COMMISSION BRIEFING SLIDES/EXHIBITS

MEETING WITH ACRS

OCTOBER 2, 2003

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

September 24, 2003

MEMORANDUM TO: Annette L. Vietti-Cook
Secretary of the Commission

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

SUBJECT: ADVISORY COMMITTEE ON REACTOR SAFEGUARDS MEETING WITH THE U. S. NUCLEAR REGULATORY COMMISSION, OCTOBER 2, 2003 -- SCHEDULE AND BACKGROUND INFORMATION

The ACRS is scheduled to meet with the NRC Commissioners between 9:30 a.m. and 11:30 a.m. on Thursday, October 2, 2003, to discuss items listed below. Background materials related to these items are enclosed.

	ESTIMATED TIME*
INTRODUCTION - NRC Chairman Nils J. Diaz	5 minutes

ACRS PRESENTATIONS

1. Overview - Mario Bonaca - ACRS Chairman — 15 mins.
   - License Renewal
   - Risk-Informing 10 CFR 50.46
   - Proposed 10 CFR 50.69
   - AP1000 Design Certification Review
   - GE/ESBWR Preapplication Review
   - Power Uprate Review Standard
   - Future ACRS Activities

2. Materials Degradation Issues - John Sieber — 15 mins.
   - Current Issues
   - Industry Response
   - Regulatory Response
   - ACRS Activities
   - ACRS Conclusions and Recommendations
   - Proactive Life Management of Materials Degradation Issues

3. Reactor Oversight Process - William Shack — 15 mins.
   - ROP Improvements Over SALP
   - ACRS Letters
   - Is the ROP effective?
   - Remaining ACRS Issues

4. Improvement of the Qualilty of Risk Information for Regulatory Decisionmaking - Thomas Kress — 10 mins.
    - May 16, 2003 Report

CLOSING REMARKS - NRC Chairman Nils J. Diaz

*NOTE: Estimated times are for presentation only and do not include the time set aside for Commission questions and answers.

1

OVERVIEW

Mario V. Bonaca
ACRS Chairman

LICENSE RENEWAL

- **Reviewed three applications since April 2003**
- **Will review another three applications between October and December 2003**
- **Responded to SRM on improvements to generic license renewal guidance documents – June 2003**

LICENSE RENEWAL (Cont'd)

- **Will review five applications in CY 2004**

- **Exploring means to further streamline the ACRS review of license renewal applications**

RISK-INFORMING 10 CFR 50.46

- **ACRS was briefed on the Use of Expert Elicitation Process to Develop LOCA Frequencies**

- **Will review results of elicitation in Fall 2003**

RISK-INFORMING 10 CFR 50.46 (Cont'd)

- Will review proposed rulemaking in response to March 31, 2003 SRM prior to it being forwarded to the Commission

- Will work with the Staff to reconcile challenging technical issues

PROPOSED 10 CFR 50.69

- Provided comments and recommendations on the proposed 10 CFR 50.69,

  March 19, 2002
- Discussed with Commission on July 10, 2002

PROPOSED 10 CFR 50.69 (Cont'd)

- **Meeting planned in Fall 2003 to discuss:**
  - **Draft Final 10 CFR 50.69**
  - **Staff's resolution of public comments**
  - **Staff's resolution of ACRS comments and recommendations**
  - **NEI's implementation guidance and staff's endorsement**

AP1000 DESIGN CERTIFICATION REVIEW

- **Held four Subcommittee meetings and one Full Committee meeting to discuss AP1000 design aspects, PRA, Thermal-Hydraulic issues, and DSER open items**

AP1000 (Cont'd)

- Reliability of ADS-4 Squib Valve still a question
- Significant number of open items remain to be resolved
- ACRS Full Committee meeting 10/2003 to discuss status of resolution of open items
- ACRS Full Committee Review of FSER—7/2004

GE/ESBWR PRE-APPLICATION REVIEW

- ESBWR Design is based on GE/SBWR and ABWR Designs with Passive Decay Heat Removal system
- Elimination of recirculation pumps, jet pumps, and associated valves and piping reduces the number of locations where reactor coolant system leakage could potentially occur

GE/ESBWR (Cont'd)

- **Held one Subcommittee meeting and one full Committee meeting to review thermal-hydraulic issues and design aspects**
- **Will continue to review ESBWR design aspects and associated Staff review efforts**

POWER UPRATE REVIEW STANDARD

- **Reviewed draft final extended power uprate review standard – ACRS Report dated September 24, 2003**
- **Expect to review up to seven power uprate applications in CY 2004**
- **Revise ACRS review criteria for Power Uprate Applications (>5%) after Staff's implementation of review standard**

FUTURE ACRS ACTIVITIES

- Risk-Informed and Performance-Based Regulation
- Advanced Reactor Designs
  - Design Certification Of AP1000
  - Pre-application Reviews
  - Early Site Permit Process/Applications
- Thermal-Hydraulic Codes

FUTURE ACRS ACTIVITIES (Cont'd)

- **Materials Degradation Program**
- **Steam Generator Action Plan**
- **Mixed Oxide Fuel Fabrication Facility**
- **License Renewal and Core Power Uprate Applications**

FUTURE ACRS ACTIVITIES (Cont'd)

- High-Burnup Fuel Issues
- Safeguards and Security Matters
- Resolution of Generic Safety Issues
- Significant Reactor Operating Events
- Safety Research Program
- Fire Protection Matters

ACRONYMS

- ABWR – Advanced Boiling Water Reactor
- ADS-4 – Automatic Depressurization System
- DSER – Draft Safety Evaluation Report
- ESBWR – Economically Simplified Boiling Water Reactor
- FSER – Final Safety Evaluation Report
- GE – General Electric

ACRONYMS (cont'd)

- LOCA – Loss of Coolant Accident
- NEI – Nuclear Energy Institute
- PRA – Probabilistic Risk Assessment
- SBWR – Simplified Boiling Water Reactor
- SRM – Staff Requirements Memorandum

ACRS LETTERS

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

June 24, 2003

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

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

Dear Chairman Diaz:

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

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

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

RECOMMENDATIONS

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

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

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

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

DISCUSSION

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

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

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

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

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

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

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

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

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

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

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

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

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

Sincerely,

Mario V. Bonaca

Mario V. Bonaca
Chairman

References:

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

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

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

UNITED STATES
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WASHINGTON, D.C. 20555-0001

September 17, 2003

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

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE LICENSE RENEWAL APPLICATION FOR THE ST. LUCIE NUCLEAR PLANT UNITS 1 AND 2

Dear Chairman Diaz:

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

CONCLUSION AND RECOMMENDATION

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

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

BACKGROUND AND DISCUSSION

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

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

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

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

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

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

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

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

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

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

Sincerely,

Mario V. Bonaca

Mario V. Bonaca
Chairman

References:

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

UNITED STATES
NUCLEAR REGULATORY COMMISSION
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WASHINGTON, D.C. 20555-0001

September 24, 2003

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

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

Dear Chairman Diaz:

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

RECOMMENDATION AND CONCLUSION

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

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

DISCUSSION

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

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

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

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

Sincerely,

Mario V. Bonaca

Mario V. Bonaca
Chairman

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

2

Materials Degradation Issues

John Sieber

Current Issues

- Cracking of PWR VHP nozzles.
- Boric acid wastage of the RPV head ferritic base metal
- RPV lower head penetration leakage
- Other Alloy 600 applications and weldments

Industry Response

Industry tasks for EPRI

- Cracking susceptibility algorithm
- Inspection protocol and techniques
- Industry inspection database
- Prediction methodology for VHP boric acid corrosion

Regulatory Response

- Bulletins 2001-01,2002-01,2002-02 and 2003-02
- Order EA-03-009
- Information Notice 2003-11
- Davis Besse Lessons Learned Task Force Action Plan

ACRS Activities

- The staff has kept the ACRS regularly informed of the progress of industry and staff work
- ACRS issued letters on July 23, 2001, June 20, 2002 and May 16, 2003

ACRS Conclusions and Recommendations

- Sound technical basis for VHP degradation plan
- Action plan needs to be augmented
- Develop capability to predict RPV lower head penetration cracking

ACRS Conclusions and Recommendations (cont'd)

- Augment current flaw evaluation guidelines
- Qualify inspection methods
- Manage other degradation modes
- NRC/Industry collaboration is needed

Proactive Life Management of Materials Degradation Issues

- Roles of utility, reactor designer, and NRC
- Requires adequate knowledge of chemistry, materials, and mechanical aspects
- Balance between degradation prediction and inspection capabilities

Proactive Life Management of Materials Degradation Issues (cont'd)

- **Concept of "Proactive Materials Degradation Assessment" plan seems appropriate**

- **Will review industry and NRC plans**

ACRONYMS

- EPRI – Electric Power Research Institute
- PWR – Pressurized Water Reactor
- RPV – Reactor Pressure Vessel
- VHP – Vessel Head Penetration

ACRS LETTERS

UNITED STATES
NUCLEAR REGULATORY COMMISSION
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WASHINGTON, D.C. 20555-0001

May 16, 2003

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

SUBJECT: VESSEL HEAD PENETRATION CRACKING AND REACTOR PRESSURE VESSEL DEGRADATION

Dear Chairman Diaz:

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

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

CONCLUSIONS AND RECOMMENDATIONS

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

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

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

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

DISCUSSION

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

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

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

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

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

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

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

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

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

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

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

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

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

Sincerely,

Mario V. Bonaca

Mario V. Bonaca
Chairman

Additional Comments by ACRS Members Dana A. Powers and Thomas S. Kress

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

References:

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

ACRSR-2001

June 20, 2002

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

SUBJECT: VESSEL HEAD PENETRATIONS AND VESSEL HEAD DEGRADATION

Dear Chairman Meserve:

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

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

CONCLUSIONS AND RECOMMENDATIONS

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

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

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

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

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

Background

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

Discussion

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

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

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

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

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

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

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

Sincerely,

/RA/

George E. Apostolakis
Chairman

References

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

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

July 23, 2001

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

SUBJECT: CIRCUMFERENTIAL CRACKING OF PWR VESSEL HEAD PENETRATIONS

Dear Chairman Meserve:

During the 484th meeting of the Advisory Committee on Reactor Safeguards, July 11-13, 2001, we heard presentations by and held discussions with representatives of the NRC staff and the Electric Power Research Institute (EPRI) Materials Reliability Program regarding industry and staff actions relative to cracking and leaking observed in pressurized water reactor (PWR) Alloy 600 reactor vessel head penetrations, including control rod drive mechanism (CRDM) nozzles. This matter was also discussed during a July 10, 2001, meeting of the Materials and Metallurgy and the Plant Operations Subcommittees. During our reviews, we had the benefit of the documents referenced.

Conclusions and Recommendations

1. The decision to issue a bulletin addressing the recent incidents of circumferential cracking of CRDM nozzles in U.S. PWRs is timely and appropriate.

2. The staff should urgently address technical issues associated with risk assessment, the effectiveness of inspection techniques, and the completeness of damage accumulation prediction.

Discussion

Cracks were recently detected during inspections of CRDM nozzles at Oconee Units 1, 2, and 3 and Arkansas Nuclear One (ANO) Unit 1. Preliminary risk assessment indicates that the issuance of a bulletin is appropriate to request operational information from the licensees as soon as possible.

The staff's in-depth analysis has raised a number of technical concerns. Although plans are in place to resolve them, the following concerns are of particular importance:

- Risk Assessment

The risk assessment activities should be expanded to include rod ejection with coincident small-break loss of coolant accident and potential damage to adjacent control rods.

- Prioritization of Inspection Schedules

Inspection schedule prioritization during the upcoming refueling outages will be based on an analysis of the susceptibility of cracking of CRDM nozzles in different plants. This approach relies on the assumption that susceptibility is determined by time of service and vessel head temperature. This has led to the grouping of each PWR into one of four "bins." The 14 reactors in the two highest susceptibility bins should receive highest priority in inspections of all CRDM nozzles in 2001. Although this approach is reasonable from a technical standpoint at present, its accuracy will become apparent as inspections proceed. It is prudent to consider potential modifications to this methodology including the following:

(a) The cracking susceptibility will depend on other conjoint plant-specific factors that can affect cracking and that are not considered explicitly in the current susceptibility algorithm, which addresses only vessel head temperature and operating time. These further factors include residual stress, material composition, heat treatment, welding practices, and local chemical environment.

(b) As more information on the cracking of CRDM nozzles accumulates from the upcoming U.S. inspections and from past observations overseas, the basis for a risk-informed methodology may be formulated.

The staff should be prepared to modify any proposed inspection program and timing depending on the results of inspections of the first group of plants (i.e., Fall 2001). These early inspection results may show that it is imperative to inspect the vessel heads of the remaining pressurized water reactors promptly. On the other hand, they may show that it is appropriate to delay the inspections of the remaining plants to allow improvements in diagnostic capabilities.

- Inspection Methods

The current visual inspection process, which relies on detecting boron crystals at the top of the annulus, indicates the possible presence of circumferential cracks at the base of the annulus, but gives no information on the size and/or orientation of these cracks in the Alloy 600 material. In addition, the absence of visible boron crystals does not give complete assurance that a concentrated chemical environment at the annulus does not exist, resulting in the rapid growth of a circumferential crack. This concern could be addressed during the fall outage by a full volumetric inspection of all CRDM nozzles (i.e., including those with no boron crystals) at Oconee Units 1, 2, and 3, and ANO Unit 1. Volumetric inspections by a qualified process in such cases makes abundant sense. Assessment of the inspection methods used to detect and size cracks in CRDM nozzles and nozzle welds is necessary, especially for the circumferential cracks initiating at the base of the annulus between the CRDM nozzles and the pressure vessel head.

- <u>Inspection Periodicity</u>

  The inspection intervals once cracks are detected depend on knowledge of crack propagation rates as a function of the local material, environmental, and stress conditions. There are data for Alloy 600 cracking as a function of stress intensity and the temperature of the PWR primary coolant. Also, there are limited data relevant to the axial cracking in the Inconel 182 J-weld connecting the CRDM nozzle to the vessel head. The quality of these data is being evaluated by separate expert committees convened by industry and the staff. There is no similar data set relevant to the circumferential cracks that initiate in and adjacent to the J-weld and that present the greatest potential structural integrity concern. The reason for this lack of cracking data is that the local environment in the annulus between the pressure vessel and the CRDM nozzle is not known with sufficient certainty. This problem is also being addressed by the staff.

Consideration of the above issues in conjunction with the issuance of the bulletin should ensure that this matter is satisfactorily addressed for the short term. The Committee wishes to be updated once the licensee responses to the bulletin are evaluated.

A crucial issue confronted in the proposed bulletin is the urgency of inspections of vessel head penetrations, especially for plants thought to be less susceptible to CRDM stress corrosion cracking. Risk would be the metric best suited for determining the urgency. Unfortunately, neither the NRC's phenomenological capabilities, such as the ability to predict time-dependent stress corrosion cracking, nor the NRC's risk assessment capabilities are sufficiently developed at this time to provide defensible bases for decisions on the urgency of vessel head inspections. Sustained research to better the agency's integrated capabilities in probabilistic fracture mechanics and risk assessment will be needed to assist NRC in confronting future issues of reactor coolant system degradation.

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

Sincerely,

George E. Apostolakis

George E. Apostolakis
Chairman

References:

1. Letter dated June 29, 2001, from A. Marion, Nuclear Energy Institute, to Brian W. Sheron, Office of Nuclear Reactor Regulation, NRC, Subject: Response to June 22, 2001, letter from Dr. Brian Sheron (NRC) to Mr. Alex Marion (NEI) transmitting NRC staff questions on EPRI Interim Report TP-1001491, Part 2 (Proprietary).
2. U. S. Nuclear Regulatory Commission Proposed Bulletin 2001-XX, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," dated June 25, 2001.
3. Memorandum dated June 21, 2001, from C. E. Carpenter, Office of Nuclear Reactor Regulation, NRC, to W. Bateman, Office of Nuclear Reactor Regulation, NRC, Subject: Summary of June 7, 2001, Meeting with the EPRI Materials Reliability Program on Generic Activities Related to CRDM Cracking.
4. U. S. Nuclear Regulatory Commission Information Notice 2001-05: "Through-Wall Circumferential Cracking of Reactor Pressure Vessel Head Control Rod Drive Mechanism Penetration Nozzles at Oconee Nuclear Station, Unit 3," dated April 30, 2001.
5. Electric Power Research Institute, TP-1001491, Part 2, "PWR Materials Reliability Program, Interim Alloy 600 Safety Assessments for US PWR Plants (MRP-44)," Interim Report, May 2001.
6. Letter dated April 17, 2001, from Brian W. Sheron, Office of Nuclear Reactor Regulation, NRC, to Alex Marion, Nuclear Energy Institute, Subject: Issues to be Addressed in a Generic Justification for Continued Operation of PWRs.
7. U. S. Nuclear Regulatory Commission, NUREG/CR-6245, "Assessment of Pressurized Water Reactor Control Rod Drive Mechanism Nozzle Cracking," October 1994.
8. Letter dated June 1, 2001, from Alex Marion, Nuclear Energy Institute, to Brian Sheron, Office of Nuclear Reactor Regulation, NRC, regarding NRC's Assessment of Topical Report MRP-44 - Summary of NRC/NEI Telecon of May 30, 2001.
9. Letter dated May 18, 2001, from Alexander Marion, Nuclear Energy Institute, to Brian Sheron, Office of Nuclear Reactor Regulation, NRC, Subject: PWR Reactor Pressure Vessel Head Penetrations, dated, May 18, 2001.
10. Letter dated December 11, 1998, from David Modeen, Nuclear Energy Institute, Subject: Responses to NRC Requests for Additional Information on Generic Letter 97-01.
11. U. S. Nuclear Regulatory Commission Generic Letter 97-01, Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations, dated April 1, 1997.
12. Letter dated November 19, 1993, from William Russell, Office of Nuclear Reactor Regulation, NRC, to W. Rasin, Nuclear Utility Management and Resources Council (now NEI), transmitting the Safety Evaluation for Potential Reactor Vessel Head Adaptor Tube Cracking.
13. Generic Letter 88-05, Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants, March 1998.

3

Reactor Oversight Process

William Shack

ROP Improvements Over SALP

- Performance based and risk informed
- Trivial issues eliminated
- More objective and systematic
- Easier public understanding
- Better management tool
- Stakeholder support

ACRS Letters

- March 15, 2000
- October 12, 2001
- February 13, 2002
- March 13, 2003

Is the ROP effective?

- Yes. Provides more objective assessment; accepted by stakeholders; excellent public outreach
- No. Insufficient emphasis on cross-cutting issues; disproportionate assessment of performance between cornerstones

Remaining ACRS Issues

- Risk arguments for PI thresholds are incorrect
- PI thresholds (colors) should be performance based, not risk based
- PIs are needed for cross-cutting issues
- Parity in the Action Matrix between PIs and SDP results needs improvement

ACRONYMS

- PI – Performance Indicators
- ROP – Reactor Oversight Process
- SALP – Systematic Assessment of Licensee Performance
- SDP –Significance Determination Process

ACRS LETTERS

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

March 13, 2003

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

SUBJECT: REACTOR OVERSIGHT PROCESS

Dear Chairman Meserve:

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

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

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

The ACRS views on the ROP are as follows:

1. The purpose of the ROP is to assess safety performance so that the agency can take appropriate action.
2. The ROP is risk-informed because it focuses on performance areas and indicators that affect safety.
3. It is incorrect to base thresholds for PIs on risk metrics such as ΔCDF (changes in core damage frequency) and ΔLERF (changes in large, early release frequency).
4. The thresholds separating all the performance levels (colors) should be performance-based and determined by expert judgement similar to the selection of the current green/white thresholds.
5. The principal role for the SDP is to assign risk characterization to inspection findings – not to be an evaluation of performance.

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

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

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

DISCUSSION

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

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

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

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

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

Sincerely,

Mario V. Bonaca

Mario V. Bonaca
Chairman

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

February 13, 2002

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

SUBJECT: THE REVISED REACTOR OVERSIGHT PROCESS

Dear Dr. Travers:

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

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

Sincerely,

George E. Apostolakis
Chairman

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

October 12, 2001

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

SUBJECT: THE REVISED REACTOR OVERSIGHT PROCESS

Dear Chairman Meserve:

During our 485th meeting on September 5-7, 2001, the Advisory Committee on Reactor Safeguards met with representatives of the NRC staff to discuss the revised Reactor Oversight Process (ROP). We continued our deliberations during our 486th meeting on October 4-6, 2001. This matter was also discussed during meetings of the ACRS Plant Operations Subcommittee on December 6, 2000, May 9, 2001, and July 9, 2001. In addition, the ACRS Subcommittees on Plant Operations and Fire Protection held meetings with licensees on June 13, 2000, and June 27, 2001, and held meetings with Regions III and IV on June 14, 2000, and June 28, 2001, respectively. During our review, we had the benefit of the documents referenced.

BACKGROUND

The ROP utilizes the results of performance indicators (PIs) and baseline inspection findings to determine the appropriate regulatory action to be taken in response to a licensee's performance. The escalation of the regulatory responses is specified in the action matrix, which the staff developed as part of the ROP. This ROP has been in effect for nearly all licensees for about one year. The staff has conducted an assessment of the state of the ROP and recognizes that it is still a process in development.

The ACRS has previously commented on various aspects of the ROP and provided recommendations to the staff regarding potential process improvements. We remain convinced that the ROP is more objective and understandable than the former oversight process and represents a significant improvement. This report discusses some specific questions that the Commission raised to the ACRS, and offers some additional thoughts on potential improvements in the ROP.

In the Staff Requirements Memorandum dated April 5, 2000, the Commission requested the ACRS to:

(1) Review the use of PIs in the ROP to ensure that the PIs provide meaningful insight into aspects of plant operation that are important to safety.

(2) Review the initial implementation of the significance determination processes (SDPs), and assess the technical adequacy of the SDP to contribute to the ROP.

The current PIs do provide meaningful insight into plant performance. However, there is a need to redefine the thresholds for some of the PIs to provide better input to the ROP. In particular, the numerical values for the white/yellow and yellow/red thresholds for the initiating event and mitigation system PIs are not useful and should be revised. The color bands for the PIs and SDPs associated with all the cornerstones have similar implications with respect to agency action and, therefore, the thresholds should be commensurate with their respective safety significance.

The most immediate and pressing need for the ROP is to improve the SDP tools. Some SDPs are incomplete and, in cases such as fire protection, overly subjective. The technical adequacy of the risk-based SDPs depends on the availability and quality of a relevant probabilistic risk assessment (PRA). Thus, the SDP for at-power situations provides meaningful risk information. For routine findings that are predominantly of very low, low, and moderate safety significance, the process is probably adequate. The threshold values for the risk-based SDPs are appropriate.

We continue to believe that a documented review of the SDP worksheets and SPAR models (as well as the underlying SAPHIRE computer code) is essential to public confidence in the ROP.

An SDP based on low-power and shutdown PRAs or other shutdown management tools is needed to characterize findings during these modes of operation. In addition, the fire protection SDP involves very qualitative inputs to a quantification process of uncertain pedigree. This SDP is probably useful for its intended purpose, however, it may be hard to defend and justify to the public. Even though this SDP calculates the change in core damage frequency (CDF), the SDP is really intended to provide an indication of the degradation of defense in depth for fire protection as defined in 10 CFR Part 50, Appendix R.

Presently, concurrent performance deficiencies are assessed collectively, as applicable, to determine the total change in CDF, but each performance deficiency is assigned a color individually. There may be instances in which conclusions could be altered if the results are considered collectively, and thus such collective results should be considered in the action matrix.

DISCUSSION

An important premise of the ROP is that there should be a graded regulatory response to inspection findings and PI results. Although a graded response to oversight findings is a desirable attribute, the inputs to the action matrix that implements this response must be produced in a way that justifies the resulting response. This is especially true for the right-hand columns of the matrix which could lead to severe regulatory responses.

The current ROP uses different technical bases to establish the thresholds for the PIs and inspection findings. In particular:

- On the basis of its review of recent operating history, the staff set the green/white thresholds for the PIs for initiating events and mitigating systems at the 95th percentile of peer performance for the given PI. By contrast, the staff based the white/yellow and yellow/red thresholds on an assessment of the value of a PI corresponding to increases in CDF of 10^{-5} and 10^{-4} per reactor year, respectively.

- The staff set the PI thresholds for barriers, emergency preparedness, occupational radiation safety, public radiation safety, and physical protection by considering technical specification limits, the number of noncompliances with regulatory requirements, and other absolute measures.

- The staff based the green/white, white/yellow, and yellow/red thresholds for SDP results on increases in CDF of 10^{-6}, 10^{-5}, and 10^{-4} per reactor year, respectively. This is true for the initiating event, mitigating system, and fire protection cornerstones. The other SDPs do not have a PRA basis and take a deterministic and defense-in-depth approach to establish thresholds for safety significant issues.

These different bases for defining the various thresholds raise questions regarding the kinds of information that the PIs and SDPs provide and the consistency of the meaning of the thresholds across the PIs and SDPs. These different thresholds are based on expert judgment that the degradation in performance associated with each color band is appropriately linked to a corresponding regulatory response[1].

It is from this viewpoint that we believe it is necessary to reconsider the definitions of the white/yellow and yellow/red thresholds for initiating events and mitigating systems, which as we noted above were based on an attempt to assess the value of a PI corresponding to increases in CDF.

We have noted previously that it is difficult to generically assess the risk impact of changes in a PI. The associated changes in risk tend to depend strongly on plant-specific features. This approach, however, has a deeper, more intractable flaw. Specifically, it focuses on the change in CDF that results from changes in a single, isolated parameter assuming that all other factors that can affect CDF remain constant. A realistic assessment of the change in CDF cannot be related to the change in a single PI. Thus, in some cases, the use of this approach to select white/yellow and yellow/red thresholds has led to values for these thresholds that, in our judgment and that of many of the staff and the industry, are too high to be meaningful. Regulatory attention would increase at much lower levels.

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[1] The color bands for the ROP are called "constructed scales" in decision analysis. Ensuring the consistency of the bands of these scales is what decision analysts commonly call "performing sanity checks," and such checks are among the most important steps in a decisionmaking process. In our report on the NRC Safety Research Program (NUREG-1635, Vol. 4), we recommended that the staff initiate a program of research to investigate how best to use formal decisionmaking methods in regulatory decisions.

The white/yellow and yellow/red thresholds for the PIs for initiating events and mitigating systems should be set in terms of an expert judgment of what values should in fact trigger the regulatory response associated with the threshold. Although general considerations for the selection of thresholds for PIs and SDPs are discussed in SECY-99-007, the expert judgment process that the staff used to develop the initial values for the thresholds for the non risk-based PIs and SDPs and the corresponding equivalency of the combination of findings in the action matrix have not been well documented. The NRC has been a pioneer in the use of scrutable expert judgment processes, and it is unfortunate that the use of expert judgment in a process as central to the NRC's mission as the ROP lacks the traceability of other NRC uses of expert judgment. Formal decision analysis could be helpful in making the selection of thresholds and the action matrix more objective and scrutable.

In assessing the need to revise the current PIs and develop new PIs, we believe that the staff responsible for the ROP should consider the work being done in other parts of the agency. For example, the review of operating experience for the reactor core isolation cooling (RCIC) system for BWRs (NUREG/CR-5500, Vol. 7) shows that the dominant failure modes involve system failures while running and human failures to recover the system (i.e., failures that are not part of the unavailability calculations that the ROP requires). In analyzing the operating experience, the analysts distinguished between two contexts of RCIC system operation: (1) short-term missions (less than 15 minutes), in which the system must inject water into the reactor vessel following a scram with feedwater available and the main isolation valves open, and (2) long-term missions, in which the system must inject water into the reactor vessel following a scram with feedwater unavailable and/or the reactor vessel isolated. The average system unreliability in these two contexts differs by a factor of 2. The ROP green/white threshold for RCIC system unavailability is 0.04 and makes no distinction between the two contexts identified in the study driven by operating experience. Since unreliability is a metric that includes all potential failure modes, it should be included in the PIs.

We continue to believe that it is important that there be consistency in the definition of terms like "unavailability" which are used in the PIs. Inconsistencies in technical terms that the agency uses in several major activities make comparisons and communication, both internally and externally, difficult.

The ROP is an evolving process. The staff has done an excellent job establishing the basic framework in a relatively short period of time considering the scope of this project. We look forward to continued interactions with the staff on this very important matter.

Additional comments by ACRS Members George E. Apostolakis, Thomas S. Kress, and Steven L. Rosen are presented below.

Sincerely,

George E. Apostolakis
Chairman

<u>References</u>:

1. Staff Requirements Memorandum dated April 5, 2000, from Annette L. Vietti-Cook, Secretary, NRC, to Dr. John T. Larkins, Executive Director, ACRS, Subject: Staff Requirements - Meeting on March 2, 2000, with ACRS on Risk Informing 10 CFR Part 50.
2. Letter dated March 15, 2000, from Dana A. Powers, ACRS Chairman, to Richard A. Meserve, Chairman, NRC, Subject: Revised Reactor Oversight Process.
3. NRC Inspection Manual, Manual Chapter 0609, Appendix A, Significance Determination of Reactor Inspection Findings for At-Power Situations, February 5, 2001.
4. NUREG/CR-5500, Vol. 7, Reliability Study: Reactor Core Isolation Cooling System, 1987 - 1993, Idaho National Engineering and Environmental Laboratory, September 1999.
5. U. S. Nuclear Regulatory Commission, SECY-99-007, Recommendations for Reactor Oversight Process Improvements, January 8, 1999.
6. U. S. Nuclear Regulatory Commission, SECY-99-007A, Recommendations for Reactor Oversight Process Improvements (Follow-up to SECY 99-007), March 22, 1999.
7. U. S. Nuclear Regulatory Commission, SECY 01-0114, Results of the Initial Implementation of the New Reactor Oversight Process, June 25, 2001.
8. U. S. Nuclear Regulatory Commission, SECY 00-0049, Results of the Revised Reactor Oversight Process Pilot Program, February 24, 2000.
9. U. S. Nuclear Regulatory Commission Inspection Manual, Manual Chapter 0305, Operating Reactor Assessment Program, March 23, 2001.
10. NRC Inspection Manual, Manual Chapter 0609, Significance Determination Process, February 27, 2001.
11. Advisory Committee on Reactor Safeguards, NUREG-1635, Vol 4, Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program, A Report to the U.S. Nuclear Regulatory Commission, May 2001.

ADDITIONAL COMMENTS BY ACRS MEMBERS GEORGE E. APOSTOLAKIS, THOMAS S. KRESS, AND STEPHEN ROSEN

We agree with the recommendations and comments of our colleagues. The intent of our comments is to elaborate on the expert judgment process.

In any decisionmaking situation, the most important requirement is that the decisionmaker's judgments be consistent. This is particularly important for the ROP because the bases for the inputs to the action matrix are different.

One of the columns of the action matrix treats two white inputs and one yellow input (for one degraded cornerstone) as being equivalent. This means that the staff's judgment is that two white inputs signify a certain degradation in performance which is about the same as that corresponding to one yellow finding in the sense that the resulting regulatory response should be the same. For consistency in defining these color bands, one would have to address questions such as the following:

- Does the yellow band for the initiating event PI indicate a degradation in performance that is similar to that indicated by the yellow band for a mitigating system PI?
- Is the yellow band of a PI twice as important as its white band?
- Is a yellow finding from an SDP of equal significance as a finding that a PI is in its yellow band?

We appreciate that judgments such as "of equal significance" and "twice as important" are subjective. Our argument is that attempting to answer questions such as these removes a good deal of the subjectivity and, in fact, will be very helpful when the thresholds are determined. This argument acquires additional significance in the present case in which the action matrix does not represent the judgments of a single individual but those of the agency. In other words, communication among the experts who make these judgments would be enhanced.

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

March 15, 2000

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

Dear Chairman Meserve:

SUBJECT: REVISED REACTOR OVERSIGHT PROCESS

During the 469th and 470th meetings of the Advisory Committee on Reactor Safeguards, February 3-5 and March 1-4, 2000, we discussed technical aspects of the revised reactor oversight process, including the technical adequacy of current and proposed performance indicators (PIs) and the significance determination process (SDP).

This report responds to the Commission request in the December 17, 1999 Staff Requirements Memorandum, that the ACRS evaluate the extent to which the PIs, collectively, provide meaningful insights into those areas of plant operations that are most important to safety. Our Subcommittee on Plant Operations met on January 20, 2000, to discuss these matters. We also had the benefit of the documents referenced.

Conclusions and Recommendations

1. The Revised Reactor Oversight Process (RROP) makes NRC assessments and actions more objective, predictable, and understandable to both the public and industry.

2. Although the RROP is a work in progress, it is ready for initial implementation at all power reactors. Further adjustments in the process may be needed as more experience is gained with a larger base of plants. Because changes are expected after the initial implementation, staff should look for methods to implement the process in ways that it can be easily changed.

3. The choices of the PIs and the associated thresholds remain controversial. Alternative views of ACRS members regarding the choice of thresholds are offered in the discussion.

4. The SDP is incomplete. Further development of this process and the analytical tools it uses is required for full implementation.

5. Additional PIs will be needed for full and effective implementation of the RROP. In particular, PIs are needed to characterize the licensee's problem identification and corrective action program (CAP), human performance, safety culture, and low-power and shutdown operations.

Discussion

The RROP pilot program was completed in November 1999 and lessons learned have resulted in changes that have improved the process prior to its initial implementation at all power reactors. The process is intended to ensure that plants continue to perform at an acceptable level and to provide early warning of adverse trends.

We recognize that the RROP is a work in progress and that certain aspects could not be fully exercised and evaluated during the 6-month pilot program. We agree that the overall process, the concept of the cornerstones, and the associated framework are sound. The new process will make NRC assessments more objective, predictable, and understandable to both the public and industry and should be approved for initial implementation at all plants. The staff has stated that continued development and implementation of the process will not adversely affect initial implementation. The staff plans to assess the effectiveness of the entire process after the first year of initial implementation.

The staff has selected a set of PIs to be used as part of the RROP, which is intended to be risk informed and performance based. The PIs are defined in the expectation that they are correlated with risk, even though in some cases the implied correlation cannot be explicitly defined or quantified. Without such an explicit connection to risk, it is difficult to determine which and how many PIs are sufficient or to determine quantitative threshold values. An added practical constraint to the selection of a set of PIs is the limited ability of the staff to obtain data from the licensees.

Recognizing that there are unavoidable limitations in the chosen set of PIs, the staff has developed a baseline inspection program for each cornerstone to complement and supplement the PIs. We agree with the staff that the technical adequacy of the proposed PIs should be evaluated in the context of the overall assessment process.

Another key element of the RROP is the licensee's problem identification and CAP. A basic tenet of the RROP is that the licensee's CAP should be relied upon to correct issues that do not result in crossing safety performance thresholds. This is based on the assumption that the improved overall industry performance over the past 10 years has demonstrated the general robustness of the CAPs. Confirmation of this assumption for individual plants requires that NRC periodically assess the effectiveness of each CAP as part of the baseline inspection program.

We believe that additional PIs will be needed for full and effective implementation of the program. In particular, PIs are needed to characterize the licensee's problem identification and CAP, human performance, safety culture, and low-power and shutdown operations.

The proposed green-white PI thresholds have been selected as the 95th percentile of the values for the whole population of operating plants. Some ACRS members believe that this approach has led to the selection of PI thresholds that are too high to provide early warning of adverse trends in performance. The proposed values are such that most indicators will always be in the green, therefore, the PIs may not contribute meaningful information to the oversight process.

Because performance in the green may be interpreted as good performance, there will be a reduced incentive for improved performance by the licensees.

Some ACRS members find the staff's approach to the selection of the green-white thresholds acceptable. Current industry practices and regulatory requirements, along with the previous inspection and oversight process, have resulted in acceptable overall industry performance. Therefore, the set of current values for the PIs does represent the range of acceptable performance values, and the 95th percentile values are to identify outliers. Obviously there is some degree of arbitrariness involved, but it is an acceptable choice for initial implementation.

Some ACRS members believe that there is a fundamental flaw with the process of selecting the PI thresholds. As noted in our report dated June 10, 1999, a lesson from the probabilistic risk assessments and Individual Plant Examinations is that the risk profile of each plant is unique. The PIs and the thresholds should reflect this finding and should be plant specific. This means that the threshold for a specific PI should be selected from a distribution of values that reflects past performance with respect to this PI at that plant. A typical value that is usually selected is the 95th percentile of this plant-specific curve. The current process, however, selects the thresholds from distributions that include plant-to-plant variability. A plant-to-plant variability curve represents the distribution of the past values of a PI across all plants. The selection of the 95th percentile of these distributions could have two significant consequences. First, the thresholds are too high for the plants whose past performance placed them below the chosen threshold value. Second, the few plants with past performance above the selected threshold value may be in the "white" category without credit for other compensating features. This situation would create pressure on those licensees to "improve" their performance with respect to the PI, thereby ratcheting up the expected performance of the plant.

The same ACRS members believe that the establishment of plant-specific thresholds is feasible. The staff has agreed that, ideally, plant-specific thresholds would be desirable, but that they cannot be established at this time. An example of such an exercise, however, was the implementation of the maintenance rule and the proposed plant-specific performance criteria by the licensees. The staff has collected and published plant-specific data, including those from studies by the former Office for Analysis and Evaluation of Operational Data, e.g., NUREG/CR-5500, Volumes 4-8, and associated updates. Alternatively, it may be possible to identify groups of plants with similar design and operational characteristics that could share the same PI threshold values.

Some ACRS members are concerned that the high PI thresholds focus on equipment performance only. The staff has stated that cross-cutting issues involving human performance and safety culture will manifest themselves through the PIs or the baseline inspections. The baseline inspections may lag adverse human performance trends and not trigger action until some PI thresholds are exceeded. PI thresholds do not appear to provide timely warning of negative trends.

The SDP is designed to provide guidance for the risk characterization of inspection program findings so that the overall licensee performance assessment process can compare and evaluate the findings on a significance scale similar to that established for PIs. The SDP is still incomplete. Findings from workshops and lessons learned on the pilot program have not been accounted for in the SDP. Because of limitations in the staff's analytical tools, very approximate risk assessment methods are used for some SDP evaluations.

It is expected that the overwhelming majority of SDP findings will be "green." We are concerned that such an outcome could mask programmatic problems. For example, weakness in a maintenance program that was manifested by the failure of an unimportant component would result in a "green" finding, but the same programmatic weakness could result in the failure of a safety-significant component. The staff recognizes the potential problem but believes that such programmatic weakness will be reflected in the PIs or identified through inspection of the problem identification and CAP. More experience with the process is needed to validate this assumption.

Notwithstanding these concerns, we believe that the staff has developed a comprehensive oversight process, which is a significant improvement over the previous one. The staff's request to proceed with initial implementation should be approved, recognizing that changes will be made to the RROP, including the SDP; that research should continue to identify better choices for PIs and associated thresholds; that the current PIs are limited in scope; and that any reduction in the baseline inspection effort will require more realistic PIs.

Once the RROP has been implemented, substantial resistance may arise toward any changes. Because changes are expected after the initial implementation, staff should look for methods to implement the process in ways that it can be easily changed.

Sincerely,

Dana A. Powers

Dana A. Powers
Chairman

References:

1. Memorandum dated February 24, 2000, from William D. Travers, Executive Director for Operations, NRC, for The Commissioners, Subject: SECY-00-0049, Results of the Revised Reactor Oversight Pilot Program.
2. Memorandum dated December 17, 1999, from Annette L. Vietti-Cook, Secretary, NRC, to John T. Larkins, ACRS, Subject: Staff Requirements - Meeting with Advisory Committee on Reactor Safeguards, November 4, 1999.
3. Nuclear Energy Institute, NEI 99-02, Draft Revision D, "Regulatory Assessment Performance Indicator Guideline," November 1999.
4. Letter dated November 23, 1999, from Samuel J. Collins, Director, Office of Nuclear Reactor Regulation, NRC, to Dana A. Powers, Chairman, ACRS, Subject: Advisory Committee on Reactor Safeguards Review of Revised Reactor Oversight Process Technical Components.
5. Memorandum dated June 18, 1999, from Annette Vietti-Cook, Secretary, NRC, to William D. Travers, Executive Director for Operations, NRC, Subject: Staff Requirements - SECY-99-007 - Recommendations for Reactor Oversight Process Improvements, and SECY-99-007A - Recommendations for Reactor Oversight Process Improvements (Follow-up to SECY-99-007).

6. Letter dated June 10, 1999, from Dana A. Powers, Chairman, ACRS, to William D. Travers, Executive Director for Operations, NRC, Subject: Pilot Application of the Revised Inspection and Assessment Programs, Risk-Based Performance Indicators, and Performance-Based Regulatory Initiatives and Related Matters.
7. Report dated February 23, 1999, from Dana A. Powers, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: Proposed Improvements to the NRC Inspection and Assessment Programs.
8. U. S. Nuclear Regulatory Commission, NUREG/CR-5500, Vol. 4, "Reliability Study: High-Pressure Coolant Injection (HPCI) System, 1987-1993," September 1999.
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12. U. S. Nuclear Regulatory Commission, NUREG/CR-5500, Vol. 8, "Reliability Study: High Pressure Core Spray (HPCS) System, 1987-1993," September 1999.
13. U. S. Nuclear Regulatory Commission, NUREG/CR-xxx Vol. X, "Reliability Study Update: High-Pressure Coolant Injection (HPCI) System, 1987-1998" (Draft), October 1999.
14. U. S. Nuclear Regulatory Commission, NUREG/CR-xxx, Vol. X, "Reliability Study Update: Reactor Core isolation Cooling (RCIC) System, 1987-1998" (Draft), October 1999.
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A

IMPROVEMENT OF THE QUALITY OF RISK INFORMATION FOR REGULATORY DECISIONMAKING

Thomas Kress

May 16, 2003 Report

- Focused on several aspects of PRA methodology and practice that need to be addressed to achieve comprehensive high quality PRAs
- Improving the scope and quality of the PRAs is very important to the advancement of risk-informed regulation
- PRA insights may be affected significantly by PRA scope and quality

- **Completeness of risk information requires that PRAs address low-power and shutdown (LPSD) modes and “external” events, such as fires, in addition to power operations**

- **Use of bounding analyses to account for the missing PRA elements does not necessarily lead to conservative decisions**

- **The staff should develop guidance on how licensees and peer-review teams should consider operating experience to improve PRA completeness**

- **It has been suggested that as many as 20% of the events evaluated by the Accident Sequence Precursor (ASP) Program involve initiating events and accident sequences not modeled in existing PRAs**

- **The assessment of uncertainties should address model uncertainties**

- **Guidance for the quantitative evaluation of such uncertainties should be developed**

ACRS LETTERS

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

May 16, 2003

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

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

Dear Chairman Diaz:

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

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

CONCLUSIONS AND RECOMMENDATIONS

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

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

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

DISCUSSION

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Sincerely,

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Chairman

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