COMMISSION MEETING SLIDES/EXHIBITS

MEETING WITH ACRS

FRIDAY, MAY 11, 2001

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

May 3, 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, MAY 11, 2001—SCHEDULE AND BACKGROUND INFORMATION

The ACRS is scheduled to meet with the NRC Commissioners between 10:30 a.m. – 12:30 p.m. on Friday, May 11, 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 Dr. George Apostolakis		
1.	Proposed Framework for Risk Informed Changes to 10 CFR Part 50 Dr. William Shack	10 minutes
2.	South Texas Project (STP) Exemption Request Mr. John Sieber	10 minutes
3.	Thermal-Hydraulic Codes Dr. Graham Wallis	10 minutes
4.	Status of Steam Generator Issues Dr. Dana Powers	10 minutes

		ESTIMATED TIME
5.	Status of ACRS Activities on License Renewal Dr. Mario 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.

Attachments: As stated

cc: ACRS Members
ACRS Staff

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS MEETING WITH THE U.S. NUCLEAR REGULATORY COMMISSION

DR. GEORGE E. APOSTOLAKIS
ACRS

MAY 11, 2001

OVERVIEW OF TOPICS AND NEAR-TERM ACTIVITIES

- Proposed Framework for Risk Informed Changes To 10 CFR Part 50
- South Texas (STP) Exemption Request
- Thermal-Hydraulic Codes
- Status of Steam Generator Issues
- Status of ACRS Activities on License Renewal

PROPOSED FRAMEWORK FOR RISK INFORMED CHANGES TO 10 CFR PART 50

DR. WILLIAM J. SHACK
ACRS

MAY 11, 2001

ACRS report dated November 20, 2000, concerning proposed Option 3 framework document (SECY-00-0198)

- **Staff identified elements important to prioritization of candidate regulations.**
- **Good Start. Improvements will be made as experience is gained.**
- **ACRS considerations focused on the treatment of defense in depth.**

DEFENSE IN DEPTH

- **Defense in depth based on prevention and mitigation:**
  - **Limit the frequency of accident initiating events, limit probability of core damage, given initiation**
  - **For mitigation limit radionuclide releases, limit public health effects**

DEFENSE IN DEPTH

- **Framework is consistent with our suggestion that, where possible, the need for defense-in-depth measures (safety margins, redundancy, and diversity) be assessed quantitatively through probabilistic risk assessment (PRA).**

- **Framework treats safety margins in terms of probabilities which permits quantification of the contribution of margins to meeting the risk goals.**

DEFENSE IN DEPTH

- **Framework should clearly state that defense in depth measures should not be imposed at lower tiers except when there are significant uncertainties**

DEFENSE IN DEPTH

- **ACRS Subcommittee meeting held on March 16, 2001, to discuss risk-informing 10 CFR 50.46 concerning emergency core cooling systems.**

- **Industry proposed to use leak-before-break and probabilistic fracture mechanics to define new large-break loss-of-coolant accident and demonstrate low probability of current double-ended guillotine pipe break and other large breaks.**

DEFENSE IN DEPTH

- **Although staff has accepted such arguments in context of dynamic loads, it argues that compliance with 10 CFR 50.46 is more complex and will require more rigorous assessment of uncertainties associated with leak-before-break.**

FUTURE ACRS ACTIVITIES

- Subcommittee meeting in late May 2001 to review results of staff's 10 CFR 50.46 feasibility study

- Full Committee review and report during June 6-8, 2001 ACRS meeting.

South Texas Project (STP) Exemption Request Option 2

John D. Sieber
ACRS
May 11, 2001

STP Exemption Request

What is involved?

STP has requested exemption from 11 NRC Regulations regarding the treatment of non-risk significant components in their facilities.

STP

Plant Description

- **Two units, 4 loop Westinghouse (W) PWRs, rated at 1250 MW each.**
- **Commercial Operation -1988.**
- **Three safety trains.**
- **Large, dry containments.**

Purpose

- **Identify components that are important to safety, from a risk standpoint, and eliminate components not important to safety from Special Treatment Requirements, including 10CFR50, App B.**
- **Identify non-"Q"* components that are "risk significant."**

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*Nuclear safety grade

Important Processes

✓Categorization

✓Treatment

STP Components

(for a 2 Unit Plant)

Total number of components in 29 safety systems.	43,690
Total number of components classified as "Q"* components	16,715
Total number of components identified in a typical PRA analysis (2 Units).	2,400

*Nuclear safety grade

STP Categorization Process

Two Methods Were Used

- **Recategorization based on plant specific PRAs and "Expert Panel."**

  2,400 Components 5.7 percent

- **Recategorization based on "Expert Panel."**

  41,290 Components 94.3 percent

STP Categorization Process
Results

1. Safety Related, Risk Significant Components 3,810 (8.7%)	2. Non-Safety Related, Risk Significant 372 (0.9%)
3. Safety Related, Non-risk Significant 12,905 (29.5%)	4. Non-Safety Related, Non-Risk Significant 26,603 (60.9%)

Important Elements

- ✓ **Robust PRA**
- ✓ **Sensitivity Studies**
- ✓ **Documented Treatment Processes**

The Future

- ✓ The staff and the licensee must complete the documentation of the treatment methodology for category-3 items.
- ✓ The staff must complete the revision of the draft safety evaluation report.
- ✓ The staff and the licensee must come to a resolution of remaining open items.
- ✓ The ACRS will write a letter after the first three items are complete

THERMAL-HYDRAULIC CODES

DR. GRAHAM B. WALLIS
ACRS

May 11, 2001

Thermal-Hydraulic Code Issues

- Codes have proven adequate to satisfy regulatory requirements when used with appropriate conservatism and judgment and when extensively examined by the staff.
- Use of codes for "realistic" or "best-estimate" analyses requires improved documentation and definitive criteria for uncertainty assessment.

Thermal-Hydraulic Code Issues

- ACRS supports NRC staff obtaining and exercising applicants' thermal hydraulic codes
  - NRC and Westinghouse have yet to agree on this matter for AP1000 pre-application review

Impact of Codes on NRC Performance Goals

- ***Maintain safety***
  - **Inadequate assessment of code uncertainties may compromise safety**

- ***Increase public confidence***
  - **Code uncertainties, errors and assumptions decrease public confidence**

Impact of Codes on NRC Performance Goals

- ***Increase efficiency and effectiveness***
  - Poor documentation and validation extend review and may require additional experiments
- ***Reduce unnecessary burden***
  - Unquantified uncertainties require conservative decisionmaking; the margins are unspecified

Review of Siemens S-RELAP5 Code to Appendix K Small-Break Loss-of-Coolant Accident Analysis

- Code is adequate for this application

- Code documentation must be improved for realistic applications

Review of EPRI RETRAN-3D Code

- **Concerns with momentum equations identified by Thermal-Hydraulic Phenomena Subcommittee**
- **Awaiting EPRI's response**

NRC Staff Activities

- Resolving comments on documents for development and review of codes:
  - Draft Regulatory Guide DG-1096
  - Standard Review Plan Section 15.0.2
  - Office of Nuclear Regulatory Research Public Workshop Held April 9, 2001
- Office of Nuclear Regulatory Research completing development of consolidated thermal hydraulic code

STATUS OF STEAM GENERATOR ISSUES

DR. DANA A. POWERS
ACRS

May 11, 2001

STEAM GENERATOR ISSUES

- **Known vulnerability of PWRs**
  - — **design basis accident analysis**
    - **steam generator tube rupture**
    - **leakage during main steamline break**
  - — **severe accident analysis/risk assessment**
    - **bypass accident sequences**
    - **risk dominant, not frequency dominant**

PLANT	DATE	CAUSE
Point Beach 1	Feb. 26, 1975	Wastage
Surry 2	Sept. 15, 1976	PWSCC
Doel 2	June 25, 1975	PWSCC
Prairie Island 1	Oct. 2, 1979	Loose parts
Ginna 1	Jan. 25, 1982	Loose parts
Fort Calhoun	May 16, 1984	ODSCC
North Anna 1	July 15, 1984	Fatigue
McGuire 1	March 7, 1989	ODSCC
Mihama	Feb. 9, 1991	Fatigue
Palo Verde 2	March 14, 1993	ODSCC
Indian Point 2	Feb. 15, 2000	PWScc
Next?		

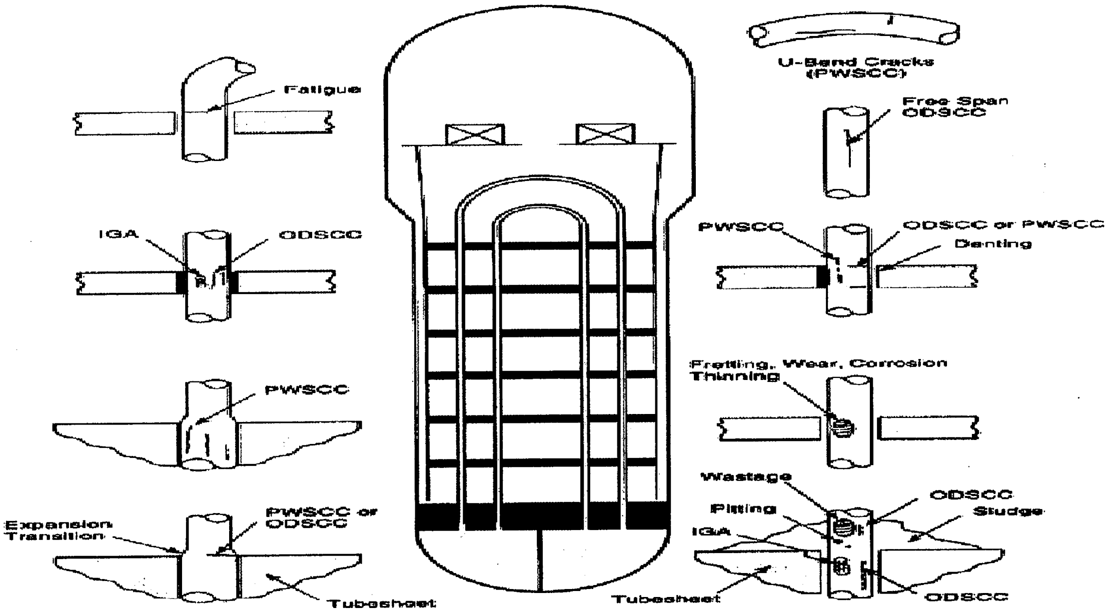
PWSCC= primary water stress corrosion cracking
ODSCC= outside diameter stress corrosion cracking

STEAM GENERATOR ISSUES

- **Continue to receive attention**
  - **by Industry**
    - **"Steam Generator Program Guidelines"**
    - **replacement generators**
    - **condition monitoring**
  - **by NRC**
    - **NRR Action Plan**
    - **research program**

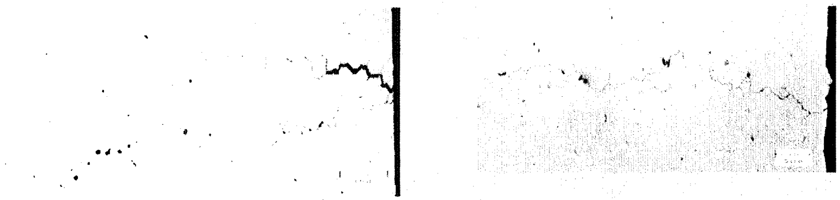
Steam Generator Tube Corrosion

Figure 1. Schematic Diagram of a Steam Generator, and Types of Corrosion and Degradation Observed at Various Locations

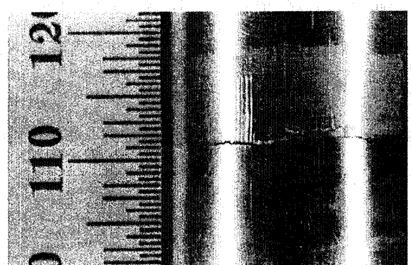


Stress Corrosion Cracks

a. Examples of a predominantly axial, intergranular stress corrosion cracks. The fiducial marks in these photographs are 0.1 mm long.



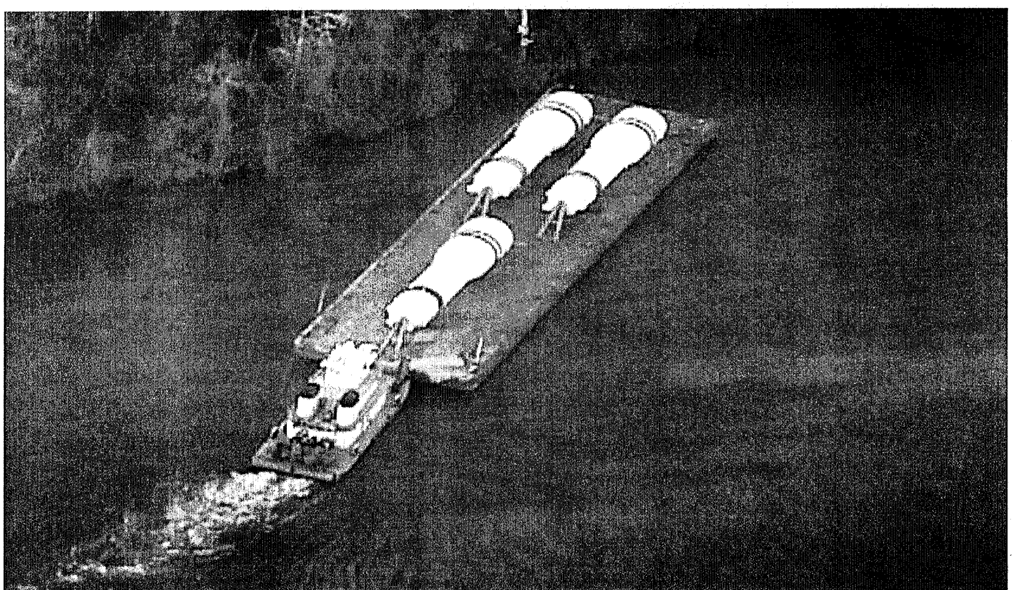
b. Example of circumferential cracks.



- **Steam Generator Tube Corrosion**
  - prompting steam generator replacements
    - alloy 690 is not immune to stress corrosion cracking
    - caution in approving requests to extend inspection intervals

STEAM GENERATOR REPLACEMENT

STEAM GENERATOR REPLACEMENT



- **Steam Generator Tube Corrosion**
  - **prompting steam generator replacements**
    - **alloy 690 is not immune to stress corrosion cracking**
    - **caution in approving requests to extend inspection intervals**

- **Alternative repair criteria**
  - **type of corrosion changing from wastage to stress corrosion cracking**
    - **inside the tube (PWSCC)**
    - **outside the tube (ODSCC)**
      - **in U bends**
      - **in free span**
      - **within tube sheets and tube support plate**

- **Alternative repair criteria**
  - adequate technology for crack detection
  - sizing cracks challenging
    - voltage criteria for repairing or removing tubes from service with flaws in the region of the tube support plates
    - condition monitoring program

- **Needs for Technical Basis of the Alternative Repair Criteria**
  - forces and effects on tubes of depressurization in a main steamline break
  - database for 7/8" tubes
  - monitoring for systematic deviations from linear, bounding crack growth rates
  - understanding of iodine spiking

- Can degraded tubes fail ...
  - as a result of blowdown forces during a main steamline break?
    - potential generic issue for research
  - as a result of heat and pressure loads during severe accidents?
    - converts all pressurized accidents into bypass accidents

- **Need better understanding of induced steam generator tube rupture**
  - **licensees are requesting reduced inspection requirements**
  - **requests and reviews are being cast in risk-informed language**

- **Need better understanding of licensee analysis methods such as the MAAP code**

- **Mechanistic understanding of stress corrosion cracking and prediction of leakage as well as burst of degraded tubes?**
  - **not any time soon**
  - **will require better data on cracks**
  - **inspection and empirical prognostication will be the prevailing approach for some time**

STATUS OF ACRS ACTIVITIES ON LICENSE RENEWAL

DR. MARIO V. BONACA
ACRS
MAY 11, 2001

LICENSE RENEWAL GUIDANCE DOCUMENTS

- ACRS reviewed proposed final generic License Renewal Guidance Documents:
  - Standard Review Plan
  - Regulatory Guide 1.188, Standard Format and Content
  - Generic Aging Lessons Learned (GALL) Report
  - Nuclear Energy Institute (NEI) 95-10, Industry Implementation Document

CONCLUSIONS

- ACRS recommended approval of these documents
- The staff has developed an effective set of guidance documents.
- Inclusion of the results of the scoping process in applications should be encouraged. These results facilitate the review process and make the license renewal applications more understandable.

CONCLUSIONS (Continued)

- The staff will update the GALL Report periodically.

- The SRP and Regulatory Guide 1.188 should be updated to make them consistent with the updated GALL Report.

REVIEW OF LICENSE RENEWAL APPLICATIONS

- The Subcommittee on Plant License Renewal reviewed the License Renewal Application for Arkansas Nuclear One, Unit 1 and the staff's interim safety evaluation report on February 22, 2001. The Sub-committee did not identify any significant issues.

INTERIM REVIEW OF THE HATCH APPLICATION

- ACRS reviewed the Hatch license renewal application and provided its interim report on April 13, 2001.
  - Review included selected Boiling Water Reactor Vessel and Internals Project (BWRVIP) Topical Reports referenced in the Hatch application.

CONCLUSIONS ON HATCH APPLICATION

- **The staff review of the Hatch license renewal application was extensive and thorough.**
- **Applicant has implemented adequate processes to identify structures and components subject to aging management review.**
- **Guidelines of the BWRVIP effectively support license renewal.**

FUTURE ACRS ACTIVITIES ON LICENSE RENEWAL APPLICATIONS

- At the staff's request, the ACRS will complete its review of the Arkansas Nuclear One, Unit 1 License Renewal Application and the staff's safety evaluation report on May 10-11, 2001: five months ahead of schedule.

FUTURE ACRS ACTIVITIES ON LICENSE RENEWAL (Continued)

- ACRS will perform an interim review of the Turkey Point License Renewal Application in October 2001.
- Final review of the Hatch application is scheduled for November 2001.
- The ACRS plans to provide views in July 2001 on the need to revise the License Renewal Rule (10 CFR Part 54).

FUTURE ACRS ACTIVITIES ON LICENSE RENEWAL (Continued)

- ACRS will form two License Renewal Subcommittees to handle the expected workload and divide the review of applications starting in 2002.