other countries that come to our attention.

Sincerely,

Mario V. Bonaca

Mario V. Bonaca
Chairman

Attachment:
As stated

An Overview of Differences in Nuclear Safety Regulatory Approaches and Requirements Between United States and Other Countries

Prepared by
H. P. Nourbakhsh

October 2004

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

ABSTRACT

This report has been prepared for use by the NRC Advisory Committee on Reactor Safeguards (ACRS) in support of its ongoing effort to inform the Commission on significant differences in regulatory approaches and requirements between the United States and other countries. This report, which is based on review of a number of documents issued by various international organizations, provides an overview of regulatory approaches and discusses differences in specific regulatory requirements of current interest in the United States.

The views expressed in this report are solely those of the author and do not necessarily represent the views of the ACRS.

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TABLES

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ACRONYMS

Acronym	Definition
ABWR	Advanced Boiling Water Reactor
ACRS	Advisory Committee on Reactor Safeguards
AGR	Advanced Gas-cooled Reactor
ALARA	As Low As reasonably Achievable
ALARP	As Low As reasonably Practicable
BWR	Boiling Water Reactor
CFR	Code of Federal Regulations
CSNI	Committee on the Safety of Nuclear Installations
CNRA	Committee on Nuclear Regulatory Activities
CSS	Containment Spray System
DCH	Direct Containment Heating
ECCS	Emergency Core Cooling System
EPR	European Pressurized Water Reactor
GCR	Gas-Cooled Reactor
GE	General Electric
IAEA	International Atomic Energy Agency
LOCA	Loss-of-Coolant Accident
LWR	Light Water Reactor
NEA	Nuclear Energy Agency
NEI	Nuclear Energy Agency
NII	Nuclear Installations Inspectorate
NRC	Nuclear Regulatory Commission
OECD	Organization for Economic Cooperation and Development
PHWR	Pressurized Heavy Water Reactor
PWR	Pressurized Water Reactor
PRA	Probabilistic Risk Assessment
RCS	Reactor Coolant System
SAM	Severe Accident Management
SRM	Staff Requirements Memorandum
SSCs	Systems, Structures, and Components
TMI-2	Three Mile Island Unit 2
U.K.	United Kingdom
U.S.	United States

1 INTRODUCTION

The purpose of this report is to provide an overview of differences in nuclear safety regulatory approaches and requirements between United States (U.S.) and other countries.

In an April 28, 2003 Staff Requirements Memorandum (SRM) [1], resulting from the April 11, 2003 meeting with Advisory Committee on Reactor Safeguards (ACRS), the Commission stated that "In the course of its routine activities of reviewing and advising the Commission on reactor issues, the Committee should explore and consider other international regulatory approaches. Where there are significant differences in regulatory approaches and requirements, the Commission should be informed." This report has been prepared for use by the ACRS in responding to the Commission request.

This report focuses on regulatory requirements pertinent to western-designed light water reactors (LWRs). It does not address requirements relating to nuclear materials and waste safety, or safeguard and security issues.

A number of documents issued by various international organizations, in particular the European Commission and the Organization for Economic Cooperation and Development/ Nuclear Energy Agency (OECD/NEA), were reviewed for the preparation of this report.

The European Commission has sponsored many studies to support its activities toward harmonization of safety requirements and practices in an enlarged European Union. The results of these studies [2-7] on review of safety philosophies and practices in European Union member states were the major source of information for developing this report.

The OECD/NEA reports [8-11] on the scientific and technological background of nuclear safety criteria, rules and guidelines, and applied assessment methods were reviewed to identify the safety issues for which there may not yet be a common technical position among international communities.

The adoption of the Convention on Nuclear Safety in 1994 legally binds the participating countries to maintain a high level of safety. The Convention obliges parties to submit reports on the implementation of their obligations for "peer review" at regular meetings of the parties held by the International Atomic Energy Agency (IAEA). The National Reports on the Convention of Nuclear Safety [12] were also utilized for the preparation of this report.

The report begins with a general overview of regulatory approaches in various countries. It then discusses differences in the specific regulatory requirements in the areas of current interest in the U.S.. They are:

- Design-basis assessment
- Periodic safety reviews
- Protection against severe accidents
- Risk-informed regulations and practices
- Materials degradation issues and aging management

2 GENERAL OVERVIEW OF REGULATORY APPROACHES IN THE WORLD

Regulatory policies differ from country to country. These differences reflect the differences in culture, social, economic, and governmental systems between countries [2]. Regulatory regimes fall broadly within two categories, prescriptive or otherwise. In a prescriptive regime, the requirements on methodologies, standards, and quality assurance are prescribed by the regulatory authority. The licensee must demonstrate that the plant complies with these regulatory requirements. In addition regulatory guidelines, describing methods acceptable to the regulatory authority, may be provided which the licensee can follow for implementing specific portions of regulations. The U.S. Nuclear Regulatory Commission (NRC) regulations fall in this category.

In a less prescriptive regime, the emphasis is on principles which are largely qualitative (except perhaps for certain parameters, e.g., dose limits). The licensee must comply with these principles, but may choose its own methodology of meeting them. European regulations are generally less prescriptive than those in the U.S. There is, however, some degree of variation among the European countries [2].

The interactions between regulator and utility vary from country to country. Generally such interactions are formal with respect to licensing, but less formal from the point of view of safety research. However, there are differences within the licensing relationships. Some regulators encourage a collaborative approach and a continuing dialog through the stages of a licensing application, others adopt a very formal approach [2]. There is a non-adversarial relationship between the plant operator and the regulatory authority in many countries. This is, in part, due to the fact that the plants are owned by the government institutions in these countries.

The safety approaches and practices in the western countries have been largely open to public knowledge and scrutiny. This has encouraged a collaborative safety consciousness over many years. The Eastern European and the former Soviet countries are moving toward more open safety practices. This has been facilitated through the influence of various IAEA, OECD, U.S., and European initiatives.

There is a strong influence of the U.S. regulatory system in setting the basis for licensing requirements in many countries. This is because a large number of plants in operation in other countries are of U.S. design or derived from U.S. designs, which must be licensable in their country of origin. Some countries (e.g., Spain, Holland, and Belgium) completely follow the regulations of the country from which their nuclear power plants were purchased. They follow the U.S. NRC regulations for their Westinghouse pressurized water reactors (PWRs) and the General Electric (GE) boiling water reactors (BWRs), and the German regulations for their Siemens (KWU) plants [2].

Operating reactors by country and by type are presented in Table 1, and Table 2 respectively. The LWR technology was initially developed in the U.S., with GE pioneering the BWRs and Westinghouse developing the PWRs. The main nuclear electricity production in Europe and the Far East, in common with the rest of the world, now comes from LWRs. The exception is the United Kingdom (U.K.) where advanced (oxide fueled) gas-cooled reactors (AGRs) provide a large fraction of the nuclear-generated electricity. The U.K. also subsequently elected to follow the LWR route. The LWR also provides the basic concepts for the WWER reactors developed

Table 1 Operating Reactors in Various Countries[1]

Country	No. of Operational Units				Total MW(e)
	PWR	BWR	Other	Total	
United States Of America	69	35	0	104	98298
France	58	0	FBR 1	59	63363
Japan	23	27	ABWR 3 FBR 1	54	45464
Russian Federation	0	0	LWGR 15 WWER 14 FBR 1	30	20793
United Kingdom	1	0	AGR 14 GCR 12	27	12052
Republic of Korea	15	0	PHWR 4	19	15850
Germany	12	6	0	18	20643
Canada	0	0	PHWR 17	17	12113
India	0	2	PHWR 12	14	2550
Ukraine	0	0	WWER 13	13	11207
Sweden	3	8	0	11	9451
Spain	7	2	0	9	7584
China	7	0	PHWR 2	9	6587
Belgium	7	0	0	7	5760
Taiwan	2	4	0	6	4884
Czech Republic	0	0	WWER 6	6	3548
Slovak Republic	0	0	WWER 6	6	2442
Switzerland	3	2	0	5	3220
Bulgaria	0	0	WWER 4	4	2722
Finland	0	2	WWER 2	4	2656
Hungary	0	0	WWER 4	4	1755
Republic of Lithuania	0	0	LWGR 2	2	2370
Brazil	2	0	0	2	1901
South Africa	2	0	0	2	1800
Mexico	0	2	0	2	1310
Argentina	0	0	PHWR 2	2	935
Pakistan	1	0	PHWR 1	2	425
Slovenia	1	0	0	1	656
Romania	0	0	PHWR 1	1	655
Netherlands	1	0	0	1	449
Armenia	0	0	WWER 1	1	376

[1] Based on information in IAEA (PRIS) database [13], last updated on May 19, 2004

Table 2 Operating and Under Construction Reactors by Type[2]

Type	Operational		Under Construction	
	No. of Units	Total MW(e)	No. of Units	Total MW(e)
PWR	214	204335	6	6111
BWR	90	78025	1	1067
WWER	50	33040	8	7534
PHWR	39	19972	8	3135
LWGR	17	12589	1	925
AGR	14	8380	0	0
GCR	12	2484	0	0
ABWR	3	3955	3	3904
FBR	3	1039	0	0

in Russia and used in other Eastern European countries and Finland.

The LWRs in other countries are quite similar to the designs developed in the U.S.. The French PWRs are very similar to the Westinghouse PWRs since France had bought the license for their design from Westinghouse. The PWRs and BWRs designed by the KWU (Siemens) are similar to the U.S. designs but are different in the configurations of their containments. The BWRs designed by ABB Atom are similar to the Mark-II BWR plants of GE, except, with some modifications (e.g., internal pumps).

The LWRs in other countries, having been commissioned after the U.S. LWRs, were designed and constructed in accordance with the design criteria and safety philosophy developed in the U.S.. The U.S. safety philosophy of defense in depth was adopted by the regulatory authorities in western Europe, Japan, and Korea, not only for the barriers to the release of radioactive substances, but also in the design, construction, quality assurance, inspection, and operational practices. However, there may be differences in the implementation of the defense-in-depth principle, e.g., in levels of diversity and redundancy required from the safety systems. Requirements for three trains of safeguard in France and four trains of safeguard in Germany (because of on-line maintenance) and the requirement for diversity of instrumentation for all safety-related measurements in Germany are examples of such differences in implementation of the defense-in-depth principle. There are also some country-

[2] Based on information in IAEA (PRIS) database [13], last updated on May 19, 2004

specific regulatory requirements regarding the effectiveness of the various barriers. One such example is the requirement in Germany to design the containment to withstand the crash of a light fighter plane.

The 1979 accident at Three Mile Island Unit 2 (TMI-2), led to the reexamination of the design basis and the consideration of regulations for protection against severe accidents. The reexamination of the design basis was prompted by the fact that the TMI-2 accident initiated with a small-break loss -of-coolant accident (LOCA), whose consequences should have been bounded by those of a large-break LOCA, but became much more severe due to misunderstanding of the event by the operators. The event-based procedures have been modified to symptom-based procedures in most Western plants [2].

The Chernobyl accident in 1986 affected opinion in Western Europe about the safety of nuclear power plants in general, contributing to the decisions of some countries (e.g., Germany) to tighten safety requirements for new plants, implying design modifications, or to phase out nuclear power stations, either immediately (e.g., Italy), or over a period of time (e.g., Sweden, Germany).

In most countries, the principles of traditional deterministic approach have been accepted over many years to demonstrate the reliability and safety of design. (deterministic approach refers to an approach that specifies certain design and operational conditions and applies bounding criteria to demonstrate acceptable plant performance.) Systems, structures, and components (SSCs) are designed and manufactured to accepted standards, regulations, codes of practice etc. to ensure that the SSCs can perform their intended functions. The single-failure criterion has been commonly adopted, as has the 30 minutes rule, i.e., that the safety objectives can be met without operator intervention within the first 30 minutes into an accident.

The majority of licensing submittals have been based on the evaluation model (EM) methodology. This was established on the premise that deliberate modeling conservatisms are included to compensate for lack of knowledge of the governing phenomena. This methodology was based on the Appendix K of the U.S. Code of Federal Regulations (10CFR Part 50). However, with improved understanding of the phenomena, there have been moves to change the conservative biases and assumptions of the evaluation model methodology, allowing the licensee to move further toward best-estimate methodologies. Within the U.S. this led to a revision of the emergency core cooling system (ECCS) rule (10CFR 50.46) in 1988 enabling licensees to apply best-estimate methodologies, with the provision that due allowance is given to any remaining uncertainties in code, data, or modeling. The move toward best-estimate methodologies is also a common trend in most countries [2].

In light of increased realization of the impact of human factors on plant safety, regulatory authorities now require the utilities to consider human factor engineering concepts in the design and operational aspects of plants. There is an international recognition of the importance of safety culture and management. There is an evolving consensus on what constitutes good performance on the part of an organization but less on how it can be measured.

The ALARA (or ALARP) principles are generally adopted to ensure that risks are reduced to a level acceptable to the regulatory body to be "as low as reasonably achievable (or practicable)." Most countries follow this approach in qualitative terms. In principle risk may be quantified via a cost-benefit analysis, whereby the costs to industry are compared with the benefits to society. The extent to which cost-benefit analysis is encouraged or allowed varies from country to country. In most European countries, safety improvements are generally introduced without the requirement for formal cost-benefit analysis [2]. Nevertheless, cost/benefit is informally considered by regulatory authorities in all of these countries. The issue of cost may become more important as competition grows in Europe.

Basic deterministic safety assessments are now generally complemented by probabilistic risk assessments (PRAs) to verify the overall design and system of operation. PRAs are conducted by many countries to demonstrate that there are no sudden increase in risk for accidents that are outside of the design basis. Most countries with nuclear power plants have performed PRAs and have found that such assessments often lead to the identification of plant vulnerabilities. However, there is not much support, so far, in many other countries for formally considering risk information in regulatory decisionmaking as it is in the U.S..

3 DIFFERENCES IN REGULATORY REQUIREMENTS

Despite considerable similarities in the objectives and actual implementation of nuclear safety regulatory approaches, there are differences in regulatory requirements across the world. Indeed, efforts to harmonize safety requirements and regulatory practices within the European Union have been unsuccessful so far.

Reasons for the differences in regulatory requirements relate to national energy policy (mainly in support of public acceptance); national industrial tradition (e.g., giving more credit to redundancy or diversity, or crediting a software-based system as opposed to hard-wired controls); consistency with national regulatory or legislative system (e.g., compliance with probabilistic safety criteria on individual and societal risk as applicable to the environmental policy); country-specific conditions (e.g., differences in geography such as flooding for Netherlands and seismic for Japan); and uncertainties associated with the severe-accident phenomena.

Some of the areas where differences in safety requirements exist are discussed below.

3.1 Design-Basis Assessment

There is an internationally accepted rule that the licensee should provide a comprehensive safety assessment to confirm that the design of an installation fulfils the safety objectives and requirements. This assessment is submitted in a safety analysis report. Specific approval by the regulatory body is required before the start of operation. The U.S. NRC regulation (10CFR50.71) requires the licensee to update periodically (the interval between updates should not exceed 24 months) the final safety analysis report (FSAR) originally submitted as part of the application for the operating license. The update should include the effects of: all changes made in the facility or procedures as described in the FSAR; all safety analyses and evaluations performed by the licensee in support of approved license amendments, and all analyses of new safety issues performed by or on behalf of licensee at Commission request. In many other countries, a safety analysis report is updated every 10 years as a part of periodic safety reviews (see section 3.2). These reviews must take account of existing operational experience and any other information relevant to safety that is currently available.

The accident sequence groups and the accidents to be analyzed in the safety analysis report may be prescribed by the regulator (e.g., U.S. NRC), but if not, are defined by the licensee as part of its safety case submission (e.g., United Kingdom). The implementation of either approach is similar. There are, however, some differences in certain acceptance criteria and the licensing calculations due to various degree of conservatism made at each step of the calculation. Some of these differences are summarized below.

3.1.1 Acceptance Criteria for Emergency Core Cooling System

Most countries use acceptance criteria for ECCS that are based on those specified in Appendix K to 10CFR Part 50. Germany has also established an additional acceptance criterion to limit the fraction of failed fuel clad under LOCA conditions. The 10% fuel failure criterion in Germany was originally established to limit the radiological consequences in case of a LOCA (see section 3.1.2). The original intention of this criterion has since been broadened. Beside the radiological aspects, this criterion has been used for the evaluation of core loading. If the core is loaded with new fuel rods or new loading strategies are applied, the compliance with the 10% fuel failure criterion has to be demonstrated again by the applicant [6].

There is a common understanding among the German licensing authorities that if the compliance with all these acceptance criteria can be proven, there is no need to limit the fuel burn-up or to restrict the core loading[6].

3.1.2 Extent of Fuel Failures that is Assumed in Radiological Assessment

The extent of fuel failure that is assumed in radiological assessments varies from country to country. In performing the design-basis accident analyses, the commonly applied practice includes the use of conservative assumptions regarding system performance and components failure. Following a large LOCA, it is assumed that a fraction of the fuel is failed allowing release of the radionuclides from the fuel into the containment atmosphere. This release of fission products into the containment ("in-containment source term") has a wide range of regulatory applications, including the basis for (1) the adequacy of the leaktightness of the containment, (2) the performance requirement of fission-product cleanup systems such as sprays and filters, (3) post-accident habitability requirements for the control room, and (4) the radiation environment for qualification of safety-related equipments.

The determination of source term inside the containment involves assumptions corresponding to to various physical stages in the release of fission products, including fraction of core failure, release from damaged fuel, airborne part of release and release into reactor coolant system and sumps, chemical behavior of iodine in the aqueous and gas phases, and natural and spray removal in the containment atmosphere.

Some countries (Belgium and Spain) follow the U.S. and assume a source term corresponding more to a core-melt accident decoupled from the LOCA thermal-hydraulic calculations, while other countries take into account the physical phenomena during a LOCA still with conservative assumptions.

Table 3 shows the extent of fuel failure that is assumed in radiological assessments in different countries. Many countries (e.g., Belgium, United Kingdom, Spain) follow the U.S. and assume 100% fuel failures during a large LOCA. Some European countries (e.g., Germany, Switzerland, Netherlands) assume 10% of fuel failure during a LOCA. In France, a 100% fuel failure assumption is used for the radiological consequences evaluation of the 900 and 1300 MWe plants. However, for N4 plants, a 33% fuel failure assumption has been proposed by the utility and is under assessment by the regulatory body (IPSN). The utility position is that this value is sufficiently conservative to constitute a decoupling assumption avoiding a specific safety demonstration for each core refueling, taking into account a previous Framatome study for the 1300 MWe French nuclear power plant design for which 7% of clad failure was predicted [6].

Table 3 The Extent of Fuel Failure That Is Assumed In Radiological Assessments

Country	Extent of Fuel Failures in Radiological Assessment
Belgium	100%
France	100% (33% proposed for N4 plants)
Germany	10%
Netherlands	10%
Spain	100%
Switzerland	10%
United Kingdom	100%
United States	100%

3.1.3 Strainer Blockage Issue

The 1992 clogging of intake strainers for containment spray water in Barsebäck-2, a BWR in Sweden, renewed the focus of regulators around the world on safety questions associated with strainer clogging which, until then, had been considered as resolved.

Although the Barsebäck incident in itself was not very serious, it revealed a weakness in the implementation of defense-in-depth concept in the design, which under other circumstances could have led to the failure of the ECCS and containment spray system (CSS). The Barsebäck-2 event also demonstrated that larger quantities of fibrous debris could reach the strainers than had been predicted by models and analysis methods developed for the resolution of the strainer blockage issue[8,14].

The Barsebäck-2 incident prompted action on the part of regulators and utilities in other countries. Research and development efforts of varying intensity were launched in many countries. Extensive studies have been performed to assess the amount of insulation materials that could be dislodged during pipe break events inside the containment. In many countries, the analyses were based on the double cone model developed by the NRC [14]. The analyses have also included specific studies of the transport of insulation materials and other debris in the containment, and of strainer pressure drops. Such efforts resulted in a number of corrective actions being taken in BWRs and some PWRs around the world. For a number of plants, actions were taken as direct responses to requirements issued by regulatory authorities, while for other plants back-fitting measures were introduced voluntarily or because of anticipated requirements [9].

The modifications of the ECCS and/or CSS suction strainers carried out in different countries are summarized in Tables 4 and 5 for BWRs and PWRs respectively. The modifications have resulted in new strainer designs with significantly enlarged filtering area. Most of the new strainers have good self-cleaning properties. In some BWRs, the

Table 4 Summary of the BWR Strainer Modifications in Different Countries After the Barsebäck-2 Event

Country	BWR Strainer Modifications	Comment
United States of America	34 out of 34 units modified	New strainers with significantly increased area Established a schedule to remove particulate and other debris from the suppression pool
Japan	None of 28 units modified	More than 95% of the insulations are replaced by non-fiber type ones (e.g., Reflective Metallic Insulation). The absence of foreign materials in the suppression pool is ensured through inspection and maintenance practices
Sweden	9 out of 9 units modified	New strainers with 15 to 40-fold area increase
Germany	4 out of 6 units modified	Stainers were enlarged
Spain	2 out of 2 units modified	New strainers with significant area increase
Switzerland	2 out of 2 units modified	New strainers with 7 to 30-fold area increase
Finland	2 out of 2 units modified	About 10-fold area increase

Table 5 Summary of the PWR Stainer Modifications in Different Countries After the Barsebäck-2 Event

Country	PWR Strainer Modifications	Comment
United States of America	One out of 69 units modified	Davis-Besse is the only U.S. PWR plant that its licensee voluntarily enlarged its sump screen area - by 30 fold
France	EDF plans Backfits to sump strainers at all 58 units	EDF is finishing design studies for the backfit, which is expected to consist of replacing the current filters with a system of pierced piping that provides more strainer surface.
Japan	None of 23 units modified	More than 95% of the insulations are replaced by non-fiber type ones (e.g., Reflective Metallic Insulation). The absence of foreign materials in the recirculation sumps is ensured through inspection and maintenance practices.
Germany	2 out of 12 units modified	Stainers were enlarged.
Spain	None of 7 units modified	
Belgium	2 out of 7 units modified	6-fold area increase
Switzerland	None of 3 units modified	
Sweden	1 out of 3 units modified	New strainers with > 7-fold area increase
Netherlands	1 out of 1 unit modified	New strainer installed (50% area increase) There is no mineral or fiberglass insulation of noteworthy importance around the primary components as well as in the sump area.
Finland	2 out of 2 WWER-440/213 units modified	New strainer design with significant area increase

design includes the capability to back-flush the strainers [9].

Large fractions of the thermal insulation materials utilized on piping and other components inside the containment have also been replaced. The newly installed insulation materials vary both within and among countries. They are primarily reflective metallic insulation, nuclear grade fiberglass, mineral wool, and calcium silicate. The same insulation material (e.g., mineral wool) are installed differently in different countries (i.e., Jacketed or encapsulated in cassettes). The administrative measures taken in other countries include a periodic cleanup of the suppression pool and containment sumps, with the aim to minimize the presence of foreign materials, and the control and eventual improvement of the containment coating.

In U.S., all BWR licensees were required to implement appropriate measures to ensure the capability of the ECCS to perform its safety function following a LOCA. The U.S. nuclear industry addressed the NRC requirements by installing large capacity passive strainers in each BWR plant and establishing a schedule to remove particulate and other debris from the suppression pools. Most U.S. BWR licensees followed the guidance prepared by the U.S. BWR Owners Group during the development of their corrective actions.

As a result of research findings related to resolving the BWR ECCS strainer blockage safety issue, the NRC conducted further research to determine if the transport and accumulation of debris in a containment following a LOCA would impede the operation of the ECCS in operating PWRs. The research program included debris transport tests, debris settling tests, debris generation tests, computational simulations, and various engineering analyses. The results of these studies indicated the need for accurate plant-specific assessment of adequacy of the recirculation function of the ECCS and CSS for each operating PWR. The Nuclear Energy Institute (NEI) also recognized this need and has developed guidance for such plant-specific assessment, which is under review by the NRC staff.

The issue of strainer blockage in PWRs have been particularly troublesome. Continuing research revealing new modes of blockage has shown that the prompt actions taken by some European plants may not have completely alleviated the problem of strainer blockage. Indeed, redesign may be required of these plants. There is a strong evidence that plant owners throughout the world do not have a definitive solution to the issue.

3.2 The Periodic Safety Reviews

In contrast to U.S. NRC, most regulatory authorities in the world have a requirement that the nuclear power plants be subject to an overall assessment on a periodic basis, in addition to the permanent supervision the regulatory body exerts on these plants. Table 6 presents a comparison of international practices with respect to periodic safety review activities.

The periodic safety review is a safety concept mainly developed in the European countries and was introduced later in the IAEA documents [16]. The periodic safety reviews are complementary to the routine reviews of nuclear power plant operation (including modifications to hardware and procedures, significant events, and operating experience) and special safety reviews following major events of risk significance. The frequency of review varies from country to country; typically every ten years (see Table 6). The periodic safety review necessitates licensees to take into account advances in technology unconstrained by licensing basis as in U.S..

Table 6 Periodic Safety Review Requirements in Various Countries

Country	Periodic Safety Review Frequency	Comment
United States Of America	None Required	Requiring the licensee to maintain the licensing basis for the facility or activity
France	Every 10 years (normally)	Linked to the statutory 10 year outage program interval, but the in-depth safety assessment performed on request , at the regulator's discretion
Japan	Every 10 years	Limited scope, concentrated mainly on aging behavior, without the evaluation of the overall plant design
United Kingdom	Every 10 years	Comprehensive safety reviews Requirements stipulated in conditions attached to licenses
Republic of Korea	Every 10 years	Comprehensive safety reviews
Germany	Every 10 years	Comprehensive safety reviews
Canada	None required	License renewed every 2-5 years subject to satisfactory safety performance
Sweden	Every 10 years	Comprehensive safety reviews
Spain	Every 10 years	Comprehensive safety reviews
Belgium	Every 10 years	Comprehensive safety reviews
Czech Republic	Every 10 years	Comprehensive safety reviews
Switzerland	Every 10 years	Comprehensive safety reviews (Regulatory requirement for facilities to comply with the state-of-the-art in science and technology)
Finland	Every 10 years	Comprehensive safety reviews
Hungary	Every 10 years	Comprehensive safety reviews
Mexico	Every 10 years	Comprehensive safety reviews
Netherlands	Every 10 years	Comprehensive safety reviews

The objective of these periodic safety reviews are to assess the cumulative effects of plant aging and plant modifications, operating experience, technical developments, and siting aspects. The reviews include an assessment of plant design and operation against current safety standards and practices in order to propose any eventual improvement. The reviews also examine an extension of the original design basis of the plant, in particular postulated initiating events (internal and external) not considered earlier. The reassessment of the original design basis in Europe is strongly linked to the research on severe accidents and their management strategies [3].

Deterministic safety analyses are used in safety reassessments made in the periodic safety reviews. However, it is now common to complement the deterministic analyses with a PRA (level 1 or 2), in particular to determine the modifications that significantly improve the safety [3].

3.3 Protection Against Severe Accidents

The desire for protection against severe accidents is shared by all of the regulatory authorities in the Western World. It has also been argued that the severe accident is a very low probability event; it deserves a response, but the cost/benefit should be a factor. This argument has been accepted by the U.S. NRC and it is a part of the regulatory practice (backfit rule, 10CFR50.109). However, most regulatory authorities of the European Union Member States do not formally accept this argument. Nevertheless, cost/benefit is silently considered by all of these authorities[2].

The first significant regulatory action for severe accident mitigation was the hydrogen rule (10CFR50.44) issued by U.S. NRC soon after the TMI-2 accident. The rule required control of the hydrogen that is produced in a severe accident. Decisions were made to inert the BWR Mark-I and Mark-II containments and install igniters for hydrogen control in BWR Mark-III and the ice condenser containments. The PWR plants with large dry containments (including those operating with a sub-atmospheric internal pressure) were exempted from hydrogen control, because of the large volume of their containments. Regarding hydrogen control, the BWR Mark-I and Mark-II plants in European countries followed suit in inerting containment atmosphere. The PWRs in Europe have gone through a long evaluation process and most of them (except the Westinghouse-designed plants) have decided to install catalytic hydrogen recombiners of sufficient capacity to address severe accident hydrogen production [2].

The phenomenology of the severe accident is extremely complicated. The severe accident evaluation methodologies are associated with large uncertainties. In fact, such uncertainties have led different parties to reach to different conclusions from research results obtained for several severe accident phenomena. For example, there is a large uncertainty associated with the coolability of a melt/debris attacking the concrete basemat, by flooding with water. This has introduced different approaches for severe accident management strategies. For example, U.K., Spain, Belgium, Sweden, and Finland will add water to their PWR cavities and their BWR lower dry-wells in order to fragment the melt, to facilitate its cooling, and possibly delay the basemat melt-through. On the other hand, the Germans do not have either the facility, or the desire to add water to their PWR cavities in order to avoid the possibility of steam explosions.

The European plant owners, with the encouragement of the regulatory authorities, have developed severe accident management measures. An excellent example is the containment filtered vent, which has been installed in the Swedish BWRs. Containment vents are being considered for installation in several European BWRs and PWRs [2]. Sand filters have been installed in French PWRs, as a backfit. The U.S. plants on the other hand, have not been partial to containment venting. Hard vents are being installed in U.S. BWRs with Mark-I containments, but no U.S. PWR is installing a filtered vent system. Correspondingly, some of the Westinghous-designed plants in Europe (e.g., in U.K., Spain, Belgium) are not considering installation of vents on their containments [2].

A severe accident management measure of very wide acceptance by the PWR plants in Europe and in the U.S. is that of reactor coolant system (RCS) depressurization in the event of a severe accident, in order to avoid the potential for early failure of the containment by direct containment heating (DCH). RCS depressurization was included in the design of the U.K. PWR, primary for accident prevention, and has been introduced in French PWRs as a backfit following high pressure melt ejection and DCH studies. This severe accident management measure, however, cannot be accomplished on some plants whose safety valves do not have sufficient relief capacity.

Another example of severe accident management measures is that of cooling the vessel from outside in order to retain the core debris inside the vessel. With the reactor intact and debris retained in the lower head, phenomena such as ex-vessel steam explosion and core-concrete interaction, which occur as a result of core debris relocation to the reactor cavity, could be prevented. This is the so-called severe accident management strategy of In-Vessel Melt Retention which has been approved for the Loviisa plant in Finland and has been incorporated in the design of the AP600 and AP1000 passive plants. Reactor vessel integrity is assumed if RCS is depressurized and the cavity adequately flooded. Cooperative, international research programs, RASPLAV and MASCA are producing results that suggest this approach may not work for plants with power densities higher than that in the Loviisa plant.

Future reactors are expected to have greater provision against severe accidents. The extension of the design to cover severe accidents, as proposed in Germany and being adopted by the French, would represent a significant departure from currently accepted safety practices in many countries. Whether such an objective becomes a regulatory requirement or not in a particular country will clearly have a major impact on different national approaches to safety [2].

In Europe, there is now a desire to extend the design basis to deal specifically with severe accidents, but the ways to achieve this have not been defined. Much of the current capability for severe accident mitigation arises from the strength of the containment. However, if (some) severe accidents are to be included in the design basis, there is a case for the containment to be designed for higher loads, possibly with a smaller safety margin. This is an area in which standards have yet to emerge, although current documents imply that "best estimate" should be sufficient for severe accident assessments [2].

Inclusion of severe accidents in the design basis poses technical challenges in other areas, such as steam explosion assessments. There are questions on how conservative should the loading be, or whether it is possible to show that the "design loading" is always conservative. Currently this is an area where probabilistic arguments, supported by deterministic analyses, have been accepted [2]. Current proposal for new reactor systems have focused on "evolutionary designs". These designs are essentially modifications to existing LWRs, usually with some advanced safety features. Examples of such safety features are more passive systems, the use of ex-vessel flooding in AP600 and AP1000, and the provision of a debris retention device in the European Pressurized Water Reactor (EPR) design and the WWER-1000 reactors currently under construction in China. .However, there is no general agreement on what additional features should be included in future designs, what the design basis should be, and how the improvements to safety should be quantified [2].

3.4 Risk-informed Regulations and Practices

The U.S. NRC has led the development of the quantitative risk analysis for nuclear power plants. Though PRAs have been used extensively in the past, they were usually limited to a variety of applications on a case by case basis as deemed necessary or useful. The NRC is now moving toward a much expanded use of PRAs in what is termed risk-informed regulatory approach. In 1995, the NRC adopted a policy that promotes increasing the use of probabilistic risk analysis in all regulatory matters to the extent supported by the state-of-the-art to complement the deterministic approach. The current regulatory framework is based largely, but not entirely on a deterministic approach that employs safety margins, operating experience, accident analyses, and probabilistic assessment of the risk, and relies on a defense-in-depth philosophy.

The NRC has applied information gained from PRAs extensively to complement other engineering analyses in improving issue-specific safety regulation, and in changing the current licensing bases for individual plants. Using risk insights, the NRC has modified its oversight process and its requirements for maintenance (10CFR 50.65). The NRC is considering further revisions to its reactor regulations (10CFR Part 50) to focus requirements on programs and activities that are most risk significant. However, these revisions would provide alternatives, that are strictly voluntary, to current requirements. The agency is also considering changes to 10 CFR Part 50 that could lead to incorporating a new set of design-basis accidents, revising specific requirements to reflect risk-informed considerations, or deleting certain regulations. The main driving force behind the move toward risk-informing the current regulations and processes is the expectation that the use of risk insights can result in both improved safety and a reduction in unnecessary regulatory requirements, hence allowing both the NRC and licensees to focus resources on equipment and activities that have the greatest risk significance.

Within Europe, deterministic safety assessments are now often complemented by PRAs to verify the overall design and system of operation. An example is that of specifying as a safety target (goal) the core melt frequency of 10^{-5} or 10^{-6} and the conditional probability of containment failure of 10^{-1} or 10^{-2}. The use of safety evaluation based on probabilistic arguments is, so far, confined to resolution of severe accident safety issues. This trend is not uniform across the European Union. For Example, the German regulatory and technical support organization views of PRA are not as favorable as those of the comparable Spanish organizations[2].

There is not much support, so far, in Europe for formally considering risk-informed regulations and practices, as it is in the U.S.. The exception is the U.K. where the current Nuclear Installations Inspectorate (NII) licensing guidelines adopt a risk-based approach. However, most regulatory authorities in Europe declare that they consider risk information informally [2].

The aim of the most European regulatory authorities is to improve safety, not just to maintain it. Therefore, they encourage the development and the use of PRAs for improving safety, and not for reducing regulatory requirements. There is a considerable reluctance to use the results of quantitative risk assessment to reduce regulatory requirements regardless of the calculated risk significance of these requirements. Uncertainty in the quantitative results, concern over the completeness of the analyses, and lack of properly dealing with organizational or safety culture issues are usually cited as the bases for this reluctance [17].

3.5 Materials Degradation Issues and Aging Management

Safe control of aging of nuclear power plants is an important concern for plant owners and safety regulatory authorities in the world. The optimal ageing management of nuclear power plants require knowledge on materials degradation phenomena and evaluation techniques.

Contrary to U.S., there is no expiration time for the operating license in many countries (see Table 7). The periodic safety review (typically every ten years) is the principal method applied to reactors to ensure that the plant is adequately safe for a further period of operation. However, according to different countries, the operating authorization given by the regulatory authority to the plant operator is not associated with the same formal process. Formal aging management evaluation processes exist in some countries, for quite short periods (i.e., one year in Spain, two in the U.K.); in others, it appears through a requirement of ability for safety demonstration at any moment (in France and Belgium). In practice, safety aging management is implemented through the periodic safety review approach, widely accepted in many countries (see section 3.2).

The material degradation issues have been the subject of numerous studies in different countries and by several international organizations [4]. These studies have led to the establishment of various programs or projects specifically dedicated to the management of aging of SSCs.

Aging management begins with plant design. Many design criteria explicitly or implicitly address aging. The "long-lived" SSCs in a nuclear plant, for example, were originally designed with sufficient margins to meet minimum lifetime requirements. Current aging management programs aim essentially at managing the gradual degradation of SSCs as a result of their physical aging in order to ensure permanently satisfying the safety criteria.

The various aging aspects leading to slow degradation of SSCs are evaluated during periodic safety assessment. However, aspects related to more quick changes (in particular those affecting active components) are managed on a continuos basis through an appropriate maintenance and component qualification.

In the U.S., the original plant life is established by the regulatory process. The Atomic Energy Act and NRC regulations limit the initial operating licenses of nuclear power plants to 40 years, but also permit such licenses to be renewed. The original 40-year term was selected on the basis of economic and antitrust considerations, rather than by technical limitations. However, the selection of this term may have resulted in individual plants being designed on the basis of an expected 40-year service life. 10 CFR Part 54, known as the "license renewal rule," establishes the technical and procedural requirements for renewing operating licenses. Under the license renewal rule, the applicant must perform a screening review of all SSCs within the scope of the rule to identify "passive" and "long-lived" structures and components. The applicant must demonstrate that it will manage the effects of aging such that the SSCs will function as intended throughout the 20-year period of extended operation.

Table 7 Operating License Periods in Various Countries

Country	License Period Approach
United States Of America	Fixed term (40 years, with 20-year renewal option)
France	Lifetime
Japan	Lifetime
United Kingdom	Lifetime
Republic of Korea	Lifetime
Germany	Lifetime
Canada	Fixed term (2-5 years)
Sweden	Lifetime
Spain	Variable (5-10 years) Case-by-case, no fixed term but moving to 10-year standard for nuclear facilities that complete periodic safety reviews
Belgium	Lifetime
Czech Republic	Lifetime
Switzerland	Lifetime (except for 2 plants with term licenses based on historical technical concerns)
Finland	Fixed term (10-20 years)
Hungary	Lifetime
Mexico	Fixed term (30 years)
Netherlands	Lifetime

4 SUMMARY AND CONCLUSIONS

Despite considerable similarities in the objectives and actual implementation of nuclear safety regulatory approaches, there are differences in nuclear safety regulatory requirements between the United States and other countries.

There is a strong influence of the U.S. regulatory system on setting the basis for licensing requirements in many countries. This is because a large number of plants in operation in other countries are of U.S. design or derived from U.S. designs. The U.S. safety philosophy of defense in depth was adopted by the regulatory authorities in Western Europe, Japan, and Korea, not only for the barriers to the release of radioactive substances, but also in the design, construction, quality assurance, inspection, and operational practices. However, there may be differences in the implementation of the defense in depth principle, e.g., in levels of diversity and redundancy required from the safety systems.

In most countries, the principles of traditional deterministic approach have been accepted over many years to demonstrate the reliability and safety of design. Systems, structures, and components are designed and manufactured to accepted standards, regulation, codes of practice etc. to ensure that the SSCs can perform their intended functions.

There is an internationally accepted rule that the licensee should provide a comprehensive safety assessment to confirm that the design of an installation fulfils the safety objectives and requirements. The accident sequence groups and the accidents to be analyzed in the safety analysis report may be prescribed by the regulator (e.g., U.S. NRC), but if not, are defined by the licensee as part of his safety case submission (e.g., U.K.). The implementation of either approach is similar.

There are, however, some differences in certain acceptance criteria and the licensing calculations due to various degree of conservatism made at each step of the calculation. Some of these differences were discussed in this report.

Basic deterministic safety assessments are now generally complemented by PRAs to verify the overall design and system of operation. However, there is not much support, so far, in many other countries for formally considering risk information in regulatory decisionmaking as it is in the U.S..

The desire for protection against severe accidents is shared by all of the regulatory authorities in the Western World. It has also been argued that the severe accident is a very low-probability event; it deserves a response, but the cost/benefit should be a factor. This argument has been accepted by the U.S. NRC and it is a part of the regulatory practice (backfit rule, 10CFR50.109). Most regulatory authorities of the European Union Member States do not formally accept this argument.

The Barsebäck-2 incident prompted a number of corrective actions being taken in BWRs and some PWRs around the world. Actions were taken as direct responses to requirements issued by regulatory authorities for many plants, while for other plants back-fitting measures were introduced voluntarily or because of anticipated requirements. The issue of strainer blockage in PWRs have been particularly troublesome. Continuing research revealing new modes of blockage has shown that the prompt actions taken by some European plants may not have completely alleviated the problem of strainer blockage. Indeed, redesign may be required of these plants. There is a strong evidence that plant operators throughout the world do not have a definitive solution to the issue.

In Europe, there is now a desire to extend the design basis to deal specifically with severe accidents, but the ways to achieve this have not been agreed. Future reactors are expected to have greater provision against severe accidents. The extension of the design to cover severe accidents, as proposed in Germany and being adopted by the French, would represent a significant departure from currently accepted safety practices in many countries. Whether such an objective becomes a regulatory requirement or not in a particular country will clearly have a major impact on different national approaches to safety.

Contrary to U.S., there is no expiration time for the operating license in many countries. The periodic safety review (typically every ten years) is the principal method applied to reactors to ensure that the plant is adequately safe for a further period of operation. However, according to different countries, the operating authorization given by the regulatory authority to the plant operator is not associated with the same formal process. Formal aging management evaluation processes exist in some countries, for quiet short periods; in others, it appears through a requirement of ability for safety demonstration at any moment.

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

November 8, 2004

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

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

During the 517th meeting of the Advisory Committee on Reactor Safeguards, November 4-6, 2004, the Committee considered the draft proposed rule 10 CFR Part 26 on drug testing combined with the draft proposed rule 10 CFR Part 26 on worker fatigue. The Committee has no objection to the staff's proposal to issue the combined rules for public comment. The Committee would like the opportunity to review the draft final combined rules after reconciliation of public comments.

Reference:
Memorandum dated August 3, 2004, from Catherine Haney, Program Director, Office of Nuclear Reactor Regulation, to John T. Larkins, Executive Director, Advisory Committee on Reactor Safeguards, Subject: Request for Deferral of ACRS Review of Proposed Rule: Fitness For Duty (FFD) Programs, 10 CFR Part 26.

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

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

November 8, 2004

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: DRAFT REGULATORY GUIDE 1127, "COMBINING MODAL RESPONSES AND SPATIAL COMPONENTS IN SEISMIC RESPONSE ANALYSIS"

During the 517th meeting of the Advisory Committee on Reactor Safeguards held on November 4-6, 2004, the Committee considered Draft Regulatory Guide 1127, "Combining Modal Responses and Spatial Components in Seismic Response Analysis." The Committee decided not to review this Guide and agreed that the staff should continue with its process for issuing this Guide for public comment. The Committee would like an opportunity to review the draft final version of this Guide after reconciliation of public comments.

Reference:
Memorandum dated October 8, 2004, from Michael E. Mayfield, Director, Division of Engineering Technology, RES, to John T. Larkins, Executive Director, ACRS, Subject: Request to Defer ACRS Review of Draft Regulatory Guide (DG) 1127, "Combining Modal Responses and Spatial Components in Seismic Response Analysis," NUREG/CR-6850, (Proposed Revision 2 of Regulatory Guide 1.92).

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

November 17, 2004

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

SUBJECT: RESOLUTION OF CERTAIN ITEMS IDENTIFIED BY THE ACRS IN NUREG-1740, "VOLTAGE-BASED ALTERNATIVE REPAIR CRITERIA"

Dear Mr. Reyes:

Thank you for your letter of August 25, 2004, responding to our May 21, 2004, letter regarding the Resolution of Certain Items Identified by the ACRS in NUREG-1740, "Voltage-Based Alternative Repair Criteria."

We would like to contribute to the ongoing studies of steam generator tube behavior under severe accident conditions by offering further explanations of some of the points raised in our letter of May 21, 2004.

Countercurrent flow in the hot leg is an essential characteristic of severe accidents that may lead to thermally induced failure of steam generator tubes. Countercurrent flow is driven by buoyancy effects due to the difference in density between the steam in the upper plenum of the reactor vessel and the steam in the lower plenum of the steam generator. These densities are determined by the corresponding temperatures, which are outputs of an analysis of the behavior of the entire system. Therefore, the countercurrent flow rate must also be an output of this system analysis. We are uncertain how SCDAP/RELAP5 can model such phenomena and would like an explanation from the staff during a future meeting.

The staff has performed an excellent computational fluid dynamics (CFD) study of flows in the steam generator. To perform a similar study to guide development and validation of a model for countercurrent flow in the hot leg requires adequate modeling of the lower plenum of the steam generator and the upper plenum of the reactor vessel. These determine the boundary conditions at the ends of the hot leg where the buoyancy effects are created. It is not necessary to apply CFD to other parts of the system such as the vessel internals or the steam generator tubes. The problem is therefore simpler than is described in your response.

We agree that the 1/7th scale tests provide an excellent basis for validating the modeling and theoretical representation of the phenomena which occur in various components and for developing a model of the entire system. However, since all of the thermal-hydraulic phenomena are not exactly scaled, the outputs from the 1/7th scale tests, such as the ratio of heat transferred to the core and to the steam generator, are not directly transferable to full scale. They must be determined from a suitable model of the whole system, applied to the entire transient. The fraction of heat transferred to the steam generator is likely to be low at the start of the transient, however, if the steam generator is cooled at all and materials do not fail, the fraction will eventually tend asymptotically to unity, though by then the temperatures would probably be unrealistically high. How this ratio evolves during a transient must be predicted and not somehow used as an input to the analysis. Therefore, we reiterate the recommendation in our May 21, 2004, letter that the staff should improve the thermal-hydraulic analyses needed for these studies to enable a realistic prediction for the fraction of heat transferred to the steam generator, rather than estimating a value for this fraction based on the $1/7^{th}$ scale test results.

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

Sincerely,

Mario V. Bonaca

Mario V. Bonaca
Chairman

References:

1. Letter dated May 21, 2004, from Mario V. Bonaca, Chairman, ACRS, to William D. Travers, Executive Director for Operations, NRC, Subject: Resolution of Certain Items Identified by the Advisory Committee on Reactor Safeguards in NUREG-1740, "Voltage-Based Alternative Repair Criteria."

2. Letter dated August 25, 2004, from Luis A. Reyes, Executive Director for Operations, NRC, to Mario V. Bonaca, Chairman, ACRS, Subject: Resolution of Certain Items Identified by the Advisory Committee on Reactor Safeguards in NUREG-1740, "Voltage-Based Alternative Repair Criteria."

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

November 18, 2004

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

SUBJECT: LESSONS LEARNED FROM THE ACRS REVIEW OF THE AP1000 DESIGN

Dear Mr. Reyes:

During the 517th meeting of the Advisory Committee on Reactor Safeguards, November 4-6, 2004, we completed our deliberations regarding lessons learned from the review of the AP1000 design. As noted in our July 20, 2004 report, issues on the safety aspects of the AP1000 design certification application were resolved to our satisfaction. As has been our practice with previous certification reviews, we have developed a number of lessons learned that could be relevant to reviews of future applications or to operating plants. These are listed below in no particular order of significance along with explanations. We also had the benefit of the documents referenced.

Lessons Learned

1. Aerosol Removal in Containment: Westinghouse used the STARNAUA code for the limiting sequence to determine an appropriate average lambda for aerosol removal from containment. The staff used the MELCOR code for the same purpose and obtained reasonable agreement. In light-water reactors, in general, the lambda is dominated by gravitational settling. In AP1000, however, lambda is dominated by thermophoresis and diffusiophoresis. The modeling of these two phenomena is generally the least validated of the aerosol models and hence the model predictions are subject to significant uncertainty. In future certifications, it is likely that the applicant will rely on the MAAP code for determination of aerosol behavior in containment. These aerosol models in MAAP code need to be reviewed and the calculations need to be accompanied by sufficient uncertainty/sensitivity analyses.

2. Pi Groups: Validation of the computer codes relied on in the certification process depends on the use of scaling analyses to demonstrate that the results from small scale integral tests performed to validate the codes are applicable to the full-scale design. The degree of fidelity of the scaled facilities is evaluated using dimensionless parameters referred to as Pi groups. In theory, values of unity for the ratios of Pi groups represent perfect scaling. In reality, these ratios will differ somewhat from unity. There does not seem to be a technical basis for how much these can differ from unity and still represent acceptable scaling. The staff has said it will investigate whether such a technical basis can be developed. This study should be completed before the next design certification.

3. In-Vessel Retention/Fuel Coolant Interaction (FCI): Advanced reactor designs sometimes have severe accident management features (e.g., the in-vessel retention feature of AP1000 and the "core-catcher" feature of EPR) that may increase the potential for an energetic FCI. These features may invalidate the current resolution of this issue, which

relies on the expected very low frequency of occurrence. If so, the current FCI models, in our view, have insufficient experimental validation to rely on their predictions of the occurrence of an energetic thermal interaction and the magnitude of the energy release if such an interaction is judged to occur. Consequently, in such cases, it will be important to conduct sufficient sensitivity analyses on such parameters as: delayed trigger time; quantity of metallic components in the melt; system pressure; coolant characteristics, and initial conditions.

4. TRACE Code: The staff plans to rely increasingly on the TRACE thermal-hydraulic code to assess the design capability of a reactor to cope with LOCAs and transients. It is important that TRACE be made fully operational promptly. Assurance of the fidelity of any new models should be provided.

5. Safety Evaluation Reports (SERs): We have seen a steady improvement in the detail that has been included in SERs to make them a more complete and useful documentation of the extent of the staff's review and the bases for its decisions. The staff should maintain this high standard.

6. Phenomena Identification and Ranking Table (PIRT) Process: The increasing use of the PIRT process in design certification is a positive development. It will be especially important for designs such as the ACR-700 where the thermal-hydraulic phenomena are less well understood and less familiar.

7. Control Room Staffing: The operators of advanced plants will rely more on automatic controls and may operate a number of modular reactors from one control room. The staff needs to develop criteria and policy on the requisite control room staffing levels for these plants.

Sincerely,

Mario V. Bonaca

Mario V. Bonaca
Chairman

References:

1. Report dated July 20, 2004, from Mario V. Bonaca, Chairman, ACRS, to Nils J. Diaz, NRC Chairman, Subject: Report on the Safety Aspects of the Westinghouse Electric Company Application for Certification of the AP1000 Passive Plant Design.

2. Letter dated March 17, 2004, from Mario V. Bonaca, Chairman, ACRS, to William D. Travers, Executive Director for Operations, NRC, Subject: ACRS Reviews of the Westinghouse Electric Company Application for Certification of the AP1000 Plant Design-Interim Letter.

3. U.S. Nuclear Regulatory Commission, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design," Volumes 1 and 2, dated September 13, 2004.

4. Westinghouse Electric Company, AP1000 Design Control Document (DCD), APP-GW-GL-700, Revision 11, dated May 20, 2004.

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

November 18, 2004

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

SUBJECT: ACRS ASSESSMENT OF THE QUALITY OF SELECTED NRC RESEARCH PROJECTS

Dear Dr. Paperiello

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

- Effects of Chemical Reactions on Debris-Bed head Loss
    - This project is found to be slightly less than satisfactory. The results meet the research objectives for the most part.
- Experimental Studies of Loss-of-Coolant Accident Generated Debris Accumulation and Head Loss
    - This project marginally satisfies the research objectives. We have identified important deficiencies.
- Improvements to the MACCS Computer Code, Plume Model Adequacy
    - This project is found to be an excellent effort.

This independent evaluation of the quality of selected research projects was undertaken to satisfy the needs of the Office of Nuclear Regulatory Research (RES) and the requirements of the Government Performance and Results Act. The methods used by the ACRS for the quality review of research projects are described in detail in the accompanying report.

Some lessons have been learned from this first effort:

- The ACRS review panels experienced some challenges understanding what exactly had been asked of the researchers and the constraints imposed upon the research. It is imperative that the review panels be provided a copy of the NRC Form 189 and descriptions of any modifications to the scope of the research made by NRC.
- In some cases, the panels encountered delays in getting the documentation. We propose that, for future reviews, RES provide the ACRS the appropriate documents and these be screened by us before the review of a particular research activity is undertaken.

- It proved difficult to delve into large, multi-task projects in sufficient detail to evaluate the work in terms of the performance measures that we are using. We propose that, for future reviews, we work with you to focus reviews on particular efforts. The larger effort can be evaluated as part of our overall review of the NRC research program.

- It appears that it will be difficult to review projects at their very beginning. We propose that formal reviews with numerical scores not be undertaken until a research project has reached sufficient maturity.

We are now poised to undertake review of four additional research projects during fiscal year 2005. In an earlier communication to us, RES suggested that the next projects for review be selected from:

- Thermal Hydraulic Experimental Programs
- LOCA Frequency Determinations
- SPAR3 Quality Assessment
- Associated Circuits Analysis
- Pressurized Thermal Shock Re-evaluation
- Steam Generator Tube Integrity Under Severe Accident Conditions

We propose to meet with your staff to select projects for review over the next 12 months in light of the lessons learned from this first set of research quality reviews.

Sincerely,

Mario V. Bonaca

Mario V. Bonaca
Chairman

Attachment:
As stated

Assessment of the Quality of Selected NRC Research Projects by the Advisory Committee on Reactor Safeguards

November 2004

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

ABOUT THE ACRS

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

The ACRS is independent of the NRC staff and reports directly to the Commission, which appoints its members (currently 11 members). The ACRS is structured to provide a forum where experts representing many technical disciplines can provide independent advice that is factored into the Commission's decisionmaking process

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

ACRS operations are governed by the *Federal Advisory Committee Act* (FACA), which is implemented through NRC regulations at Title 10, Part 7, of the *Code of Federal Regulations* (10 CFR Part 7). ACRS operational practices encourage the public, industry, state and local governments, and other stakeholders to become involved in Committee activities.

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ABSTRACT

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

- Effects of Chemical Reactions on Debris-Bed Head Loss

  - This project was found to be slightly less than satisfactory . The results meet the research objectives for the most part.

- Experimental Studies of Loss-of-Coolant Accident Generated Debris Accumulation and Head Loss with Emphasis on the Effects of Calcium Silicate Insulation

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

- Improvements to the MACCS Computer Code, Plume Model Adequacy Evaluation

  - This project was found to be an excellent effort.

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ACRONYMS

Acronym	Definition
ACRS	Advisory Committee on Reactor Safeguards
BWR	Boiling Water Reactor
GPRA	Government Performance and Results Act
LOCA	Loss-Of-Coolant Accident
MAUT	Multi-Attribute Utility Theory
MACCS	MELCOR Accident Consequence Code System
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
PWR	Pressurized Water Reactor
RES	Office of Nuclear Regulatory Research

1 INTRODUCTION

The Nuclear Regulatory Commission (NRC) maintains a safety research program to ensure that its regulatory framework has a sound technical basis. The research effort is needed to support regulatory activities and agency initiatives while maintaining an infrastructure of expertise, facilities, analytical tools, and data to support regulatory decisions.

The Office of Nuclear Regulatory Research (RES) is required to have an independent evaluation of the effectiveness (quality) and utility of its research programs. This evaluation is mandated by the Government Performance and Results Act (GPRA). The Advisory Committee on Reactor Safeguards (ACRS) has agreed to assist RES by performing independent assessments of the quality of selected research projects. Quality assessment of individual research projects constitutes a new undertaking for the Committee; one that is quite different in scope and depth in comparison to the ACRS biennial review of the NRC research activities. During its March 2004 meeting, the ACRS approved a strategy for conducting such reviews [Ref. 1].

In this report, the ACRS presents the results of its quality assessment of three research projects. Two projects were reviewed within the general category of sump blockage research - Effects of Chemical Reactions on Debris-Bed Head Loss and Experimental Studies of Loss-of-Coolant Accident Generated Debris Accumulation on Head Loss. The third project evaluated was Improvements to the MACCS Computer Code: Plume Model Adequacy Evaluation. These projects, which deal with subjects of high current interest, were selected from a list of eight candidate projects suggested by RES.

A panel of three ACRS members was formed to review the sump blockage research projects and a second three-member panel reviewed the MACCS project. Each panel consisted of a chairman, a member with special expertise in the general area of the research program, and one other ACRS member. The panels conducted their detailed reviews of the assigned projects and presented their assessments to the full Committee. The discussions by the full Committee, which were concluded during the November 2004 meeting of the Committee, were intended to ensure consistency among the reviews of the various research projects.

An analytic/deliberative decisionmaking framework was adopted for evaluating the quality of NRC research projects. The ACRS considered the following general attributes in assessing the quality of the NRC research projects:

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

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

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

- Clarity of presentation
- Identification of major assumptions

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

2 METHODOLOGY FOR EVALUATING THE QUALITY OF RESEARCH PROJECTS

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

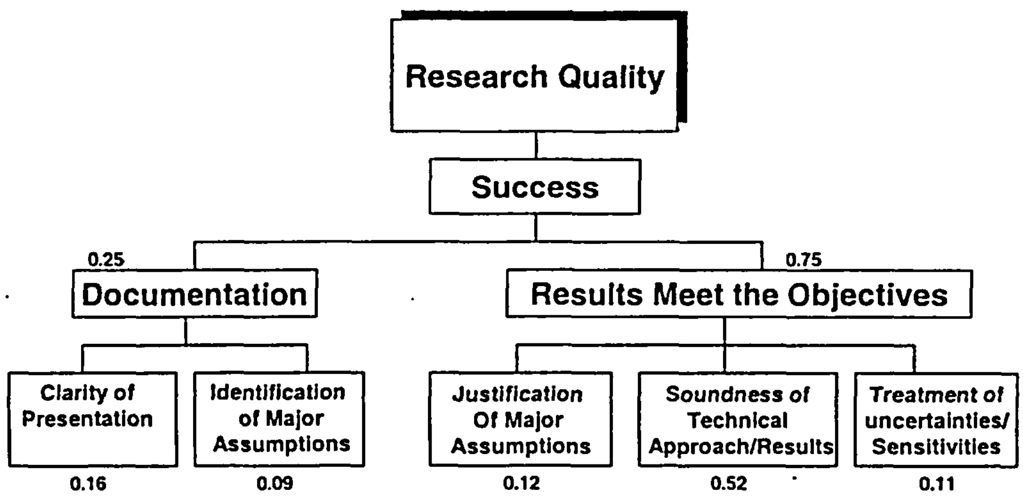

Figure 1 The value tree used for evaluating the quality of research projects

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

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

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

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

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

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

Table 1. Constructed Scales for the Performance Measures

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

3 RESULTS OF QUALITY ASSESSMENT

3.1 Sump Blockage Research

The 1992 clogging of intake strainers for containment spray water in Barsebäck-2, a boiling water reactor (BWR) in Sweden, renewed the safety questions associated with strainer clogging which, until then, had been considered as resolved. In response to the Barsebäck-2 event, the NRC launched research and development efforts to assess the vulnerability of U.S. BWRs to the loss of net positive suction head (NPSH) margin caused by excessive debris accumulation on suction strainers. Such efforts resulted in a number of corrective actions being taken in U.S. BWRs.

The NRC conducted further research to determine if the transport and accumulation of debris in a containment following a loss-of-coolant accident (LOCA) would impede the operation of the Emergency Core Cooling System (ECCS) in operating pressurized water reactors (PWRs). The research program included debris transport tests, debris settling tests, debris generation tests, debris-bed head loss tests, computational simulations, and various engineering analyses.

Two experimental projects in the area of sump blockage research were selected for review and quality evaluation. The results of these evaluations are discussed below.

3.1.1 Effects of Chemical Reactions on Debris-Bed Head Loss

This study was performed to assess the potential for chemically induced corrosion products to impede the performance of ECCS recirculation after a LOCA at PWR plants. A number of small-scale tests were performed to determine whether post-LOCA debris generation and sump-screen head loss in a PWR containment can be affected by chemical interactions between the ECCS/containment -spray water (which contains boric acid and sodium hydroxide at elevated temperatures) and exposed materials such as a metal surfaces, inorganic zinc-based paint chips, and fiberglass insulation debris. These tests were conducted in the Department of Civil Engineering of the University of New Mexico under the direction of Los Alamos National Laboratory. The results of this study were documented in Reference 6.

The consensus scores for this project are shown in Table 2. This project was found to be slightly less than satisfactory. The results meet the research objectives for the most part. Comments and conclusions within the evaluation categories are:

Documentation

- Clarity of presentation **(Consensus score = 5.0)**

  The report [Ref. 6] on the work included all appropriate material. The testing approach and apparatus were adequately described. Peer review comments were included. The report properly described the chemical environment to be expected in the containment during a large break LOCA. The appropriate literature was consulted for corrosion rates, solubilities and likely precipitants.

However, the executive summary of the report is not fully consistent with the body of the report. Major results were not clearly presented in Table 4-2 of the report and material in the table was not consistent with the text. The report did not indicate how peer review comments were addressed and it appeared that in some cases these comments had not been addressed in the project. Some conclusions did not reflect important insights.

No systematic attempt was made to pull the results together into a summary figure and compare them with theoretical predictions and previous work.

- Identification of major assumptions **(Consensus score = 5.0)**

  A number of explicit and implied assumptions arise in the research and these were identified satisfactorily.

Table 2. Summary Results of ACRS Assessment of the Quality of the Project on Effects of Chemical Reactions on Debris-Bed Head Loss

Performance Measures	**Consensus Scores**	**Weights**	**Weighted Scores**
Clarity of presentation	5.0	0.16	0.80
Identification of major assumptions	5.0	0.09	0.45
Justification of major assumptions	4.0	0.12	0.48
Soundness of technical approach/results	4.5	0.52	2.34
Treatment of uncertainties/sensitivities	4.0	0.11	0.44
		Overall Score:	**4.51**

Results Meet Objectives

- Justification of major assumptions **(Consensus score = 4.0)**

  Some assumptions were not justified sufficiently:

Limiting the test conditions to just those associated with large-break LOCAs was not justified.

Though a reference was cited for concluding "A high pH is essential to prevent fibers and small particles from coagulating and depositing on the sump screen," no further justification or discussion was provided.

It was asserted that not including phosphate and thiosulphate was a conservative conclusion, but no justification was provided.

It was assumed without justification or confirmation that "qualified application of coating systems are robust with respect to LOCA chemical environments."

- Soundness of technical approach and results **(Consensus score = 4.5)**

  The researchers were asked to conduct tests to determine the technical basis for deciding whether chemical interactions play a significant role in loss of NPSH. Implied in this request is the question of whether conditions will arise that lead to chemical interactions. The approach adopted in the research was insufficient to address this implied question.

  Corrosion rates under well-oxygenated spray conditions were not addressed though such conditions are likely to occur.

  The amount of data developed in the research is insufficient to be useful for quantitative analyses of the overall sump blockage issue.

  The oven used for corrosion tests did not have the capability to cover the range of temperature conditions of interest.

  Small sample sizes and short test durations yielded data that were insufficient to quantify and to resolve the uncertainties.

  Stirred vessels were not used in the corrosion rate studies. Consequently, corrosion rates may be underestimated because of local saturation effects.

  The small-scale loop test was not adequate. It could not maintain either constant approach velocities or constant temperatures that are basic requirements for the tests.

  Setting up artificial saturation conditions with chemicals that are known to precipitate in gelatinous form is not an appropriate approach to answer the fundamental questions to be addressed in the research.

  Labeling weight gains in samples as "negative corrosion" is misleading. A method to convert these weight gain results into corrosion rates should have been developed.

  The formation of gelatinous material is artificial and unrelated to LOCA conditions.

- Treatment of uncertainties and characterization of sensitivities **(Consensus score = 4.0)**

  Head loss characteristics of fibrous beds with different metal species, concentrations, approach velocities, temperature etc. were determined and thereby the important sensitivities were examined.

  The uncertainties in the results were not addressed.

3.1.2 Experimental Studies of Loss-Of-Coolant Accident Generated Debris Accumulation and Head Loss

This experimental program was conducted to generate data on the head loss associated with calcium silicate (CalSil) insulation accumulated on PWR sump screens, with or without other insulation materials such as fiberglass or reflective metallic insulation, and to determine the suitability of the NUREG/CR-6224 [Ref. 7] head loss correlation for CalSil head loss calculations. The results of this study were documented in Reference 8

The consensus scores for this project are shown in Table 3. This project marginally satisfied the research objectives. The ACRS identified important deficiencies. Comments and conclusions within the evaluation categories are:

Table 3 Summary Results of the ACRS Assessment of the Quality of the Project on Experimental Studies of LOCA Generated Debris Accumulation and Head Loss

Performance Measures	**Consensus Scores**	**Weights**	**Weighted Scores**
Clarity of presentation	6.5	0.16	1.04
Identification of major assumptions	4.5	0.09	0.41
Justification of major assumptions	3.0	0.12	0.36
Soundness of technical approach/results	3.0	0.52	1.56
Treatment of uncertainties/sensitivities	2.5	0.11	0.28
		Overall Score:	**3.65**

Documentation

- Clarity of presentation **(Consensus score = 6.5).**

  The report on the work was complete and included all appropriate materials such as tables of raw data and photographs of Scanning Electron Microscopy of debris-bed morphologies. The report adequately described the test apparatus and procedures. The appropriate results as well as the conditions of tests were included in the report. However, there were some concerns about the inclusion of the results of failed and shakedown tests in the report.

- Identification of major assumptions **(Consensus score = 4.5)**

  The report listed a number of major assumptions. However, the list was not complete and the assumptions were scattered throughout the report rather than being listed separately.

Results Meet Objectives

- Justification of major assumptions **(Consensus score = 3.0)**

Many assumptions were not justified sufficiently:

No justification was provided on how artificially generated debris properly simulated real debris.

No technical basis was provided for the assumption that small/medium break LOCAs would be bounded by screen loading of 1 ft^3/ft^2 and large-break LOCA by 10 ft^3/ft^2.

No justification was provided for the ratio of the volume of CalSil to volume of fibrous debris used in the tests. The report noted that "experience and engineering judgment" were used to "reasonably" represent this parameter. This is a weak justification without having more backup information.

No real justification was provided for the values of the selected parameters used in the test compared to expected conditions during a PWR LOCA.

It was assumed that NUKON was representative of the broad class of fiber insulations without any justification for this assumption.

The report did not clearly define what is meant by a "uniform" accumulation pattern.

The report did not provide a clear definition of "thin bed", what criterion should be used to determine its occurrence, and how it is possible for the effective surface area of the particles to increase by almost an order of magnitude when it occurs.

- Soundness of technical approach and results **(Consensus score = 3.0)**

  The technical approach was to simulate appropriate PWR containment debris accumulation and head-loss conditions with flexibility for controlling local flow conditions, debris quantity and other important parameters and with the capability of taking applicable measurements and visual observations of the phenomena under examination. As this was strictly a test of the applicability of a previously developed empirical correlation, there was an insufficient test matrix and number of tests to provide a good assessment. The major objective of the work was not accomplished since there was only a limited set of data, half of which was under failed conditions.

  The range, quality, and amount of data are insufficient to resolve the inconsistencies and anomalies observed.

  Overly optimistic statements are made about validation of the correlation. This is in sharp contrast to the numerous anomalous and inconsistent features of the results which were insufficiently investigated and not resolved.

  The test apparatus appears to be under-designed for the purpose. There was no independent control of temperature or pump speed.

  Not much insight was developed on the real issue of "the thin bed" effect.

  The process of adjusting the value of a major independent variable (specific surface area) to provide a fit to the largest delta-P data point was inappropriate. The obtained data were presented on head-loss versus approach velocity plots along with the NUREG/CR-6224 correlation prediction, and the CalSil specific surface area (and the sludge density when bed compaction occurred) was adjusted until the correlation predicted the higher velocity data points. This circular approach does not "establish a technical basis for extending the applicability of the NUREG/CR-6224 correlation from porous debris beds on boiling water reactor (BWR) suppression pool strainers to debris beds on PWR sump screens or other flow blockages," as set forth in the objectives of the test program.

  This research did not provide a better understanding of the behavior of particulate filters nor did it provide a good predictive capability.

  The range of the head loss data was limited by the apparatus and cannot be extrapolated with confidence to large LOCA conditions where the bed thickness may be an order of magnitude higher than was tested.

- Treatment of uncertainties and characterization of sensitivities **(Consensus score = 2.5)**

  Sensitivity of the head loss characteristics of fibrous beds to some parameters such as different CalSil volume ratio, approach velocity, and temperature were examined. However, no attempt was made to characterize or quantify the many uncertainties.

The objective of validating a previously developed correlation requires enough data to characterize the uncertainties. However there were insufficient data to do this.

3.2 Improvements to the MACCS Computer Code

The NRC uses MACCS consequence code system for Level 3 Probabilistic Risk Assessment (PRA) consequence analyses, planning for emergencies, and cost-benefit analyses. MACCS uses a Gaussian plume model for atmospheric transport and dispersion . This model has been criticized as being overly simplistic. The justification for its use has been that only average values of metrics of interest over numerous weather consequences are used in the regulatory arena and that this averaging compensates for the loss of fine structure in the meteorology that occurs away from the point of release. The simple Gaussian plume model has been retained because of the desire to have models covering the entire path through the environment, including the food and water pathways and covering essentially a lifetime of exposure to a contaminated environment, which can run in short times on personal computers.

The NRC initiated a research effort to test the assumption about the adequacy of simplified Gaussian plume model through comparison with more complex models. The models compared were: MACCS, the simplified model; LODI (Lagrangian Operational Dispersion Integrator), a state-of-the-art, 3-dimensional advection-diffusion code using a Lagrangian stochastic, Monte Carlo method; and RASCAL (Radiological Assessment System for consequence analysis), which uses a Lagrangian trajectory, Guassian puff model. The objective of this study was to see if the average atmospheric transport and dispersion results from these three codes were sufficiently close that a more complex model is not required for the NRC purposes of emergency planning and cost -benefit analysis or different enough that the NRC code should be modified to provide more rigorous atmospheric transport and dispersion. The results of this study were documented in Reference 9.

The consensus scores for the project are shown in Table 4. This project was found to be an excellent effort. Comments and conclusions within the evaluation categories are:

Documentation

- Clarity of presentation **(Consensus score = 9.0)**

  The research had a clear and unambiguous direction that was followed appropriately.

  A complete, well-written and understandable report was issued. The report was indexed and referenced. Graphics were useful. All appropriate material was included in the main body of the report or in appendices.

  Enough material was included for an independent investigator to make comparisons to results from other codes.

  A peer-reviewed paper was developed from this work.

Table 4 Summary Results of ACRS Assessment of the Quality of the Project on Improvements to the MACCS Computer Code,
Plume Model Adequacy Evaluation

Performance Measures	Consensus Scores	Weights	Weighted Scores
Clarity of presentation	9.0	0.16	1.44
Identification of major assumptions	9.5	0.09	0.86
Justification of major assumptions	7.0	0.12	0.84
Soundness of technical approach/results	8.0	0.52	4.16
Treatment of uncertainties/sensitivities	4.5	0.11	0.50
		Overall Score:	**7.80**

- Identification of major assumptions **(Consensus score = 9.5)**

  All major assumptions that differentiate the modeling of the codes for comparison were identified.

  The approach used for making the code inputs consistent was discussed clearly.

  Few assumptions were necessary in the work.

Results Meet Objectives

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

  Assumptions made on input for the codes were well justified.

- Soundness of technical approach and results **(Consensus score = 8.0)**

  Directions given in the statement of work were followed appropriately.

  The project analyzed the only available site for which there are sufficient three dimensional data on year-around wind directions and temperatures to sufficient distances for a useful comparison of code predictions.

The figures of merit adopted for the comparisons of code predictions were appropriate. Only the meteorological aspects of the codes were exercised in the effort.

Comparisons of the predictions of the RASCAL and RACHET codes added to the utility of the results.

The project did not necessarily define the maximum possible differences among code predictions. It might have been possible to do this by developing a fictional site and manufactured meteorological data deliberately chosen to maximize differences in code predictions. This, however, would have been well beyond the scope of the project.

- Treatment of uncertainties and characterization of sensitivities **(Consensus score = 4.5)**

The uncertainties and sensitivities of the MACCS code and the 3-D LODI code have been addressed extensively in many other studies. The report provided sufficient references to these studies.

The purpose of the exercise was to gain insight on the level of uncertainty in code predictions and this was accomplished well. The comparison of the discrepancies of the code results, however, should have been placed in the larger context of the uncertainties that are present in such evaluations.

5. REFERENCES

1. Letter Dated April 26, 2004, from Mario V. Bonaca, Chairman, ACRS, to Ashok C. Thadani, Director, Office of Nuclear Regulatory Research, NRC, Subject: Proposed Approach to Assess the Quality of NRC Research Projects.

2. National Research Council, *Understanding Risk: Informing Decisions in a Democratic Society*. National Academy Press, Washington, D.C, 1996.

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5. Keeney, R.L., and H. Raiffa, *Decisions with Multiple Objectives: Preferences and Value Tradeoffs*, Wiley, New York, 1976.

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9. Molenkamp C.R., N.E. Bixler, C. W. Morrow, J.V. Ramsdell Jr., and J.A. Mitchell, "Comparison of Average Transport and Dispersion Among a Gaussian Model, a Two-Dimensional Model and a Three-Dimensional Model," Prepared for the U.S. Nuclear Regulatory Commission, Lawrence Livermore National Laboratory, Draft NUREG/CR-6853, 2004.

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

November 19, 2004

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

SUBJECT: DRAFT PROPOSED RULE ON POST-FIRE OPERATOR MANUAL ACTIONS

Dear Chairman Diaz:

During the 517th meeting of the Advisory Committee on Reactor Safeguards, November 4-6, 2004, we met with representatives of the NRC staff to review the draft proposed rule on post-fire operator manual actions. Our Fire Protection Subcommittee also reviewed this matter during a meeting on October 27, 2004. During our reviews, we had the benefit of discussions with representatives of the NRC staff and the nuclear industry and interested members of the public. We also had benefit of the documents referenced.

RECOMMENDATION

The draft proposed rule should be published for public comment.

BACKGROUND

Nuclear power plant fire protection regulations and associated guidelines prescribe fire protection features that are intended to ensure that at least one means of achieving and maintaining safe shutdown conditions will remain available during or after any postulated fire. Currently, paragraph III.G.2 of Appendix R specifies three acceptable methods for protecting the safe shutdown capability of one of the redundant shutdown trains from a fire when these trains are located in the same fire area. These options are:

- a 3-hour fire barrier, or
- a 1-hour fire barrier with fire detectors and an automatic fire suppression system, or
- 20 feet of horizontal separation with no intervening combustibles and with fire detectors and an automatic fire suppression system.

During recent NRC inspections of licensee fire protection programs, the staff has identified concerns regarding licensee compliance with these requirements. Currently, licensees relying on operator manual actions which have not been reviewed and approved by NRC in accordance with 10 CFR 50.12 are considered to be in noncompliance with NRC regulations. In addition to the compliance issue, the staff is also concerned that some unapproved operator manual actions may not be feasible. However, the staff has acknowledged that certain operator manual actions may be safe and effective when performed under appropriate conditions.

DISCUSSION

To address this issue, the staff initiated rulemaking to explicitly permit the use of operator manual actions in lieu of using physical barriers or separation to achieve and maintain safe shutdown in the event of a fire where redundant trains are located in the same fire area. The staff has also developed a draft regulatory guide that includes acceptance criteria for evaluating and demonstrating the feasibility and reliability of post-fire operator manual actions as an acceptable alternative to the physical barriers and separation.

In the proposed rule, crediting operator manual action would be predicated on the requirement that the area where the fire occurs is equipped with fire detection and automatic suppression systems. The staff contends that fire detection and automatic suppression systems are necessary to preserve the physical component of a plant's fire protection defense-in-depth.

The staff has stated that the proposed change to paragraph III.G.2 is intended to (1) maintain safety and increase public confidence, (2) provide quality and uniformity in licensee assessments and documentation, (3) reduce unnecessary regulatory burden associated with the exemption or deviation process, and (4) result in more efficient use of resources by licensees and the NRC. We agree that the draft proposed rule and the accompanying regulatory guide will be effective in meeting objectives 1 and 2.

We have been informed by nuclear industry representatives that, with the inclusion of the requirement for fire detection and automatic suppression in the proposed rule, objectives 3 and 4 are unlikely to be achieved since many exemption and/or deviation requests will continue to be required. We believe, however, that many plants will use the new rule rather than the exemption process in 10 CFR 50.12.

During our reviews, we questioned the inclusion of a provision in paragraph III.P.2 of the proposed rule which requires the consideration of security events when evaluating the feasibility of operator manual actions. We have been informed by the staff that this provision will be removed and handled separately in a manner that properly protects sensitive information. We agree with the staff's decision.

The proposed rule requires time-authenticated walkdowns of each credited operator manual actions at intervals not to exceed 12 months to demonstrate the feasibility of the action. Since some plants may be seeking credit for numerous actions, the requirement in paragraph III.P.2 (d) for repetition of such walkdowns for all credited operator manual actions may be excessive and unnecessary. An initial demonstration for each credited action combined with routine operator training may be sufficient.

Sincerely,

Mario V. Bonaca

Mario V. Bonaca
Chairman

Additional Comments by ACRS Member Stephen L. Rosen

I agree with the Committee's recommendation that the staff should publish the draft proposed rule for public comment. However, I believe that the staff's strategy on this issue misses an important opportunity to move the industry and agency towards a more risk-informed, performance-based approach to the control of fire risk.

As noted in the rulemaking plan attached to SECY 03-100, the staff has acknowledged that replacing a passive fire barrier or automatic suppression system with human performance activities can increase risk but for some simple operator manual actions, the risk increase associated with human performance may be minimal. The staff also stated that the introduction of feasible operator manual actions could result in a minimal increase in overall risk and has concluded (on a plant-specific basis) that the use of certain specific operator manual actions for the operation of co-located safe shutdown trains provides an adequate level of fire safety and satisfies the underlying purpose of the fire protection regulations.

I agree with the staff and I would add that each fire scenario must be evaluated on its own merits, taking into account passive features, combustibles, ventilation, detection, automatic suppression (if available), fire brigade activities and feasibility and reliability of operator manual actions.

Yet, in the draft proposed rule, the staff has chosen to take a more deterministic approach by including provisions in the rule that require that the affected fire area be equipped with suppression and detection before licensees may take credit for operator manual actions even when the operator actions are performed in other remote areas of the plant that may be unaffected by the fire.

The rule requires a time-line analysis of postulated fires. These analyses done in accordance with the regulatory guide accompanying the rule will be adequate to reveal circumstances where credit should not be granted for post-fire operator manual actions.

The staff should pursue a more risk-informed and performance based approach to the control of fire risk. The requirement for automatic fire suppression in the area of the postulated fire should be removed from the draft proposed rule.

References:

1. U.S. Nuclear Regulatory Commission, Draft Federal Register Notice, Subject: Proposed Rule, Fire Protection Program - Post-Fire Operator Manual Actions.

2. Memorandum from Catherine Haney, NRR to John T. Larkins, ACRS, Subject: Review of Post-Fire Operator Manual Actions Proposed Rule, September 20, 2004.

3. Draft Regulatory Guide DG-1136, Guidance for Demonstrating the Feasibility and Reliability of Operator Manual Actions in Response to Fire, September 2004.

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

December 7, 2004

MEMORANDUM TO: Luis A. Reyes
Executive Director for Operations

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

SUBJECT: PROPOSED RULE- AP1000 DESIGN CERTIFICATION

During the 518th meeting of the Advisory Committee on Reactor Safeguards, December 2-4, 2004, the Committee considered the proposed AP1000 design certification rule. The Committee has no objection to the staff's proposal to issue this proposed rule for public comments. The Committee would like the opportunity to review the draft final rule after reconciliation of public comments.

Reference:
Memorandum dated December 1, 2004, from William D. Beckner, Program Director, Office of Nuclear Reactor Regulation, to John T. Larkins, Executive Director, Advisory Committee on Reactor Safeguards, Subject: Proposed Rule- AP1000 Design Certification.

cc: A. Vietti-Cook, SECY
W. Dean, OEDO
I. Schoenfeld, OEDO
J. Dixon-Herrity, OEDO
M. Crutchley, NRR
W. Beckner, NRR
L. Quinones-Navarro, NRR
J. Wilson, NRR
A. Levin, RES

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

December 9, 2004

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

SUBJECT: INTERIM LETTER – REGULATORY STRUCTURE FOR NEW PLANT LICENSING: TECHNOLOGY-NEUTRAL FRAMEWORK

Dear Mr. Reyes:

During the 518th meeting of the Advisory Committee on Reactor Safeguards, December 2-4, 2004, we met with representatives of the NRC staff and discussed the staff's draft Commission Paper, "Second Status Paper of the Staff's Proposed Regulatory Structure for New Plant Licensing and Update on Policy Issues Related to New Plant Licensing." We also had the benefit of the documents referenced.

It is inevitable that the agency will be asked to certify reactor designs that are markedly different from current light-water reactors (LWRs). It will be difficult to apply current regulations to such designs. Thus, we strongly support the effort to develop a technology-neutral regulatory framework. We consider the completion of this effort to be essential for the efficient and effective certification of non-LWR designs. It can also be useful for risk-informed decisionmaking in general.

While it may be some time before an application for the certification of a non-LWR design is submitted, the development of a technology-neutral framework is likely to be a long-term effort. Thus, the ongoing commitment is appropriate. Our discussions with the staff to date have revealed that the staff has a strategic approach and is articulating and addressing difficult technical and policy issues. We believe the work has the potential to provide a more efficient and more coherent regulatory system. We look forward to continued discussion of the staff's progress.

Sincerely,

Mario V. Bonaca

Mario V. Bonaca
Chairman

References:
1. DRAFT Commission Paper, Second Status Paper of the Staff's Proposed Regulatory Structure for New Plant Licensing and Update on Policy Issues Related to New Plant Licensing (Predecisional).

2. U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, NUREG-xxxx, Regulatory Structure for New Plant Licensing, Part 1: Technology-Neutral Framework (working draft Predecisional), December 2004.

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

December 10, 2004

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

SUBJECT: ESTIMATING LOSS-OF-COOLANT ACCIDENT FREQUENCIES THROUGH THE ELICITATION PROCESS

Dear Chairman Diaz:

During the 518th meeting of the Advisory Committee on Reactor Safeguards on December 2-4, 2004, we reviewed the draft NUREG Report, "Estimating Loss-of-Coolant Accident (LOCA) Frequencies through the Elicitation Process," (Reference 1). Our Subcommittee on Regulatory Policies and Practices reviewed this matter during a meeting on November 16, 2004. During these reviews, we had the benefit of discussions with the NRC staff and of the documents referenced.

RECOMMENDATION

- The draft NUREG Report should be revised prior to being issued for public comment.

DISCUSSION

In a staff requirements memorandum (SRM) dated March 31, 2003 (Reference 2), the Commission directed the staff to develop a risk-informed alternative to the current requirements in 10 CFR 50.46 related to the analysis of the performance of emergency core cooling systems (ECCS) during LOCAs. The focus of this effort is the selection of a risk-informed transition break size (TBS) for the alternative design-basis LOCA. In an SRM dated July 1, 2004 (Reference 3), the Commission directed the staff to use LOCA frequencies derived from an expert-opinion elicitation process, supported by historical data and fracture mechanics and other relevant information to determine an appropriate alternative break size. This alternative break size could be the break size that has a mean frequency of occurrence of 10^{-5} per reactor year.

Expert-opinion-based probability distributions of uncertain quantities have been used extensively in probabilistic risk assessments (PRAs) starting with WASH-1400 (Reference 4). The NUREG-1150 studies (Reference 5) formalized the process of elicitation and utilization of expert judgments. Later, major studies sponsored by both government and industry refined the process and applied it to seismic risk assessments (References 6 and 7).

Generating distributions from expert opinions involves the selection of the experts, elicitation of their judgments, and the processing of the individual judgments to produce a composite distribution.

An important question is what kinds of sources of uncertainties the expert-opinion produced distribution of LOCA frequencies should represent. Ideally, this distribution should reflect the uncertainties due to all scenarios and mechanisms with the potential of causing or contributing to a LOCA. Plant-to-plant variability is an important source of uncertainty. In addition, these uncertainties should reflect the opinions of the expert community at large (i.e., the composite distribution should represent the uncertainties in the state of the art).

Of course, this ideal situation is very difficult to achieve. It is impossible to elicit the opinions of the whole community of experts and the analysts have to rely on a group of experts that is representative of the range of the community's views. The expert panel in this study was selected carefully to represent a broad range of expertise.

In the elicitation process, it is very important that the analysts ask the experts questions that will lead to the development of a composite distribution useful to the decisionmakers. The experts must fully understand the questions and the underlying assumptions. In this context, we have identified several issues that must be addressed.

The Report does a good job describing the limitations of the results with respect to the scenarios and mechanisms considered. The elicitation process assumed normal plant operational cycles and did not consider the effects of operating profile changes (e.g., due to power uprates). The effects of "rarer" transients, such as seismic events, were also not considered.

It is unclear to what extent the experts considered plant-to-plant variability. The Report states that the elicitation was focused on developing generic, or average, values for the fleet of plants. The panelists were instructed to account for broad plant specific factors. It states further that "the uncertainty bounds do not represent LOCA frequency estimates for individual plants that deviate from the generic values." We conclude that plant-to-plant variability may not be fully reflected in the composite distribution. This conclusion is consistent with the statement in Section H-1 that "Several panelists expressed that safety culture deficiencies at a single plant could increase the LOCA frequencies at that plant by a factor of 10 or more."

The decisionmakers will have to compensate for the uncertainties created by these limitations by evaluating their impact and resorting to structuralist defense-in-depth measures (e.g., by adding conservatism to the ultimate results of the study). The Report should include a better explanation of what a generic frequency value for the fleet of plants means and to what extent plant-to-plant variability affected the results.

The LOCA size categories are defined by determining an effective break size using correlations that relate break size to the flow rate associated with the break. We were told that some experts assumed that the calculated break size corresponded to double the flow rate while others did not. The question is whether one uses the flow rate from one end of the severed pipe or this flow rate is doubled to include coolant loss from both ends. The analysts should correct the results to make them consistent. The Report should state clearly what the understanding of the experts was when they answered questions about the LOCA size categories.

The Report acknowledges that possible ways for correcting the individual expert opinions to compensate for potential biases and the method of aggregation of these opinions can have a significant impact on the results. Sensitivity analyses are presented to show the impact of a number of approaches.

The aggregation method chosen is what the Report calls "geometric" averaging, e.g., the group's 95th percentile is taken to be the nth root of the product of the 95th percentiles provided by n experts. The results from "arithmetic" averaging are also presented as a sensitivity analysis. This means that the group's estimate is taken to be the sum of the individual estimates divided by n. We note that these averaging methods deal with the characteristic values of the individual distributions directly [i.e., the group median and the group 95th percentile are the geometric (or arithmetic) average of the individual medians and 95th percentiles, respectively]. This practice is at variance with the methods employed in References 5-7, in which the arithmetic averaging method is applied to the probability distributions of the experts.

As we stated above, the analysts performed numerous sensitivity analyses. Yet, the Executive Summary lists only the "baseline" results and states: "This study does not recommend whether the LOCA frequency estimates corresponding to the baseline or a particular sensitivity analysis should be used in applications." By not stating what, in their judgment, the most appropriate distribution is, the analysts place an extraordinary burden on the users of the results who are generally not familiar with the intricacies of expert opinion elicitation and aggregation. The final distribution reported in the Executive Summary should be the composite distribution that the analysts, based on the sensitivity analyses, believe represents the expert community's current state of knowledge regarding LOCA frequencies.[1] Providing such a distribution would also be consistent with PRA practice, which utilizes epistemic distributions for the frequencies of initiating events (in this case, LOCA frequencies) and not confidence intervals for individual percentiles. Thus, the results would be useful to a broader class of applications than just the selection of the TBS.

During our December, 2004 meeting, the analysts presented to us results from the aggregation method that averages probability distributions (what they called a "mixture distribution"). They also provided us with a revised chapter of the Report. It is evident that this work is still in progress and is not ready for public comment.

We look forward to reviewing the Report after the staff responds to our comments.

Sincerely,

Mario V. Bonaca

Mario V. Bonaca
Chairman

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[1]This means that the analysts should act as a Technical Facilitator/Integrator, a concept described in detail in NUREG/CR-6372 (Reference 7).

References:
1. Memorandum dated November 4, 2004, from Michael E. Mayfield, Director, Division of Engineering Technology, RES, to John T. Larkins, Executive Director, ACRS, Subject: Transmittal of Draft NUREG on Passive System LOCA Frequency Development for use in Risk-Informed Revision of 10 CFR 50.46, Appendix K to Part 50, and GDC and Appendices (Pre-Decisional For Internal ACRS Use Only).
2. Staff Requirements Memorandum dated March 31, 2003, from Annette L. Vietti-Cook, SECY, NRC, to William D. Travers, EDO, NRC, Subject: SECY-02-0057 - Update to SECY-01-0133, "Fourth Status Report on Study of Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50 (Option 3) and Recommendations on Risk-Informed Changes to 10 CFR 50.46 (ECCS Acceptance Criteria)."
3. Staff Requirements Memorandum dated July 1, 2004, from Annette L. Vietti-Cook, SECY, NRC, to Luis A. Reyes, EDO, NRC, Subject: Staff Requirements - SECY-04-0037, "Issues Related to Proposed Rulemaking to Risk-Inform Requirements Related to Large Break Loss-of-Coolant-Accident (LOCA) Break Size and Plans for Rulemaking on LOCA With Coincident Loss-of-Offsite Power".
4. U.S. Nuclear Regulatory Commission, Reactor Safety Study, an Assessment of Accident Risks in U.S. Nuclear Power Plants, Report NUREG-75/014 (WASH-1400), 1975.
5. U.S. Nuclear Regulatory Commission, Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants, Report NUREG-1150, 1990.
6. Electric Power Research Institute, *Seismic Hazard Methodology for the Central and Eastern United States*, EPRI Report NP-4726, 1988.
7. R.J. Budnitz, G. Apostolakis, D.M. Boore, L.S. Cluff, K.J. Coppersmith, C.A. Cornell, and P.A. Morris, *Recommendations for Probabilistic Seismic Hazard Analysis: Guidance and Use of Experts*, Report NUREG/CR-6372, 1997.

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

December 10, 2004

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

SUBJECT: SAFETY EVALUATION OF THE INDUSTRY GUIDELINES RELATED TO PRESSURIZED WATER REACTOR SUMP PERFORMANCE

Dear Mr. Reyes:

Thank you for your letter of November 26, 2004, which responded to our letter of October 18, 2004, on the staff safety evaluation (SE) of the Industry Guidelines Related to Pressurized Water Reactor Sump Performance.

We appreciate the staff's desire to move ahead to resolve Generic Safety Issue-191, "Assessment of Debris Accumulation on Pressurized Water Reactor (PWR) Sump Performance." As licensees attempt to use the guidance, we anticipate that they will have to cope with several technical problems due to errors in the suggested methods. We disagree with your statement that the knowledge limitations are clearly identified and addressed in the SE. In our letter, we identified a number of these limitations. The purpose of this letter is to restate several of the limitations, and to respond to some of the staff's replies.

The head loss correlation in NUREG/CR-6224 (Ref. 1) is not entirely empirical, as claimed by the staff, but rests in part on the theoretical representation of two physical phenomena: the mechanical compression of the bed and the limit of this compression. The theoretical models for these phenomena are erroneous. Although some results may be predicted with apparent adequacy, the faulty models lead to some conclusions that are obviously at odds with reality. For example, correlating bed compression with the pressure gradient is inconsistent with standard methods in the literature and cannot explain the compression of a fiber bed by the imposed pressure from a superposed particulate bed, as in the "thin bed effect." In addition, the NUREG/CR-6224 equation for the compression limit would predict that a fiber bed could be compressed up to the limiting particulate bed density even when there are no particles present, which makes no sense. The foundation of the correlation of data must be theoretically sound if the user of the guidance is to extrapolate a very limited range of data to real plant conditions.

The Committee commented in its letter that the effect on coatings of a two-phase jet is not well understood. The staff agreed "that the nature and effects of a two-phase LOCA jet on coatings are not well understood and that there is a lack of data on coatings." However, the staff still believes that the guidance is acceptable because of "precedents set by past applications approved by the staff and accepted by the ACRS or based on the staff approach of applying conservative assumptions to bound the unknowns." Unfortunately, because the phenomena are not well known, the uncertainties are also not well known, so the staff's "conservative assumptions" are only engineering judgment, without any technical basis.

We are pleased that the staff has alerted the American Nuclear Society to our technical comments on the 1988 ANSI/ANS standard (Ref. 2). However, the claim that Appendix I of the SE contains a "detailed evaluation" of this model is incorrect. Appendix I explains how to use the model, but repeats the technical errors contained in the model, such as the assumption of an "asymptotic plane" beyond which there are no supersonic effects, and the use of a stagnation density to describe a high-velocity stream. As a result, we have not seen convincing arguments that it is conservative to use the ANSI/ANS standard to determine the size of the zone of influence.

The staff claims that it is appropriate to assume that the debris bed is homogeneous, with the particles uniformly distributed through it. The staff also claims to supply guidance about the "thin bed effect," which is the extreme case where all the particles concentrate in a single layer. These two arrangements of the debris are limiting situations of the general case in which various degrees of inhomogeneity occur; they cannot be true simultaneously. The guidance should address a wider range of possible inhomogeneities. It should allow the user to predict how much inhomogeneity occurs and the resulting head loss. There also needs to be better guidance on how the head loss evolves with time (as observed in experiments documented by NRC contractors), apparently because of the development of inhomogeneities, and on how extreme inhomogeneity can give rise to anomalously high head loss.

The guidance is also inadequate for evaluating downstream effects. It merely lists issues to be considered. It does not explain how to determine whether the issues are resolved, or how to perform an "integrated evaluation". Licensees will have to derive the acceptance criteria themselves.

There is also no useful guidance on chemical effects. The staff has only told the industry not to get caught by unexpected results from the ongoing experimental program.

We continue to believe that both the SE and the Nuclear Energy Institute guidance document contain technical faults and limitations that will have to be corrected at some stage in order for the methods to be sufficiently robust and durable to support sound regulatory decisions.

Sincerely

Mario V. Bonaca

Mario V. Bonaca
Chairman

References:
1. NUREG/CR-6224, "Parametric Study of the Potential for BWR WCCS Strainer Blockage Due to LOCA Generated Debris," G. Zigler et.al., October 1995.
2. ANSI/ANS-58.2-1988, "Design Basis for Protection of Light-Water Nuclear Power Plants Against the Effects of Postulated Pipe Rupture," American Nuclear Society, October 6, 1988.

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

December 17, 2004

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

SUBJECT: RISK-INFORMING 10 CFR 50.46, "ACCEPTANCE CRITERIA FOR EMERGENCY CORE COOLING SYSTEMS FOR LIGHT-WATER NUCLEAR POWER REACTORS"

Dear Mr. Reyes:

During the 518th meeting of the Advisory Committee on Reactor Safeguards on December 2-4, 2004, we reviewed a draft version of a proposed rule for a voluntary alternative to 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems (ECCS) for Light-Water Nuclear Power Reactors" (Reference 1). We also reviewed draft proposed rule language (Reference 2) during the 517th meeting on November 4-6, 2004. Our Subcommittee on Regulatory Policies and Practices reviewed this matter during a meeting on October 28-29, 2004. During these reviews, we had the benefit of discussions with the NRC staff, the Nuclear Energy Institute, Westinghouse Owners Group, and members of the public. We also had the benefit of the documents referenced. Although the proposed rule language has not been finalized, we present our views on some of the basic elements of a risk-informed 10 CFR 50.46.

CONCLUSIONS AND RECOMMENDATIONS

1. A risk-informed 10 CFR 50.46 should maintain defense in depth by including requirements intended to provide reasonable assurance of a coolable core geometry for breaks up to the double-ended guillotine break (DEGB) of the largest pipe in the reactor coolant system.

2. The results of the expert opinion elicitation need to be further reviewed and assessed by the staff before finalizing the selection of the transition break size. Nevertheless, it appears that a transition break size corresponding to the single-ended rupture of the largest pipe attached to the reactor coolant system bounds the range of break sizes corresponding to a frequency of 1 $\times 10^{-5}$/year.

3. A better quantitative understanding of the possible risk benefits of a smaller transition break size is needed to arrive at a final choice of the transition break size. If the defense-in-depth capability to mitigate breaks greater than the transition break size is maintained, a smaller choice of transition break size may be supportable.

DISCUSSION

Loss-of-coolant accidents (LOCAs) have been the focus of nuclear plant safety since the first commercial reactor designs. LOCAs can arise from many causes, and the current design basis requires the demonstration of the capability to mitigate a spectrum of break sizes up to the

DEGB of the largest pipe in the reactor coolant system. Since the Three-Mile Island accident and the earliest probabilistic risk assessments, it has been recognized that small-break LOCAs are more risk significant than large-break LOCAs (LBLOCAs). This has been reflected in operator training, procedures, etc., but it has not been fully reflected in the regulations.

Although the design-basis LBLOCA requirements have led to the development of robust safety systems, the burdens imposed by the design-basis requirement to deal with the DEGB of the largest pipe in the reactor coolant system are not commensurate with its risk importance, and the resulting requirements may have inhibited opportunities to optimize the system response for the entire range of challenges that must be met including those more likely to occur. For example, the current LBLOCA requirements result in rapid diesel start times. The testing necessary to demonstrate that these start times can be achieved increases wear on the diesel and reduces the reliability of the diesel in the case of more risk-important sequences that do not require such rapid start times.

A risk-informed 50.46 rule will be an enabling rule. It will not impose any specific changes that would be made in the design or operation of nuclear power plants. It will permit licensees to make changes that may decrease risk by optimizing system responses to accidents that are more likely to occur, and changes such as power uprates that will result in risk increases.

In a Staff Requirements Memorandum (SRM) dated July 1, 2004, the Commission approved the development of a proposed rule to risk-inform the requirements addressing LBLOCAs. The proposed rule was to use the initiating event frequencies from the expert elicitation process and other relevant information to guide the determination of an appropriate alternative break size. The staff was also to ensure that any changes to the plant or operating procedures would follow a change process consistent with Regulatory Guide (RG) 1.174. RG 1.174 permits only small increases in risk as long as it is reasonably assured that sufficient defense in depth and margins are maintained.

In our report, dated April 27, 2004, we concluded that the process and criteria in RG 1.174 are appropriate for evaluating the acceptability of changes proposed under a revised rule, but recommended explicit consideration of late release frequency (LRF) in addition to core damage frequency (CDF) and large early release frequency (LERF) to ensure that all of the safety objectives are addressed. The SRM and the proposed rule language posit a process, akin to the current 10 CFR 50.59 process, to permit licensees to make changes that result in "inconsequential" changes in risk without prior NRC review and approval. We agree that a process for making such changes is needed. The staff argues that the existing 10 CFR 50.59 process is not suitable, since it addresses design-basis issues, while the new process must address the acceptability of changes with respect to risk. Additional input on the need for a new change process can be obtained when a draft rule is issued for public comment.

In the proposed rule language, the staff introduces a transition break size (TBS). The TBS is chosen to ensure that the frequency of LOCAs corresponding to breaks larger than the TBS is less than $1x10^{-5}$/reactor-year. This frequency is consistent with the goal set in SECY-00-198, Attachment 1, "Framework for Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50" for rare initiators and the criterion proposed for a vessel failure frequency due to pressurized thermal shock, when it is recognized that those are unmitigated events, and that a substantial mitigative capability will be maintained for LOCAs beyond the TBS.

For LOCAs corresponding to break sizes smaller than the TBS, the requirements are equivalent to those in the current 10 CFR 50.46. We agree that defense in depth should be maintained through the requirement that sufficient mitigating capability be available to prevent severe core damage (i.e., loss of coolable geometry) for breaks greater than the TBS up to the DEGB of largest pipe in the reactor coolant system. Because of the low frequency of such breaks, it should not be necessary to postulate a simultaneous loss of offsite power and single failure of the most critical component. Credit may be taken for operation of any equipment supported by appropriate availability data. Nominal operating conditions rather than technical specification limits, actual fuel burnup in decay heat predictions, and actual operating peaking factors can be used. Some increase in the degree of core damage beyond that implied in the current 10 CFR 50.46 should also be considered acceptable. The integrity of the reactor containment structure should be maintained using realistically calculated pressure, temperature, and containment capacity.

Because breaks with sizes greater than the TBS are not risk significant, and hence equipment needed to mitigate such breaks might be considered unimportant in 10 CFR 50.65(a)(4) assessments of acceptable configurations, the staff has included additional configuration control requirements to ensure the capability to mitigate such large breaks during all modes of operation when the reactor is critical. We agree that such configuration control requirements are appropriate.

The draft version of a proposed rule discussed with us proposes a TBS that is reactor specific and equivalent in area to a double-ended rupture of the largest pipe attached to the reactor coolant system. For a pressurized water reactor (PWR) this would correspond to surge, shutdown cooling, or safety injection lines that are typically 12-14 inches in diameter Schedule 160 pipe. For a boiling water reactor (BWR) these would be residual heat removal or feedwater lines, which are typically 20 inches in diameter Schedule 80 pipe.

The selection of the TBS requires estimates of LOCA frequencies as a function of break size. The most comprehensive assessment of this information is the expert opinion elicitation conducted by the Office of Nuclear Regulatory Research (RES). We believe that additional work needs to be done to complete the expert opinion elicitation and have issued a separate report on this matter, dated December 10, 2004. Hence, some of our judgments below on the implications of the elicitation must be considered preliminary.

The elicitation sought to develop LOCA frequency estimates for PWR and BWR piping and non-piping passive components. It focused on developing average values for the fleet of operating plants, and thus the uncertainty bounds represent bounds on these average values and not on LOCA frequency estimates for individual plants. Thus they are only applicable to plants that can demonstrate that they have no additional degradation mechanisms, no significant differences in the conditions that produce degradation, and no significant differences in their capability to detect degradation than is typical of most plants in the fleet.

The elicitation also did not consider the impact on the frequency of LBLOCAs of the power uprates that could likely result from a risk-informed 10 CFR 50.46. Such uprates could have substantial impacts on flow-assisted corrosion rates in secondary systems in PWRs. PWR power uprates are not likely to have a significant impact on primary system piping. The BWR feedwater piping is susceptible to flow-assisted corrosion. The potential impact of power uprates on LOCA frequency will have to be addressed as part of the licensing reviews of the uprates.

In its efforts to develop a new rule, the staff has considered other potential mechanisms that could cause pipe failure that were not explicitly considered in the expert elicitation process such as active system LOCAs, seismic loading, heavy load drops, and LOCA–induced waterhammer loading. No active system LOCAs were identified that would result in break sizes greater than about 4 inches. The staff concluded that heavy load drops would have little effect on the choice of the TBS. For seismic loads with magnitudes of occurrence of 1 $x10^{-5}$/yr, the staff has found that undegraded piping or piping with minor degradation has little likelihood of failure. More severely degraded piping could fail under such seismic loads, but the relatively low frequency of degradation in primary piping and the low frequency of the expected loading suggest that these will not have a significant impact on the choice of the TBS. RES is still performing some confirmatory research in this area.

Thus it appears that the expert elicitation has addressed the important potential contributors to the LBLOCA frequency. However, the choice of a TBS is strongly dependent on how the uncertainties in the elicitation are addressed.

For PWRs the break size (i.e., the equivalent diameter of the flow area) corresponding to a frequency of 1 x 10^{-5}/reactor-year from the expert opinion elicitation reported in Reference 3 ranges from 4-11 inches depending on the approaches used to aggregate and assess the expert opinions, whether the mean or 95th percentiles of the resulting distributions are used, and how the results are interpolated between the discrete break sizes in the elicitation.

The staff's choice of a break size corresponding to a double-ended break of the largest piping attached to the reactor coolant system appears to conservatively bound the range of values determined through the elicitation. The large disparity in size between the main reactor coolant system piping and the largest attached piping also provides an argument for the selection of the failure of the attached piping as the TBS. Although uncertainties in the elicitation could affect the choice of the TBS in the range of sizes up to the diameter of the attached piping, the physics of the failure processes give a very-high confidence in the low-failure probability of the main coolant piping. The staff notes that this choice for the TBS makes it very unlikely that any future reevaluations of the break frequency versus break size will result in the need for licensees to make any plant modifications as a result of implementing the revised 10 CFR 50.46 thus helping to ensure a more stable regulatory environment. It also bounds the flow areas associated with breaks of components such as bolted connections. Although these connections were considered in the elicitation, they are more likely to be affected by human errors and are thus perhaps subject to even greater uncertainty than the piping failure.

Based on our current understanding of the results of the expert opinion elicitation, it appears that the choice of the double ended rupture is overly conservative. Choosing the TBS as the diameter of the largest attached pipe (i.e., a single-ended rupture) would still bound the elicitation results and would be consistent with the argument that the failure of the main coolant piping is much more unlikely than the failure of the smaller attached piping. If the defense-in-depth capability to mitigate breaks greater than the TBS is maintained, a less conservative choice of TBS (e.g., one based on the mean value of the final "best estimate" distribution from the elicitation) may also be supportable.

A better quantitative understanding of the impact of the TBS on parameters, such as required diesel start time, is needed to help optimize the choice of a TBS to balance the defense in depth provided by the larger TBS in any new draft rule with the possible risk benefits of smaller break sizes. Since much of this may be plant specific and will require detailed plant information, it may have to be sought when a draft rule is issued for public comment. Any discussion of risk benefits should also include consideration of the impact of power uprates, which are the likely consequence of a risk-informed 10 CFR 50.46, on such risk benefits.

We would like to review any new draft rule before it is issued for public comment.

Sincerely,

Mario V. Bonaca

Mario V. Bonaca
Chairman

References:

1. Memorandum dated December 2, 2004, from Catherine Haney, Program Director, Policy and Rulemaking Program, NRR, to various members NRR, Subject: Office Concurrence on Proposed Rule - Risk Informed Changes to Loss-of-Coolant Accident Technical Requirements (Pre-Decisional For Internal ACRS Use Only).
2. Memorandum dated October 14, 2004, from Catherine Haney, Program Director, Policy and Rulemaking Program, NRR, to John T. Larkins, Executive Director, ACRS, Subject: Review of Risk-Informed 10 CFR 50.46 Proposed Rule Executive Summary and Draft Proposed Rule Language (Pre-Decisional For Internal ACRS Use Only).
3. Staff Requirement Memorandum dated July 1, 2004, from Annette L. Vietti-Cook, SECY, NRC to Luis A. Reyes, EDO, NRC Subject: SECY-04-0037 - Issues Related to Proposed Rulemaking to Risk-Inform Requirements Related to Large Break Loss-of-Coolant Accident (LOCA) Break Size and Plans for Rulemaking on LOCA with Coincident Loss-of-Offsite Power.
4. Regulatory Guide 1.174, entitled, "An Approach for using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Rev. 1, Office of Nuclear Regulatory Research, November 2002.
5. ACRS Report dated April 27, 2004, from Mario V. Bonaca, Chairman, ACRS to Nils J. Diaz, Chairman, NRC, Subject: SECY-04-0037, "Issues Related to Proposed Rulemaking to Risk-Inform Requirements Related to Large Break Loss-of-Coolant Accident (LOCA) Break Size and Plans for Rulemaking on LOCA with Coincident Loss-of-Offsite Power."
6. 10 CFR 50.59, "Changes, tests, and experiments."
7. 10 CFR 50.65(a)(4), "Requirements for monitoring the effectiveness of maintenance at nuclear power plants."
8. ACRS Report dated December 10, 2004, from Mario V. Bonaca, Chairman, ACRS to Nils J. Diaz, Chairman, NRC, Subject: Estimating Loss-of-Coolant Accident Frequencies through the Elicitation Process.
9. Memorandum dated November 4, 2004, from Michael E. Mayfield, Director, Division of Engineering Technology, RES, to John T. Larkins, Executive Director, ACRS, Subject: Transmittal of Draft NUREG on Passive System LOCA Frequency Development for use in Risk-Informed Revision of 10 CFR 50.46, Appendix K to Part 50, and GDC and Appendices (Pre-Decisional For Internal ACRS Use Only).

NRC FORM 335 (9-2004) NRCMD 3.7

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

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

NUREG-1125, Volume 26

2. TITLE AND SUBTITLE

A Compilation of Reports of the Advisory Committee on Reactor Safeguards: 2004 Annual

3. DATE REPORT PUBLISHED

MONTH	YEAR
June	2005

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

6. TYPE OF REPORT

Compilation

7. PERIOD COVERED *(Inclusive Dates)*

Jan. thru Dec. 2004

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

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

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Same as above

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

This compilation contains 47 Advisory Committee on Reactor Safeguards reports submitted to the U. S. Nuclear Regulatory Commission (NRC), or to the NRC Executive Director for Operations, during calendar year 2004. In addition, a report to the Commission on the NRC Safety Research Program, NUREG-1635, Volume 6, is included by reference only. All reports have been made available to the public through the NRC Public Document Room, the U. S. Library of Congress, and the Internet at http://www.nrc.gov/reading-rm/doc-collections. The reports are organized in chronological order.

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

Core Power Uprate
Digital Instrumentation and Control
Human Factors
License Renewal
Nuclear Reactors
Nuclear Reactor Safety
PWR Sump Performance
Reactor Operations
Risk-Informed
Safety Engineering
Safety Research

13. AVAILABILITY STATEMENT

Unlimited

14. SECURITY CLASSIFICATION

(This Page) Unclassified

(This Report) Unclassified

15. NUMBER OF PAGES

16. PRICE

Printed on recycled paper

Federal Recycling Program

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, DC 20555-0001

OFFICIAL BUSINESS