COMMISSION MEETING EXHIBITS

MEETING WITH ACRS

WEDNESDAY, DECEMBER 5, 2001

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

November 28, 2001

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, DECEMBER 5, 2001 - SCHEDULE AND BACKGROUND INFORMATION

The ACRS is scheduled to meet with the NRC Commissioners between 1:30 p.m. - 3:30 p.m. on Wednesday, December 5, 2001, to discuss the items listed below. Background materials related to these items are attached.

		ESTIMATED TIME
INTRODUCTION - NRC Chairman, Dr. Richard A. Meserve		5 minutes
PRESENTATIONS - Advisory Committee on Reactor Safeguards		
OVERVIEW OF TOPICS AND NEAR-TERM ACTIVITIES G. E. Apostolakis, Chairman, ACRS		
1.	Reactor Oversight Process J. D. Sieber	20 minutes
2.	Regulatory Challenges for Future Plant Designs T. S. Kress	10 minutes
3.	ACRS Activities Associated with Core Power Uprates G. B. Wallis/D. A. Powers	10 minutes
4.	ACRS Activities Associated with License Renewal M. V. Bonaca	10 minutes
CLOSING REMARKS		5 minutes

***NOTE:** Estimated times are for presentation only and do not include time for Commission Questions and Answers.

Annette L. Vietti-Cook

Attachments: As stated

cc: ACRS Members
ACRS Staff

ACRS MEETING WITH THE U.S. NUCLEAR REGULATORY COMMISSION

G. E. Apostolakis

December 5, 2001

OVERVIEW OF TOPICS NEAR-TERM ACTIVITIES

- Reactor Oversight Process
- Regulatory Challenges for Future Plant Designs

OVERVIEW (CONT'D)

- ACRS Activities Associated with Core Power Uprates
- Status of ACRS Activities on License Renewal

1

REACTOR OVERSIGHT PROCESS

J. D. Sieber

4

COMMISSION REQUEST

- Review the use of Performance Indicators in the Reactor Oversight Process to ensure that the PIs provide meaningful insight into aspects of plant operation that are important to safety.

COMMISSION REQUEST

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

RESPONSE: PIs

- Current PIs do provide meaningful insight into plant performance.
- The numerical values for the white/yellow and yellow/red thresholds for initiating events and mitigation system PIs are not meaningful. They should be revised.

RESPONSE: PIs(CONT'D)

- Definitions of terms such as "unavailability" should be consistent among agency activities.
- Unreliability should be a PI.
- Consider other related work, such as reliability studies, when assessing need to revise and develop new PIs.

RESPONSE: SDPs

- The most pressing need is to improve the SDP tools.
- The technical adequacy of risk-based SDPs depends on the availability and quality of a relevant probabilistic risk assessment (PRA).

RESPONSE: SDPs

- Threshold values for risk-based SDPs are appropriate.
- Some SDPs are incomplete and some, such as fire protection, are overly subjective.
- SDPs for at-power situations are meaningful.

RESPONSE: SDPs

- An SDP based on low-power and shutdown PRAs or other shutdown management tools is needed.
- Documented review of the SDP worksheets and Simplified Plant Analysis Risk (SPAR) models is necessary for public confidence.

PIs and SDPs

- PIs and SDPs and the corresponding equivalency of the combination of findings in the action matrix have not been well documented.

PIs and SDPs

- Formal decision analysis could be helpful in making the selection of thresholds and the action matrix more objective and scrutable.

CONCLUSIONS

- The ROP is an evolving process.
- The staff has done an excellent job establishing the basic framework.
- The ROP is more objective and understandable than the former oversight process.

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

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

References:
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.

2

REGULATORY CHALLENGES for FUTURE PLANT DESIGNS

T.S. Kress

ACRS WORKSHOP

- Various reactor designs and potential regulatory and policy issues were discussed at an ACRS Workshop on "Regulatory Challenges for Future Reactor Designs" – June 4-5, 2001.

ACRS WORKSHOP

- Attended by about 100 stakeholders.
- Presenters included representatives from DOE, NEI, MIT, Exelon Generation Co., Westinghouse, General Atomics, General Electric, ORNL, and NRC Staff.

SUMMARY

- The workshop was a success.
- A good tone was set in the keynote address by Commissioner Diaz.
- A list of regulatory challenges for future plant designs was developed.
- The workshop proceedings are in preparation.

OTHER ACTIVITIES

- Participated in a RES workshop (October 10-12, 2001) on high-temperature gas-cooled reactor safety and research issues.
- Met with NRC staff and Exelon's representatives in October 2001 to discuss:

ACTIVITIES (CONT'D)

- NRC readiness for reviewing future plant designs
- Exelon's proposed licensing approach for the Pebble Bed Modular Reactor (PBMR)

ACTIVITIES (CONT'D)

- Met with the NRC staff and stakeholders in November 2001
  - Discussed the staff's evaluation of Exelon's proposed licensing approach for the PBMR.
- Scheduled additional meetings with the staff to discuss pertinent issues.

3

ACRS ACTIVITIES ASSOCIATED WITH CORE POWER UPRATES

G. B. Wallis

D. A. Powers

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BACKGROUND

- Current economic conditions strongly favor power uprates/plant life extension.
- Many licensees are actively planning or have initiated power uprate programs.

BACKGROUND (CONT'D)

- In early 1990s, General Electric initiated a generic power uprate program.

BACKGROUND (CONT'D)

- Westinghouse/Combustion Engineering have recently approached the staff regarding power uprate plans (10-20% uprates).

GE GENERIC UPRATE PROGRAM

- GE Generic Uprate Program (initiated in 1991)
  - -Limited to 5% power uprates
  - -Lead Plant: Fermi Unit 2 (ACRS review 9/92)

GENERIC UPRATE PROGRAM (CONT'D)

- Most operating BWRs will utilize this program

GE EXTENDED UPRATE PROGRAM

- GE Extended Power Uprate Program (initiated in 1995)
  - -Uprates of 5% - 20%
  - -Lead Plant: Monticello (6.3%) (ACRS review 7/98)

EXTENDED UPRATE PROGRAM (CONT'D)

- Encouraged staff/applicant reviews of impact on plant risk
- GE Generic Topical Report addresses program scope/content

SIGNIFICANT ISSUES

- Anticipated Transients Without Scram (ATWS)
  - ATWS recovery
  - Operator response times
- Core Instability

SIGNIFICANT ISSUES (CONT'D)

- Material degradation
  - Irradiation-assisted stress corrosion cracking
  - Embrittlement of pressure vessel

SIGNIFICANT ISSUES (CONT'D)

- Flow-assisted corrosion
- Fatigue

• Containment response

RECENT REVIEWS OF EXTENDED UPRATES

- Duane Arnold Energy Center (15.3% uprate) – ACRS letter dated October 17, 2001
  - Recommended approval of uprate application

RECENT REVIEWS

- Dresden/Quad Cities Nuclear Power Stations (17/17.8% uprates)
  - ACRS letter to be issued in December 2001

RECENT REVIEWS

- Improved guidance required from staff on detail in safety evaluations.
- Need for confirmatory analyses to complement applicant submittals.

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

October 17, 2001

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

SUBJECT: DUANE ARNOLD ENERGY CENTER EXTENDED POWER UPRATE

Dear Chairman Meserve:

During the 486th meeting of the Advisory Committee on Reactor Safeguards, October 4-6, 2001, we met with representatives of the NRC staff and the Nuclear Management Company to review the license amendment request for an increase in core thermal power for the Duane Arnold Energy Center (DAEC), pursuant to the General Electric Nuclear Energy Extended Power Uprate Program. Our subcommittee on Thermal-Hydraulic Phenomena also reviewed this matter during meetings held on June 12 and September 26-27, 2001. During our review, we had the benefit of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

1. The DAEC application for the extended power uprate should be approved.

2. The Safety Evaluation Report (SER) should be revised to document adequately the technical resolution of the issues raised by the staff.

3. The staff should develop improved guidance on the detail to be provided in SERs and criteria for when independent assessments should be performed to complement its reviews of applicant submittals.

DISCUSSION

The Nuclear Management Company has requested an amendment to the DAEC operating license for a 15.3% increase over the plant's current operating power limit. Previously, the staff had approved a smaller power uprate. Consequently, the current application is for a power uprate of 20% over the originally licensed power. This is the largest power uprate ever considered for boiling water reactors (BWRs) in the United States. It is anticipated that many other licensees will request similarly large increases in the operating powers of BWRs. Consequently, we anticipate that staff review of the DAEC power uprate will be a template for future reviews and will set the expectations for many future power uprate applications.

A generic methodology for evaluating and justifying power uprates of up to 20% for BWRs has been developed by General Electric. This generic methodology has been approved by the staff. The DAEC application has adopted this methodology and, in fact, the NRC staff has used the methodology to guide its review of this power uprate application.

The power increase at DAEC will be achieved by increasing steam production, while holding liquid flow in the core, dome pressure and temperatures quite near current values. The increased steam production is achieved by "flattening" the core power profile, which involves increasing power generation in the outer regions of the core. There is an increase in feedwater flow to match the increased production of steam. Balance-of-plant modifications are required and will cause the DAEC power increase to be performed in two steps.

Many technical issues must be addressed in an application for power uprate. Of these, we consider five to be especially significant:

1. Susceptibility of the plant to ATWS (Anticipated Transients Without Scram)

2. ATWS recovery

3. Reduction in some of the times available for operator actions because of higher decay heat

4. Material degradation due to irradiation-assisted stress corrosion cracking (IASCC) of reactor internals and flow-assisted corrosion and fatigue of feedwater piping

5. Containment response to accident events involving higher decay heat levels

Our examinations of the staff's SER and Requests for Additional Information submitted by the staff to the applicant persuaded us that the staff had raised numerous, pertinent issues concerning the conformance of the power uprate to approved methodologies. Though we persuaded ourselves eventually that the DAEC power uprate could be accomplished safely, we found it difficult to obtain information on the technical resolution of the issues either in the staff's SER or in our meetings with the staff. An exception to this common difficulty was the resolution of issues concerning containment response to design-basis accident events. In this case, the staff provided us a report on comparisons of applicant analyses with analyses done using an independent computational tool.

We found it far more difficult to assure ourselves that the DAEC core is susceptible only to global power oscillations and does not need to consider local power oscillations. It was similarly difficult to assure that ATWS recovery methods were applicable to cores with flattened power profiles, that critical human actions had been identified with adequate independence by the staff, and that material degradation sensitivities had been adequately assessed.

Many of the challenges that we encountered in our review of the DAEC power uprate application could have been eased if the staff had improved guidance on the detail to be provided in SERs and developed criteria for when independent assessments should complement reviews of applicant submittals.

ACRS Members Mario Bonaca and F. Peter Ford did not participate in the Committee's review of this matter.

Sincerely,

George E. Apostolakis
Chairman

References:

1. Memorandum dated September 5, 2001, to John T. Larkins, ACRS, from J. Zwolinski, Office of Nuclear Reactor Regulation, NRC, Subject: Draft Safety Evaluation for Duane Arnold Energy Center Extended Power Uprate (draft Predecisional report).
2. GE Nuclear Energy, Topical Report, NEDC-32424P-A, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate," February 1999 (Proprietary).
3. GE Nuclear Energy, Topical Report, NEDC-32523P-A, "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate," February 2000 (Proprietary)
4. GE Nuclear Energy, Topical Report, NEDC-32523P-A, Supp 1, Volume 1, "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate - Supplement 1, Volume I," February 1999, and Volume II, April 1999 (Proprietary).
5. GE Nuclear Energy, Topical Report, NEDC-32992P, "ODYSY Application for Stability Licensing Calculations," October 2000 (Proprietary).
6. BWR Owners' Group Letter dated March 8, 1996, transmitting GE Nuclear Energy Licensing Topical Report, BWR Owners Group Long-Term Stability Solutions Licensing Methodology, NEDO-31960-A, November 1995.
7. Report (draft final) from A. Cronenberg, ACRS, "Margin Reduction Estimates for Re-Licensed/Uprated Plants: Hatch Case Study," August 2001.
8. Response by Nuclear Management Company to ACRS Thermal-Hydraulic Phenomena Subcommittee question, undated, attached to October 3, 2001 Memorandum from P. Boehnert to ACRS Members.
9. U.S. Nuclear Regulatory Commission, Technical Evaluation Report, ISL-NSAD-NRC-01-001, "Duane Arnold Energy Center Extended Power Uprate Containment Analysis Audit Calculation," B. Gitnick, Information Systems Laboratory, Inc., July 2001.
10. Memorandum (undated) from J. Zwolinski, Office of Nuclear Reactor Regulation, NRC, Subject: Responses to Advisory Committee on Reactor Safeguards (ACRS) Subcommittee Questions Regarding Duane Arnold Energy Center Extended Power Uprate, attached to October 3, 2001, Memorandum from P. Boehnert, to ACRS Members (contains Proprietary information).
11. GE Nuclear Energy Licensing Topical Report, NEDC-32980P, Rev. 1, "Safety Analysis Report for Duane Arnold Energy Center Extended Power Uprate," April 2001 (Proprietary).
12. Nuclear Management Company Memorandums: Response to Request for Additional Information - Duane Arnold Energy Center Extended Power Uprate, dated April 9, March 23, April 16, April 16 (Proprietary), May 8 (Proprietary), May 10, May 11, May 11 (Proprietary), May 22, May 29 (Proprietary),and June 5, 2001.
13. Nuclear Management Company Memorandums: Response to Request for Additional Information - Extended Power Uprate, June 11, June 18, June 21, June 28, July 11, July

19, July 25, August 1 (proprietary), August 1(proprietary), August 10 (proprietary), August 16 (proprietary), and August 21, 2001.

4

STATUS OF ACRS ACTIVITIES ON LICENSE RENEWAL

M. V. Bonaca

REVIEWS SINCE LAST COMMISSION MEETING

- Possible revision to 10 CFR Part 54, License Renewal Rule
- Final reviews of Arkansas Nuclear One, Unit 1 (ANO-1) and Hatch applications
- Initial review of Turkey Point

RECOMMENDATIONS ON REVISING 10 CFR 54

- 10 CFR Part 54 is effective and efficient. Rule need not be revised at this time.
- Avoiding rulemaking will maintain stability of the existing process.

RECOMMENDATIONS

- Resolution of open technical issues can be incorporated in future updates of the generic license renewal guidance documents.

ANO-1 and HATCH APPLICATIONS

- The Committee completed its reviews of the ANO-1 and Hatch applications in May 2001 and November 2001, respectively.
- The requirements of 10 CFR Part 54 were effectively implemented.

ANO-1 and HATCH APPLICATIONS

- The staff has performed effective reviews of the applications.

ANO-1 and HATCH

- The resolution of open items was appropriate.

ANO-1 and HATCH

- Adequate programs have been established to manage the effects of aging so that plants can be operated safely in accordance with their current licensing basis for the period of extended operation.

ANO-1 and HATCH

- Review of ANO-1 application was completed five months ahead of schedule.
- Hatch SER clarifications should eventually be incorporated into the Generic License Renewal Guidance Documents.

TURKEY POINT APPLICATION

- The application was complete and scrutable, and the draft SER was comprehensive.
- The ACRS did not issue an interim report, because only four open items remained to be addressed.

TURKEY POINT APPLICATION (CONT'D)

- The ACRS plans to issue a report on the application in the Spring of 2002.

OBSERVATIONS

- Applications are becoming more scrutable and complete.
- The ACRS expects this trend to continue as applications follow the now available generic license renewal guidance documents.

PLANNED ACTIVITIES IN CY 2002

- Initial reviews of the Surry and North Anna, McGuire and Catawba, and Peach Bottom applications
- Final reviews of the Turkey Point, and Surry and North Anna

PLANNED ACTIVITIES

- Review of revisions to the Generic Guidance Documents
- Two License Renewal Sub-committees starting in CY 2002

UNITED STATES
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WASHINGTON, D.C. 20555-0001

November 16, 2001

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

Dear Chairman Meserve:

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE LICENSE RENEWAL APPLICATION FOR THE EDWIN I. HATCH NUCLEAR PLANT, UNITS 1 AND 2

During the 487th meeting of the Advisory Committee on Reactor Safeguards, November 8-10, 2001, we completed our review of the Southern Nuclear Operating Company's (SNC's) application for license renewal of the Edwin I. Hatch Nuclear Plant, Units 1 and 2, and the related final Safety Evaluation Report (SER). We issued an interim letter concerning this application and the SER with open items on April 16, 2001, and our Plant License Renewal Subcommittee held discussions with representatives of the staff and SNC on October 25, 2001. We also had the benefit of the documents referenced.

Conclusions and Recommendations

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

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

3. The staff has performed a comprehensive review of SNC's application. The open items identified in the February 2001 draft SER have been resolved satisfactorily.

4. The SER clarifies staff positions on non-safety-related seismic II-over-I piping systems, long-lived passive components of skid-mounted complex assemblies, fan housings, and damper frames. These clarifications provide significant guidance that could prevent these issues from becoming open items in future applications. They should be incorporated into the generic license renewal guidance documents.

Background and Discussion

This report fulfills the requirement of 10 CFR 54.25 that the ACRS review and report on license renewal applications. SNC requested renewal of the operating licenses for Hatch, Units 1 and 2, for a period of 20 years beyond the current license terms, which expire on August 6, 2014, for Unit 1, and June 13, 2018, for Unit 2. The final SER documents the results of the staff's review of information submitted by SNC, including those commitments that were necessary to resolve open items identified by the staff in its February 2001 draft SER. The staff's review included the verification of the completeness of structures, systems, and components (SSCs) identified in the application, the validation of the integrated plant assessment process, the identification of the possible aging effects associated with each passive long-lived component, and the verification of the adequacy of the aging management programs. The staff also conducted site inspections to verify the adequacy of the implementation of the methodology described in the application.

As noted in our April 16, 2001 interim letter, the SNC's approach to identifying SSCs that are within the scope of the License Renewal Rule is function-based, rather than the system-based approach used in previous applications. This approach was adequate, but made it difficult for the reviewers to ascertain which SSCs were in scope and which were not. The staff's review relied heavily on supporting documents located at the site and on requests for additional information. In addition, the staff performed a "walk-through" of the process for three systems that are within scope. On the basis of its extensive review, the staff identified some additional components that the applicant should have included within the scope of license renewal, and classified them as open items. These open items have been resolved by including the additional components in scope. We concur with the staff that the applicant has now properly identified SSCs requiring an aging management review.

Components brought into scope through the resolution of open items include non-safety-related seismic II-over-I piping systems, long-lived passive components of skid-mounted complex assemblies, fan housings, and damper frames. The inclusion of these components was contested in previous license renewal applications. The issue of seismic II-over-I piping is an open item in an application that is currently under review. The Hatch SER includes effective clarifications of why these components need to be included within scope. The guidance provided by these clarifications could prevent these issues from becoming open items in future applications. Consequently, these clarifications should be incorporated into the generic license renewal guidance documents.

SNC has conducted a comprehensive aging management review of SSCs that are within scope. Aging effects were identified on the basis of component material, operating environment, and operating stresses using plant-specific and industry-wide operating experience. Topical reports developed by the Boiling Water Reactor Vessel and Internals Project (BWRVIP) were also used to identify aging effects and to develop aging management programs that support the Hatch application. We reviewed a number of BWRVIP topical reports and commented on their effectiveness in supporting license renewal in our April 16, 2001 letter.

Appendix A to the Hatch application describes 17 existing programs, 5 modified programs, and 7 new programs that SNC has implemented to manage aging effects during the period of extended operation. The resolution of open items has resulted in added commitments to these programs, including a one-time inspection of plant service water piping in the diesel generator building and a one-time inspection of small-bore butt-welded stainless steel piping.

One of the added commitments resulting from resolution of open items involves periodic testing of fire-protection system sprinkler heads that are within the scope of license renewal. SNC had proposed a one-time test of such sprinkler heads at or before the start of the period of extended operation. The staff did not agree with the one-time test, because the design life (50 years) of the sprinkler heads does not cover the period of extended operation. As recommended by the staff, SNC has committed to perform the sprinkler head tests as specified in the National Fire Protection Association (NFPA) Standard 25, Section 2.3.3.1, "Sprinklers." The application of this Standard will result in periodic testing of the sprinkler heads at 10-year intervals, with the first test taking place during the third year of the renewal period. This program is acceptable because it confirms the effectiveness of the periodic inspections to which the sprinkler heads are subjected and ensures testing of the sprinkler heads early in the renewal period.

The staff requested that SNC perform a one-time inspection of the four buried emergency diesel generator (EDG) fuel oil storage tanks. SNC responded by performing visual inspections and ultrasonic testing of one of the four tanks. Ultrasonic testing of 144 locations along the lower shell of the tank indicated that there was no thinning of the wall. Visual inspections of the internal surface revealed very little corrosion. SNC and the staff concluded that the one-time inspection demonstrated that loss of material of the diesel fuel oil storage tanks was not an aging effect requiring management during the period of extended operation.

We also considered the possibility that the external coating of a tank could be damaged at some location during installation and result in localized fuel oil leakage. Such damage would be of concern during the current license term and, thus, would not be specific to the period of extended operation. The safety consequences would not be significant because the potential leakage would not cause substantial depletion of the fuel oil inventory before it would be detected. We concur with the staff's determination that loss of material of the diesel fuel oil storage tanks is not an aging effect requiring management during the period of extended operation.

Jet pump assemblies and fuel supports contain cast austenitic stainless steel (CASS) components that are within the scope of license renewal. These components may be exposed to neutron fluence levels that would make them susceptible to neutron irradiation embrittlement and loss of fracture toughness. Since neutron embrittlement becomes a concern when cracks are present in the components, the staff requested that SNC propose a one-time inspection of the jet pump assemblies and fuel supports to confirm that these CASS components have not experienced cracking. Following this request, the staff recognized that cracking of CASS components has not been observed to date. Furthermore, BWRVIP-41, "BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines," requires inspections of jet pump assembly welds that are

generally believed to be more susceptible to cracking than the CASS components and, therefore, provide a leading indicator for inspection of CASS components. SNC has committed to perform the weld inspection required by BWRVIP-41. In addition, the BWRVIP and the NRC's Office of Nuclear Regulatory Research plan to conduct confirmatory research to determine the effects of high levels of neutron fluence on BWR internals. SNC has committed to implement any requirements resulting from this research. Given the above, the staff concluded that the requested one-time inspection is not warranted at this time. We agree with the staff's conclusion.

Time-limited aging analyses (TLAA) have shown that neutron irradiation embrittlement during the extended period of operation will have no significant impact on the integrity of the Hatch reactor vessels. At the end of the renewal period, the vessels will still have margin over applicable regulatory limits. In order to monitor time-dependent parameters used in the TLAA, SNC plans to implement the provisions of the integrated surveillance program (ISP) described in BWRVIP-78, BWR integrated surveillance program plan, and BWRVIP-86, BWR integrated surveillance program implementation plan. Since these topical reports have not yet been approved by the staff, SNC committed to implement either a staff-approved ISP or a plant-specific program that meets specific staff requirements on periodic removal of capsules to monitor neutron fluence and the impact of irradiation on the reactor vessels. SNC committed to provide the staff with program details prior to the period of extended operation. The staff made this commitment a license condition.

The staff has performed a comprehensive review of SNC's application. The applicant and the staff have identified plausible aging effects associated with passive and long-lived components. Adequate programs have been established to manage the effects of aging so that Hatch, Units 1 and 2, can be operated safely in accordance with their current licensing bases for the period of extended operation.

Sincerely,

George E. Apostolakis
Chairman

References:

1. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the License Renewal of the Edwin I. Hatch Nuclear Plant, Units 1 and 2," issued October 2001.
2. U. S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the License Renewal of the Edwin I. Hatch Nuclear Plant, Units 1 and 2," issued February 2001.
3. Letter dated February 29, 2000, from H. L. Sumner, SNC, to the U.S. Nuclear Regulatory Commission, "Edwin I. Hatch Nuclear Plant Application for Renewed Operating Licenses."
4. Letter dated April 16, 2001, from George E. Apostolakis, Chairman ACRS, to

William D. Travers, Executive Director for Operations, NRC, Subject: Interim Letter Related to the License Renewal of Edwin I. Hatch Nuclear Station, Units 1 and 2.

5. Topical Report BWRVIP-41, "BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines," October 1997.
6. Topical Report BWRVIP-78, "BWR Integrated Surveillance Program - Unirradiated Charpy Reference Curves for Surveillance Material," December 1999.
7. Topical Report BWRVIP-86, "BWR Vessel and Internals Project, BWR Integrated Surveillance Program Implementation Plan."

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

July 20, 2001

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

SUBJECT: RECOMMENDATION ON THE NEED TO REVISE 10 CFR PART 54, "REQUIREMENTS FOR RENEWAL OF OPERATING LICENSES FOR NUCLEAR POWER PLANTS"

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 Nuclear Energy Institute (NEI) regarding the need to revise 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," to resolve generic technical issues associated with license renewal. We also discussed this matter during our 483rd meeting on June 6-8, 2001. During our review, we had the benefit of the documents referenced.

Recommendation

10 CFR Part 54 is effective and efficient. It does not need to be revised at this time.

Discussion

In a Staff Requirements Memorandum (SRM) dated August 27, 1999, regarding SECY-99-148, "Credit for Existing Programs for License Renewal," the Commission asked the staff to prepare a detailed analysis and provide recommendations on whether it would be appropriate to resolve generic technical issues, including any credit for existing programs, by rulemaking. These recommendations were to be based on the accumulation of more data from license renewal applications of different designs and on experience gained from reviewing more applications.

Since the SRM was issued, the staff has reviewed license renewal applications for three pressurized water reactor plants and renewed their licenses. We have reviewed and commented on the Safety Evaluation Reports (SERs) associated with these applications. On the basis of our review, we believe that the license renewal process developed by the staff, with feedback from stakeholders, under the current rule is effective. This process is documented in a set of guidance documents: Generic Aging Lessons Learned (GALL) report, Standard Review Plan, and Regulatory Guide 1.188 that endorses NEI 95-10, Revision 3, "Industry Guideline for Implementing the Requirements of 10 CFR Part 54 – The License Renewal Rule."

These guidance documents incorporate the resolution of technical issues, such as credit for existing programs, thus making the license renewal process understandable and predictable. Future updates of the guidance documents will provide the means for incorporating the resolution of remaining outstanding technical issues without amending the rule. Although review of the first boiling water reactor application for Hatch, Units 1 and 2, has not been completed, resolution of the open items in the interim SER does not appear to require rulemaking.

License renewal applications and their reviews have become increasingly efficient with subsequent applications. We expect them to become even more efficient when licensees endorse the approaches suggested by the now-approved guidance documents. Avoiding rulemaking at this time will further stabilize the existing process and facilitate the submittal and review of future applications.

Sincerely,

George E. Apostolakis
Chairman

References
1. Memorandum dated August 27, 1999, from Annette L. Vietti-Cook, Secretary, to William D. Travers, Subject: SECY-99-148 - Credit for Existing Programs for License Renewal.
2 Letter dated June 4, 2001, from Douglas J. Walters, Nuclear Energy Institute, to Christopher I. Grimes, Office of Nuclear Reactor Regulation, NRC, Subject: License Renewal Rulemaking.
3 Letter dated June 26, 2001, from David Lochbaum, Union of Concerned Scientists, to Christopher I. Grimes, Office of Nuclear Reactor Regulation, NRC, Subject: License Renewal Rulemaking.
4. Letter dated April 13, 2001, from George E. Apostolakis, Chairman, ACRS, to Richard A. Meserve, Chairman, NRC, Subject: Proposed Final License Renewal Guidance Documents.
5. Letter dated November 15, 2000, from Dana A. Powers, Chairman, ACRS, to Richard A. Meserve, Chairman, NRC, Subject: License Renewal Guidance Documents.
6. U. S. Nuclear Regulatory Commission, NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," dated March 1, 2001.
7. U. S. Nuclear Regulatory Commission, NUREG-1801, Vols. 1 and 2, "Generic Aging Lessons Learned (GALL) Report," dated March 1, 2001.
8. U. S. Nuclear Regulatory Commission, Regulatory Guide 1.188, "Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses," March 2001.
9. Nuclear Energy Institute, NEI 95-10, Revision 3, "Industry Guideline for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule," March 2001.
10. U. S. Nuclear Regulatory Commission, "Safety Evaluation Report With Open Items Related to the License Renewal of Edwin I. Hatch, Units 1 and 2," February 2001.

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

May 18, 2001

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

Dear Chairman Meserve:

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE LICENSE RENEWAL APPLICATION FOR ARKANSAS NUCLEAR ONE, UNIT 1

During the 482nd meeting of the Advisory Committee on Reactor Safeguards, May 10-11, 2001, we completed our review of Entergy Operations, Inc., application for license renewal of Arkansas Nuclear One, Unit 1 (ANO-1), and the related final Safety Evaluation Report (SER). Our review included two meetings with the staff and the applicant. We had the benefit of the documents referenced.

Conclusions and Recommendations

1. Entergy has properly identified the structures, systems, and components (SSCs) that are subject to aging management review consistent with the requirements of 10 CFR Part 54.

2. Aging mechanisms associated with passive, long-lived SSCs have been appropriately identified.

3. The programs instituted to manage aging-related degradation of the identified SSCs are appropriate and provide reasonable assurance that ANO-1 can be operated in accordance with its current licensing basis for the extended license term without undue risk to the health and safety of the public. The programs do not explicitly address the potential for circumferential cracking in control rod drive mechanism (CRDM) nozzle penetrations, such as has been observed at the Oconee Nuclear Plant, Unit 3. We expect that this current problem will be resolved and that the resolution will be incorporated into the current licensing basis and carried over into the license renewal period.

4. The staff has performed a comprehensive and thorough review of Entergy's application, and the open items identified in the January 2001 draft SER have been satisfactorily resolved .

5. The staff should determine whether modification of the current guidance in NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," is required to reflect the lessons learned from the ANO-1 application regarding aging management of small-bore piping and medium-voltage buried cable.

Background and Discussion

This report fulfills the requirement of 10 CFR 54.25 that the ACRS review and report on license renewal applications. Entergy requested renewal of the operating license for ANO-1 for a period of 20 years beyond the current license term, which expires on May 20, 2014. The final SER documents the results of the staff's review of information submitted by Entergy, including those commitments that were necessary to resolve open items identified by the staff in its January 2001 draft SER. The staff's review included verification of the completeness of the SSCs identified in the application, the validation of the integrated plant assessment process, the identification of the possible aging mechanisms associated with each passive long-lived component, and the adequacy of the aging management programs.

Our Subcommittee on Plant License Renewal met with the applicant and the staff on February 22, 2001, to review the SER with open items. The Subcommittee did not identify any issues to be addressed other than the six open items identified by the staff. This remarkably small number of open items is due, in large part, to the fact that the applicant implemented relevant lessons learned from the previous license renewal applications. In addition, the applicant structured the application using the standard application format and the guidance in Nuclear Energy Institute (NEI) Report 95-10, which facilitated the review. Because of the small number of open items and the scrutability of the application, we decided that there was no necessity to provide an interim report and have reviewed the SER on an accelerated basis.

The process implemented by the applicant to identify SSCs within the scope of the License Renewal Rule is effective. Reactor coolant system (RCS) components were identified using the generic Babcock & Wilcox Owners Group (BWOG) topical reports that address aging of RCS piping, pressurizer, reactor vessel, and reactor vessel internals. These topical reports, which have been approved by the staff, are applicable to ANO-1 and were used to support the license renewal application for Oconee. All other components in scope were determined on a plant-specific basis. At ANO-1, the safety-related SSCs included in the quality assurance program ("Q" list), as required by 10 CFR Part

50, Appendix B, are those that meet the definition of "safety related" in 10 CFR 54.4(a)(1). Furthermore, the majority of SSCs whose failure could prevent satisfactory accomplishment of any of the safety-related functions in 10 CFR 54.4(a)(1) are also classified as safety-related and included in the ANO-1 "Q" list. Therefore, the applicant was able to use the "Q" list to identify the bulk of the ANO-1 SSCs within the scope of the License Renewal Rule. This process has also resulted in the conservative inclusion of some SSCs that do not meet the criteria of 10 CFR 54.4(a)(2). We concur with the staff that the applicant has properly identified SSCs requiring an aging management review.

The applicant conducted a comprehensive aging management review of SSCs in scope. Aging effects of RCS components were identified using the aforementioned BWOG topical reports. Aging effects of all other SSCs were identified based on component material, operating environment, and operating stresses using plant-specific and industry-wide operating experience. Appendix B of the application describes the 22 existing or modified programs and the seven new programs implemented to manage aging during the period of extended operation.

ANO-1 has proposed a significantly smaller number of one-time inspections than did previous applicants. This is due, in part, to the fact that existing or modified ANO-1 programs manage aging effects that previous applicants do not manage during their current license terms. Consequently, previous applicants had to implement a larger number of one-time inspections to support license renewal. For example, aging of small-bore piping is managed at ANO-1 by a plant-specific risk-informed inspection program, and therefore, does not require a one-time inspection. We agree with the staff that the applicant has properly identified possible aging mechanisms associated with passive, long-lived SSCs and that the programs instituted to manage aging degradation of the identified SSCs are appropriate.

The ANO-1 application identifies cracking at welded joints of the CRDM pressure boundary as an aging effect to be managed. Appendix B of the application describes the aging management program instituted to deal with this aging degradation mechanism; i.e., "CRDM nozzle and other vessel closure penetration inspection program." This program identifies primary water stress corrosion cracking of Alloy-600 nozzles with partial penetration welds as the aging effect of concern and ties programmatic elements, such as the frequency of inspections, to the results of plant-specific and sister plant inspection findings. The initiatives included in this program are adequate to deal with this identified aging effect during the remaining portion of the current license term and during the period of extended operation. However, it is likely that the recent observations of stress corrosion cracking at the outer surface of CRDM nozzle penetrations may require some revisions to the program. We have noted

previously that aging management programs may have to be revised if it is found that new modes of degradation are occurring.

The ANO-1 application includes time limited aging analyses (TLAA) to evaluate the impact of neutron embrittlement on reactor vessel integrity. These analyses determine reactor vessel resistance to failure during pressurized thermal shock (PTS) events and the maintenance of acceptable Charpy upper-shelf energy levels. The TLAA used the methodology described in topical report BAW-2251A, "Demonstration of the Management of Aging Effects for the Reactor Vessel." This topical report was reviewed and approved by the staff and reviewed by the ACRS. Based on the composition of the limiting welds, Entergy projected that the ANO-1 reactor vessel will not reach the PTS and Charpy upper-shelf energy screening limits until well after 60 years of operation. The ANO-1 reactor vessel integrity program will be utilized to ensure that the time-dependent parameters used in the TLAA evaluations are tracked so that the TLAA remain valid during the license renewal period.

Entergy committed to implementing a plant-specific program to manage the effects of fatigue. Using the correlations published in NUREG/CR-5704, Entergy has found that the surge line, the high pressure injection/makeup nozzles, and safe ends may reach the limits of acceptable fatigue during the period of extended operation. To address this condition, Entergy has proposed a program that will include one or more of the following options: refinement of the fatigue analyses, repair, replacement, or management of fatigue effects using a program that will be reviewed and approved by the staff. We concur with the staff that Entergy's proposed program is an acceptable plant-specific approach for resolving the concerns of Generic Safety Issue-190, "Fatigue Evaluation of Metal Components for 60 Year Plant Life."

ANO-1 region 1 spent fuel storage racks currently use Boraflex as a neutron absorber. Aging of Boraflex was identified in the application as a time limited aging analysis. During the staff's review of the ANO-1 application, Entergy informed the staff that Boraflex had been found to degrade more rapidly than previously expected, and was not expected to last through the current 40-year licensing term. Therefore, a corrective action plan for the remainder of the 60-year operating term would be identified and committed to before the end of 2002. In Open Item 4.7.2-1 associated with Boraflex degradation, the staff requested that Entergy continue to recognize aging of Boraflex as a time limited aging analysis and provide details on the required monitoring program. Entergy has now provided the requested programmatic details. We concur with the staff that either the implementation of a permanent solution during the current licensing period or the Boraflex monitoring program provided by Entergy and described in the SER provides acceptable management of Boraflex degradation during the period of extended operation.

The staff has performed a comprehensive and thorough review of Entergy's application. The applicant and the staff have identified possible aging mechanisms associated with passive long-lived components. Adequate programs have been established to manage the effects of aging so that ANO-1 can be operated safely in accordance with its current licensing basis for the extended license term.

The review of the ANO-1 application has provided significant new information on small-bore piping and medium-voltage buried cable aging degradation and related management programs. As described above, ANO-1 has implemented a small-bore piping inspection program because it has identified small-bore piping in safety-significant locations that is susceptible to aging degradation. The staff should determine whether current guidance in the GALL report needs to be modified to reflect this experience. Also, ANO-1 has implemented a medium-voltage buried cable aging management program that includes the options of cable testing or periodic replacement of buried cables. ANO-1 has included the replacement option because it has found that in a number of instances testing was not effective in identifying cable degradation. The staff needs to evaluate the adequacy of testing of buried cables and provide appropriate guidance in the next update of the GALL report.

Dr. William J. Shack did not participate in the Committee's deliberations on aging-induced degradation.

Sincerely,

George E. Apostolakis
Chairman

References:

1. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the License Renewal of Arkansas Nuclear One, Unit 1," dated April 2001.
2. Letter dated January 31, 2000, from C. R. Hutchinson to the U.S. Nuclear Regulatory Commission, Subject: Arkansas Nuclear One, Unit 1, License Renewal Application.
3. Letter dated March 14, 2001, from J. D. Vandergrift to the U.S. Nuclear Regulatory Commission, Subject: Arkansas Nuclear One, Unit 1, License Renewal Safety Evaluation Report Open Item Responses.

4. Babcock and Wilcox Owners Group Generic License Renewal Program Topical Report, BAW-2251A, "Demonstration of the Management of Aging Effects for the Reactor Vessel," dated June 1996.
5. U. S. Nuclear Regulatory Commission, NUREG/CR-5704, "Effects of LWR Coolant Environment on Fatigue Design Curves of Austenitic Steels," dated April 1999.
6. U. S. Nuclear Regulatory Commission, Generic Safety Issue - 190, "Fatigue Evaluation of Metal Components for 60-Year Plant Life."