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

October 10, 2001

MEMORANDUM TO: ACRS Members

Noel Dudley

FROM: Noel Dudley, Senior Staff Engineer
ACRS\ACNW

SUBJECT: CERTIFICATION OF THE MINUTES OF THE ACRS SUBCOMMITTEE MEETING ON MATERIALS AND METALLURGY CONCERNING THE STEAM GENERATOR ACTION PLAN, SEPTEMBER 26, 2001-ROCKVILLE, MARYLAND

The minutes of the subject meeting, issued on October 4, 2001, have been certified as the official record of the proceedings of that meeting. A copy of the certified minutes is attached.

Attachment: As stated

cc: Technical Support Branch
Operations Support Branch (3 copies)

cc via e-mail:
J. Larkins
S. Bahadur
ACRS Fellows and Technical Staff
E. Barnard

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

MEMORANDUM TO: Noel Dudley, Senior Staff Engineer
ACRS/ACNW

FROM: Dr. F. Peter Ford, Chairman
Materials and Metallurgy Subcommittee

SUBJECT: CERTIFICATION OF THE MINUTES OF THE ACRS MATERIALS AND METALLURGY SUBCOMMITTEE MEETING CONCERNING THE STEAM GENERATOR ACTION PLAN, SEPTEMBER 26, 2001-ROCKVILLE, MARYLAND

I hereby certify that, to the best of my knowledge and belief, the minutes of the subject meeting issued on October 4, 2001, are an accurate record of the proceedings for the meeting.

Dr. F. Peter Ford, Chairman
Materials and Metallurgy Subcommittee

October 5th, 2001
Date

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

October 4, 2001

MEMORANDUM TO: Dr. F. Peter Ford, Chairman
Materials and Metallurgy Subcommittee

FROM: Noel Dudley, Senior Staff Engineer
ACRS/ACNW

SUBJECT: WORKING COPY OF THE MINUTES OF THE ACRS MATERIALS AND METALLURGY SUBCOMMITTEE MEETING CONCERNING THE STEAM GENERATOR ACTION PLAN, SEPTEMBER 26, 2001 - ROCKVILLE, MARYLAND

A working copy of the minutes for the subject meeting is attached for your review. I would appreciate your review and comment as soon as possible. Copies are being sent to the Materials and Metallurgy Subcommittee members for information and/or review.

Attachment: As stated

cc: W. Shack
M. Bonaca
T. Kress
J. Sieber
D. Powers

cc via E-Mail:
J. Larkins
S. Bahadur

Issued: 10/4/01
Certified: 10/5/01

CERTIFIED

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
MINUTES OF SUBCOMMITTEE MEETING ON
MATERIALS AND METALLURGY
THE STEAM GENERATOR ACTION PLAN
SEPTEMBER 26, 2001
ROCKVILLE, MARYLAND

The ACRS Subcommittee on Materials and Metallurgy met on September 26, 2001, to hold discussions with the NRC staff concerning the Steam Generator Action Plan, including issues related to the differing professional opinion and South Texas Unit 2 steam generator leakage issues. The entire meeting was open to public attendance. Mr. Noel Dudley was the cognizant ACRS staff engineer for this meeting. The meeting was convened at 8:30 a.m. and adjourned at 12:30 p.m.

ATTENDEES

ACRS

P. Ford, Chairman
W. Shack, Vice Chairman
M. Bonaca, Member
T. Kress, Member
J. Sieber, Member
D. Powers, Member
N. Dudley, ACRS Staff

NRC REPRESENTATIVES
W. Bateman, NRR
E. Sullivan, NRR
E. Murphy, NRR
M. Banerjee, NRR
C. Boyd, RES
K. Karwoski, RES
J. Muscara, RES
C. Tinkler, RES
S. Bajorek, RES

NUCLEAR ENERGY INSTITUTE
James Riley, NEI

There were no written comments or requests for time to make oral statements received from members of the public. No members of the public attended the meeting. Approximately 4 members of the public attended the meeting. A list of meeting attendees is available in the ACRS office files.

INTRODUCTION

Dr. F. Peter Ford, Chairman of the Materials and Metallurgy Subcommittee, stated that the purpose of the meeting was to gather information regarding the status of the Steam Generator Action Plan and the South Texas, Unit 2, steam generator tube leakage issue. Dr. Ford noted that the staff issued the Steam Generator Action plan on November 16, 2000 and updated the Plan on May 11, 2001 to include items associated with the differing professional opinion (DPO) associated with steam generator tube integrity. He called upon Mr. Edmund Sullivan, Jr., Office of Nuclear Reactor Regulation (NRR), to begin.

STATUS OF THE STEAM GENERATOR ACTION PLAN

Mr. Edmund Sullivan, Jr., NRR, introduced the staff's presentation. Ms. Maitri Banerjee, NRR, presented an overview of the recent history of the Steam Generator Action Plan. She described the purpose, scope, and status of the Plan. Ms. Banerjee identified the following items as Steam Generator Action Plan significant activities:

- issue a regulatory issues summary on the lessons learned;
- provide guidance for NRC inspectors;
- consider steam generator performance indicators;
- provide guidance for reviewing steam generator tube inspection results;
- hold a steam generator workshop with stakeholders;
- provide guidance for license amendment reviews;
- prepare a safety evaluation for the latest revision of NEI 97-06;
- develop milestones for activities addressing ACRS recommendations on DPO issues;
- develop milestones for resolving GSI 163, GSI 188, and DG-1073; and
- establish a Steam Generator Action Plan web page.

Ms. Banerjee described how the resolution of the Steam Generator Action Plan items would be managed.

Mr. Sullivan provided background information concerning the staff's initiative to revise the regulatory framework for steam generator tube inspections and repairs. He presented an overview of the development of the Nuclear Energy Institute's (NEI) steam generator generic change package, which includes revised technical specifications and new administrative technical specifications. Mr. Sullivan explained that licensees are developing steam generator inspection programs in accordance with NEI 97-06, "Steam Generator Program Guidelines." He noted that NEI 97-06 references sub-tier Electric Power Research Institute (EPRI) guideline documents, which the staff had not intended to review.

Mr. Sullivan explained that in response to the NEI's proposal to extend the interval for steam generator tube inspections to 22 EFPY, the staff reviewed portions of the EPRI guidelines that are critical to the effectiveness of condition monitoring. During its review, the staff identified a number of issues related to industry practices. Mr. Sullivan concluded that the staff can proceed with its review of the generic change package provided appropriate regulatory restrictions are maintained on the length of the inspection interval. He outlined the milestones necessary for the staff to complete its review of the generic change package.

The Subcommittee members and the staff discussed why the staff review of NEI 97-06 was put on hold and why the target dates were slipping.

Dr. Powers, ACRS, questioned why the staff decided not to include a performance indicator for steam generator integrity, given that steam generator tube ruptures are risk dominate at most plants. The staff explained that three performance indicators were considered. The staff rejected the proposed performance indicators associated with tube degradation assessments and condition monitoring since these indicators would not provide early indication of loss of tube integrity. The staff rejected the proposed performance indicator for primary to secondary leakage since it was a continuously monitored parameter that had no correlation to risk.

Mr. Sieber, ACRS, stated that steam generator tube integrity is a measurable quantity and that exceeding the performance criteria will not result in an off-site release. Mr. Ernmett Murphy, NRR, replied that exceeding the structural integrity performance criteria could result in exceeding a risk thresholds, which would be unacceptable.

The Subcommittee members and the staff discussed the following aspects of the administrative technical specifications:

- establishing performance criteria,
- identifying defects,
- determining crack growth rate,
- staff control over licensees' selection of tubes to be pulled, and
- required actions when the condition monitoring results exceed the operational assessment predictions.

The Subcommittee members and the staff discussed long term protocols, such as, the basis for extending the inspection frequency for new materials like 690 stainless steel and the inability of some licensees to incorporate industry experience into their programs. They also discussed when the staff's review of NEI 97-06 would be provided to the ACRS for its review.

SOUTH TEXAS UNIT 2 STEAM GENERATORS

Mr. Kenneth Karwoski , NRR, described the steam generator design and operating experience at South Texas, Unit 2. He summarized the implementation of the voltage-based repair criteria at South Texas Unit 2 and described the license's actions in response to steam generator tube operation leakage. Mr. Karwoski presented a comparison of the number of predicted and observed indications of tube defects over the last three operating cycles. He concluded by explaining the staff's activities and noting that the steam generators are scheduled for replacement in December 2002.

The Subcommittee members and Mr. Karwoski discussed the difference between carbon and stainless steel tube support plates, the reliability of burst technology, the use of the 3 volt alternate repair criteria, the probability of detection of tube defects, and use of average growth rates in the operational assessment.

RESPONSE TO ACRS RECOMMENDATIONS

Mr. Karwoski addressed the ACRS recommendation that the 7/8" tube leakage database needs to be greatly improved to be useful. He agreed with the ACRS recommendation and noted that licensees had committed to periodically remove tubes for destructive examination. He conceded, however, that there were no regulatory requirements to force licensees to remove additional tubes beyond their present commitments.

In response to Subcommittee members' questions, Mr. Karwoski explained that the voltage-based correlation for 7/8" tubes is not improving as additional data is added the database. He speculated that if the correlation did not improve the staff may have to conclude that the probability of tube leakage is independent of voltage readings.

Mr. Karwoski addressed the ACRS recommendation that the staff establish a program to monitor the prediction of flaw growth for systematic deviations from expectations. He explained that the staff will continue to review licensees' 90 day reports and is formalizing its review of inspection summary reports.

Dr. Joseph Muscara, RES, addressed the ACRS recommendation concerning crack propagation in steam generator tubes from pressure and main steamline-break loads. He described the planned calculations and analyses, which are intended to estimate what loads would be required to propagated a range of axial and circumferential cracks. He explained that the staff would conduct tests of degraded tubes under pressure to validate the analytical results. In response to questions, Dr. Muscara explained that the test results would be used to validate and not develop the calculational model.

Dr. Muscara addressed the ACRS recommendation for evaluating damage progression via jet cutting. He presented initial test results that indicated low erosion rates. Dr. Muscara indicated that the testing would be completed and draft reports would be available at the end of December 2001.

Dr. Muscara addressed the ACRS recommendation concerning use of a constant probability of detection. He presented the result of round robin tests at the Argonne National Laboratory mockup. Complete analysis and documented research results from the round robin will be available at the end of December 2001.

Dr. Ford, ACRS, questioned how the staff correlated crack growth rates that are linear to those that are non-linear over time. He also stated that voltage and crack growth are not related. Dr. Muscara agreed that there are problems associated with measuring crack growth rates. He noted that better measurement techniques are being developed in the laboratory but that it will be years before these techniques are available for use in the field.

Dr. Muscara addressed the ACRS recommendation concerning developing a better understanding of stress corrosion cracking. He presented plans to conduct tests and described the use of these test results and operating experience to develop models for predicting cracking behavior of steam generator tubes in operating environments.

Mr. Charles Tinkler, RES, addressed the ACRS recommendation concerning the development of a better understanding of the behavior of degraded steam generator tubes under severe accident conditions. He identified the analytic codes and data that the staff plans to use in its research efforts. He listed the severe accident issues to be analyzed and indicated whether the issues would be evaluated using analyses, test results, or both.

Mr. Stephen M. Bajorek, RES, addressed the DPO contentions concerning the effect of blowdown forces caused by the depressurization of the reactor coolant system during a main steamline break and the affect of tube support plate lift during a steam generator depressurization. He presented the background and evaluation plan associated with these concerns.

Mr. Christopher Boyd presented his analysis of steam generator inlet plenum mixing using computational fluid dynamics (CFD). He summarized the background and advantages of CFD

and showed qualitative and quantitative results of inlet mixing predicted by CFD. Mr. Boyd explained that the CDF predictions of parameters are generally within experimental uncertainty of experimental data. He concluded by summarizing the staff's plans for further analysis of mixing in the inlet plenum during main steamline breaks.

SUBCOMMITTEE COMMENTS, CONCERNS, AND RECOMMENDATIONS

None.

STAFF AND INDUSTRY COMMITMENTS

None.

SUBCOMMITTEE DECISIONS

The Subcommittee requested that the staff present at the October 4, 2001 ACRS meeting, an abbreviated version of its presentation concerning research programs that address ACRS recommendations related to DPO issues.

FOLLOW-UP ACTIONS

The Subcommittee requested the opportunity to review and comment on the staff's assessment of the revised NEI 97-06 when it becomes available.

PRESENTATION SLIDES AND HANDOUTS PROVIDED DURING THE MEETING

The presentation slides and handouts used during the meeting are available in the ACRS office files or as attachments to the transcript.

BACKGROUND MATERIAL PROVIDED TO THE SUBCOMMITTEE:

1. NRR Director's Quarterly Status Report on Generic Activities, Action Plans, and Generic Communication and Compliance Activities, dated April 5, 2001, "Steam Generators," pp. 5-23.
2. Memorandum dated May 11, 2001, from Samuel Collins, Director NRR, and Ashok Thadani, Director RES, to William D. Travers, Executive Director for Operations, NRC, Subject: Steam Generator Action Plan Revision to Address Differing Professional Opinion on Steam Generator Tube Integrity.
3. Memorandum dated March 23, 2001, from John A. Zwolinski, NRR, et. al., to Brian Sheron and R. William Borchardt, NRR, Subject: Steam Generator Action Plan Revision and Completion of Items Nos. 1.1, 1.2, 1.3, 1.4, 1.7, 1.8, 1.15, 2.1 and 2.2.
4. Memorandum dated November 16, 2000, from Brian Sheron and Jon Johnson, NRR, to Samuel Collins, Director NRR, Subject: Steam Generator Action Plan.
5. Memorandum from dated April 30, 2001, from Jack Strosnider, NRR, to Brian Sheron and Jon Johnson, NRR, Subject: Steam Generator Action Plan Item 1.11A – Review and Revise the Baseline Inspection Program Related to Steam Generator Inspections.
6. Memorandum from dated April 30, 2001, from Jack Strosnider, NRR, to Brian Sheron and Jon Johnson, NRR, Subject: Steam Generator Review Guidance (milestones 1.10 and 1.12).

7. Memorandum dated April 12, 2001, from John A. Zwolinski, NRR, et. al., to Brian Sheron and R. William Borchardt, NRR, Subject: Steam Generator Action Plan Item No. 2.3.
8. Memorandum dated April 3, 2001, from Bruce Boger, NRR, to Brian Sheron and R. William Borchardt, NRR, Subject: Steam Generator Action Plan Revision and Completion of Item Nos. 2.4 and 2.5.
9. Memorandum dated March 30, 2001 from Jack Strosnider, NRR, to Brian Sheron and R. William Borchardt, NRR, Subject: Steam Generator Action Plan Item 1.9 – Guidance to Inspectors Overseeing Facilities with Known Steam Generator Tube Leakage.
10. Memorandum dated June 1, 2001, from William D. Travers, Executive Director for Operations, to Chairman Meserve, NRC, Subject: Differing Professional Opinion on Steam Generator Tube Integrity Issues.
11. Memorandum dated June 24, 2001, from J. Hopenfeld, RES, to the Commissioners, Subject: Differing Professional Opinion Steam Tube Integrity Issues.
12. Memorandum dated March 5, 2001, from William D. Travers, Executive Director for Operations, to Joram Hopenfeld, RES, Subject: Differing Professional Opinion on Steam Generator Tube Integrity Issues.
13. Memorandum dated March 5, 2001, from William D. Travers, Executive Director for Operations, to Samuel J. Collins, NRR, and Ashok C. Thadani, RES, Subject: Differing Professional opinion on Steam Generator Tube Integrity.
14. Letter dated February 1, 2001, from D.A. Powers, Chairman, ACRS Ad Hoc Subcommittee, to William D. Travers, Executive Director, Subject: Differing Professional Opinion on Steam Generator Tube Integrity.
15. Letter dated June 28, 2001, from Mark E. Kanavos, South Texas Project Nuclear Operating Company, to Document Control Desk, NRC, Subject: Steam Generator Tube Voltage-Based Repair Criteria 90-Day Report.
16. Letter dated June 7, 2001, from T. J. Jordan, South Texas Project Nuclear Operating Company, to U.S. Nuclear Regulatory Commission Document Control Desk, Subject: Steam Generator Tube Burst and Accident Leakage Information Requested by NRC.
17. South Texas Project, Unit 2, Slides presented at the NRC Meeting April 19, 2001.
18. Meeting Summary dated May 15, 2001, Subject: Summary of Meeting with STPNOC/Westinghouse Regarding Results of Steam Generator Tube Inspections and In Situ Tube Pressure Tests Conducted During End-of-Cycle 8 Refueling Outage for South Texas Project Unit 2.

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NOTE: Additional details of this meeting can be obtained from a transcript of this meeting available in the NRC Public Document Room, One White Flint North, 11555 Rockville Pike, Rockville, MD, (301) 415-7000, downloading or viewing on the Internet at "http://www.nrc.gov/ACRSACNW," or can be purchased from Neal R. Gross and Co., 1323 Rhode Island Avenue, NW, Washington, D.C. 20005, (202) 234-4433 (Voice), 387-7330 (Fax), e-mail: nrgross@nealgross.com.

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
MATERIALS AND METALLURGY SUBCOMMITTEE
STEAM GENERATOR ACTION PLAN
SEPTEMBER 26, 2001
ROCKVILLE, MARYLAND

- PROPOSED AGENDA -

	TOPIC	PRESENTER	TIME	
I.	Opening Remarks	P. Ford, ACRS	5 min.	8:30-8:35 a.m.
II.	Introductory Remarks	E. Sullivan, NRR	5 min.	8:35-8:40 a.m.
III.	Status of Steam Generator Action Plan A. Background B. Current Status C. NEI 97-06	NRR Staff	80 min.	8:40-10:00 a.m.
	BREAK		15 min.	10:00-10:15 a.m.
IV.	Status of Action Plan DPO Issues	RES Staff	75 min.	10:15 - 12:30 p.m. ~~10:15-11:30 a.m.~~
V.	Overview of South Texas Steam Generator Tube Integrity Issues	NRR Staff	30 min.	10:15 - 10:55 a.m. ~~11:30-12:00 noon~~
VI.	Adjournment	P. Ford, ACRS		12:30 p.m. ~~12:00 noon~~

NOTE:

Presentation time should not exceed 50 percent of the total time allotted for specific item. The remaining 50 percent of the time is reserved for discussion.

Number of copies of the presentation materials to be provided to the ACRS - 25

NRD

6888) between 7:00 a.m. and 3:45 p.m. (EDT). Persons planning to attend this meeting are urged to contact the above named individual one or two working days prior to the meeting to be advised of any potential changes to the agenda, etc., that may have occurred.

Dated: August 30, 2001.

Sher Bahadur,

Associate Director for Technical Support, ACRS/ACNW.

[FR Doc. 01–22763 Filed 9–10–01; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards Meeting of the ACRS Subcommittee on Materials and Metallurgy; Notice of Meeting

The ACRS Subcommittee on Materials and Metallurgy will hold a meeting on September 26, 2001, Room T–2B3, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance.

The agenda for the subject meeting shall be as follows: *Wednesday, September 26, 2001—8:30 a.m. until the conclusion of business*

The Subcommittee will discuss the status of the Steam Generator Action Plan and the South Texas Project, Unit 2, tube integrity issues. The purpose of this meeting is to gather information, analyze relevant issues and facts, and to formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Oral statements may be presented by members of the public with the concurrence of the Subcommittee Chairman; written statements will be accepted and made available to the Committee. Electronic recordings will be permitted only during those portions of the meeting that are open to the public, and questions may be asked only by members of the Subcommittee, its consultants, and staff. Persons desiring to make oral statements should notify the cognizant ACRS staff engineer named below five days prior to the meeting, if possible, so that appropriate arrangements can be made.

During the initial portion of the meeting, the Subcommittee, along with any of its consultants who may be present, may exchange preliminary views regarding matters to be considered during the balance of the meeting.

The Subcommittee will then hear presentations by and hold discussions with representatives of the NRC staff and other interested persons regarding this review.

Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, and the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor, can be obtained by contacting the cognizant ACRS staff engineer, Mr. Noel F. Dudley (telephone 301/415–6888) between 7:00 a.m. and 3:45 p.m. (EDT). Persons planning to attend this meeting are urged to contact the above named individual one or two working days prior to the meeting to be advised of any potential changes to the agenda, etc., that may have occurred.

Dated: August 30, 2001.

Sher Bahadur,

Associate Director for Technical Support, ACRS/ACNW.

[FR Doc. 01–22764 Filed 9–10–01; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Sunshine Act Meeting

DATE: Weeks of September 10, 17, 24, October 1, 8, 15 2001

PLACE: Commissioners' Conference Room, 11555 Rockville Pike, Rockville, Maryland

STATUS: Public and Closed

MATTERS TO BE CONSIDERED:

Week of September 10, 2001

There are no meetings scheduled for the Week of September 10, 2001.

Week of September 17, 2001—Tentative

There are no meetings scheduled for the Week of September 17, 2001.

Week of September 24, 2001—Tentative

Friday, September 28, 2001

9:25 a.m. Affirmation Session (Public Meeting) (if needed)

9:30 a.m. Briefing on Decommissioning Activities and Status (Public Meeting) (Contact: John Buckley, 301–415–6607)

1:30 p.m. Briefing on Threat Environment Assessment (Closed-Ex. 1)

Week of October 1, 2001—Tentative

Thursday, October 4, 2001

9:25 a.m. Affirmation Session (Public Meeting) (if needed)

Week of October 8, 2001—Tentative

There are no meetings scheduled for the Week of October 8, 2001.

Week of October 15, 2001—Tentative

Thursday, October 18, 2001

9:00 a.m. meeting with NRC Stakeholders—Progress of Regulatory Reform (Public Meeting) (Location—Two White Flint North Auditorium)

The schedule for Commission meetings is subject to change on short notice. To verify the status of meetings call (recording)—(301) 415–1292. Contact person for more information: David Louis Gamberoni (301) 415–1651.

The NRC Commission Meeting Schedule can be found on the Internet at: http://www.nrc.gov/SECU/smj/schedule.htm

This notice is distributed by mail to several hundred subscribers; if you no longer wish to receive it, or would like to be added to the distribution, please contact the Office of the Secretary, Washington, DC, 20555 (301–415–1969). In addition, distribution of this meeting notice over the Internet system is available. If you are interested in receiving this Commission meeting schedule electronically, please send an electronic message to dkw@nrc.gov.

Dated: September 6, 2001.

David Louis Gamberoni,

Technical Coordinator, Office of the Secretary.

[FR Doc. 01–22852 Filed 9–7–01; 11:50 am]

BILLING CODE 7590–01–M

OFFICE OF MANAGEMENT AND BUDGET

Cumulative Report on Rescissions and Deferrals

August 1, 2001.

Section 1014(e) of the Congressional Budget and Impoundment Control Act of 1974 (Public Law 93–344) requires a monthly report listing all budget authority for the current fiscal year for which, as of the first day of the month, a special message had been transmitted to Congress.

This report gives the status, as of August 1, 2001, of two deferrals contained in one special message for FY 2001. The message was transmitted to Congress on January 18, 2001.

Deferrals (Attachments A and B)

As of July 1, 2001, $1.4 billion in budget authority was being deferred from obligation. Attachment B shows the status of each deferral reported during FY 2001.

Information From Special Message

The special message containing information on the deferrals that are

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEE MEETING ON MATERIALS AND METALLURGY

SEPTEMBER 26, 2001
Date

PLEASE PRINT

NAME	NRC ORGANIZATION
FAROUK ELTAWILA	NRR/DE
Ken Karwoski	NRR/DE/EMCB
Louise Lund	NRR / DE / EMCB
Jim Riley	NEI
Joe Muscara	NRC/RES/DET
Cheryl Khan	NRC/NRR/DE
Steve Long	NRC/NRR/DSSA/SPSB
Franklin Coffman	NRC/RES/DET/MEB
Bill Bateman	USNRC/NRR/DE/EMCB
J. Bongarra	USNRC/NRR/DORPM/IOLB
Eric Thornsbury	NRC/RES/DRAA/PRAB
Lane Hay	SERCH Bechtel
John Rajan	NRC/NRR/SPSB
Steve Bajorek	NRC / RES
Maitri Banerjee	NRC/NRR/DLPM
Edmund Sullivan	NRC/NRR/DE/EMCB
Mark Kanavos	STPNOC
Alvin Gutterman	Morgan, Lewis & Bockius LLP
Joel Page	RES / DET / MEB

Steam Generator Action Plan

Briefing for the ACRS
Materials and Metallurgy Subcommittee
September 26, 2001

Maitri Banerjee, NRR/DLPM

Historical Overview

- 2/15/00 - IP2 Tube Failure Event
- 2/28/00 - NRR request to RES for independent review
- 3/16/00 - RES response to NRR
- 5/24/00 - Task Group Charter issued
- 8/29/00 - OIG Report issued
- 8/30/00 - Chairman's request for staff review of OIG Report
- 10/23/00 - Lessons-Learned Report issued
- 11/3/00 - Staff Review of OIG Report issued
- 11/16/00 - Steam Generator (SG) Action Plan issued
- 2/1/01 - ACRS Ad Hoc Subcommittee Report issued
- 3/5/01 - EDO tasking memo on ACRS Report (SG DPO)
- 5/11/01 - SG Action Plan revised (SG DPO issues)

SG Action Plan - Purpose

- SG Action Plan was issued on 11/16/00. The purpose of the action plan is to:
  - Direct and monitor the NRC's efforts
  - Ensure issues are appropriately tracked and dispositioned
  - Ensure the NRC's efforts result in an integrated SG regulatory framework

SG Action Plan - Scope

- Consolidates numerous activities related to SGs:
  - Recommendations from Task Group report
  - Recommendations from the OIG report
  - NEI 97-06, "Steam Generator Program Guidelines "
  - Resolution of SG DPO
  - GSI-163, "Multiple Steam Generator Tube Leakage "
- Anticipate revision to include resolution of GSI-188 "SG Tube Leaks/Ruptures Concurrent with Containment Bypass," and DG-1073, "Plant Specific Risk-Informed Decision Making : Induced SG Tube Rupture "

SG Action Plan - Scope (cont.)

- Also includes some non-SG related issues (e.g., Emergency Planning issues from OIG report)
- Does not address plant-specific reviews or industry proposed modifications related to implementation of GL 95-05 (voltage-based tube repair criteria)

SG Action Plan Status

- Action Plan dated 11/16/00 - 29 milestones
  - Items 1.1 - 1.21 (SG-related)14 of 21 complete
  - presently scheduled to be complete by 2/01/02
  - Schedule for NEI 97-06 items being reevaluated

  - Items 2.1 - 2.8 (non-SG related) 6 of 8 complete
  - scheduled to be complete by 02/02, except for one item TBD
- Revision dated 5/11/01 added 11 milestones
  - Items 3.1 - 3.11 (SG DPO-related)
  - Scheduled to be complete by 12/31/06

SG Action Plan- Significant Activities

- Regulatory Issues Summary - SG Lessons Learned
- Guidance for NRC Inspectors

  -Baseline inspection program
  -Risk-informed SDP
  -Inspector training
  -Facilities with known SG tube leak

- SG Performance Indicators
- Guidance for review of licensee SG inspection results, conference calls during outages
- SG workshop with stakeholders

SGAP - Significant Activities (Cont'd)

- Guidance for license amendment reviews
- Review and develop SE for NEI 97-06
- Establish SG Action Plan web page
- Planned initiative on risk communication and outreach to the public
- Milestones for ACRS recommendations on DPO
- Milestones for GSI 163, 188, DG-1073

SG Action Plan - Management

- Completion of significant milestones will be documented via memo from lead Division Director to Associate/ Deputy Office Director
- Resolution of issues will be coordinated with internal and external stakeholders
- Status of milestones will be updated in Commission Tasking Memo, NRR Director's Quarterly Status Report, and RES Operating Plan
- Overall management of the plan is the responsibility of DLPM

NEI Steam Generator Generic License Change Package
-Discussion
- NRC Review Status

ACRS Materials and Metallurgy Subcommittee

September 26, 2001

Ted Sullivan
Division of Engineering
Office of Nuclear Reactor Regulation
(301) 415-2796

Background

- Regulatory requirements for SG inspection/repair are prescriptive and out of date

  - Requirements not focused on key objective of ensuring tube integrity for entire period between inservice inspections

  - Meeting these requirements does not, in and of itself, ensure tube integrity is being maintained

- Staff initiative for a revised regulatory framework has evolved over time

  – Rulemaking
  – Generic Letter
  – Consideration of industry's NEI 97-06 initiative
  – Review of NEI SG Generic Change Package (GCP)

NEI SG Generic Change Package (GCP)

- Initially submitted February 4, 2000
  - Revised submittal dated December 11, 2000
- Staff did not initiate review until January 2001
  - due to followup activities relating to IP-2 SG tube failure on February 15, 2000
- Staff's review of GCP has included consideration of issues identified in the staff's action plan, including:
  - NRC IP-2 SG Tube Failure Lessons Learned Report
  - Regulatory Information Summary (RIS) 2000-22
  - DPO Action Plan

NEI SG Generic Change Package (GCP) (Cont'd)

GCP represents the culmination of efforts to develop a revised and updated regulatory framework that include the following characteristics:

- **performance based**: establishes performance criteria for ensuring tube structural integrity and leakage integrity under normal and accident conditions
- performance criteria are in terms of parameters which are **measurable** and **tolerable**
- **flexible**: methods for meeting the performance criteria are up to the licensee
- **adaptable** to changing degradation mechanisms and technology
- **risk informed**: ensures no significant increase in risk

NEI SG Generic Change Package (GCP) (Cont'd)

Revised Technical Specifications:

- Revised LCO Spec for operational leakage: 500 gpd to 150 gpd
- New LCO Spec, "Steam Generator Tube Integrity"
  - SR: Verify performance criteria are met in accordance with **Steam Generator Program**
- New Administrative Tech Spec: "**Steam Generator Program**"

New Administrative Tech Spec: "Steam Generator Program"

An **SG Program** shall be established and implemented to ensure SG tube integrity performance criteria are maintained

- Condition monitoring assessments of as-found tube condition vs the performance criteria shall be performed at each SG inspection outage. Requirements for condition monitoring are defined in the **SG Program**

- Use NRC approved performance criteria

- Use NRC approved tube repair criteria and repair methods

SG Program

- Details of the SG program will be located outside of tech specs

- Licensee's will commit to developing the SG Program in accordance with NEI 97-06

- NEI 97-06 provides general guidance for a performance based, programmatic strategy for ensuring SG tube integrity. Programmatic elements include:

  - performance criteria
  - tube integrity assessment
  - inservice inspection
  - tube repair limits & repair methods
  - leakage monitoring

- NRC staff is reviewing NEI 97-06 for endorsement as part of its review of the NEI SG GCP

Detailed EPRI Guideline Documents

- NEI 97-06 references sub-tier, detailed EPRI guideline documents concerning each of the programmatic elements. These include:

  - EPRI SG examination guidelines
  - EPRI tube integrity assessment guidelines
  - EPRI in-situ pressure test guidelines
  - EPRI guidelines for monitoring primary-to-secondary leakage
  - EPRI water chemistry guidelines
  - EPRI sleeve and plug assessment guidelines

Detailed EPRI Guideline Documents

- It had not, initially, been the staff's intent to review or endorse the sub-tier, detailed EPRI guideline documents
  - staff considers some of these guidelines to be "work in progress"
  - staff expectation that guidelines would be sufficiently well developed to lead to improved tube integrity performance
  - staff expectation that guidelines would continue to evolve over time in response to technology changes, lessons learned from operating experience, and results of industry and NRC studies (e.g., NRC SG mockup and ECT round robin, DPO action plan)

Inspection Interval Issue

- At the NRC SG Workshop in February 2001, industry representatives discussed draft revisions to the EPRI SG examination guidelines to permit inspection intervals for SGs with Alloy 600 TT or Alloy 690 TT tubing well beyond current EPRI guidelines and regulatory requirements:
  - Initial industry draft would have permitted inspection intervals ranging to 22 EFPY
- Staff is concerned that certain EPRI guidelines are not sufficiently well developed to ensure that inspection intervals beyond current requirements will be implemented so as to ensure that:
  - tube integrity performance criteria will continue to be met
  - tubing conditions not meeting the performance criteria will be promptly detected

Inspection Interval Issue (Cont'd)

- The staff has reviewed provisions of the EPRI guidelines critical to the effectiveness of condition monitoring (CM) to fulfill Appendix B obligation
  - A number of issues in this respect identified (ltr. dated 8/2/01)
  - Issues relate to industry practices that exist under the current regulatory framework and would continue to exist under the new framework
  - These issues not expected to reduce assurance of tube integrity or increase risk under new regulatory framework, assuming inspection intervals do not increase relative to current requirements and practice
  - However, the safety significance of these issues may be increased by longer inspection intervals depending on the specific methodology employed to justify the intervals

Inspection Interval Issue (Cont'd)

- The staff has also reviewed most of the industry responses to issues identified in the NRC IP 2 Lessons Learned Report and in RIS 2000-22
  - Preliminary findings documented by 8/2/01 letter to NEI
  - These issues also relate primarily to the EPRI guidelines, and some overlap issues above on CM and inspection intervals
  - The staff finds that a number of these issues remain unresolved, including issues extending beyond CM and inspection intervals
  - Issues not expected to reduce assurance of tube integrity or increase risk under new regulatory framework, assuming inspection intervals do not increase relative to current rqmts. and practices
  - Safety significance of additional issues may be increased by longer inspection intervals <u>for plants with degradation activity</u> depending on the specific methodology employed to justify the intervals

Preliminary Conclusion

- Pending resolution of the guideline issues, the staff has concluded preliminarily that it can proceed with review and approval of the GCP provided appropriate licensing restrictions are maintained on inspection intervals
  - The GCP would reduce assurance of tube integrity only in cases where longer inspection intervals than currently permitted are implemented without adequate justification.
- NRC staff is exploring with the industry alternatives to current requirements, particularly for improved tube materials, pending resolution of the guideline issues with reasonable assurance that the tube integrity performance criteria will be maintained
- NRC staff is working with industry to establish a protocol agreement for resolving outstanding technical issues (current and future)

NRC Review Status

- The staff is working with the industry to identify acceptable inspection interval restrictions
  - GCP would need to be revised accordingly
- Staff plans to issue SE concerning the GCP in a Regulatory Issue Summary (RIS)
  - Drafts of the SE and RIS will be issued for public comment
- Target Date for Completion: Previous target of 10/31/01 has slipped six months due to the inspection interval issue. New target date is April 2002. This is contingent, of course, on resolution of the inspection interval issue in the very near term

ACRS MATERIALS AND METALLURGY SUBCOMMITTEE

September 26, 2001

Office of Nuclear Regulatory Research

Status of Action Plan DPO Issues

September 26, 2001

Joseph Muscara, Division of Engineering Technology, RES - Materials Engineering Overview and Current Results
301-415-5844

Charles Tinkler, Division of Systems Analysis and Regulatory Effectiveness, RES - Severe Accidents and Thermal Hydraulics Overview
301-415-6770

Stephen Bajorek, Division of Systems Analysis and Regulatory Effectiveness, RES - Thermal Hydraulics
301-415-7574

Christopher Boyd, Division of Systems Analysis and Regulatory Effectiveness, RES - CFD Predictions
301-415-0244

SG Action Plan Milestones Associated with the DPO - Materials Engineering

3.1: Crack Propagation in SG tubes from pressure and MSLB loads

- Obtain loads, including cyclic loads, acting on tubes during MSLB from thermal-hydraulic calculations 12/31/02

- Obtain load and displacement information experienced by SG structures under MSLB conditions from existing analyses and submittals 12/31/02

- Using above information estimate upper bound loads, cycles and displacements 12/31/02

- Estimate crack growth, if any, for a range of crack types and sizes using the bounding loads, displacements and cycles in addition to the pressure stresses 12/31/02

3.1: Crack Propagation in SG tubes from pressure and MSLB loads (Cont.)

- Estimate loads required to propagate a range of axial and circumferential cracks and compare to MSLB loads (Margins)
  12/31/02

- Conduct tests of degraded tubes under pressure and with axial and bending loads to validate the analytical results
  06/30/03

3.2: Damage Progression Via Jet Cutting

- Complete tests of jet impingement under MSLB conditions 12/31/01

- Conduct long duration tests of jet impingement under severe accident conditions 12/31/01

- Draft reports 12/31/01

Jet Impingement and the Potential for Propagation of Failure

- In NUREG–1570 the issue of propagating failures due to erosion by steam jets or high temperature gas/particle streams was considered.

- Additional experiments to address jet cutting issues were performed.
  - High temperature tests, representative of severe accident conditions performed by W. Tabakoff at University of Cincinnati
  - Steam jet erosion tests, representative of design basis accident conditions performed at Argonne National Laboratory
- Initial results indicated low erosion rates (10 min test)
  - ACRS suggested longer term tests to verify rates were steady state

    Longer tests (30 min) gave data indistinguishable from shorter tests

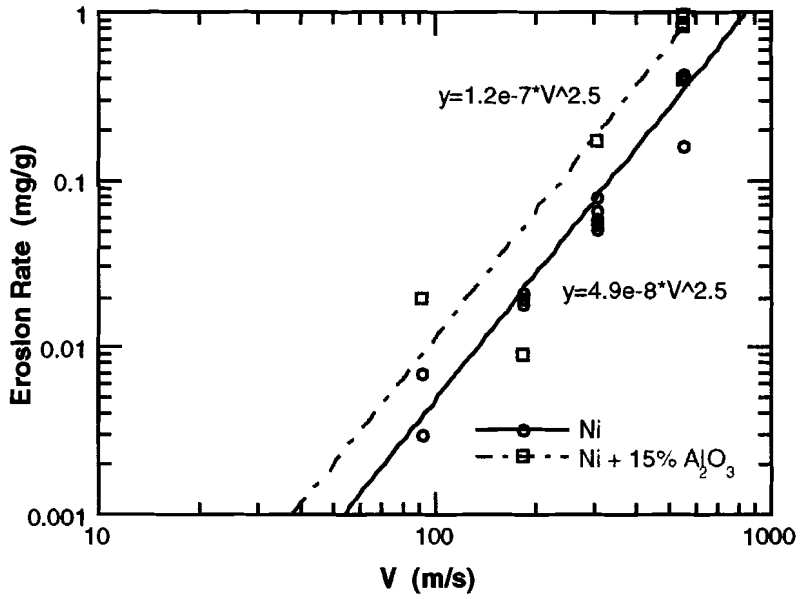
- High Temperature Erosion Tests
  - Particle loading due to aerosol transport from the molten core. Accident analyses suggest

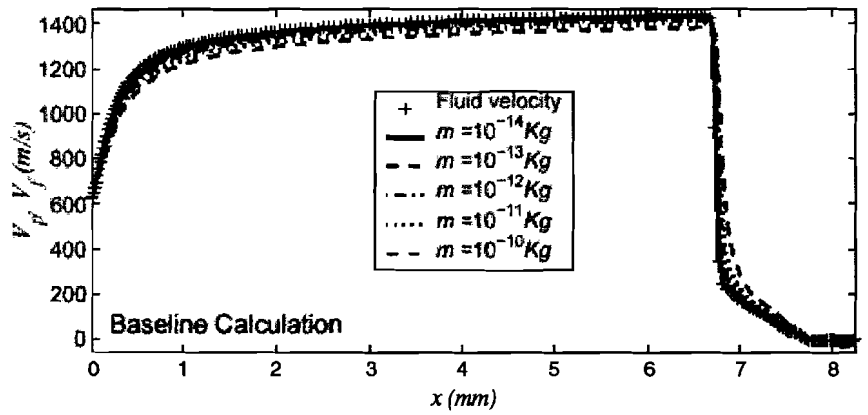
    Particulate in jet consists primarily of Ag plus lesser amounts of In_2O_3, $CsMoO_4$, SnO_2, CsI, and other species

    Median particle diameter is $\approx 1.5\ \mu m$, bulk of particles $< 3\ \mu m$
  - Surrogate particles Ni and NiO were chosen for the tests. Particle sizes were taken as 3–7 μm. In terms of size (larger particles are more damaging) and hardness at temperature, the surrogate particles should give conservative estimates of erosion rates.

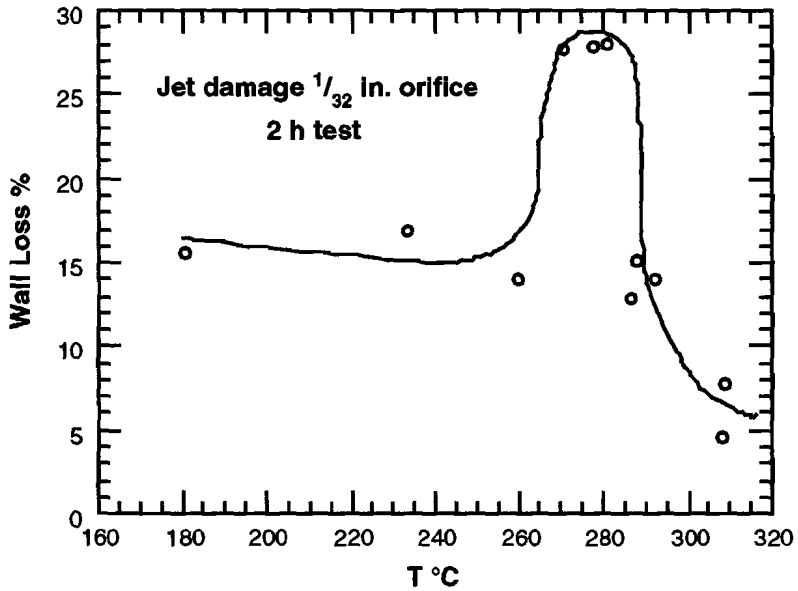
    Initial tests with NiO gave deposition rather than erosion. Subsequent tests were performed with Ni + 15% Al_2O_3. High hardness and angularity of Al_2O_3 particles additional conservatism.

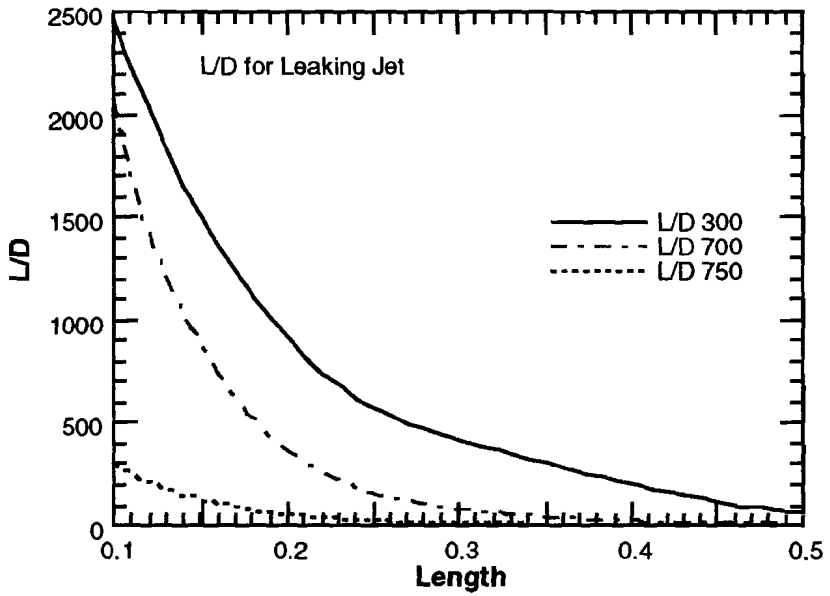
    Erosion rates with Ni + 15% Al_2O_3 about 2.4 times higher than Ni





- Erosion rates in accident < 2 mil/h (Ni), 5 mil/h (Ni + 15%Al_2O_3)
  - Assumed velocity of 200 m/s conservative based on CFD analysis with L/D=16 (>200 for 0.25 in. cracks)
  - Erosion rates are conservative because of particle choices
  - Mass loading based on conservative accident calculations



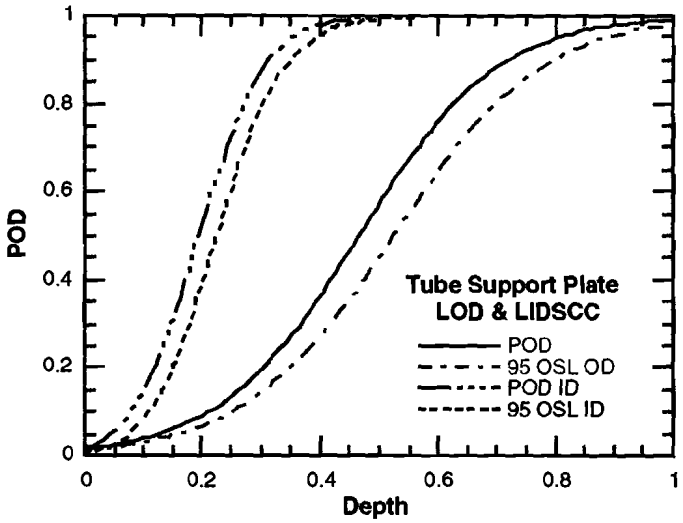


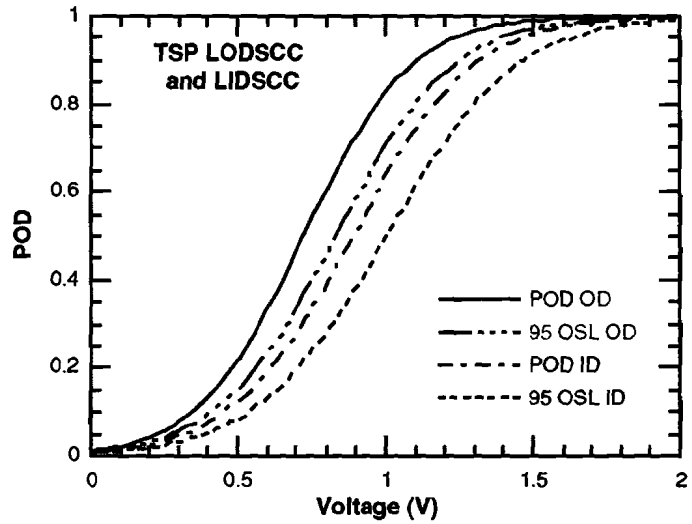
- Steam jet erosion has been studied at ANL under MSLB pressure drop and temperatures
  - Erosion rate depends on degree of subcooling. Peaks at cold leg conditions (less likelihood of cracks).
  - Rate appears to drop when flashing occurs before jet impacts target (detectable by size of impact area)
  - Results for $1/32$ orifice where L/D=8 are conservative compared to cracks
  - Confirmatory tests on cracks in progress

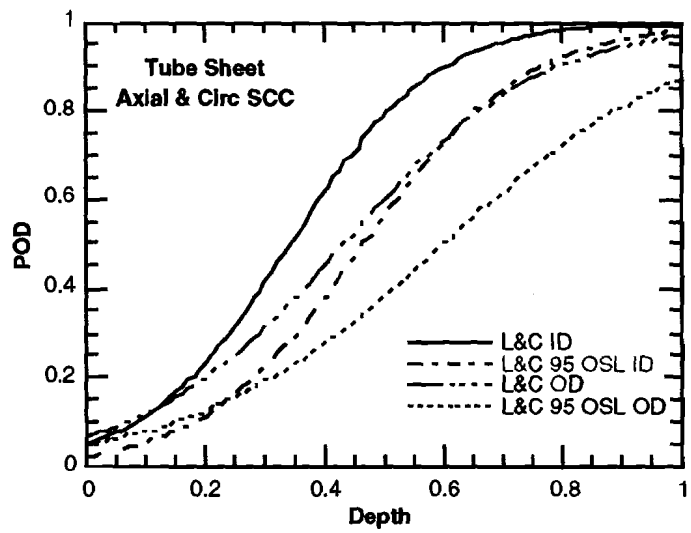
3.6: Improvements Over Use Of a Constant POD

- Complete analysis and document research results from the SG mockup NDE Round Robin 12/31/01

Insights into Probability of Detection from Round Robin Tests on ANL Mockup



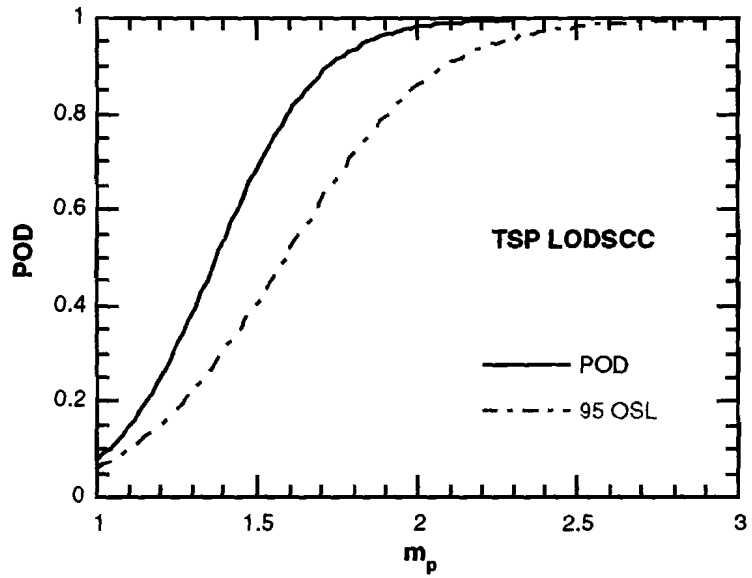


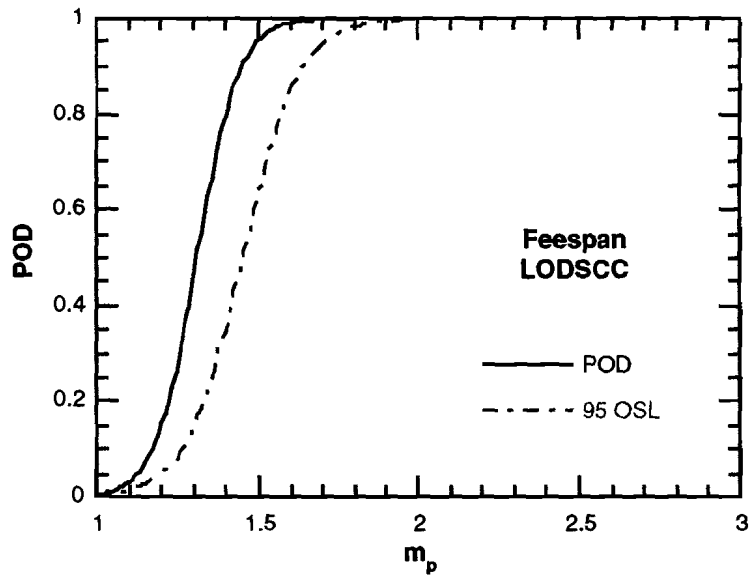


ID cracks are easier to detect than OD cracks

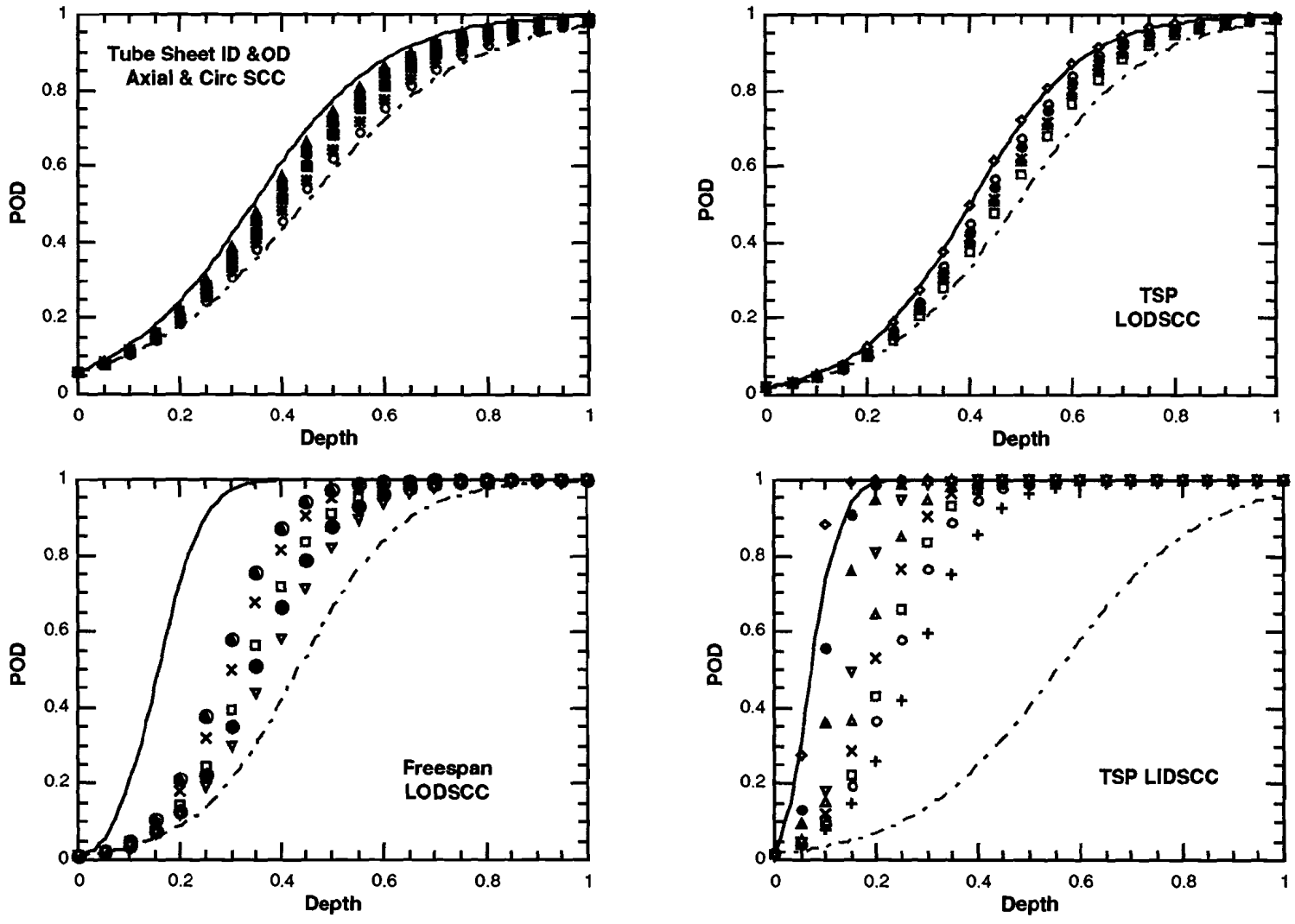
POD vs Voltage characterizes interpretation; a signal is seen, is it correctly characterized as a crack?

TS MRPC results also show ID OD variation. False call rate is higher than for TSP or freespan.





Depth does not characterize structural integrity completely; short, deep cracks can be less damaging than longer, shallower cracks. m_p parameter measures stress magnification on the remaining ligament and includes length and depth; cracks that violate $3\Delta p$ correspond to $m_p > 2.3$



- Training and procedures minimize team to team variation in some cases, but in other cases significant variations were observed.

Develop a more technically defensible position on the treatment of radionuclide release to be used in the safety analyses of design basis events.

Due	Task	Description
10/31/01	A	Assess Adams and Atwood and Adams and Sattison spiking data with respect to the ACRS comments.
12/31/01	B	Develop a response to ACRS comments based upon work performed in Task 3.9.a.
2/15/02	C	Publish for Public Comment response to ACRS comments.
6/30/02	D	Complete Review of Public Comments
8/15/02	E	Based upon Task 3.9.d. determine if additional work is required.

3.10: Stress Corrosion Cracking

- Conduct tests to evaluate crack initiation, evolution and growth 12/31/05
- Use operating experience and results from laboratory testing to develop models for predicting cracking behavior of SG tubes in the operating environment 12/31/06

Division of Systems Analysis and Regulatory Effectiveness Severe Accident and Thermal-Hydraulic Analysis of Steam Generator Tube Integrity

Presentation to the Advisory Committee on Reactor Safeguards
September 26, 2001

Charles G. Tinkler	Overview / Severe Accident
Stephen Bajorek	Thermal Hydraulics
Christopher Boyd	CFD Predictions

Recommendations of the ACRS Ad Hoc Subcommittee on a Differing Professional Opinion (Steam Generator Tube Integrity)

Severe and Design Basis Accident Conditions

- Develop a better understanding of the behavior of degraded steam generator tubes under severe accident conditions
  - Specifically addressed by SGAP Item 3.4
- Evaluate the potential for progression of damage to (multiple) steam generator tubes during rapid depressurization caused by a main steamline rupture.
  - Specifically addressed by SGAP Item 3.1

Steam Generator Action Plan

Task 3.4 Develop better understanding of reactor coolant conditions and the corresponding component behavior (including steam generator tubes) under severe accident conditions.

- Major components of research
  - System level code analysis (SCDAP/RELAP)
  - Computational fluid dynamics (CFD) code analysis (FLUENT)
  - Assessment of 1/7th scale data
  - New experimental data

Steam Generator Plan Activities – Severe Accident

Task 3.4 Near term items

<u>Subtask 3.4a</u> Perform system level analyses to assess impact of plant sequence variations. Due October 2001.

- Reactor coolant pump seal leakage
- Tube leakage
- Alternate steam generator depressurization
- Calculations completed, report in progress, on schedule.
- Next Subtask 3.4b re-evaluate system level code assumptions and simplifications. Due March 30, 2001.

<u>Subtask 3.4e.1</u> Benchmark CFD methods against 1/7 scale test data.

- Completed on schedule - August 2001
- Conclusion : FLUENT calcs, with no tuning, show good agreement with data
- Next milestone (Subtask 3.4.e.2) to develop full scale CFD model for plant calculations. March, 2002.

Severe Accident Issues

• Plant design differences –	System level analysis
• Plant sequence variations –	System level analysis
• Uncertainty analysis quantified for – system level mixing parameters	System level analysis
– Core melt progression uncertainty	
• Loop seal clearing –	System level analysis
• Effect of tube leakage on – Inlet plenum mixing	CFD analysis, considering testing
• Hot leg/inlet plenum orientation –	CFD analysis, considering testing
• Tube to tube variations	CFD analysis, considering testing
• FP deposition –	Planned testing (ARTIST)

SG DPO Action Plan Activities - DBA T/H

Task 3.1 Develop better understanding of potential for damage progression of multiple steam generator tubes due to depressurization accidents (e.g. MSLB)

Scheduled completion 12/02

- **ΔP loads on tube support plates and SG tubes**
- **Flow induced vibrations**

Thermal-Hydraulic Issues Related to Steam Generator Transient Loads

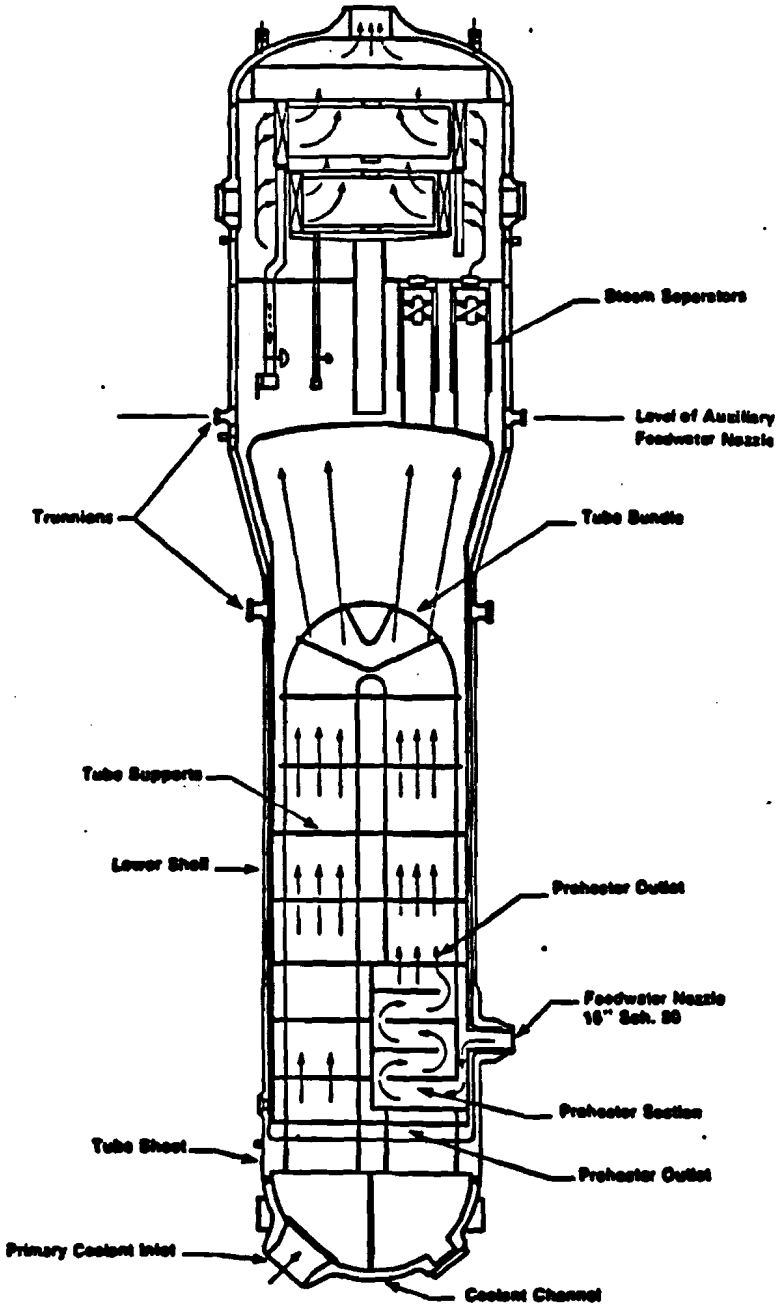
Presentation to the Advisory Committee on Reactor Safeguards
September 26, 2001

Stephen M. Bajorek
Safety Margins and Systems Analysis Branch
Division of Systems Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research

DPO Contentions:

(b) Blowdown Forces: Depressurization of the reactor coolant system during a main steam line break will produce shock waves and violent sympathetic vibrations that will cause cracks to form, to grow and unplug, leading to higher leakage from the primary to secondary sides of the reactor coolant system than has been considered by the staff.

(e) Tube Support Plate Lift: Tube support plates can be lifted during the sudden depressurization of a main steamline break and this can cause cracks in tubes to penetrate through the tube walls and lead to additional flow from the primary coolant system to the secondary side of the coolant system.



Background

Depressurization waves propagate at sonic velocity though a fluid media. As depressurization waves move through a system and interact with each other, the ΔP created causes complex loads and motion in some components.

Typical Velocities

Primary at Tcold :
c(280 C, 15.5 Mpa) = 1074 m/s

Secondary at Operating Pressure:
c(Tsat, 6.55 Mpa) = 1002 m/s (Saturated liquid)
c(Tsat, 6.55 Mpa) = 493 m/s (Saturated vapor)

Evaluation Plan:

(1) Initial approach to use hand calculations assuming a 1 msec BOT and a 3D TRAC-M model of a steam generator to generate a transient pressure history on the secondary side at the tubesheet. The $\Delta P(t)$ across the tubesheet is the forcing function for tubesheet vibration. Goal of the hydraulic effort is to produce a conservative forcing function, $\Delta P(t)$.

(2) The forcing function $\Delta P(t)$ across the tubesheet will be used to analyze tubesheet motion and stresses affecting the tubes. A finite element model of the affected region will be developed and used to determine the transient stresses.

(3) Long-term effort to consider results of (1) and (2) :

a) Determine most appropriate analysis tool for investigating SG hydraulic forcing functions.

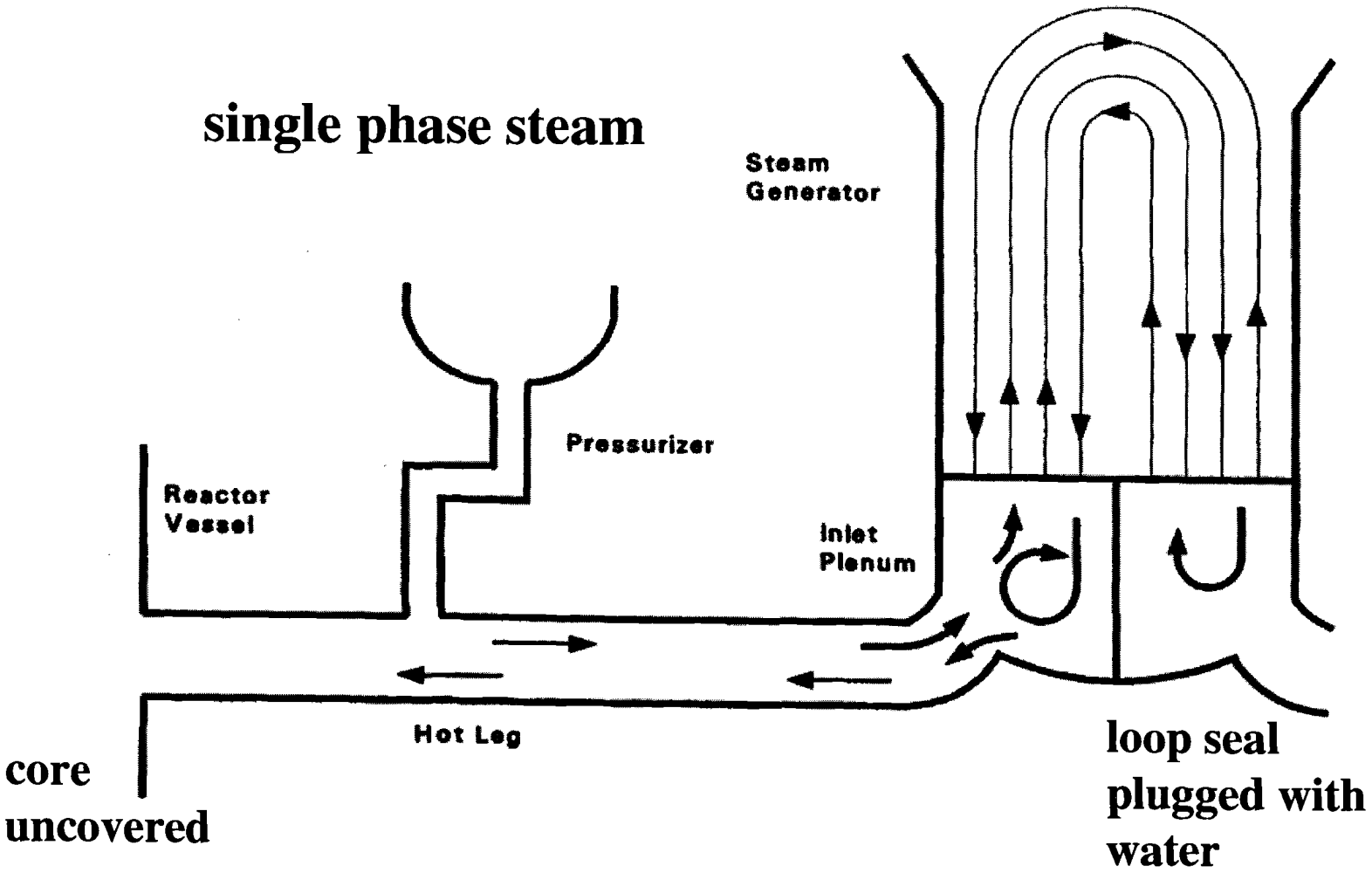
b) Perform experimental testing and validation as necessary.

Analysis of Steam Generator Inlet Plenum Mixing using Computational Fluid Dynamics (CFD)

Presentation to the Advisory Committee on Reactor Safeguards
September 26, 2001

Christopher Boyd
Division of Systems Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research

Thermal-Hydraulics of Interest

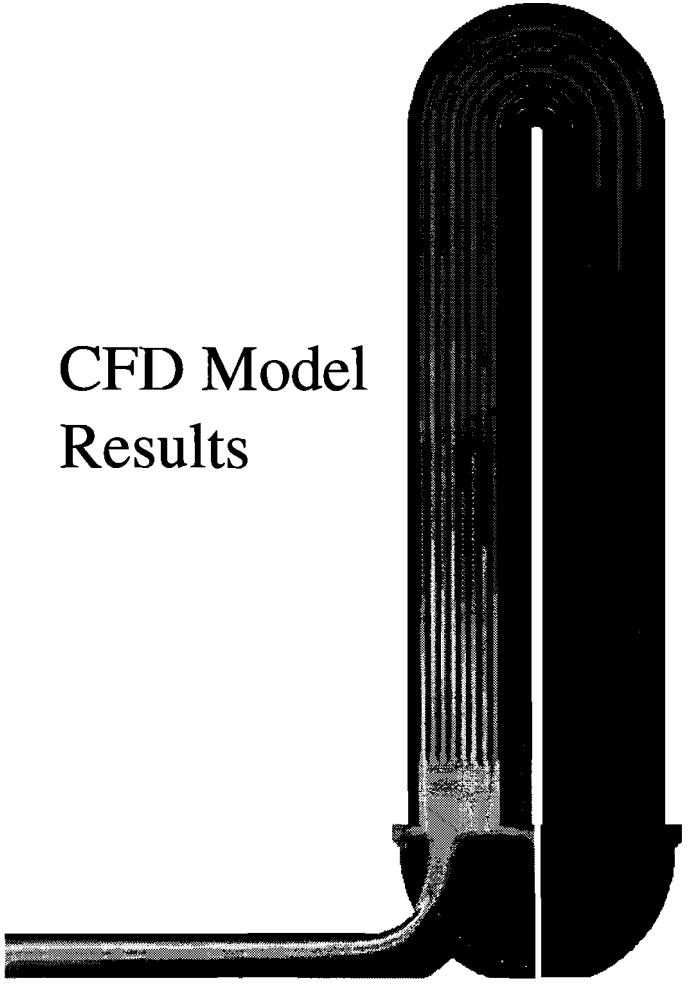


Background

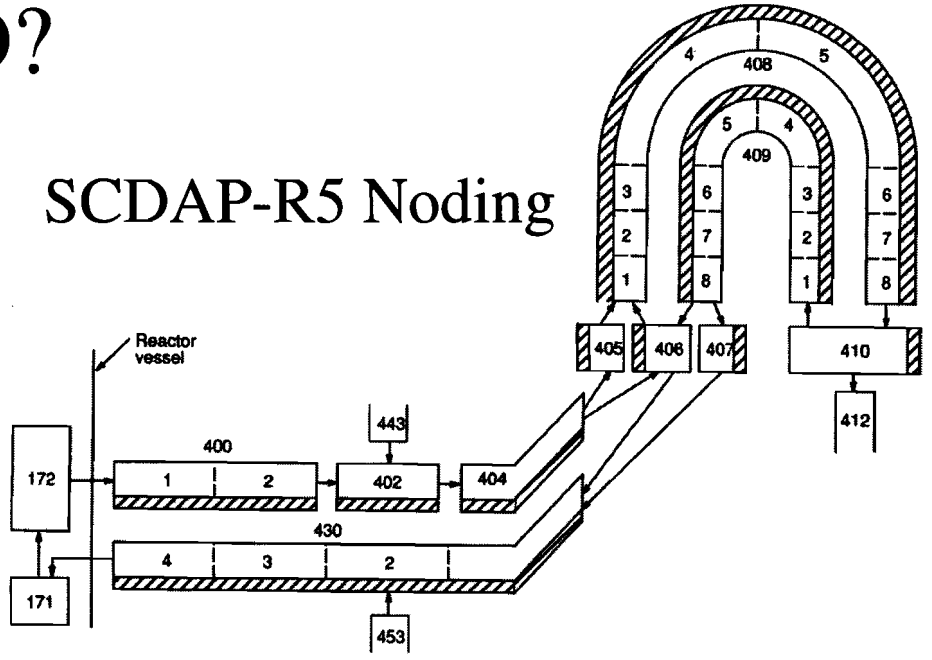
- Thermal-hydraulic predictions of plant behavior rely on SCDAP-RELAP5. (lumped parameter code)
- Tube temperature predictions during the re-circulating natural circulation phase are directly influenced by mixing parameters which are fixed into the SCDAP-RELAP5 models.
- These mixing parameters are determined from experiments that do not cover the full range of conditions which are of interest to the NRC.

Advantages of CFD?

CFD Model Results

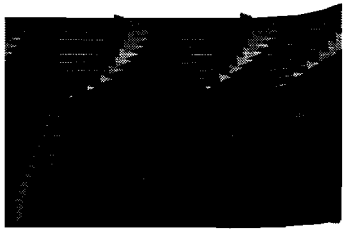
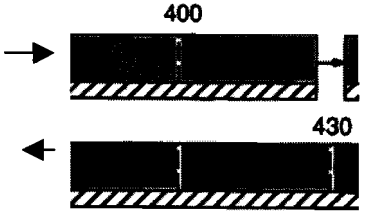
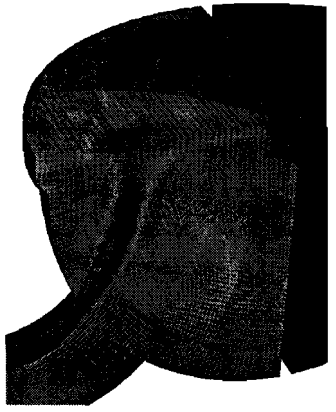
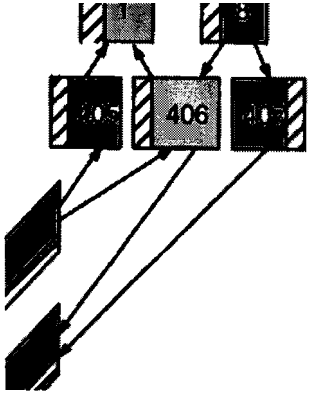
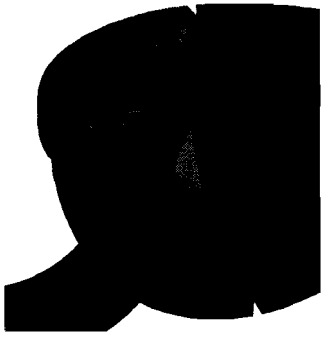
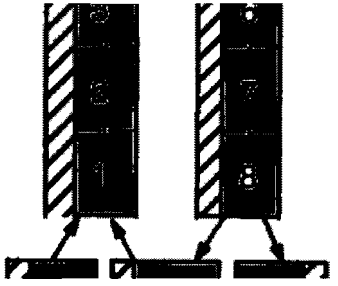


SCDAP-R5 Noding



- less expensive than experiments
- direct resolution of mixing
- no fixed mixing parameters
- extension of data
  - full scale, full P,T, steam
  - inlet geometry effects
  - tube leakage effects

Flow Physics Comparison

	CFD Approach		Lumped Parameter Codes	
Hot Leg		Direct prediction of counter current flow.		Flow broken into two 1D pipes set prior to run.
Inlet Plenum		Hot plume is modeled in detail using CFD. Temperature variations and mixing are predicted implicitly from flow equations.		Mixing fraction and re-circulation ratio are fixed in code input. Temperatures represent average (lumped) values.
Tubes		Flow into the tube sheet is computed directly. Tube fraction is determined implicitly from the results. Temperature variations are predicted.		Tube fractions are fixed. Average temperatures used.

CFD Will Provide…

- Improved understanding of 1/7th scale data
- Extension to full scale
  - does scale affect mixing parameters?
- Effect of tube leakage
- Effect of inlet geometry variations
  - CE plants
- Tube to tube variations within hot tubes

Schedule / Approach

8/01 • Validate technique through a comparison with Westinghouse 1/7th scale data (complete).

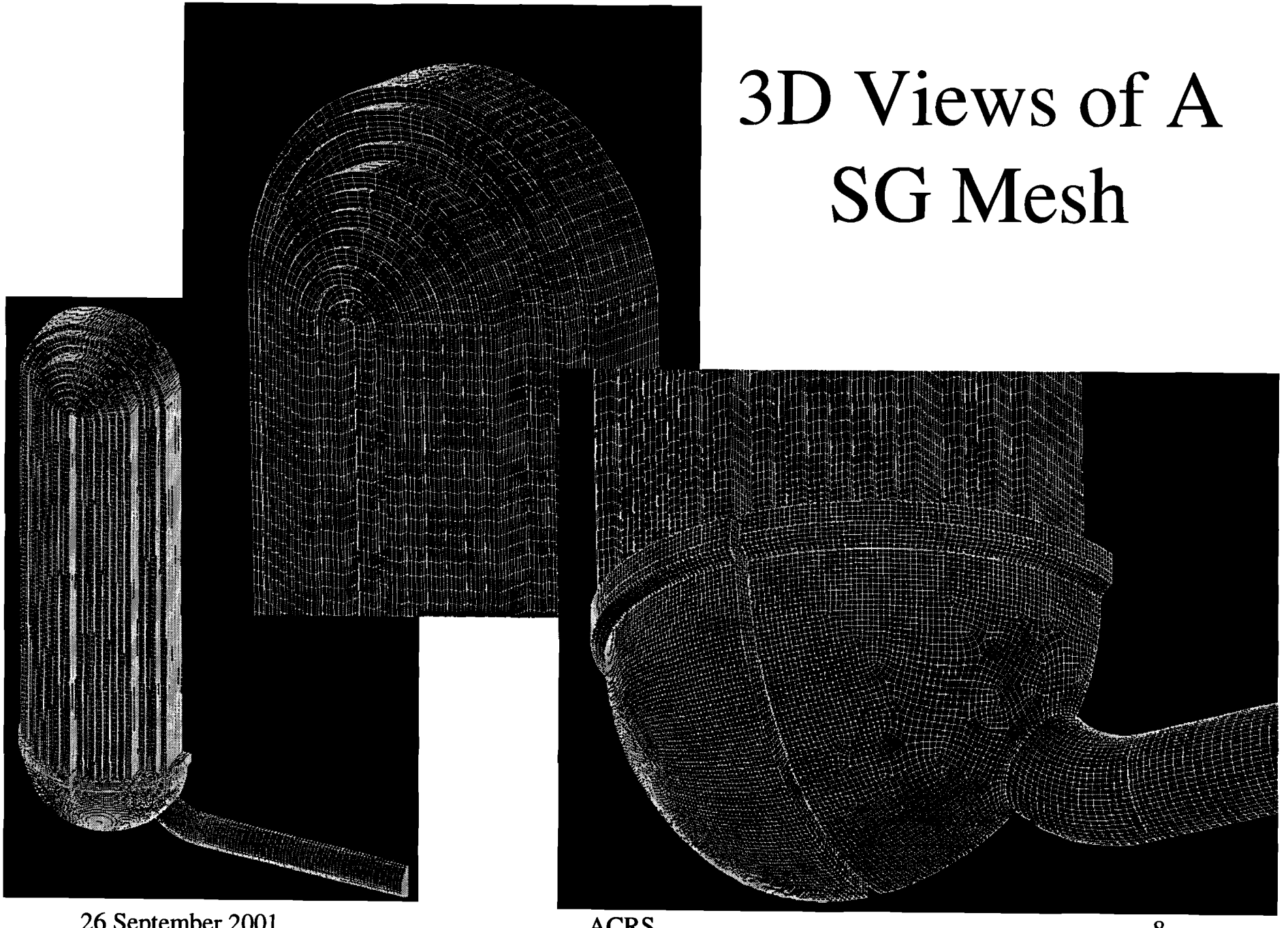
3/02 • Extend predictions to full scale using best estimate conditions.

- What is the effect of scale on the mixing parameters?

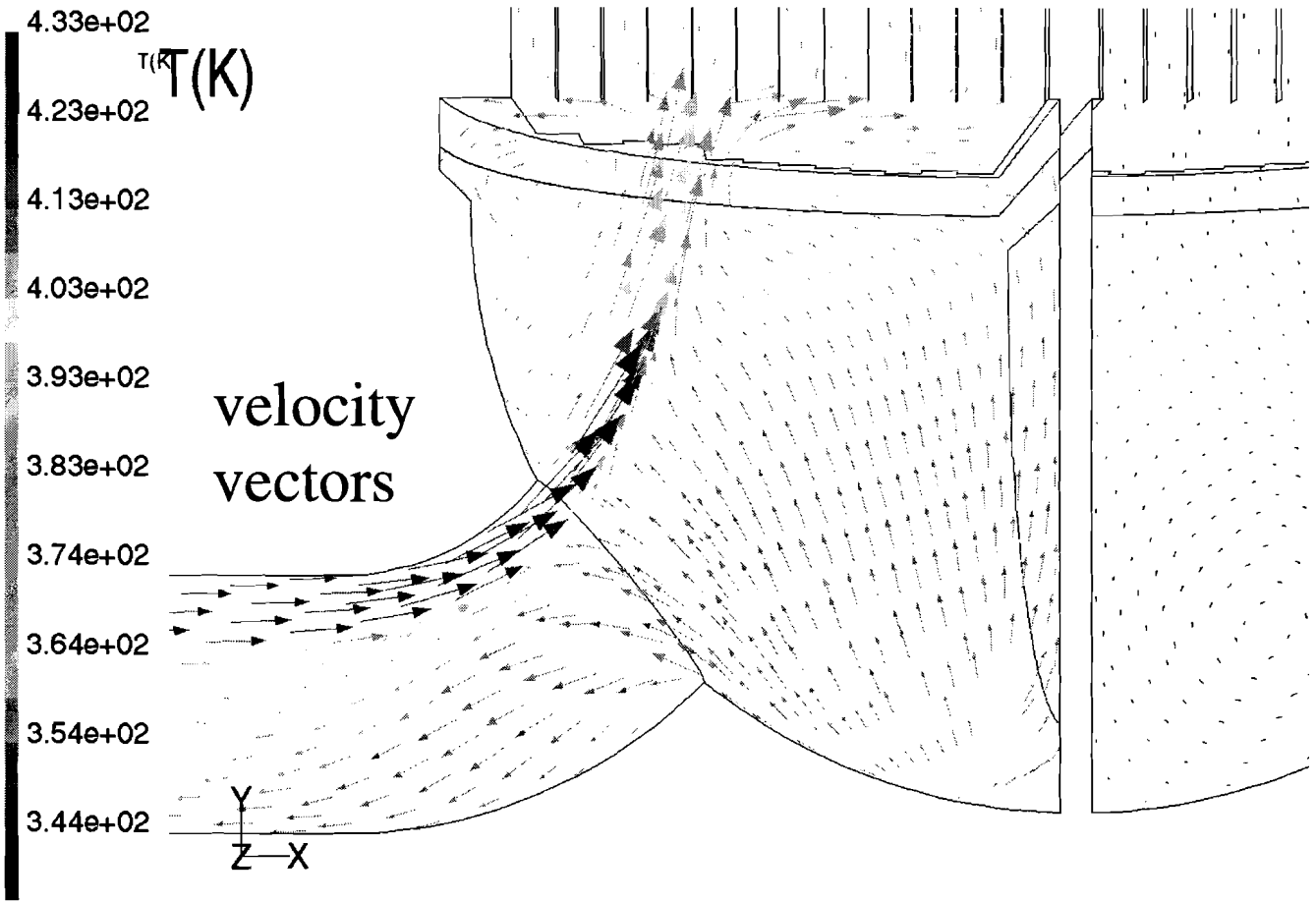
7/02 • Complete additional studies.

- What is the effect of tube leakage?
- What effect can inlet geometry have on mixing?
- Other sensitivities…

3D Views of A SG Mesh



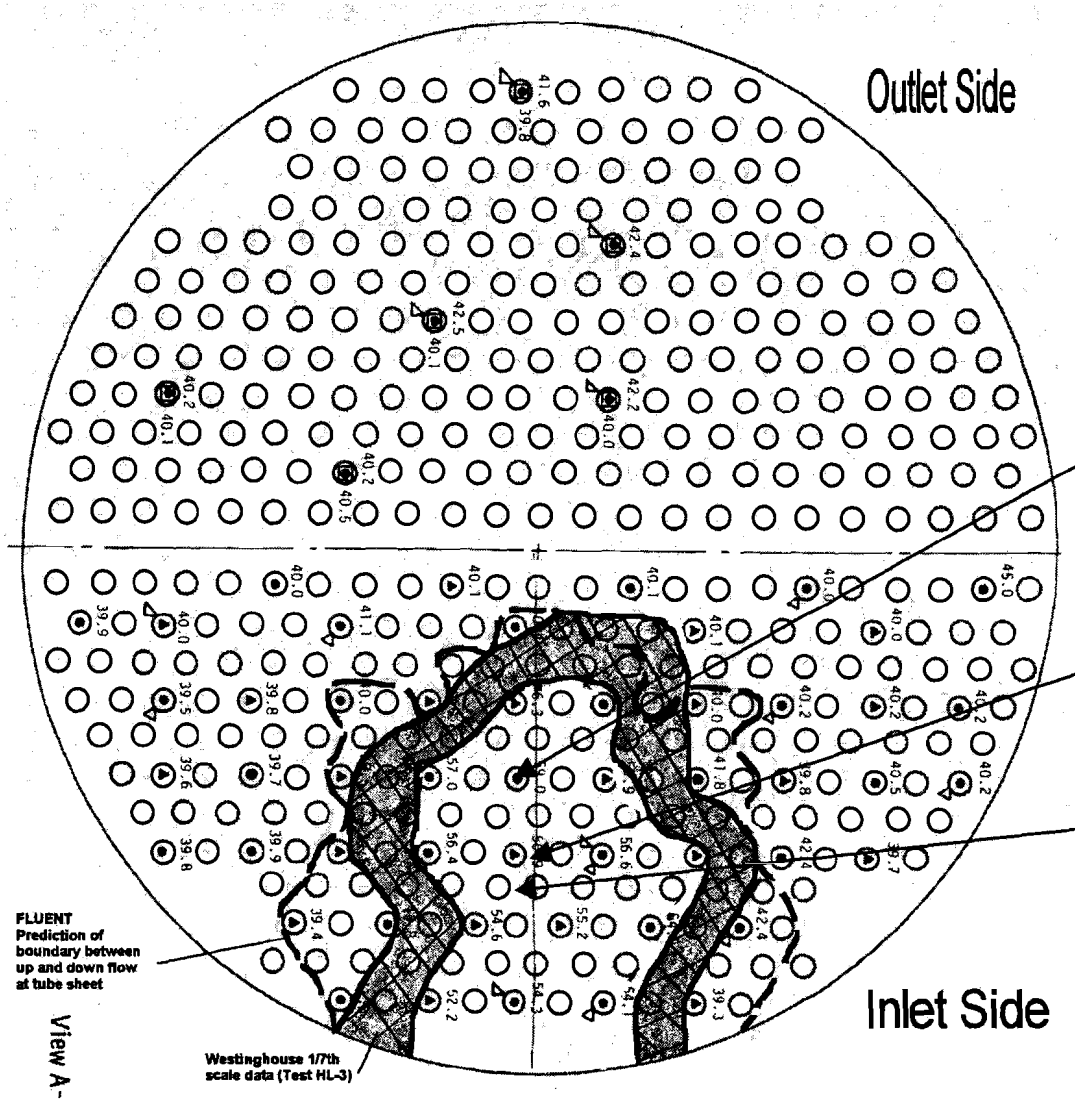
Qualitative Results



Quantitative Results (Test SG-S3)

Parameter or Value	Westinghouse 1/7th Reported Test Data	FLUENT prediction	difference
Heat Loss at Tubes (kW)	3.56	3.48	-2.3 %
# Hot Tubes	75	98	23 tubes
# Cold Tubes	141	118	-23 tubes
T_h Hot Leg (average Hot T (C))	159.3	149.9	9.4 °C
T_c Hot Leg (average Cold T (C))	86.8	86.9	0.1 °C
m Hot Leg (mass flux (kg/s))	.0599	.0646	7.8%
T_{ht} Tubes (average Hot T (C))	100.8	99.6	-1.2 °C
T_{ct} Tubes (average Cold T (C))	64.7	65.6	0.9 °C
mt Tubes (mass flux (kg/s))	.1197	.1088	9.1 %
mt/m (recirculation ratio)	2.01	1.7	-15.4 %
f (mixing fraction (ref. 2))	0.85	0.91	7.1%

Tube Sheet, Test HL-3



Peak Temperature

data (1 row off ctr)
Tmax = 59 (C)

prediction (center line)
Tmax = 61.5

prediction (1 row off ctr)
Tmax = 58.5

Results of Validation

- The CFD predictions of parameters are generally within 10% of the Westinghouse 1/7th scale data.
- The predictions generally are considered to lie within the experimental uncertainty.
- The phenomena observed during the tests are predicted by the CFD code.
- Work on the full scale predictions will proceed with a high degree of confidence in the technique.

Summary

- The CFD technique has been demonstrated to be applicable for the prediction of mixing parameters.
- This work provides a high degree of confidence that CFD can be used to evaluate conditions not explicitly covered by the experimental facilities.
- Further analysis is planned.
  - full scale analysis
  - tube leakage
  - geometry effects
  - sensitivity studies

GL 95-05 DPO ISSUES

AND

SOUTH TEXAS UNIT 2 STEAM GENERATORS

Briefing of the ACRS Subcommittee on Materials and Metallurgy

September 26, 2001

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7/8" TUBE LEAKAGE DATABASE

ACRS RECOMMENDATION: Database needs to be greatly improved to be useful. Consider requiring a near-term expansion of the database.

Staff agrees that the 7/8" leakage database does not demonstrate as strong a correlation as the 3/4" database

- Expansion of the database may firm up the correlation

Licensee committed to periodically remove tubes for destructive examination and data are reflected in the database. Utilities focus on larger voltage indications (i.e., ones that are likely to leak).

- No regulatory vehicle in methodology to require removal of additional tubes beyond these current commitments

Staff will continue to monitor the effects of additional data on the correlation

FLAW GROWTH

ACRS RECOMMENDATION: Staff should establish a program to monitor the predictions of flaw growth for systematic deviations from expectations. Develop a database on predictions and observed voltage distributions.

Staff has, and will continue, to review "90-day reports" submitted by licensees with this in mind

Staff formalizing review of inspection summary reports (including 90-day ODSCC reports) in conjunction with Steam Generator Action Plan Item 1.10

Some instances where predictions were non-conservative either in the number or severity (voltage) of degradation

SOUTH TEXAS UNIT 2 STEAM GENERATOR DESIGN

Westinghouse Model E2 Steam Generators (SGs) - 4864 tubes per steam generator

3/4" outside diameter alloy 600 mill annealed tubes - 15 thermally treated tubes in one steam generator

Drilled hole stainless steel tube support plates (TSPs) - bottom support is a flow distribution baffle with enlarged tube holes

Similar to Doel 4 and Tihange 3 Steam Generators - no other domestic plants with stainless steel drilled hole TSPs

Tubes hydraulically expanded into tubesheet

Shot peened hot leg at end of 1st cycle and cold leg at end of 2nd cycle

U-Bend heat treatment for R1 and R2 prior to operation

Preheater - tubes expanded to reduce likelihood of tube wear (high crossflow velocity of feedwater)

STEAM GENERATOR OPERATING EXPERIENCE AT STP 2

T-hot approximately 625°F

At the End-of-Cycle 8 (EOC 8) in March 2001, the SGs were in operation for 8.9 EFPY

Primary degradation mechanism - axial outside diameter stress corrosion cracking (ODSCC) at TSPs

Other degradation mechanisms observed during this inspection and prior inspections

- Free span axial ODSCC associated with ding locations (9 at EOC 7, 22 at EOC 8)
  - 16 of 22 in cold leg, most are OD initiated
- 4 free span volumetric indications in EOC 8

STEAM GENERATOR OPERATING EXPERIENCE AT STP 2 (CONT'D)

New mechanisms observed at EOC 8

9 circumferential indications at hot leg expansion transition - 8 OD initiated, one possible ID indication (in situ tested some indications)

Degradation at paired dings - dings separated by thickness of TSPs and believed to be introduced during a "bending moment during tube insertion" - 1 circumferential indication at upper ding and axial at lower ding (in situ tested)

Row 1 U-bend indication - 1 OD initiated axial crack near apex (in situ tested)

Single axial indication at U-bend transition in Row 36 - OD initiated

OD volumetric indication at expansion transition of one of the preheater expanded tubes - possible foreign part damage

9% of tubes currently plugged

Replacement scheduled for 12/02

VOLTAGE BASED REPAIR CRITERIA

Implementation of voltage-based tube repair criteria at STP 2

Cycle 7 (1998-1999) - GL 95-05, 1-volt repair criteria

Given I-131 levels, 15.4 gpm primary-to-secondary leakage could be tolerated during a SLB

Cycle 8 (1999 - 2001) - GL 95-05 1.0 volt repair criteria

Cycle 9 (2001 - 2002) - 3.0 volt locked tube support plate repair criteria

Select tubes expanded at TSPs 2, 3, and 4 to limit TSP motion during SLB

Similar to Braidwood/Byron 1, 3-volt repair criteria

LEAKAGE DURING NORMAL OPERATION

Steam generator tube leakage observed for first time during Cycle 8

SG A: 10.5 gpd	SG B: 7.5 gpd
SG C: 8.0 gpd	SG D: 9.0 gpd

Secondary side pressure test conducted for 4 days - approximately 100 "damp" tubes located

- Leakage attributed to ODSCC at TSPs

No other domestic plant has observed measurable operating leakage from axial ODSCC at tube-to-tube support plate intersections

- South Texas 2 is only one with drilled hole stainless steel TSPs
- 2 foreign plants had drilled hole stainless steel TSPs (Doel 4 and Tihange 3) - during a secondary side pressure test at Doel 4, some leakage was attributable to ODSCC at TSPs

Because of operational leakage, licensee essentially implemented a 1.5 volt repair criteria

- Performed plus-point inspections and depth sized to determine tubes that should be preventively plugged

NRC REVIEW OF SOUTH TEXAS 90-DAY REPORT (EOC 8/BOC 9)

Predicting EOC voltage distributions

- Number of indications - POD of 0.6
- Severity of indications - growth rate distribution

Correlation of in-situ test results to observed leakage during operation

- In-situ tested tubes with most severe degradation
- When corrected for operating conditions (e.g., temperature), leakage measured during test does not appear to account for observed leakage

Probability of Leakage Model

- Several tubes were "damp" following the secondary side pressure test - are the results of this test consistent with the probability of leakage model?

Comparison of Number of Indications Predicted versus Observed
Cycles 6, 7, and 8

Steam Generator	Cycle 6 (1997-1998)		Cycle 7 (1998-1999)		Cycle 8 (1999-2001)	
	Projected	Actual	Projected	Actual	Projected	Actual
A	322	188	293	330	509	611
B	565	500	836	815	1294	1229
C	437	456	749	602	927	972
D	437	340	558	515	792	767
Total	1761	1484	2436	2262	3522	3579

Observation: For cycle 8, under predicted the number of indications in 2 of 4 steam generators

Comparison of Severity of Indications Predicted versus Observed
Cycles 6, 7, and 8 (greater than 2 volt indications)

Steam Generator	Cycle 6 (1997-1998)		Cycle 7 (1998-1999)		Cycle 8 (1999-2001)	
	Projected > 2 V	Actual > 2 V	Projected > 2 V	Actual > 2 V	Projected > 2 V	Actual > 2 V
A	2	4	6	7	18	43
B	1	0	1	8	34	33
C	0	1	2	11	32	41
D	0	1	3	8	24	46
Total	3	6	12	34	108	163

Observation: For cycle 8, under predicted the severity of the “large” voltage indications in 3 of 4 steam generators

Average Growth Rates
Cycles 5, 6, 7, and 8

Cycle	Period	Duration (EFPD)	Number of Indications	Average BOC Voltage	Average % Growth per EFPY
5	1995-1997	450	703	0.31	31
6 (2RE06)	1997-1998	564.9	1484	0.31	27
7 (2RE07)	1998-1999	342.5	2262	0.41	45
8 (2RE08)	1999-2001	458	3580	0.37	82
9	2001-2002	485 (planned)			

Observation: Average growth rate is increasing

NEXT STEPS

Questions/Issues posed to licensee

Monitoring for operational leakage

None observed presently

Licensee plans to replace steam generators in 12/02