ACRS Subcommittee on Differing Professional Opinion Issues

UNITED STATES NUCLEAR REGULATORY COMMISSION

October 12 and 13, 2000

Office of Nuclear Reactor Regulation
Office of Nuclear Regulatory Research

STEAM GENERATOR ISSUES INTRODUCTION

Presented to

ACRS STEAM GENERATOR DPO COMMITTEE

By

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October 13, 2000

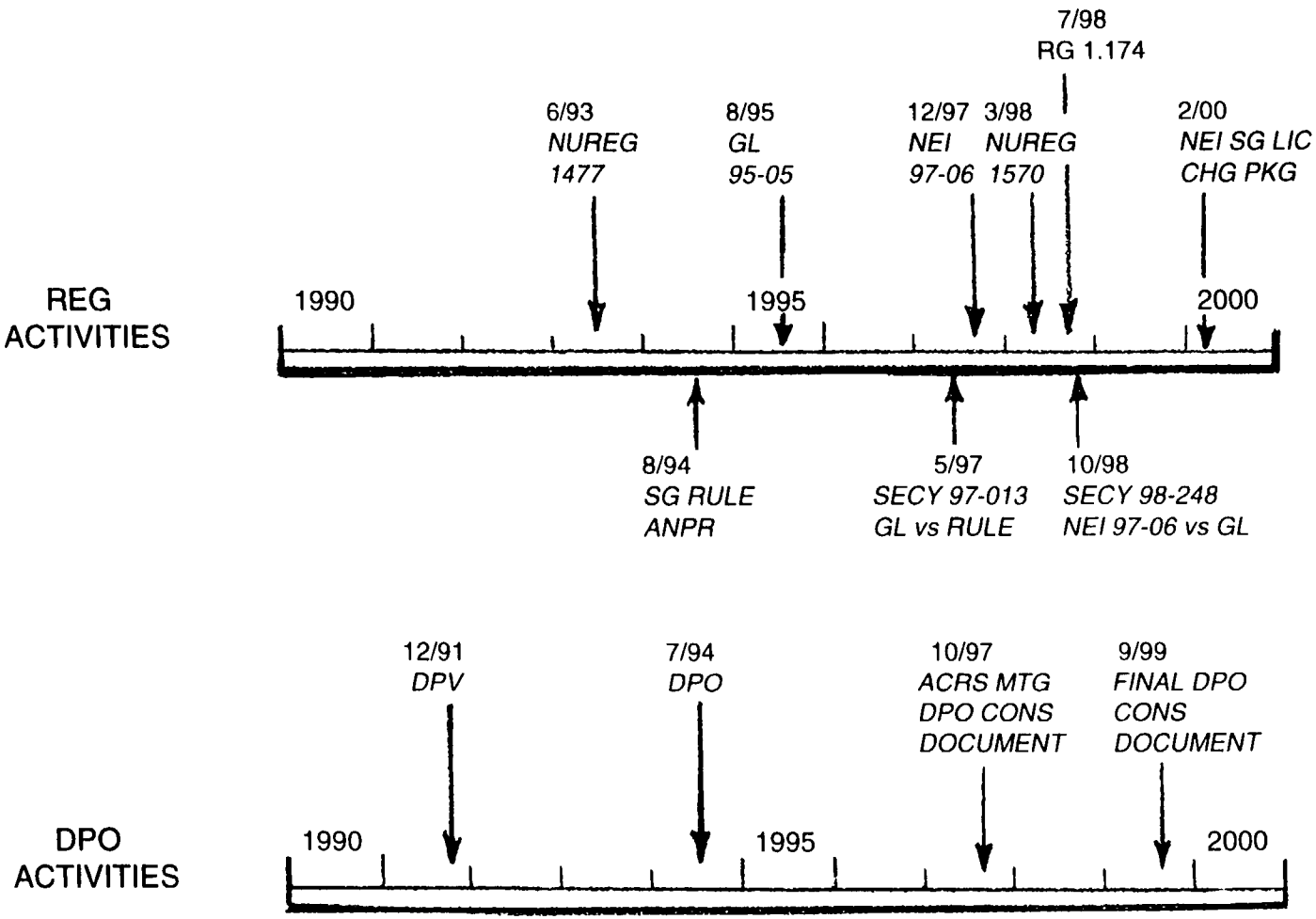
AGENDA

INTRODUCTION

- IODINE SPIKING (2 hrs)
- NDE AND CRACKING PHENOMENA (4 hrs)
- DAMAGE PROPAGATION (2 hrs)
- ACCIDENT ASSESSMENT (4 ½ hrs)
- INTEGRATED DECISION MAKING (1 hr)

SUMMARY

STEAM GENERATOR ACTIVITIES TIME-LINE



APPROACHES TO RESOLUTION

- SPECIFIC TECHNICAL ANSWERS
- PROCESS

ITEM 9
IODINE SPIKING

JACK HAYES - PROBABILISTIC SAFE TY ASSESSMENT BRANCH
October 12, 2000

Presentation

- DPO Author's Concern
- Staff's Reassessment of Iodine Spiking
- Staff's Conclusions with Respect to Iodine Spiking
- Staff's Assessment of DPO Concern

DPO Author's Premise

Postulate: Reduction of 48 Hour Primary Coolant Activity Level of DE ^{131}I to low activity levels may result in spiking factors > 500

Consequence: Part 100 dose limits may be exceeded

Background

- Iodine Spiking
- Calculation of Releases for a MSLB
- Voltage-base Criteria for SG tubes

What is the Spiking Factor

Increase in release rate from the fuel to primary coolant resulting from a transient.

$$SF = \frac{\text{Release Rate Post-trip}}{\text{Release Rate @steady-state}}$$

Iodine Spiking Background

Release from Fuel Gap caused by Pressure Transients

Typical Occurrences - Power Transients
- Startup/Shutdown

SGTR & MSLB Analyses

Performance of MSLB & SGTR Accident Dose Assessments

Typical Assumptions for Calculating Releases:

- Primary to Secondary Leak Rate = 1 gpm
- Primary Coolant Activity = 1 μCi/g DE ^{131}I
- Spiking Factor = 500

Dose Acceptance Criteria = 30 rem thyroid @EAB, LPZ and control room

Incorporation of the Spiking Factor

Determine release rate from fuel for equilibrium activity level in primary coolant determined ,

e.g. 5 Ci/hr of ^{131}I for 0.4 μCi/g of DE ^{131}I

Application of Spiking Factor, e.g. 500

∴ 2500 Ci/hr ^{131}I released into primary coolant

Voltage-based Criteria

Allows tubes normally required to be plugged to remain in service.

Introduces a new release mechanism into the accident

- accident-induced primary to secondary leakage

Impact of Voltage-Based Criteria

Goal becomes minimize # of plugged tubes

Net increase in primary to secondary leakage

Increased accident-induced primary to secondary leakage requres reduction in primary coolant activity level of DE ^{131}I

Staff's Reassessment of Iodine Spiking

Part of re-assessment of SGTR & MSLB accident analyses methodology

Iodine Spiking Re-assessment

Review of Postma Report

Review of Lewis & Iglesias Report

Review of Adams Articles

Adams & Sattison Article, "Consequences of a SGTR Event"

Spiking Factor - 58 events

Spiking Factor Range - 1.7 - 908

Initial DE ^{131}I = 0.004 μCi/g - 0.943 μCi/g

Spiking Factor > 500 in 3 cases

Two Spiking Factors ≈ 900

Initial DE ^{131}I for 3 cases - 0.013, 0.014 & 0.02 μCi/g

Maximum activity in 3 cases - 3.5 μCi/g

Adams &Sattison Conclusions

Large spiking factors tended to be associated with small coolant activity levels & small iodine release rates.

Spiking factor of 500 in association with a DE ^{131}I is overly conservative for SGTR

Expand data base to reduce uncertainties

Adams & Atwood
"The Iodine Spike Release Rate During a Steam Generator Tube Rupture"

Bound each spiking event

Information based upon LERs

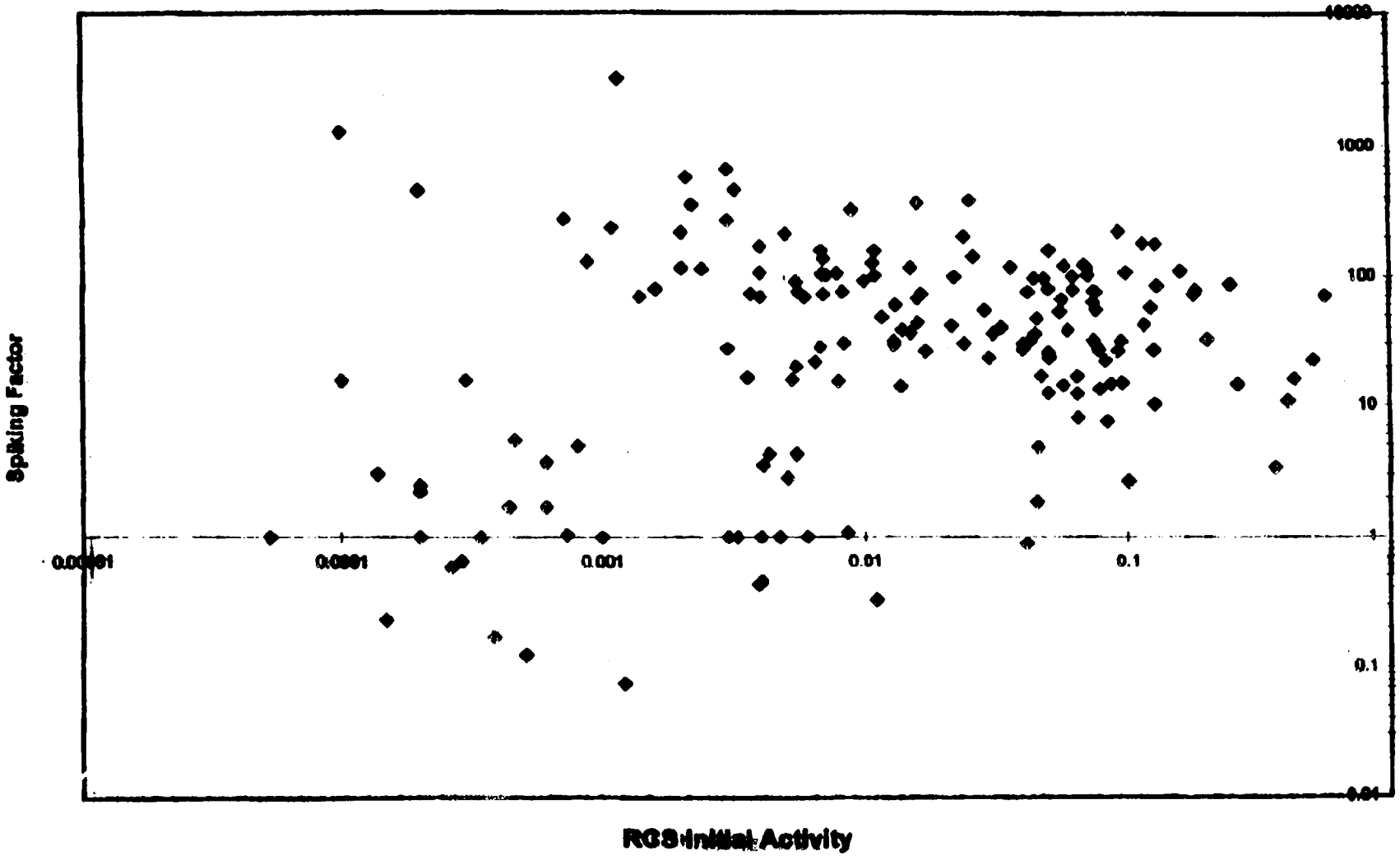
Postulated maximum DE ^{131}I no greater than 3 x value measured between 2-6 hours following the trip

Bounded values multitplied measured values of by 3

Presumed maximum activity occurred 2 hours following trip

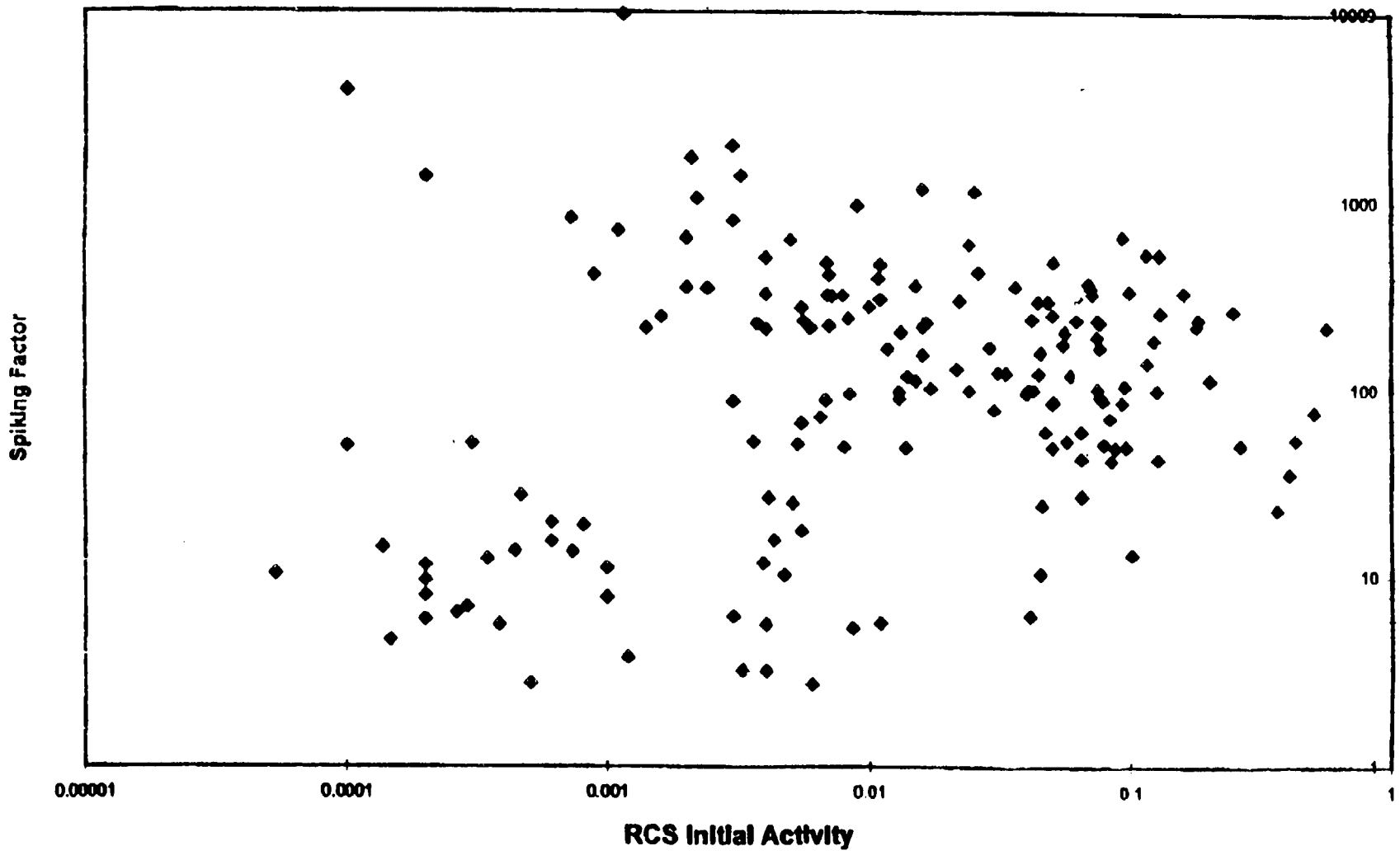
ADAMS Chart 1

Spiking Factor versus RCS Initial Activity



ADAMS Chart 1

Spiking Factor (R(3)/Ro) versus RCS Initial Activity



Adams & Atwood Conclusions

Spiking Factor could be reduced substantially (factor of 15) and still be conservative

Large spikes not because the post-trip release rate is high but because the steady-state release rate is low.

Spiking data representative of SGTR rather than a MSLB

Staff Assessment of Spiking Factor

Reviewed Adams & Atwood data w/o factor of 3

Assessed spiking factor as f(DE ^{131}I)

Staff Conclusion on Iodine Spiking

Spiking factor not f(DE ^{131}I)

Enveloped Spiking Data

95% Value = 335

Staff Conclusions on Spiking for MSLB

Spiking data representative of SGTR

Cannot extrapolate spiking data to a MSLB

Assessment of MSLB Spiking Factor

Linear relationship between Δp rate and resultant iodine activity level or release rate

Esitmate MSLB spiking factor 2-3 x factor for SGTR

Quadrature rather than linear, release rate for a MSLB 4-9 > SGTR.

Relationship between Δp rate and DE ^{131}I unknown

Reasonable assumption - no higher than a quadrature

Estimate primary coolant activity/release rate magnitude for MSLB within a factor of 10 of SGTR

Staff Conclusion on Spiking for MSLB

Uncertainty factor of 10 for MSLB

Adams states spiking factor for SGTR conservative by 15

Two factors offset each other

Absent MSLB data, only recourse - lab tests

Probability of MSLB does not justify research

Staff concludes spiking factor for MSLB should remain 500

Assesment of DPO Concern

Parametric Analysis Performed
Base Case - 3 Loop Westinghouse Plant

Analysis Consistent with SRP 15.1.5 - MSLB

Assumptions

- Primary Coolant = 1μCi/g DE ^{131}I
- Primary to Secondary Leak Rate = 150 gpd (Intact Steam Generators)
- Primary to Secondary Leak Rate = 1290 gpd (Faulted Steam Generator)
- Spiking Factor = 500
- EAB (0-2 hr.) & LPZ (0-8 hr.) Releases Calculated

Other Critical Assumptions

8 hours for faulted steam generator to be isolated

All primary to secondary leakage assumed released directly to the environment.

Spiking assumed to occur for the duration of the accident.

Assumed releases result in a 30 rem thyroid dose

Performance of Parametric Analysis

- Assumed Primary to Secondary leakage rates
  - 10 gpm
  - 35 gpm
  - 100 gpm

Assumed Primary Coolant Activity

- 0.5 μCi/g
- 0.1 μCi/g
- 0.05 μCi/g
- 0.001 μCi/g
- 0.005 μCi/g

Spiking Factors Which Would Result in Releases Equal to Those of the MSLB SRP Case#

RCS Activity Level (µCi/g Dose Equivalent ^{131}I)	Primary to Secondary Leak Rate (gpm total)	Spiking Factors## 0-2 Hour Releases	Spiking Factors## 0-8 hour Releases
0.5	100	*	8.18
	35	12.5	26.6
	10	86.3	98.6
0.1	100	37	57
	35	133	150
	10	502	511
0.05	100	91	118
	35	283	305
	10	1020	1030
0.01	100	526	606
	35	1490	1540
	10	5180	5150
0.005	100	1070	1220
	35	2999	3090
	10	10400	10300

SRP Case based upon reactor coolant activity level of 1 µCi/g of dose equivalent ^{131}I, a spiking factor of 500 and 1 gpm total primary to secondary leak rate.

Spiking factors would need to be multiplied by a factor of 10 in order to equate to a dose of 300 rem thyroid.

* Maximum allowable leak rate would be limited to approximately 60 gpm.

An Example

Assumptions: Primary Coolant Activity - 0.5 μCi/g DE ^{131}I

Primary to Secondary Leak Rate - 35 gpm

Conclusion: If spiking factor > 12.5 release criteria exceeded.

Must reduce primary coolant activity level until spiking factor of 500 obtained

Result: 0.01 μCi/g ≤ Primary Coolant ≤ 0.05 μCi/g

Conclusions from Table

Some combinations of leak rate and primary coolant activity levels require spiking factor ≤ 500

For ARC amendments this would necessitate reducing primary coolant activity levels

Spiking Factor ≥5000 (500 x 300 rem/30 rem) for Dose ≥ Part 100 Limits

For Primary Coolant ≤ 0.01 μCi/g, Spiking Factor must be 5,200- 51,800 to exceed Part 100

Staff's Conclusion on DPO Concern

Spiking Factors > 500 can occur@ low DE ^{131}I activity levels

MSLB Spiking Factor ≤ 5,000

@Primary Coolant DE ^{131}I ≤ 0.01 μCi/g Spiking Factors of 526-5,150 for base case 0-2 hour or 0-8 releases to be exceeded.

At Primary Coolant DE ^{131}I ≤ 0.01 μCi/g, curie content in primary coolant is small as is the equilibrium release rate

Maximum dose for the base case is 10% of Part 100

To exceed Part 100 doses, @DE ^{131}I ≤ 0.01 μCi/g Spiking Factor ≥5,260-51,800

Spiking Factors not that large

LIMITATIONS OF NDE METHODS AND CRACKING PHENOMENA
ACRS Subcommittee on Differing Professional Opinion Issues

October 12, 2000

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OUTLINE OF PRESENTATION

Regulatory framework and operating experience

Capabilities/Limitations of NDE including detection and sizing of degradation

Use of NDE in tube integrity analysis including growth rate analysis and correlations for burst and leakage

CURRENT REGULATORY FRAMEWORK

Requirements/Guidance pertaining to steam generator tube integrity located in 10 CFR 50 Appendices A and B, 10 CFR 100, Regulatory Guide (RG) 1.121, RG 1.83, Standard Review Plan Sections 5.2.3, 5.3.2.1, and 5.4.2.2, ASME Code, and Technical Specifications (TSs)

Present technical specifications (TSs) developed about 25 years ago when prevalent forms of degradation involved general wall thinning

- TSs typically specify a depth-based tube repair criteria (typically 40% through-wall) regardless of degradation mode

Requirements for tube inspection and repair and normal operating primary-to-secondary leakage limits are contained in technical specifications

TYPICAL SG TECHNICAL SPECIFICATION REQUIREMENTS

Sampling program in TSs

- 3% initial sample which is expanded based on inspection classification (C-1, C-2, C-3)
- Re-examination of previously degraded areas

Frequency of Inspection

- Once every 12 to 24 calendar months
- May be lengthened to 40 months. May need to be shortened to 20 months
- After tube leak in excess of limits, seismic occurrence, LOCA, major SLB

Inspection Extent

- Hot leg and U-bend are required. Cold leg inspections "recommended" in GL 85-02 and in EPRI Guidelines

Inspection Techniques

- Not specified but licensees primarily use eddy current testing (ECT)

Repair Criteria

- Typically require plugging at 40% through-wall - applicable to all degradation mechanisms

TYPICAL TUBE PLUGGING/REPAIR LIMITS

Tube repair criteria developed using Regulatory Guide 1.121 methodology

Regulatory Guide 1.121 lists various structural and leakage criteria the tubes should meet

- Typically most limiting structural criteria is the tube should be able to withstand a differential pressure of 3 times the normal operating pressure differential ($3\Delta P_{NO}$) or 1.4 times the main steam line break (MSLB) differential pressure

- Tube leakage criteria should be based on "limiting crack length"

Tube repair criteria based on uniform thinning of tube wall and must include allowances for flaw growth and measurement uncertainty

- Determine tube wall thickness necessary to meet structural criteria ($3\Delta P_{NO}$ or 1.4xMSLB) - this is approximately 40% TW (i.e., 60% of tube wall could be "eliminated" if no measurement error and no progression of degradation feasible)

- Margin for measurement uncertainty (~10% TW)

- Margin for degradation growth between inspections (~10% TW)

ISSUES WITH CURRENT REGULATORY FRAMEWORK

Major goal of SG tube inspections is to ensure tube integrity for the operating interval between inspections (structural integrity per RG 1.121 and ASME Code, leakage integrity per Part 100 and GDC 19)

Technical Specifications do not reflect either current degradation modes or inspection technology and are inappropriate for some forms of degradation

- Repair criteria (i.e., 40%) tends to be conservative for cracks
- Inspection sample size, expansion criteria, and frequency does not explicitly take into consideration the severity of the degradation (TSs mainly based on number of tubes with degradation)

Leakage limits do not prevent tube bursts

"NEW" REGULATORY FRAMEWORK

Industry and NRC have recognized for years the issues associated with the current regulatory framework

Industry performance has improved as a result of their (and NRC) efforts

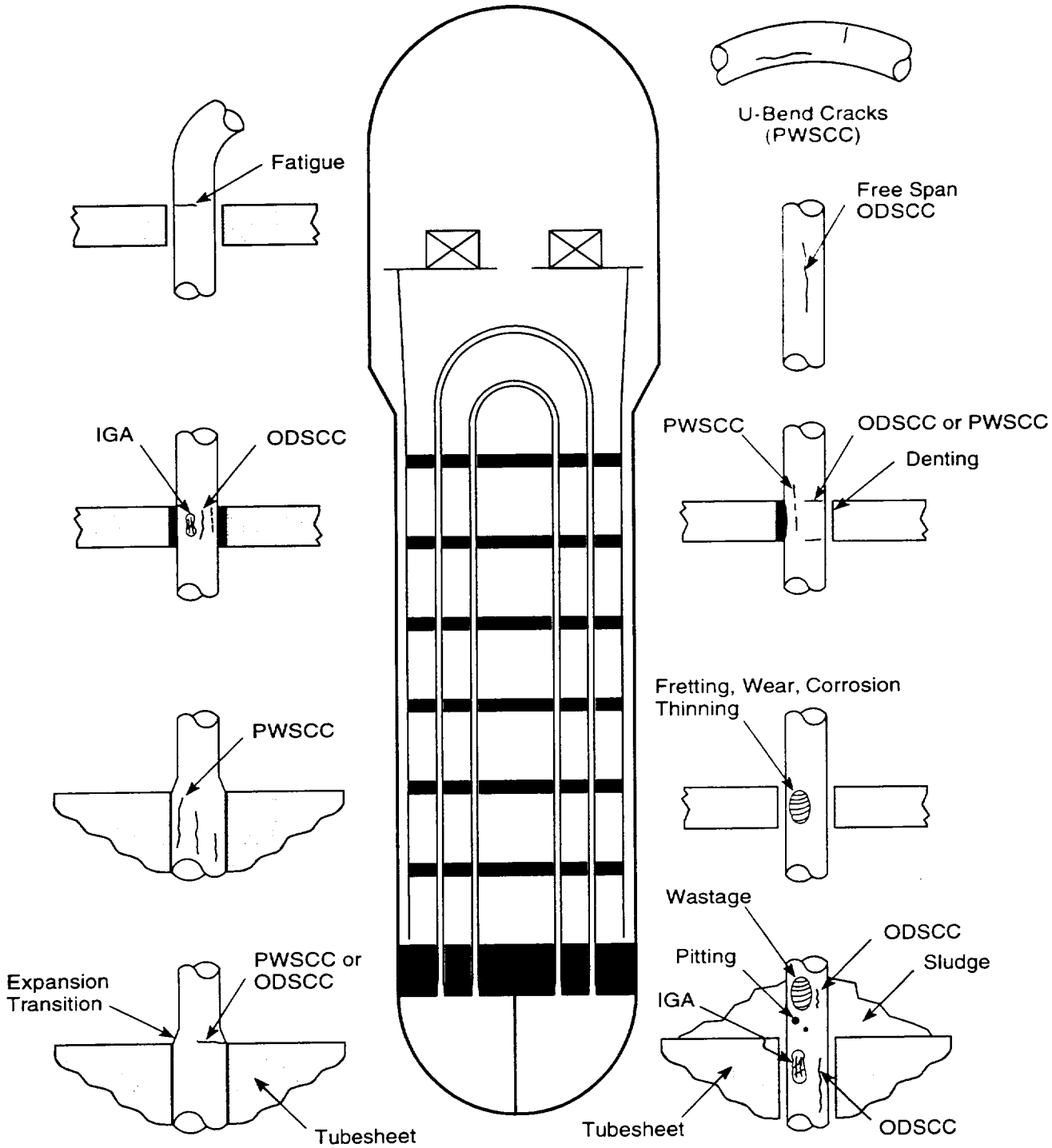
EPRI Examination Guidelines	Generic Letters
SGMP	Information Notices
CMOA Philosophy	Research

NEI developed SG guidelines: NEI 97-06

NRC developed draft Regulatory Guide 1074 (DG-1074) addressing tube integrity

Industry initiative

EXAMPLES OF SG TUBE DEGRADATION MECHANISMS



FACTORS AFFECTING TUBE DEGRADATION

Tube material including heat treatment: alloy 600 mill annealled and thermally treated, alloy 690 thermally treated

Tube microstructure: grain size, carbide distribution, etc.

Fabrication of tubes and stresses - expansion joints (partial and full depth hard roll expanded, full depth explosive, full depth hydraulic)

Tube support plate design/material

Operating temperature and stresses (pressure)

Water chemistry including secondary system design (copper, condenser, etc.) - sludge pile and denting

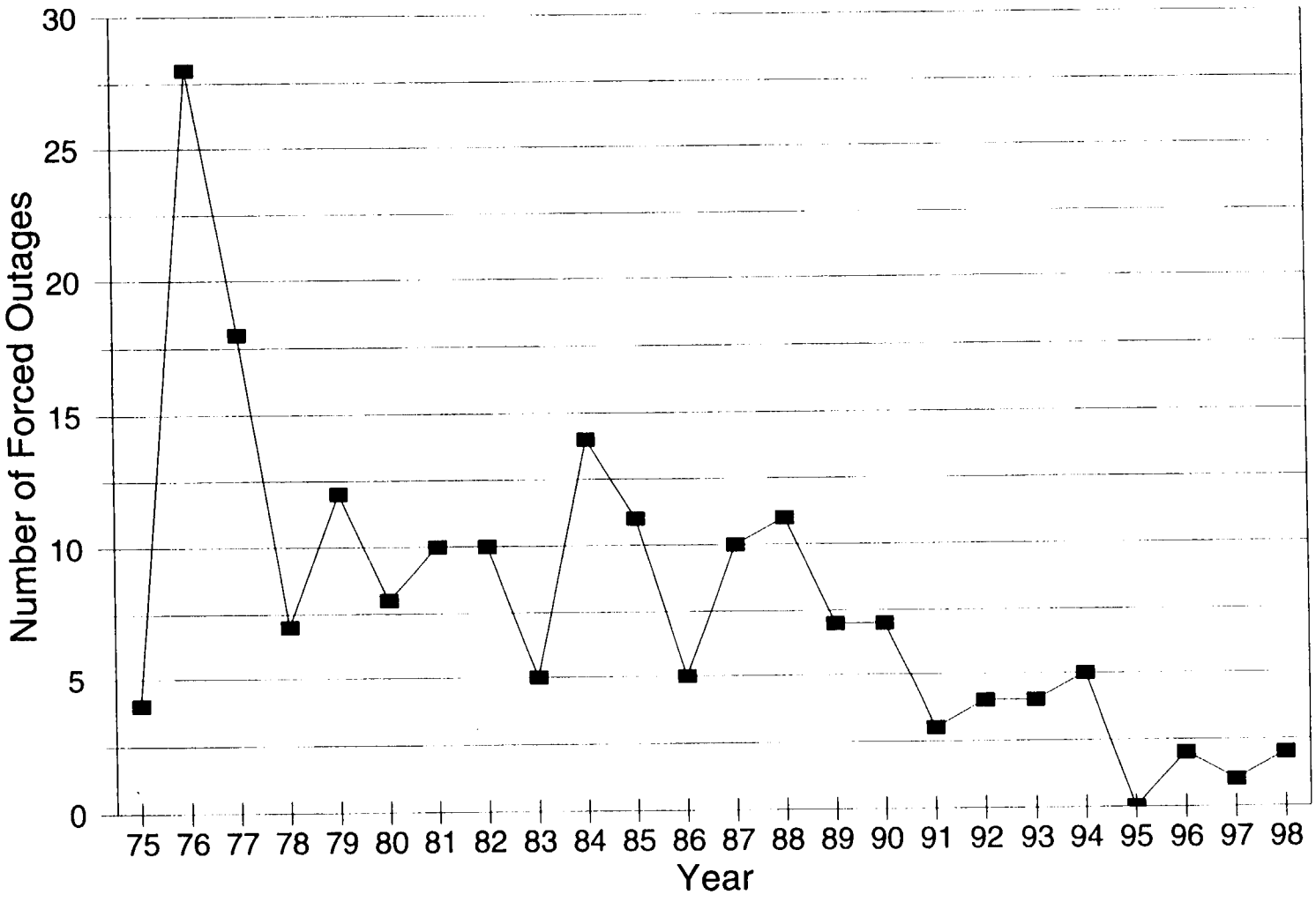
Operating time

Crevices

SGTR EVENTS

PLANT	DATE	LEAKAGE RATE	MECHANISM
Point Beach 1	2/75	125 gpm	Wastage and caustic SCC
Surry 2	9/76	330 gpm	Axial PWSCC - U-Bend
Doel 2	6/79	135 gpm	Axial PWSCC - U-bend
Prairie Island 1	10/79	390 gpm	Foreign Object
Ginna	1/82	760 gpm	Foreign Object
North Anna 1	7/87	550 to 650 gpm	Fatigue - U-Bend Tangent (TSP)
McGuire 1	3/89	500 to 600 gpm	Axial ODSCC - Free Span
Mihama 2	2/91	~700 gpm	Fatigue - U-Bend Tangent (TSP)
Palo Verde 2	3/93	240 gpm	Axial ODSCC - Free Span
Indian Point 2	2/00	150 gpm	Axial PWSCC - U-Bend

US SG Tube Leak Forced Outages



SG TUBE LEAKAGE EVENTS

Number of plants shutting down as a result of primary-to-secondary leakage during normal operation has generally decreased over the years

Some shutdowns initiated prior to exceeding technical specification leakage limits

Some plants have operated with leakage until refueling outage

Causes of shutdowns due to leakage in 1990s include:

- Sleeves (B&W kinetically expanded)
- Plugs (alloy 600 - NRC Bulletin 89-01)
- Loose parts
- Fatigue (B&W lane/wedge region)
- SCC (expansion transition, tube supports, freespan)

STEAM GENERATOR REPLACEMENTS

Plant Name	No. of Loops	SG Manufacturer/Model		Completion Date
		Original	Replacement	
Surry 2	3	W/51	W/51F	9/80
Surry 1	3	W/51	W/51F	7/81
Turkey Point 3	3	W/44	W/44F	4/82
Turkey Point 4	3	W/44	W/44F	5/83
Point Beach 1	2	W/44	W/44F	3/84
Robinson 2	3	W/44	W/44F	10/84
Cook 2	4	W/51	W/54F	3/89
Indian Point 3	4	W/44	W/44F	6/89
Palisades	2	CE	CE	3/91
Millstone 2	2	CE-67	BWC	1/93
North Anna 1	3	W/51	W/54F	4/93
Summer	3	W/D3	W/D75	12/94
North Anna 2	3	W/51	W/54F	5/95
Ginna	2	W/44	BWC	6/96
Catawba 1	4	W/D3	BWC	9/96
Point Beach 2	2	W/44	W/D47	12/96
McGuire 1	4	W/D2	BWC	5/97
Salem 1	4	W/51	W/F	7/97
McGuire 2	4	W/D3	BWC	12/97
St. Lucie 1	2	CE-67	BWC	1/98
Byron 1	4	W/D4	BWC	1/98
Braidwood 1	4	W/D4	BWC	11/98
STP-1	4	W/E	W/D94	5/00
Farley-1	3	W/51	W/54F	5/00
Cook -1	4	W/51	BWC	TBD
ANO-2	2	CE/2815	W/D109	In Progress
Indian Point-2	4	W/44	W/44F	In Progress

PLANNED SG REPLACEMENTS

A number of plants have indicated they plan on replacing their SGs

SG replacement and license renewal

SGs from each of the 3 major U.S. vendors are planned for replacement in the next 10 years

SG ALTERNATE TUBE REPAIR CRITERIA

Extensive tube degradation led to development of alternate tube repair criteria in 1980s

Alternate tube repair criteria in tubesheet region: F*, L*, W*

Alternate tube repair criteria for predominantly axially oriented outside diameter stress corrosion cracking (ODSCC) at tube support plates

Beaver Valley 1/2, Comanch Peak 1, Diablo Canyon 1/2, Farley 2, Kewaunee, Prairie Island 1/2, Sequoyah 1/2 , South Texas 2

Alternate tube repair criteria for axially oriented primary water stress corrosion cracking (PWSCC) at/near dented tube support plates

Sequoyah 1 /2 (2 cycles)

SG TUBE INSERVICE INSPECTION TECHNIQUES

Eddy current testing (ECT)

Bobbin probe - relatively fast (24 to 48 in/sec), but relatively insensitive to circumferentially oriented degradation and poor at characterizing degradation

Rotating probes (pancake and plus-point) - slow (~0.1 to 0.6 in/sec), but able to detect circumferentially oriented degradation and good at characterizing degradation - better inspection of regions with geometric changes (dents, expansion transitions, and U-bends)

Cecco Probe (array probe) - medium speed (12 in/sec) "in-between" rotating and bobbin probe

Ultrasonic Testing - not used for production inspections in U.S. - occasionally used as supplemental technique

EVOLVING TECHNOLOGY AND DEGRADATION MECHANISMS

Bobbin coil probe was primary inspection tool in 1970s

- Single frequency, good at detecting general wall loss types of degradation (wastage, thinning, wear, pitting)

Multiple frequency bobbin coil and rotating pancake coil probe in late 1970s and 1980s

- Multiple frequencies allowed optimizing techniques to mix out unwanted signals
- Rotating pancake coil probe (mid-to-late 1980s) better at detecting and characterizing stress corrosion cracking in locations with geometry changes - initially used at expansion transitions

Widespread use of rotating pancake coil probe began in late 1980s and early 1990s (inspection of expansion transitions)

Plus-point coil use emerged in mid 1990s

- 1st major application at Maine Yankee - better at detecting and characterizing degradation

New probes (e.g., X-probe and array probe) and data analysis software and techniques (e.g., ANL) continue to be developed

FLAW DETECTION HISTORY 1

Economics and regulatory concerns resulted in improvement of techniques

A large number of leaker/mid-cycle outages were occurring in 1970s/early 1980s

- Technique capability???
- Analyst reliability???
- High growth rates???

Evaluation of laboratory prepared tube specimens and removed tube samples from the Surry steam generator by Pacific Northwest Laboratory (under contract by NRC)

- Determined probability of detection (POD) for stress corrosion cracking (SCC) from laboratory samples to be 0.6

FLAW DETECTION HISTORY II

In 1980s/early 1990s industry develops ISI guidelines pertaining to sizing

- Minimum detection capability (80% POD at 90% confidence)
- Analyst qualification (qualified data analyst program in mid 1990s))
- "Generic" vs "plant specific" qualification

In mid-to-late 1990s, industry and NRC develop additional guidelines

- Plant-specific considerations
- Technique versus system

Evaluation of steam generator mockup samples by Argonne National Laboratory (under contract by NRC) in 2000

FACTORS AFFECTING DETECTION

Essential variables

Equipment variables (system (e.g. MIZ-30), probe size, probe type, cable type and length)

Technique variables (frequencies, drive voltage, coil excitation mode, calibration method, digitizing rate, scan pattern (pitch and direction))

Analysis variables (method of calibration including mixes), data review requirements, reporting requirements, system for analysis (e.g., algorithms used for data analysis))

Analyst reliability

Plant specific considerations for detection include

deposits (conducting, non-conducting, ferromagnetic)

dents and/or geometry changes (location of degradation)

support structures (TSPs, top of tubesheet)

crack orientation/morphololgy (axial/circumferential, PWSCC/ODSCC)

noise (electrical and tube)

SPECIFIC FLAW/TECHNIQUE DETECTION CONSIDERATIONS I

Industry qualifies specific probes (e.g., plus-point) with specific sizes to detect specific degradation mechanisms (circumferential PWSCC) under specific circumstances (frequency) at specific locations (dents)

- Each "qualified" technique has a list of essential variables

Axial versus circumferential cracks

- Bobbin probe will not reliably detect circumferential cracks/degradation
- Bobbin probe and axial degradation (dents, U-bends, expansion transitions)
- Pancake coil probe and plus-point coil probe are qualified for detecting axial and circumferential cracks

Isolated cracks versus cracks in cluster

- In general, easier to detect larger volume loss (e.g., cracks in cluster) but characterization of flaw more difficult

SPECIFIC FLAW/TECHNIQUE DETECTION CONSIDERATIONS II

"Plugged" cracks

To some extent depends on nature of deposits - if conducting, will lower ECT response making it more difficult to detect. Ferromagnetic deposits will increase ECT response

If deposits not conducting or ferromagnetic and no significant effect on tube noise, detection thresholds should be similar from tube-to-tube

Crack location relative to TSPs, bends,....

More difficult to detect; however, techniques are qualified for the particular situation and sometimes need a "specialized" (i.e., non-bobbin) probe

Operation speed (linear speed thru tubes)

Utilities have assessed effects of increased speeds - "blind" tests

Increased speed (12 to 24"/sec) sometimes reduces noise and makes detection comparable

Too fast a speed can cause probe "jumping" if run into geometry change - noise problem, missed data, frequency effects if improper compensating software

Appendix H Technique Qualification

WEAR 400/100 DIFF AMP MIX - BOBBIN
FRACTION DETECTED & POD @ 90% C/L

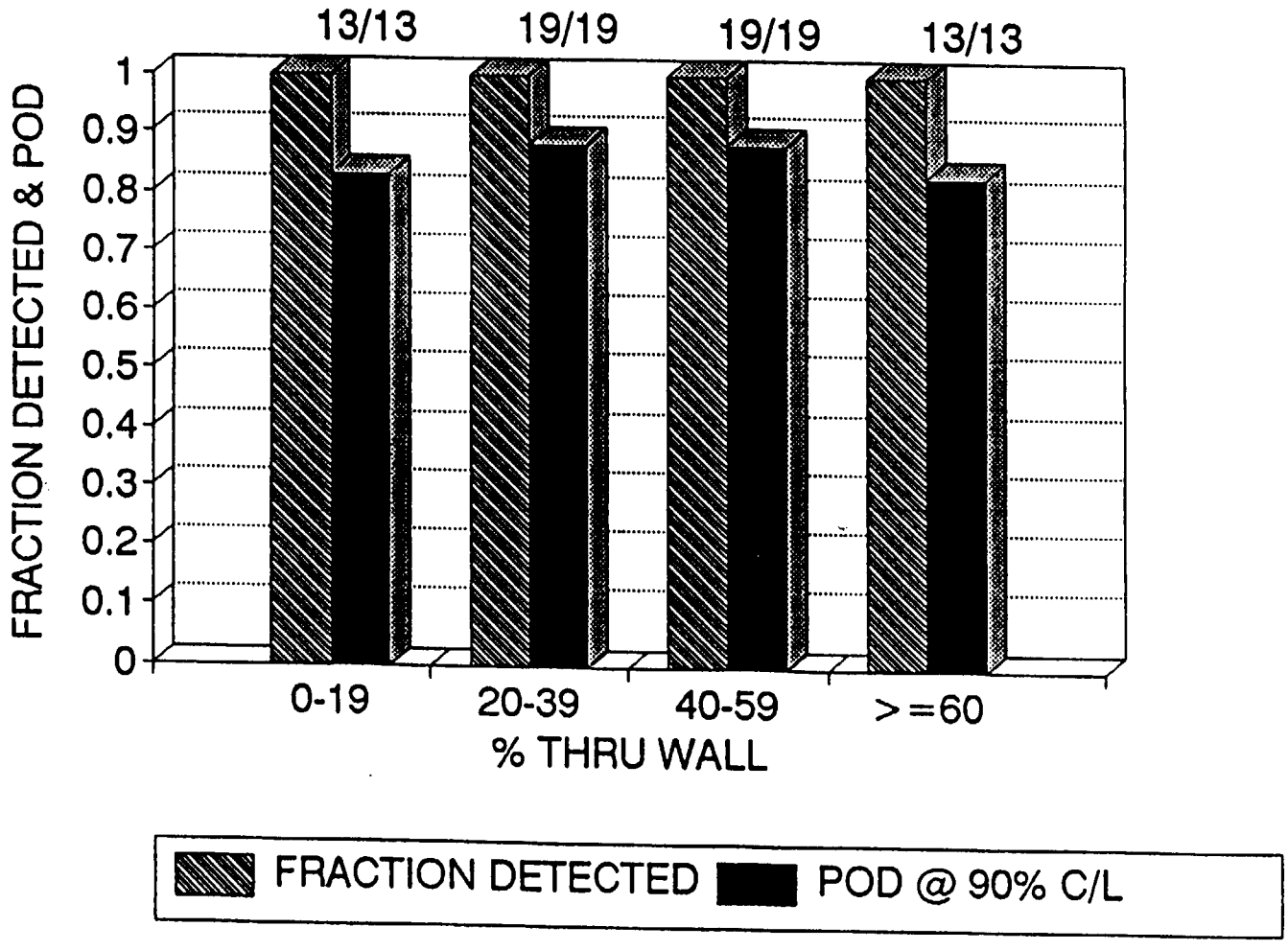


Figure 4.3-1. Indian Point-2 Sludge Pile POD vs. Maximum Depth + Point Coil, Log Logistic Function

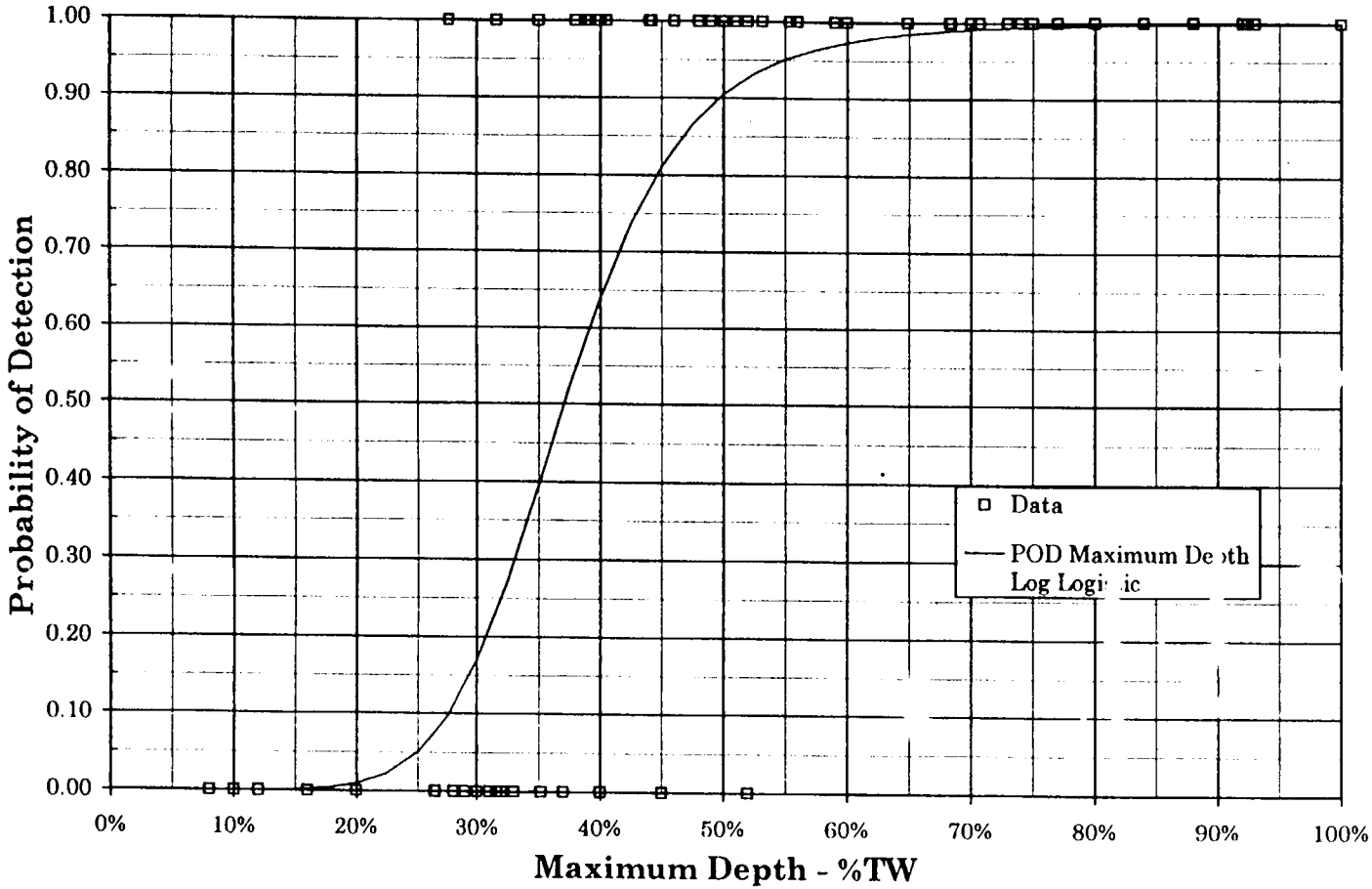
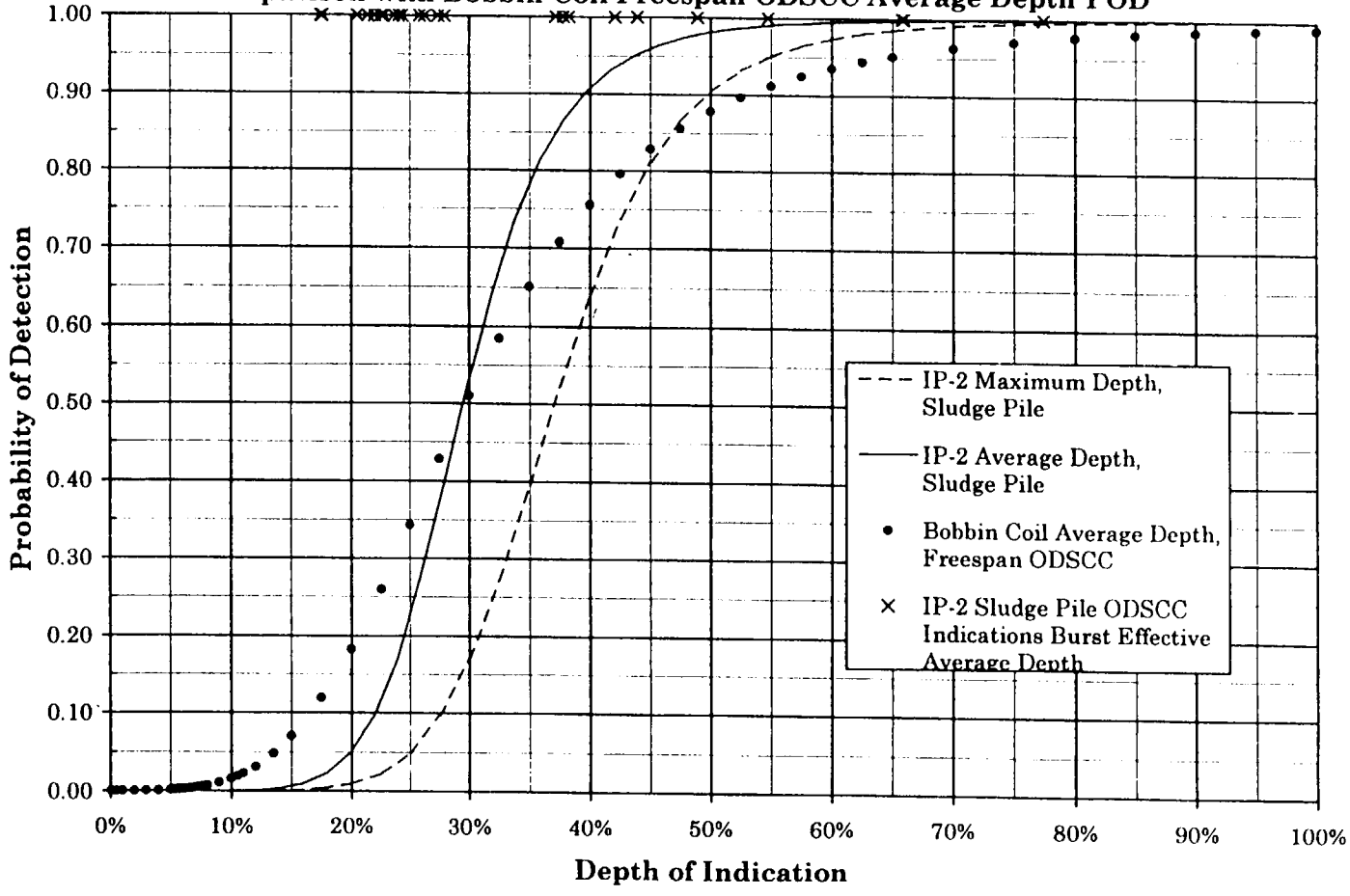


Figure 4.3-2. Indian Point-2 Sludge Pile PODs for Max. and Avg. Depth + Point Coil, Log Logistic Function Comparison with Bobbin Coil Freespan ODSCC Average Depth POD



Appendix H Technique Qualification

ODSCC PRIME/QTR DIFF MIX-BOBBIN FRACTION DETECTED & POD @ 90% C/L

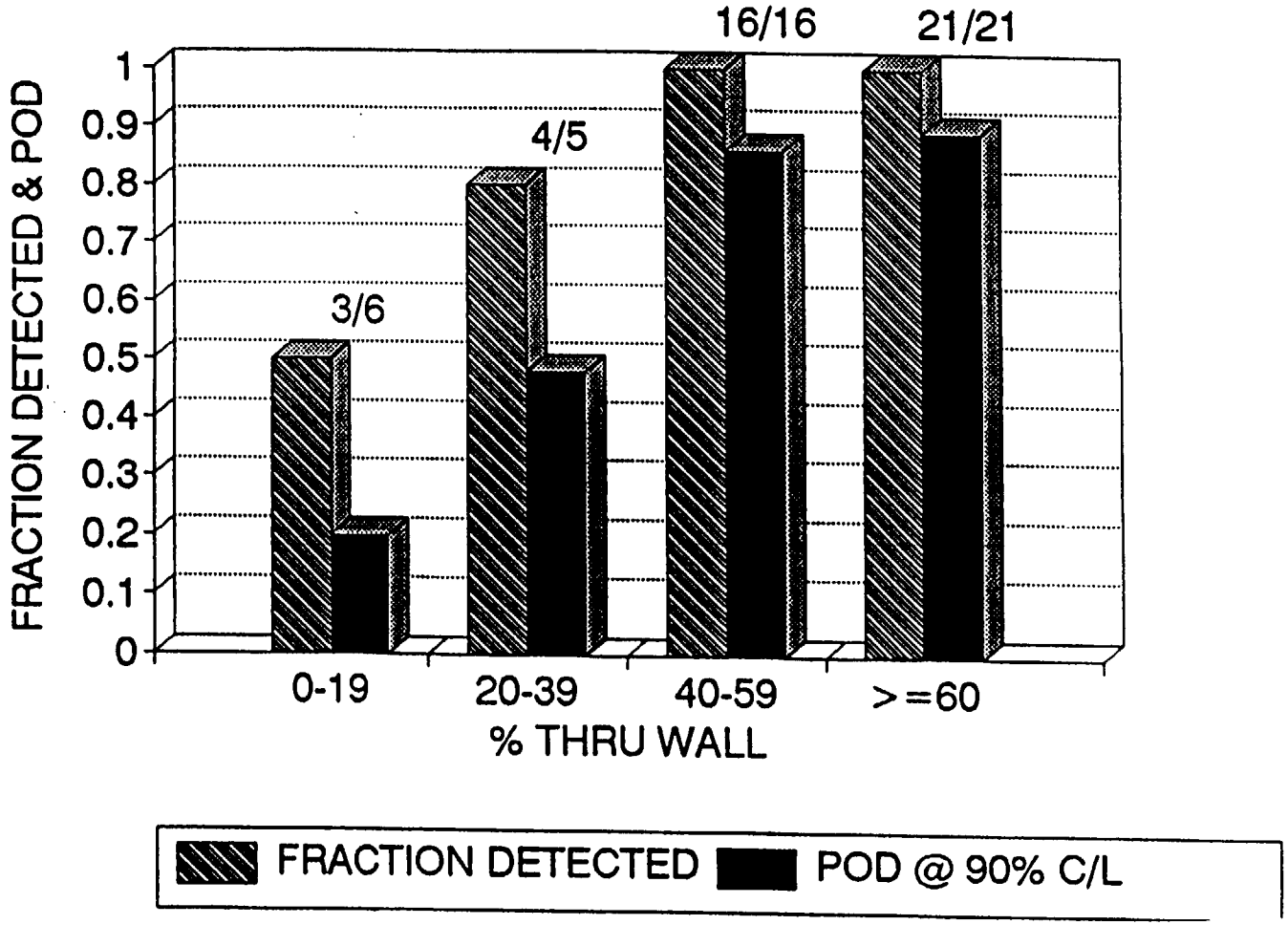
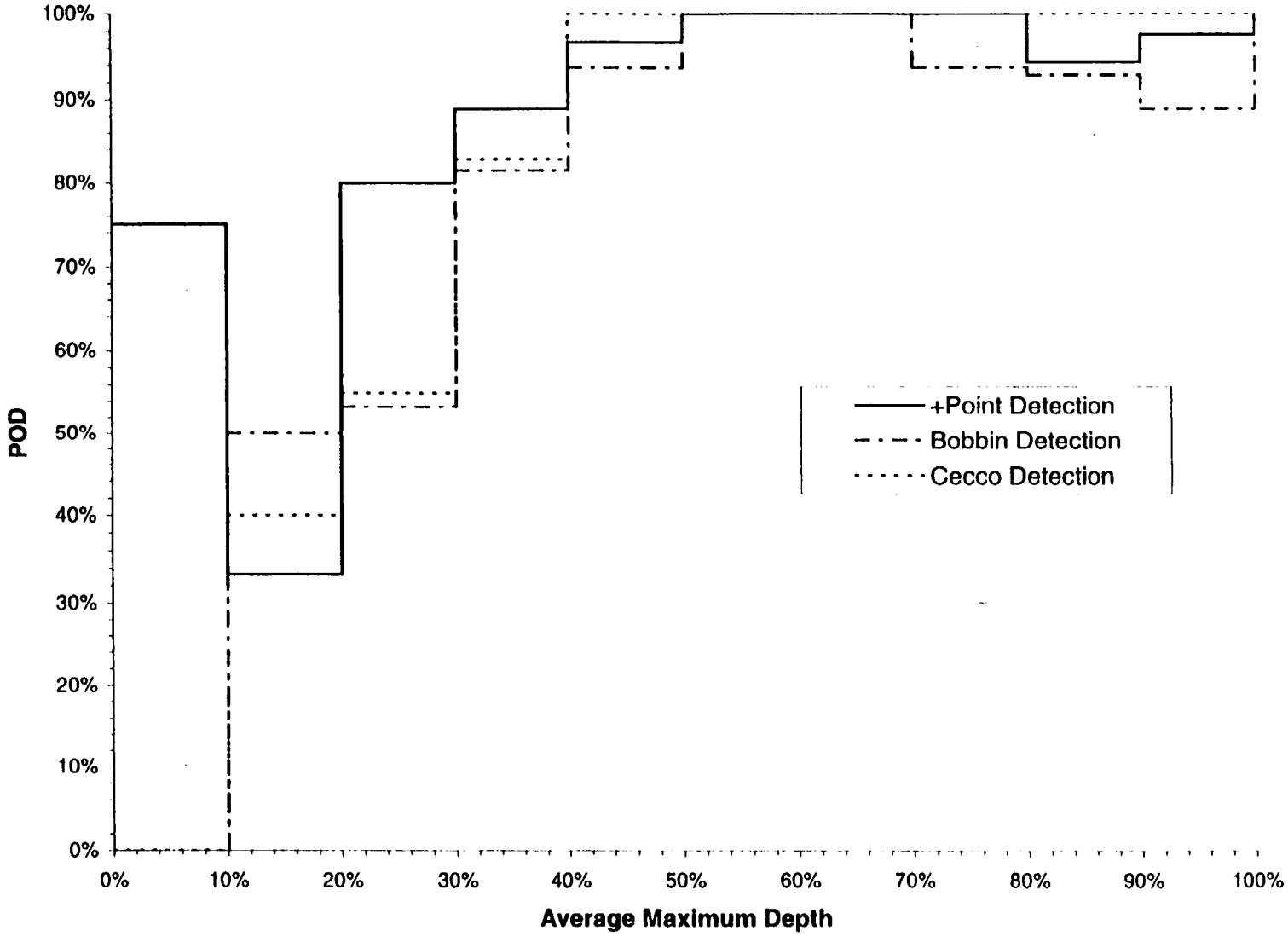
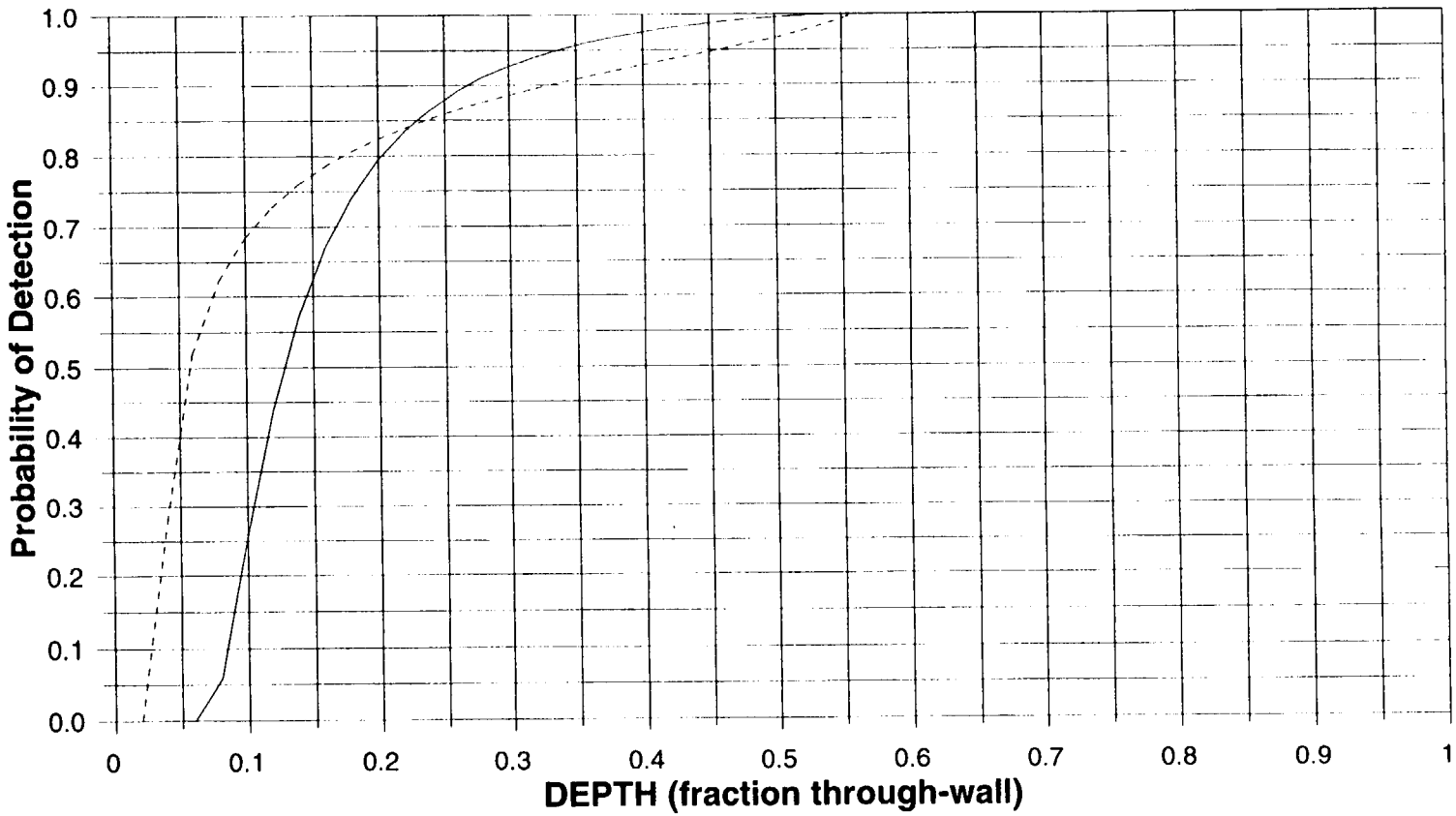


Figure 5-1
Blind Analyses Test Results
Average +Point POD as a Function of Avg. Maximum Depth



BOBBIN AND +Pt POD

PWSCC AT DENTED TSP INTERSECTIONS



Bobbin (Average Depth) +Point (Average Depth)

USE OF PROBABILITY OF DETECTION (POD)

Probability of detection (POD) curves reviewed and approved by NRC for alternate tube repair criteria, as applicable

- Used in GL 95-05 tube repair criteria (axial ODSCC at TSPs) and for axial PWSCC at dented TSP repair criteria

In both cases, staff uses a POD of 0.6 regardless of depth/severity of degradation

- Based on PNL round-robin study of laboratory specimens in 1980s (20% to 100% through-wall SCC)

How is the POD applied in methodology?

- Used not only to account for missed indications but also potentially new indications

Is 0.6 the correct value?

- Industry proposals/data
- NRC perspective (operational assessments)
- Conservative given inspection techniques today - based on GL 95-05 operational assessment data

POD ISSUES

Lab data versus field data

Overall system versus technique capability

Role of plant-specific factors including noise

POD - a function of average depth, maximum depth, and/or structurally significant depth

False calls

SIZING VERSUS DETECTION

A technique may be qualified for detecting degradation but not for sizing it

Utilities routinely used bobbin coil for detection and rotating probes for characterizing (including sizing) degradation

Sizing can be in terms of length (axial or circumferential), depth (average or maximum), or voltage

QUALIFIED SIZING TECHNIQUES

What constitutes qualification???

- Root Mean Square Error Approach
- Understanding uncertainties and applying in condition monitoring and operational assessments

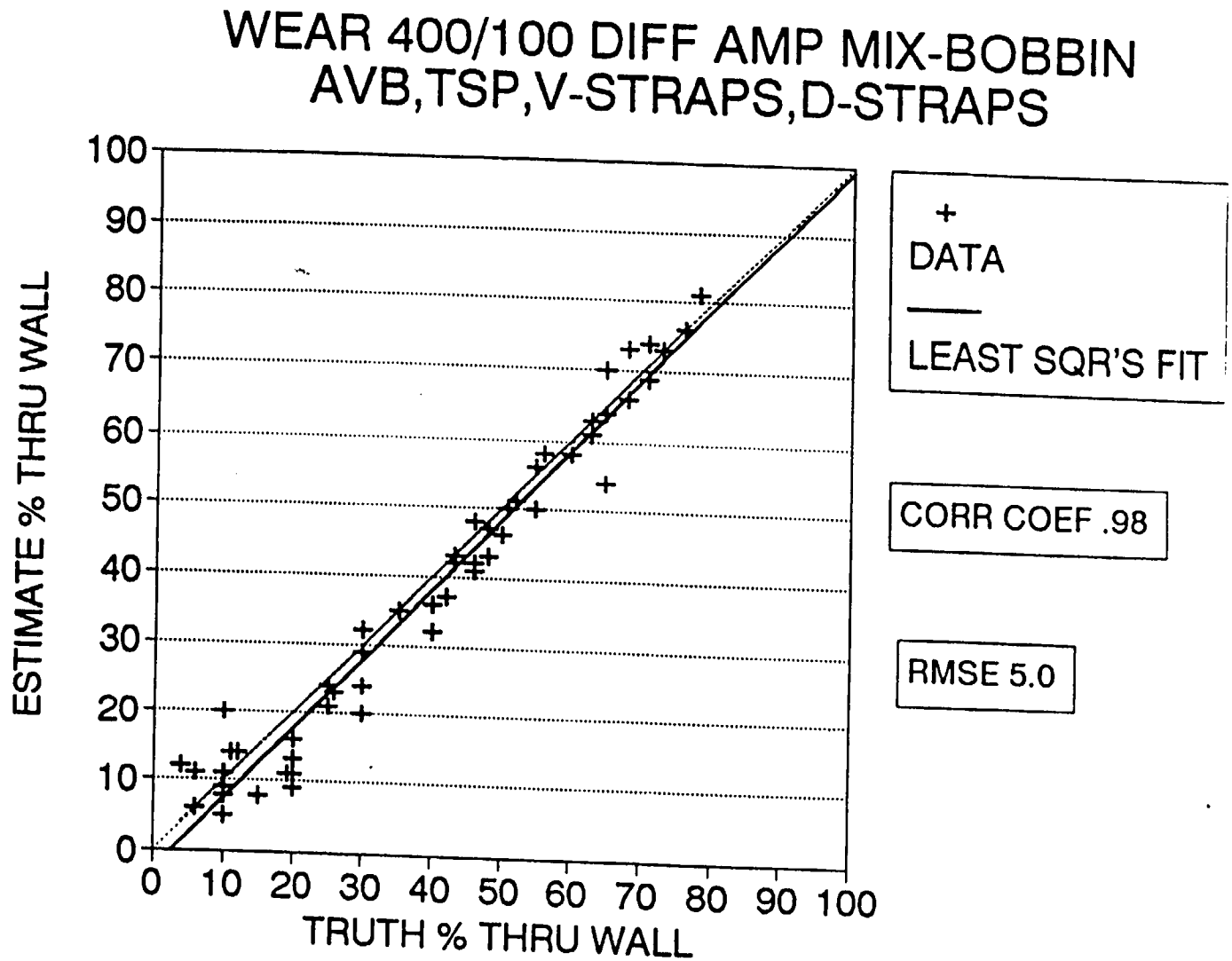
Bobbin coil generally considered as a "qualified" sizing technique for wear, thinning, and pitting

NRC has approved models for voltage sizing predominantly axially oriented SCC at tube support plate (TSP) elevations with a bobbin coil (i.e., GL 95-05)

- 2 components of model: probe wear and analyst variability

NRC has approved use of plus-point coil for depth sizing axially oriented PWSCC at dented TSP elevations

Appendix H Technique Qualification



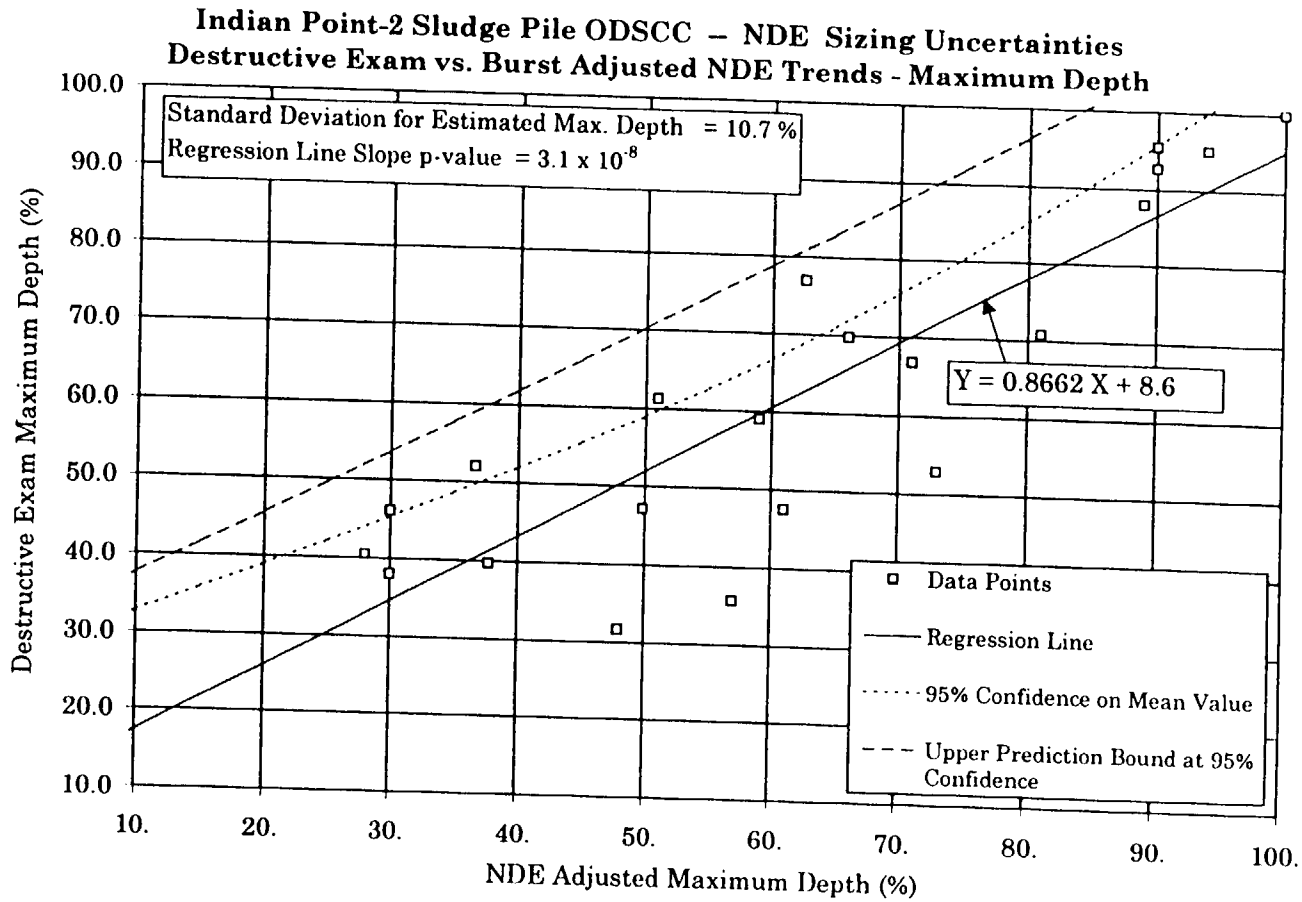
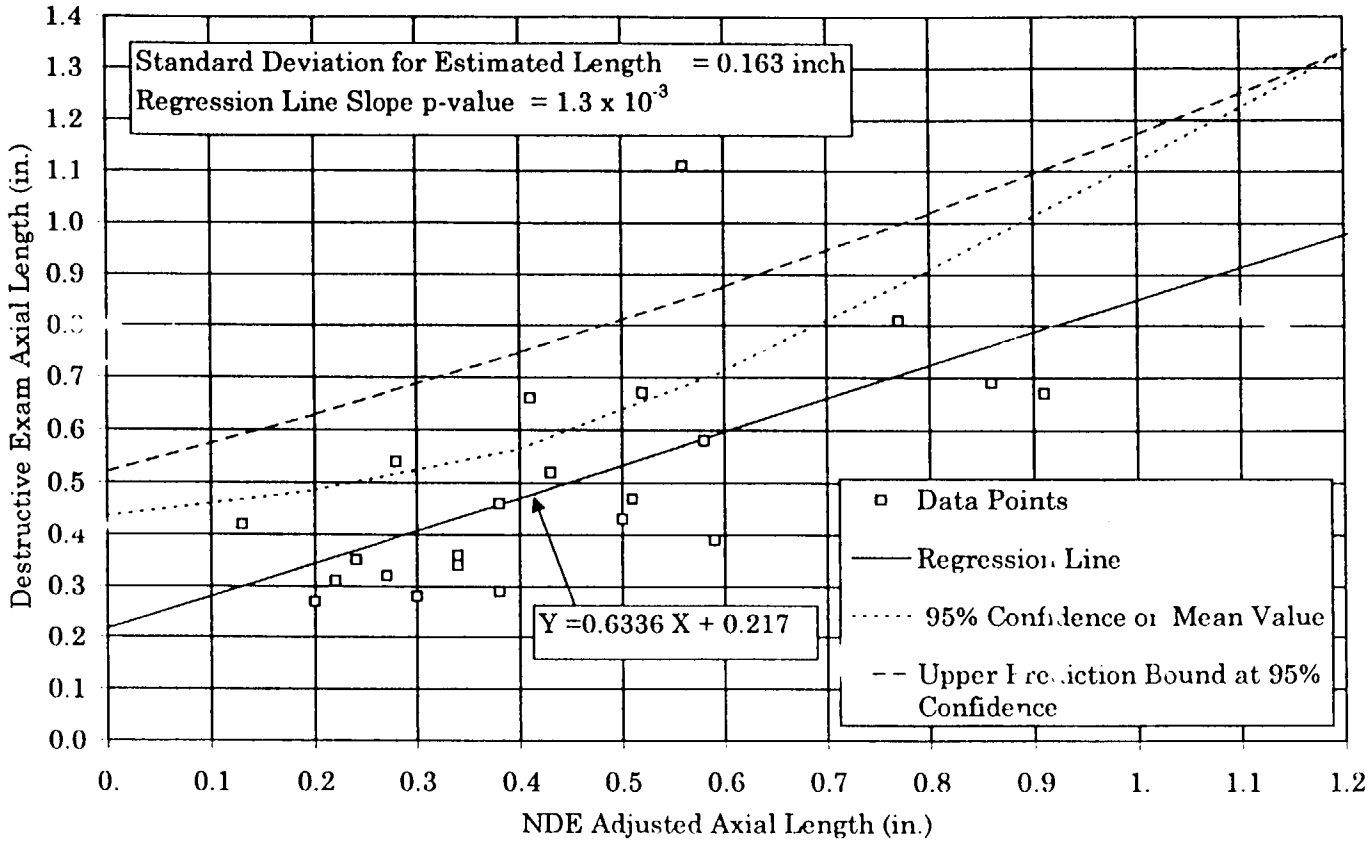
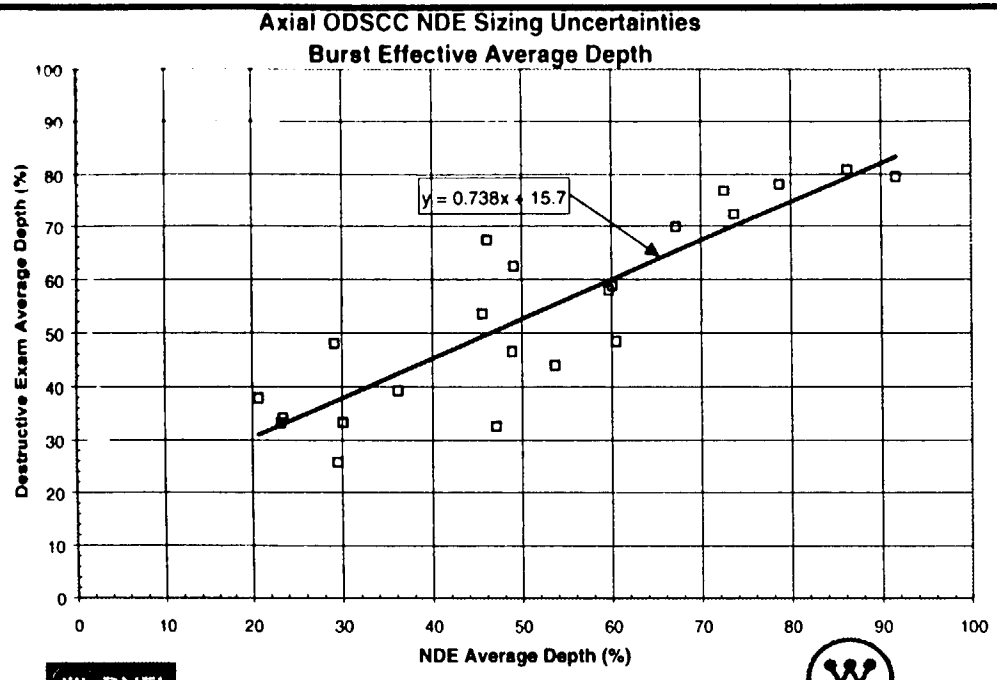

Indian Point-2 Sludge Pile ODSCC – NDE Sizing Uncertainties
Destructive Exam vs. Burst Adjusted NDE Trends - Maximum Depth
Standard Deviation for Estimated Max. Depth = 10.7 %
Regression Line Slope p-value = 3.1 x 10^-8
Y = 0.8662 X + 8.6
Data Points
Regression Line
95% Confidence on Mean Value
Upper Prediction Bound at 95% Confidence
Destructive Exam Maximum Depth (%)
NDE Adjusted Maximum Depth (%)
100.0
90.0
80.0
70.0
60.0
50.0
40.0
30.0
20.0
10.0
10.
20.
30.
40.
50.
60.
70.
80.
90.
100.

Figure 4.4 -1

Indian Point-2 Sludge Pile ODSCC – NDE Sizing Uncertainties
Destructive Exam vs. Burst Adjusted NDE Trends - Axial Length

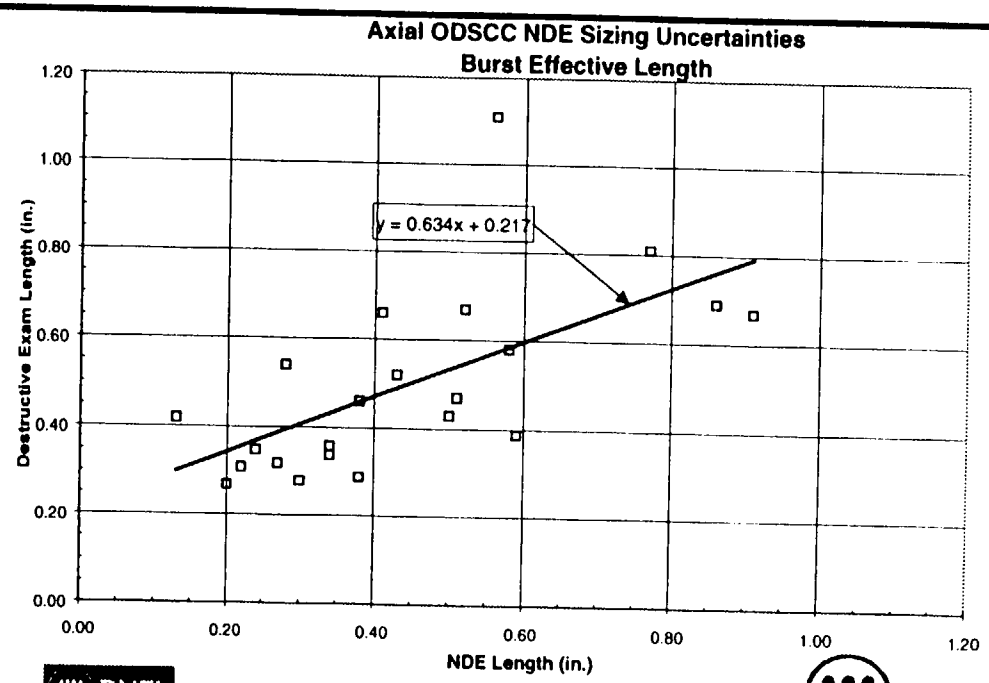


Average Depth NDE Sizing Correlation



BNFL

Length NDE Sizing Correlation



BNFL

Figure 7-1
APPENDIX H QUAL - Dented TSP Axial PWSCC
Average Depth Destructive Exam vs. NDE Trends

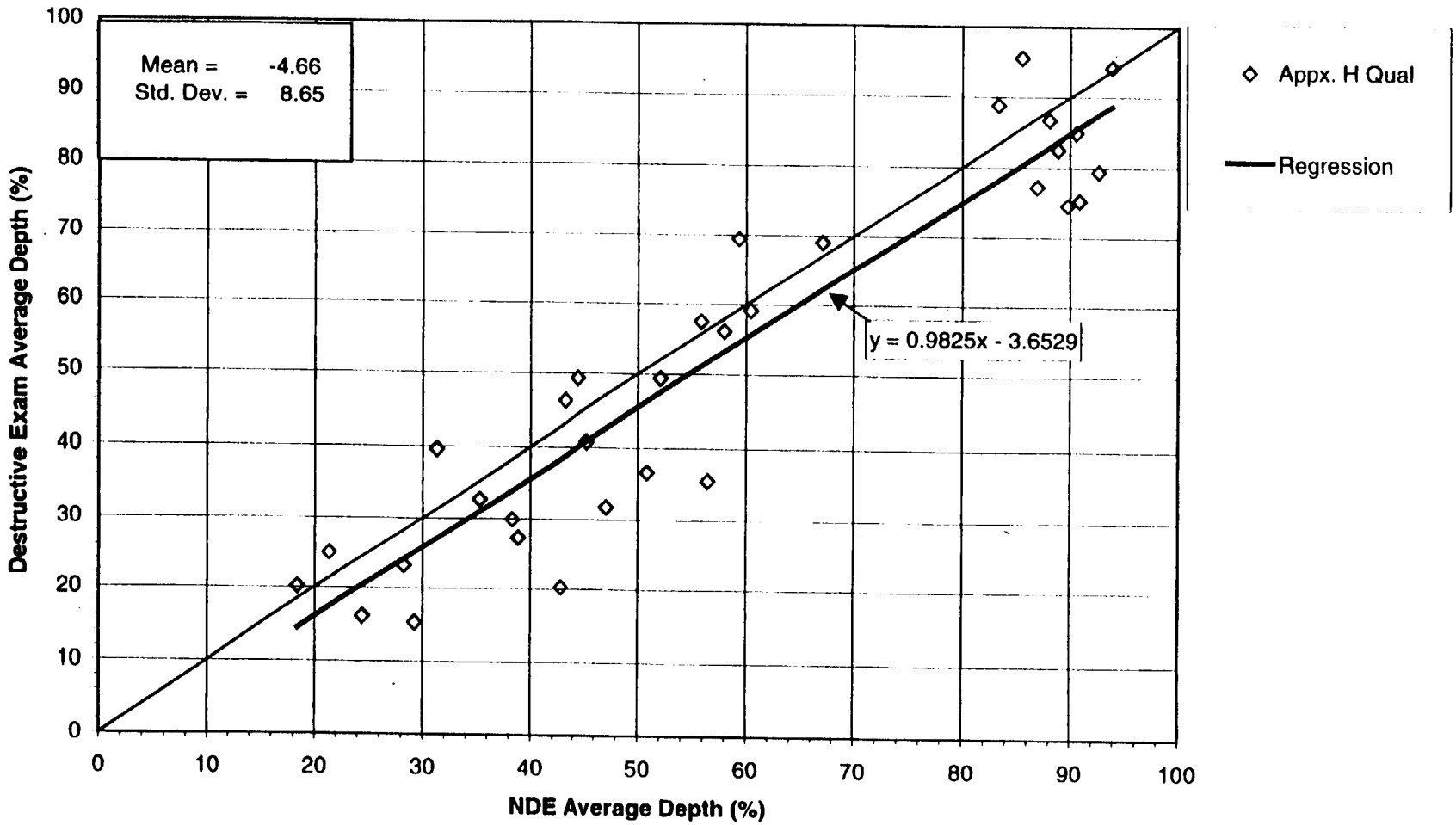


Figure 7-3
APPENDIX H QUAL - Dented TSP Axial PWSCC
Maximum Depth Destructive Exam vs. NDE Trends

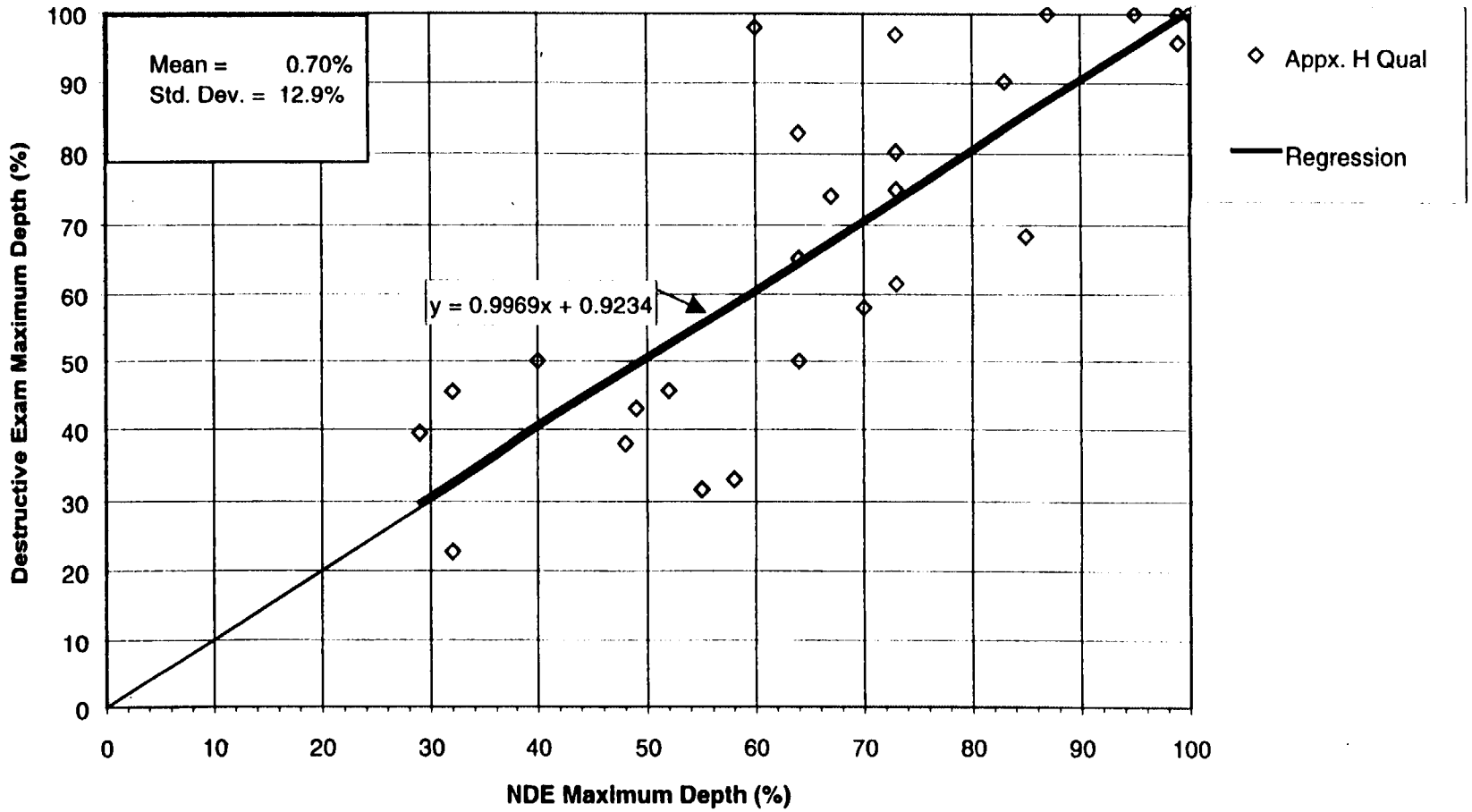


Figure 7-4
APPENDIX H QUAL - Dented TSP Axial PWSCC
Local Max. Depth for NDE and RA Max. Depth for DE
Destructive Exam vs. NDE Trends

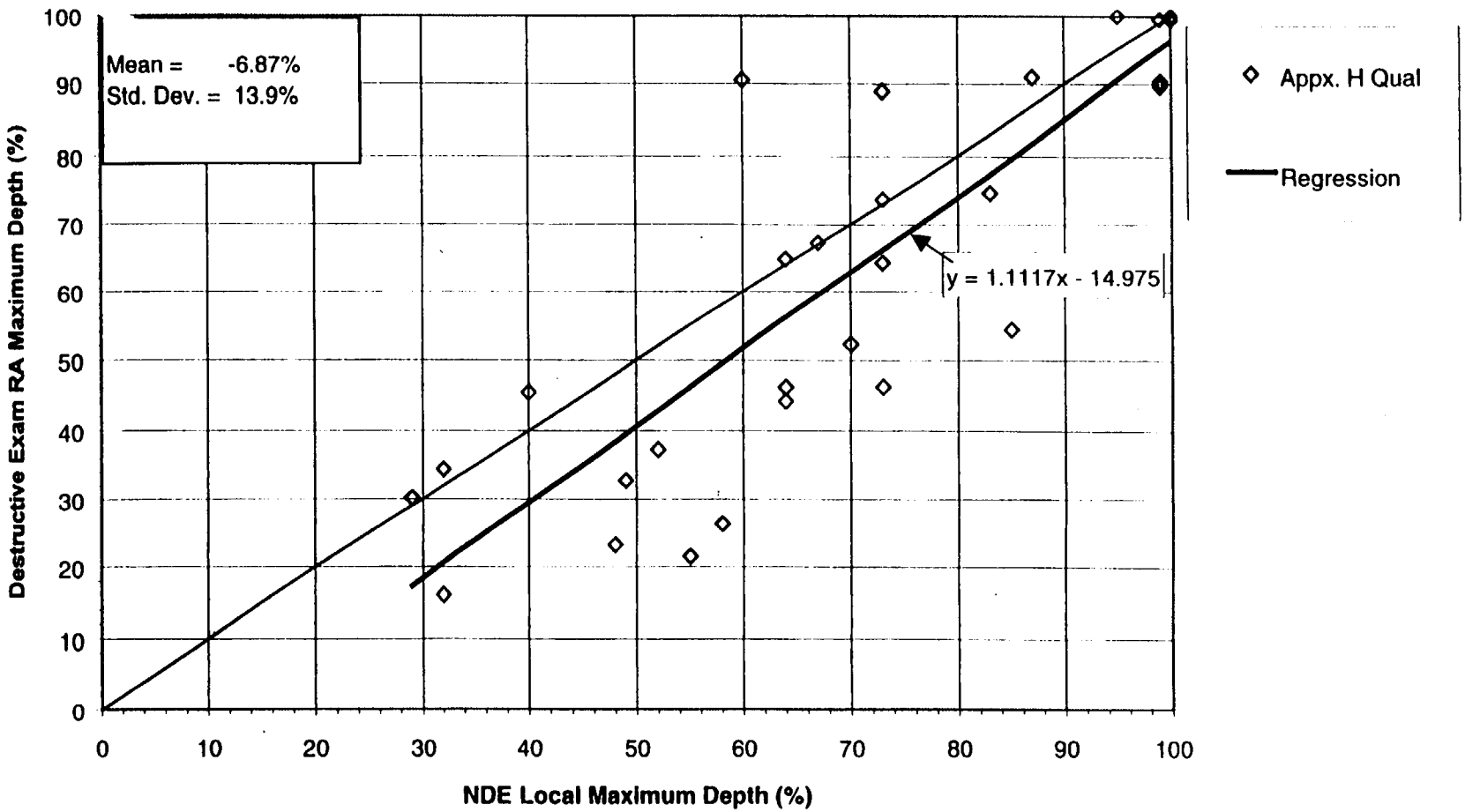
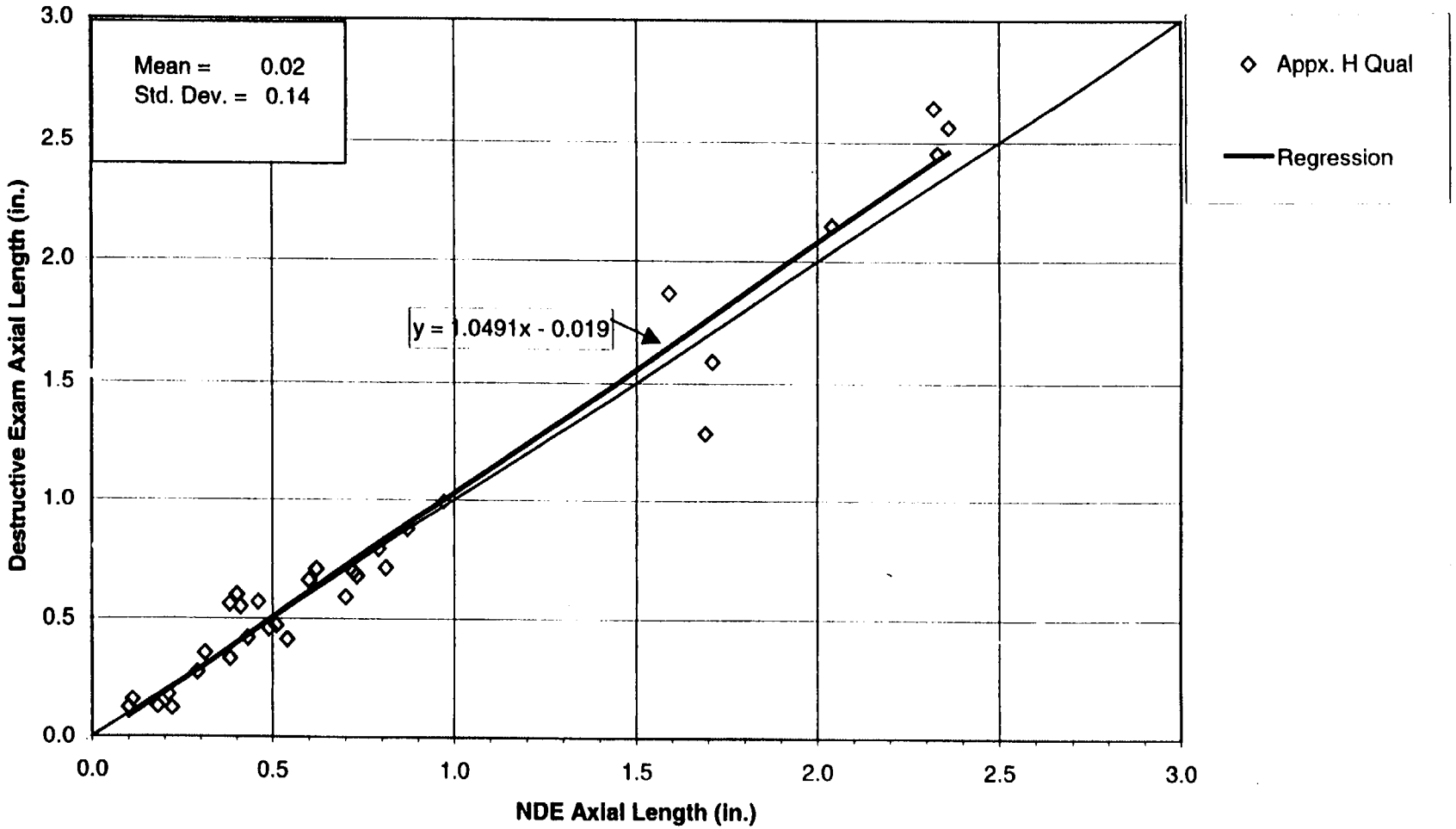


Figure 7-2
APPENDIX H QUAL - Dented TSP Axial PWSCC
Axial Length Destructive Exam vs. NDE Trends

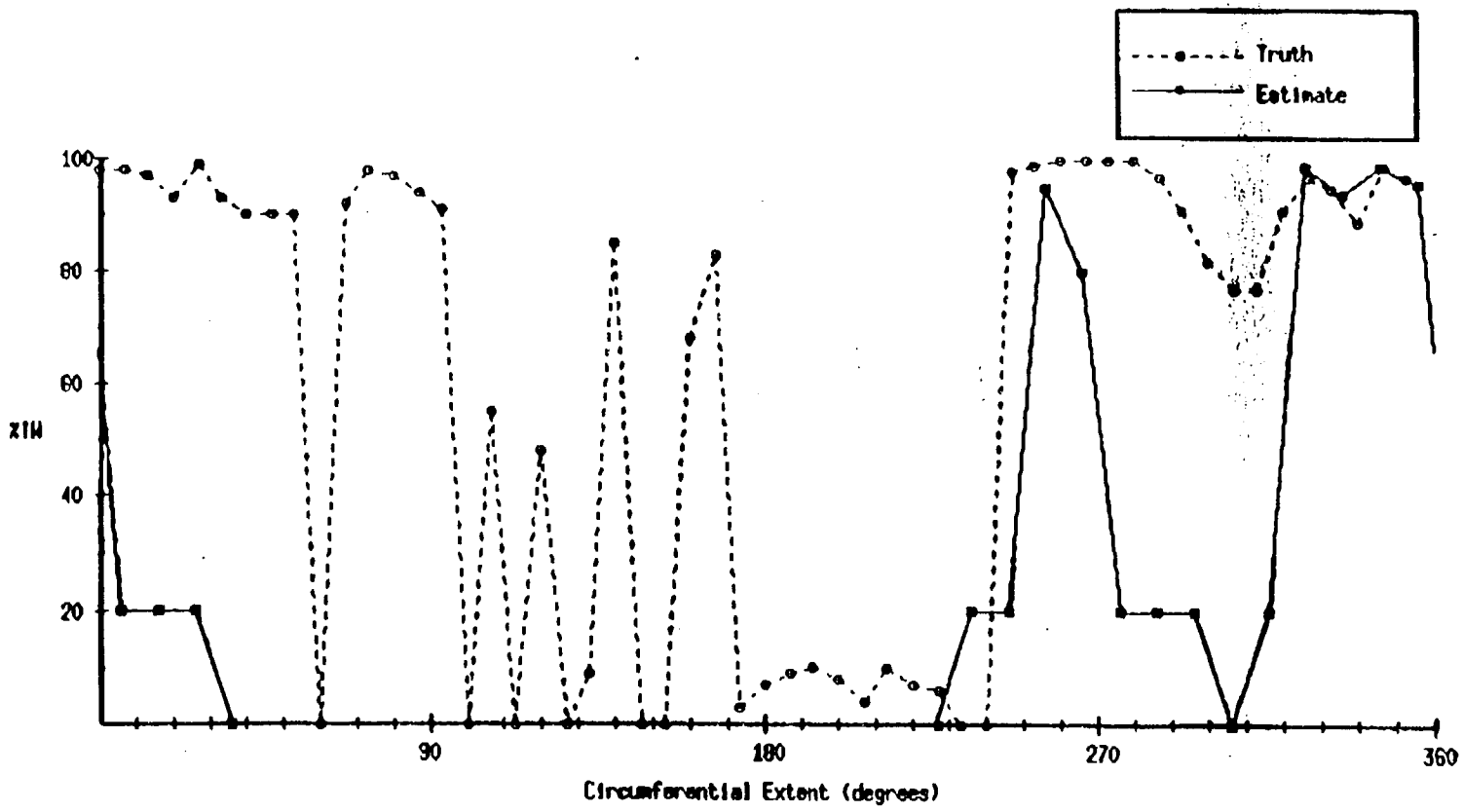


R 24 C 42

.080 Pancake Axial Average Filter

Group B

TRUTH	Flaw Length Degraded Area = 63.24%	Length = 340 deg.	RMSE = 56.32
	360 Degree Degraded Area = 62.45%	Max Depth = 100%	COEF = 0.48
EST.	Flaw Length Degraded Area = 43.71%	Length = 170 deg.	Dia. = 0.7500
	360 Degree Degraded Area = 20.64%	Max Depth = 99%	Wall = 0.0430



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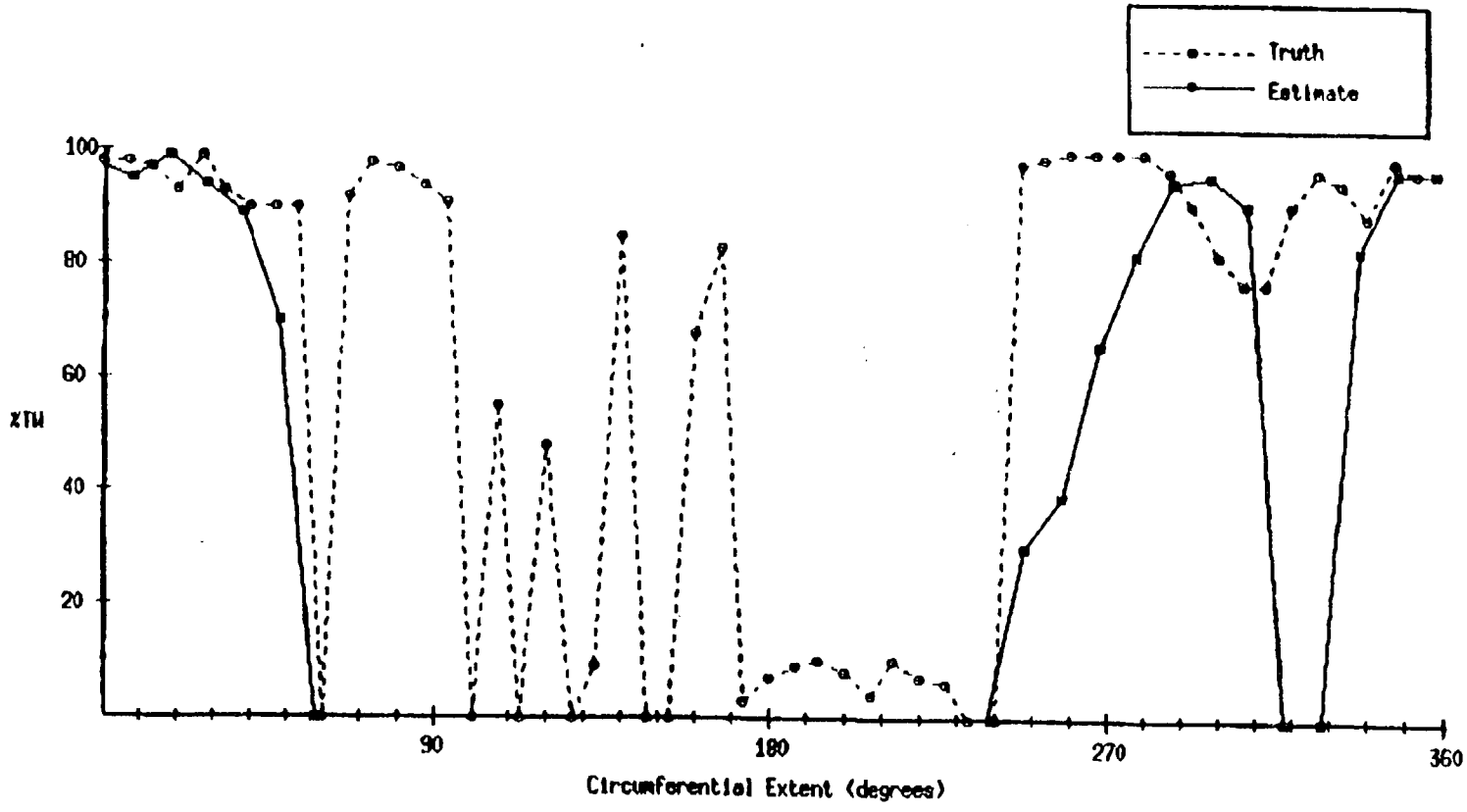
R 24 C 42

.115 Pancake Axial Average Filter

Group B

TRUTH	Flaw Length Degraded Area = 63.24%	Length = 340 deg.	RMSE = 45.06
	360 Degree Degraded Area = 62.45%	Max Depth = 100%	COEF = 0.63
EST.	Flaw Length Degraded Area = 71.94%	Length = 170 deg.	Dia. = 0.7500
	360 Degree Degraded Area = 33.97%	Max Depth = 99%	Wall = 0.0430



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R 24 L 42

Plus Point Axial Average Filter

Group B

TRUTH	Flaw Length Degraded Area = 63.24%	Length = 340 deg.	RMSE = 42.15
	360 Degree Degraded Area = 62.45%	Max Depth = 100%	COEF = 0.69
EST.	Flaw Length Degraded Area = 72.94%	Length = 170 deg.	Dia. = 0.7500
	360 Degree Degraded Area = 34.44%	Max Depth = 95%	Wall = 0.0430

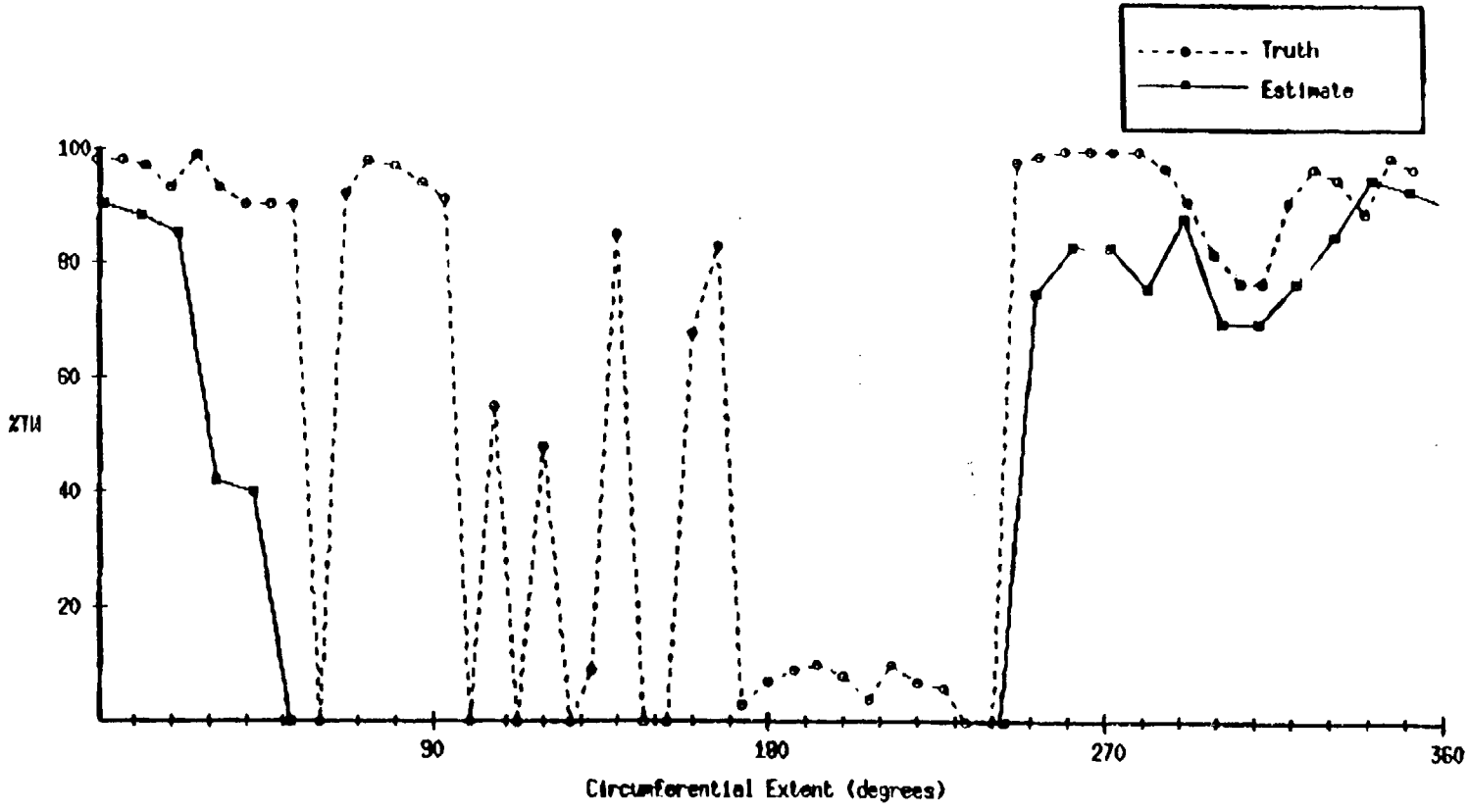

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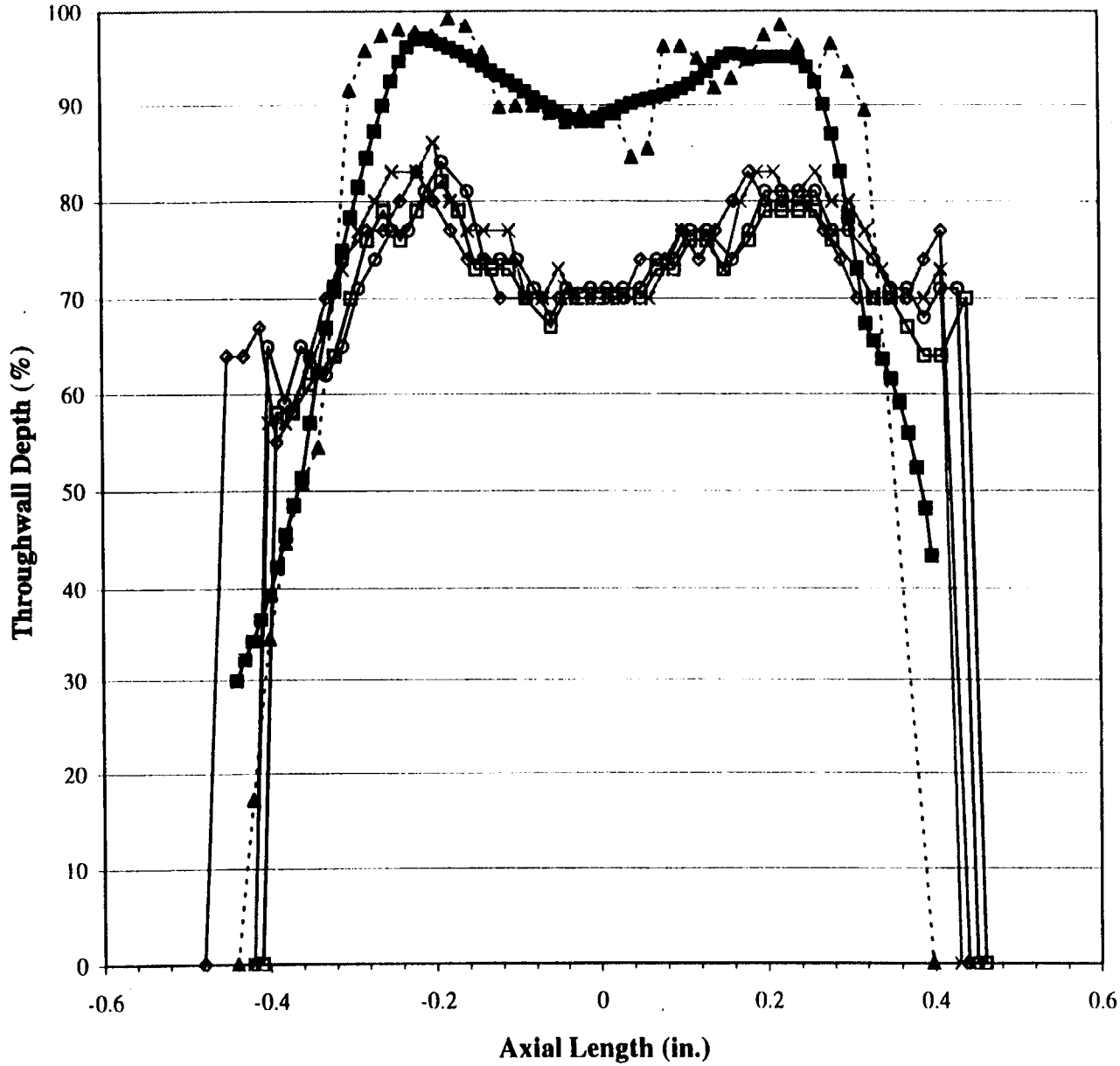
Figure C-54
Sample 11, TSP 2H- Crack 1
Mid-Range +Point, 300 kHz
NDE Depth vs. Axial Length with Destructive Exam



Analyst 1
Analyst 2
Analyst 3
Analyst 4
Destructive Exam w/RA
Destructive Exam w/o RA

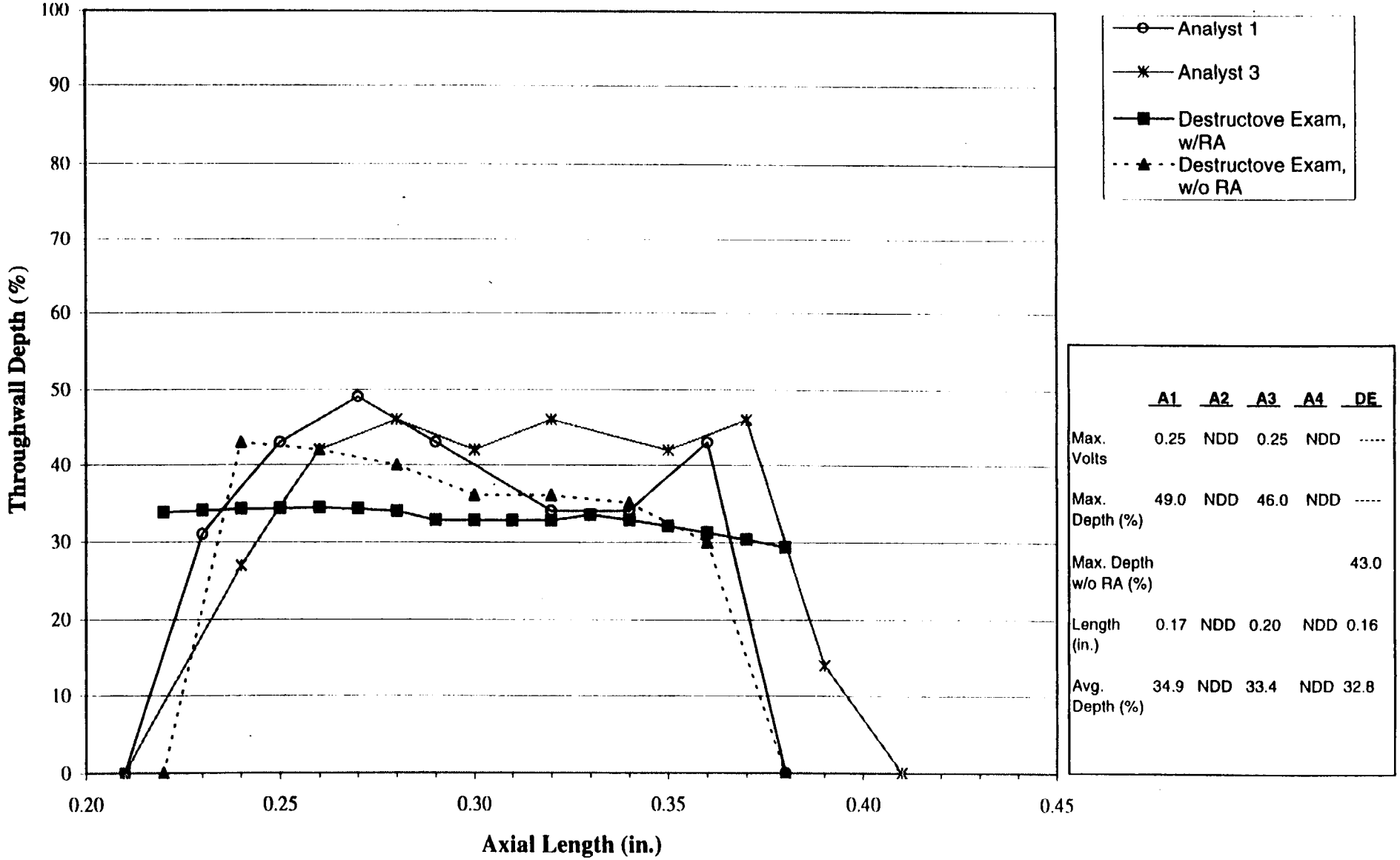
	A1	A2	A3	A4	DE
Max. Volts	3.25	3.16	3.21	3.18	------
Max. Depth (%)	84.0	82.0	86.0	83.0	96.8
Max Depth w/o RA (%)					99.2
Length (in.)	0.87	0.87	0.85	0.92	0.84
Avg. Depth (%)	71.7	70.2	72.7	70.5	80.9

Figure C-55

Sample 11, TSP 2H- Crack 2

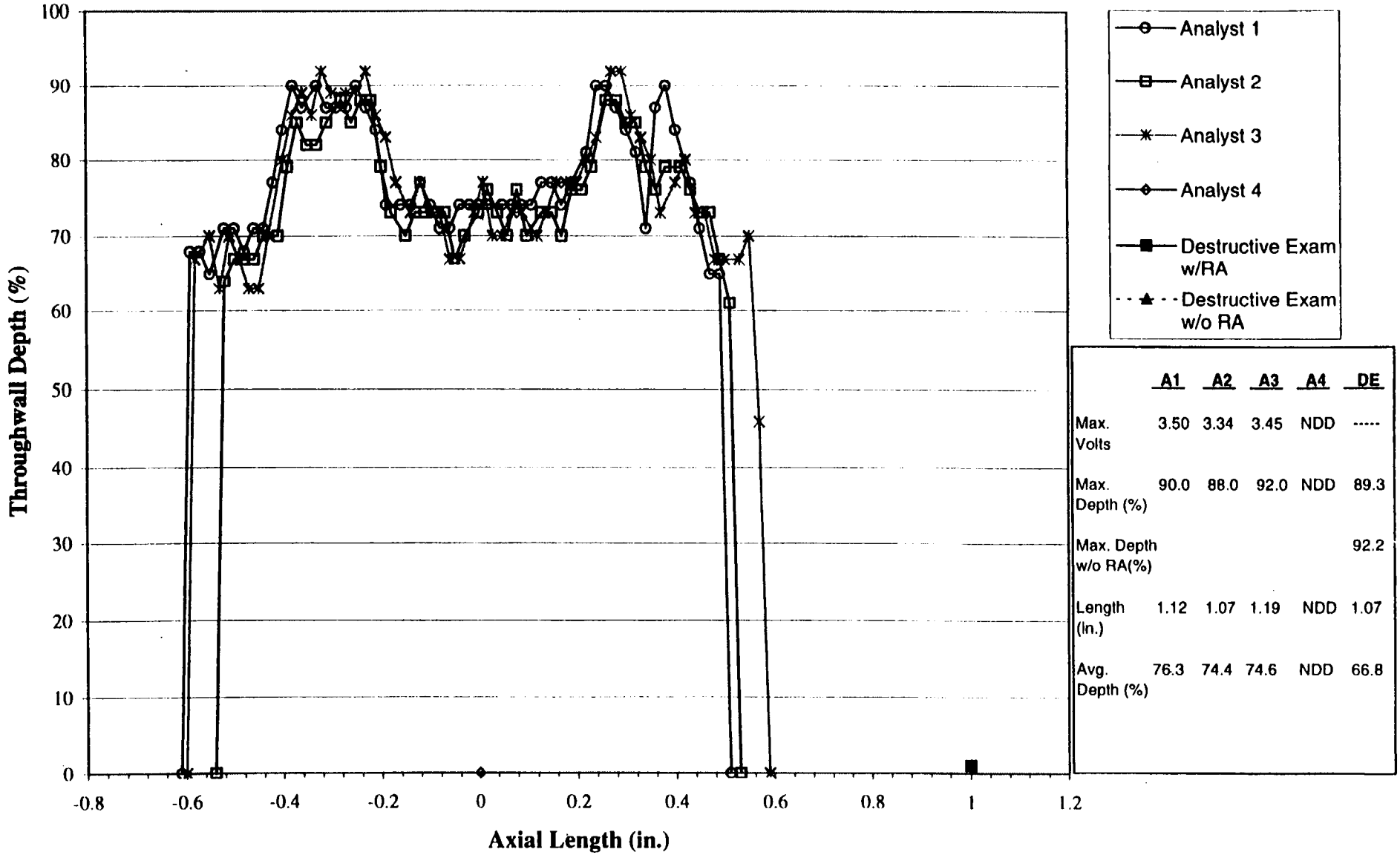
Mid-Range +Point, 300 kHz

NDE Depth vs. Axial Length with Destructive Exam



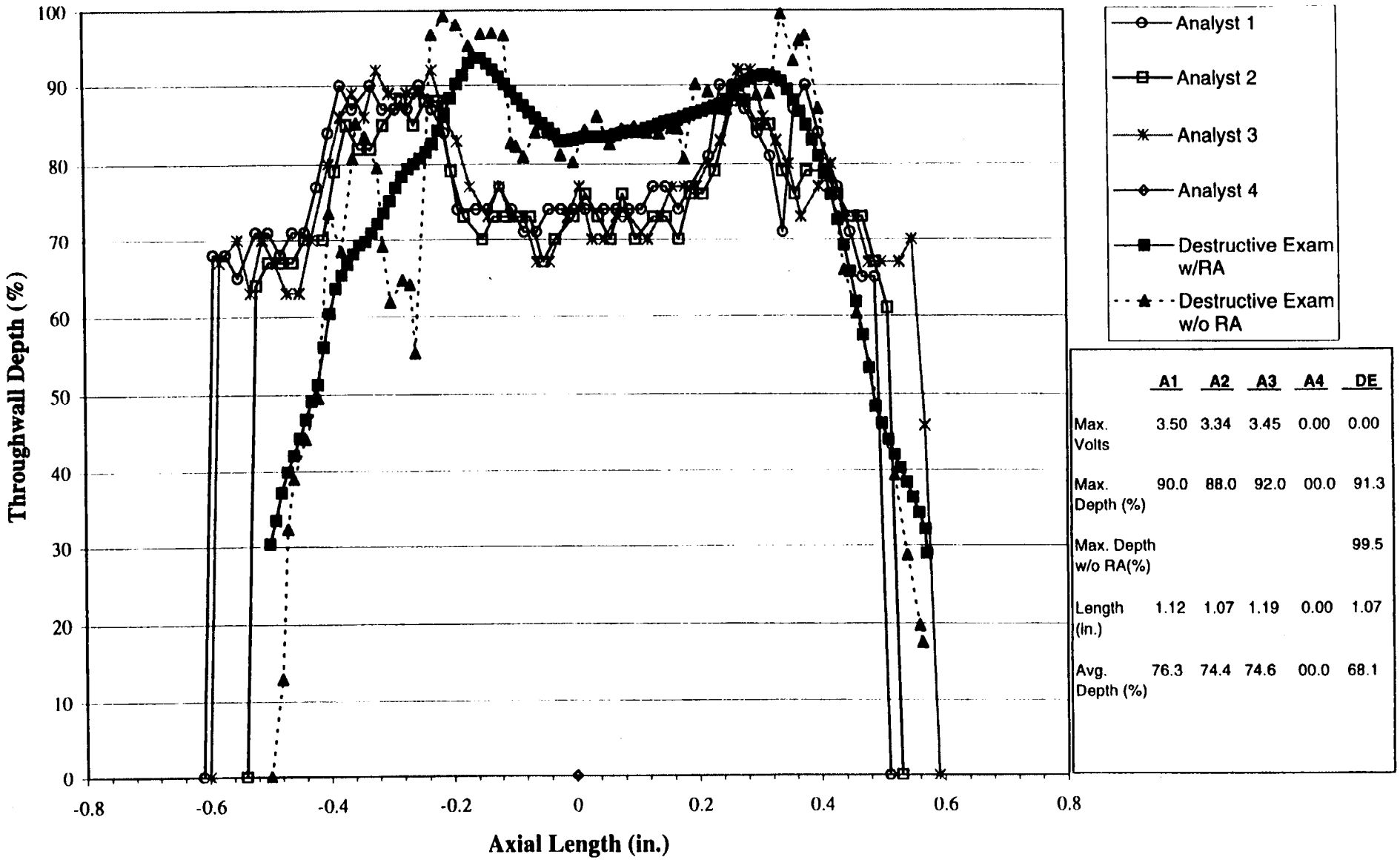
	A1	A2	A3	A4	DE
Max. Volts	0.25	NDD	0.25	NDD	-----
Max. Depth (%)	49.0	NDD	46.0	NDD	-----
Max. Depth w/o RA (%)					43.0
Length (in.)	0.17	NDD	0.20	NDD	0.16
Avg. Depth (%)	34.9	NDD	33.4	NDD	32.8

Figure C-56
Sample 11, TSP 3H - Crack 1
Mid-Range +Point, 300 kHz
NDE Depth vs. Axial Length with Destructive Exam



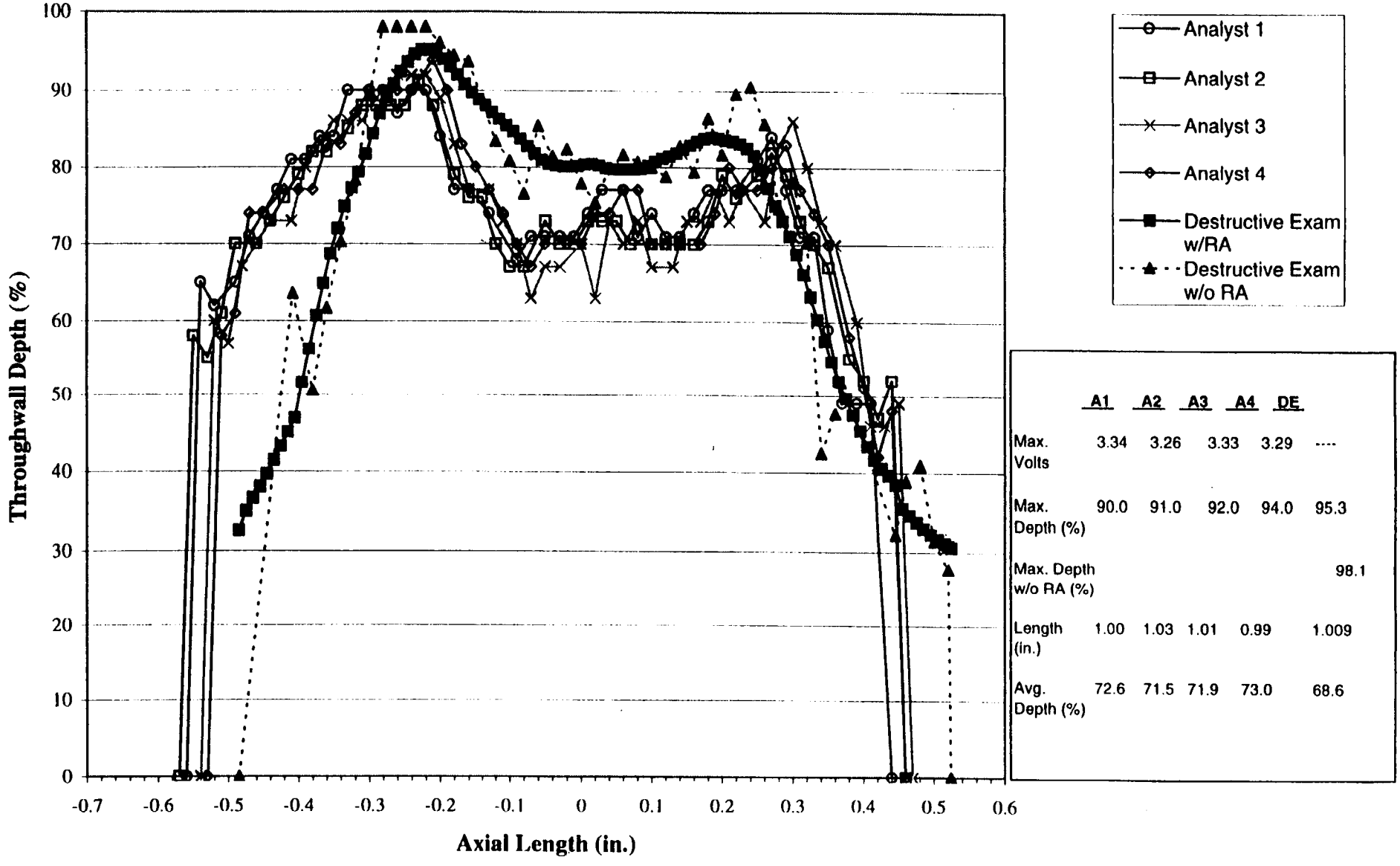
	A1	A2	A3	A4	DE
Max. Volts	3.50	3.34	3.45	NDD	-----
Max. Depth (%)	90.0	88.0	92.0	NDD	89.3
Max. Depth w/o RA(%)					92.2
Length (in.)	1.12	1.07	1.19	NDD	1.07
Avg. Depth (%)	76.3	74.4	74.6	NDD	66.8

Figure C-56
Sample 11, TSP 3H - Crack 1
Mid-Range +Point, 300 kHz
NDE Depth vs. Axial Length with Destructive Exam



	A1	A2	A3	A4	DE
Max. Volts	3.50	3.34	3.45	0.00	0.00
Max. Depth (%)	90.0	88.0	92.0	00.0	91.3
Max. Depth w/o RA(%)					99.5
Length (in.)	1.12	1.07	1.19	0.00	1.07
Avg. Depth (%)	76.3	74.4	74.6	00.0	68.1

Figure C-58
Sample 11, TSP 4H- Crack 1
Mid-Range +Point, 300 kHz
NDE Depth vs. Axial Length with Destructive Exam



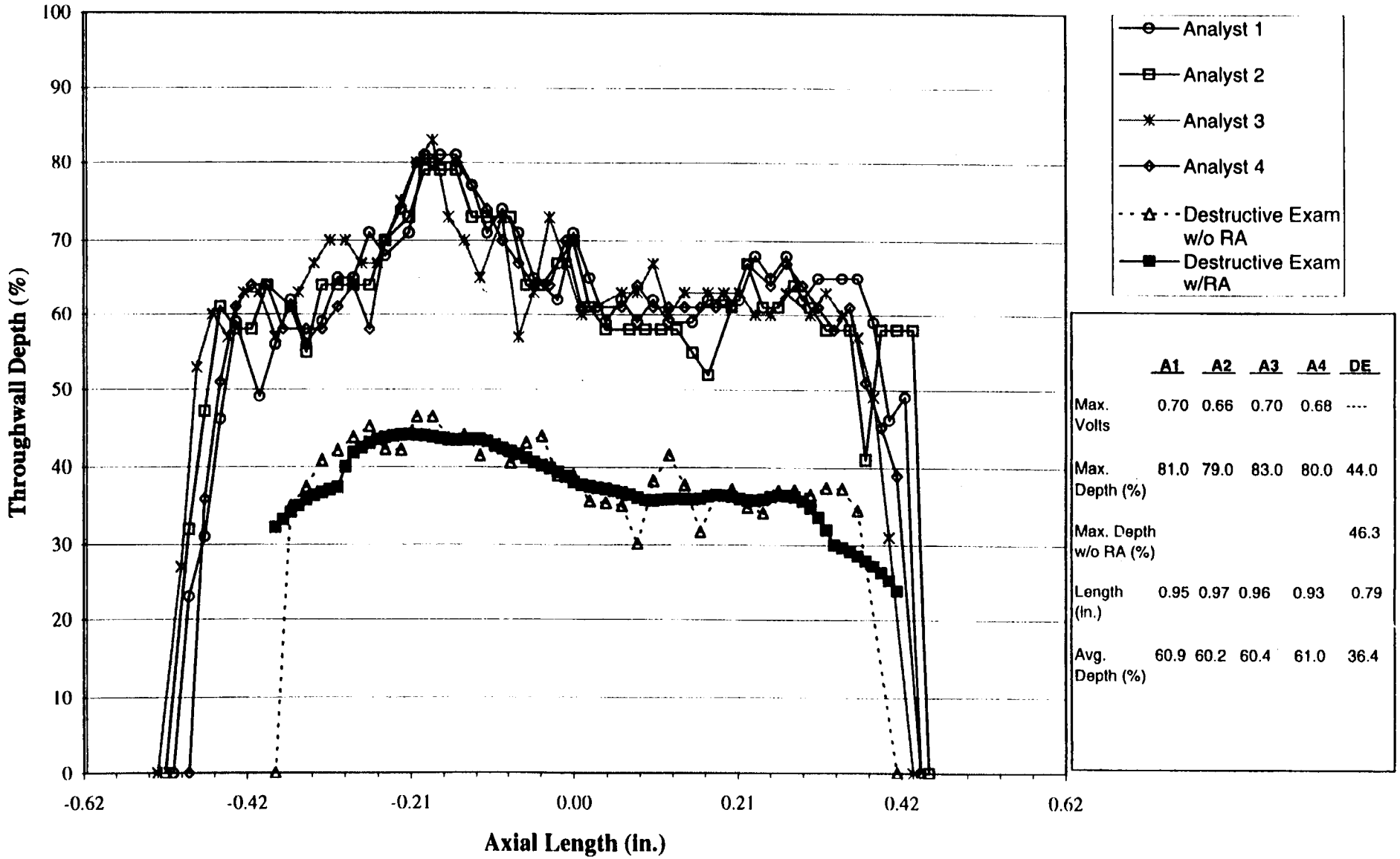
	A1	A2	A3	A4	DE
Max. Volts	3.34	3.26	3.33	3.29	----
Max. Depth (%)	90.0	91.0	92.0	94.0	95.3
Max. Depth w/o RA (%)					98.1
Length (in.)	1.00	1.03	1.01	0.99	1.009
Avg. Depth (%)	72.6	71.5	71.9	73.0	68.6

Figure C-59

Sample 11, TSP 4H- Crack 2

Mid-Range +Point, 300 kHz

NDE Depth vs. Axial Length with Destructive Exam



	A1	A2	A3	A4	DE
Max. Volts	0.70	0.66	0.70	0.68	----
Max. Depth (%)	81.0	79.0	83.0	80.0	44.0
Max. Depth w/o RA (%)					46.3
Length (in.)	0.95	0.97	0.96	0.93	0.79
Avg. Depth (%)	60.9	60.2	60.4	61.0	36.4

FACTORS AFFECTING ABILITY TO SIZE I

Many factors affect ability to size including:

- Technique (UT, ECT)
- Type of coil (bobbin, plus-point, pancake, ...)
- Location of degradation
- Frequency
- Interfering signals
- Noise

Plant-specific considerations play a role (e.g., interfering signals from deposits and dents, noise,.....)

"Qualification" is generic but utilities assess applicability of technique to their plant/needs

Although quantitative size of degradation may not be determined, the general severity can be inferred and further testing (in-situ pressure testing) performed to ensure tube integrity

FACTORS AFFECTING ABILITY TO SIZE II

Crack orientation

Circumferential cracks frequently located in regions with geometry changes (expansion transitions, dents, U-bends) which generally make detection and sizing more difficult

Isolated cracks versus cracks in clusters

The closer 2 cracks are to each other the more difficult to distinguish them (due to coil size)

May result in overestimate of length of degradation

Plugged cracks or cracks occluded with crud

Depends on nature of deposits - insulating, conductive, ferromagnetic

Crack location relative to support plates

More interfering signals the more difficult to size

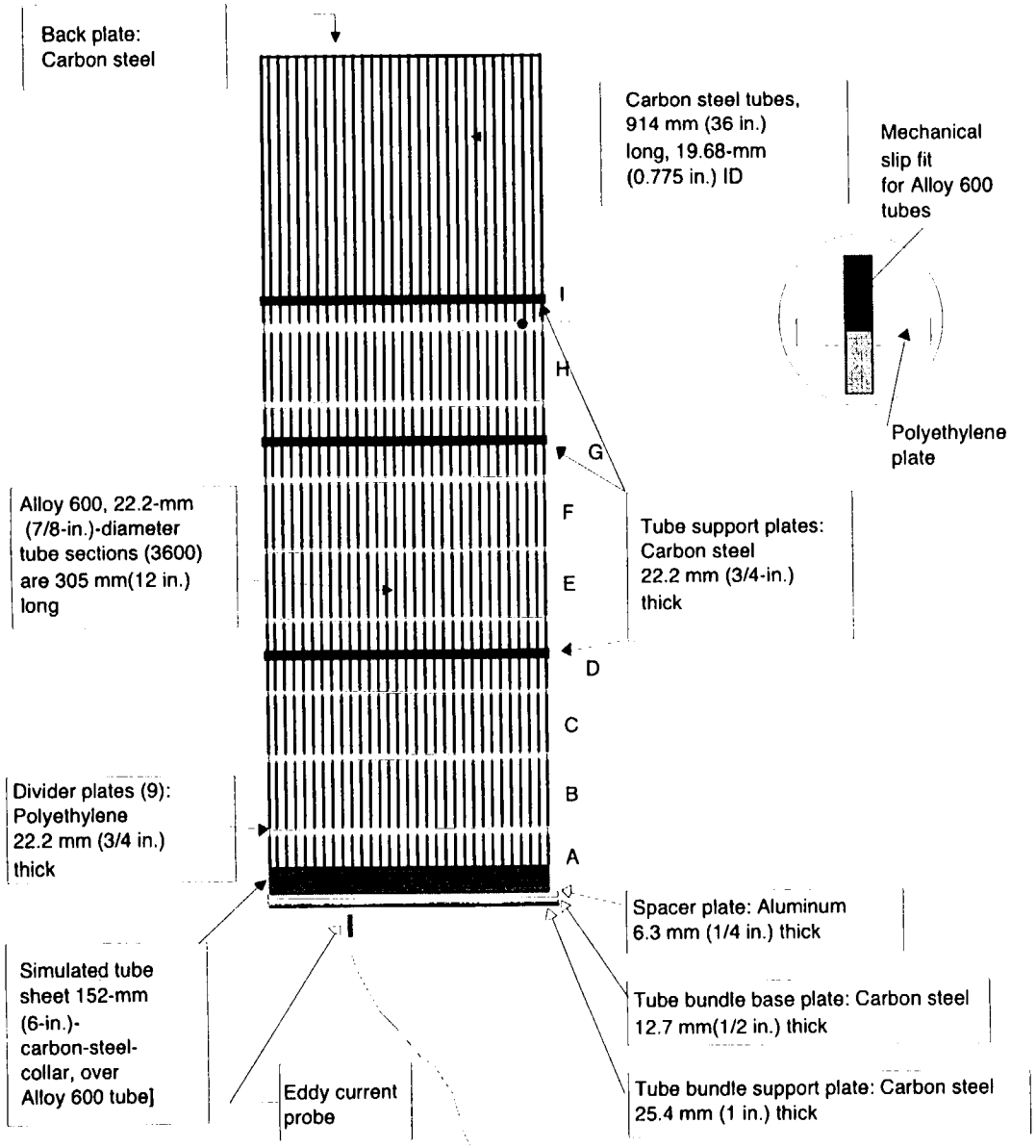
Steam Generator Mockup and Round Robin

(agenda item 10)

Purpose of round robin is to assess current state of SG tubing ISI technology

- Quantify reliability and accuracy
- Determine probability of detection (POD) as function of flaw size
- Assess flaw sizing capability

Utilizes mockup facility constructed at ANL



Mockup contains 3600 1–ft. test sections

Contains hundreds of samples of circumferential and axial OD and ID SCC, IGA, wear, and wastage

Includes simulations of dents, sludge piles, magnetite deposits, tube support plates, and roll transition at tube sheet

Additional tubes with laser and EDM notches.

As part of round robin an NDE Task Group with members from ANL, NRC, EPRI, FTI, ABB–CE, Zetec, Westinghouse, NSP, Com Ed, and Duke was formed

- Input from members helped assure documentation and procedures for RR tests were comparable to those provided by utility for inspections
- Members reviewed EC signals from laboratory cracked tubes and simulation of artifacts (TSP, tube sheet roll transitions, corrosion products, dents, etc.) to verify as representative of those encountered in the field

All cracks were created in the laboratory. Characteristics such as tightness, morphology were compared to results of destructive analyses of field cracks available in the literature. As noted, EC characteristics were reviewed by NDE Task Group members

This is an analysis round robin for detection. The actual eddy current measurements were taken by a industry qualified team from Zetec. Measurements were then provided on optical disk to teams.

- Proctor administers written and practical exams to all analysts
- Analysis of EC data is performed per written procedures and monitored by proctor
- Eight industry teams (5 members as in field inspections) have participated thus far. Analysis takes 6–8 working days to complete.

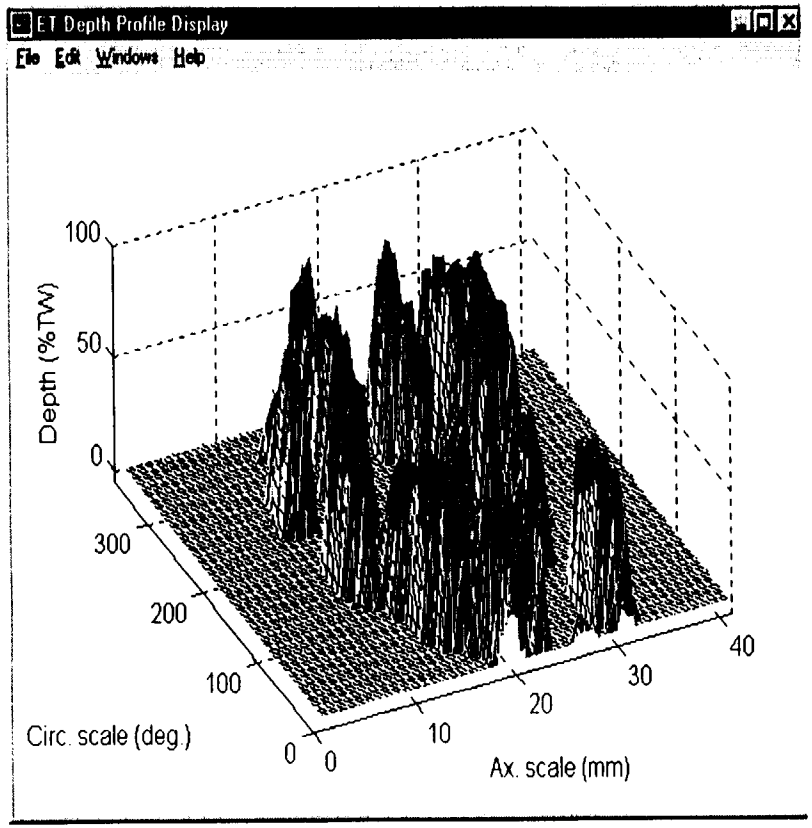
Sizing is done in the industry by a specialized group of experts. A selected set of specimens from the mockup will be profiled by a number of experts (too time consuming to analyze all specimens)

To determine POD “true” state must be known

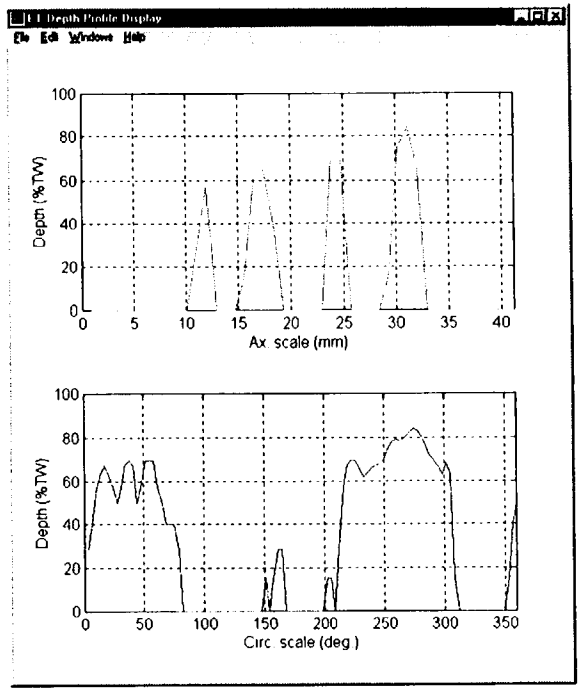
A variety of characterization methods have been used

- ANL staff have the advantage of knowing where the artifacts are and are able to analyze data taken with and without artifacts

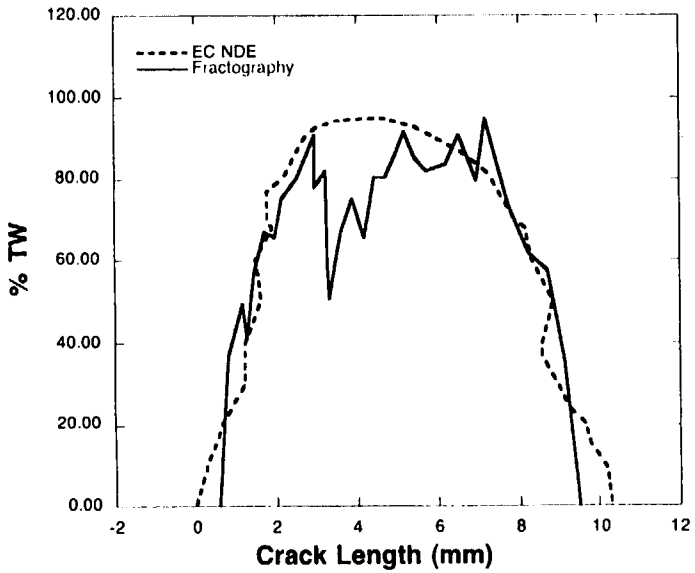
- Currently benchmarking a multifrequency EC analysis algorithm against destructive analysis. Method provides graphical display which helps visualize cracking especially in cases like the roll transition where geometry greatly complicates analysis

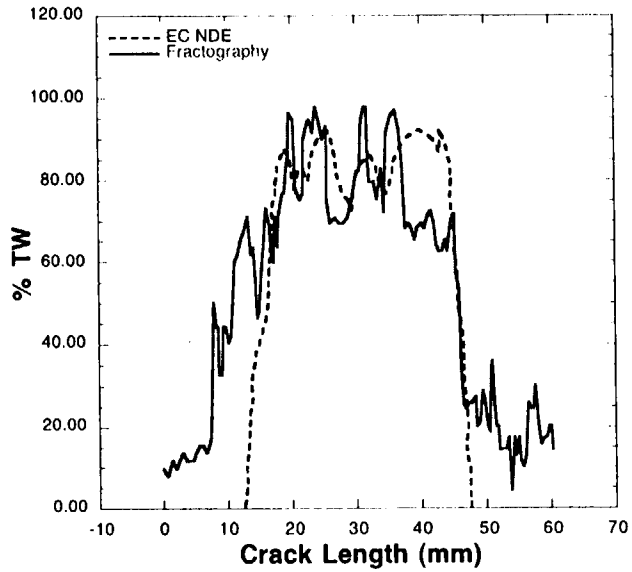


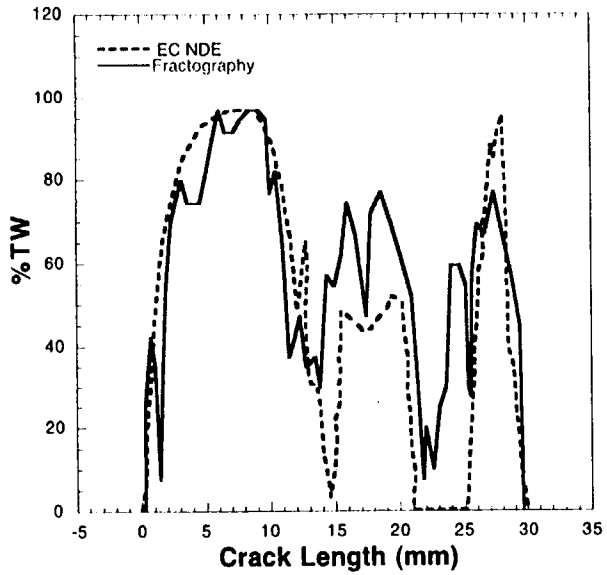
Lab-grown circumferential ODSCC ≈360° staggered cracking with maximum depth >80% TW.



Axial or circumferential cross–sections can be taken to get profiles of the crack







Destructive analysis of 29 tube set performed to validate capability of multi–frequency phase analysis to better size and characterize a variety of flaws

Preliminary Observations from Round Robin Activities

Preliminary results for ODSCC suggest that for depths >50%, POD >80%. POD estimates required by GL95–05 are conservative for deep SCC.

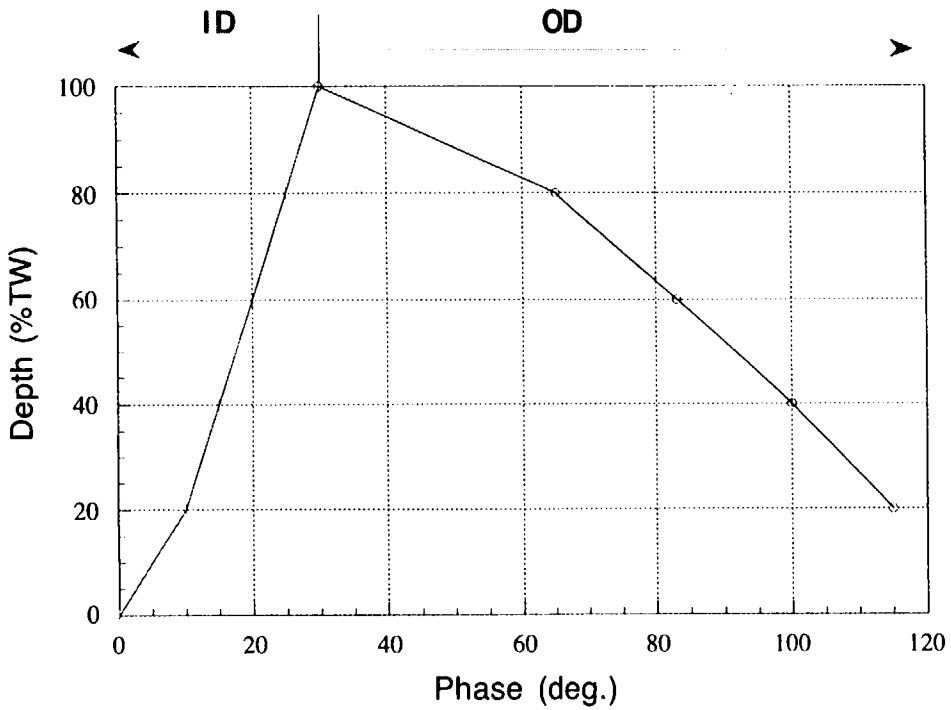
Roll transition significantly complicates detection and characterization of SCC including orientation (e.g., array of short axial cracks can be interpreted as a continuous circumferential crack) and origin .

Bobbin coil voltage measurements for TSP indications show small variability among inspection teams.

Circumferential SCC are more difficult to characterize than axial SCC at TTS where the inspection is complicated by the presence of the roll transition and sludge

Although ID cracks are detected more easily than OD cracks, ID cracks in dents are difficult to detect unless crack is deep

Clusters of cracks are often seen as a single indication, since spatial resolution is limited by the size of the coil



Qualified sizing techniques are currently limited to primary side cracking at specific locations. They are based on comparison of the MRPC phase angle for a signal from an indication with that from a notch of known depth

Phase angles for ID indications vary rapidly with depth. This can lead to uncertainties in depth sizing for ID degradation.

OD depth measurements are limited by signal-to-noise ratio (S/N), since signal amplitude drops exponentially with distance from ID.

The portions of the crack that can be detected with good confidence (> 50% depth) can be sized reasonably accurately(± 3–4 mm).

For SCC depths less than 50%TW, profiling with conventional MRPC using phase analysis at 300kHz is not reliable. The depth of shallow cracks is often overestimated, but measurements are not consistently conservative.

Determination of orientation and circumferential extent of cracking at roll transition and TTS regions is difficult.

GL 95-05: TECHNICAL OVERVIEW

GL 95-05 addresses predominantly axially oriented outside diameter stress corrosion cracking at tube support plate elevations

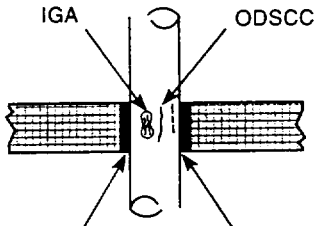
2 fundamental goals of repair criteria: Ensure adequate structural and leakage integrity

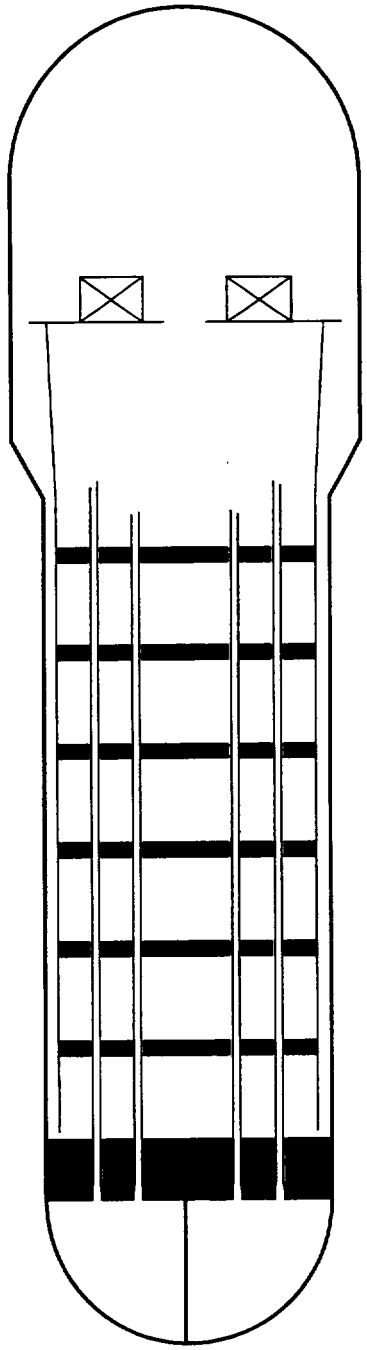
Evaluation of structural and leakage integrity require:

- Periodic inspections including understanding of capabilities/limitations of NDE
- Correlation of inspection parameters to the structural and leakage integrity of the tubing
- Evaluation of tubes accepted for continued service

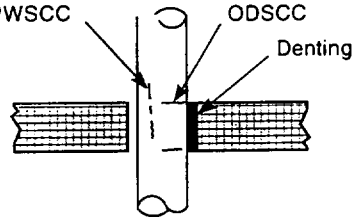
TUBE SUPPORT PLATE ELEVATION DEGRADATION

Voltage-based repair criterion applies to the following degradation:

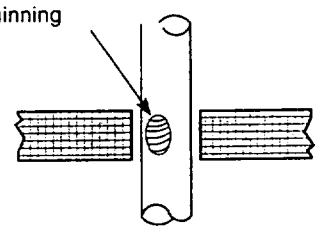




Voltage-based repair criterion does not apply to the following degradation:



Fretting, Wear, Corrosion Thinning



GL 95-05 INSPECTIONS

100% bobbin coil examinations

- Rapidly screens tubes for defects
- Extent of degradation measured in terms of voltage response for defects at the tube support plate elevations
- Detailed procedures to ensure voltage response of degradation in field is comparable to those in the structural and leakage integrity database

Rotating pancake coil examinations

- Permits better characterization of defect to ensure degradation is confined within tube support plate and is predominantly axial
- Performed at intersections with degradation exceeding specific voltage limits (1.0 volt and 1.5 volts), dents exceeding 5.0 volts, TSP elevations with copper deposits, and large mix residuals

STRUCTURAL AND LEAKAGE INTEGRITY CORRELATIONS

Correlations include data from 2 sources: Tubes removed from operating steam generators and specimens produced in model boiler facilities

Destructive examinations typically performed:

- Leak Testing
  - Used in Probability of Leakage correlations
  - Used in Conditional Leak rate correlation if tube leaks
- Burst Testing
  - Used in burst pressure correlation
- Metallurgical examination
  - Ensure degradation mode is consistent with those from other sources/plants

REGULATORY CRITERIA

Regulatory Guide 1.121 provides structural criteria for determination of tube repair limits

Typically the most limiting regulatory guide 1.121 structural criteria are:

3 times normal operating differential pressure ($3\Delta P$)

1.4 times the maximum postulated accident condition differential pressure (1.4 x MSLB)

$3\Delta P$ limit is inherently satisfied under normal operating conditions for corrosion at the tube support plate (TSP) elevation due to presence of TSP

Burst pressure/bobbin voltage correlation is used to determine the voltage at which the 1.4xMSLB limit is satisfied

DETERMINATION OF TUBE REPAIR LIMIT

Structural limit determined from intersection of limiting RG 1.121 structural criteria (1.4 x MSLB) and the burst curve evaluated at the lower 95% prediction interval adjusted to the lower 95% material properties

Structural limit adjusted for flaw growth and NDE uncertainty to determine upper voltage repair limit

A lower repair limit was established to account for

- higher than average growth rates
- limited pulled tube data above ~ 4.0 volts
- limited operating experience

BURST PRESSURE vs BOBBIN VOLTAGE
7/8-Inch Diameter Tubing

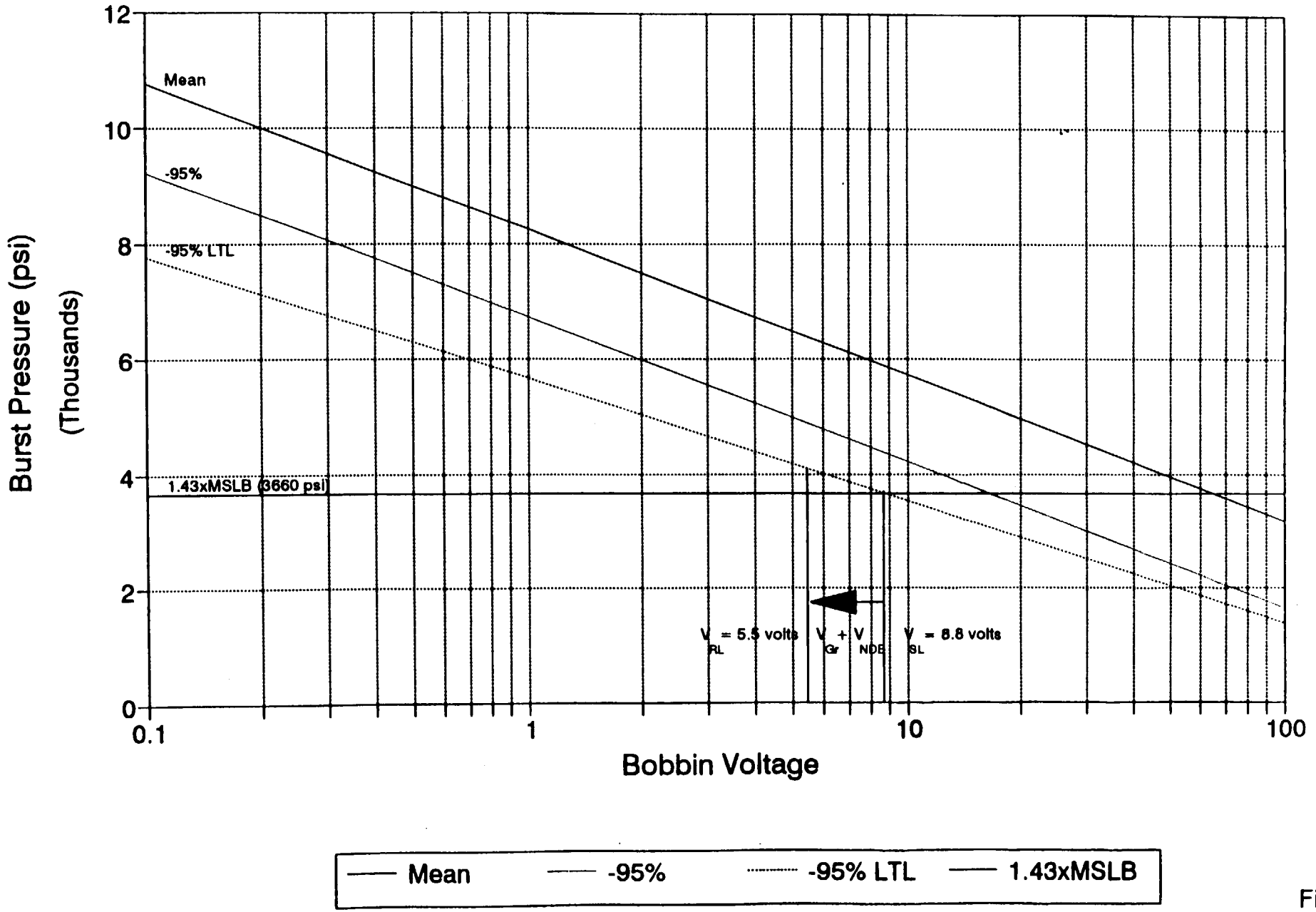


Figure 3

GENERIC LETTER 95-05 CALCULATIONS

Conditional probability of burst given a MSLB

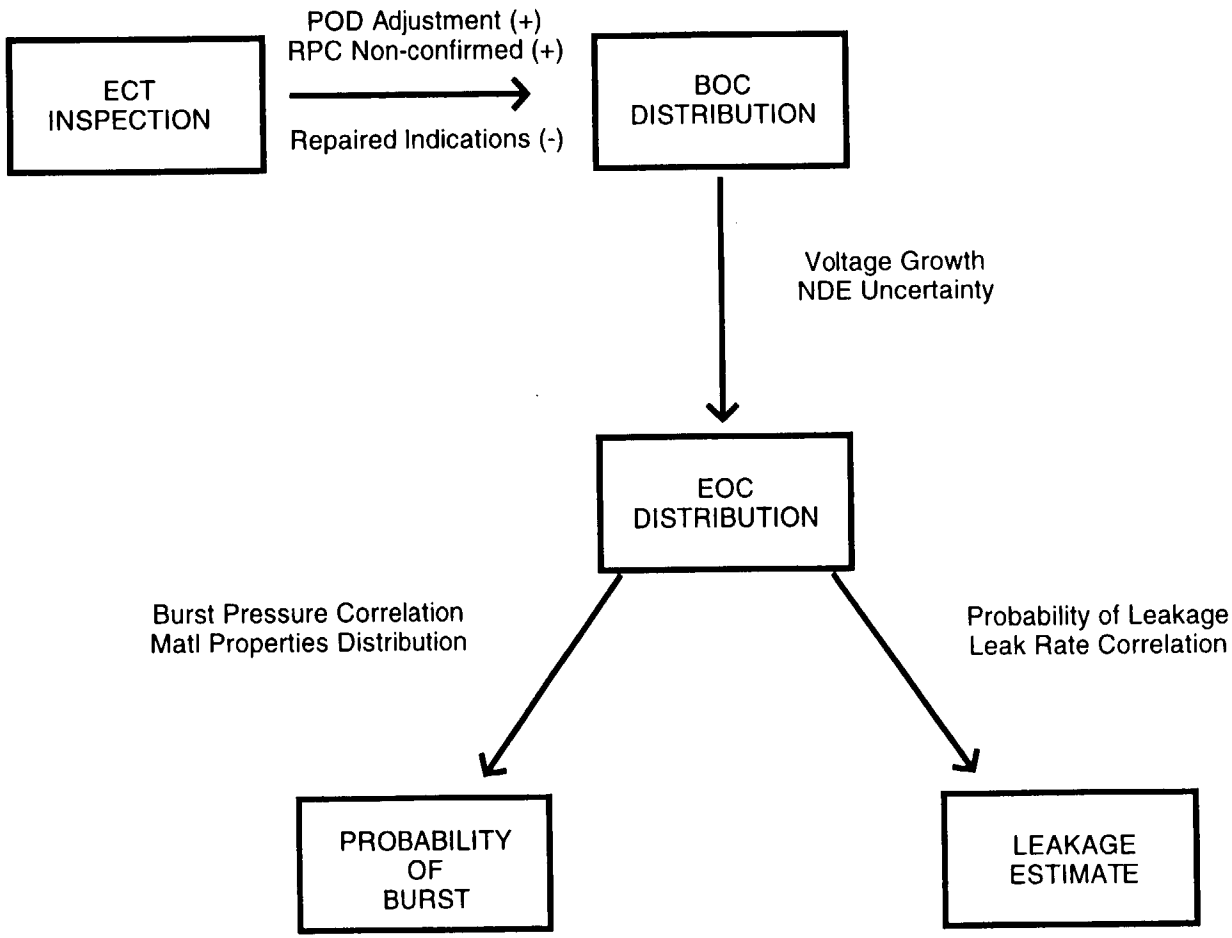
- Necessary to ensure repairs are adequate from a structural integrity standpoint
  - Values of growth and NDE uncertainty may exceed those assumed in deterministic determination of the voltage repair limit
  - Lower 95% prediction interval for burst pressure correlation used in deterministic determination of repair limits

Leakage under postulated accidents

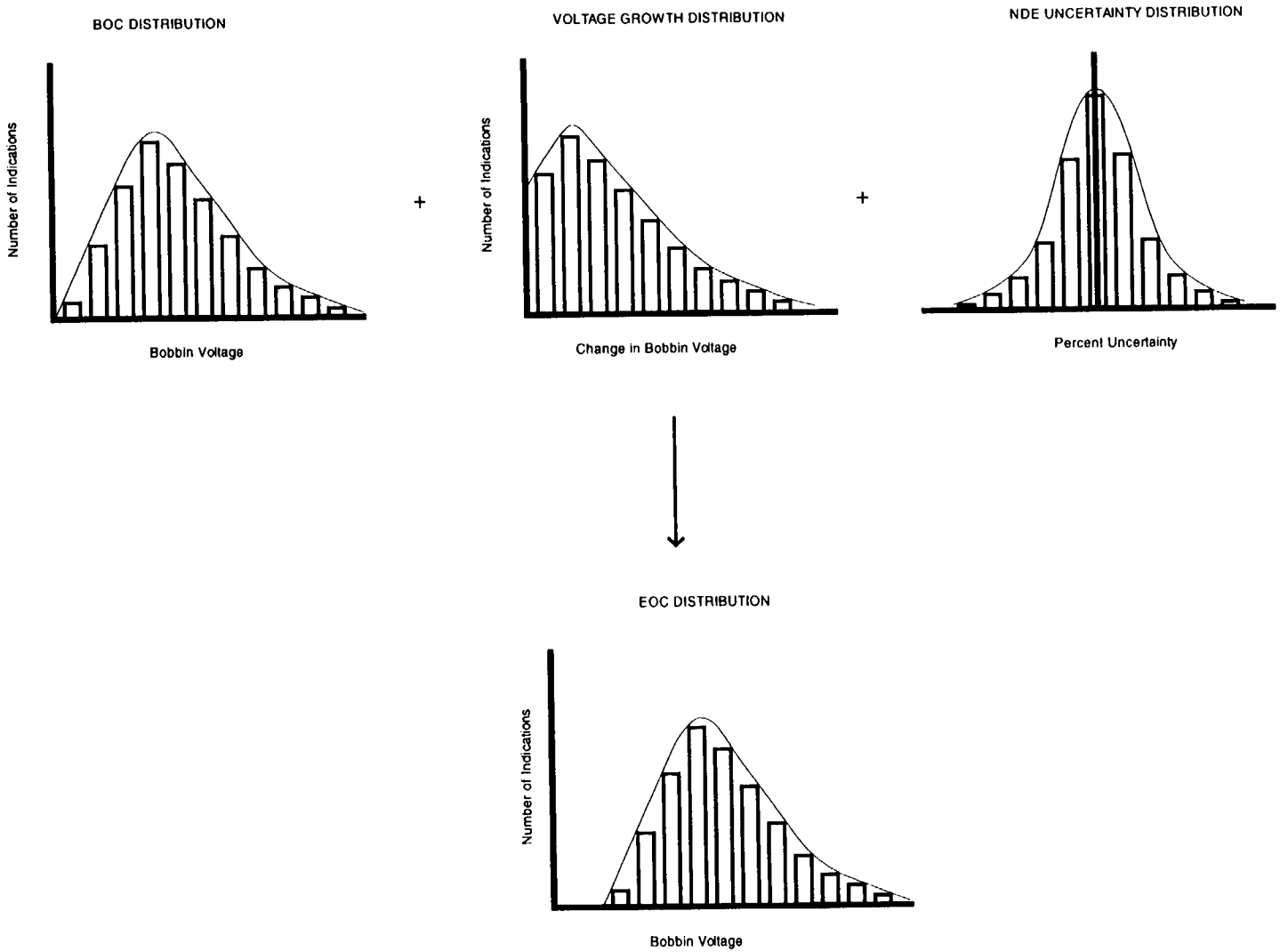
- Through wall cracks (or near through-wall cracks) may be left in service or may develop during service
- Methodology for determining leakage under accident condition is necessary

Above calculations require determination of end-of-cycle (EOC) distribution of indications

VOLTAGE-BASED PLUGGING CRITERIA METHODOLOGY

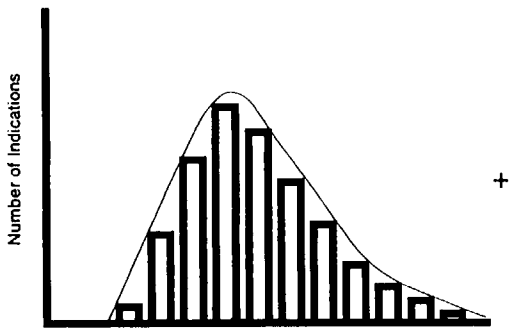


PREDICTING THE EOC VOLTAGE DISTRIBUTION



CALCULATING THE PROBABILITY OF RUPTURE

EOC DISTRIBUTION



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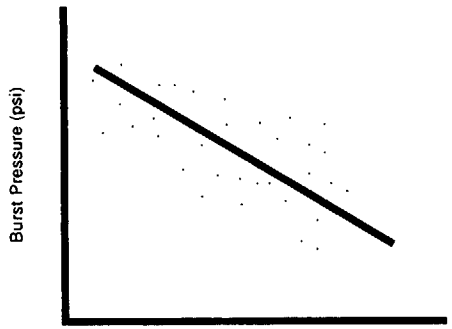
BURST PRESSURE CORRELATION



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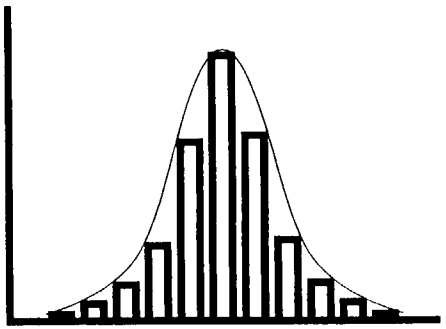
DISTRIBUTION OF MATL PROPERTIES



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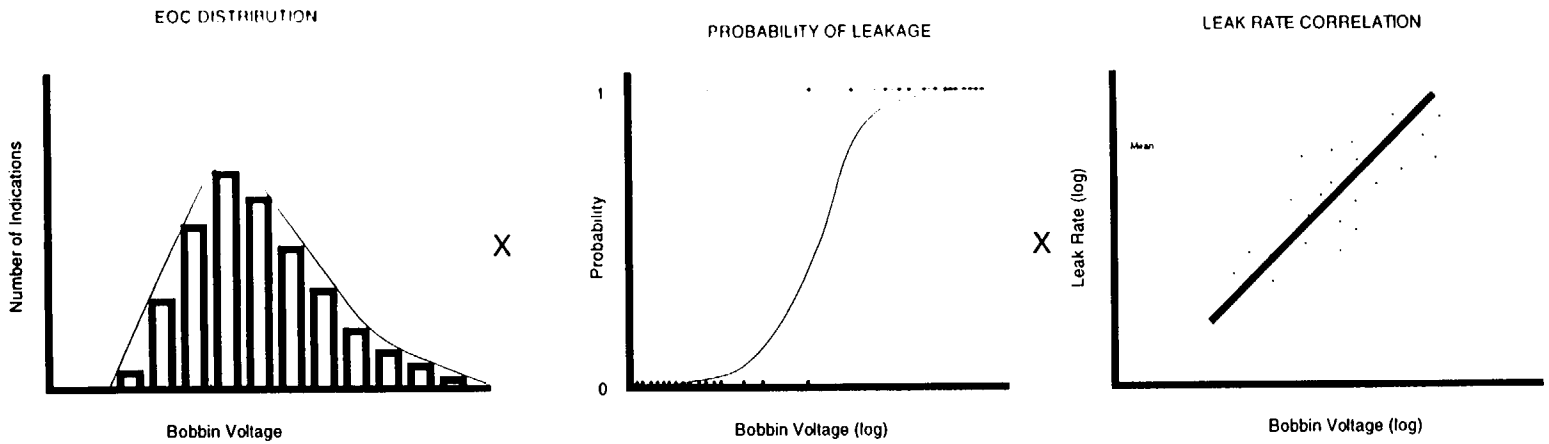
Probability of Rupture
Given a MSLB

LEAK RATE METHODOLOGY



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Postulated MSLB Leakage

ASSESSING RESULTS

Implementation of GL 95-05 methodology "requires" licensees to submit results of their analysis which permits staff to compare projected and actual EOC voltage distributions

Comparisons have generally shown methodology to be conservative

Some issues have been identified during a comprehensive review of 90-day reports (e.g., voltage dependent growth rates)

No "significant" operational leakage

CONDITION MONITORING AND OPERATIONAL ASSESSMENTS

Condition Monitoring - monitoring and evaluating the "as-found" condition of the steam generator tubes with respect to tube integrity performance criteria (a "backwards look" to ensure adequate tube integrity was maintained during the previous cycle)

Operational Assessment -a "forward looking" assessment for demonstrating the tube integrity performance criteria will be satisfied until the next inspection

- Requires knowledge of NDE uncertainties, growth rates, and uncertainties in burst and leakage correlations

USE OF CRACK GROWTH IN TUBE REPAIR CRITERIA

Crack growth assumption made in development of 40% depth-based tube repair criteria

- Based on data available at the time and degradation mechanisms observed
- Only a depth-based growth rate (i.e., infinitely long degradation)

Crack growth rate for alternate tube repair criteria (Axial ODSCC at TSP elevations, Axial PWSCC at dented TSP elevations) based on plant-specific or generic data depending on the amount of data

LABORATORY CRACK GROWTH RATE DATA

Many factors influence crack growth

- Operating parameters - temperature
- Water chemistry - bulk versus crevice, different types of chemistry regimes (all-volatile, phosphate)
- Tube material and microstructure

Many of these factors are not only plant but steam generator and/or tube specific

Difficult to apply laboratory growth rate data to field

- Assumptions made in laboratory experiments are usually conservative which can result in relatively "high" growth rates
- Plant-specific factors which affect growth rates results in regulators questioning appropriateness of laboratory results

DETERMINATION OF "CRACK" GROWTH RATES IN FIELD (GL 95-05)

GL 95-05 methodology for growth "requires" use of plant-specific growth rate data unless there is an insufficient amount of plant-specific data

A distribution of growth rates is used in determining the probability of burst. An average growth rate is used in determining the upper voltage repair limit (30% per EFPY or plant-specific value whichever is larger).

Development of growth rates requires indications to be present in 2 consecutive inspections (unless > 2 volts)

For distribution of growth rates, evaluate voltages of indications from 2 prior consecutive inspections (e.g., 1996 to 1998 growth rate, 1998 to 2000 growth rate)

- If less than 200 data points in either of the distributions, must combine
- If greater than 200 data points in both distributions, use most limiting
- If combined distribution contains less than 200 data points must use bounding industry growth rate distribution from similarly designed and operated units

For predicting the EOC voltage distribution, the growth rate distribution assumes negative growth rates are zero

ABILITY TO PREDICT EOC DISTRIBUTIONS

Results to-date indicate GL 95-05 methodology provides reasonable predictions of EOC voltage distributions

Licensees routinely evaluate inspection results

CONCLUSIONS/SUMMARY

Ability to detect and size cracks has improved over the years

More rigorous approach for addressing tube integrity - Condition monitoring and operational assessment

NRC and industry are addressing technical issues

Damage Propagation

(agenda items 11 and 14)

October 12–13, 2000

Joseph Muscara, Division of Engineering Technology, Office of Nuclear Regulatory Research
301-415-5844

Steven Arndt, Division of Systems Analysis and Regulatory Effectiveness, Office of Nuclear Regulatory Research
301-415-6502

Steven Long, Division of Systems Safety and Analysis, Office of Nuclear Reactor Regulation
301-415-1077

William Shack, Argonne National Laboratory
630–252–5137

ACRS Ad Hoc Subcommittee on the Differing Professional Opinion Issues

Jet Impingement and the Potential for Propagation of Failure (J. Muscara)

Jet Velocities and Particle Motion (S. Arndt)

Technical Basis for Critical Size for Cracks for Jet Cutting (S. Long)

Crack Growth and Tube Failure (W. Shack)

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.

- Cited experience in superheater of fossil power plant, extrapolated data from coal gasification studies

Ablation or erosion can be due to mechanical processes, corrosion processes, or a combination of the two.

- In many particle and droplet impingement problems the removal mechanism is almost solely mechanical in nature, although the actual damage processes could be either mechanical cutting or fatigue.

- For particulates in a corrosive atmosphere, the removal mechanism varies from predominantly corrosion at low velocities to mixed mechanical processes/corrosion at intermediate velocities and finally predominantly mechanical at higher velocities.

To further address the potential for propagation of tube failures by erosion the literature on particle erosion was reviewed and an Experts Meeting on Jet Impingement Effects and Leak Rates from Steam Generator Tubes During Severe Accidents was held at Argonne National Laboratory on November 19, 1999 (Minutes December 10, 1999 PDR).

Erosion —Ian Wright, ORNL and John Stringer, EPRI

Severe Accidents—Jason Schaperow, NRC and Mati Merilo, EPRI

High temperature fracture mechanics—Ashok Saxena, Georgia Tech

NRC, ANL staff and other industry representatives

Stringer and Wright strongly felt that fossil experience with superheater tubing could not be extrapolated to steam generator conditions. Fireside atmosphere of a fossil plant contains a heavy loading of large, hard ash particles which would be entrained in the jet and greatly accelerate erosion. Such types of particulates would not be present in the secondary side of a steam generator.

Stringer also felt that droplet erosion during design basis accidents was unlikely because the jet would expand into fine droplets, and droplet erosion is strongly size dependent ($\sim d^3$)

Water droplet erosion in steam turbines occurs primarily from the condensation of the fine droplets entering the turbine into larger drops, which are then struck by the rotating blades ("baseball bat erosion")

In NUREG–1570 extrapolation of data from coal gasification plant studies ablation rates are proportional to the density of the fluid and to the cube of its velocity and are affected by the temperature only through fluid density.

Inconsistent with the observed velocity and temperature dependence in the gasification data.

Cubic velocity dependence based on mechanical removal processes, not corrosion dominated processes. Effect of lower temperature on the corrosion rates is ignored.

Based on literature review and experts meeting, the jet velocities and associated particle motion were recognized as key parameters in erosion process. Assistance was sought from the Division of Systems Analysis and Regulatory Effectiveness to carry out the needed calculations.

Additional experiments to address jet cutting issues were also planned.

- 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

Numerical Study of Jet-Impingement Flow due to Severe Accident Induced Steam Generator Tube Leak

- In 1997 NRR did an analysis of velocities at adjacent tubes due to tube leakage and estimated that the average velocity would be between 775 feet/sec and 997 feet/sec. In February 2000, RES developed another calculation of jet velocity that ranged from 233 feet/sec to 520 feet/sec.

- A more detailed study of this phenomena was conducted using Computational Fluid Dynamics (CFD) to support RES experimental investigation of jet cutting on Steam Generator tubes.

- This work was done for the NRC by Prof. Ugo Piomelli using the NPARC CFD code, a Air Force developed code for high velocity flows. This work shows jet velocity and aerosol particle velocity at the adjacent tubes to be less than 1000 feet/sec.

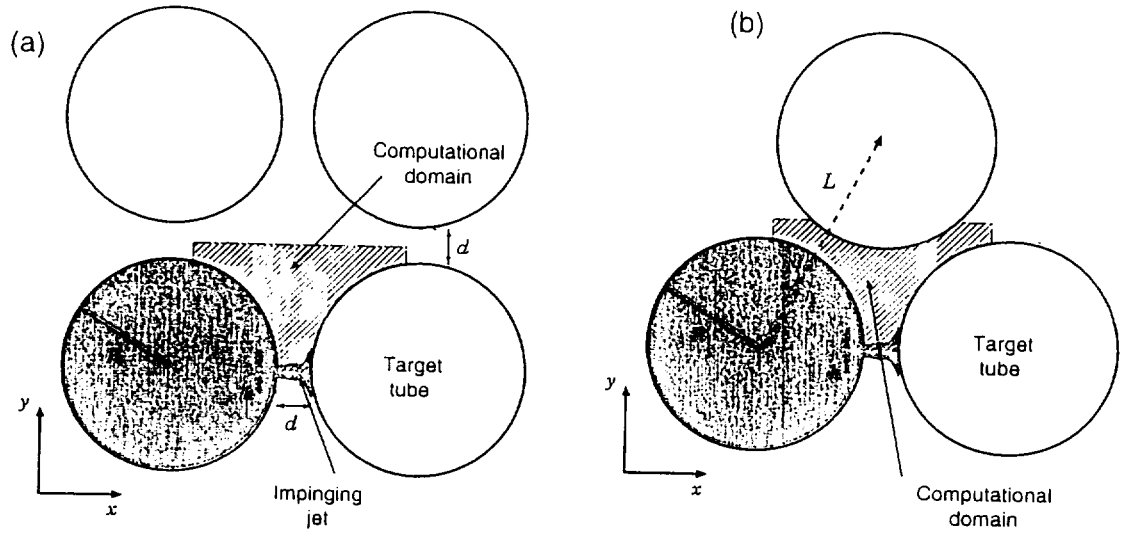
Scope of Study

- The study included a base case and several variations including studies to validate the computational grid. Other variations investigated the dependance of steam temperature, pressure, Steam Generator geometry, and crack thickness.

- The crack was assumed to be of infinite length

- The particle size and density was developed by a VICTORIA calculation and assumed to be equally distributed in the steam in the primary system.

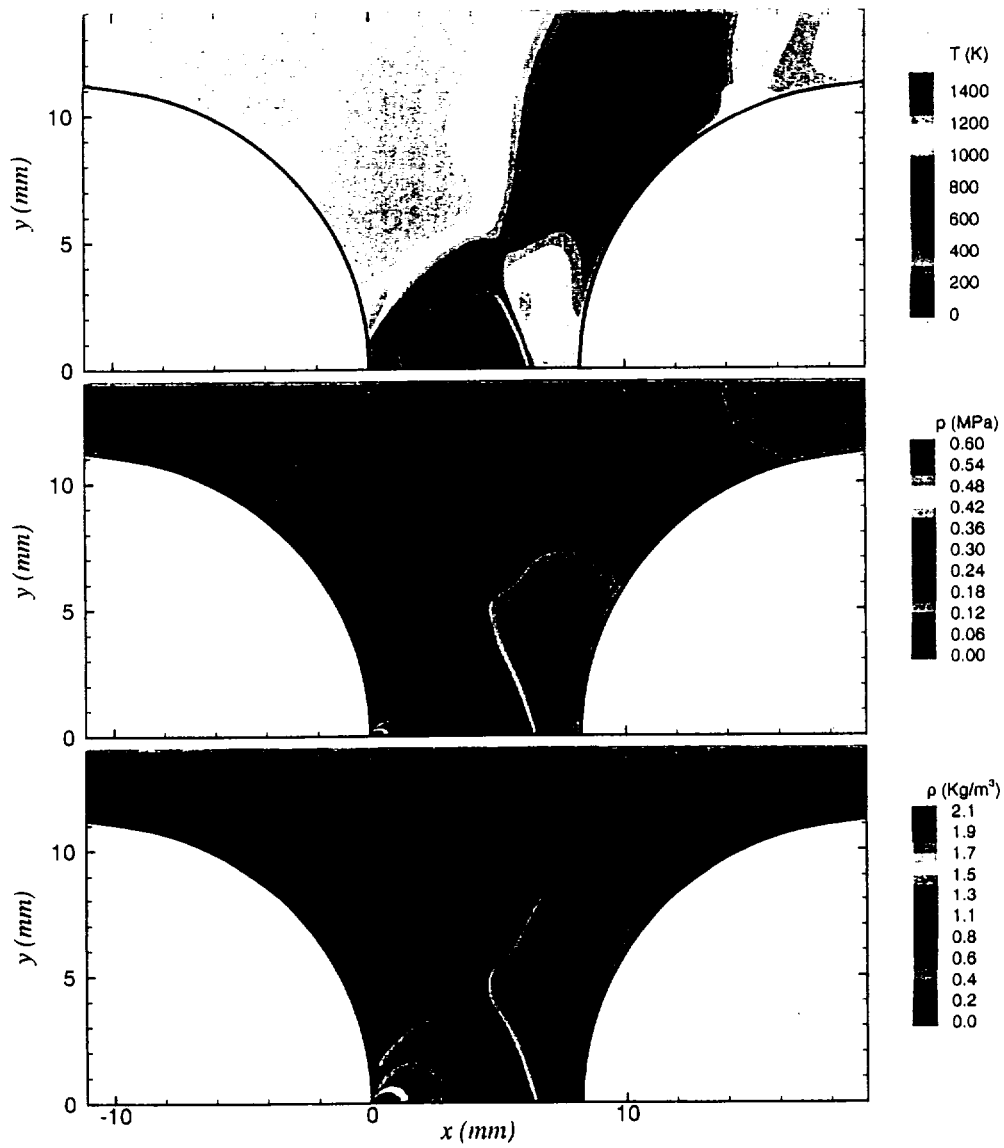
- Particle velocities were calculated along with fluid velocities.



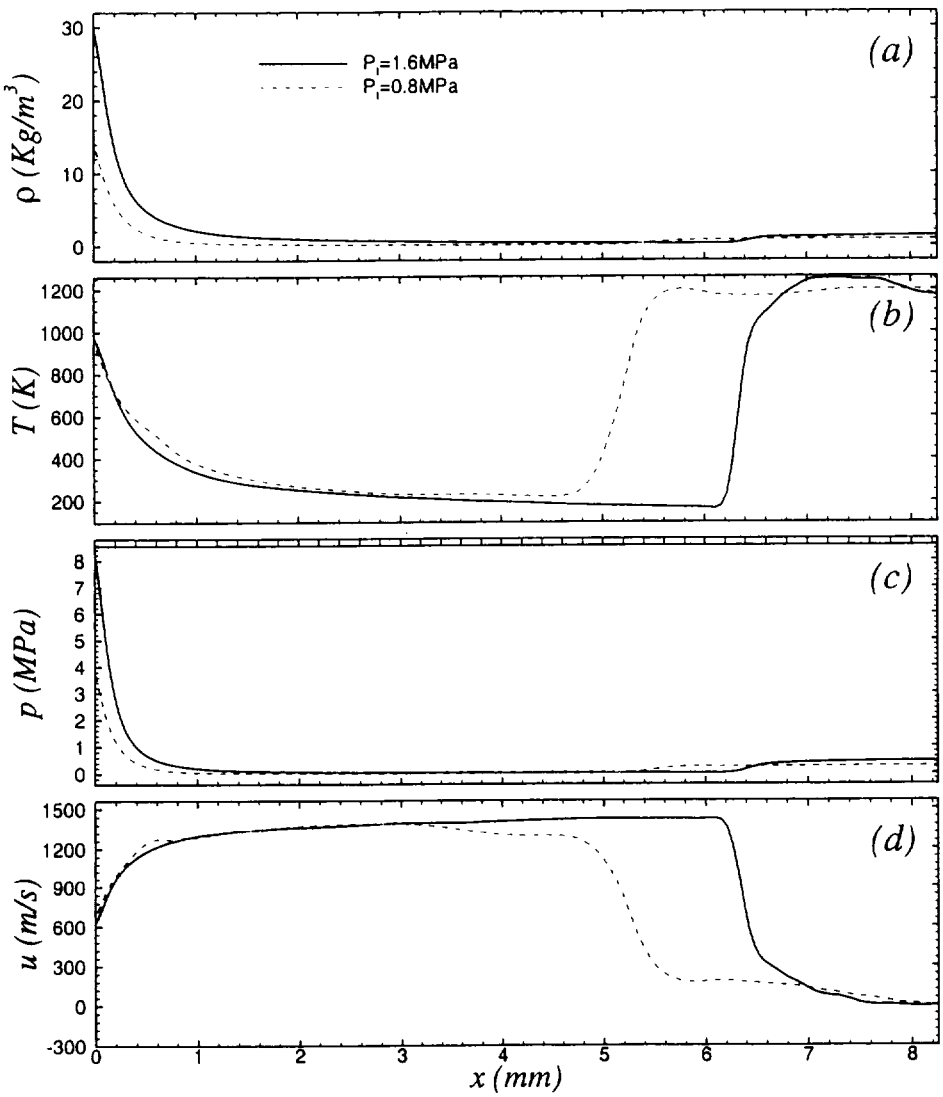
Sketch of the physical configuration (not to scale). (a) Square grid; (b) Triangular grid.

Calculation	Configuration	R mm	d mm	h mm	P_t MPa	T_t K	$\dot{m}$ Kg/(s·m)
Baseline	Square	11.25	8.25	0.25	16	1175	4.7
Dense grid	Square	9.53	7.87	0.25	16	1175	4.7
Triangular grid	Triangular	9.53	6.35	0.25	16	1175	4.7
High T_t	Square	11.25	8.25	0.25	16	975	4.7
Large crack	Square	11.25	8.25	2.5	16	1175	47.

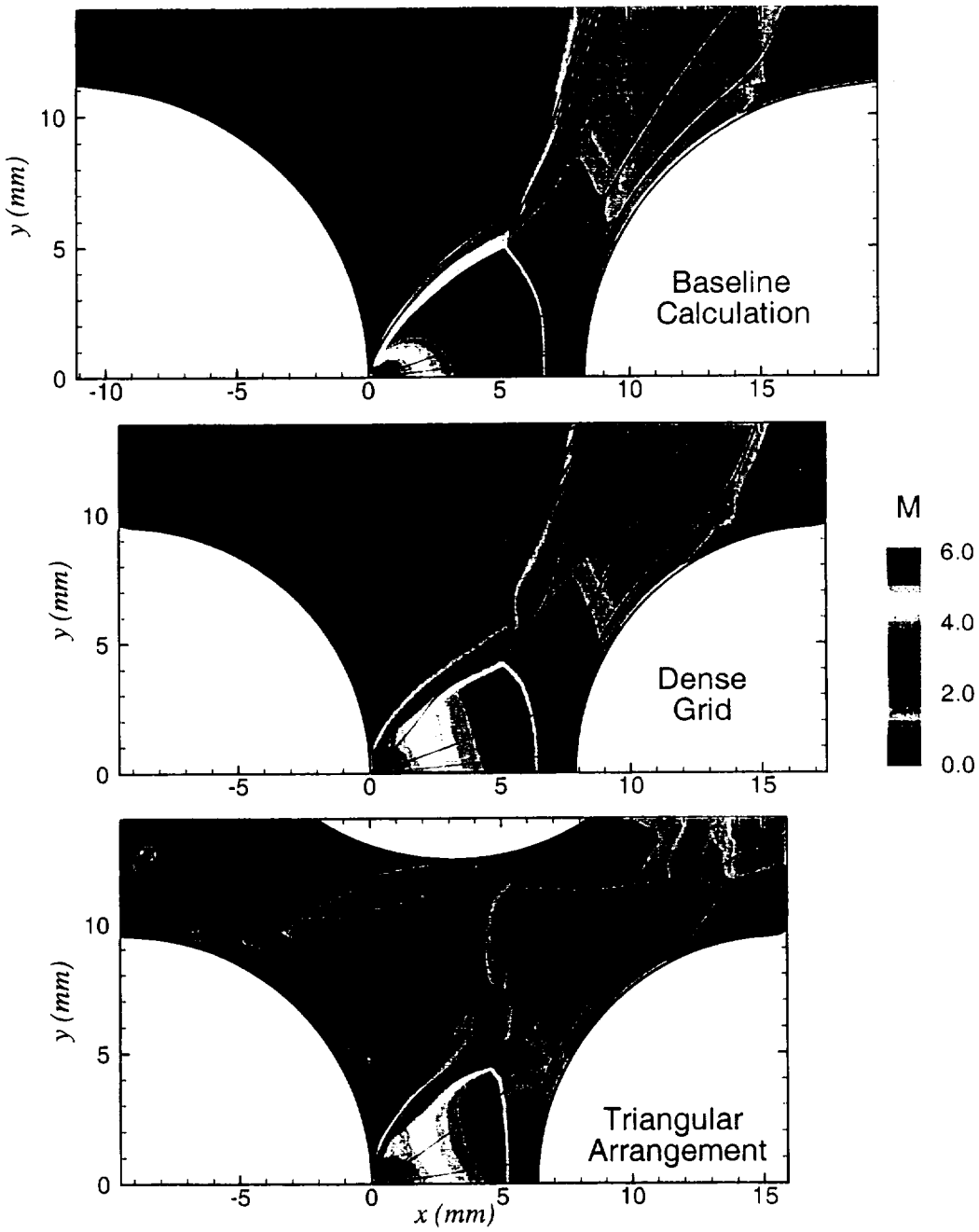
Table 1: Flow parameters.



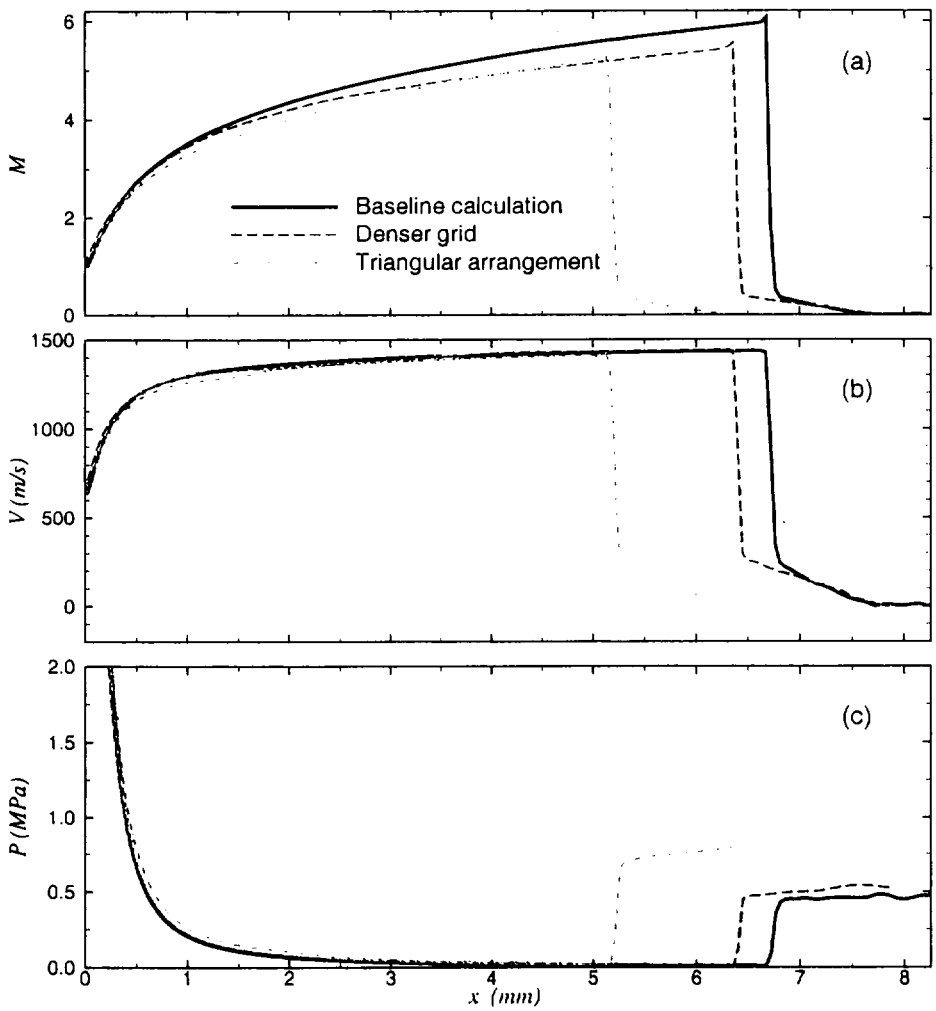
Thermodynamic variables. Fine grid, $P_t = 16MPa$ (Case F16a). (a) Temperature, T; (b) pressure, p; (c) density, ρ.



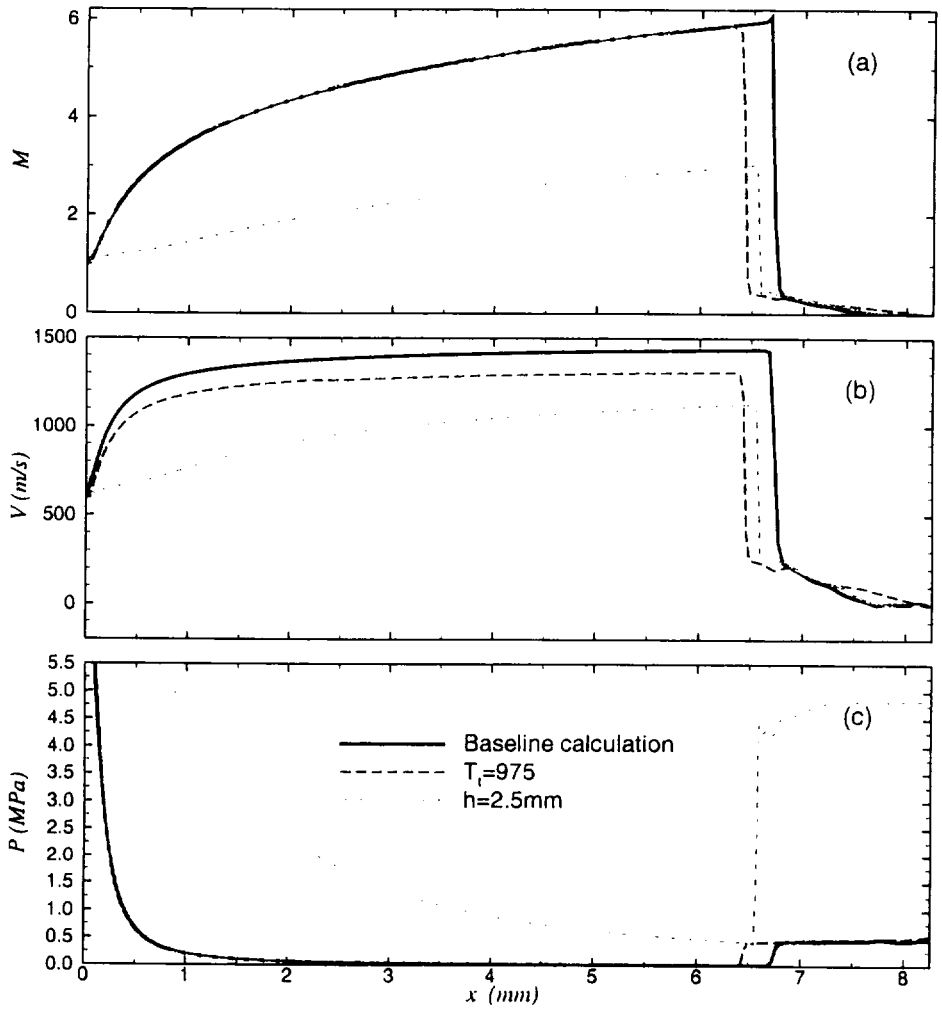
Profiles of (a) density, (b) pressure, (c) temperature and (d) u velocity component on the line of symmetry. Fine grid. —— $P_t = 16MPa$; - - - $P_t = 8MPa$.



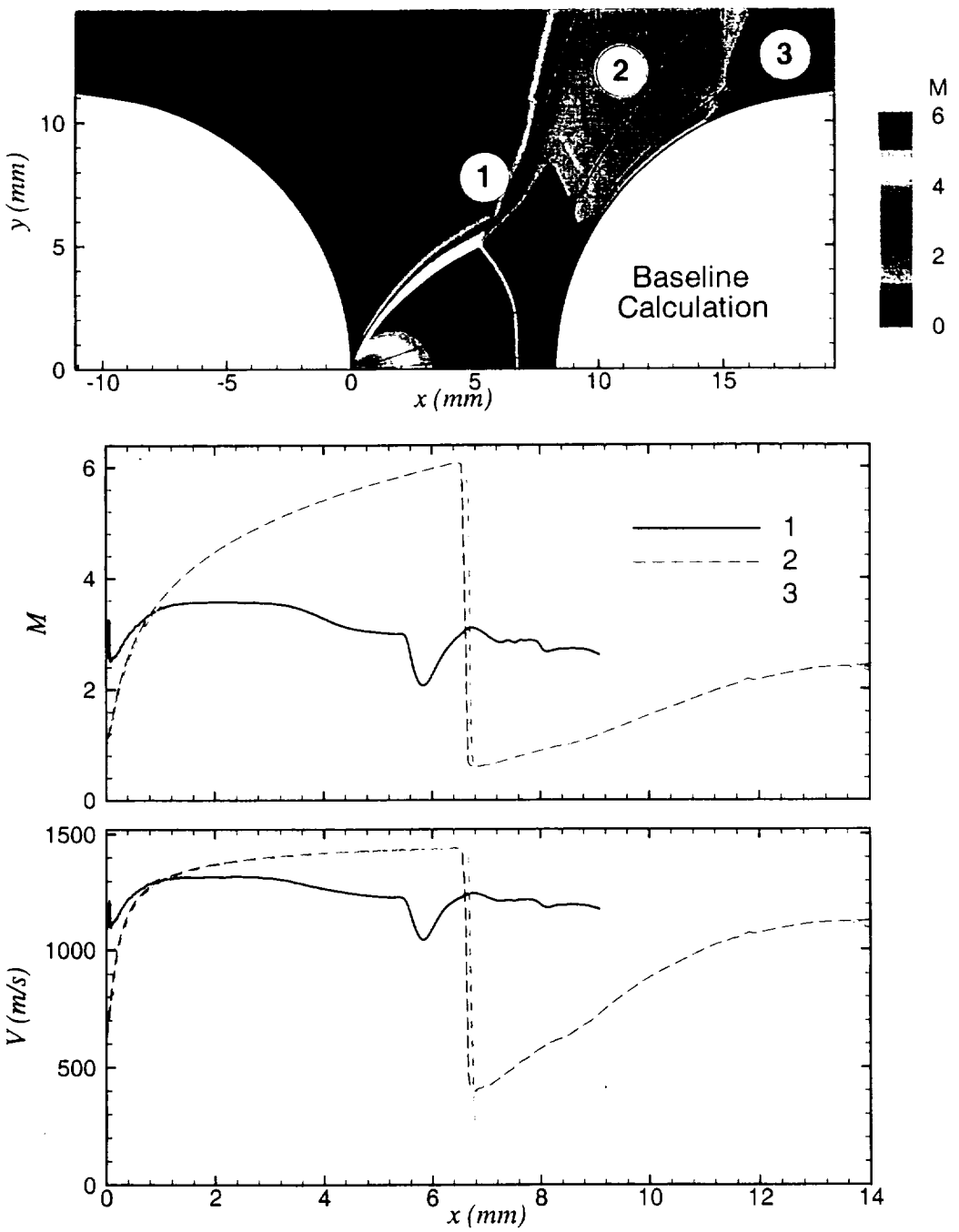
Mach number contours and selected streamlines.



Mach number, velocity and pressure on the line of symmetry.



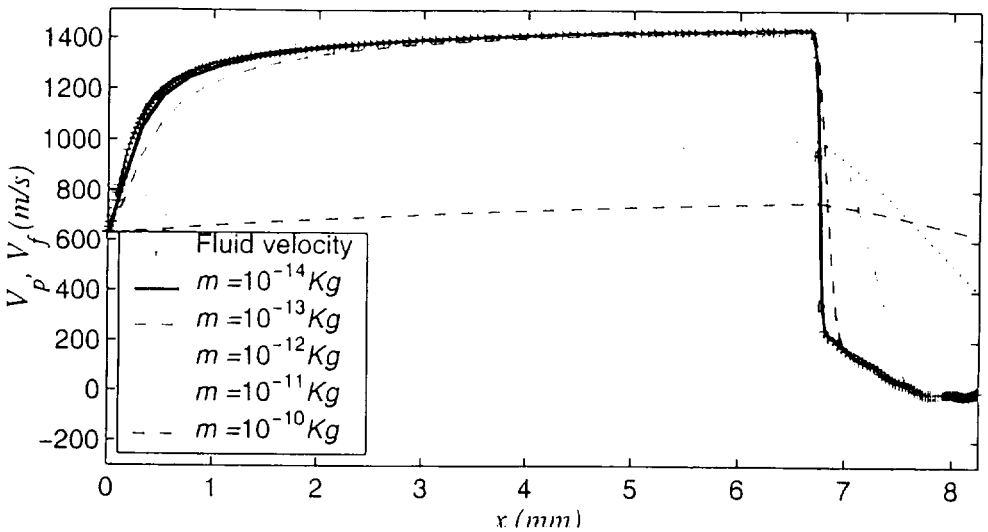
Mach number, velocity and pressure on the line of symmetry.



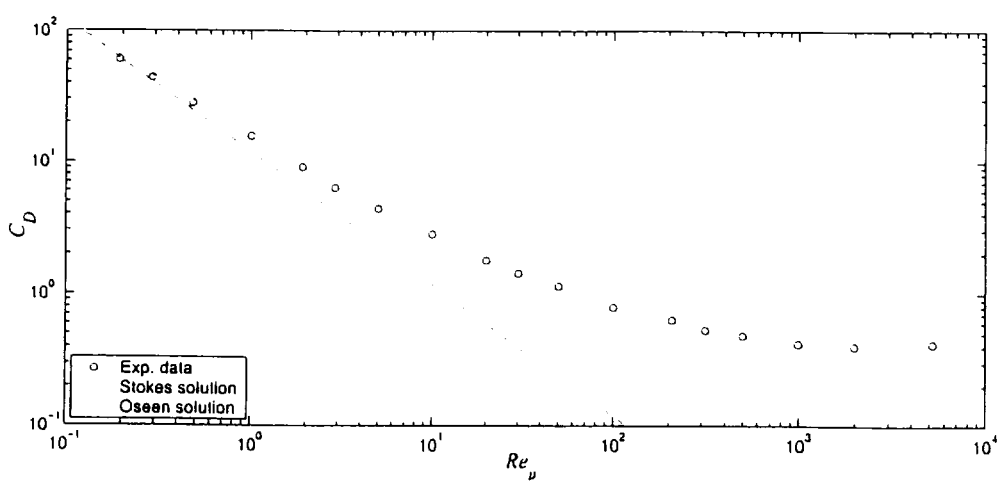
Mach number and velocity distributions along selected streamlines. Baseline calculation.

Particle Velocities

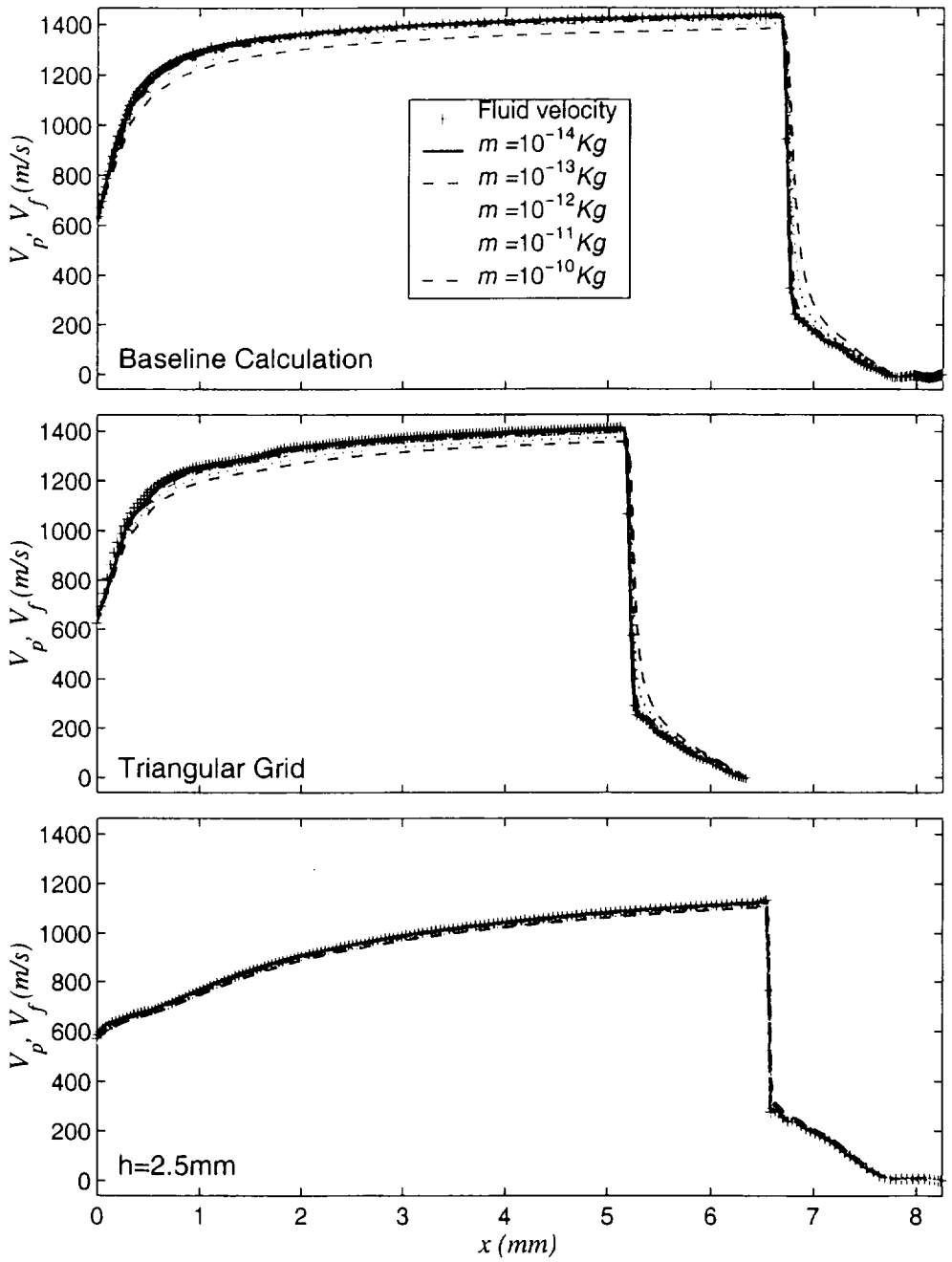
- It is assumed that the particles accelerate to fluid velocity in the crack.
- The first calculation assumed Re_p using Stokes law.
- Because $Re_p > 500$ for most of the particle trajectory, Stokes law is invalid.
- Used Oseen solution and data for high Re_p.
- Final calculations show low velocities <300 meters per sec for particles of interest.



Velocities of particles moving on the line of symmetry. Drag coefficient calculated from the Stokes law (5).



Drag coefficient of a sphere.



Velocities of particles moving on the line of symmetry. C_D given from an interpolation of the experimental data.

High Temperature Erosion Tests

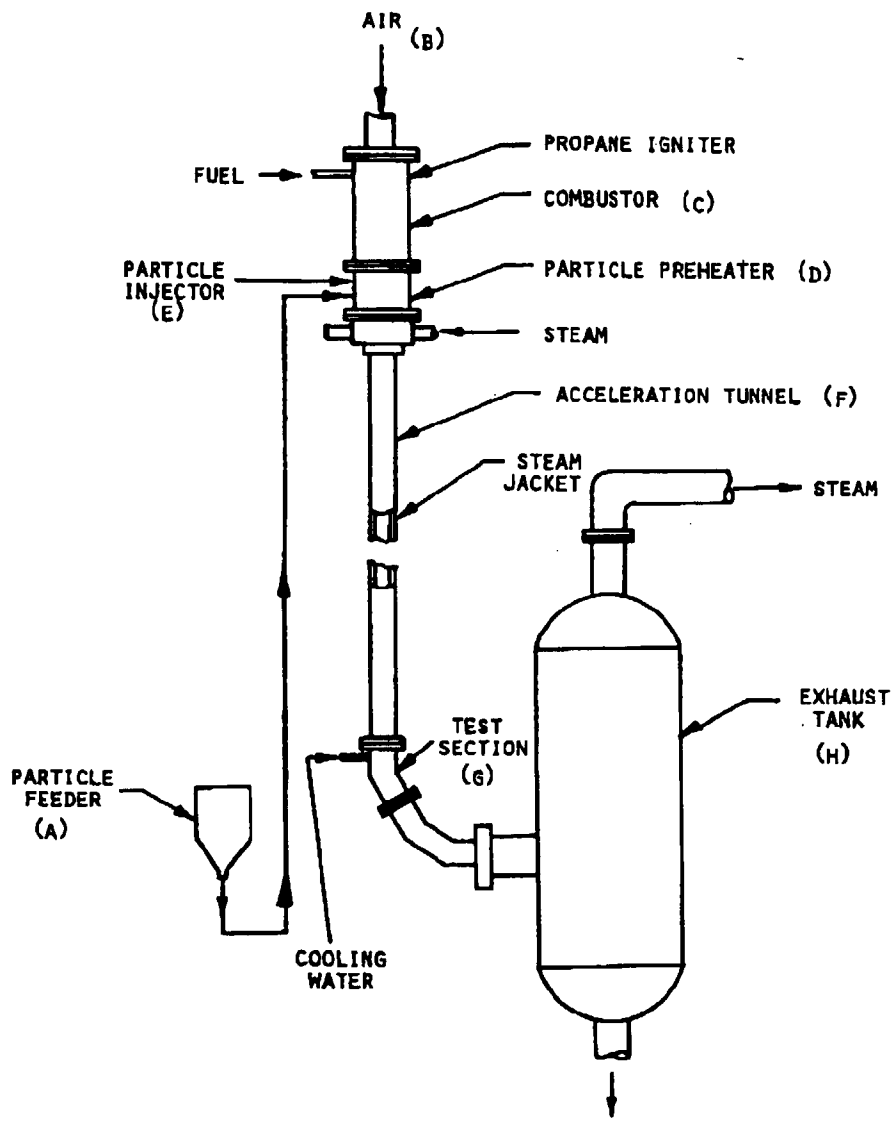
Accident conditions

High temperature steam, nominal temperature 700°C, pressure 2350 psi

Particle loading due to aerosol transport from the molten core. Accident analyses suggest following concentration of particles in stream:

Species	Conc. (g/m^3)	Species	Conc. (g/m^3)
Ag	98.4	MoO_2	0.8
In_2O_3	7.4	In	0.7
$CsMoO_4$	2.5	SnO	0.6
SnO_2	2.2	Sn	0.5
CsI	1.1	CdO	0.1
Cd	1.0	Ag_2Te	0.1
		Total = 115.4 g/m^3	

Median particle diameter is ≈1.5 μm, bulk of particles < 3 μm.



The erosion apparatus uses a propane combustor to generate a high temperature gas stream.

Resulting atmosphere is different from steam environment, although like steam it is an oxidizing environment

To assure that the temperature at the target is correct, the combustor must run a somewhat higher temperature.

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. Oxides are much harder than the metals

Material	Density, (g/cm^3)	Hardness at RT	Melting Temp.(°C)	Fraction of Melting Temp
Ag	10.49	27 Vickers	962	0.8
Ni	8.91	64 Vickers	1453	0.6
NiO	6.67	5.5-6 Mohs	1985	0.4
SnO_2	6.95	6-7 Mohs (≈600 Vickers)	1632	0.5

Initial series of tests was run at 1000 ft/s and incident angles of 20, 30, and 45° to determine the angle of incidence that produced the highest erosion rates. As expected it was 30°.

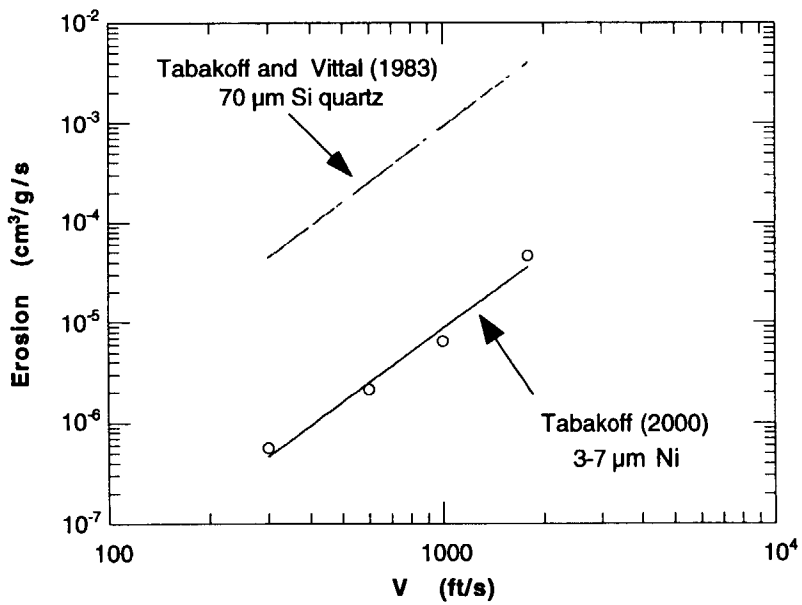
Subsequent tests performed at 30° angle of incidence over a range of velocities with Ni and NiO powders

Initial results for Ni particles

Velocity		Erosion Rate	Erosion Rate
(ft/s)	(m/s)	(mg/g)	(cm^3/g)
300	91	0.005	5.94×10^{-7}
600	183	0.019	2.26×10^{-6}
1000	305	0.057	6.77×10^{-6}
1800	549	0.415	4.93×10^{-5}

Tests were also run with NiO powders. Because of the greater hardness of the oxides, this was expected to produce a very conservative bounding estimate of the possible erosion rates. Instead deposition occurred at all velocities.

Additional tests are being performed with Ni + 15% NiO and Ni + 15% Al_2O_3 to determine if there is a synergistic interaction that could result in increased erosion rates.



Erosion rates for Ni particles are much lower than existing data for 70 µm Si quartz particles, but velocity dependence is typical of that observed for particle erosion ($\sim V^{2.5}$)

To determine rate of thinning of tubes, this erosion data must be coupled with the results of the CFD analyses of the steam behavior:

Wall thinning = Erosion Rate (cm^3/g) x Mass flux of particles ($g/cm^2/s$)

The total mass flux from the crack is 4.6 $g/cm^2/s$. The CFD analyses indicate considerable spreading of the jet, which reduces the mass flux on the tube, and that the velocities of the particles at impact are on the order of 300 m/s. Preliminary estimates based on the Ni particle data and assuming the spread of the jet decrease the mass flux by a factor of 10 and velocity of 300 m/s gives a wall thinning rate of 4 mil/h.

Jet Cutting Under Design Basis Accident Conditions

High pressure single phase jets are very effective at cutting soft materials. Much less damaging on harder materials

- A four hour duration test at ANL with a 2500 psi jet from a 1/32 in. orifice at room temperature produced only a light burnish mark on the Alloy 600 target
- Liquid droplets can erode hard materials through fatigue damage. Such droplet damage has been studied for a range of materials from acrylic plastics (aircraft windshields) through hard turbine blade materials.

Droplet damage has strong dependence on droplet size ($\sim d^3$) and velocity ($\sim V^5$)

Tests will be performed at ANL under the bounding conditions, 300°C and 2500 psi, to determine potential for such damage in steam generators. More prototypical pressure–temperature histories will also be considered.

"Justification" of ¼-Inch Threshold for Cracks Capable of Steam Jet Cutting

Originated during NUREG-1570 work.

RES (Dominion Engineering) provided estimates of flaw sizes with *uncorrelated* length and depth distributions

Majority of cracks in (gamma function) distributions were artefacts of fit, below available data

This resulted in large number of very short, nearly through wall cracks in distribution

Staff wanted to reflect potential for gas jet cutting, but did not believe that a very short crack would open enough to produce a damaging jet.

Dilemma (seemingly) was resolved with classical logic for stress-corrosion crack aspect ratio that indicates length should be at least 5-times depth. (~0.25" for 50 mil tube wall)

"Justification" of ¼-Inch Threshold for Cracks Capable of Steam Jet Cutting (Continued)

Farley Amendment Application Revealed Problem with Basis:

- Westinghouse evaluated *segments* of crack profiles from Farley inspection
- Result was high probability that at least one short segment would propagate through wall in event of severe accident conditions
- But most of this probability was for very short segments
- So, staff needed another basis to set limit for assuming gross failure due to cutting

Impromptu conference call among NRC and licensee personnel and contractors reached consensus to continue to use 0.25" threshold because:

- None had a way to accurately quantify the threshold with available data or analysis
- Those with relevant experimental experience were of the judgement that 0.25" cracks were not likely to produce gross failure by cutting.
- Consistency with previous analyses while basis was reexamined.

"Justification" of ¼-Inch Threshold for Cracks Capable of Steam Jet Cutting (Continued)

SER was written and DPO Considerations Document was rewritten to acknowledge this problem.

RES subsequently initiated experimental effort to quantify cutting phenomena.

Results to date indicate that 0.25" threshold was conservative with respect to cutting potential.

Conclusions

- Damage by jet impingement does not appear to be a concern.
- Models have been developed and are available for predicting the structural behavior of good and degraded steam generator tubes under normal operating, accident, and severe-accident conditions.

Extrapolation of Coal Gasification Erosion/Corrosion Data to SG Behavior

In NUREG–1570 data from coal gasification plant studies were used to estimate erosion rates in severe accidents. These resulted in estimates of failure times of 4.9 and 62.6 seconds with and without erodent particles

Experiments used a gas stream made up of 31% H_2, 17% CO, 15% CO_2, 3% CH_4, 1% NH_3, 0.1 to 1.0% H_2S, and the balance H_2O. Corrosion/erosion data for tests without erodent are

	Temperature C (F)	
	899 (1650)	982 (1800)
Velocity ft/s	Material Loss (mils/100 h)	
10	2.9	13.2
100	8.4	27.6

Material loss is due to the removal of corrosion product rather than base metal. Nickel base alloys are very susceptible to corrosion at high temperatures in environments with sulfur.

In NUREG–1570 extrapolation to steam generator conditions assumed that the ablation rates are proportional to the density of the fluid and to the cube of its velocity and are affected by the temperature only insofar as it impacts the fluid density.

Inconsistent with the observed velocity and temperature dependence in the gasification data used for the analysis.

Cubic velocity dependence is appropriate for some mechanical removal processes, but not for corrosion dominated processes. Effect of lower temperature on the corrosion rates is ignored.

Extrapolation probably overestimates erosion rates by orders of magnitude.

Fatigue Damage Model for Droplet Impact Erosion

A semi–empirical model (Springer) seems to describe process reasonably well for materials ranging from Plexiglas to Udimet 700:

$$\frac{dm_e}{dt} = \frac{\rho_s \pi d^3}{4} \frac{dn_p}{dt} \left[93.4 \left(\frac{P}{S} \right)^4 \right]$$

, where m_e is mass/area loss, $\frac{dn_p}{dt}$ the particle flux, ρ_s the density of the target, d the droplet diameter, P the impact pressure, S a strength parameter

$S = 10\ \sigma_{ult}\ (b-1)$, where b is the fatigue strength exponent, $N = \left(\frac{\sigma_{ult}}{\sigma} \right)^b$ for fatigue life N under stress σ and σ_{ult} is the ultimate stress

$P = \rho_w C_w V$, where ρ_w is the density of water, C_w the speed of sound

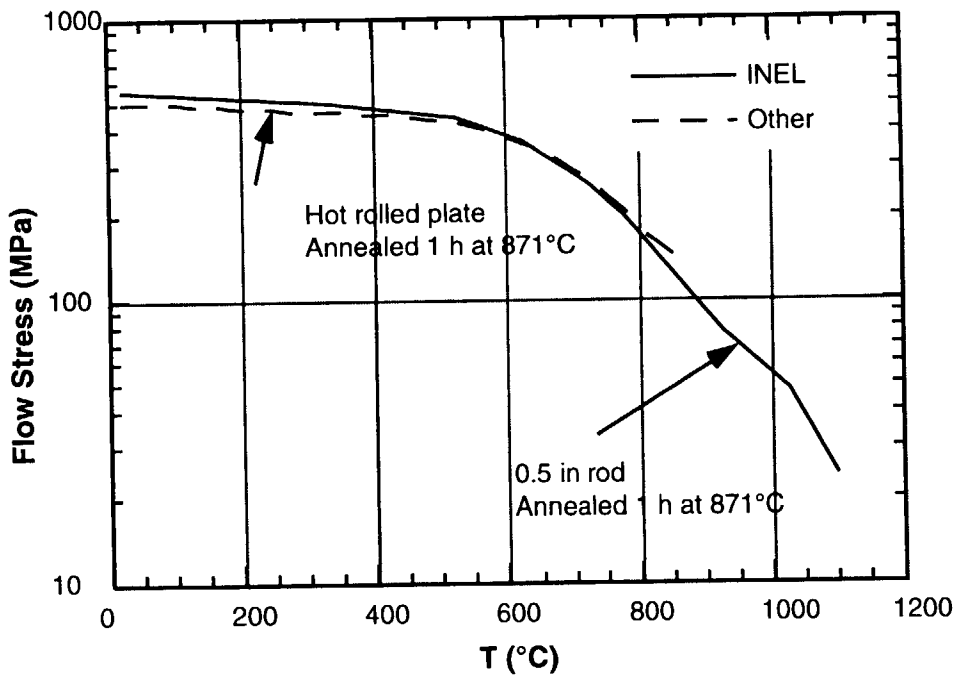
Since $\frac{dn_p}{dt} \sim V$ and $P \sim V$, $\frac{dm_e}{dt} \sim V^5$

Failure of Degraded Steam Generator Tubes

Extensive work by NRC and industry during 70's and 80's had developed and verified models for failures of flawed steam generator tubes under design basis accident conditions at 300°C at pressures up to failures of unflawed tubes (10–11000 psi)

In addition to the "classical" failure of degraded tubes under design basis accidents, there is also a potential for risk due to failure under severe accidents conditions

> "High–dry" scenarios with depressurized secondary side and core melting leads to tube temperatures of 700°C when failure of other primary components is predicted with up to 2350 psi internal pressure.



At high temperatures the flow stress of Alloy 600 decreases markedly and creep effects become significant

Failures at low strain rates are controlled by creep, at high strain rates controlled by flow stress

For a part throughwall crack

"Failure" can mean growth of the crack through the wall to create a leaking crack without a change in length of the crack. Any change in length would require a further increase in pressure

"Unstable failure" or "burst" indicates that the crack grows in length without a further increase in pressure

Failure Under Design Basis Conditions

A variety of models have been developed to describe unstable failure throughwall cracks and ligament failure of part–through axial cracks for design basis conditions. That by Hahn et al. and Erdogan is widely used

$$p_{cr} = \frac{\bar{\sigma} t}{m R_m} = \frac{p_b}{m} \text{ where}$$

$$p_b = \frac{\bar{\sigma} t}{R_m} = \text{burst pressure of unflawed tube}$$

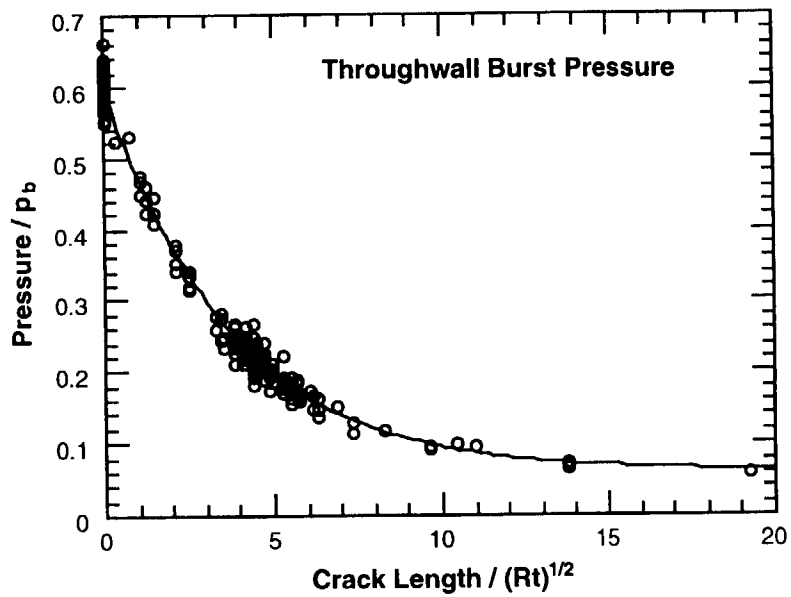
$\bar{\sigma} =$ flow stress $= k(\sigma_y + \sigma_u)$, R_m is mean tube radius, t wall thickness

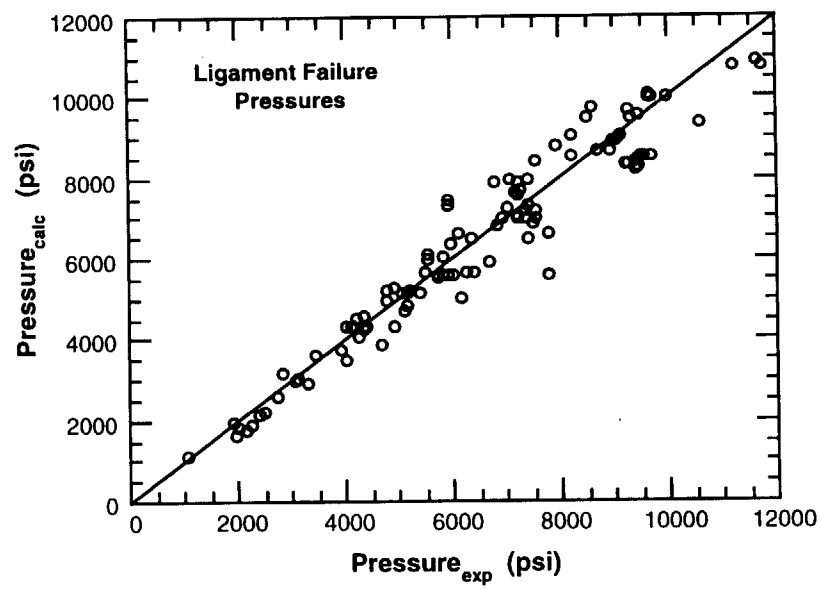
$$m = 0.614 + 0.481\lambda + 0.386\exp(-1.25\lambda), \quad \lambda = \left[12(1-\nu^2)\right]^{\frac{1}{4}} \frac{c}{\sqrt{R_m t}} = \frac{1.82c}{\sqrt{R_m t}}$$

For part through cracks a similar formulation can be used with a different expression for the stress magnification factor m:

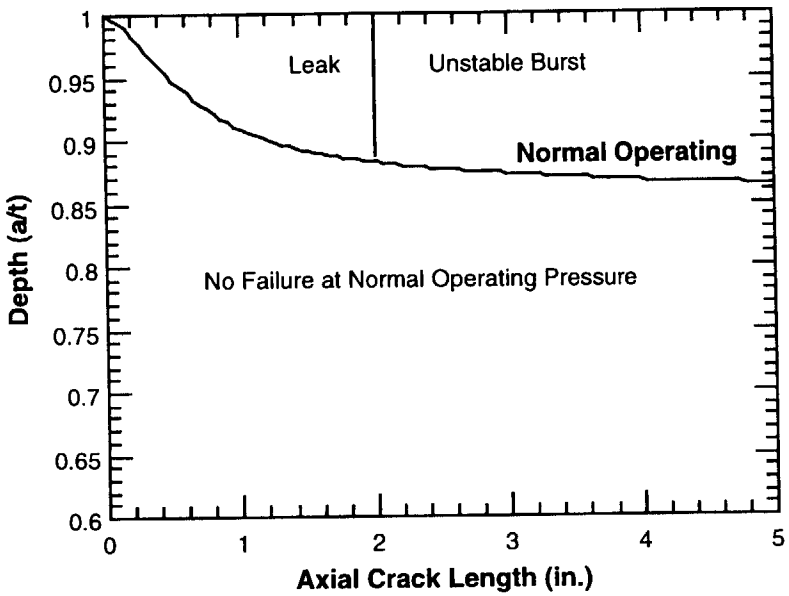
$$p_{cr} = \frac{\bar{\sigma} t}{m_p R_m} = \frac{p_b}{m_p} \text{ where}$$

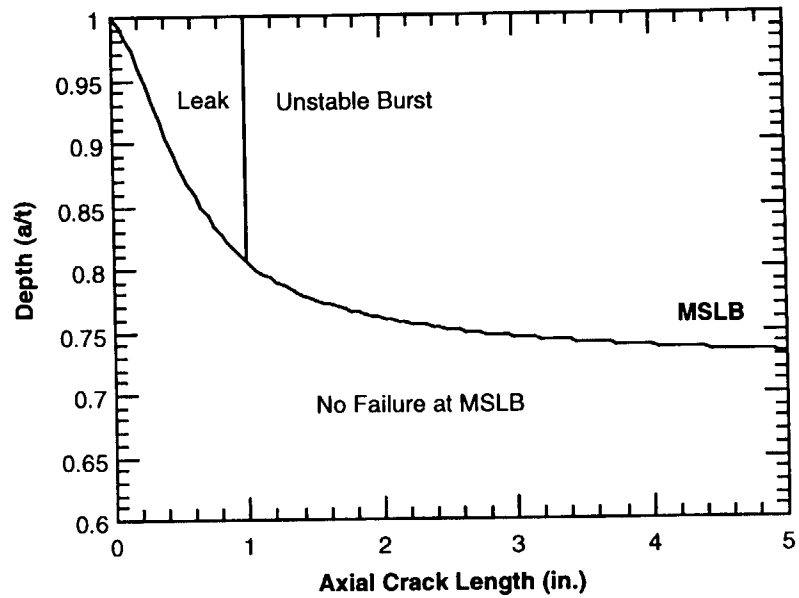
$$m_p = \frac{1 - \alpha\left(\frac{a}{t}\right)\frac{a}{mt}}{1 - \frac{a}{t}} \text{ where } \alpha\left(\frac{a}{t}\right) = 1 + \left(\frac{a}{t}\right)^2\left(1 - \frac{1}{m}\right)$$

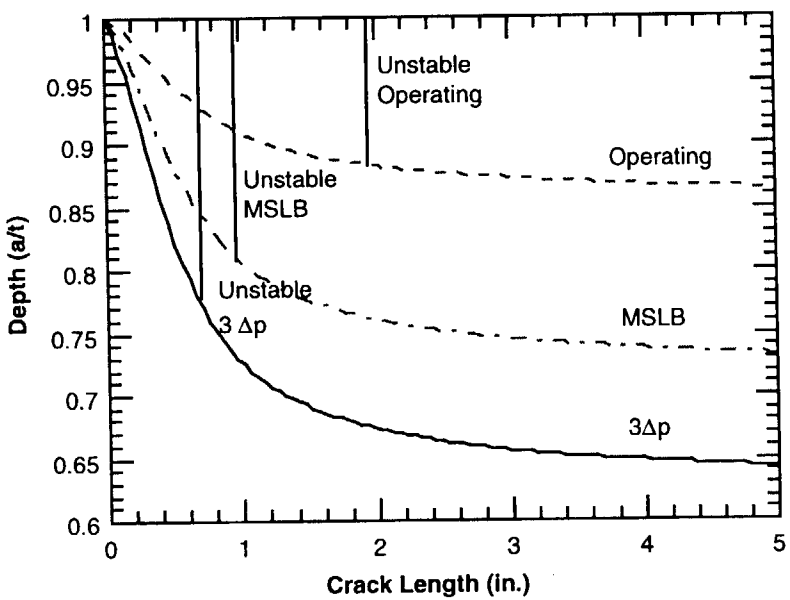




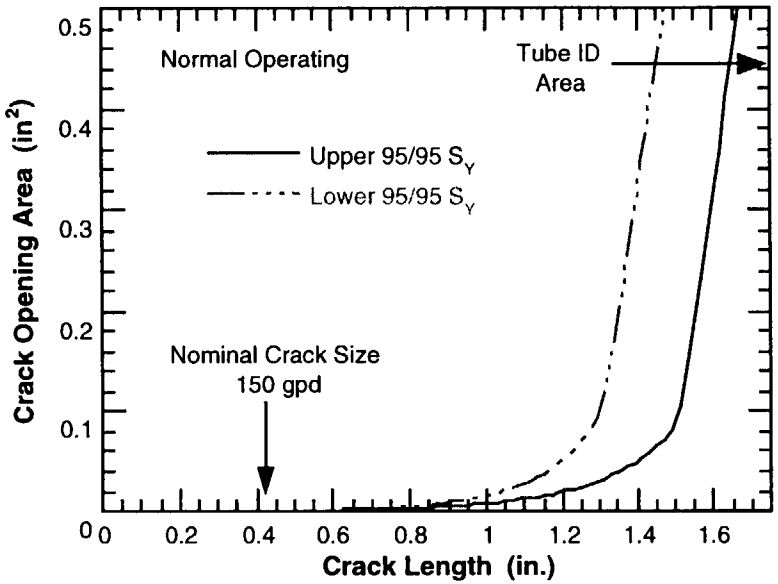
Extensive databases have been developed by NRC and industry to validate models for operating temperatures.

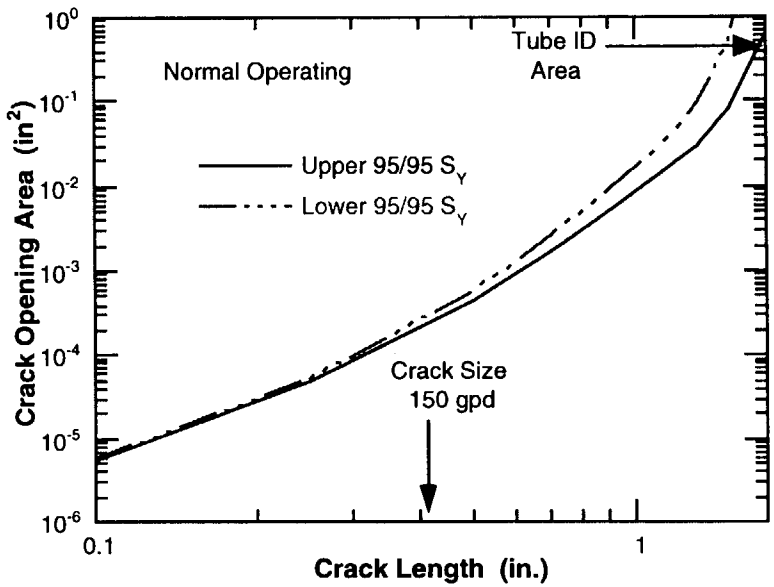


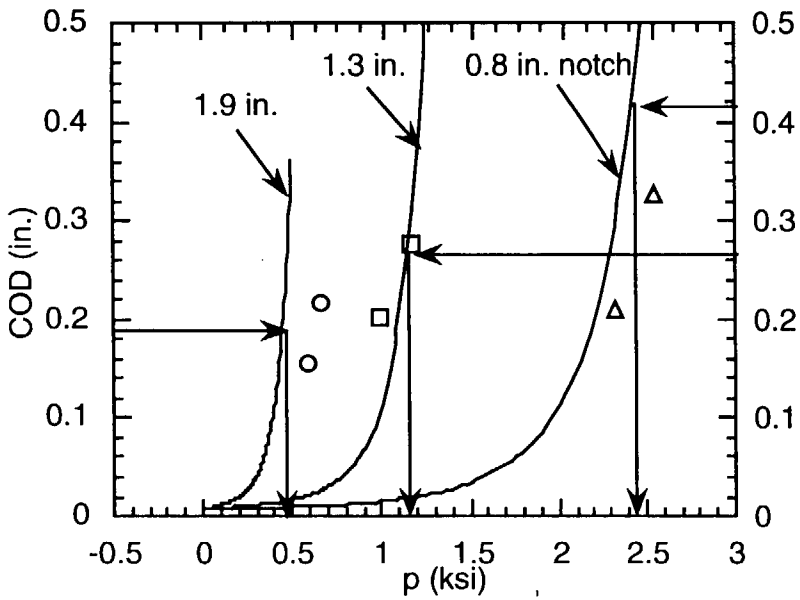




Effect of crack length and depth on behavior under normal operating, MSLB, and 3Δp test conditions.

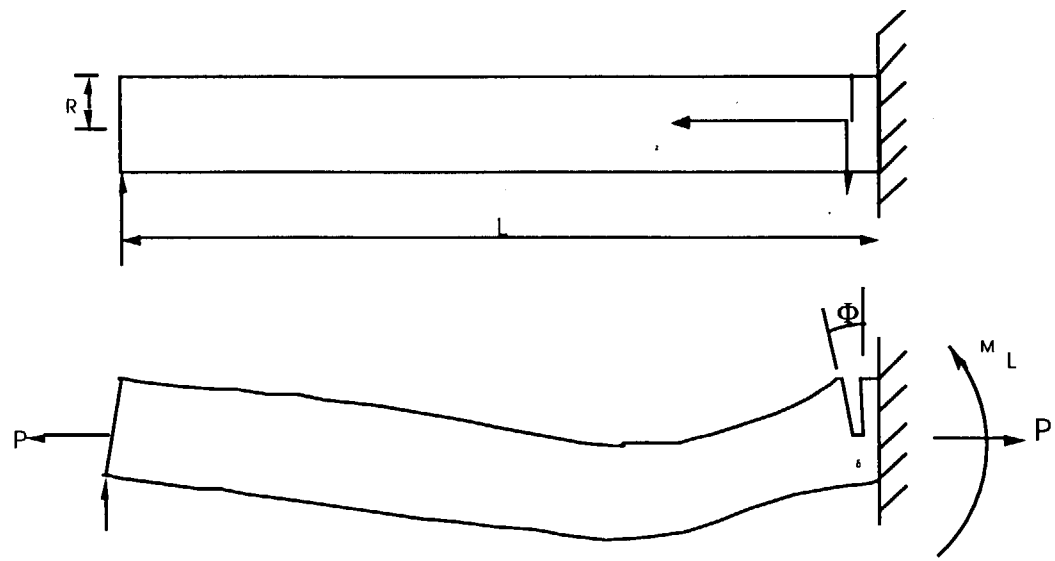




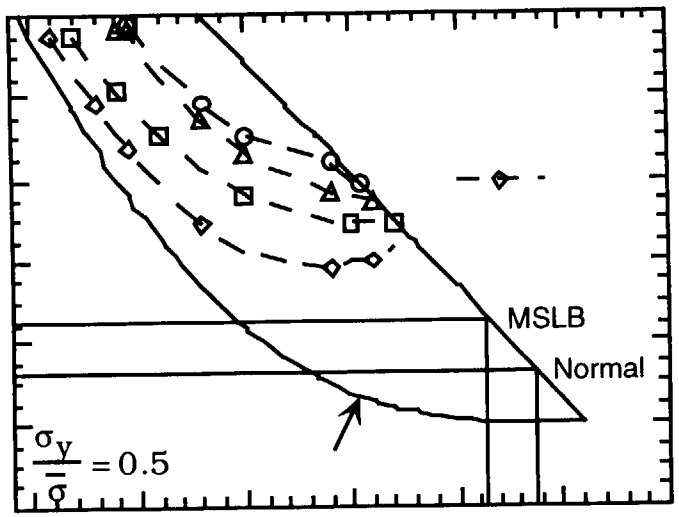


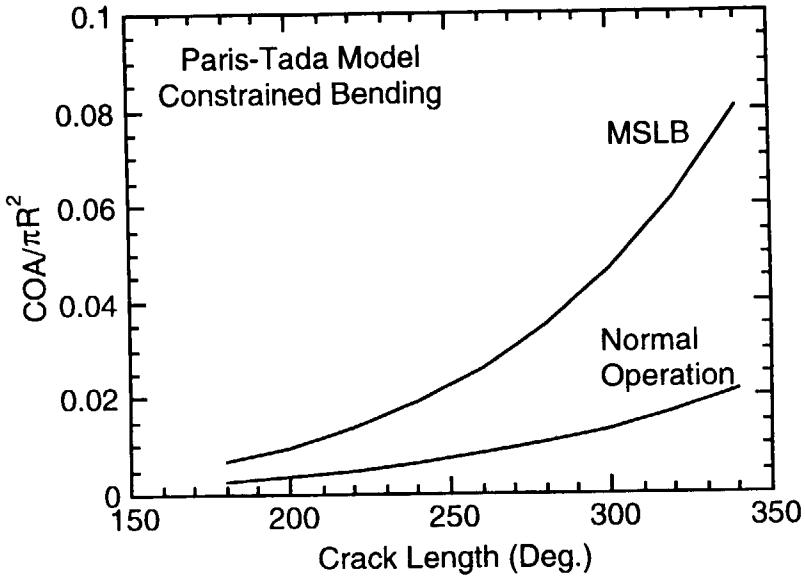
Crack opening areas can be computed by fracture mechanics analyses. Tests and finite–element analyses at ANL show that a simple model due to Zahoor gives accurate description.

Circumferential Cracks



Presence of a crack in a pressurized tube produces bending. Behavior depends strongly on whether the bending is constrained and fracture toughness of material.





Constraint due to tube support plate corresponds to $c \geq 0.3$. For circumferential cracks less than 100°, tubes fail by axial burst. Even under MSLB conditions throughwall cracks remain stable until greater than 300° in extent.

Implications for Steam Generator Tube Failures

Primary mode of interest current generators is SCC

SCC primarily associated with regions of high residual stresses or aggressive chemistries

- Tube support plate; crevice regions promote aggressive chemistry
- Roll transition; high residual stresses promote ID (PWSCC). Sludge and residual stresses also promote OD SCC both axial and circumferential.
- Small radius U–bends; residual stresses introduced during fabrication process. Additional stresses can be introduced by “hour glassing” of flow slots by denting

Cracks in small radius U–bends have the greatest potential for gross failure

- TSP cracks are limited in length by thickness of TSP and opening and leakage is constrained by TSP.
- High stress region at roll transition is limited in extent (typically less than 10 mm).
- Circumferential cracking can be extensive but because of loading does not result in concomitant reduction in margin or increase in leakage.

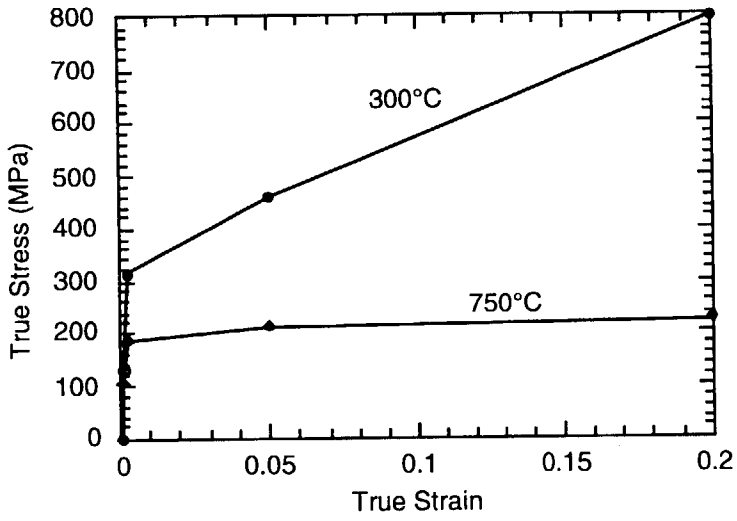
Failure of Steam Generator Tubes at High Temperatures

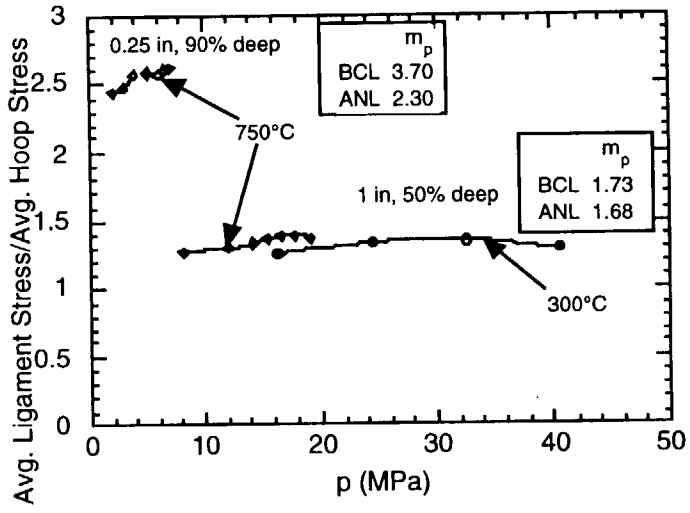
Prediction of temperature of steam generator tubes during severe accident is complex. Temperature rates (strain rates) for tests were chosen to bound those predicted by thermohydraulic analyses (3–13°C/min for the transients considered)

If temperature ramps are sufficiently rapid, burst temperature should depend only on short–term tensile properties and be history-independent, i.e., *Flow stress model.*

If temperature ramps are sufficiently slow, burst temperature will depend on the pressure and temperature history, i.e., *Creep rupture model.*

Failure model at normal operating temperature accounts for crack geometry through stress magnification factor m_p and is confirmed by an extensive data base





Finite element analyses show that m_p is insensitive to the stress strain curve and agrees well with ANL correlations at normal operating and severe accident temperatures

At high temperatures we assume m_p factors derived from low temperature tests are applicable and failure can be determined by a creep time–fraction model

Creep rupture model (linear damage rule)

$$\int_0^{t_f} \frac{dt}{t_R(T, m_p\sigma)} = 1,$$ where t_R = time to creep rupture at temperature T and stress $m_p\sigma$

Unlike flow stress model, the time to failure t_f is dependent on the history of loading.

Validation Tests for Creep Rupture Model

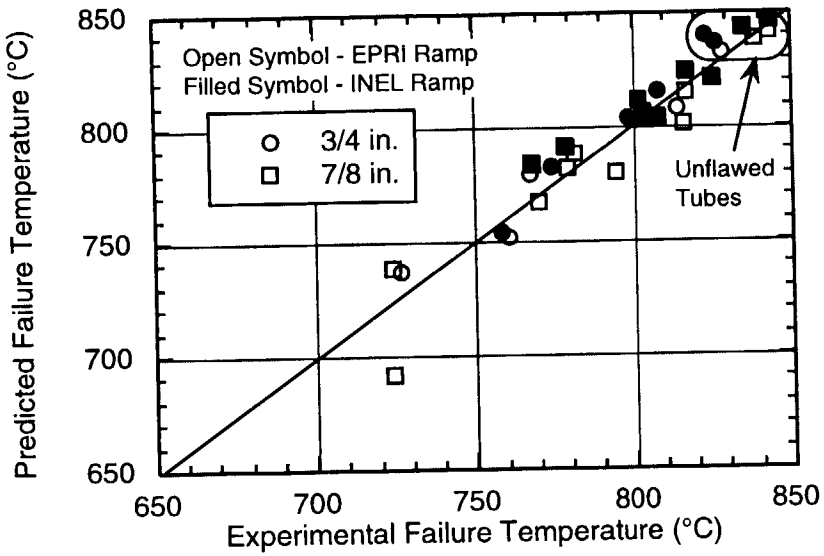
Isothermal, constant pressure creep failure tests

- Tests with deep cracks (≥90%)

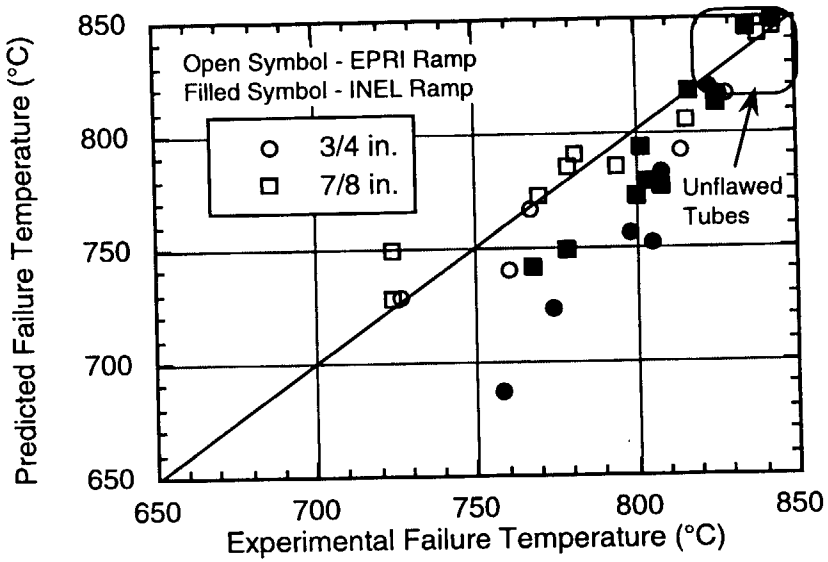
Constant ramp rate tests

- Constant-pressure temperature ramp
- Isothermal pressure ramp

"Prototypical" tests under varying temperature history

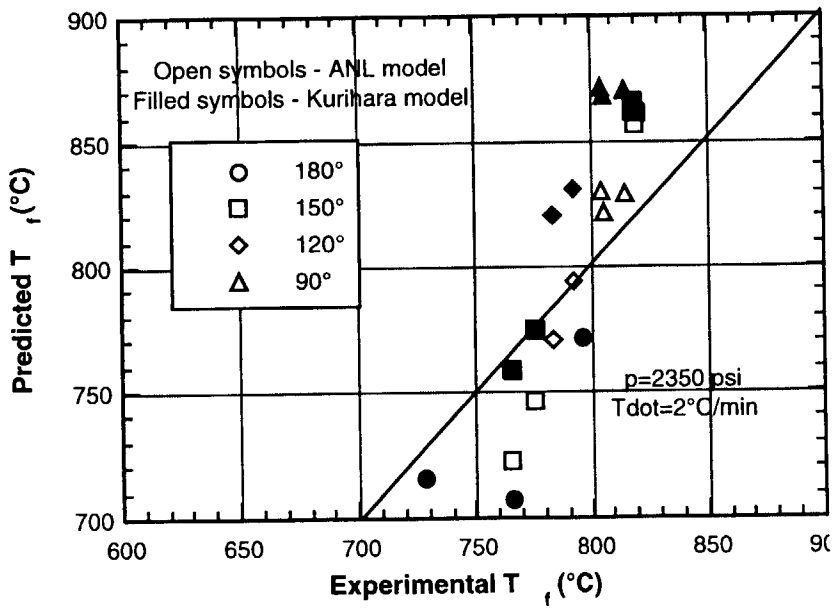


(a)



(b)

Creep rupture model (a) gives consistently better predictions than flow stress model (b) for high temperature tests.



Failure of circumferential cracks can also be expressed in terms of a m_p multiplier on the axial stress in the ligament. Model predictions agree reasonably well with experiments

Crack Opening Area at High Temperatures

Structural models determine at what temperature (and time) during a severe accident the ligament at the tip of a part-throughwall crack will rupture

To determine the leak rate after crack becomes throughwall, need to know

Whether there is any creep crack growth before the crack goes unstable?

How does the crack opening area vary with time?

Analytical predictions are based on analogy between power law plasticity models and creep behavior. Strain in plasticity model is replaced by strain rate in creep solution

$$\frac{\varepsilon}{\varepsilon_0} = \alpha\left(\frac{\sigma}{\sigma_0}\right)^{m'} \quad \text{and} \quad \dot{\epsilon}=A\sigma^n$$

Plasticity solution in EPRI NP-1931 for COD of center–cracked plate was used to predict COD assuming that the effective stress on the crack is $m\sigma_{hoop}$

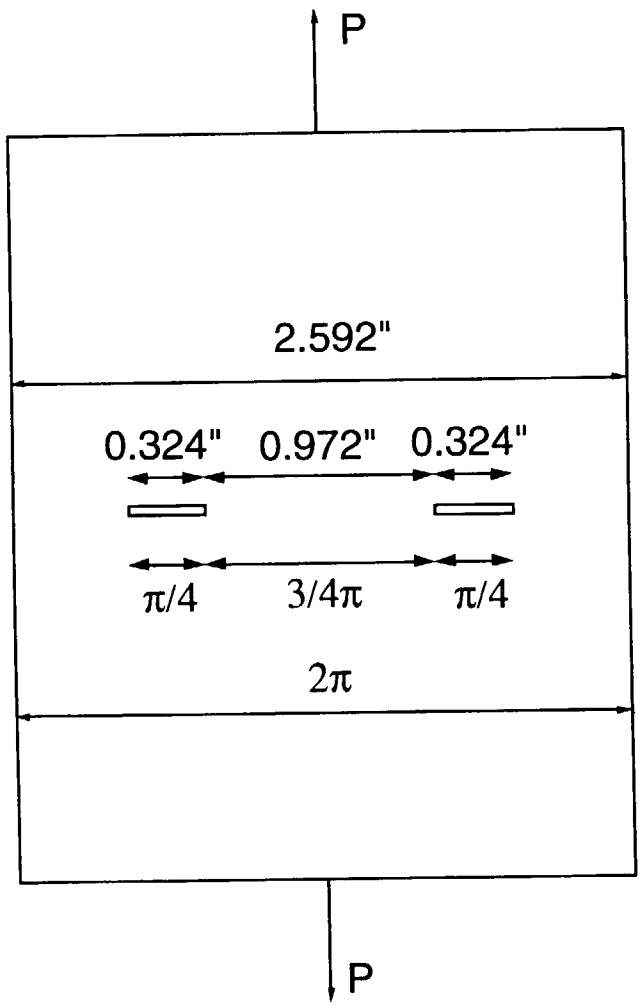
Validation Tests for Crack Opening Area

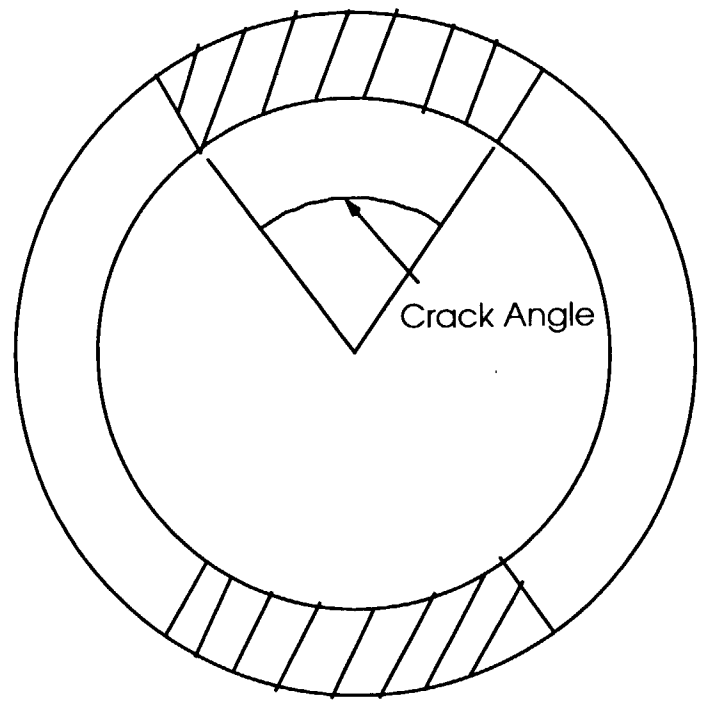
Extremely difficult to conduct creep tests on throughwall axially flawed tubes under internal pressure.

Initially considered tests on flat plates in tension, but simpler experimental solution was to conduct creep tests on axially loaded tube specimens with symmetrically located throughwall circumferential notches

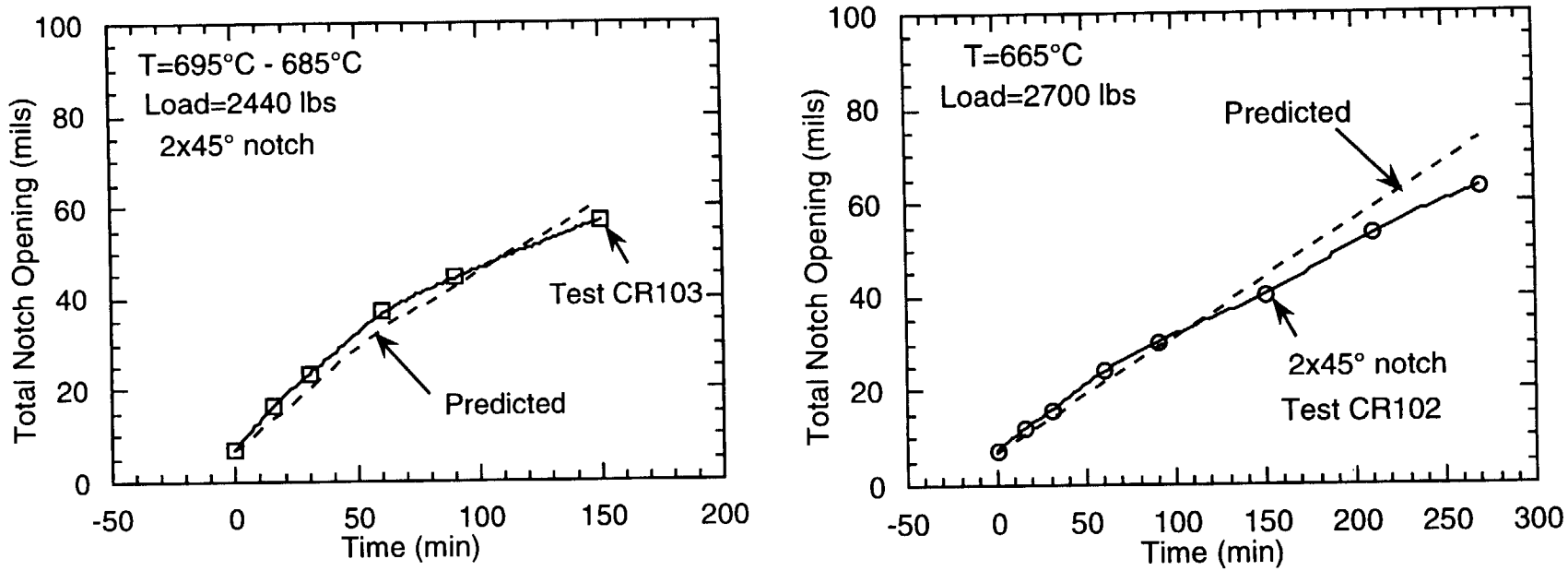
- Tube curvature effect can be ignored for small notches.
- Interaction between the two notches (small in any case) is accounted for by using solutions applicable to plates of finite width.

Geometry for Validation Tests





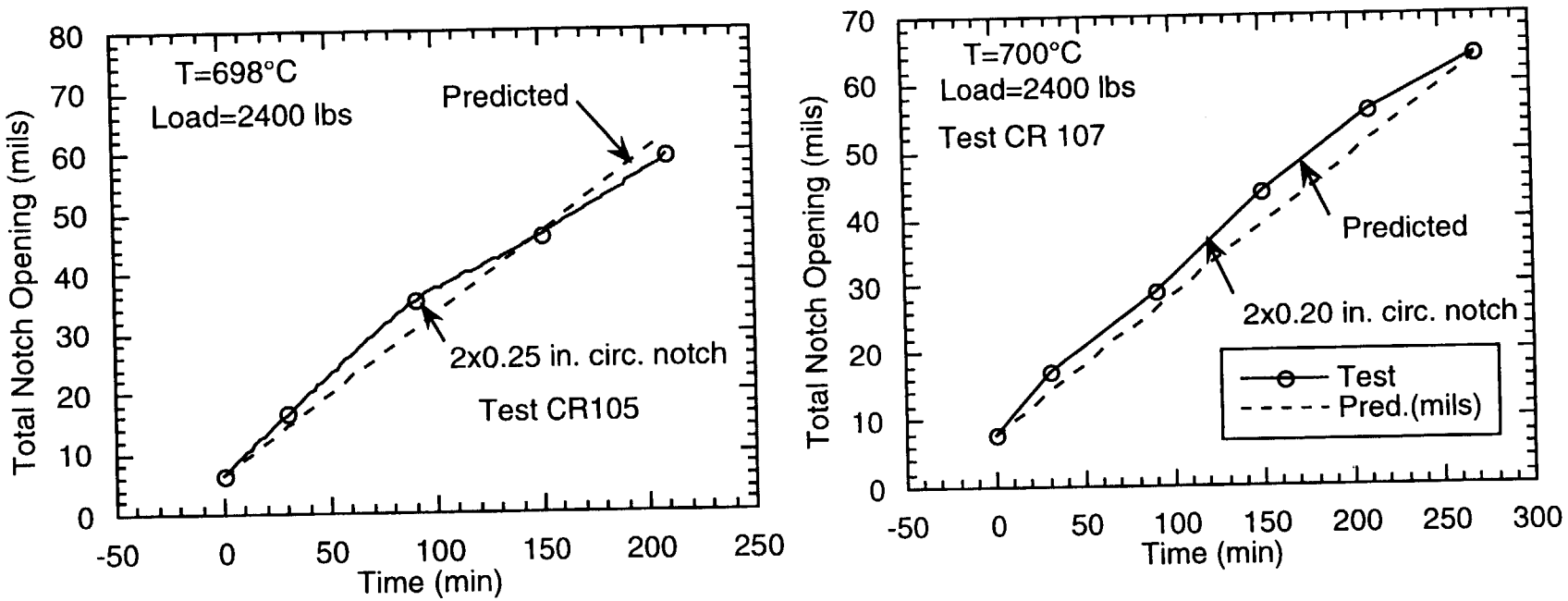
Isothermal Validation Tests



(a) (b)

Experimentally measured and predicted variation of total notch opening with time for specimens with 2x45° circumferential notches loaded at (a) 2440 lbs and (b) 2700 lbs.

Isothermal Validation Tests (Cont'd)

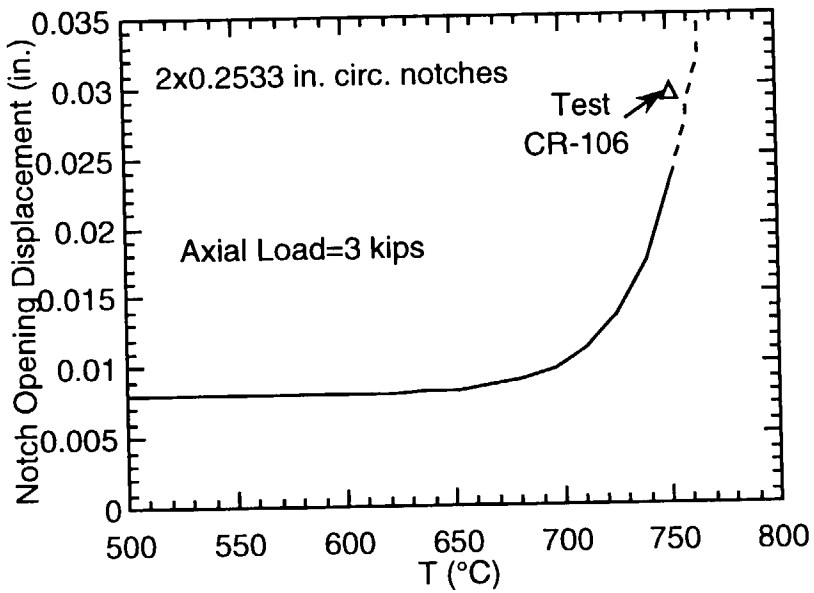


(a) (b)

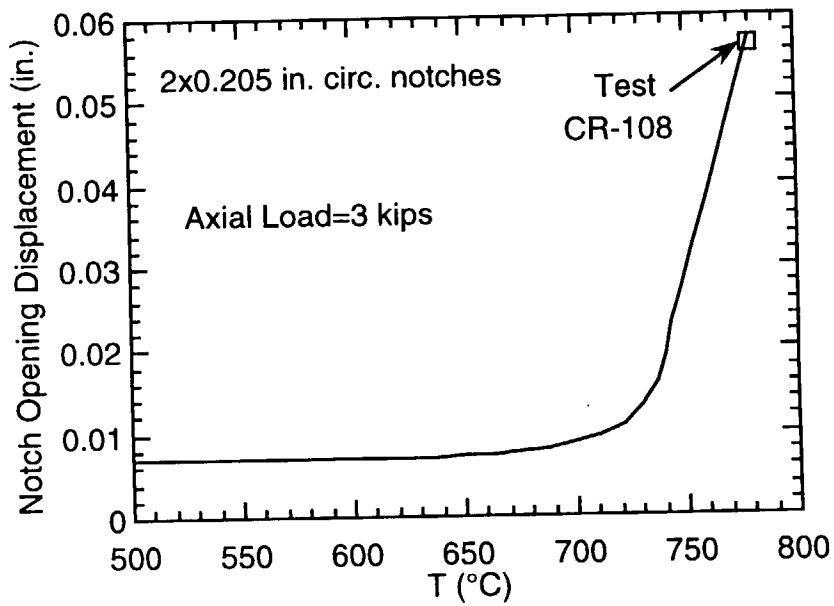
Experimentally measured and predicted variation of total notch opening with time for specimens with 2 symmetrical circumferential notches loaded at 2400 lbs for notch lengths (a) 0.25 in. and (b) 0.20 in.

Non-Isothermal Validation Tests

Used Temperature ramps simulating Case 6RU



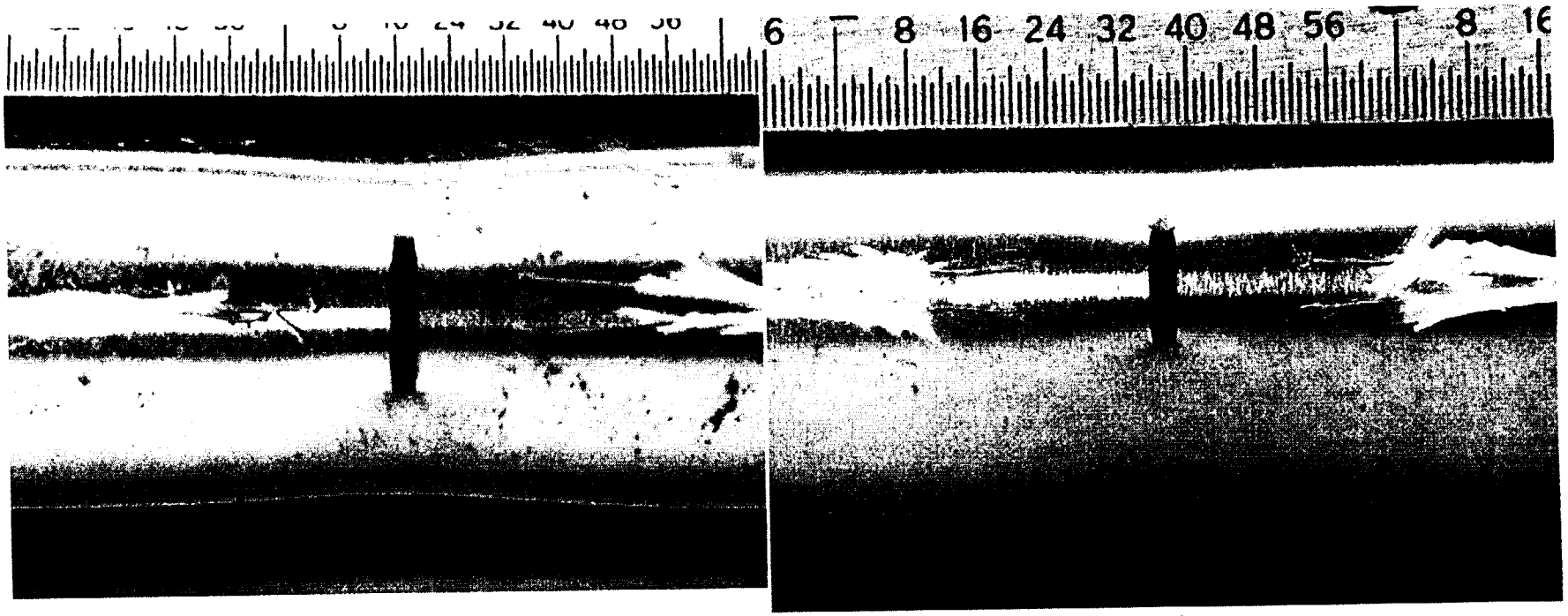
(a)



(b)

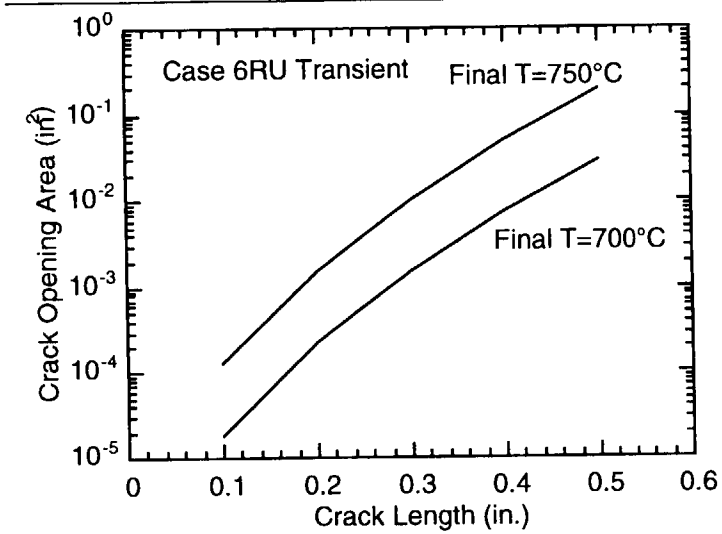
Comparison of predicted (solid line) and experimentally measured (symbols) notch opening displacements for (a) Test CR 106 and (b) Test CR 108.

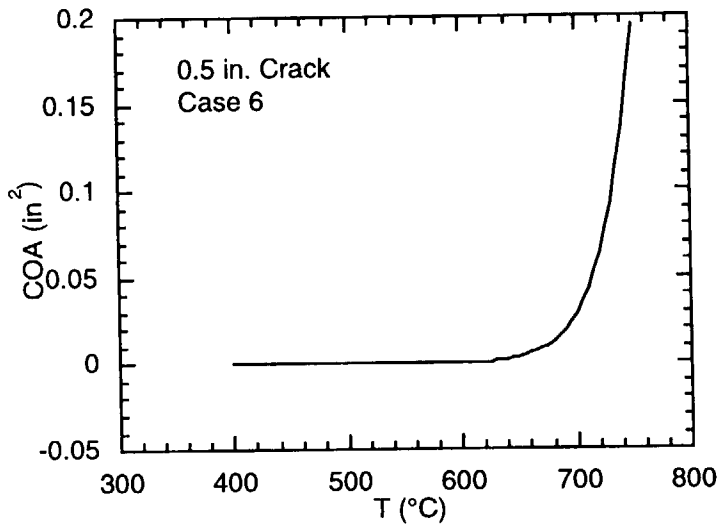
No extension of notches due to creep was observed.

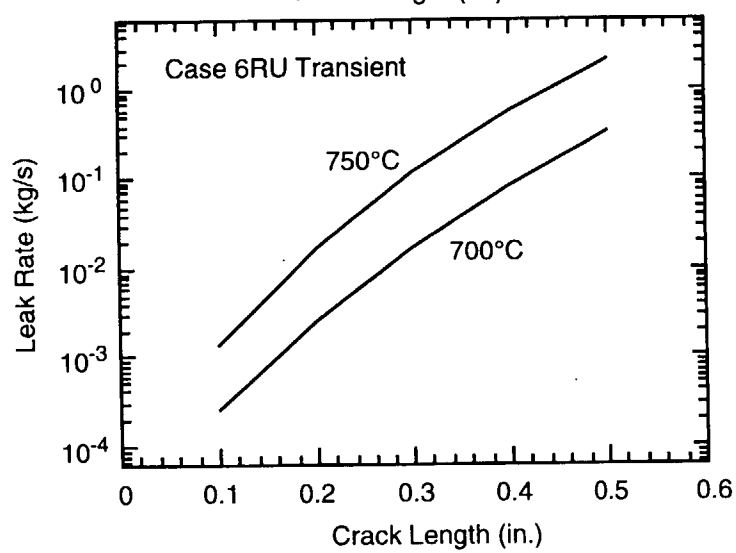


(a) (b)

Crack opening at the end of tests for (a) isothermal loading (CR 102) and (b) non-isothermal loading (CR 108)







COA begins to increase rapidly with temperature for T > 650°C

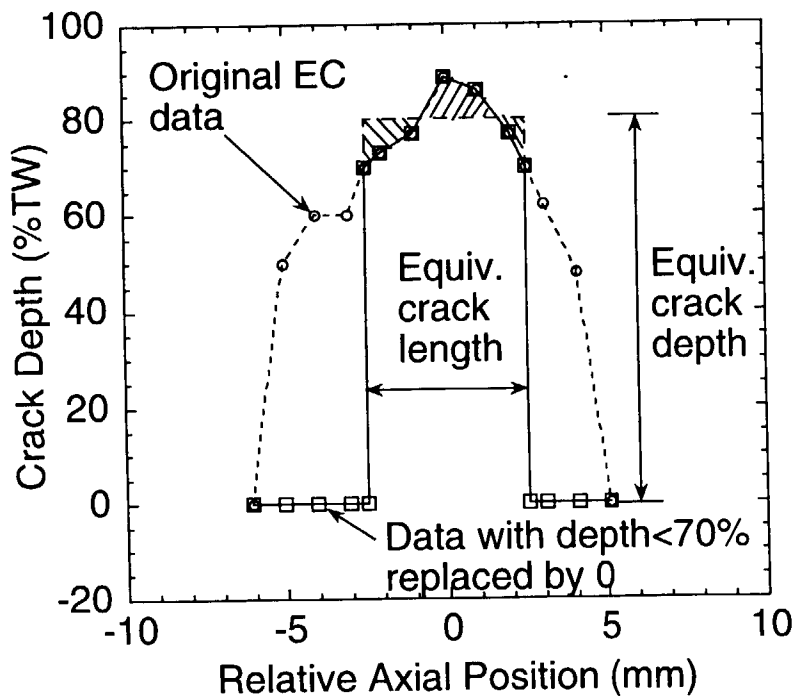
Strongly dependent on crack size. At 750°C COA for a 0.25 in. crack still <1% tube area, 0.5 in. crack COA is close to that for a rupture.

Crack Opening Area, Leak Rates During Case 6RU Transient

Application to nonuniform crack geometries

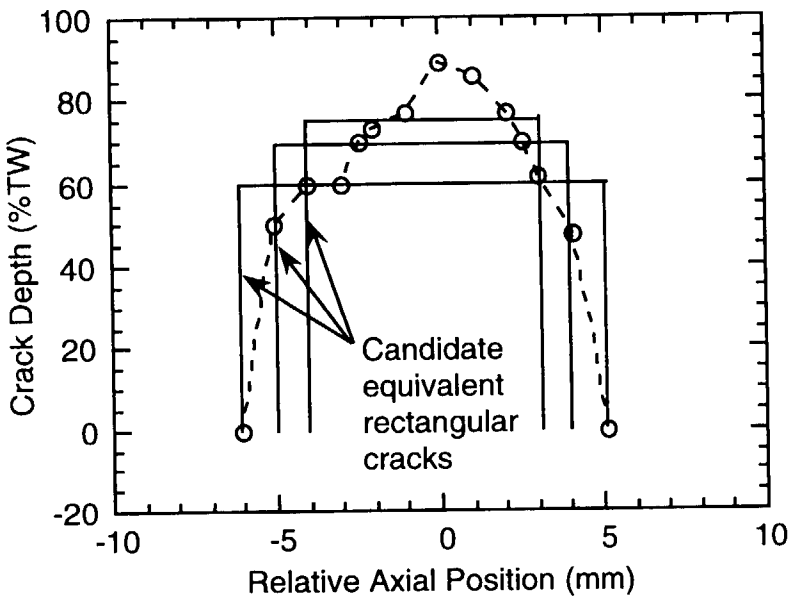
For bounding calculations replace crack with bounding rectangular crack

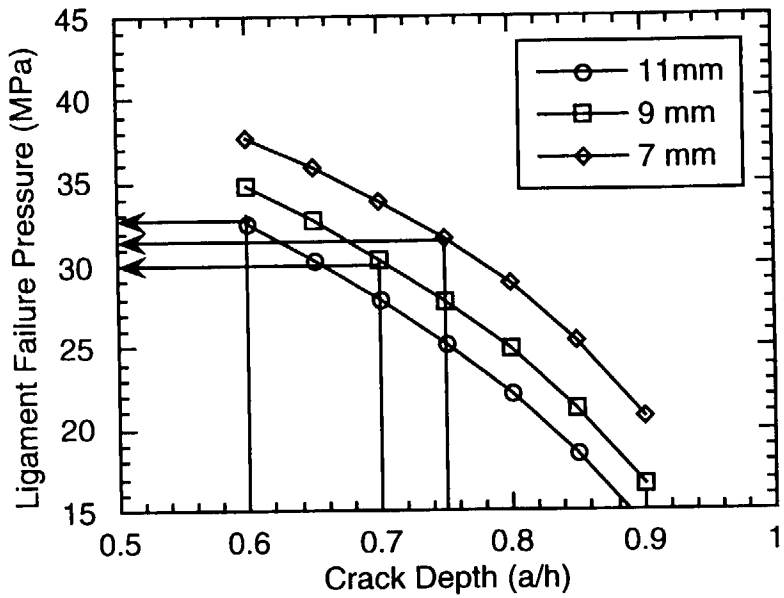
For more realistic estimates an alternative approach can be used. A candidate equivalent rectangular crack can be defined as shown:



Choose a crack depth (e.g., 70% in this example). This sets equivalent crack length. Ignore crack depths < this depth (70%).

Determine depth of candidate equivalent rectangular crack by equating hatched areas.



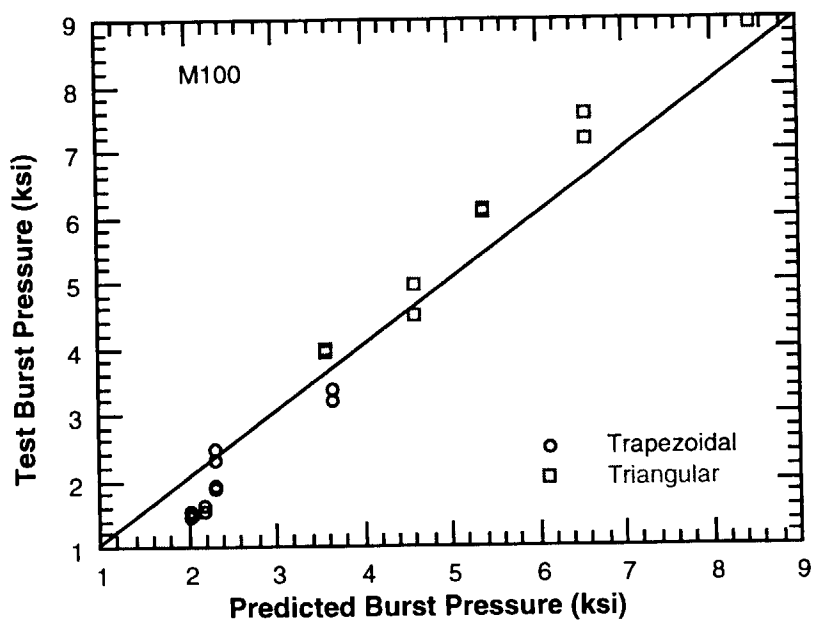


Determine a series of candidate equivalent rectangular cracks corresponding to various crack depths.

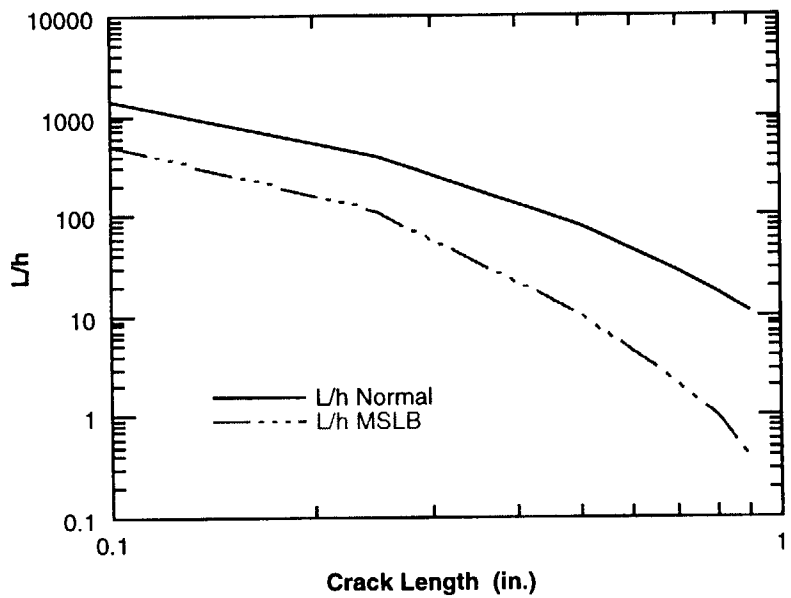
Plot the predicted ligament rupture pressure vs. crack depth and length, using a correlation for rectangular cracks.

The equivalent rectangular crack is defined by the one corresponding to the minimum ligament rupture pressure (in this case 9 mm long by 70% deep).

An analogous procedure can be used to determine the burst pressure



Reasonable agreement with preliminary experimental data on triangular and trapezoidal notches.



Stress corrosion cracks in piping typically have L/h > 500, in SG tubing for MSLB conditions L/h < 10 for crack sizes of interest

Advisory Committee on Reactor Safeguards
Meeting of the Ad Hoc Subcommittee
on Differing Professional Opinion Issues

October 13, 2000

Agenda Item 15.A

"Design Basis Accidents"

Joseph Donoghue	415-1131
Steven Long	415-1077
Gareth Parry	415-1464

What is, and What Is Not a Design Basis Accident

The design basis accidents for pressurized water reactors include:

Spontaneous rupture of a single steam generator tube with a concurrent failure of a single active component.

Main steam line rupture with primary to secondary leakage within technical specification limits and a concurrent failure of a single active component.

Licensees are responsible for maintaining the ability to mitigate these accidents without exceeding the dose guidelines in 10CFR100

The accidents that are addressed in the following slides may exceed the design basis accidents in that they include consideration of greater numbers of tube ruptures or larger amounts of leakage.

Ability to Predict Phenomena in Steam Generators During Design-Basis Accidents

- Issue: Use of thermal-hydraulic codes to predict dynamic effects in steam generators (SG) during main steam line breaks (MSLB)
- DPO Considerations Document considered elevated primary-to-secondary differential pressure during MSLB, not dynamic effects
- Safety Evaluation for Byron/Braidwood Amendment - November 1995
  - Considered SG tube support plate (TSP) deflection during MSLB
  - Licensee used RELAP5/Mod 3 to estimate blow-down loads

Ability to Predict Phenomena in Steam Generators During Design-Basis Accidents (cont.)

- Licensee also modeled Model Boiler (MB)-2 SG blow-down tests using RELAP5/Mod 3 to substantiate modeling assumptions
- Modeled hot-standby conditions using 2-phase, thermal equilibrium approach
  - Licensee found that using non-equilibrium assumption yielded instability related to interfacial heat transfer model error in code
  - Staff audit calculations using TRAC-PF1/Mod2 and comparison to MB-2 data supported use of equilibrium option
- Staff reservations with quality of MB-2 data prevented its use as basis for thorough qualification of calculational method

Ability to Predict Phenomena in Steam Generators During Design-Basis Accidents (cont.)

- Safety Evaluation conclusion:
  - Licensee used conservative approach:
    - Hot standby conditions yield highest loads
    - Peak loads applied to all TSPs
    - Applied safety factor to loads
  - Calculated peak loads considered acceptable for estimating conditions during MSLB
  - Not a generically applicable approach
- ACRS commented on staff safety evaluation during meeting in June 1996

"Justification" for Assumption of Maximum Leakage Rate Due to "Equilibrium" With Injection Flow

RES projected extremely high leakages in event of MSLB when prioritizing DPO GSI

Postulated cause was extremely large number of undetected cracks opening and leaking (but <u>not rupturing</u>) due to increased ΔP

However, thermal-hydraulic analysis of MSLB indicates that ΔP initally does not increase very much because blowdown of SG cools and depressurizes RCS, also.

ΔP does increase when SG blowdown is complete and ECCS begins to refill RCS

So, if ΔP is the reason for leakage to start, it will start from a low value at this point in the event.

Cracks are not expected to all begin leaking in unison - - more cracks should start leaking as ΔP continues to increase.

Cracks that are leaking should increase rate of leakage as ΔP increases, perhaps not smoothly on an individual basis, but not step-wise in unison.

Repressurization of RCS can only continue to point that ECCS can add water faster than leakage can remove it from RCS

"Justification" for Assumption of Maximum Leakage Rate Due to "Equilibrium" With Injection Flow
(Continued)

Leak rate increases with RCS pressure while ECCS flow rate decreases with RCS pressure, so large leakages will eventually stop repressurization at some point.

If equilibrium RCS pressure is above normal operating ΔP value, then tubes will remain at higher stress level than experienced during operation.

Laboratory analyses of individual cracks indicates that they may slowly increase leak rate or suddenly begin leaking when held for several minutes at a steady pressure that is above their normal operating pressure.

So, leakage may still increase after leak rate has reached ECCS flow rate.

However, laboratory tests indicate that flaws stabilize when ΔP is significantly decreased.

So, if total leakage increases to point that ΔP falls all the way back to the normal operational value, all flaws are expected to be stable by that point (probably much earlier).

Therefore, ECCS flow rate at back pressure equal to SG normal operating ΔP is maximum flaw leak rate that might be achieved by increasing the ΔP with secondary depressurization and ECCS injection.

Tolerable Primary-to-Secondary Leakage During Design-Basis Accidents

- Addressed in DPO Considerations Document as Issue 2: MSLB Leakage Issue
  - Discussed analyses of SG tube leaks and ruptures during secondary depressurizations in NUREG 1477 and an INEL Report
- NUREG 1477, "Voltage-Based Interim Plugging Criteria for SG Tubes"
  - Analyzed secondary system ruptures with assumed leakage from 100 gpm to 1000 gpm
  - Concluded that reactor water storage tank (RWST) inventory could be maintained based on operator actions in accordance with WOG Emergency Response Guidelines (ERGs)

Tolerable Primary-to-Secondary Leakage During Design-Basis Accidents (cont.)

- INEL Report, "SG Tube Rupture Induced from Operational Transients, Design-Bases Accidents, and Severe Accidents"
  - Analyses supported rulemaking. Provided a basis for staff to judge that more effort needed to assess severe-accident risk contribution.
  - Report summarizes analyses of MSLB with 1, 10, 15, and 20 ruptured tubes
  - Used RELAP5 model of Surry
  - Assumed that injection flow was "throttled" to maintain RWST inventory

Tolerable Primary-to-Secondary Leakage During Design-Basis Accidents (cont.)

- Concluded that adequate RWST inventory could be maintained to mitigate tube leakage associated with multiple SGTR

- Throttling Assumption:
  - Calculation used throttled injection flow in event response
  - WOG Emergency Response Guidelines have contingency for SGTR and depressurized SGs
  - Objectives of ERGs are to maintain RCS inventory and makeup water supply, and to minimize radiological release

Tolerable Primary-to-Secondary Leakage During Design-Basis Accidents (cont.)

- ERGs direct operator to reduce ECCS flow under applicable conditions (RCS subcooling, RCS pressure, Pressurizer level)
- Flow reduced by securing pumps or realignment via charging lines
- Surry charging configuration supports throttling

- Analyses concluded that RWST inventory sufficient up to 15 ruptured tubes, provided operator actions taken to depressurize RCS, reduce injection flow, and place RHR in operation.
- Low frequency assigned to multiple SGTRs suggested that additional analysis not needed to support rulemaking.

Primary-to-Secondary Leakage During Plant Transients

- Secondary depressurization events have occurred
- Usually associated with stuck-open steam system valves or loss of feedwater
- Tube leakage has not been a consideration
- Depressurization events resulting in increased leakage would prompt action by licensees
- Plants returning to power following depressurization events have not reported higher primary-to-secondary leakage.

Advisory Committee on Reactor Safeguards
Meeting of the Ad Hoc Subcommittee
on Differing Professional Opinion Issues

October 13, 2000

Agenda Item 15.B

"Severe Accidents"

Steven Long	415-1077
Gareth Parry	415-1464
Charles Tinkler	415-6770

What is Included in Severe Accident Discussion

Severe accidents are considered to be those accident sequences that result in substantial core damage.

Focus of the discussion is core damage with containment bypass due to SG tube failure

Sequences addressed:

- Sequences initiated by spontaneous rupture of a degraded tube
- Sequences initiated by secondary depressurization that induce rupture(s) of degraded tubes due to increased pressure differential
- Sequences initiated by primary system over-pressurization that induce rupture(s) of degraded tubes due to increased pressure differential
- Sequences that cause core damage by mechanisms unrelated to tube degradation, but which include conditions that would lead to containment bypass by inducing rupture of degraded tubes due to
  - increased pressure differential
  - increased tube temperatures

Severe Accidents Due to Spontaneous Steam Generator Tube Rupture

Not included in original Reactor Safety Study (WASH-1400)

First rupture event at Point Beach unit 1 in 1975 brought it into subsequent PRAs

Included in all IPEs with widely varying results:

- Initiating event frequencies all about 10^{-2}/RY
- Resulting CDF contributions vary from high 10^{-6} to low 10^{-9}/RY

Reasons for variations not fully understood by NRC

Dominant sequences for high results involve human error probabilities.

Dominant sequences for low results are usually hardware failures with non-recovery.

Human error probability estimation appears to be a major cause of differences.

These sequences are usually one of the dominant contributors to level 3 consequence results in current PRAs

Station Blackout Accidents

Actually, any core damage accident with the RCS at relatively high pressure and the SGs dry:

station blackout
loss of DC buses
small LOCAs with loss of secondary cooling

Concerns for inducing SGTR:

loss of secondary integrity causing rupture by increasing ΔP across tubes

transfer of heat from damaged core to tubes weakening tubes

Event trees for these sequences need to be expanded to account for various events that control the timing of primary and individual SG secondary depressurization and the rate of introduction of heat to the tubes and other RCS components

Station Blackout Accidents
(Continued)

Factors to incorporate include:

RCS leakage from RCP seals, SG tubes, Pzr valves partially or fully stuck open, etc.

SG leakage from stuck MSL SVs, TDAFW supply, MSIVs, MSIV bypass valves, blowdown valves and MFW check valves

Thermodynamics of RCS component heatup, including loop seal and downcomer skirt clearing, counter-current circulation flow mixing, etc.

Potential for tube failures depends on probability of occurrence of flawed tubes that cannot withstand conditions that occur during the various sequences.

ΔP may vary between generators

Temperatures may vary within generators

Accidents Initiated By Other Means That Evolve Into Steam Generator Tube Ruptures

Tube rupture(s) may also be induced by increased ΔP across tubes due to:

- Secondary depressurizations events
- Primary over-pressurization events

Accidents Initiated By Secondary Depressurization Events

Potential Initiators:

- stuck MSL SVs
- steam dump control problems
- spontaneous breaks in MSL and associated valve headers
- SG overfills resulting in MSL valve header breaks

Conditional probability of tube rupture depends on probability that susceptible flaw(s) will be in free-span of tubes in affected generator(s)

Successful mitigation requires cool down to cold shutdown unless secondary integrity can be restored.

Human errors are most important failures in dominant cutsets.

Mitigation of about 10 full ruptures appears possible, but human errors become much more probable as limit is reached.

We are not currently aware of mechanisms that appear to have a significant probability of creating such a large number of ruptures.

Accidents Initiated By Primary Over-Pressurization Events

Initiator : ATWS

ATWS events that exceed 3200 psi in primary system are usually considered to be core damage accidents, but this produces a tube ΔP of only about 2200 psi which is less than MSLB ΔP

For ATWS events reaching lower pressures, PI-SGTR is expected to have low conditional probabilities.

This makes potential complication of ruptures in multiple generators for the same event unlikely.

So, mitigation is expected to be similar to spontaneous tube rupture, following success in dealing with ATWS event.

For core damage events due to ATWS exceeding 3200 psi in the RCS, conditional probability of PI-SGTR creating containment bypass ranges from less than conditional probability of PI-SGTR during MSLB to a higher value, depending on ATWS peak pressure.

Accidents Initiated By Primary Over-Pressurization Events
(Continued)

We do not have conditional probability curves for RCS pressure during ATWS for currently installed cores.

Rupture of tubes reduces ATWS pressure peak.

Due to low ATWS frequency and tube integrity requirements, ATWS is not estimated to make a substantial contribution to increased CDF or LERF due to induced SGTR

Analysis of Steam Generator Tube Conditions During Severe Accidents

Presentation to the ACRS DPO Panel on Steam Generator Tube Integrity Issues

October 13, 2000

Charles G. Tinkler
Safety Margins and Systems Analysis Branch
Division of Systems Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research

Outline

- Introduction/Summary
- Plant Calculation
- Code Validation
- Sensitivity Studies
- Independent Review
- SG Tube Leakage
- New Work on T/H Uncertainties
- Fission Product Deposition
- JAERI Analysis
- Conclusions

Introduction

Background

- As part of the NRC assessments on SG tube integrity, consideration has been given to the performance of tubes (flawed and electrosleeved) and their likelihood of failure during a severe accident when RCS remains at high pressure in relation to the secondary system. Concern over tube performance arises due to hot gases, as a result of core heat up and oxidation, circulating through the SG tubes.

- Circulation of high temperature gases may induce tube failure due to elevated temperature of SG tubes.

- Fission product releases from the core may deposit in the SG tubes further elevating the heat up of the SG tubes.

- Redistribution of heat from the core to the hot legs, surge line, and SGs both delays core heat up and melting and increases the probability of RCS failure (thermally induced) and depressurization. Lessens the probability of RPV failure at high pressure but increases the probability of SGTR given a core melt.

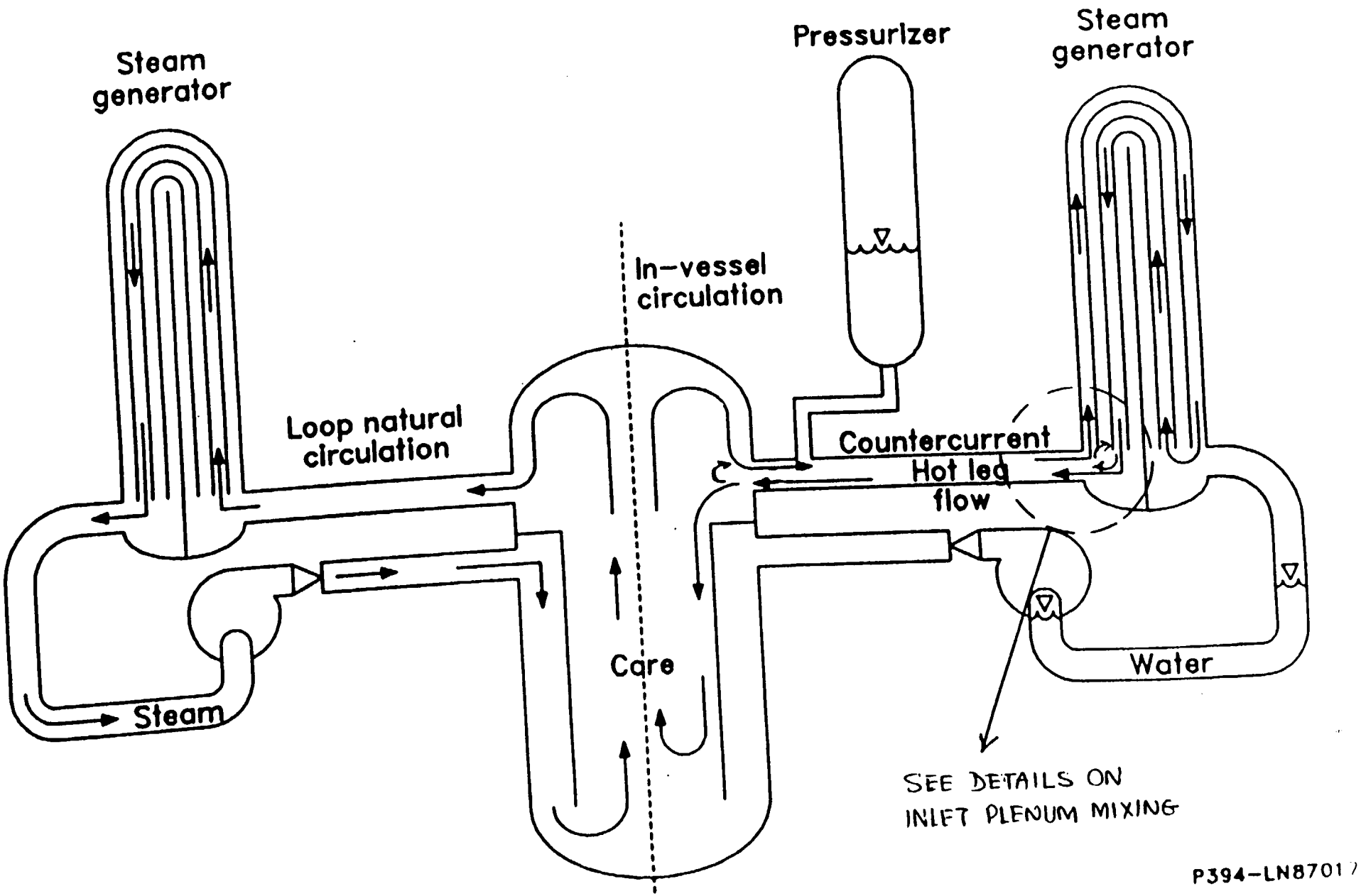


Figure 1. Severe accident natural circulation flows.

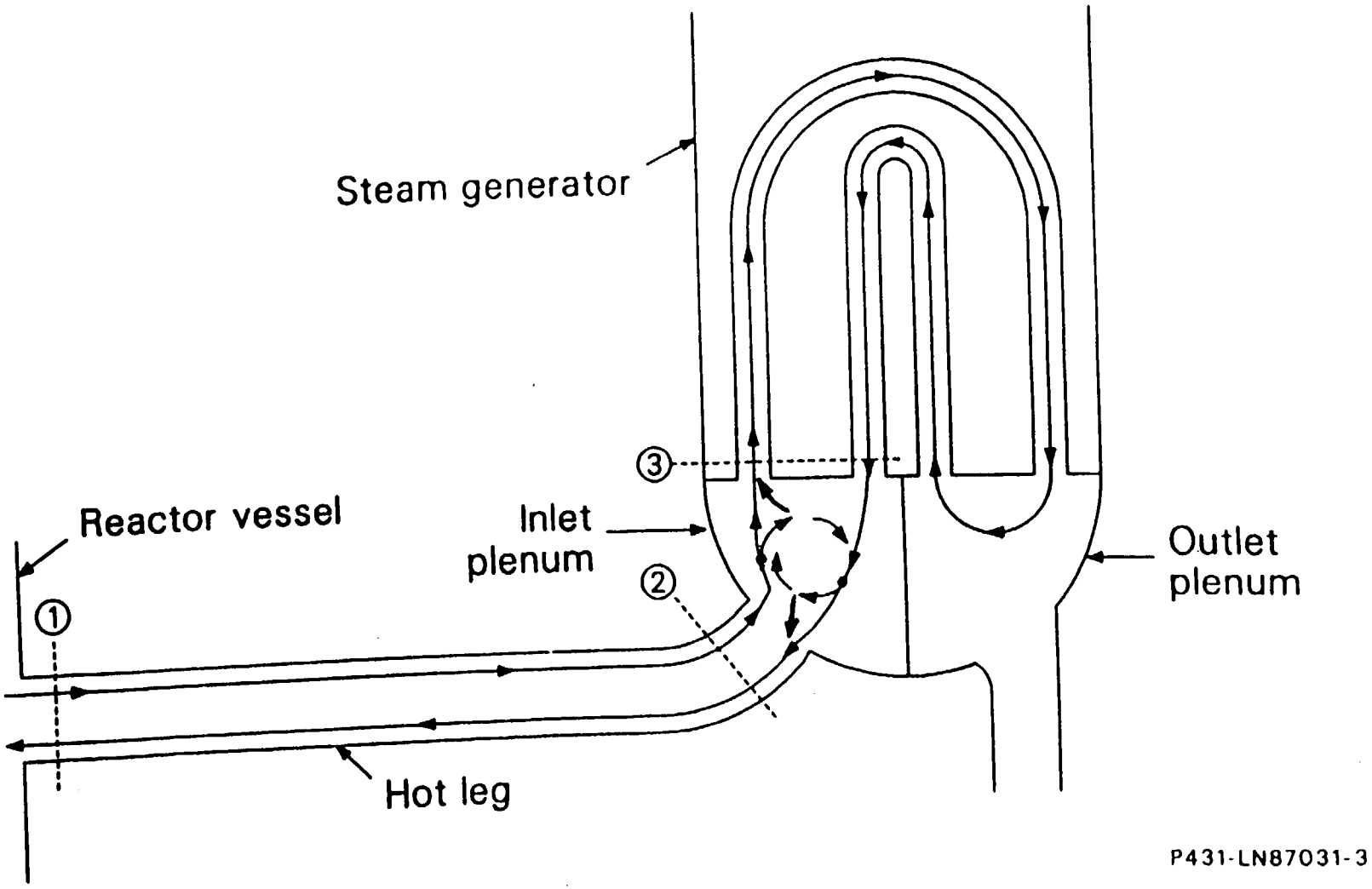


Figure 2. Hot leg natural circulation stream flows.

Summary

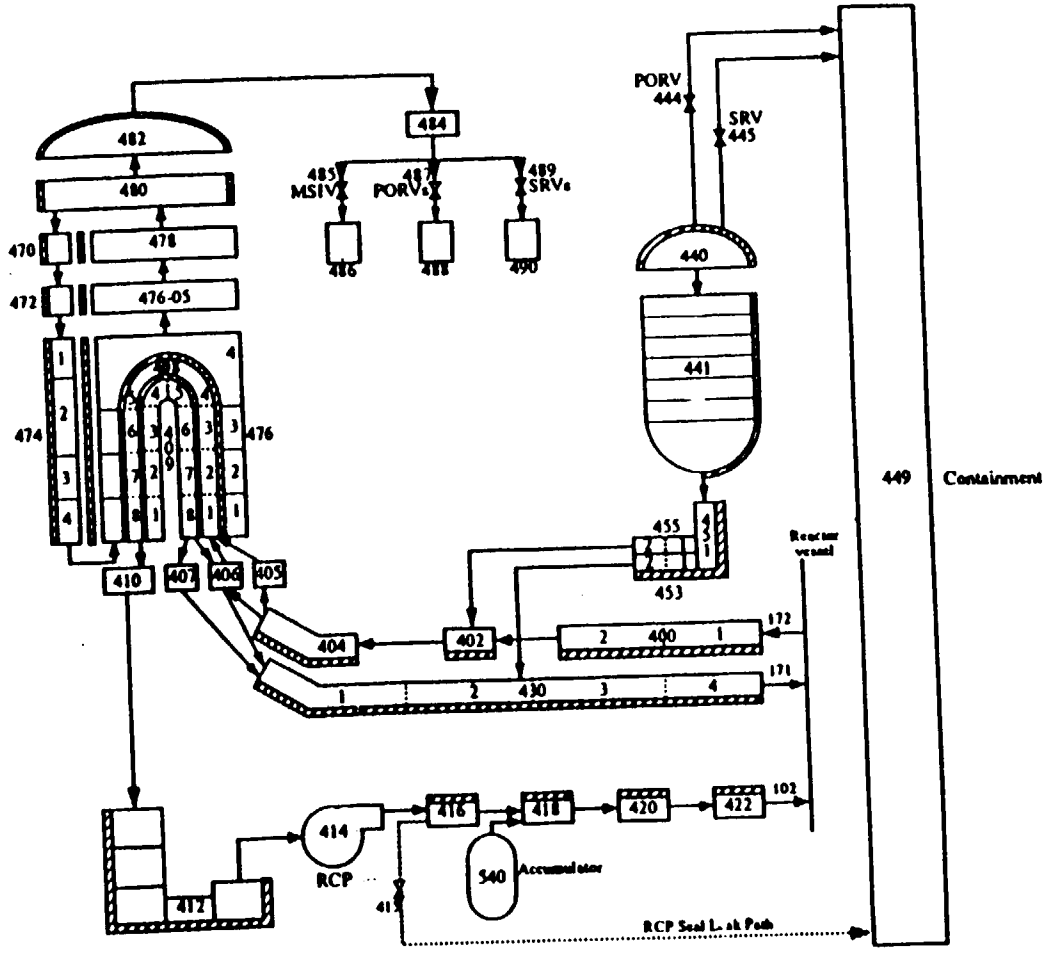
- Analysis using SCDAP/RELAP5 have been performed for representative plants for potentially risk-significant scenarios (high pressure TMLB' sequences with depressurized secondary side) to estimate effects of high temp fluid circulation.
  - SR5 analyses predict failure of hot leg or surge line before unflawed SG tubes.
  - Sensitivities on T-H Modeling did not alter conclusion on tube integrity but margins are relatively small.

Example Calculation

Application of SR5

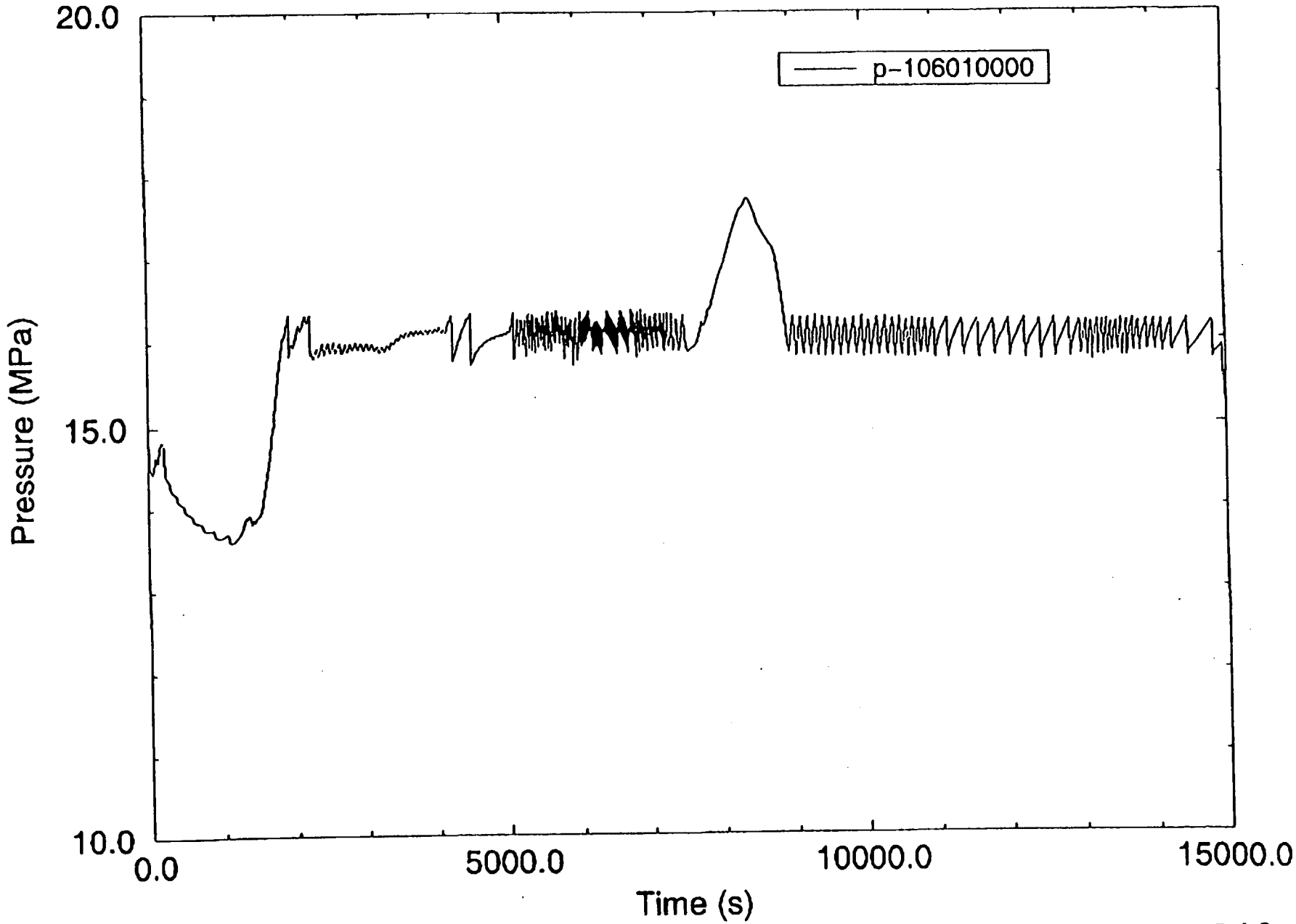
- Surry Plant calculation
  - TMLB' transient with SG secondary side depressurization.
  - Base case: #SG tubes participating in forward flow → 53%.
    mixing fraction = 0.87
    recirculation ratio = 1.9

Pressurizer Loop Nodalization



Surry Case 6

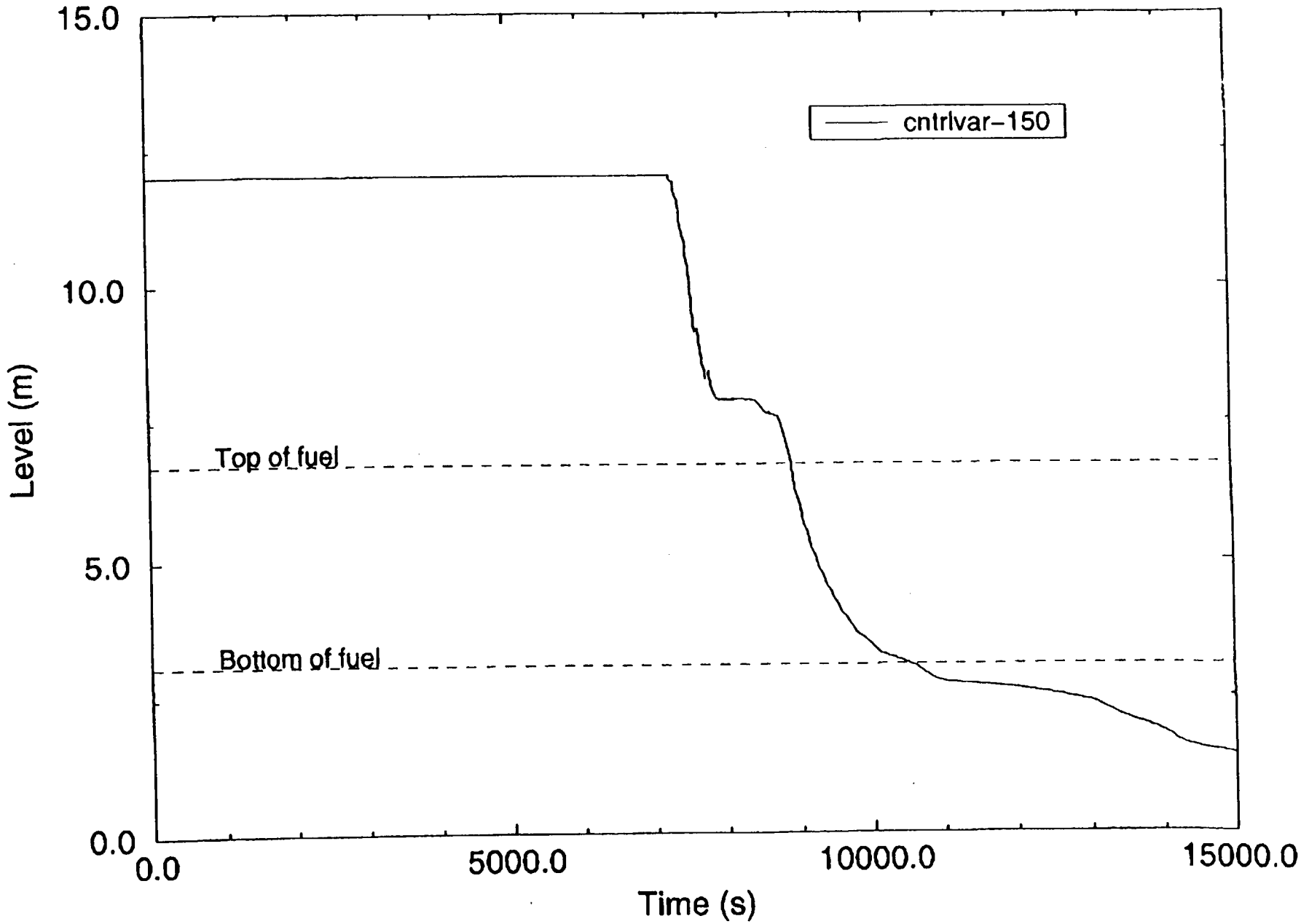
RCS Pressure



15.B.4-9

Surry Case 6

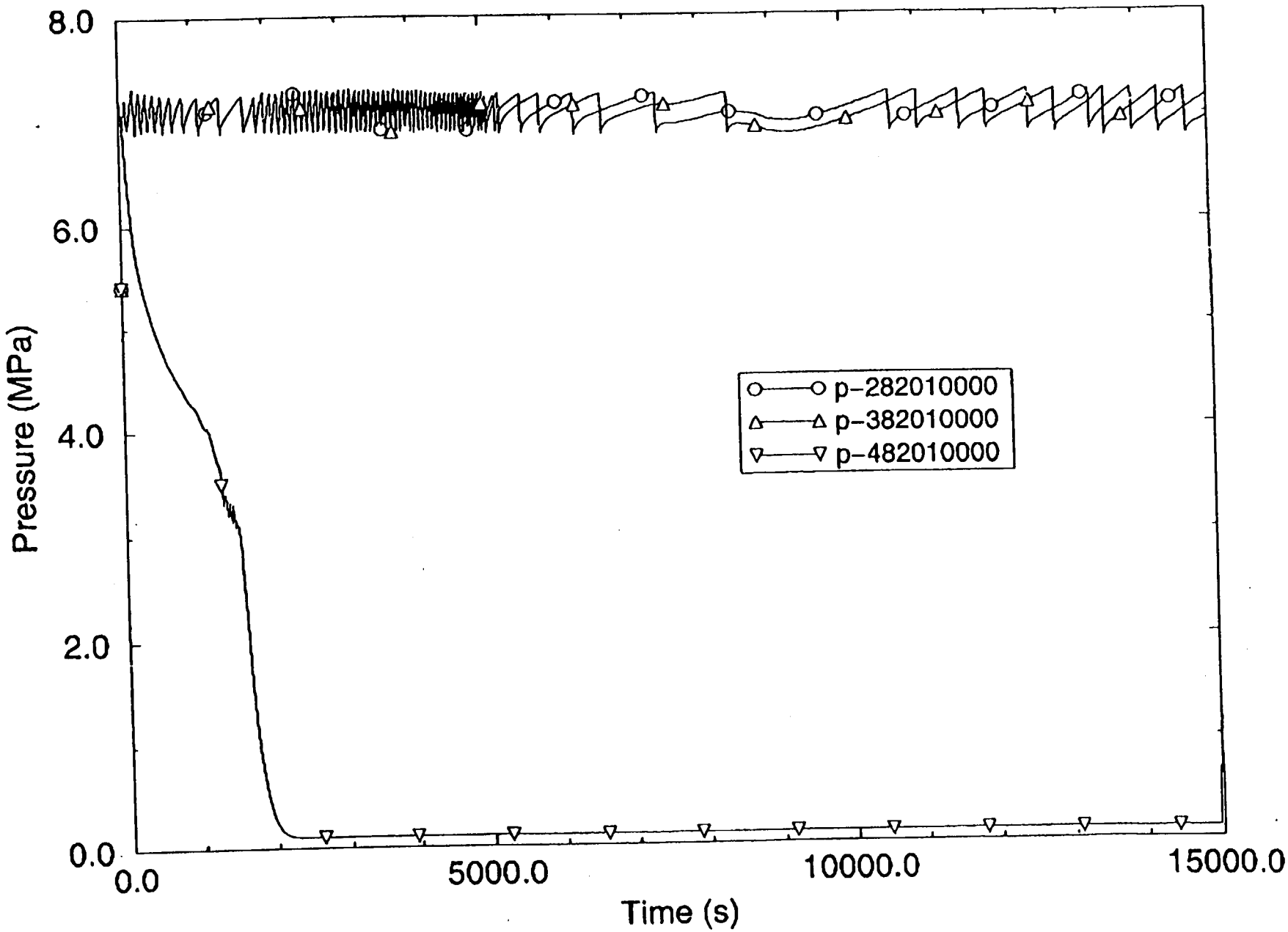
Vessel Collapsed Liquid Level



15.B.4-10

Surry Case 6

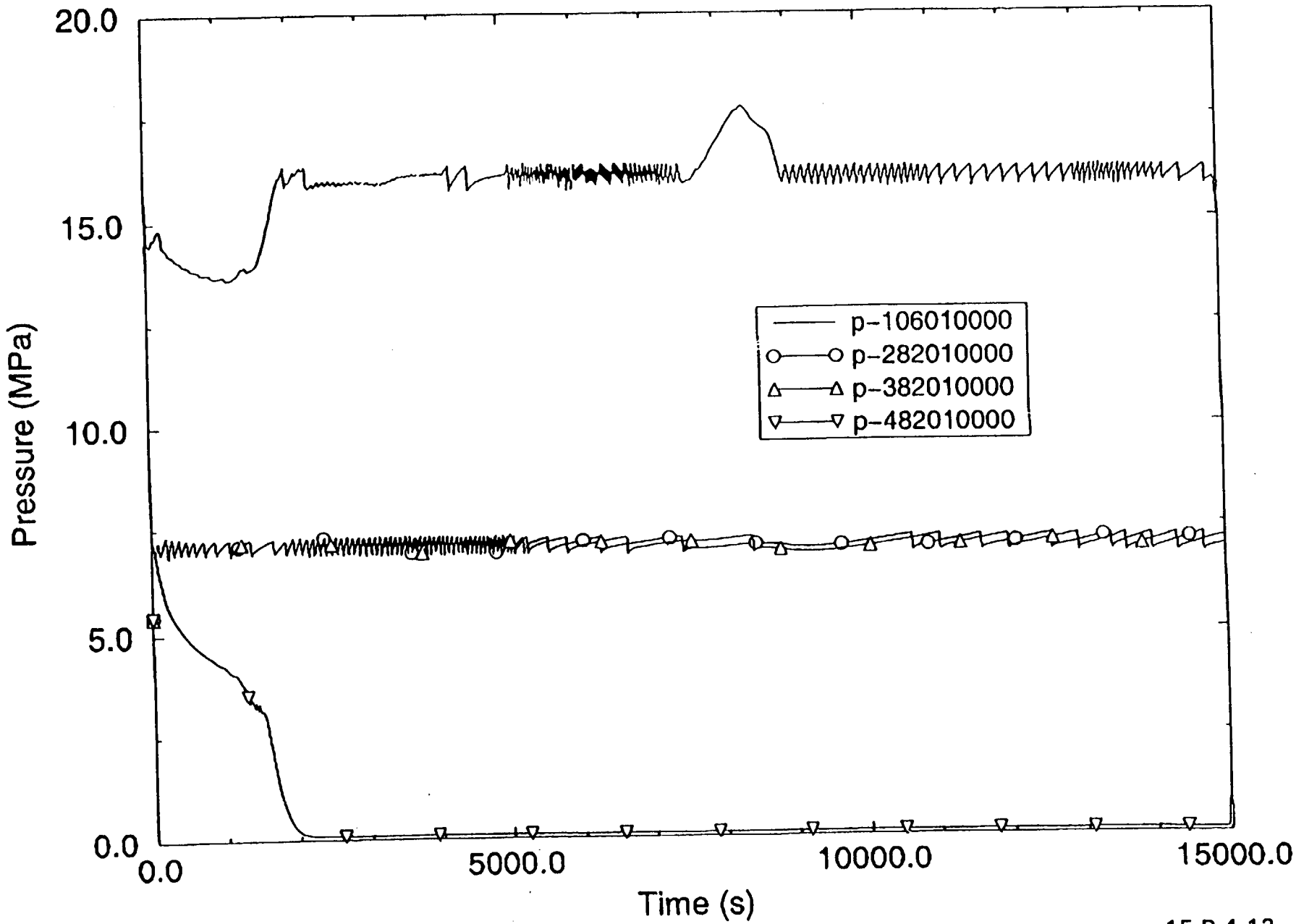
SG Secondary Pressures



15.B.4-11

Surry Case 6

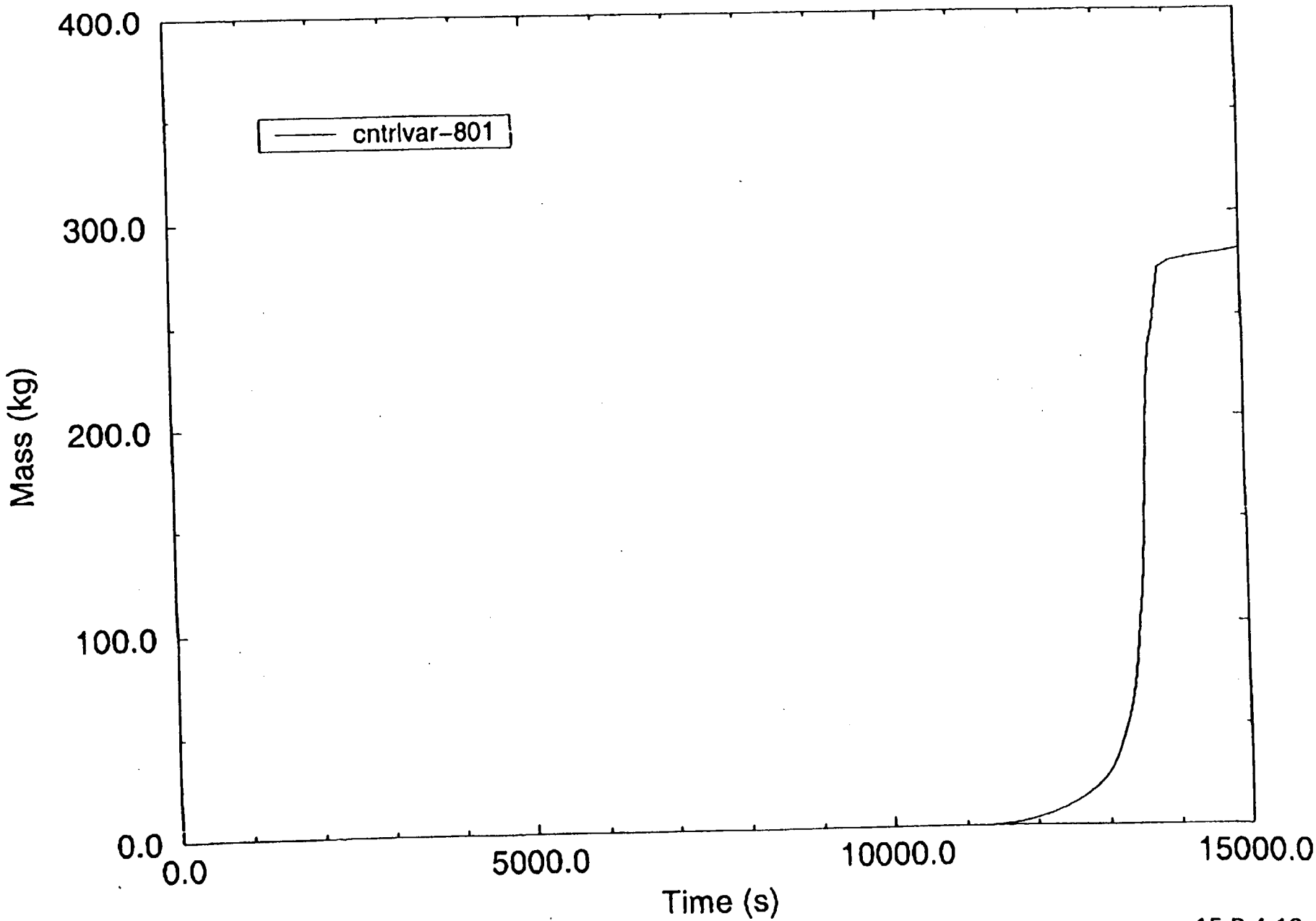
RCS and SG Secondary Pressures



15.B.4-12

Surry Case 6

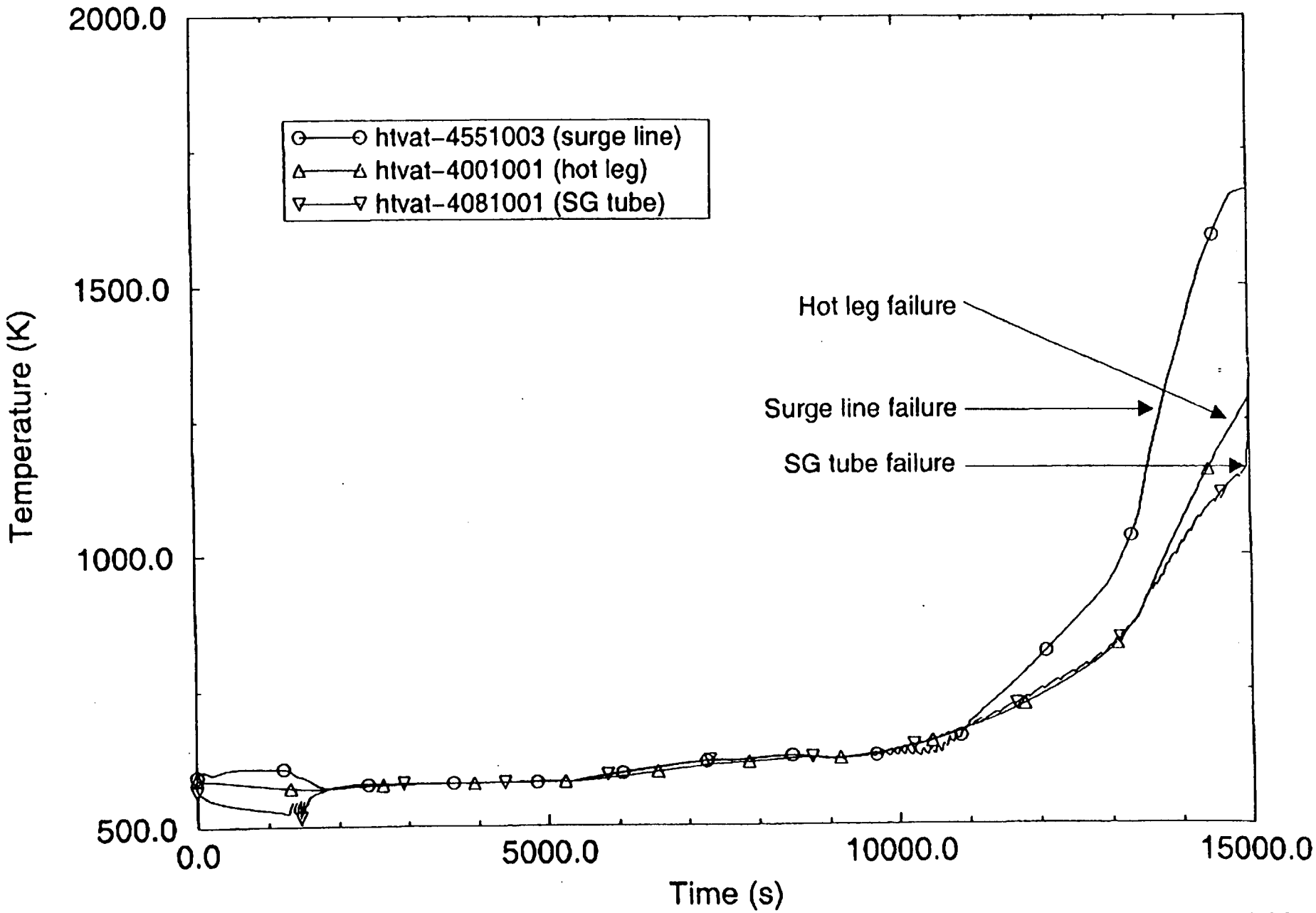
Integrated Hydrogen Generated



15.B.4-13

Surry Case 6

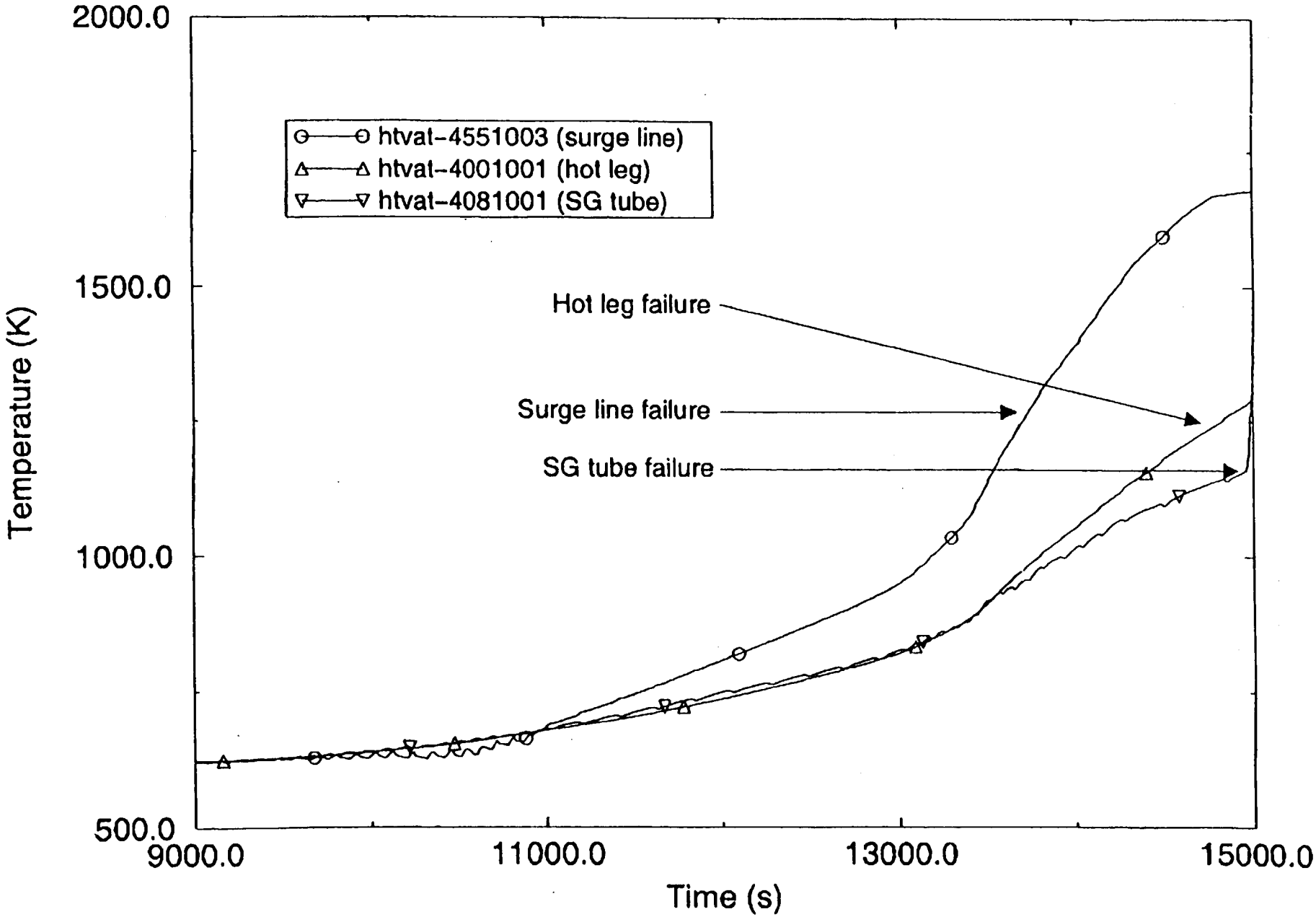
Pressurizer Loop Structure Temperatures



15.B.4-14

Surry Case 6

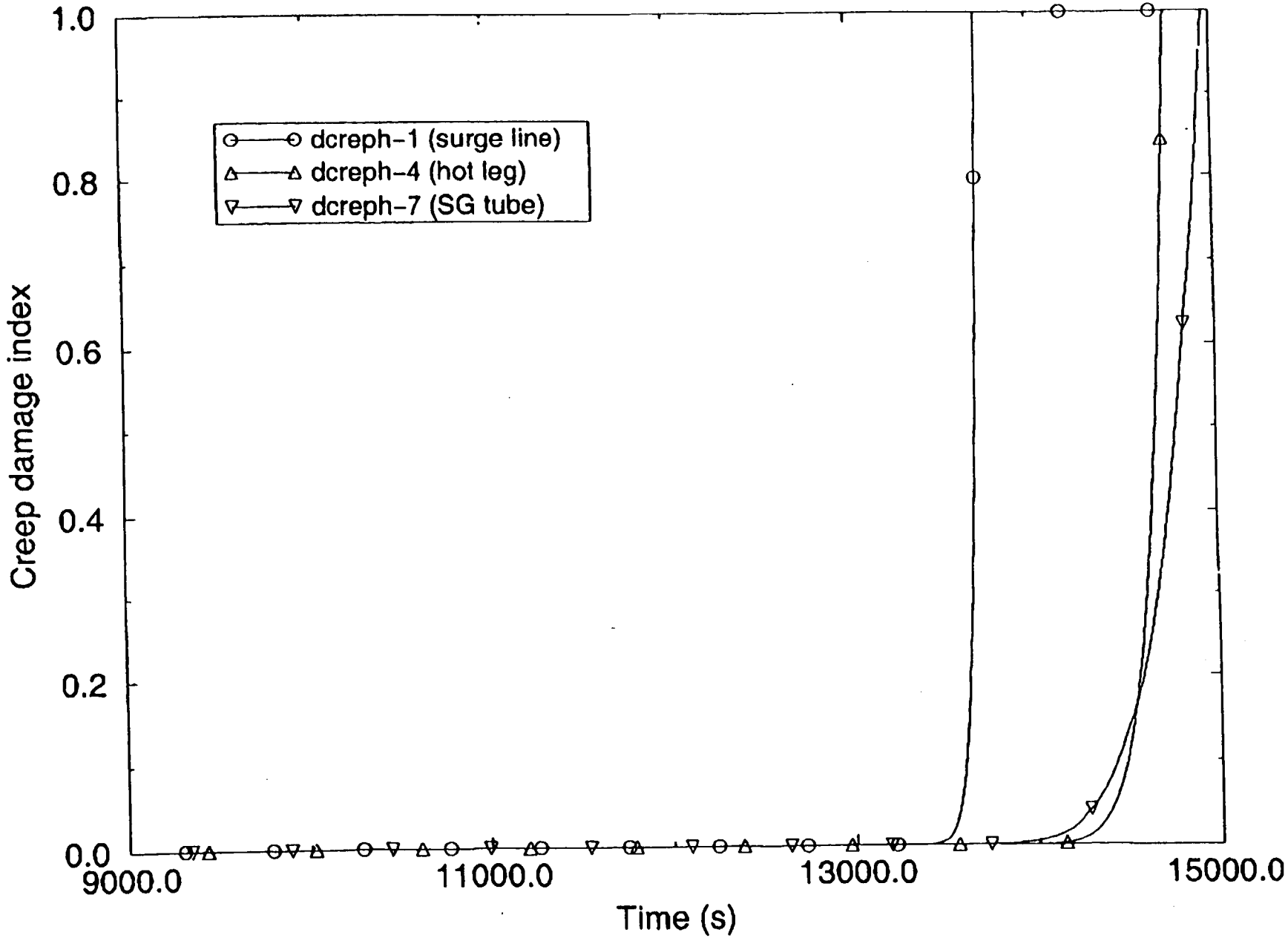
Pressurizer Loop Structure Temperatures



15.B.4-15

Surry Case 6

Pressurizer Loop Creep Damage Indices



15.B.4-16

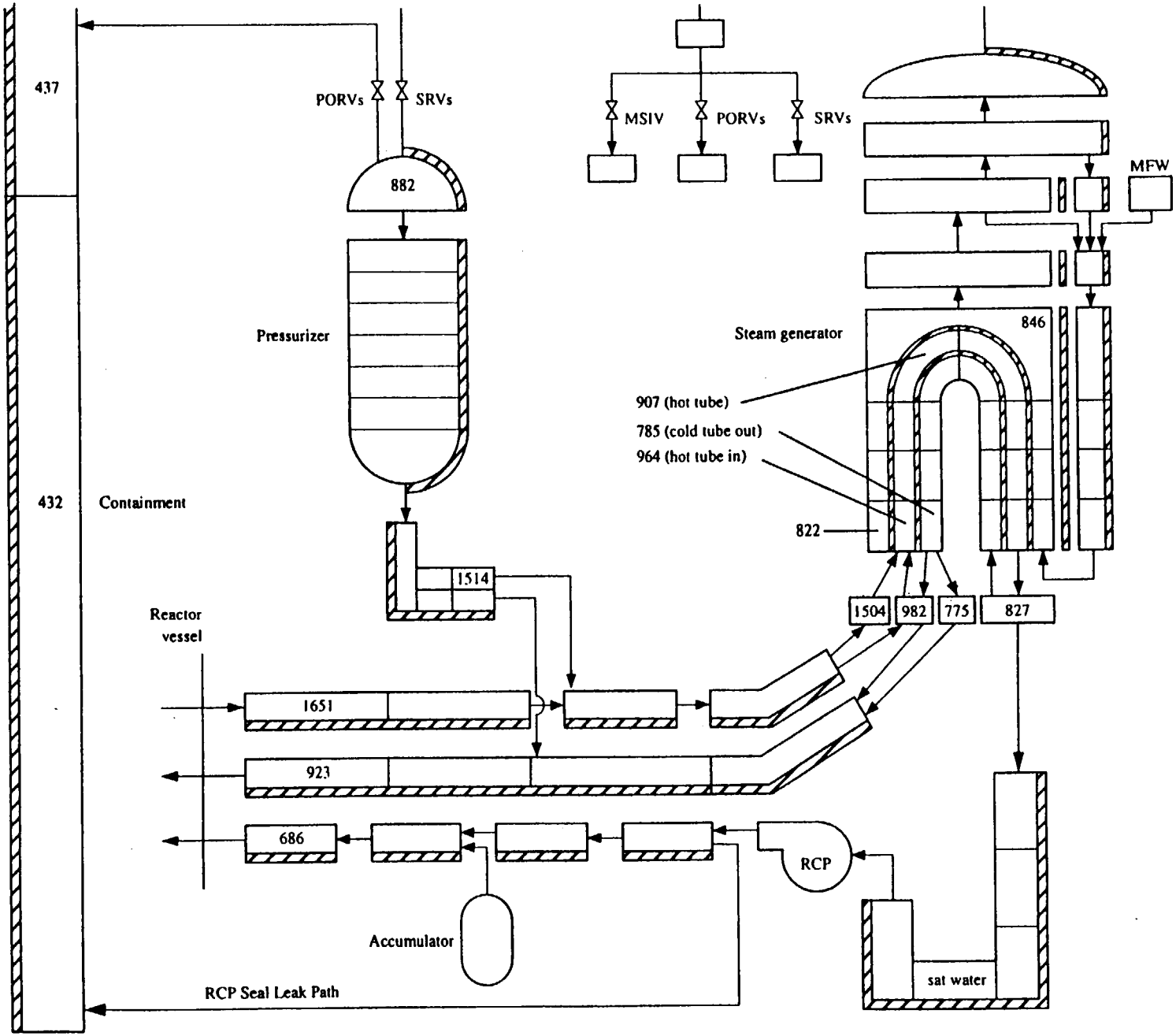


Figure 1. Surry Case 6 vapor temperatures (K) near the time of surge line failure (13,730 s).

Example Calculation (continued)

Application of SR5 (continued)

- Surry Plant calculations

  Conclusions:

  - pressurizer surge line creep rupture was the first RCS pressure boundary failure

    - Surge line rupture occurred at 13,730 sec
    - SG tube temp at this time – 957 °K

  - pressurizer surge line failed in early phase of core damage (before the onset of fuel melting

  - if pressurizer surge line and hot leg failures are ignored AND a SG secondary ADV fails open, SG tube rupture occurs ~ 20 minutes after the first RCS pressure boundary failure.

Conclusions

- SG tube T/H boundary conditions are most directly influenced by variations in the accident sequence which determine pressurization/depressurization of the primary and secondary system

  - Failure of primary and secondary relief valves
  - Primary side PORV operation (Accident Management)
  - Pump seal leakage

- SG tube T/H boundary conditions are also influenced by phenomenological issues associated with counter current natural circulation and RCS T/H.

  - SG inlet plenum mixing
  - heat transfer modeling
  - loop seal clearing

- Variations in the treatment of these issues within reasonable ranges did not significantly worsen SG tube boundary conditions.

Code Validation

- SCDAP/RELAP5 code was used to assess fluid heating of SG tubes.
  - Benchmarked against W 1/7-scale test data
  - Considerable experience developed in similar applications (DCH, unintentional depressurization) over nearly 10 years
  - Peer reviewed for this specific application (counter current natural circulation and SG tube analysis.

Code Validation (continued)

- Integral experiments were conducted by W at 1/7 scale under an EPRI/NRC cooperative program to investigate severe accident natural circulation in PWRs with U-tube SGs.

  - Several series of tests conducted, using water, low pressure SF_6.

- Low pressure tests showed (by using dye in the fluid), that a stable countercurrent flow was present in the hot legs. Flow patterns were consistent over a wide range of conditions.

- High pressure SF_6 tests provided data for validation of codes. Five series of experiments with high pressure SF_6 were conducted. Temperature measurements in the SG inlet plenum and tube inlets indicated that the fluid in the inlet plenum was well mixed.

Benchmarking of SCDAP/RELAP5 Model and Scaling

Further assessment of SR5

- Developed SR5 model for W 1/7 scale experiment. Same modeling approach used in Surry calculations was used to model the facility.

- Assessed against two high pressure SF_6 tests (S-6 and S-7).

- SR5 calculations were in reasonable agreement with the measured data (e.g., flow patterns were the same as in the experiment, temperature profiles were the same as in the experiment).

  - Calculated hot leg and SG mass flow rates within 11% of measure valves.

  - Calculated vapor temps in the SG tubes were within 5% of the measured values.

Preliminary Results

Application of SR5

- Surry: Sensitivity calculations

  - Single loop model extracted from Surry plant calculation for TMLB' sequence.

  - Calculations started at the onset of countercurrent flow (t = 9200s) and extend for an interval of 5000s.

  - Parameters varied –
    number of SG tubes participating in forward (hot) flow: 29% to 61%
    mixing fraction: 0.76 to 0.89
    recirculation ratio: 1.69 to 2.25

  - For each sensitivity study, one parameter was varied at a time.

  - Conclusions: minimal impact ($< \pm 20$K) on the peak average tube temperature for each of the individual variations.

Independent Peer Review

- Obtain independent assessment of the adequacy of SCDAP/RELAP5 modeling of natural circulation under severe accident conditions for the purpose of calculating the relative timing and failure of RCS components in order to evaluate the risk associated with thermally-induced SG tube ruptures.

- Reviewers: Raymond Viskanta (Purdue University)
  Mamoru Ishii (Purdue University)
  Peter Griffith (Massachusetts Institute of Technology)

- Review on August 19–20, 1996.

SCDAP/RELAP5 Analysis

- Additional SR5 calculations were performed to address recommendations made by ACRS and/or peer-reviewers:
  - Variations in heat transfer coefficients in the surge line, hot leg, reactor vessel upper plenum, and SGs.
  - Account for radiation heat exchange between the hot and cold streams of steams in the hot leg, and the circumferential wall heat conduction in the hot leg.
  - Synergistic effects (by changing two or more key parameters, for example the number of SG tubes carrying hot flow, mixing fraction, and recirculation flow simultaneously).

SCDAP/RELAP5 Analysis

Case	First tube failure minus first failure time (s)	Pressurizer-loop SG tube structure temperature @ surge line failure time (K)
6B ("h" x 0.8)	1180 (~ 20 mins)	964
6D ("h" x 1.3 in SG tube entrance)	1220 (~ 20 mins)	957
6 (nominal "h")	1230 (~ 21 mins)	957
6C ("h" x 1.3) in all entrances	1250 (~ 21 mins)	944
6Z ("h" x 1.2)	1310 (~ 22 mins)	938
6E fluid to fluid h.t. and circumferential wall conduction in hot leg)	1510 (~ 25 mins)	937

SCDAP/RELAP5 Analysis (continued)

- To explore synergistic effects (change the number of SG tubes carrying hot flow, mixing fraction, and recirculation flow simultaneously assuming that there is no relationship among these factors), use 5% confidence level from W 1/7 scale high-pressure transient tests.

		Case 6	Case 6F
hot/cold SG tube split	→	53%/47%	43%/57%
mixing fraction	→	0.87	0.73
recirculation ratio	→	1.9	1.78

Case	First tube failure minus first failure time (s)	Pressurizer-loop SG tube structure temperature @ surge line failure time (K)
6 (53%/47% hot/cold SG tube split)	1230 (~ 21 mins)	957
6F (5% confidence level for W high-pressure transient tests)	800s (~ 13 mins)	1007

Effect of Leakage on SG Inlet Plenum Mixing

Concern: SG tube leakage during severe accident could alter mixing in inlet plenum.

- 1/7 Scale tests did not simulate tube leakage.

Staff's evaluation of issue.

- Tube leakage effect on inlet plenum mixing likely to be dispersed (allowable leakage is an aggregate)
- Leak area equivalent to 1 gpm would produce, during severe accident, flow rate which is a very small fraction ($<<1\%$) of tube bundle flow.
- CFD code analysis should be able to provide additional insights.

New Research in the Area of Severe Accident-Induced Steam Generator Tube Rupture

- NRR user need letter, February 8, 2000, requested RES to develop a confirmatory research program to address SG tube integrity.

- The T/H part of this program (already started) includes additional work in the following areas:

  - Accident sequence variations
  - Plant design differences
  - Inlet plenum mixing
  - Tube to tube variations
  - Core melt progression

Inlet Plenum Mixing

- The SCDAP/RELAP5 code will continue to be used as the principal tool for analysis of the tube T/H boundary conditions.

- Phenomenological uncertainty in the natural circulation calculation has centered on the issue of mixing in the SG inlet plenum.

- Additional uncertainty relates to heat transfer assumptions.

- Earlier work considered single and multiple simultaneous variations in inlet plenum mixing characteristics as well as variations in heat transfer coefficients.

Inlet Plenum Mixing (continued)

- New analysis performed under this plan will involve a more rigorous treatment of uncertainties.

- Distributions will be developed for the individual mixing parameters and heat transfer coefficients and sampled using Monte-Carlo techniques. SCDAP/RELAP5 analysis will then be performed for sampled points to develop a probabilistically weighted picture of SG tube temperatures.

- The current plan is to use the following parameters for the analysis: mixing fraction, recirculation ratio, number of tubes carrying flow forward, SG tube heat transfer coefficients, hot leg and surge line heat transfer coefficients.

Effects of Leakage and Tube to Tube Variations

- SCDAP/RELAP5 analysis limited in ability to resolve variation in T/H effects among tubes.

- To estimate tube-to-tube variations, re-examine the experimental basis for the modeling (i.e., the 1/7th scale test data) to determine the appropriate variability for plant conditions.

- Use computational fluid dynamics (CFD) codes to predict inlet plenum mixing and tube to tube variations, including the effects of leakage.

- CFD code will need benchmarking against experimental data, but fundamentally, CFD codes have greater inherent capabilities for solving this type of fluid flow problem.

Current Status of Work

- SCDAP/RELAP5 work will be done at INEEL. The contract is in place and the work was stared in July 2000.

- The work to date has included establishing a new baseline calculation using the newest version of SCDAP/RELAP5, MOD 3.3, and beginning the development of parameter distributions for the inlet plenum mixing task.

- Work on this part of the user need is now planned to be complete in March 2002, with intermediate products available as work on individual tasks are completed.

- The CFD work will begin with a validation of the code to the experimental data and will begin in the first quarter of FY 2001.

Fission Product Deposition

- Objective of the analysis

  To determine the effect of fission product transport and deposition in the RCS on SG tube integrity.

- Summary of approach

  Used the VICTORIA fission product code with SCDAP/RELAP5 calculated T/H conditions as input.

- Conclusion

  Fission product transport and deposition in the RCS have a negligible effect on SG tube integrity, because the fission product release is relatively small and late in the transient.

  - Volatile fission product release represents 5–10% of decay heat.

  - Fission products spread among upper plenum, hot leg, SG plena and tubes.

Fission Product Deposition (continued)

VICTORIA

- Used VICTORIA to analyze fission product behavior under station blackout conditions

- Major VICTORIA models:

  - fuel release
  - chemistry and condensation/revaporization
  - aerosol deposition (laminar deposition, turbulent deposition, settling, thermophoresis, pipe blend)

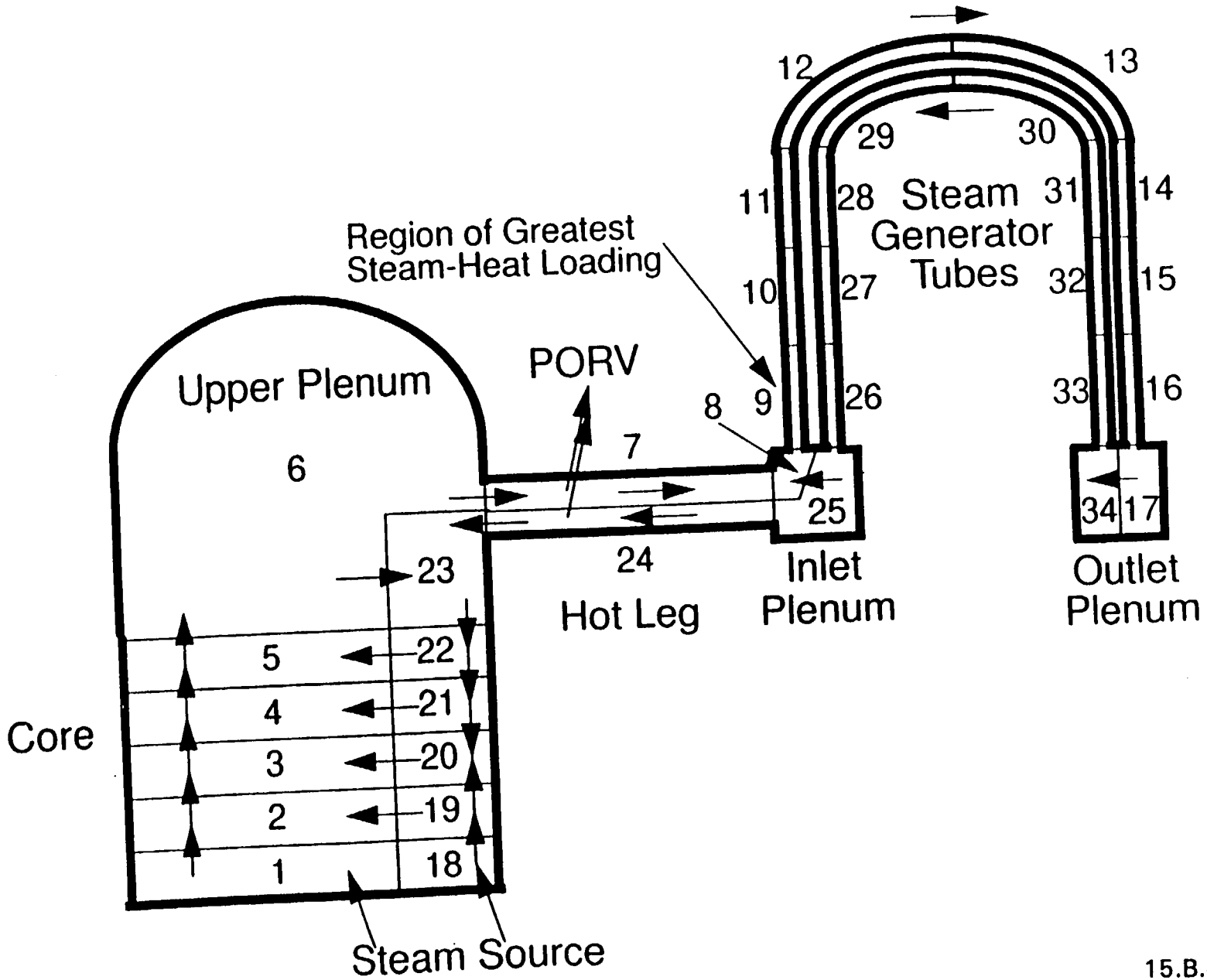
- Major VICTORIA inputs:

  - temperatures (fuel, gas, heat structure)

- From SCDAP/RELAP5:

  - flow rates
  - geometric data (cell size, flow area, hydraulic diameter, surface area)

VICTORIA Nodalization



Fission Product Deposition (continued)

RESULTS

- Fission product heating of SG tubes is minimal ~ .15 MW/SG
  - Total decay heat ~ 20 MW
- Relatively small mass of deposited material
  - Total release ~ 400 Kg
  - Fission product release ~ 50 Kg
- Deposited thickness is small ~ 30 μ
  - Tube thickness ~ 1270 μ
  - Tube diameter ~ 20,000 μ
- Dominant mechanisms are vapor condensation, settling, turbulent deposition.
- Revolatization occurs in upper plenum, hot leg, and inlet plenum.

JAERI Evaluation of Steam Generator Tube Integrity

- JAERI used:
  - SCDAP/RELAP5 for T/H
  - MELCOR for fission product concentration in the hot leg
  - ART for fission product deposition in SG tubes
- JAERI concluded: surge line fails first, but also substantial fission product heating of tubes.
- Main issue with JAERI fission product analysis is that the temperature of steam entering the tubes is assumed to be equal to the hot leg steam temperature.
  - Resulted in gas-to-wall temperature difference of up to 250K leading to substantial fission product deposition by thermophoresis. (SCDAP/RELAP5-predicted temperature difference was about 15K.)
  - Use of hot leg steam temperature in tubes would fail tubes first, irrespective of fission product deposition.

JAERI Evaluation of Steam Generator Tube Integrity (continued)

- Other issues:
  - Deposition due to entrance effects (such as in the JAERI analysis) would be inside the .5 m-thick tube sheet.
  - Dominant mechanism in NRC analysis was gravitational settling in the U-bends of the tubes, resulting in deposition over the width of the bundle (about 5 m). This mechanism was not considered in JAERI analysis. Instead, JAERI temperature assumptions caused deposition to be concentrated over a much shorter distance at the tube entrance.

Conclusions

- Tube heating during severe accidents has been analyzed using benchmarked models (validated against sealed experimental data), undergone peer review, and sensitivities examined through parametric variations.

- Variation in tube temperatures estimated at 20–50 °K.

- Evaluation of tube performance during severe accidents would benefit from resolution of uncertainty regarding T/H conditions.

- Confirmation of temperature variation/uncertainty underway.

  - More rigorous consideration of uncertainties in SG inlet plenum mixing.

  - Additional sequence/plant variation.

  - More detailed CFD modeling.

Risk Metrics
("ΔLERF or Δrisk for alternative repair criteria")

Depending on definition of LERF, actual core damage source terms with SGTR may or may not exactly meet definition.

However, they are expected to be closer to LERF than to contained core damage event source terms. So they are treated as LERF.

Use of full PRA level 3 consequence calculations for the various SGTR sequences is not yet feasible, due to the current lack of ability to predict accurately some level 2 phenomena.

Effects of RCS blowdown through faulted SG during core damage are not yet evaluated:

- tube temperatures and additional creep damage
- jet cutting damage
- primary-to-secondary flow rates as a function of time
- secondary side flow rates and velocities
- secondary structure temperatures
- effects of radionuclide deposition before release

So, source term for SGTR sequences not yet refined

Steve Long

Risk Metrics
("ΔLERF or Δrisk for alternative repair criteria")
(Continued)

Criteria for acceptance of consequences not clear, either.

Safety Goal numerical objectives apply only to close-in populations (1 mile and 10 miles)

But bulk of health consequences occur at greater distances, especially if evacuation is credited.

Steve Long

Human Reliability Analysis

1. Accident sequences are developed for plausible initiating events on the basis of functional success criteria, which provide the plant model context

2. Operator responses are assumed to be driven by
   - procedural guidance
   - perception of plant condition
   - training
   - experience

- Operator responses are identified through an understanding of the developing plant status and corresponding procedural cues and directions

- Innovative actions not addressed by procedures are not typically credited

-

Parry

DEFINITION OF HUMAN FAILURE EVENTS

- Current PRA practice is to define human failure events associated with those responses that, if not performed in a timely manner, lead to a failure of a function or system required to respond to an initiating event

- The consequences of performing actions that have a detrimental impact on mitigation are typically not modeled

- The plant conditions implied by the accident sequence provide the contextual definition of the HFE (cues, time available, equipment available, etc.)

QUANTIFICATION

- There is no consensus on how to estimate the probabilities of human failure events

- In general, operator performance is recognized as being a function of a number of factors, including
  - clarity and definitiveness of indications
  - time available for completion of response
  - experience and/or training
  - procedural guidance and completeness
  - plant ergonomics

- Methods differ in how these factors are addressed and there is significant variability between results from different methods

- There is little, if any, actuarial data to calibrate the probabilities

PRACTICAL IMPLICATIONS FOR USE OF HRA IN DECISION-MAKING

- Key task is determination of whether indications, procedures, and available equipment are adequate to provide a success path for each scenario considered credible and not screened on likelihood

- Uncertainty in probability values should be addressed by taking into account the variability in methods, or adopting a conservative method

Uncertainties In Risk Assessments

Human error probabilities

NDE detection of flaws (probability of detection)

Tube strength estimates based on NDE characterization of flaws

Steve Long

Uncertainties In Risk Assessments (Continued)

Thermal-hydraulic modeling:

RELAP/SCDAP models one *average* hot tube, but we need to know *hottest* tube temperature to assure survival of all tubes

We need *distribution* of tube temperatures to predict failure probability of significant flaws and high temperatures coinciding in same tube.

Mixing of counter-current flows in SG inlet plenum is not well enough understood to take into account:

different plenum geometry of different reactor vendor types

effects of leakage in tubes

Sequences with small leakages (elsewhere) in RCS appear to be more challenging than no leakage or large leakage, but effects of variations in leak rate and initiation timing are not explored for small leakage scenarios.

Consistent differences in results between MAAP and SCDAP/RELAP not fully understood.

Effects of radiative heat transfer between fluids and walls.

Steve Long

Uncertainties In Risk Assessments (Continued)

Materials and structural issues:

Creep models for RCS components assume infinitely long, thin-wall, straight tubes, although bends and welds are expected to dominate failure behavior.

Differential thermal growth of different temperature tubes is not considered as a possible mechanism for displacing degraded tube sections from confining structural supports.

Creep of tubes at high temperatures may result in tubes contacting each other before failure.

Potential for flows through cracks for eroding holes and cutting adjacent tubes are not yet adequately known.

Steve Long

Integrated Decision Process

RG-1.174 delineates 5 “principles” to be addressed in making a risk-informed decision

Farley and ANO-2 SERs provide examples of how those principles were addressed for SG degradation - related plant license amendment requests.

Principles of Integrated Decision Making

1. The proposed change meets the current regulations unless it is explicitly related to a requested exemption or a rule change.

2. The proposed change is consistent with the defense in depth philosophy.

3. The proposed change maintains sufficient safety margins.

4. When proposed changes result in an increase in core damage frequency or risk, the increases should be small and consistent with the intent of the Commission's Safety Goal Policy Statement.

5. The impacts of the proposed change should be monitored using performance measurement strategies.

PLUS: Consideration of uncertainties and their potential effects on the decision.

Application of Principles to Farley Decision

Principles:

1. Condition of the tubes was projected to have 99%+ probability for withstanding design basis accidents.

2. Condition of tubes was projected to have a 90% probability of withstanding severe accidents.

3. Condition of tubes was projected to have about 50% probability of meeting 3 x NOΔP. (Deterministic process normally requires 95%.)

4. Projected ΔLERF met RG-1.174 numerical guidance.

5. Impacts were not expected to be monitorable, but repetition of non-compliance was not an issue due to SG replacement. (A license requirement was added to address potential for obvious failure to meet expectations during last part of operating cycle.)

Application of Principles to Farley Decision

Uncertainties:

1. Uncertainties involving thermal-hydraulic projections were incorporated into ΔLERF estimates by integrating over estimated distributions.

2. Those involving flaw size projections were addressed by a sensitivity study indicating need for improvement from previous inspections. Staff's judgement that sufficient improvement had occurred was addressed in SER.

3. Acknowledged uncertainty and sensitivity to 0.25" crack length threshold for cutting assumption.

Application of Principles to ANO-2 Decision

Requested license amendment was denied.

NDE uncertainty and its effects on projection of flaw sizes was fundamental reason.

POD for flaws that could grow to be susceptible to MSLB was not very high during previous inspections.

Licensee's projections credited inspection improvements when none could be credited by staff.

Uncertainty in flaw strength estimates based on NDE results were found to be large. (At estimated strength of 4000 psi, 5% confidence value was 2700 psi.)

Other Issues of Interest from ANO-2 Application

1. Licensee's analysis showed that "high-dry" severe accident sequences with depressurized SGs would fail tubes with flaws that were likely to be present.

   This resulted in low ΔLERF for leaving larger flaws in service.

   But, this reversed the nature of what was conservative or non-conservative in the analysis.

2. SCDAP/RELAP analysis of licensee's proposed depressurization strategy showed large margins for success in protecting the tubes during "high-dry" sequences.

3. Slow RCS depressurization sequences were investigated for the first time.

   They were found to be substantial challenges to flawed tubes and sensitive to the size and initial opening time of the small hole.

STEAM GENERATOR ISSUES SUMMARY

Presented to

ACRS STEAM GENERATOR DPO COMMITTEE

By

Jack Strosnider, Director
Division of Engineering, NRR

October 13, 2000

DPO / STEAM GENERATOR ISSUES ARE TREATED SERIOUSLY

- Assessment of Issue Extensively Documented
  e.g., DPO Considerations Document, NUREG Reports

- Development of Regulatory Framework
  e.g., treatment of alternate repair criteria and methods

- Plant Specific Evaluations
  e.g., Farley, Arkansas 2

- Research Activities
  e.g., tube cutting/erosion, vibration

MAINTAINING SAFETY

- MAINTENANCE OF STEAM GENERATOR TUBE INTEGRITY HAS BEEN IMPROVING
  - Fewer Tube Ruptures and Leaks
- RISK INFORMED APPROACH
- MAINTAINING PUBLIC HEALTH AND SAFETY IS OUR HIGHEST PRIORITY

FUTURE ACTIONS

- INDIAN POINT 2 LESSONS LEARNED
- NEI 97-06 LICENSE CHANGE PACKAGE
- CONTINUED SUPPORT FOR ACRS REVIEW