PDR

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

AUGUST 14 1981

Docket No. (All Operating PWRs)

Licensees: B&W, Westinghouse & CE Owners Groups

SUBJECT: SUMMARY OF MEETINGS WITH THE BABCOCK & WILCOX, WESTINGHOUSE, AND COMBUSTION ENGINEERING OWNERS GROUPS ON JULY 28, 29 AND 30, 1981 RESPECTIVELY, CONCERNING PRESSURIZED THERMAL SHOCK TO REACTOR PRESSURE VESSELS (RPV).

Introduction

The meetings were held in Bethesda, MD at the request of the NRC staff in order to 1) present, briefly, the staff's analysis of the problem and the actions the staff intends to take and 2) to hear, from the PWR owners groups, the results of their analyses to date and their proposed actions concerning the problem. The meeting followed the published agenda, Enclosure 1. The attendees for each meeting are identified in Enclosure 2. Visual aids for the staff's presentation are included in Enclosure 3. Visual aids for the owners groups' presentations are included in Enclosure 4 (B&W), Enclosure 5 (Westinghouse), and Enclosure 6 (CE).

Summary of the Staffs Presentation

The staff has reviewed the owners groups responses of May 15 and the licensees' responses of May 22, 1981 to our letter dated April 20, 1981 concerning the issues. As a result of these reviews and the staff's independent review of the concern, the staff has concluded that:

1. Efforts should continue to reduce the probability and severity of overcooling transients.

2. Reliance on operator action to prevent repressurization is not an acceptable long term resolution.

3. An upper limit should be established for RT_{NDT} for continued operation.

The staff also has concluded that, for plants where the reactor pressure vessels have reached or are approaching the threshold referenced temperature stated in 10 CFR 50, Appendix G IV C, action is needed now to begin resolving the long term problems. The plants involved include the six plants which have the highest RT_{NDT} Values. This conclusion is based in part on the fact that any proposed corrective action must allow adequate lead time for planning, review, approval and procurement. Accordingly, the staff intends to request appropriate action by the six plants and the staff expects the owners group participation in the resolution of these concerns.

The NRC takes the position that there is a need now to take prompt positive action toward defining specific actions designed to prevent the potentially damaging transients or mitigate their effects. To ensure that such action is initiated, at least with respect to the most vulnerable plants, a draft letter has been prepared requesting submittal of plans for resolution. At each meeting the Owners Group representatives were given draft copies for review and comment.

Summary and Conclusion of Meeting with B&W Owners Group

The B&W Owners Group has continued on the program defined in the May 15, 1981, response. The Small Break LOCA (SBLOCA) is the bounding transient of concern. Plant specific analyses will be provided. Assumptions used in the analyses will be based on realistic assumptions. Using realistic assumptions, preliminary analysis for Oconee 1 shows a vessel lifetime of 32 Effective Full Power Years (EFPY) for SBLOCA. This assumes operator action within 30 minutes to control subcooling. The staff has concerns on the acceptance criteria of the 32 EFPY, the clarity of procedures and the adequacy of the operator training. The B&W Owners Group is not considering annealing or removing fuel assemblies. The path that is being followed is realistic analyses in plant specific analyses. The schedule for completion of these analyses is as follows: (1) Oconee 1, December 31, 1981; (2) Rancho Seco, March 1, 1982; (3) others, later in 1982. The staff discussed Item III of the agenda which are subjects which the staff wishes to see addressed in the reports. The B&W Owners Group committed to have their Material Subcommittee meet with the staff to appraise the staff of their progress. This meeting will address the systms concerns also.

Summary and Conclusions of Meeting with Westinghouse Owners Group

The discussion was directed toward the efforts underway to provide a generic solution; plant specific comments did not go into detail. The Owners Group plan goes further than the scope of the May 15th letter. A committee ("Utility Task Force") has been formed to deal with the issue which they agree is a problem demanding near-term action. They have an orderly generic program in place with a planned transition into plant-specific efforts. The accident sequences were described with emphasis on the operator instructions and actions. The Westinghouse guidelines assume that operators will take action in transients so the instructions are aimed to minimize operator error. The result has been to minimize operator actions but in a large steam line break (SLB) termination of safety injection (SI) and auxiliary feedwater (AFW) depends on operator action. The fracture mechanics calculations of crack initiation and arrest were reviewed. Remedial actions are being considered and were discussed. They believe that all Westinghouse operational transients are bounded by the small SLB and, since their RPVs can get through that accident without operator action, no operator action would be needed for operational transients. The Owners Group took the position that a request to provide plant specific analyses in 90 days would be an impediment to the generic program.

Summary and Conclusions of Meeting with CE Owners Group

The presentation was generic and the Owners Group requested that plant specific questions be directed to the appropriate licensee. Work has been going on under the direction of the Analysis and Operations Sub-Committee of the Owners Group and results to date indicate that no action is required for the most limiting plant for the next five EFPY. CE stated that they prefer to use fracture mechanics in the analysis rather than RT_{NDT} data. No engineering analyses have yet been performed to assay any of the suggested (or any other) remedial measures but CE will provide solutions within the 5 EFPY margin they have established. Currently, operators are instructed to achieve 50°F subcooling in a LOCA event and they may discontinue HPI under guidelines provided. The statements found in some CE submittals that "no operator action" is expected in some emergencies was qualified to mean: none beyond prescribed procedures. The most limiting event for CE plants is the SLB as was stated in the April NRC meeting. The analyses aim at both crack initiation and arrest with the final condition of cracked with no RPV wall penetration, preserving vessel integrity, being judged acceptable. The fracture mechanics analyses follow essentially the same method as is used by ORNL and the NRC staff except that the computer program calculates the J-integral rather than the stress intensity factor. It was agreed that a meeting will be held in September or October, 1981, to update the NRC on the technical progress in the CE program and that the Owners Group will review and comment on the draft NRC letter to licensees.

Guy Vissing

Guy Vissing, Project Manager
Operating Reactors Branch #4
Division of Licensing

Richard E. Johnson

Richard E. Johnson, USI Task Manager
Generic Issues Branch
Division of Safety Technology

Enclosures:
As Stated

cc w/enclosures:
See next page

MEETING SUMMARY DISTRIBUTION

Licensee: B&W, Westinghouse & CE Owners Groups

* Copies also sent to those people on service (cc) list for subject plant(s).

Docket File
NRC PDR
L PDR
ORB#4 Rdg
TNovak
JStolz
Project Managers (31)
Licensing Assistants (5)
OELD
Heltemes, AEOD
IE-3
SShowe
Meeting Summary File-ORB#4
RFraley, ACRS-10
Program Support Branch
DL Branch Chiefs (4)
JOlshinski
BGrimes, DEP
SSchwartz, DEP
SRamos, EPDB
FPagano, EPLB

R. E. Johnson
T. Murley
E. Throm
L. Lois
W. Hazelton
B. W. Klecker
J. Pellet
E. J. Brown
E. Abbott
H. Ornstein

F. Ashe
T. Marsh
F. Shroeder
B. D. Liaw
H. Denton
M. Vagins
D. L. Basdekas
P. N. Randall
J. H. Austin
W. Bock

W. Lanning
D. Eisenhut
G. Lainas (2)
R. Vollmer
R. Mattson
T. Speis
S. Hanauer
L. Shao
R. Bernero
I. Igne
R. Crouse

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

Enclosure 1

July 20, 1981

Docket: (All Operating PWR's)

MEMORANDUM FOR: Thomas Novak, Assistant Director for Operating Reactors, DL

FROM: Guy Vissing, Project Manager
Operating Reactors Branch #4, DL

SUBJECT: FORTHCOMING MEETING WITH THE BABCOCK & WILCOX, COMBUSTION ENGINEER, AND WESTINGHOUSE NSSS OWNERS GROUPS CONCERNING THE THERMAL SHOCK TO RPV ISSUE

Time & Date: According to following schedule

	B&W Group	Westinghouse Group	CE Group
Time & Date:	July 28, 1981 1:00-4:15pm	July 29, 1981 1:00-4:15pm	July 30, 1981 1:00-4:15pm
Location:	P-118 Phillips Building Bethesda, MD	P-118 Phillips Building Bethesda, MD	P-118 Phillips Building Bethesda, MD

Purpose: To discuss the Thermal Shock to Reactor Pressure Vessel Issue according to the attached agenda.

Requested Participants:

NRC
T. Murley, et. al.

B&W Owners Group
John Mattimoe, Chairman
B&W Representatives
Operating Plant Owners Group Representatives

Westinghouse Owners Group
Robert Jurgensen, Chairman
Westinghouse Representatives
Operating Plant Owners Group Representatives

Combustion Engineers Group
Ken Morris, Vice Chairman
CE Representatives
Operating Plant Owners Group Representatives

Guy A Vissing

Guy Vissing, Project Manager
Operating Reactors Branch #4
Division of Licensing
Phone (301) 492-8136

cc: See next page

AGENDA

Meetings with PWR Owners' Group Representatives and Vendor Representatives

Pressurized Thermal Shock

I. Introduction - NRR — 15 min.

- Purpose of Meeting
- Summary of NRC Activities
- NRC Plans

II. Owners' Group Representatives — 60 min.

- Summarize Current and Planned Activities
- Specific Questions
  - What is each plant's current RT_{NDT} and what is the expected yearly change over next 10 years?
  - What are each licensee's instructions to operators for operation of HPI after automatic initiation?
  - What are the estimated costs and benefits for
    - removing outer fuel assemblies?
    - raising temperature of ECC water?
    - in-place annealing?
    - other potential fixes?

III. Detailed Technical Discussions — 120 min.

- Bounding Overcooling Transients
- Estimated Probability of Bounding Transients
- Systems Response
- Assumed Operator Response
- Systems Analytical Model (Computer Codes Used)
- Verification of Systems Model
- Mixing Assumptions
- Downcomer Pressure/Temperature vs. Time for Selected Transients (Symmetric/Asymmetric?)
- Heat Transfer Assumptions
- Treatment of Clad Effects on Relatively Small Cracks
- Material Properties Used (Temperature Dependence?)
- Initial Crack Size/Shape Assumptions
- Temperature/Stress/LEFM Algorithms
- Treatment of Upper Shelf

Enclosure 2

MEETING ON THERMAL SHOCK TO RPV FOR B&W PLANTS
7/28/81
ATTENDENCE LIST

Name	Organization
Richard E. Johnson	NRC (Generic Issues Branch)
Tom Murley	NRC-DST
William A. Cross	FLA Power Corp.
J. H. Roy	B&W
B. J. Short	B&W
John Mattimoe	SMUD
Robert Dieterich	SMUD
Edward D. Throm	NRC/RSB
Lambros Lois	NRC/CPB
Warren S. Hazelton	NRC/DE
Guy T. Vissing	NRC/DL/ORB#4
John F. Stolz	NRC/DL
B. W. Klecker	NRC/DE
T. M. Novak	NRC/DL
Don K. Croneberger	GPU
Harvey W. Slager	CPLo
Daniel F. Spond	Arkansas P&L
Larry D. Young	Arkansas P&L
Ted J. Myers	Toledo Edison Co.
Frank Walters	B&W
Louis Redd	B&W
John Bohard	B&W
Robert Gill	Duke Power
Robert Borsum	B&W

Name	Organization
John Pellet	NRC/AEOD
Earl J. Brown	NRC/AEOD
E. T. Murphy	Westinghouse
Lynn Connor	Doc-Search Associates
Ed. Abbott	ACRS/NRC
Hal Ornstein	NRC/AEOD
Paulette Tremblay	NUS for NSAC/EPRI
Philip C. Wagner	NRC/DL/ORB#4
Daniel Garner	NRC/DL/ORB#4
Frank Ashe	AEOD/NRC
Joanne Donn	McGray-Hill
Tad Marsh	NRC/OCM
Frank Schroeder	NRR
B. D. Liaw	NRC/OCM
H. R. Denton	NRR
Roger Huston	Consumers Power
Milton Vagins	NRC/RES

NRC Meeting of July 30, 1981

SUBJECT: RPV THERMAL SHOCK PROBLEM IN CE PLANTS

Name	Organization
Edward D. Throm	NRC/RSB
R. K. Mattu	NUS Corp.
D. L. Basdekas	NRC/RES
Lambros Lois	NRC/CPB
P. N. Randall	NRC/IE
T. Murley	NRC/DST
R. E. Johnson	NRC/DST
R. W. Klecker	NRC/DE
W. S. Hazelton	NRC/DE
K. J. Morris	OPPD
J. K. Gasper	OPPD
William E. Burchill	C-E
Dan Peck	C-E
S. K. Mathavan	Fla Power & Light
Vernon T. Carson	Fla Power & Light
M. D. Patterson	Baltimore Gas & Electric Co.
D. R. Howard	Arkansas Power & Light
David L. Schreder	Arkansas Power & Light
David J. Ayres	C-E
Fred L. Carpentino	C-E
T. M. Novak	NRC/DL
B. D. Liaw	NRC/OCM
Earl J. Brown	NRC/AEOD
F. Schroeder	NRR/DST

Name	Organization
John H. Austin	NRC/EDO
Joe Longo	C-E
Robert A. Clark	NRC/DL/ORB#3
Monte Conner	NRC/DL/ORB#3
Vincent Callaghan	C-E
Robert Jewell	C-E
Jake Westhoven	C-E
Mike Keller	WPPSS -Washington Public Power Supply System
Jom Pfeifer	C-E
Dave Earles	C-E
Paul Bergeron	Yankee Atomic
Roger Huston	Consumers Power
Lynn Connor	Doc-Search Associates
Robert Borsum	B&W
William Bock	ACRS
John Garrity	Maine Yankee
Robert Shore	MY
John F. Hoffman	Yankee Atomic (MY)
George M. Solan	YAEC
Eric C. Biemiller	Yankee Atomic Elec. Co./ Maine Yankee
Thomas F. Starr Jr.	Northeast Utilities. (Millstone 2)

NRC Meeting of July 30, 1981 With Westinghouse Owners Group

Name	Organization
Jan S. Teraszkiewicz	PASNY IP-3
Joseph P. DeRoy	PASNY
Robert Borsum	B&W
Lynn Connor	Doc-Search
M. B. Shelton	VEPCO
D. W. Lippard	VEPCO
Joanne Conn	McGraw-Hill
William Bock	ACRS Fellow
Wayne Lanning	NRC/AEOD
T. N. Taylor	Public Service E&G Co.
J. E. McEwen Jr.	Portland General Electric
Pres Rahe	Nuclear Safety
Bruce King	Westinghouse
Daniel M. Spayer	West. Owners Group
D. L. Basdekas	NRC/RES
R. C. Mecredy	Rochester Gas & Electric
D. G. Maire	Westinghouse
E. T. Murphy	Westinghouse-Bethesda
W. J. Johnson	Westinghouse
B. S. Monty	Westinghouse
S. S. Palusamy	Westinghouse
J. T. Crane	Westinghouse
G. E. Campbell	Westinghouse
C. A. Schrock	Wisconsin Public Service Corp.
R. K. Mattie	NUS Corp.

Name	Organization
R. E. Johnson	NRC/DST
T. Murley	NRC/DST
T. Novak	NRC/DL
S. L. Anderson	Westinghouse
R. W. Klecker	NRC/DE
W. S. Hazelton	NRC/DE
T. A. Meyer	NTD
R. L. Kinsaul	SCS/APC
S. K. Mathavan	Fla Power & Light
Frank Gilman	CP&L
Edward D. Throm	NRC/RSB
Lambros Lois	NRC/CPB
Frank Dodd	Pacific Gas & Electric Co.
M. Kupinski	Northeast Utilities
D. Rawlins	Westinghouse
F. Schroeder	NRR/DST
Earl J. Brown	NRC/AEOD

SUMMARY OF THE PROBLEM

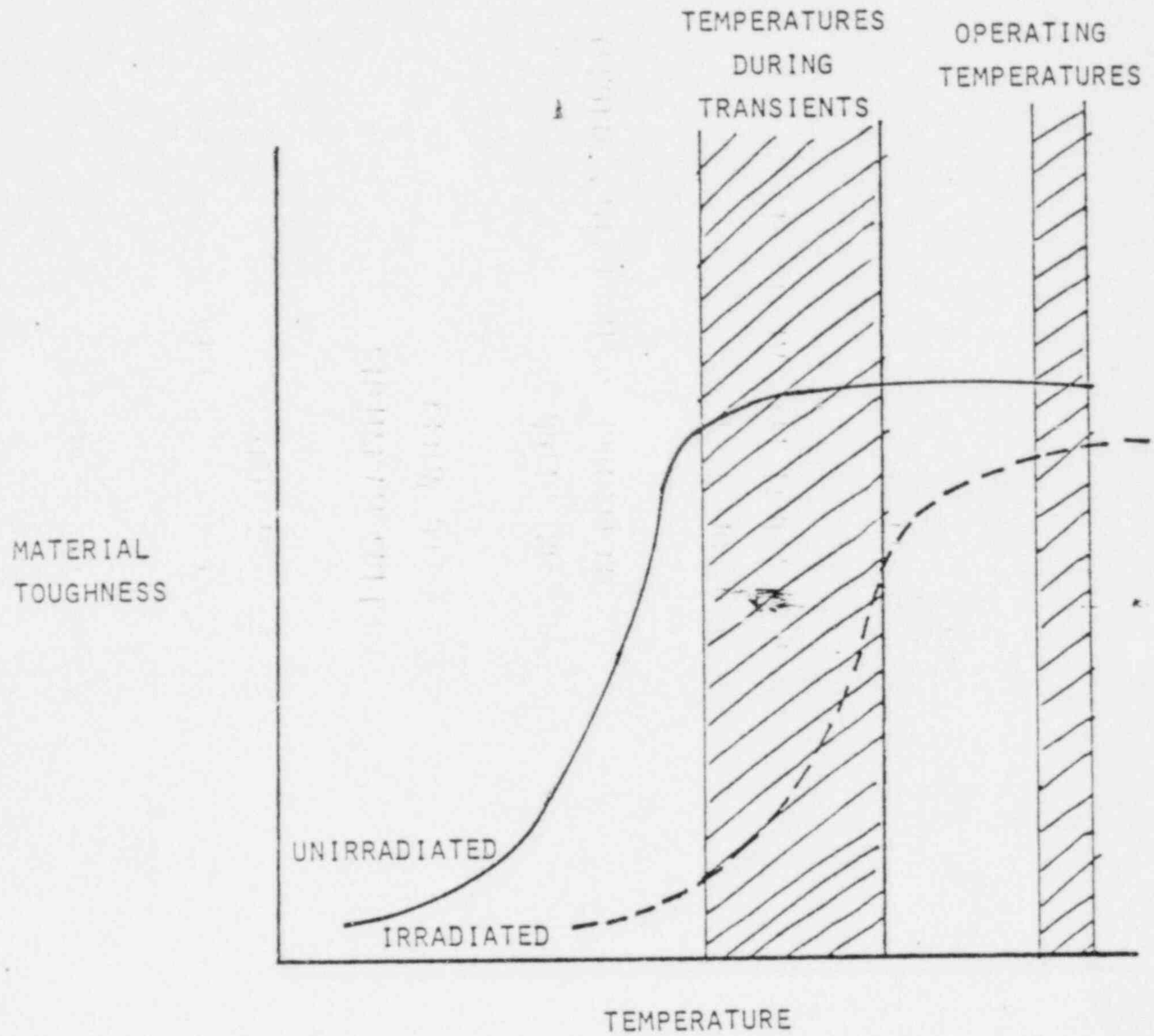
- SEVERE OVERCOOLING TRANSIENTS CAN PRODUCE HIGH THERMAL STRESSES IN THE VESSEL.

- CRACK INITIATION CAN OCCUR IF A FLAW EXISTS AND IF THE FRACTURE TOUGHNESS HAS BEEN REDUCED BY NEUTRON IRRADIATION.

- REPRESSURIZATION DURING THE TRANSIENT CAN CAUSE A CRACK TO PROPAGATE THROUGH THE WALL.

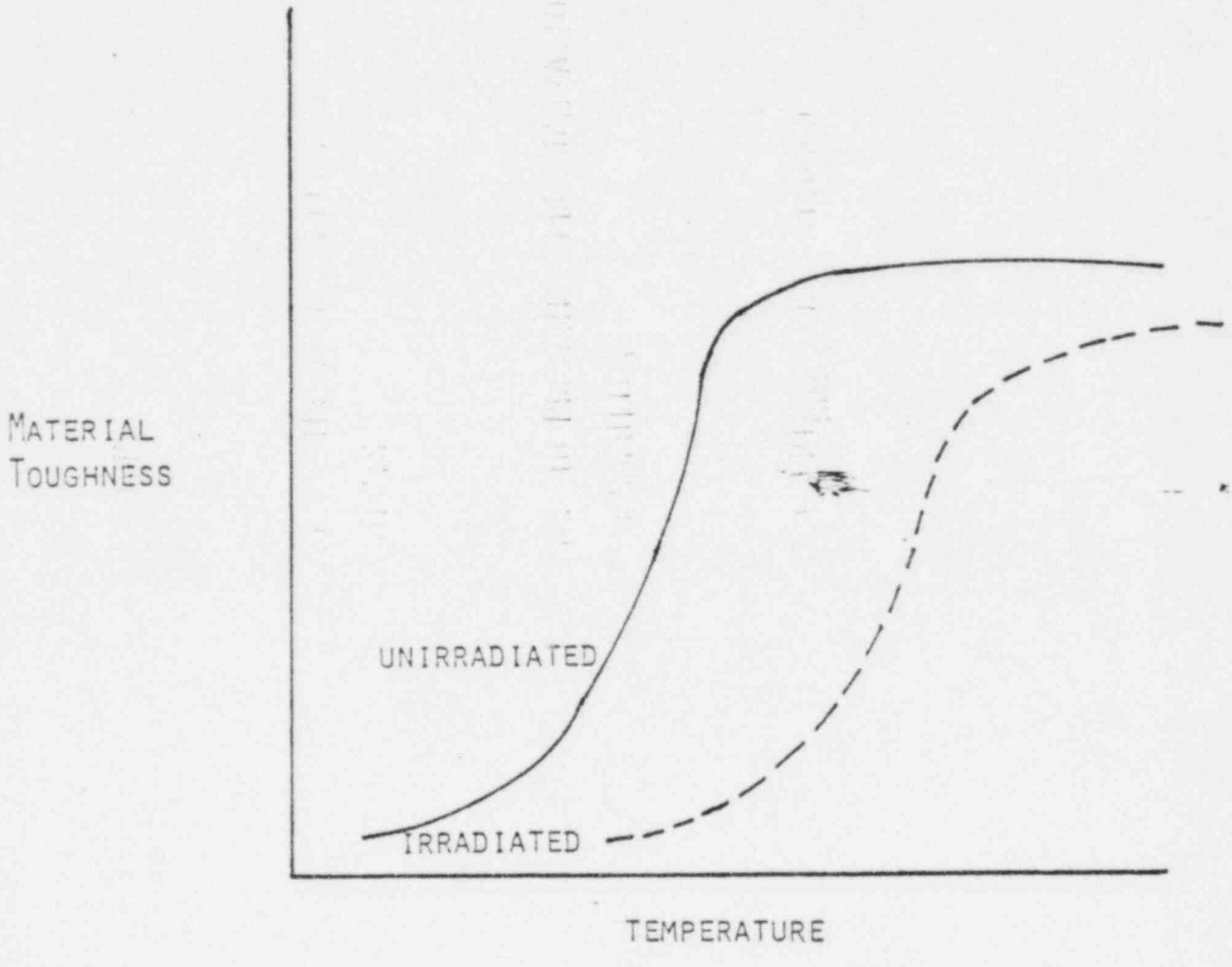
- SOME PWR VESSELS ARE APPROACHING HIGH RT_{NDT} LEVELS WHERE REDUCED FRACTURE TOUGHNESS BEGINS TO BE OF CONCERN.

Enclosure 3



TEMPERATURES DURING TRANSIENTS
OPERATING TEMPERATURES
MATERIAL TOUGHNESS
UNIRRADIATED
IRRADIATED
TEMPERATURE

NEUTRON-RADIATION-INDUCED TOUGHNESS CHANGES



POSSIBLE CORRECTIVE ACTIONS

1. REDUCE THE PROBABILITY OF OVERCOOLING TRANSIENTS
   (E.G., MORE RELIABLE CONTROL SYSTEM)

2. REDUCE THE SEVERITY OF OVERCOOLING TRANSIENTS
   (E.G., RAISE TEMPERATURE OF ECC WATER)

3. REDUCE THE LIKELIHOOD OF REPRESSURIZATION
   (E.G., OPERATOR PROCEDURES OR CONTROL SYSTEM MODIFICATIONS)

4. MAINTAIN HIGH FRACTURE TOUGHNESS OF VESSEL
   (E.G., REMOVE OUTER FUEL TO REDUCE FLUX AT VESSEL)

5. REGAIN FRACTURE TOUGHNESS OF VESSEL
   (E.G., IN-PLACE ANNEALING)

<u>NRC CONCLUSIONS</u>

1. EFFORTS SHOULD CONTINUE TO REDUCE THE PROBABILITY AND SEVERITY OF OVERCOOLING TRANSIENTS

2. RELIANCE ON THE OPERATOR ACTION TO PREVENT REPRESSURIZATION IS NOT AN ACCEPTABLE LONG-TERM SOLUTION

3. A LIMIT SHOULD BE ESTABLISHED ON RT_{NDT} FOR CONTINUED OPERATION

REFERENCE TEMPERATURE FOR THE PWR VESSELS OF INTEREST, ADJUSTED FOR RADIATION DAMAGE TO DATE

PLANT	NSSS	VESSEL MFR	RT_{NDT}*	(°F)
FORT CALHOUN	CE	CE	250	- 280
ROBINSON 2	W	CE	250	- 280
SAN ONOFRE	W	CE	250	- 280
MAINE YANKEE	CE	CE	200	- 230
PALISADES	CE	CE	190	- 220
YANKEE ROWE	W	B&W	180	- 210
OCONEE 1	B&W	B&W	160	- 190
ZION 1	W	B&W	150	- 180
ARKANSAS ANO-1	B&W	B&W	150	- 180
INDIAN POINT 2	W	CE	150	- 180
TMI-1	B&W	B&W	140	- 170
RANCHO SECO	B&W	B&W	130	- 160
SURRY 1	W	B&W	120	- 150
CRYSTAL RIVER 3	B&W	B&W	110	- 140

*PEAK FLUENCE, WORST WELD

Enclosure 4

THERMAL SHOCK

- BACKGROUND
- MAY 15, 1981 LETTER REPORT ACTIONS IDENTIFIED
- B&W OWNERS PLANT SPECIFIC PROGRAM (ONGOING)
  - SCOPE
  - ASSUMPTIONS
  - EXPECTATIONS
  - SCHEDULE

BACKGROUND

- Item II.K.2.13 of NUREG-0737, SBLOCA
- Generic, Conservative Analysis - BAW-1648
- Concluded:
  Lead B&W Plant (Rancho Seco) OK Into 1983
- NRC Requests of March 31, 1981, and April 29, 1981, To Assess Overcooling/Repressurization
- May 15, 1981 Letter Report Concluded The Previous Analysis Was Bounding

MAY 15, 1981, LETTER REPORT
ACTIONS IDENTIFIED

A. COMPLETED

1. ICS/NNI AND EFW SYSTEMS UPGRADES
2. REVISED SBLOCA OPERATING GUIDELINES ISSUED
3. RECOMMENDATION TO MAINTAIN BWST TEMPERATURE HIGHER THAN TECHNICAL SPECIFICATION MINIMUM
4. ATOG CONSIDERATION OF PROBLEM
5. OWNERS GROUP DOCUMENT ON R.V. MATERIALS (BAW-1511P)

B. CURRENTLY UNDERWAY

1. OWNERS GROUP R.V. MATERIALS PROGRAM
2. SURVEILLANCE PROGRAMS (APPENDIX H OF 10CFT50)

C. IMMEDIATE FUTURE

1. PLANT SPECIFIC EVALUATIONS
2. 2-D VESSEL HEAT CONDUCTION EVALUATIONS
3. NON-LOCA EVENTS EVALUATION

D. LONG TERM

1. CONSIDER THERMAL MIXING TEST
2. CONSIDER ENHANCED INSERVICE INSPECTION
3. EVALUATE IN-PLACE R.V. THERMAL ANNEALING
4. INVESTIGAT· .MPROVED DOSIMETRY AND FLUENCE CALCULATIONS

B&W OWNERS PLANT SPECIFIC PROGRAM

- SCOPE

1. SBLOCA ANALYSIS

   A. ECCS

   B. MIXING

   C. 2-D HEAT CONDUCTION

   D. FRACTURE MECHANICS

2. OVERCOOLING/REPRESSURIZATION

   A. SELECTION OF TRANSIENT

   B. HEAT CONDUCTION

   C. FRACTURE MECHANICS

- ASSUMPTIONS IN ONGOING PROGRAM

1.* PLANT-SPECIFIC LOCATION OF WELDS
(WELDS PREVIOUSLY ASSUMED TO BE DIRECTLY UNDER COLD LEG NOZZLE)

2. 8 VENT VALVES
(4 PREVIOUSLY ASSUMED, TO BOUND ALL PLANTS)

3. MORE REALISTIC HPI FLOW
(PREVIOUS ANALYSIS ASSUMED BOUNDING FLOW)

4. EXPANDED MIXING ANALYSIS
(PREVIOUS ANALYSIS WAS BASED ON LIMITED EVALUATION)

5. ASME CODE LINEAR ELASTIC FRACTURE MECHANICS ANALYSIS WITH CREDIT FOR WARM PRESTRESSING WHERE APPLICABLE
(PREVIOUSLY NO CREDIT TAKEN FOR WARM PRESTRESSING)

6.* 2-DIMENSIONAL HEAT CONDUCTION IN VESSEL
(PREVIOUSLY ASSUMED 1-D HEAT CONDITION)

7.* PLANT-SPECIFIC UPDATED MATERIAL CHEMISTRY AND FLUENCES
(PREVIOUSLY USED BOUNDING VALUES)

8. OTHER ASSUMPTIONS ARE SIMILAR TO PREVIOUS ANALYSIS BUT BEING EVALUATED (E.G. OPERATOR ACTION)

* BASED ON PRELIMINARY ANALYSIS FOR OCONEE-1, ITEMS 1, 6, & 7 COMBINED [redacted] SHOW VESSEL LIFETIME OF 32 EFPY FOR SBLOCA.

- EXPECTATIONS FOR EXTENDED LIFETIME*

  1. 32 EFPY FOR SBLOCA

  2. 32 EFPY FOR OVERCOOLING/REPRESSURIZATION

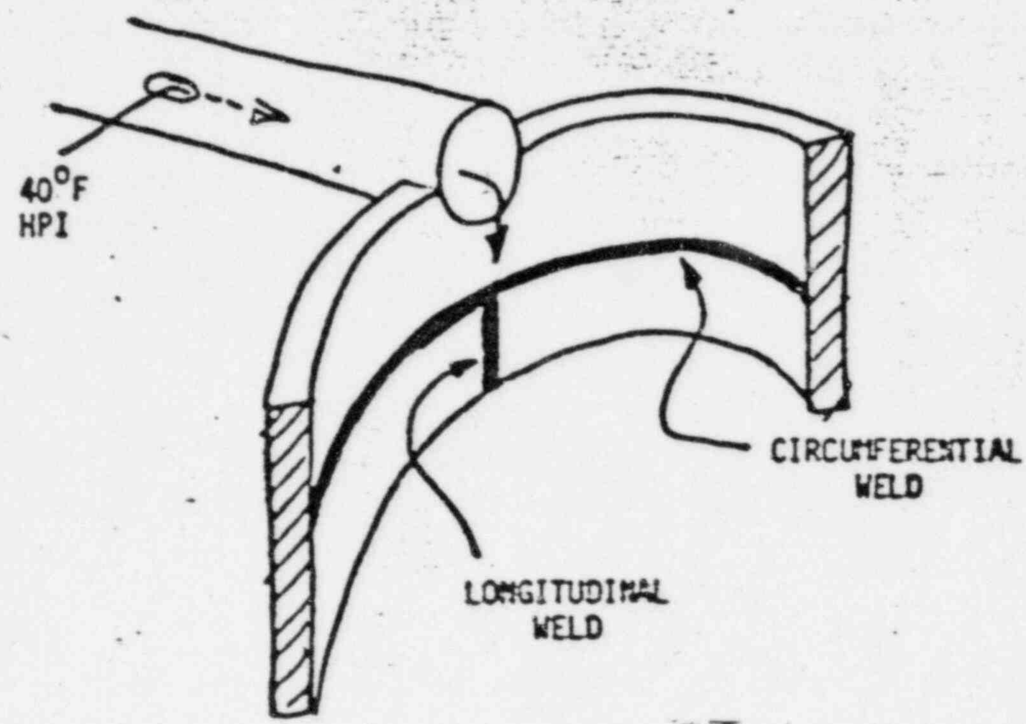
- SCHEDULE

  1. OCONEE-1 COMPLETE 12-31-81

  2. RANCHO SECO COMPLETE BY 3-1-82

  3. OTHERS LATER IN 1982, AS APPLICABLE

*WITH OPERATOR ACTION


40°F
HPI
CIRCUMFERENTIAL
WELD
LONGITUDINAL
WELD

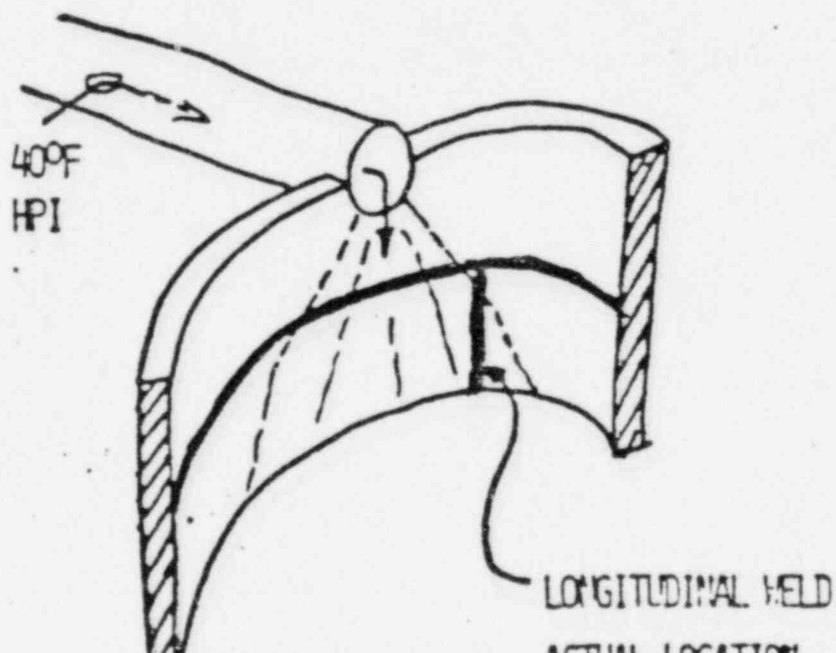

40°F
HPI
LONGITUDINAL WELD
ACTUAL LOCATION

Figure 2-1. HPI Flow Rate Vs RCS Pressure

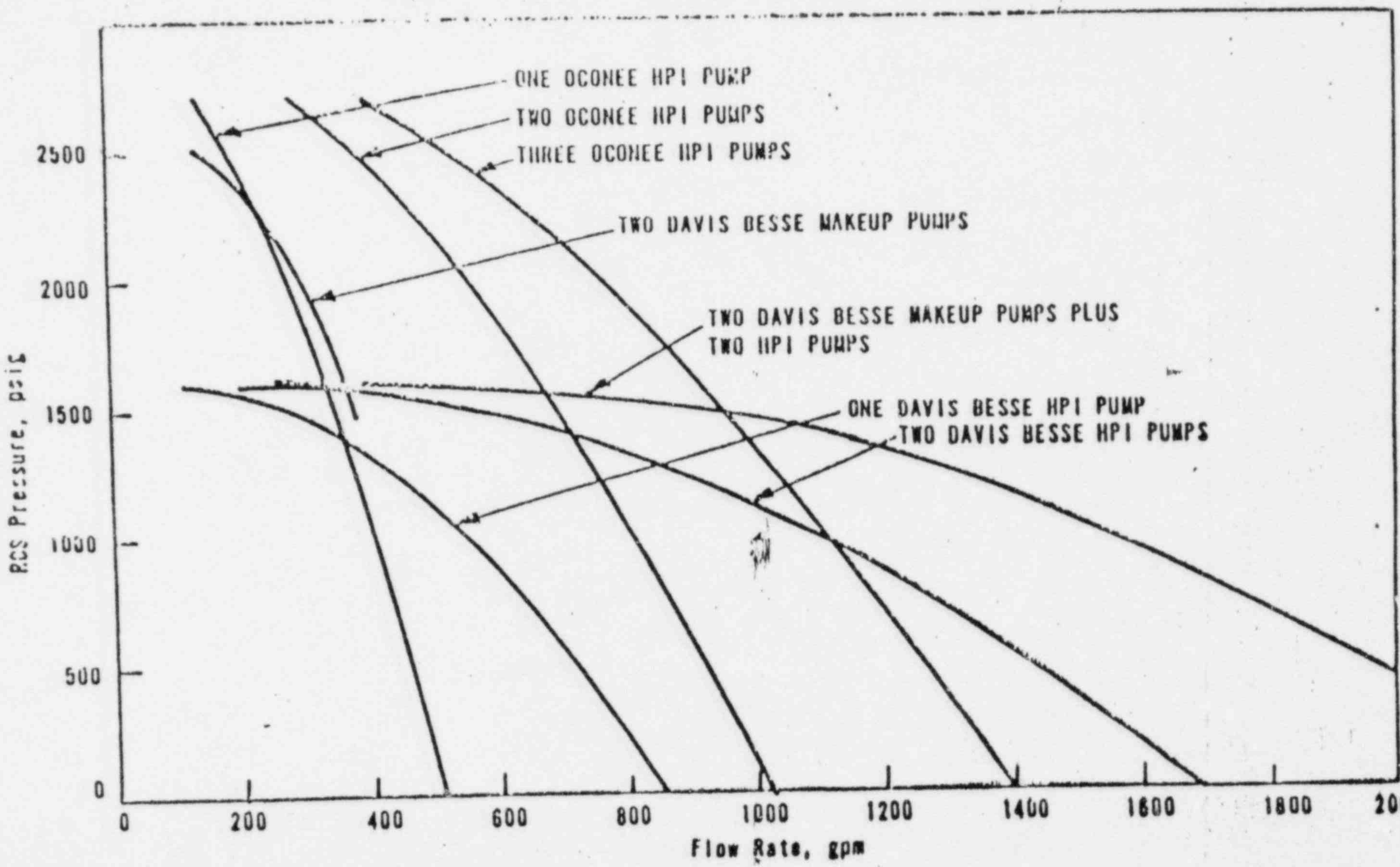
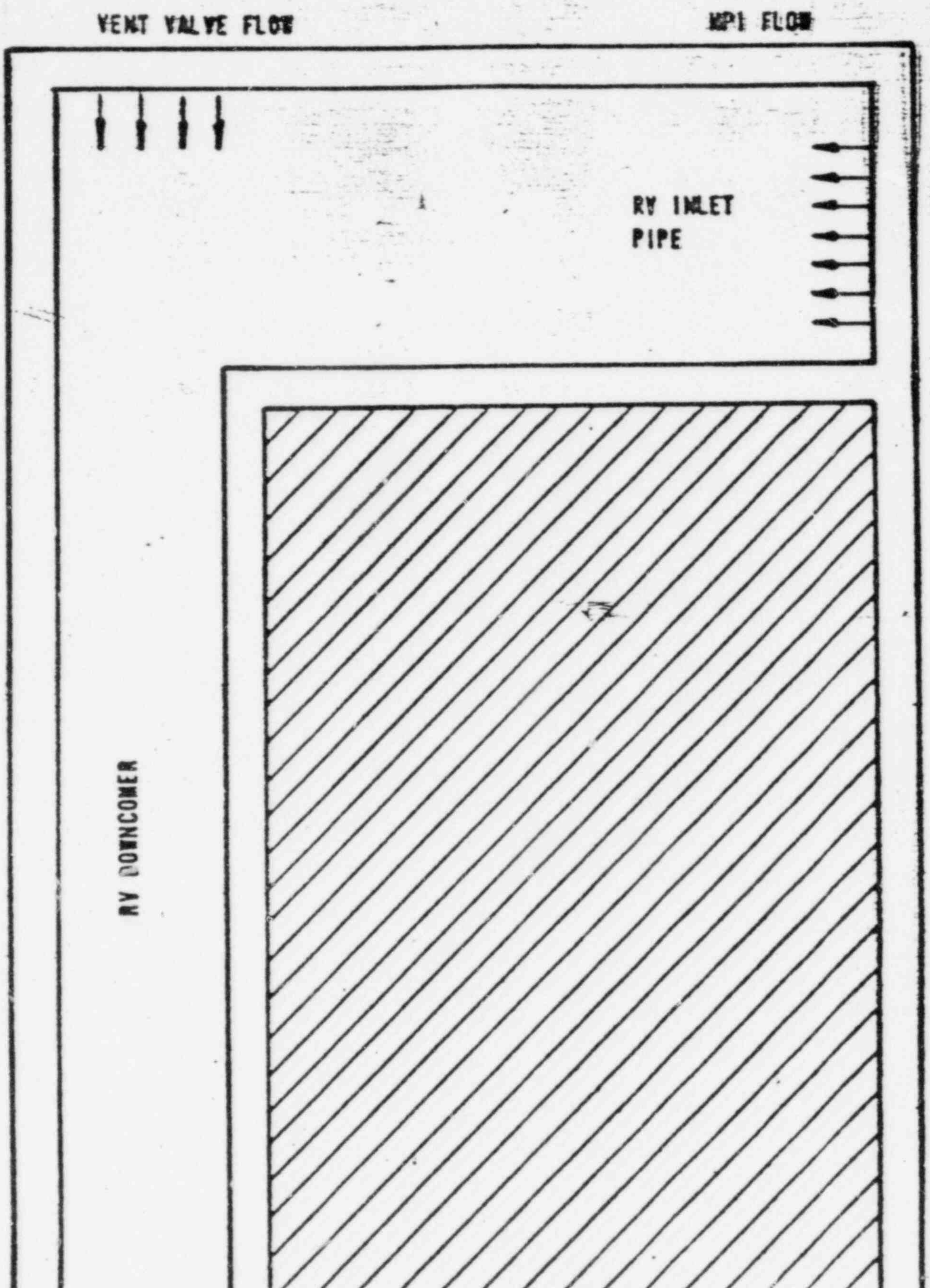


Figure 3-1. Numerical Model of MIX2 Analysis



PLANT	CURRENT RT_{NDT} AND LIMITING MATERIAL*	RT_{NDT} @ +8 EFPY
OCONEE I	152/LW	226 F
OCONEE II	168/CW	264 F
OCONEE III	166/CW	267
TMI-1	170/LW	324
ANO-1	111/CW	198
RANCHO SECO	137/LW	248
CR-3	145/CW	301
DB-1	73/CW	176

* LW = LONGITUDINAL WELD

CW = CIRCUMFERENTIAL WELD

W OWNERS GROUP EOI'S

- CURRENT REV. 2 - APRIL 1980
- GENERIC INSTRUCTIONS
- REFERENCE TECHNICAL CONTENT
- SI TERMINATION CRITERIA
  - INVENTORY
  - HEAT SINK
  - PRESSURIZED THERMAL SHOCK

Enclosure 5

GUIDANCE FOR HPI OPERATION

* DEPENDENT ON EVENT SYMPTOMS

* LOCA

* LOSS OF SECONDARY COOLANT

* REDUCE RISK OF OPERATOR ERROR

  - MINIMIZE OPERATOR ACTIONS

  - MAXIMIZE PROCEDURE UNIFORMITY

SI TERMINATION CRITERIA

LOCA

* RCS P GREATER THAN 2000 PSIG
* RCS SUBCOOLING GREATER THAN X DEG.
* PSZR LEVEL GREATER THAN 50 PERCENT
* SG HEAT SINK AVAILABILITY VERIFIED

SI TERMINATION CRITERIA

STEAM BREAK

* THOT LESS THAN 350 DEG.
* RCS P GREATER THAN 700 PSIG
* RCS SUBCOOLING GREATER THAN X DEG.
* PSZR LEVEL GREATER THAN 20 PERCENT
* SG HEAT SINK AVAILABILITY-VERIFIED

OPERATOR ACTION

* NO SI TERMINATION INDICATED FOR LOCA

* EARLY SI TERMINATION FOR STEAM BREAK

PRESSURIZED THERMAL SHOCK

- SMALL LOCA

LARGE STEAM BREAK (ZERO POWER)

SMALL LOCA SYSTEMS RESPONSE

* REACTOR AND RCP TRIP

* SI START ON LOW PRESSURE

* NORMAL FEED TRIP

* AUX FEED INITIATION

SMALL LOCA OPERATOR RESPONSE

* NO ACTIONS ASSUMED

SMALL LOCA MODELS

* WFLASH COMPUTER PROGRAM

* ECCS MODEL OCTOBER 75 VERSION

* HOT LEG BREAK NODE ADDED

SMALL LOCA MODEL VERIFICATION

- SEMISCALE PREDICTIONS
  - S-07-10D
- LOFT TEST PREDICTIONS
  - L3-1 AND L3-6

MIXING ASSUMPTIONS

* SI-COLD LEG FLOW MIXING

* COLD LEG FLOW-DOWNCOMER MIXING

SI-COLD LEG MIXING

- RAMSEY, GOLDSTEIN, ECKERT CONDUCTION MODEL

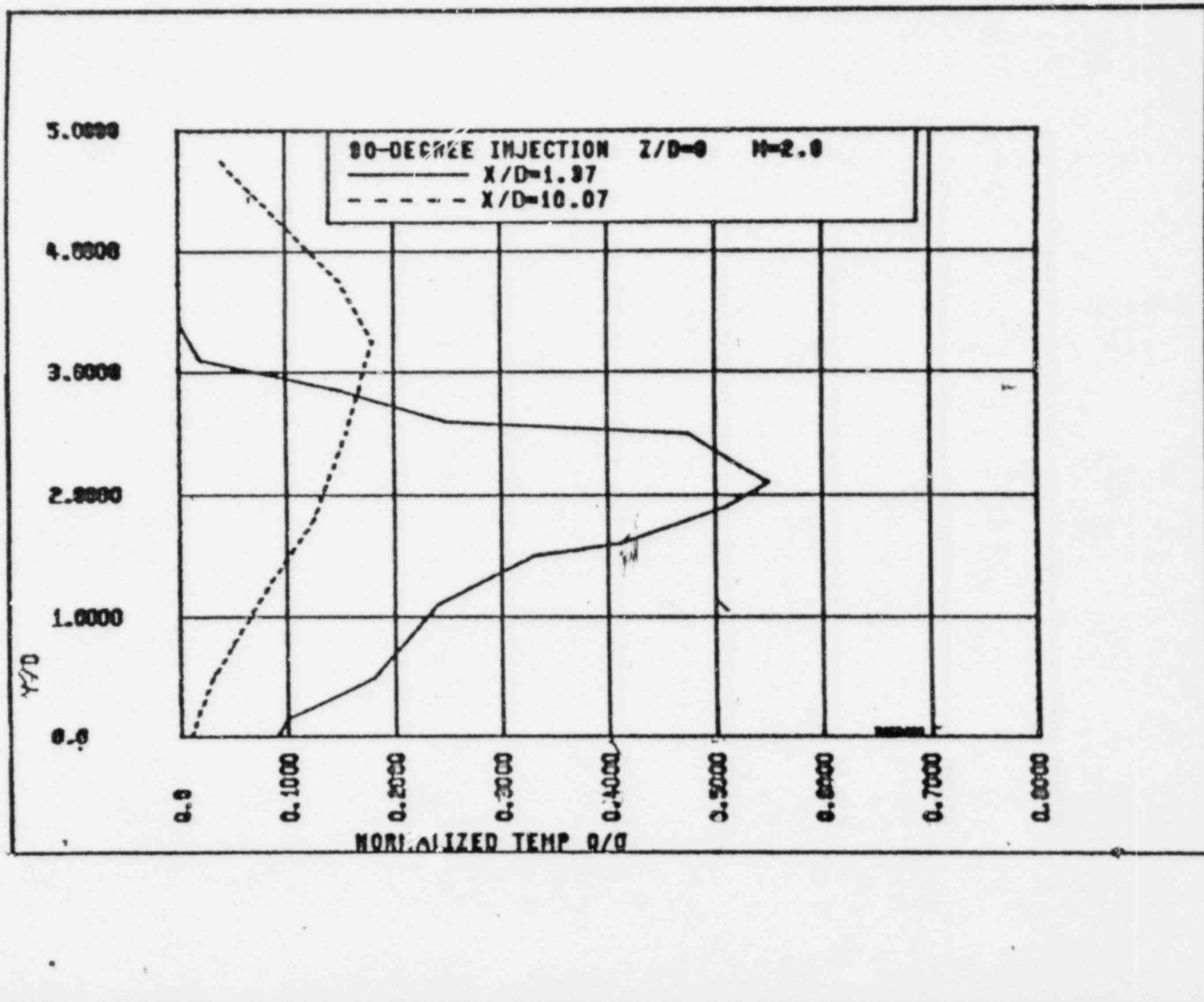
- CONSIDERABLE MIXING IN COLD LEG

COLD LEG FLOW-DOWNCOMER MIXING

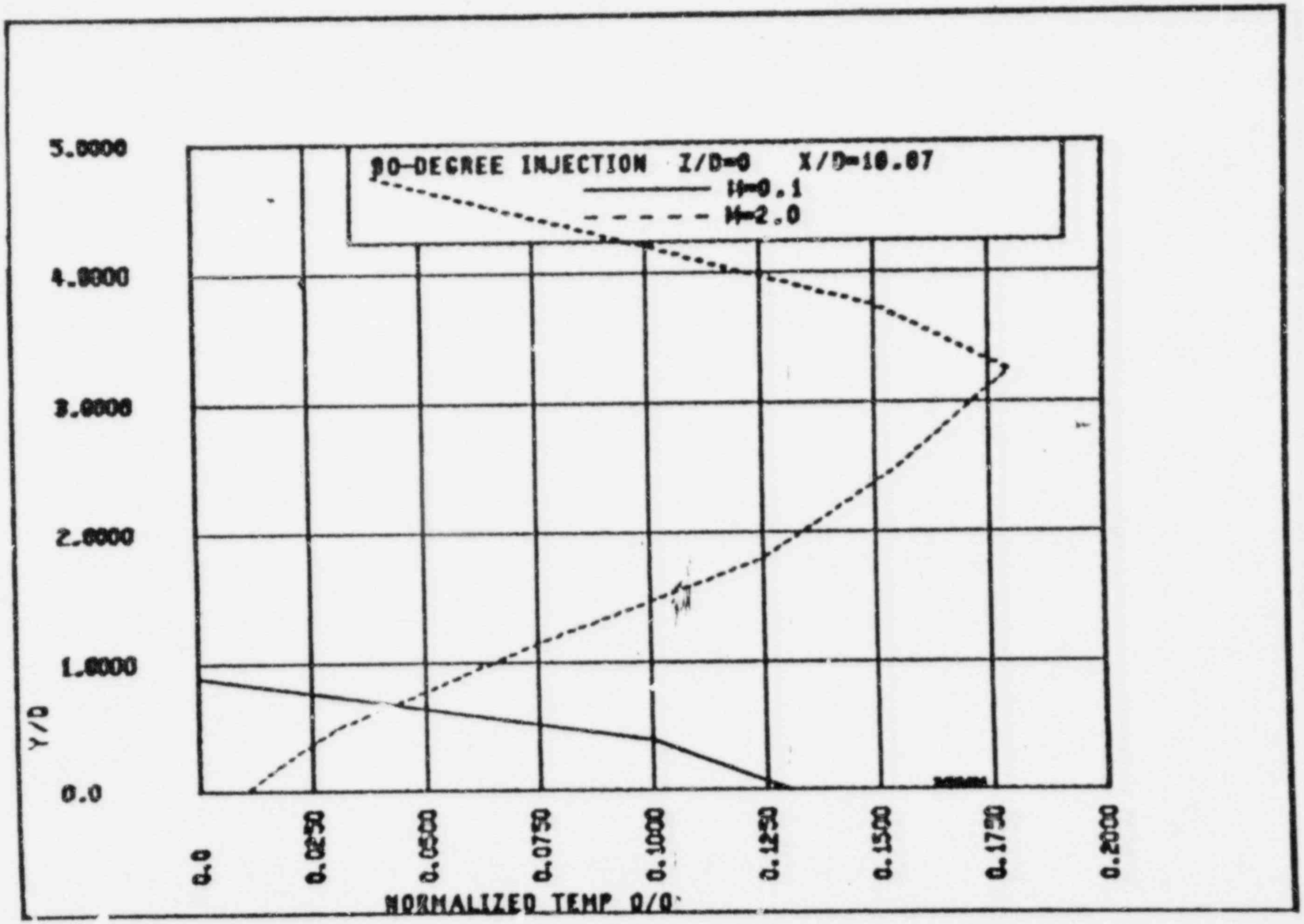
- SHORT DOWNCOMER FLOW TRANSIT TIME
- LOOP FLOW SYMMETRY
- AXIAL PROFILE NEGLIGIBLE

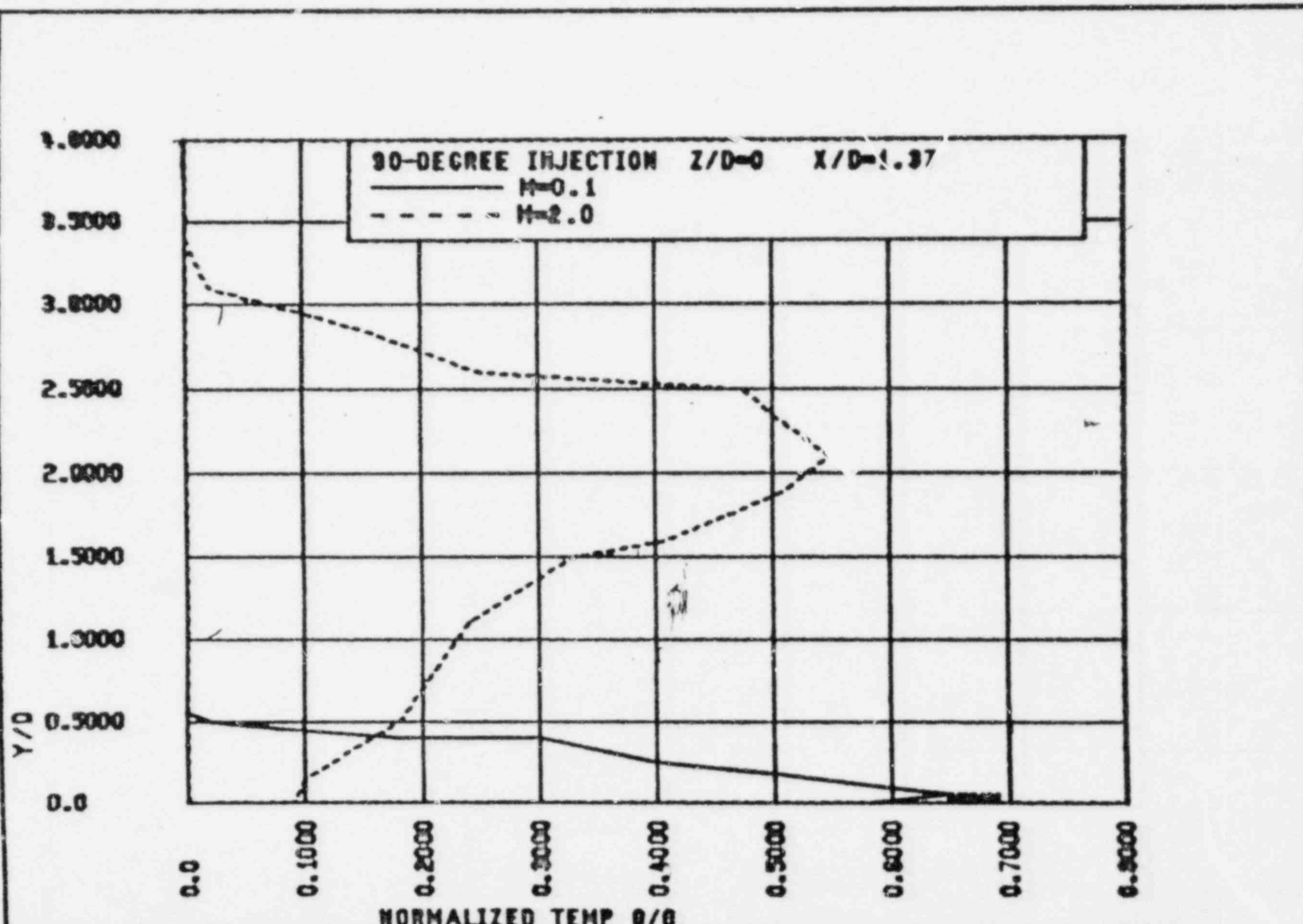
SMALL LOCA GENERIC TRANSIENTS

- 4 INCH BREAK DIAMETER
- HOT LEG BREAK LOCATION
- MAXIMUM SI FLOW DELIVERY
- MINIMUM SI AND ACCUMULATOR TEMPERATURE
- 2,3,4 LOOP PLANT TRANSIENTS

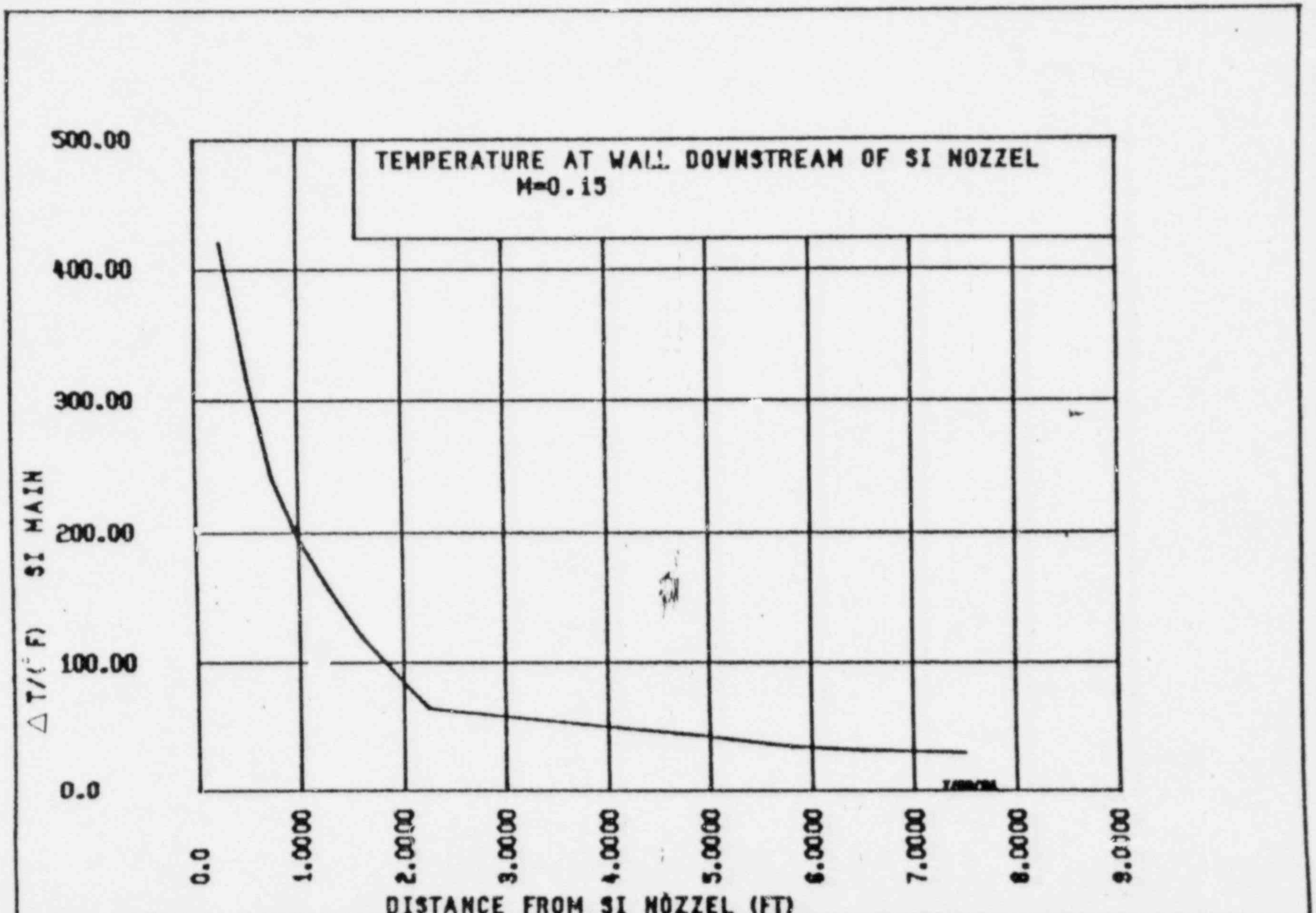


90-DEGREE INJECTION Z/D=0 M=2.0
X/D=1.37
X/D=10.07
5.0000
4.0000
3.0000
2.0000
1.0000
0.0
Y/D
0.0
0.1000
0.2000
0.3000
0.4000
0.5000
0.6000
0.7000
0.8000
NORMALIZED TEMP θ/θ

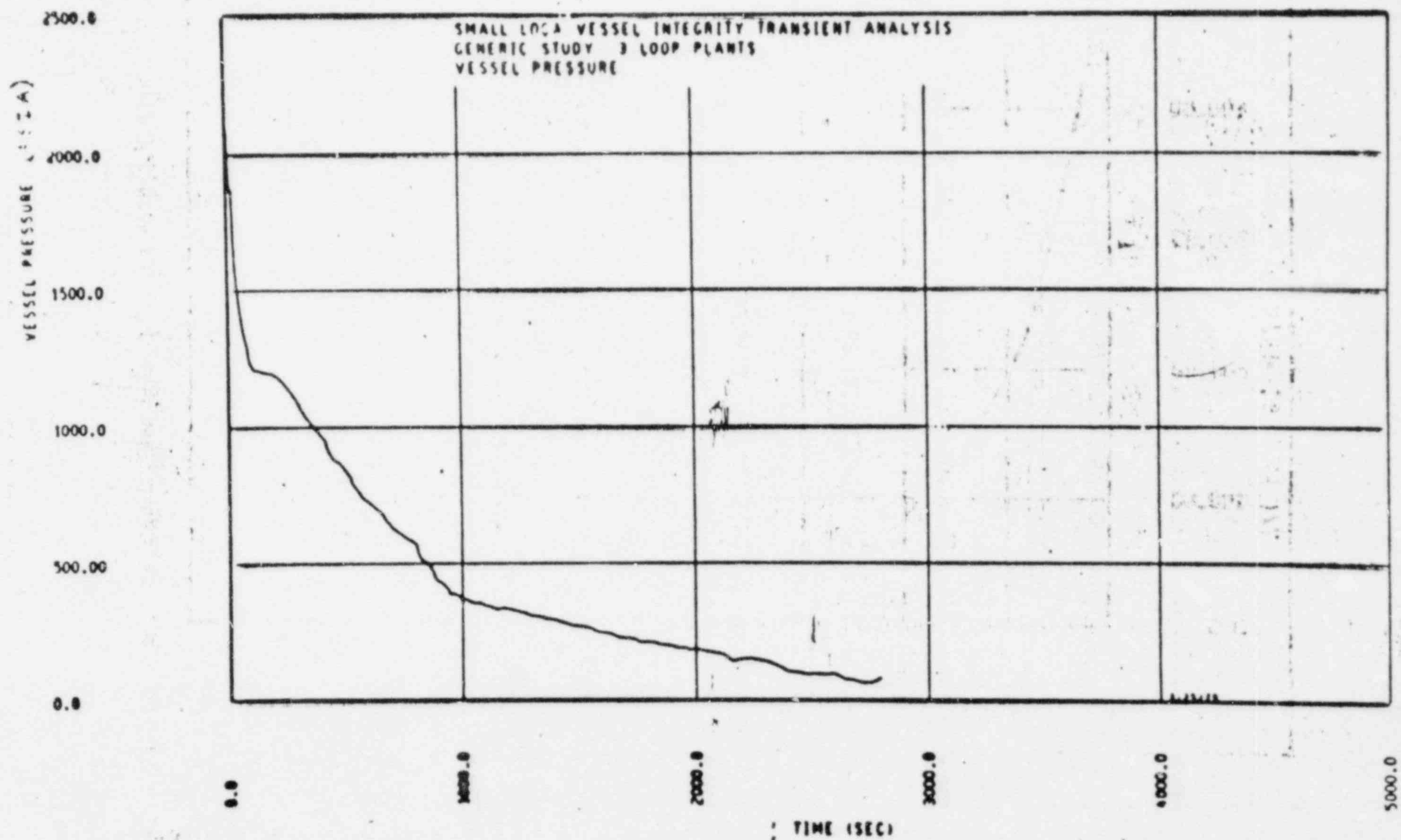

90-DEGREE INJECTION Z/D=0 X/D=10.07
M=0.1
M=2.0
5.0000
4.0000
3.0000
2.0000
1.0000
0.0
Y/D
0.0
0.0250
0.0500
0.0750
0.1000
0.1250
0.1500
0.1750
0.2000
NORMALIZED TEMP Θ/Θ


90-DEGREE INJECTION Z/D=0 X/D=1.37
M=0.1
M=2.0
4.0000
3.5000
3.0000
2.5000
2.0000
1.5000
1.0000
0.5000
0.0
Y/D
0.0
0.1000
0.2000
0.3000
0.4000
0.5000
0.6000
0.7000
0.8000
NORMALIZED TEMP θ/θ

FRAME 01 F



TEMPERATURE AT WALL DOWNSTREAM OF SI NOZZEL
M=0.15
500.00
400.00
300.00
200.00
100.00
0.0
△ T/(° F) SI MAIN
0.0
1.0000
2.0000
3.0000
4.0000
5.0000
6.0000
7.0000
8.0000
9.0000
DISTANCE FROM SI NOZZEL (FT)


SMALL LOCA VESSEL INTEGRITY TRANSIENT ANALYSIS
GENERIC STUDY 3 LOOP PLANTS
VESSEL PRESSURE
2500.0
2000.0
1500.0
1000.0
500.00
0.0
VESSEL PRESSURE
0.0
1000.0
2000.0
3000.0
4000.0
5000.0
TIME (SEC)

BOUNDING OVERCOOLING TRANSIENTS FOR PRESSURIZED THERMAL SHOCK

LARGE STEAM LINE BREAK ACCIDENT

INITIATING EVENT: DOUBLE - END GUILLOTINE SEVERANCE OF MAIN STEAM LINE

SUBSEQUENT EVENTS: RAPID COOLDOWN OF RCS

- MASS AND ENERGY RELEASE DUE TO BREAK
- COLD AFW & SI

RAPID DEPRESSURIZATION OF RCS

- VOLUME SHRINKAGE DUE TO TEMPERATURE DROP

GRADUAL REPRESSURIZATION OF RCS

- SI SYSTEM

GRADUAL INCREASE OF RCS TEMPERATURE

- TERMINATION OF AFW & SI

SYSTEMS RESPONSE

EVENTS:

- ONE SG BLOWS DOWN TO ATM
- REACTOR TRIP / TURBINE TRIP
- REMAINING SG ARE ISOLATED VIA STEAM ISOLATION VALVE
- MAIN FW PUMPS TRIPPED $\leq$ 10 S
- AFW PUMPS TURNED ON $\leq$ 60 SEC (REQUIREMENT)
- CHARGING / SI PUMPS TURNED ON
- RCP TRIPPED IF COINCIDENT LOSS OF OFF-SITE POWER
- AFW / SI SYSTEMS TERMINATED

ASSUMPTIONS: SELECTED TO MINIMIZE TEMP; MAXIMIZE PRESS RECOVERY

1) OPERATOR ACTION 10 MIN AFTER BREAK TO TERMINATE AFW FLOW AND CHARGING / SI FLOW.

2) NEGLECT DECAY HEAT, THICK METAL STORED ENERGY, AND SG REVERSE HEAT TRANSFER FOR PRIMARY COOLANT TEMP / CONSIDER FOR PRIMARY COOLANT PRESSURE.

3) SI SYSTEM OPERATES @ DESIGN CAPACITY.

4) AFW SYSTEM OPERATES @ DESIGN CAPACITY (25-36% FULL FLOW).

5) TEMP SI & AFW IS 32^{0} F.

SYSTEMS ANALYTICAL MODEL (COMPUTER CODES USED)

MARVEL - A DIGITAL COMPUTER CODE FOR TRANSIENT ANALYSIS OF A MULTILOOP PWR SYSTEM (WCAP - 9235)

CALCULATES MULTI-LOOP DETAILED TRANSIENT BEHAVIOR OF PWR SYSTEMS

SIMULATES TWO REACTOR COOLANT LOOPS INCLUDING TWO STEAM GENERATORS AND ASSOCIATED SYSTEMS

ALSO SIMULATES REACTOR KINETICS, REACTOR CONTROL AND PROTECTION SYSTEM, SAFEGUARDS SYSTEM AND OTHER SUBSYSTEMS

MODELS FLUID AS CONTROL VOLUMES

TYPICAL ANALYTICAL CASES:

MAXIMIZE COOLDOWN OF PRIMARY SYSTEM

A) WITH OFF-SITE POWER (RCP RUNNING)

B) WITHOUT OFF-SITE POWER (RCP TRIPPED)

MAXIMIZE RCS REPRESSURIZATION

VERIFICATION OF SYSTEMS MODEL (MARVEL COMPUTER CODE)

STEAM LINE BREAK ACCIDENT

MARVEL RESULTS COMPARED TO LØFTRAN AND WIT - 6

SENSITIVITY STUDIES

MAIN FEEDLINE RUPTURE ACCIDENT

MARVEL RESULTS COMPARED TO LØFTRAN AND WFLASH

LOSS OF FEEDWATER ACCIDENT

MARVEL RESULTS COMPARED TO BLKØUT AND LØFTRAN

NRC REGION IV INSPECTION REPORT 81 - 02

CONTROL OF COMPUTER PROGRAMS

SPECIFICALLY ADDRESSED MARVEL AND LØFTRAN

NO NONCONFORMANCE, UNRESOLVED, OR FOLLOW UP ITEMS WERE IDENTIFIED

DESIGN VERIFICATION

SPECIFICALLY ADDRESSED MARVEL AND LØFTRAN

NO NONCONFORMANCE OR UNRESOLVED ITEMS WERE IDENTIFIED

ONE FOLLOW UP ITEM (LOCATE SPECIFIC CALC NOTE)

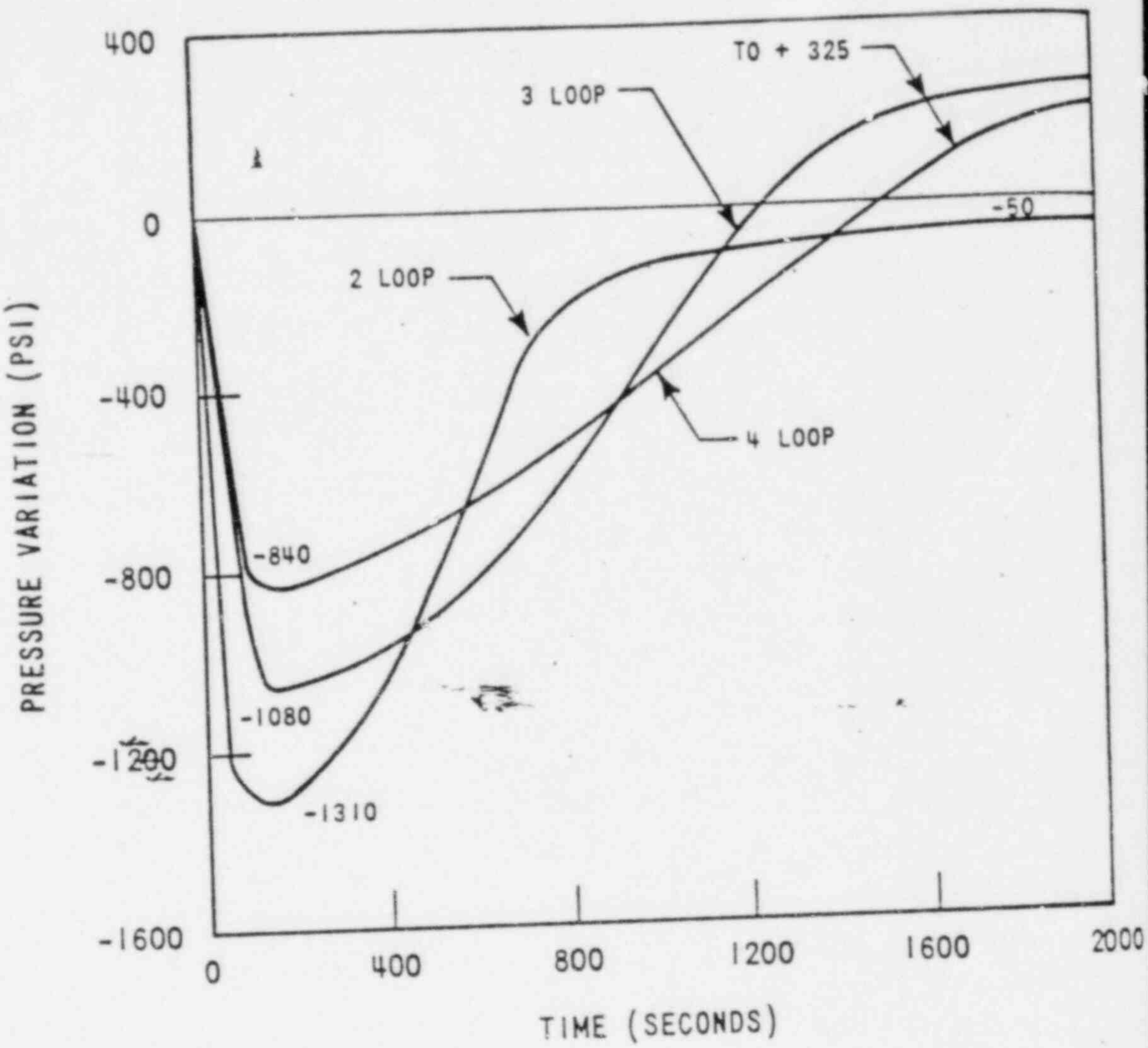


Figure 2.2.4-2A. Large Steam Line Break - Reactor Coolant Pressure Variation

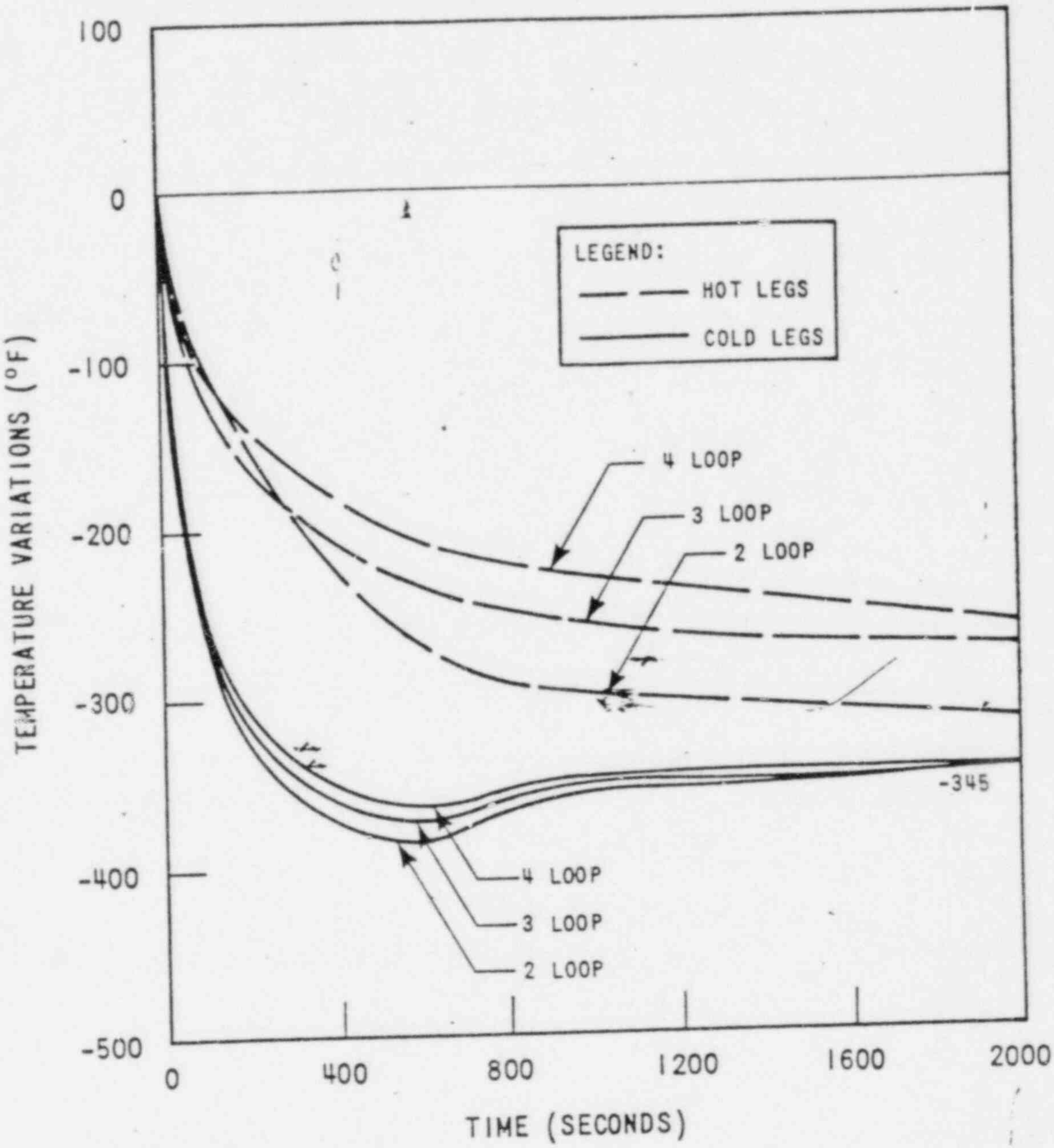


Figure 2.2.4-2B. Large Steam Line Break-Reactor Coolant Temperature Variation

FRACTURE MECHANICS METHODS

- FLAW ASSUMPTIONS
- MATERIAL PROPERTIES
- VESSEL HEAT TRANSFER
- THERMAL/STRESS/LEFM ANALYSIS

FLAW ASSUMPTIONS

- TYPICALLY 25 FLAW DEPTHS
- CONTINUOUS FLAW (2D)
- SEMI-ELLIPTICAL

  1/6 ASPECT RATIO TYPICAL
- LONGITUDINAL, CIRCUMFERENTIAL FLAWS
- IMPACT OF CLAD ON SMALL FLAWS

MATERIAL PROPERTIES

- ASME Section III Material Properties
- Carbon Steel Conductivity = 25.2 Btu/Hr-Ft-°F
- Stainless Steel Conductivity = 9.8 Btu/Hr-Ft-°F
- Carbon Steel Specific Heat = .128 Btu/Lb-°F
- Stainless Steel Specific Heat = .125 Btu/Lb-°F
- Temperature Dependent Fracture Toughness

VESSEL HEAT TRANSFER

ON THE VESSEL SURFACE

- Forced Convection (Turbulent)

$$h = \frac{K}{D}\ .023\ R^{.8}\ P^{.4}$$

(Dittus-Boelter)

- Free Convection (Turbulent)

$$h = \frac{K}{D}\ .021\ R^{.4}$$

- Boiling Heat Transfer

$$h_{FILM} = 200\ \text{Btu/Hr-Ft}^2\text{-}^{o}\text{F}$$

$$h_{NUCLEATE} = 10{,}000\ \text{Btu/Hr-Ft}^2\text{-}^{o}\text{F}$$

- Thermal Analysis Assumes Existence of Cladding

- Transient Impact on Heat Transfer

THERMAL AND STRESS ANALYSIS

- THERMAL ANALYSIS
  - FINITE DIFFERENCE CALCULATION
  - CLAD ON INNER VESSEL SURFACE
  - ADIABATIC OUTER VESSEL SURFACE
  - DETERMINE THROUGH WALL TEMPERATURE DISTRIBUTION
  - CALCULATION PERFORMED 100 SECOND INTERVALS FOR ~ 6000 SECONDS

- STRESS ANALYSIS
  - THERMAL STRESS FOR HOLLOW CYLINDER
  - PRESSURE STRESS IN A CYLINDRICAL VESSEL
  - DETERMINED AS A FUNCTION OF WALL THICKNESS
  - EXCLUDES CONSIDERATION OF THE CLAD

- FRACTURE MECHANICS
  - LINEAR ELASTIC FRACTURE MECHANICS
  - CONSIDERS TEMPERATURE, STRESS AND FLUENCE DISTRIBUTION THROUGH THE WALL THICKNESS
  - CALCULATES K_I, K_{IC}, K_{IA} EVERY 100 SECONDS
  - DETERMINES MINIMUM AND MAXIMUM FLAW INITIATION DEPTHS
  - DETERMINES MAXIMUM FLAW ARRENT DEPTH

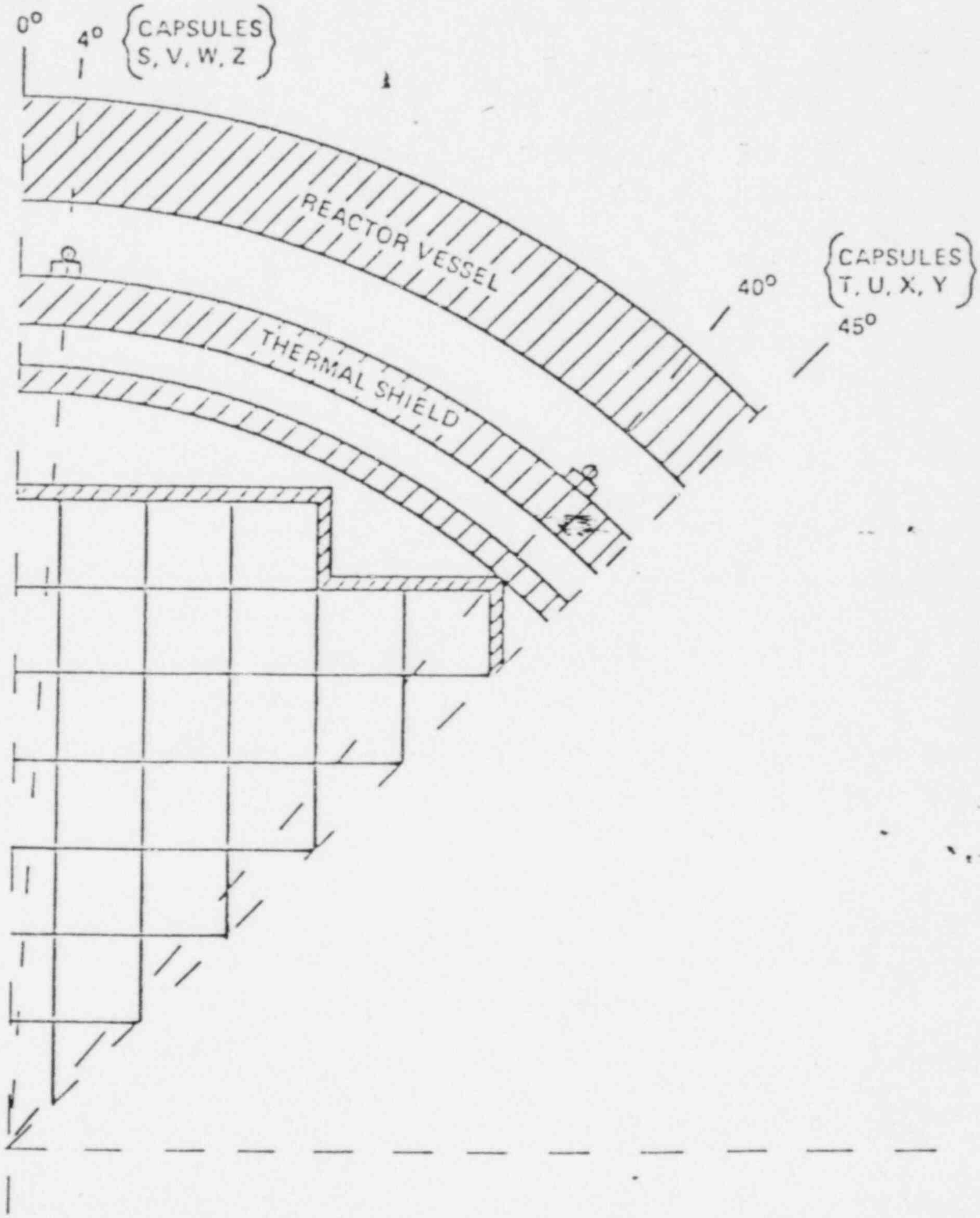
FUEL ASSEMBLY REMOVAL

- REDUCTION OF FLUENCE THROUGH REMOVAL OF PERIPHERAL ASSEMBLIES

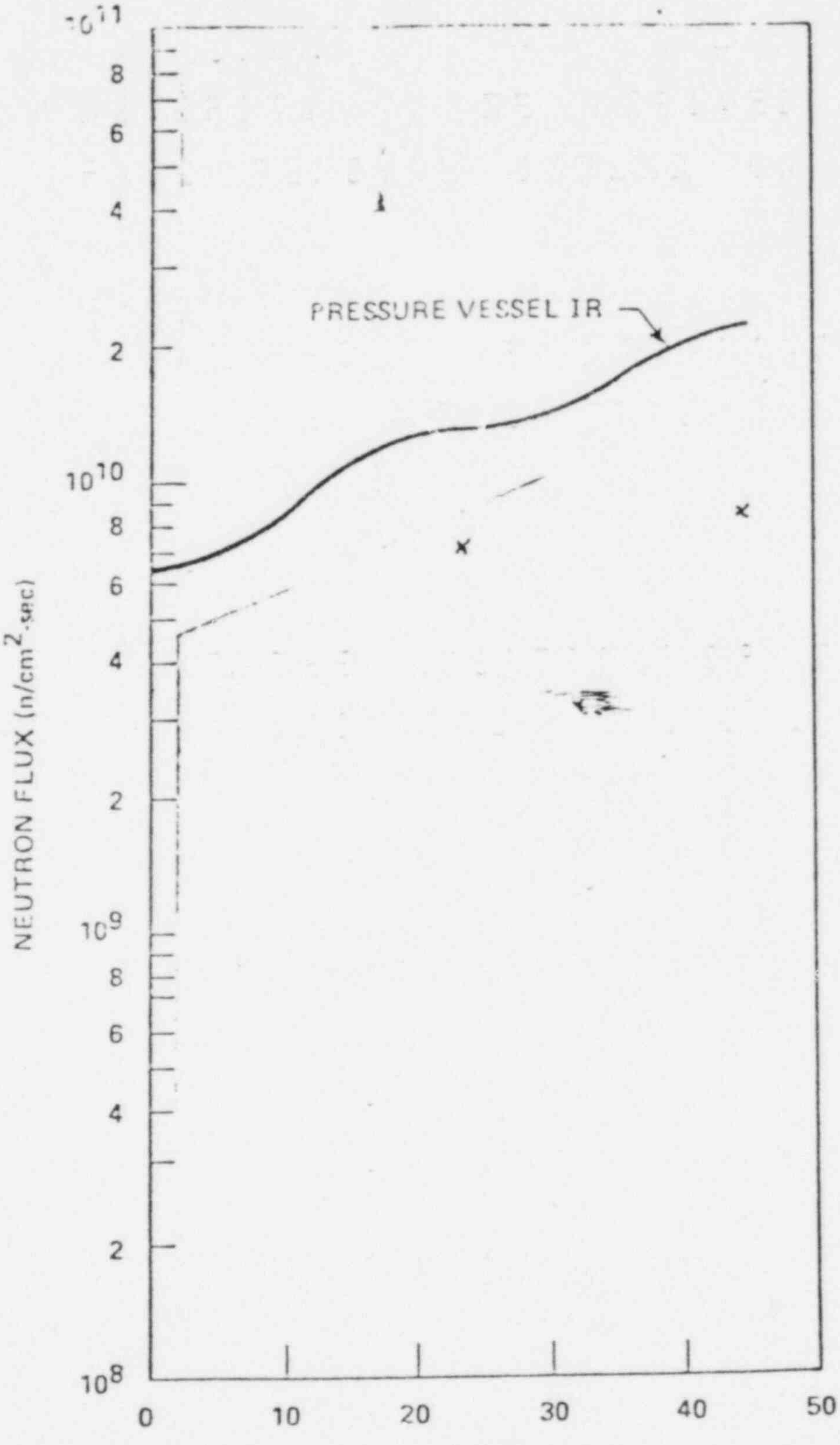
  - 2-LOOP - 12 ASSEMBLIES

  - 3-LOOP - 12 ASSEMBLIES

  - 4-LOOP - 16 ASSEMBLIES



0°
4°
CAPSULES S, V, W, Z
REACTOR VESSEL
THERMAL SHIELD
40°
45°
CAPSULES T, U, X, Y



Calculated Azimuthal Distribution of Maximum Fast Neutron Flux (E > 1.0 Mev)

- REMOVAL OF A PERIPHERAL FUEL ASSEMBLY MAY RESULT IN A FACTOR OF 2-3 REDUCTION IN FAST NEUTRON FLUX IN AREAS OF THE PRESSURE VESSEL WHICH ARE ADJACENT TO THE CORE VACANCY

- FAST NEUTRON FLUX IN AREAS OF THE VESSEL AWAY FROM THE CORE PERTURBATION MAY INCREASE SLIGHTLY IF CONSTANT REACTOR POWER IS MAINTAINED

- BENEFITS TO BE DERIVED FROM FUEL REMOVAL DEPEND ON

  1. THE TIME IN PLANT LIFE WHEN THE CORE CHANGE IS MADE

  2. THE LOCATION OF THE MOST CRITICAL MATERIAL IN THE REACTOR VESSEL BELTLINE

- ACTUAL BENEFITS MUST BE ESTABLISHED ON A CASE BY CASE BASIS

CORE MARGIN PENALTY

- CORE AVERAGE KW/FT INCREASES
  - 2-LOOP - 10%
  - 3-LOOP - 8%
  - 4-LOOP - 8%

- LOCA LIMIT DECREASES
  - 2-LOOP - 10%
  - 3-LOOP - 8%
  - 4-LOOP - 8%

- POTENTIAL RADIAL PEAKING PENALTY

ECONOMICS PENALTIES

- ~700 MWD/MTU LOSS IN CYCLE LIFETIME

- ~2.5% INCREASE IN FUEL CYCLE COST

LOW LEAKAGE PATTERN OPTION

- No Change in Core Average KW/FT

- No LOCA limit margin reduction with the use of burnable poisons

- Fuel Cycle Costs unchanged or potential benefit

Enclosure 6

CE Owners Group Presentation

PRESSURIZED THERMAL SHOCK

TRANSIENTS ANALYSES

MATERIAL PROPERTIES

FLUENCE DISTRIBUTIONS

VESSEL INTEGRITY EVALUATIONS

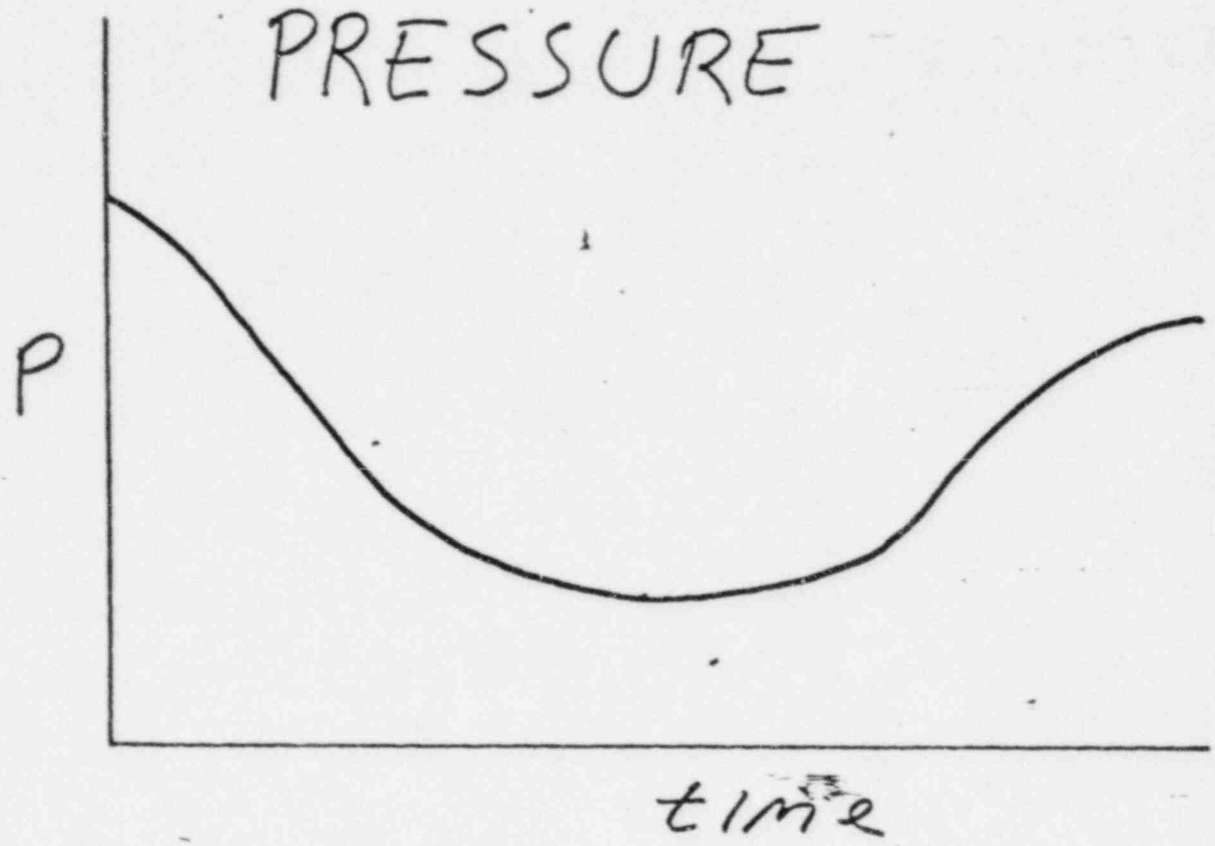
RV THERMAL SHOCK WITH REPRESSURIZATION

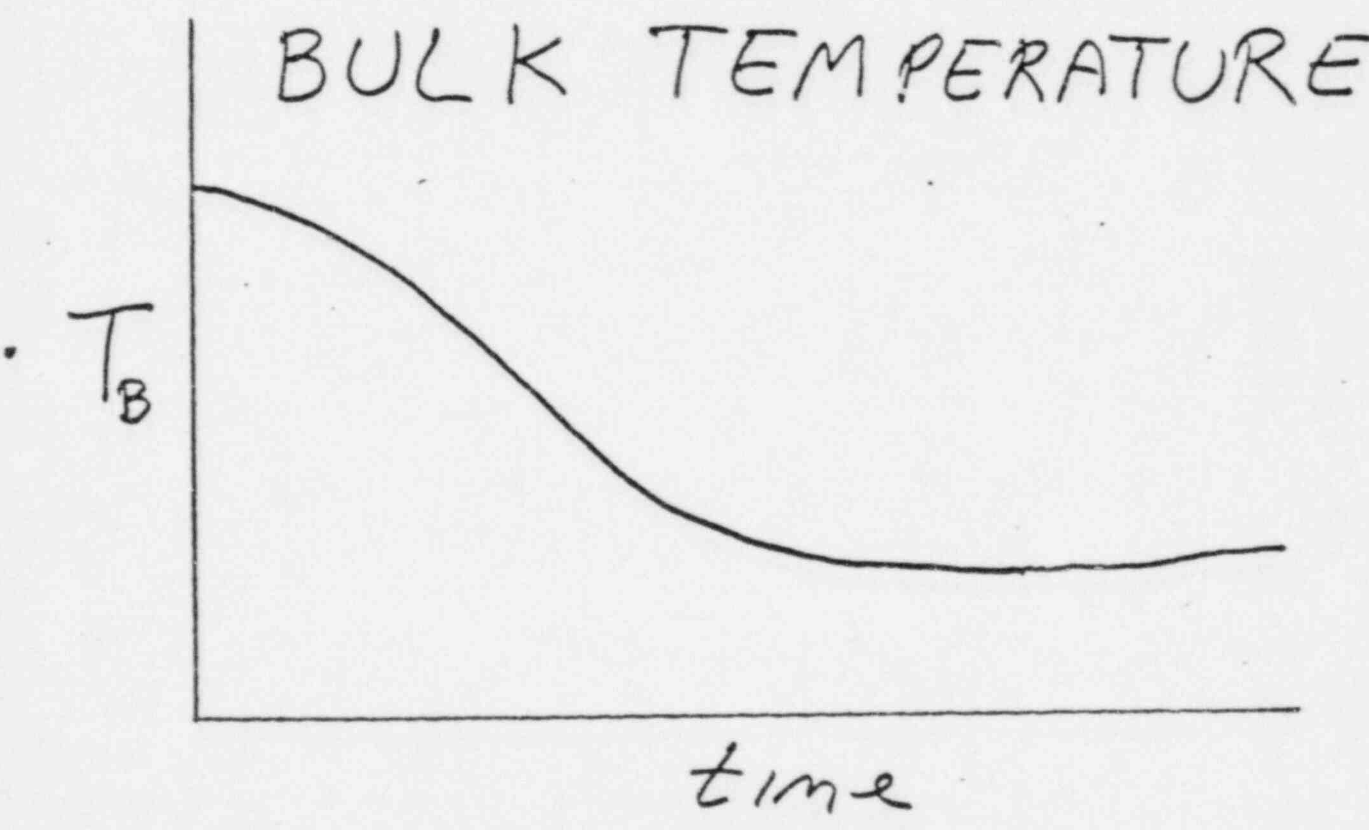
OVERVIEW OF SCOPING ANALYSIS METHODS

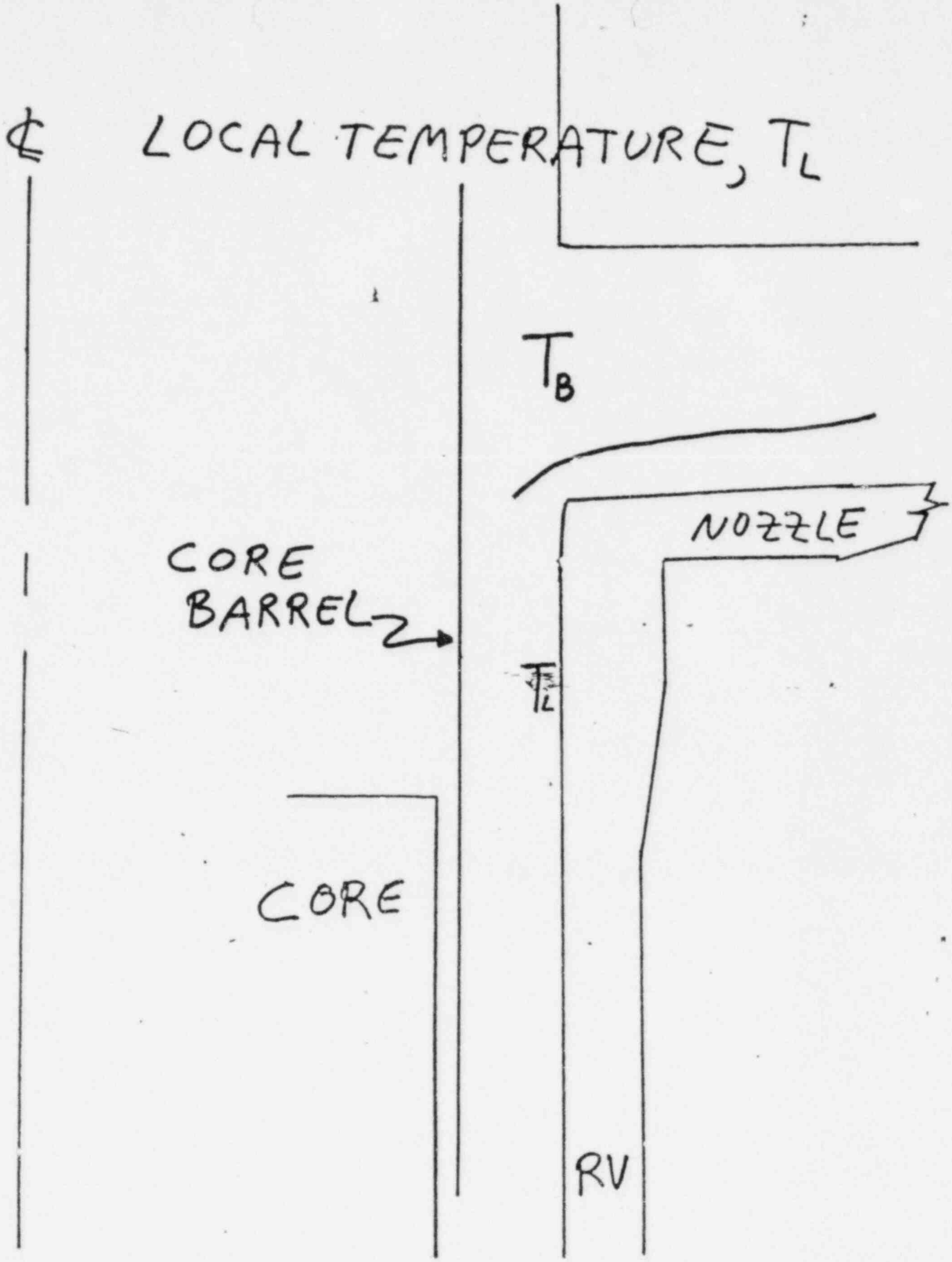
AND SOME PRELIMINARY RESULTS

D. J. AYRES

JULY 30 1981

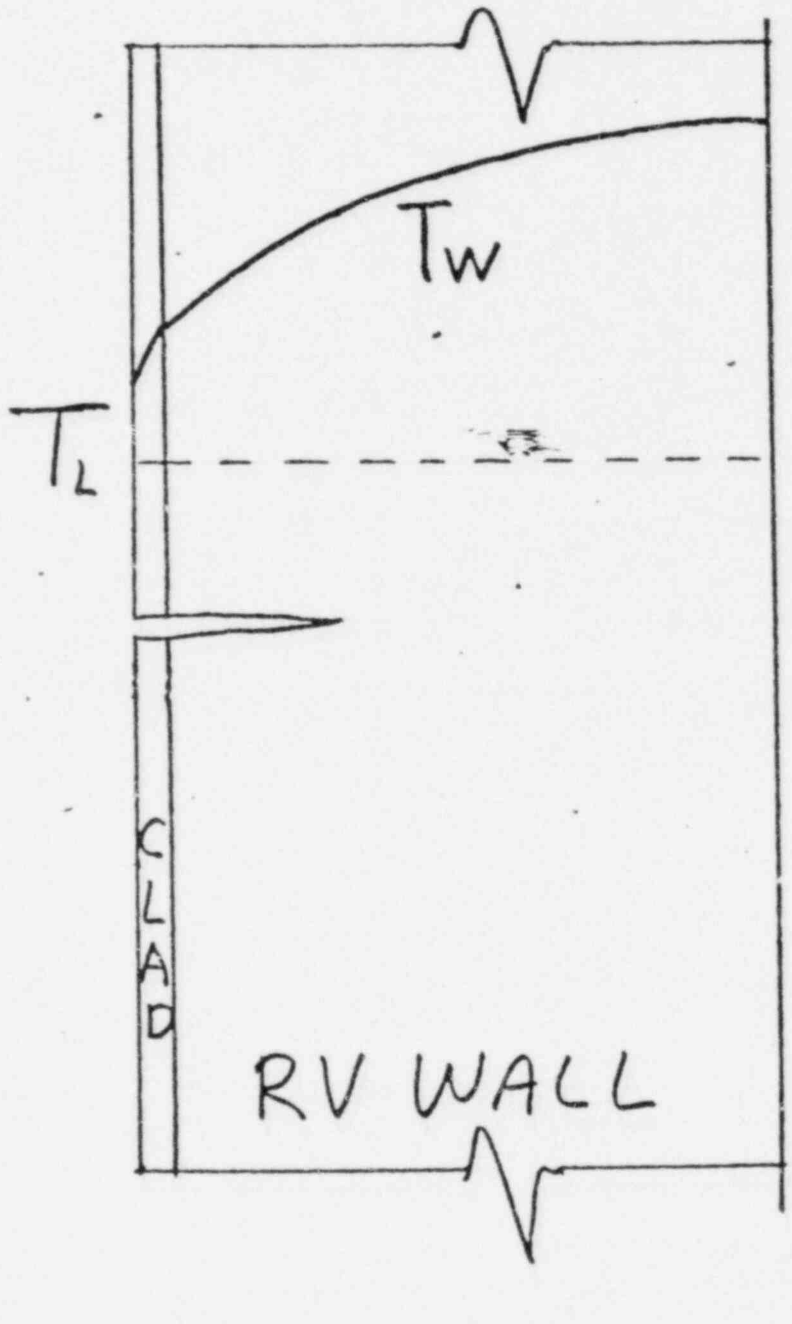

PRESSURE
P
time


BULK TEMPERATURE
T_B
time

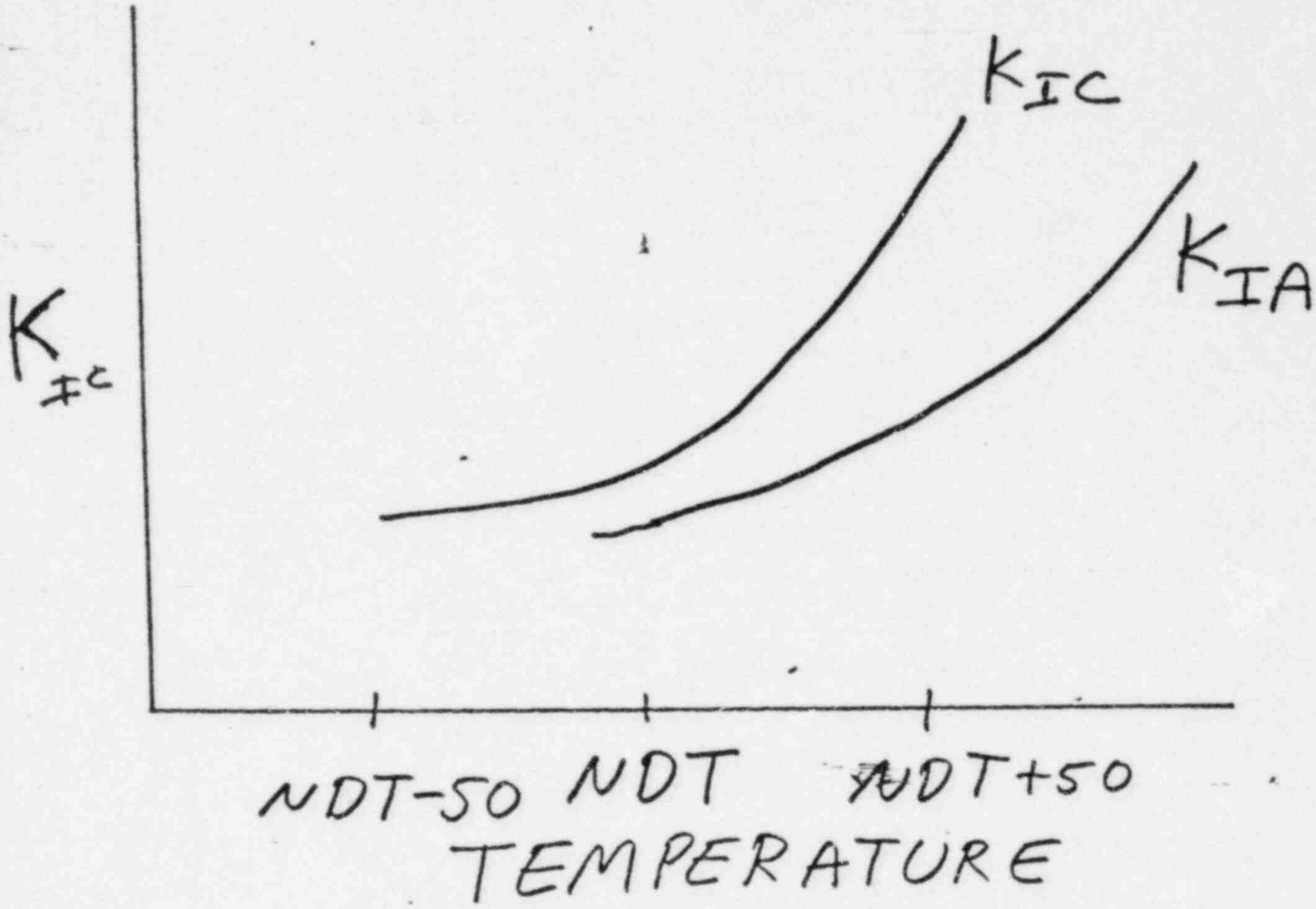


LOCAL TEMPERATURE, T_L
T_B
NOZZLE
CORE BARREL
T_L
CORE
RV

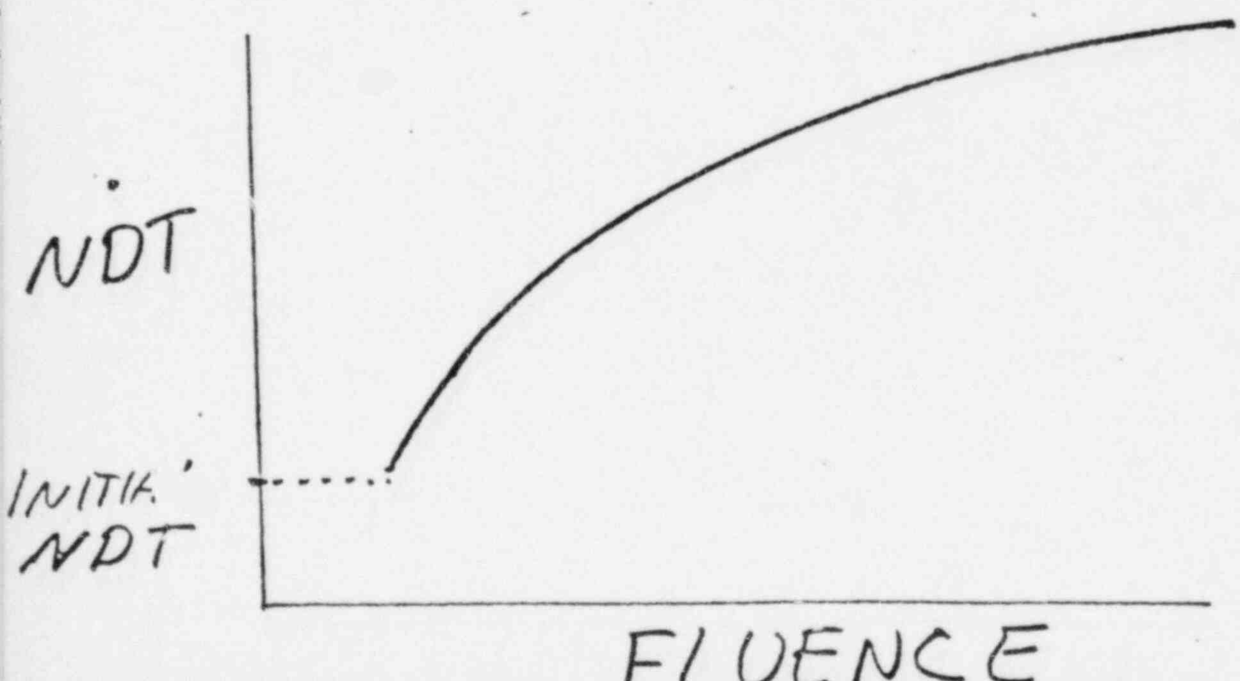
LOCAL HEAT TRANSFER



TOUGHNESS

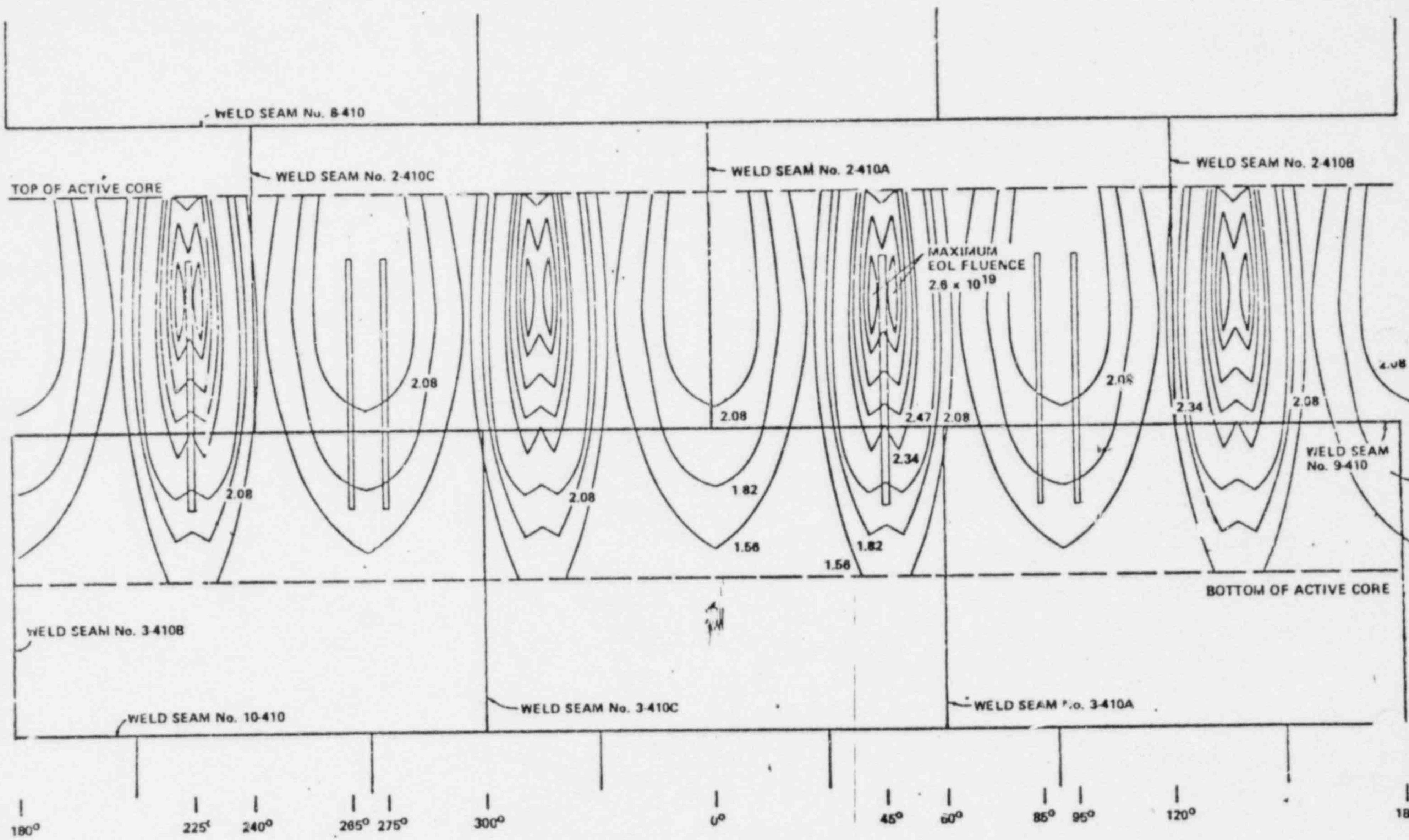


K_{IC}
K_{IA}
K_{IC}
NDT-50
NDT
NDT+50
TEMPERATURE



NDT
INITIAL NDT
FLUENCE

SUPERIMPOSED FLUENCE AND WELD MAPS



LINES OF ISOFLUENCE, E>1MEV (x 10^{19} n/cm^2)

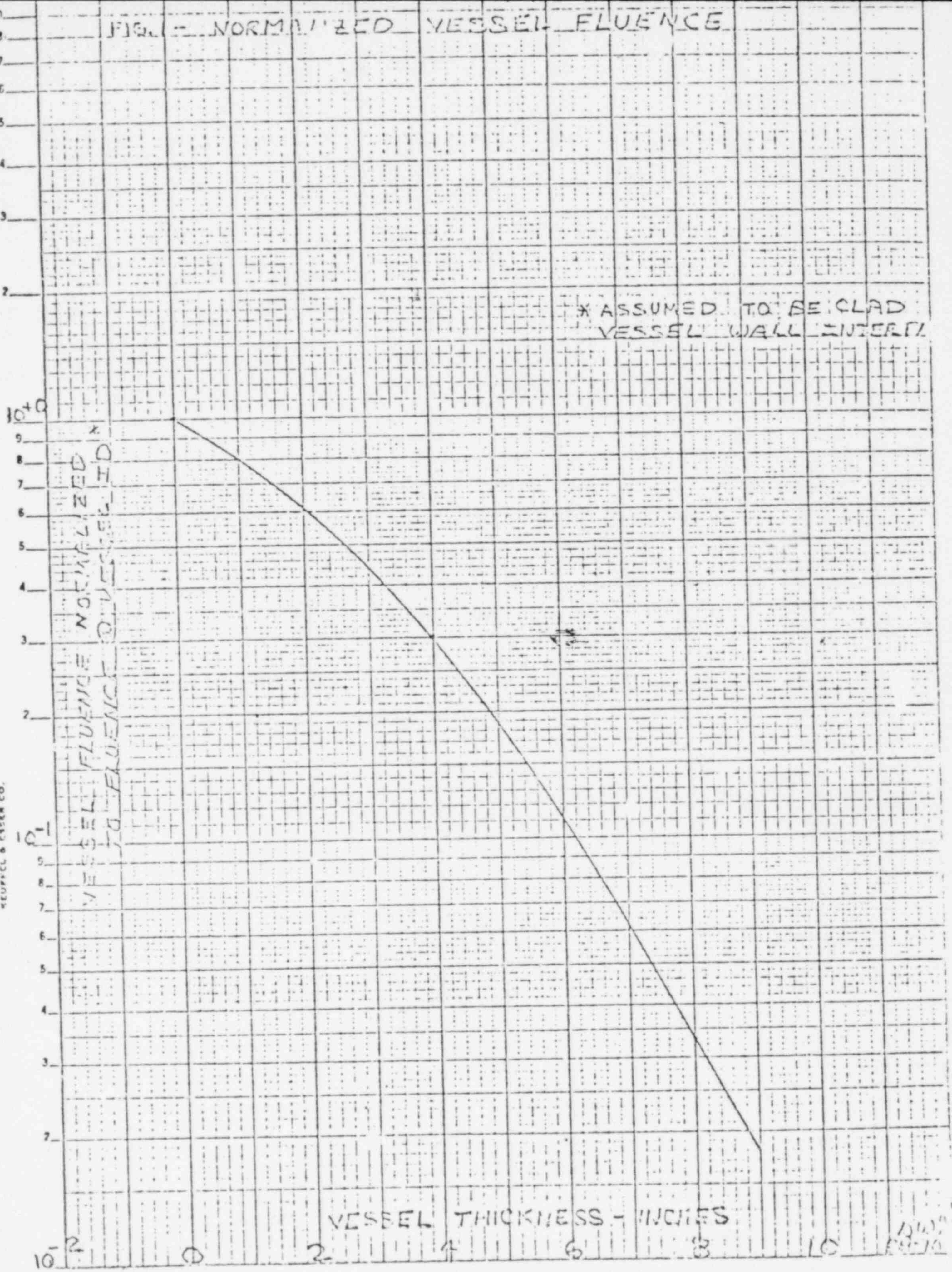


FIG. - NORMALIZED VESSEL FLUENCE
* ASSUMED TO BE CLAD VESSEL WALL
NORMALIZED VESSEL FLUENCE* TO FLUENCE @ VESSEL I.D.
VESSEL THICKNESS - INCHES
10^{+0}
10^{-1}
10^{-2}
0
2
4
6
8
10

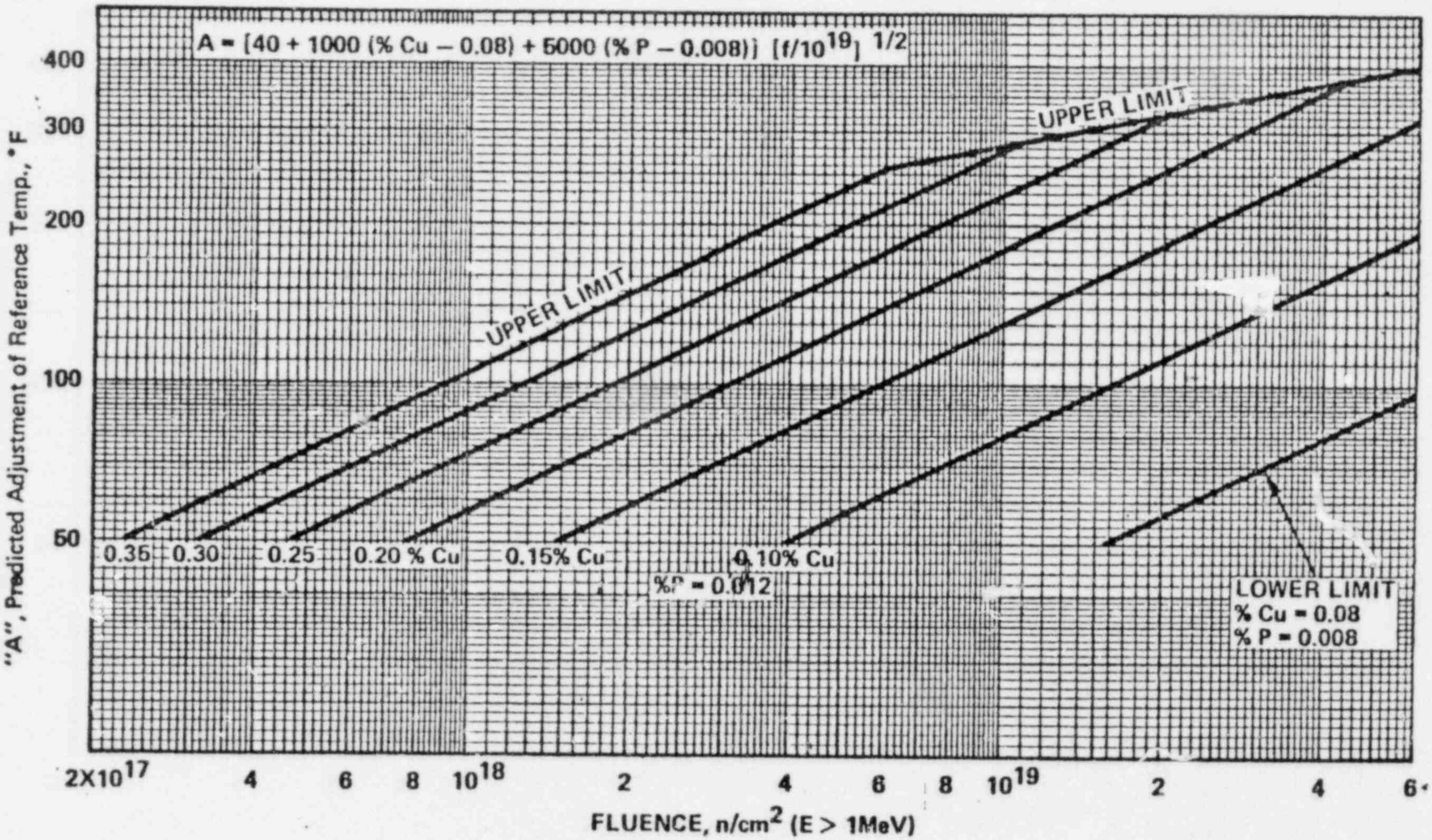
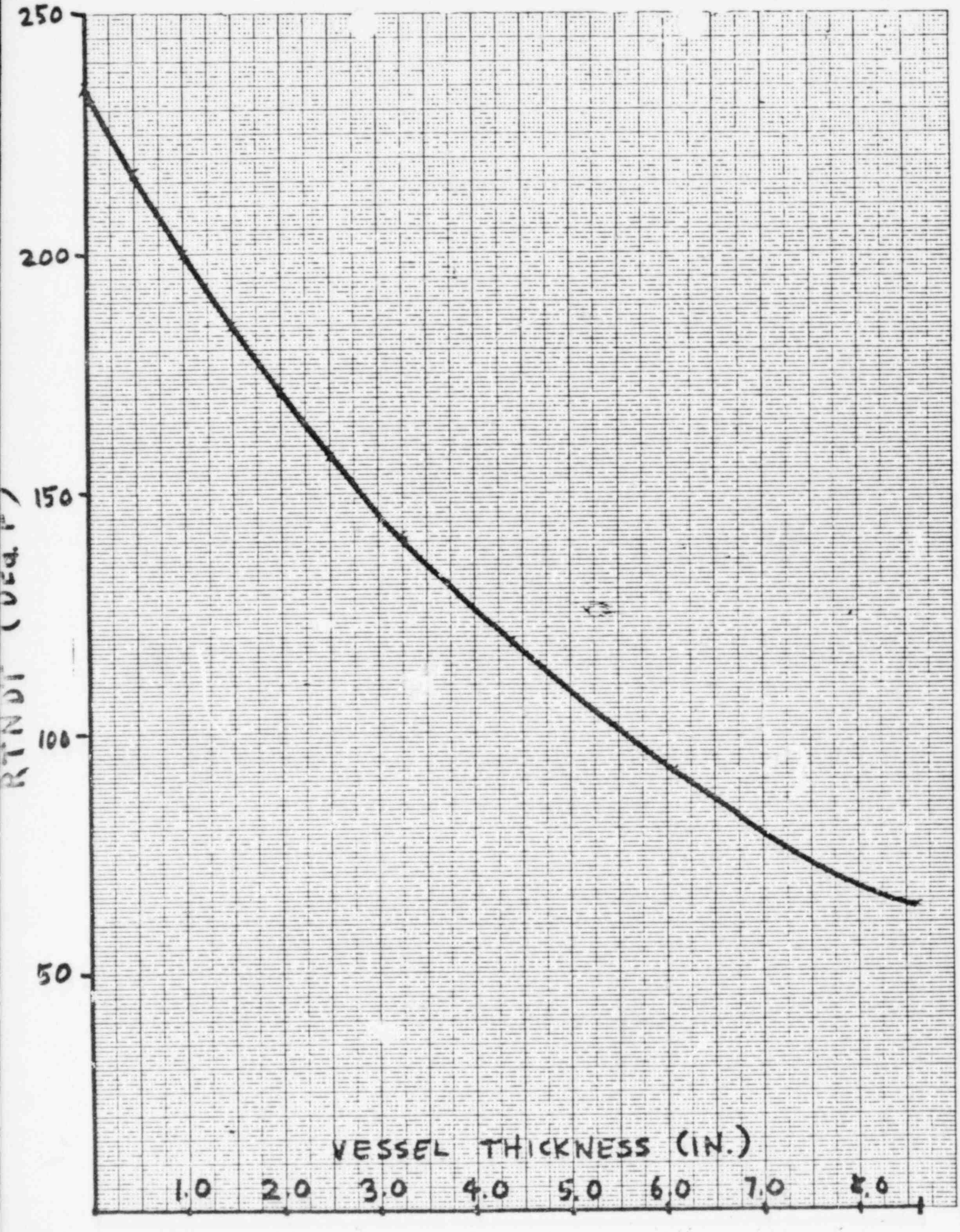
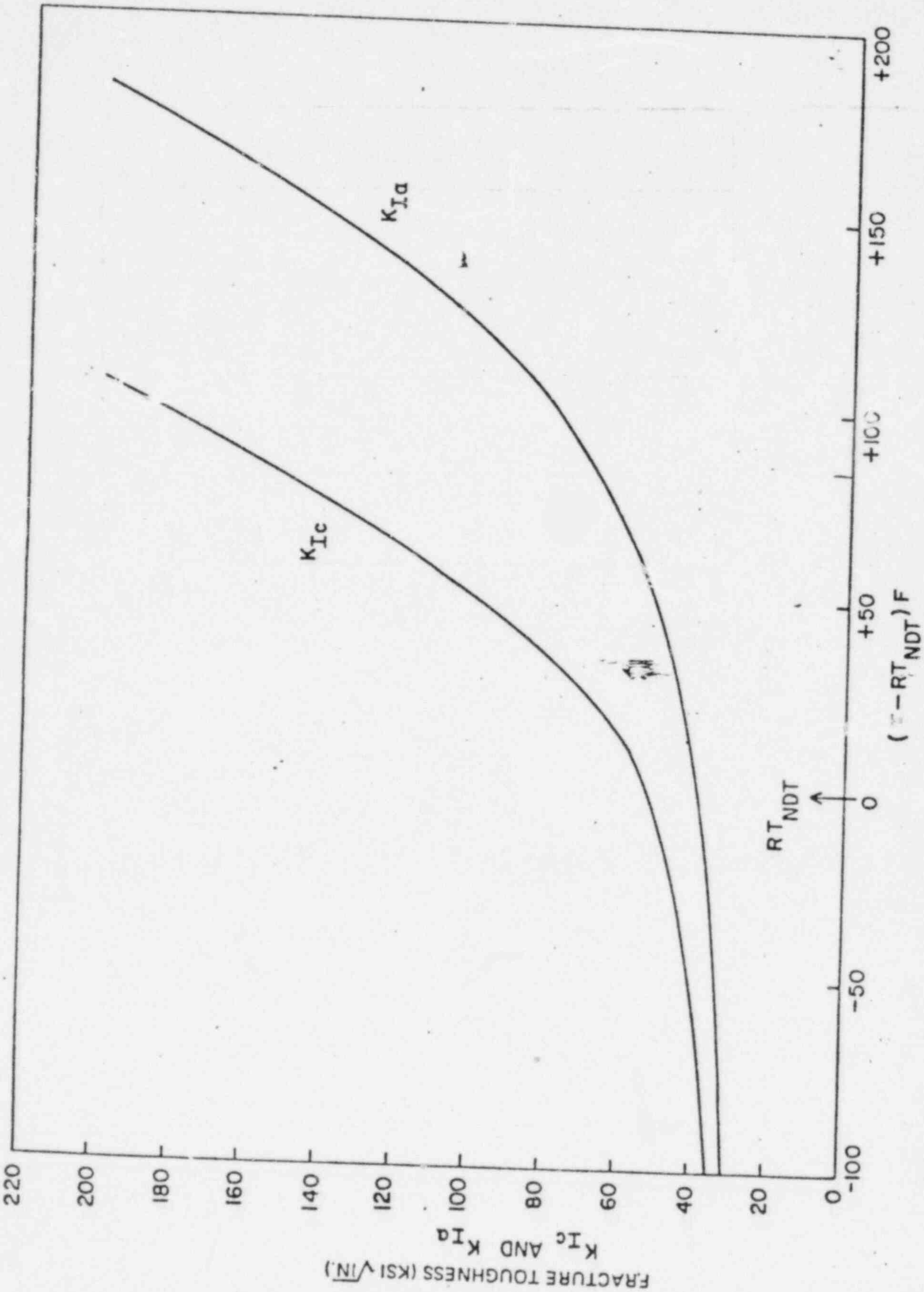
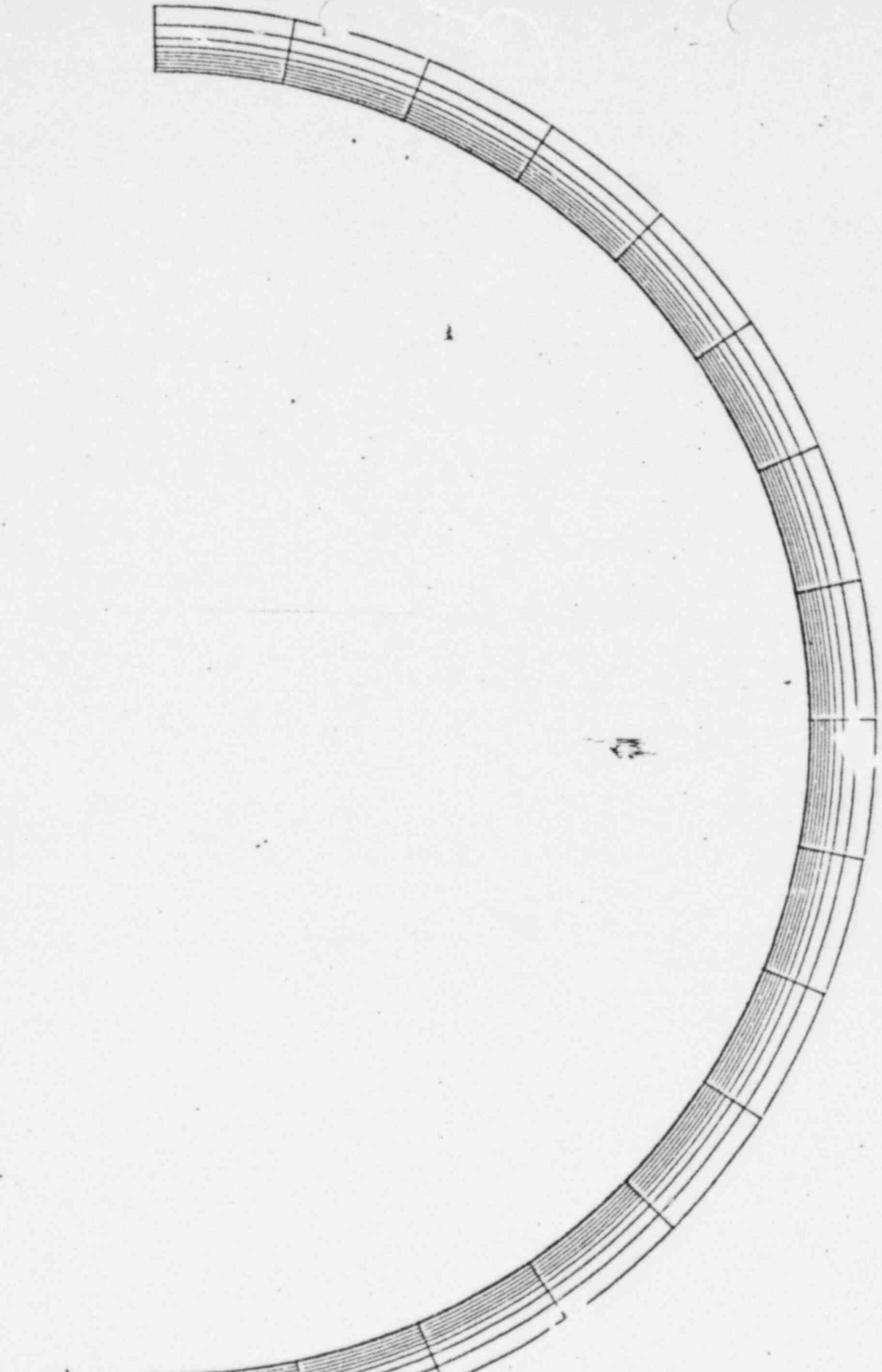


Figure 1 Predicted Adjustment of Reference Temperature, "A", as a Function of Fluence and Copper Content.
For Copper and Phosphorus Contents Other Than Those Plotted, Use the Expression for "A" Given on the Figure.

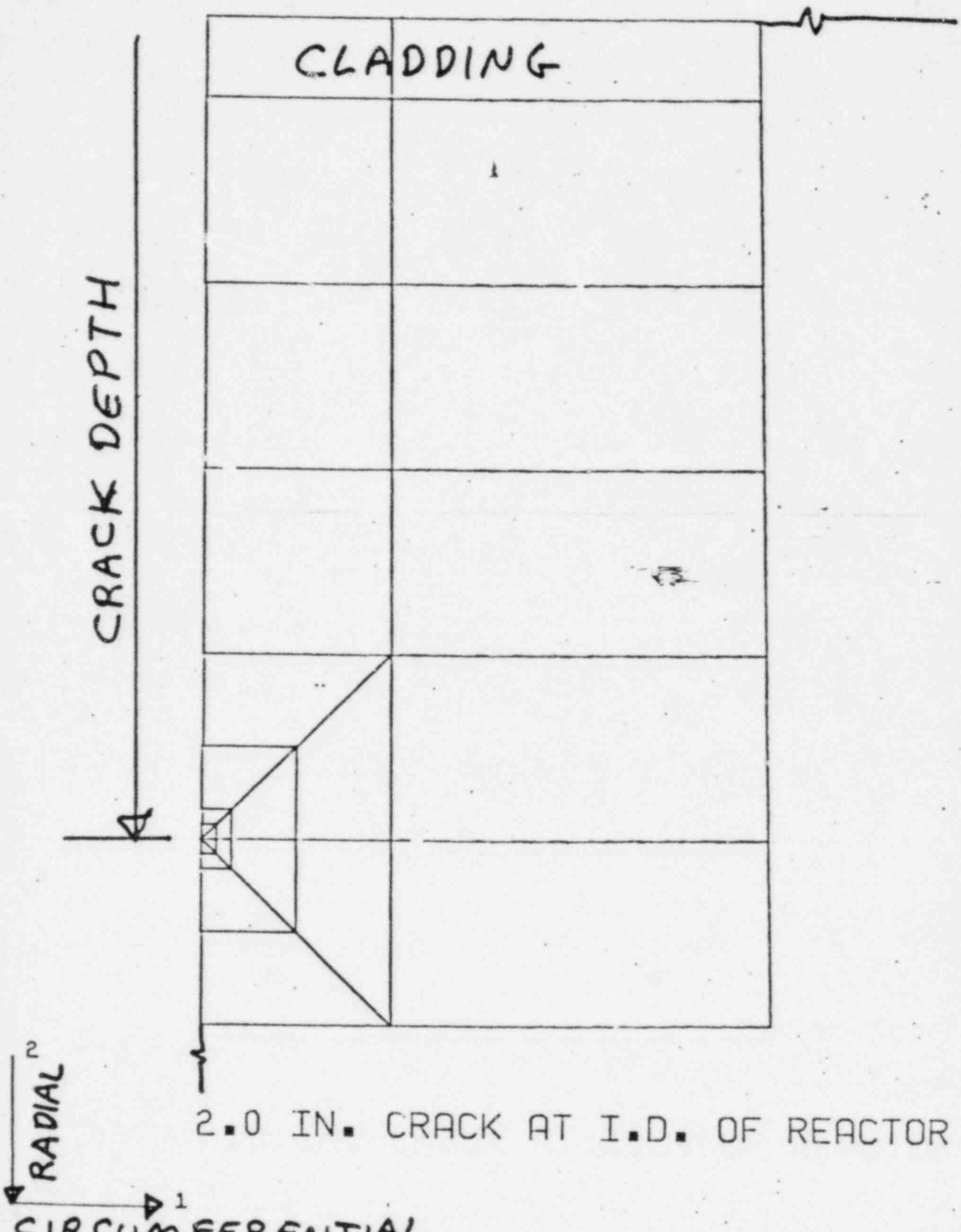


250
200
150
100
50
RTNDT (DEG F)
VESSEL THICKNESS (IN.)
1.0
2.0
3.0
4.0
5.0
6.0
7.0
8.0


FRACTURE TOUGHNESS (KSI √IN.)
K_{Ic} AND K_{Ia}
K_{Ia}
K_{Ic}
RT_{NDT}
$(T - RT_{NDT})$ F
0
20
40
60
80
100
120
140
160
180
200
220
-100
-50
0
+50
+100
+150
+200

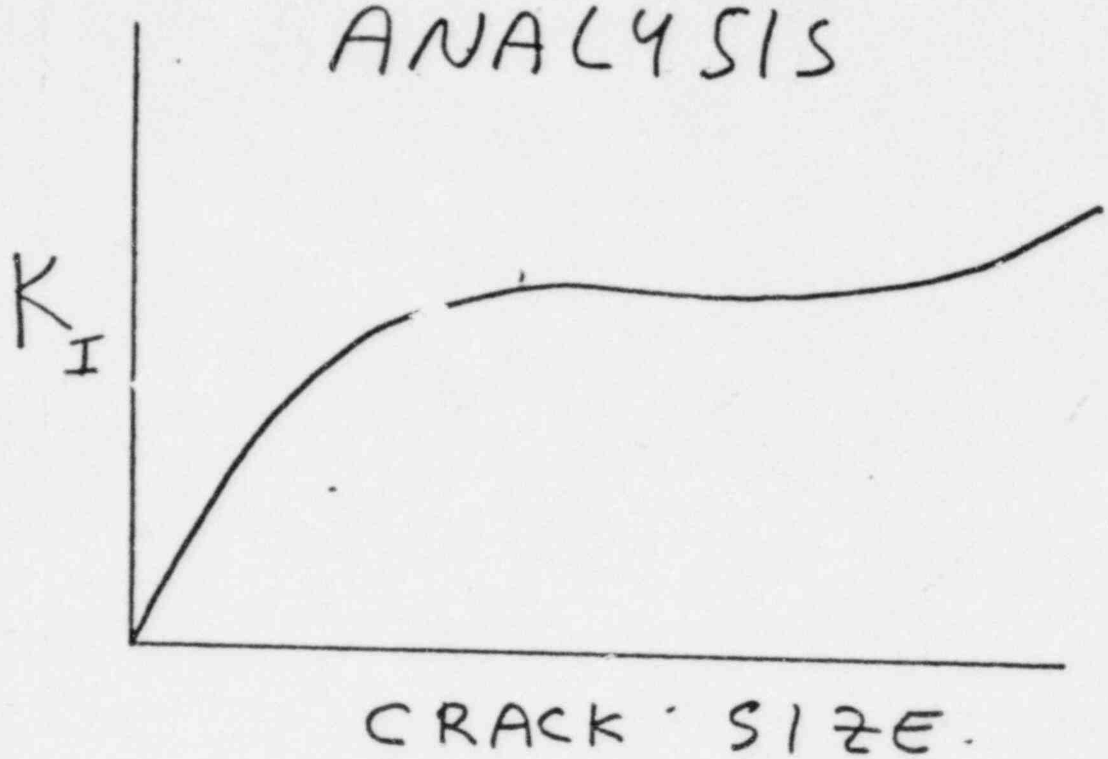


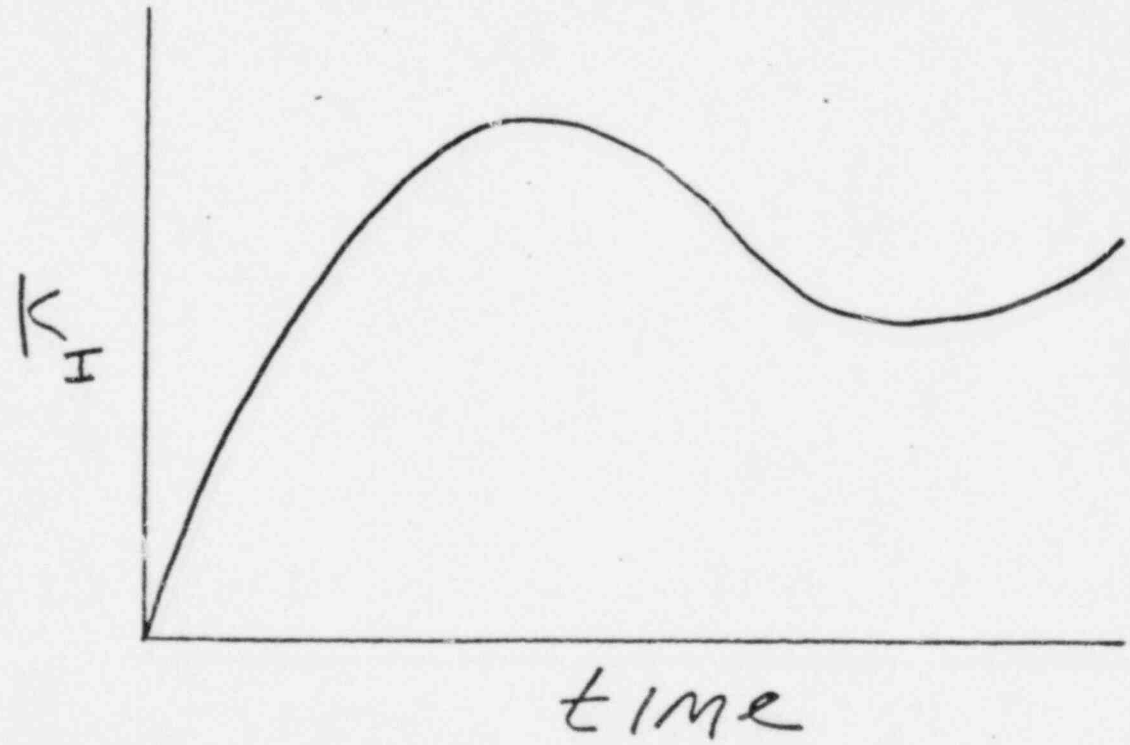
REACTOR VESSEL WITH AXIAL CRACK



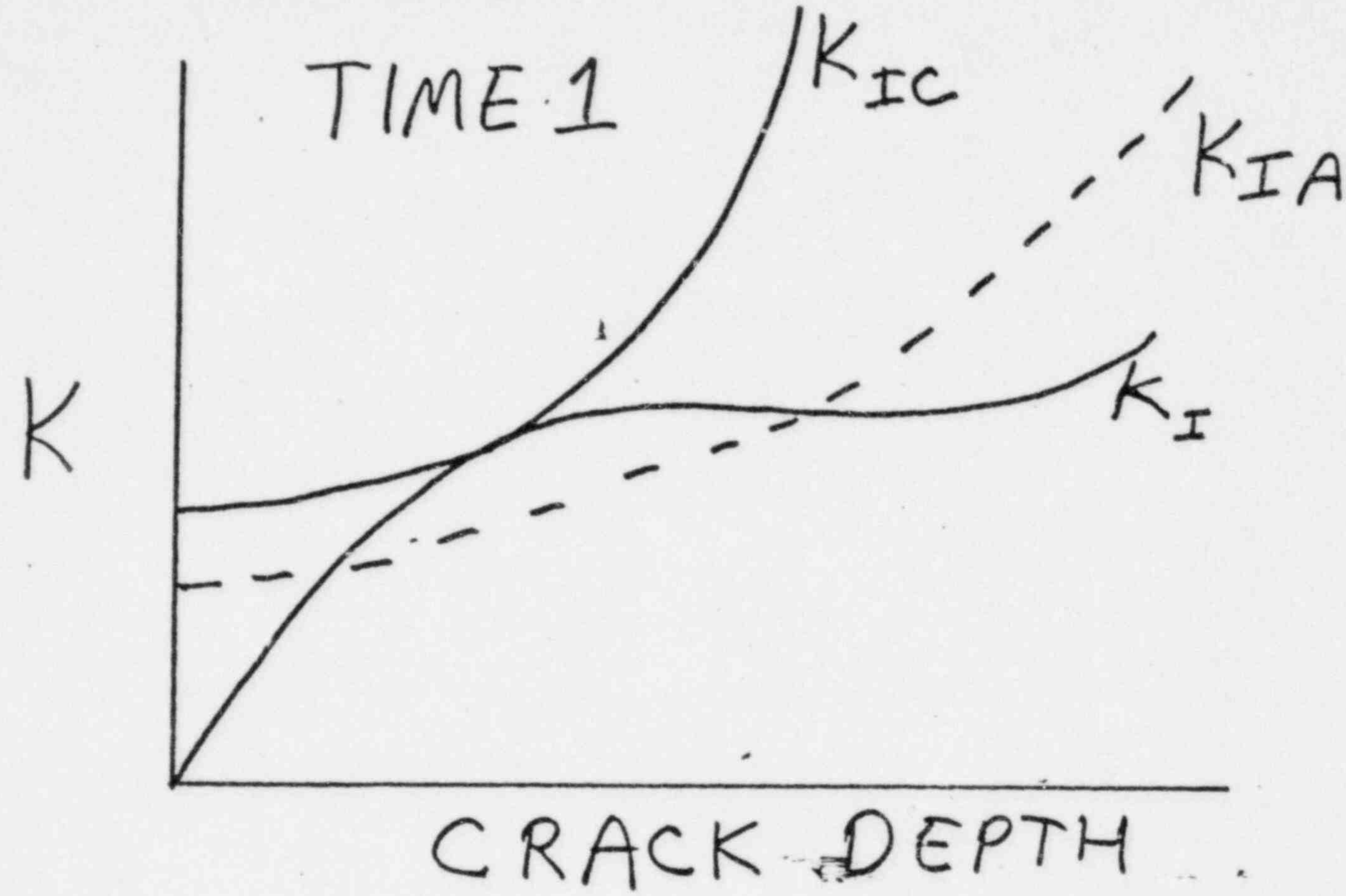
2.0 IN. CRACK AT I.D. OF REACTOR VESSEL

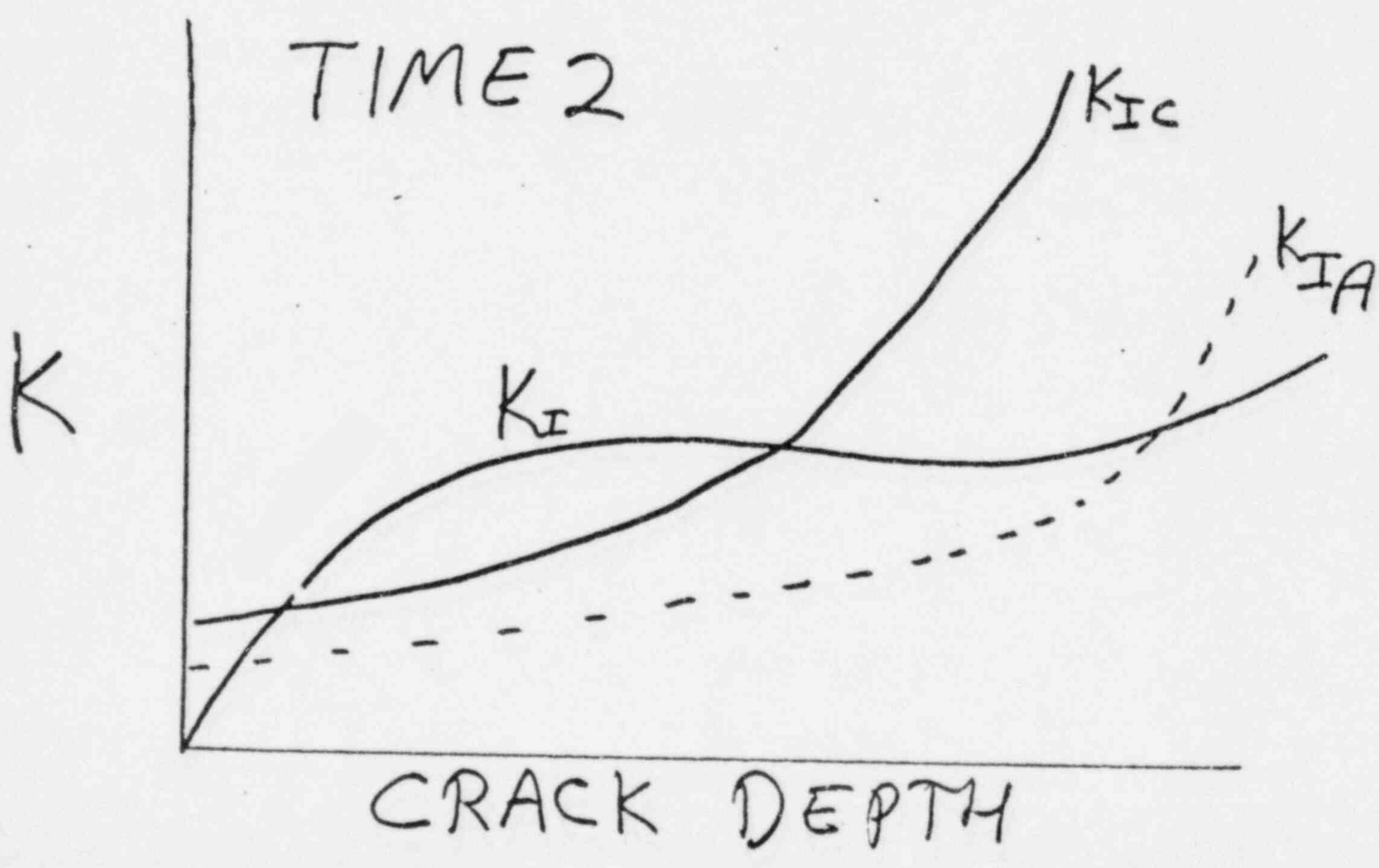
FRACTURE MECHANICS ANALYSIS



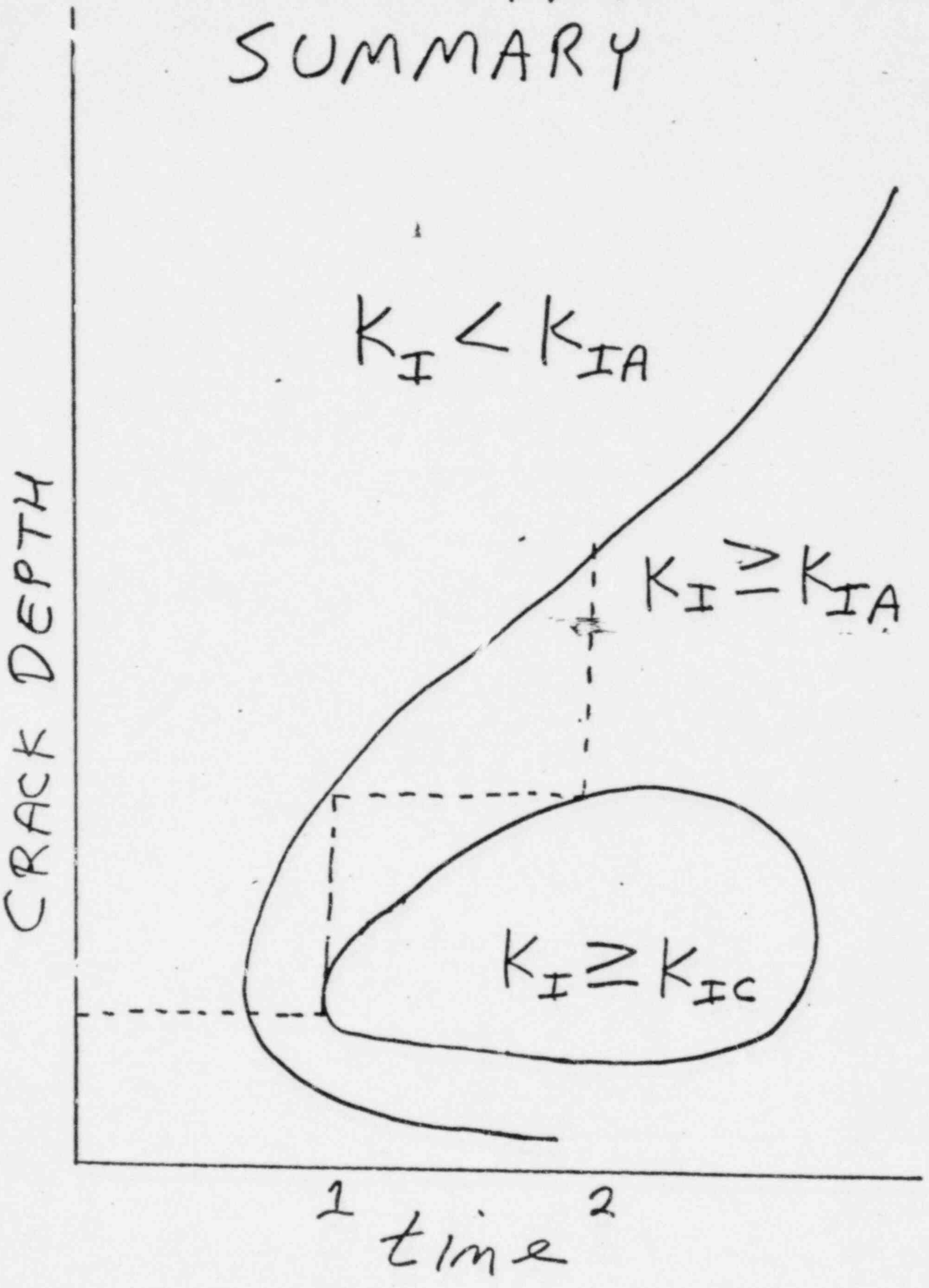


EVALUATION

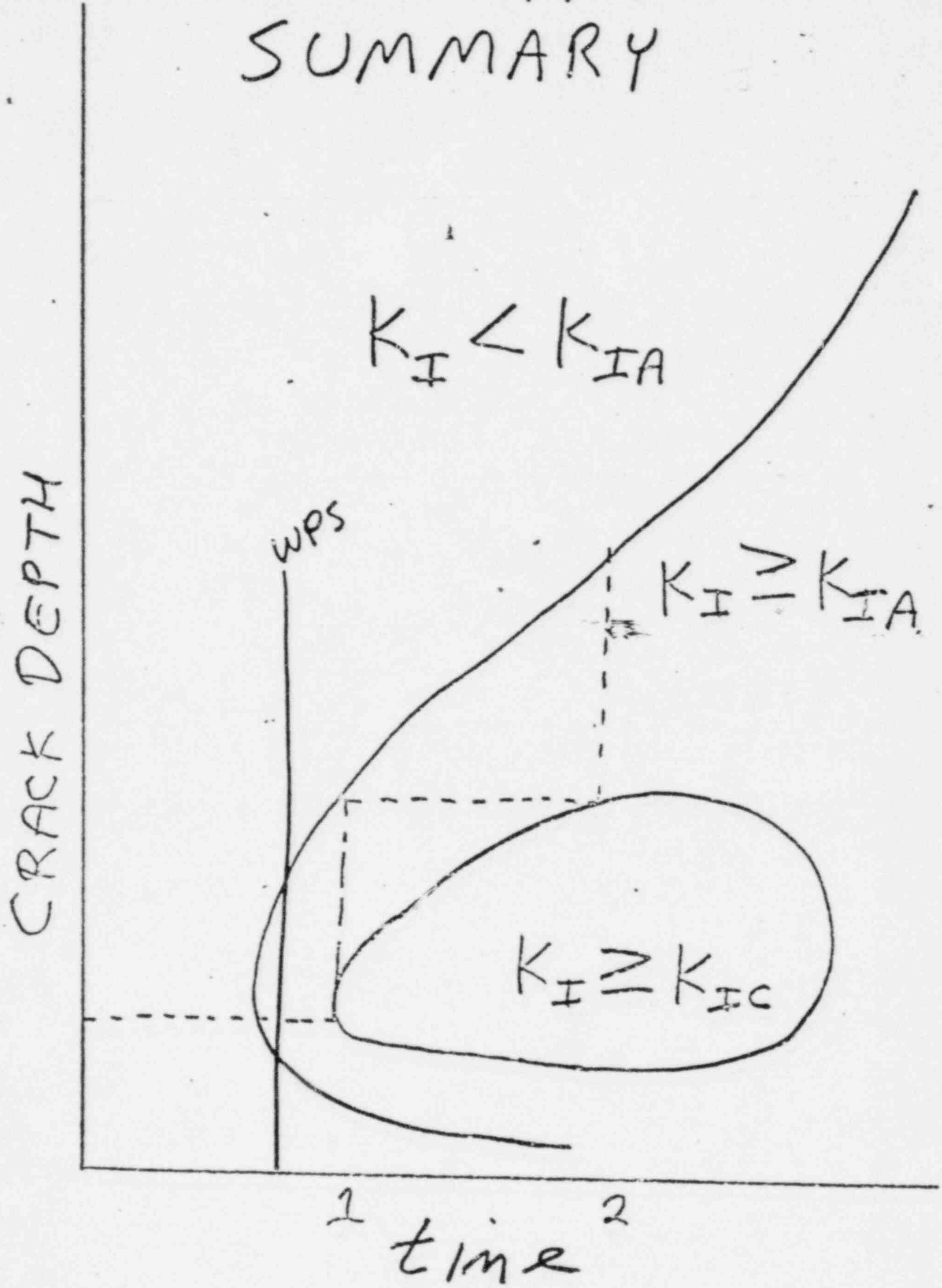




EVALUATION SUMMARY



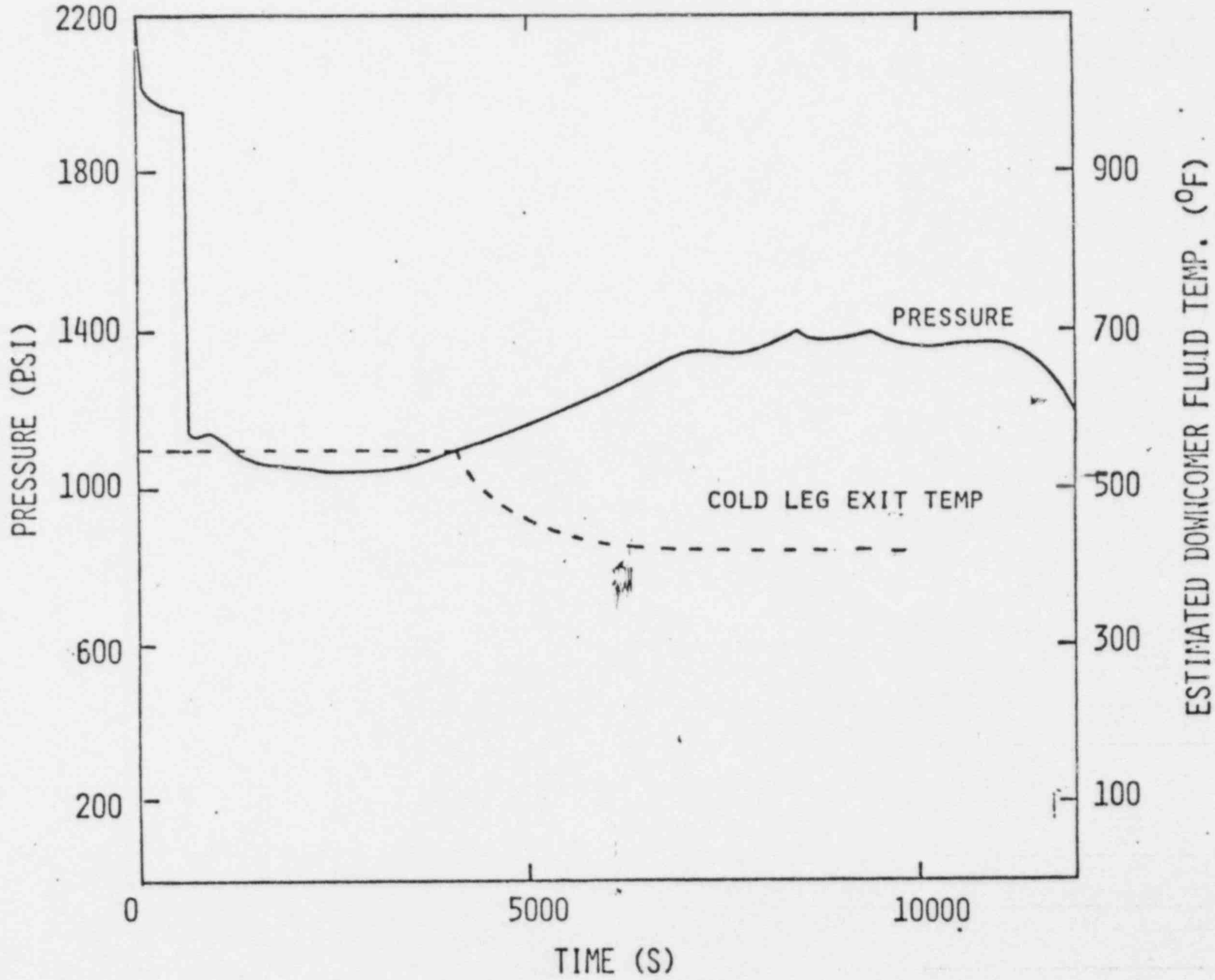
EVALUATION SUMMARY

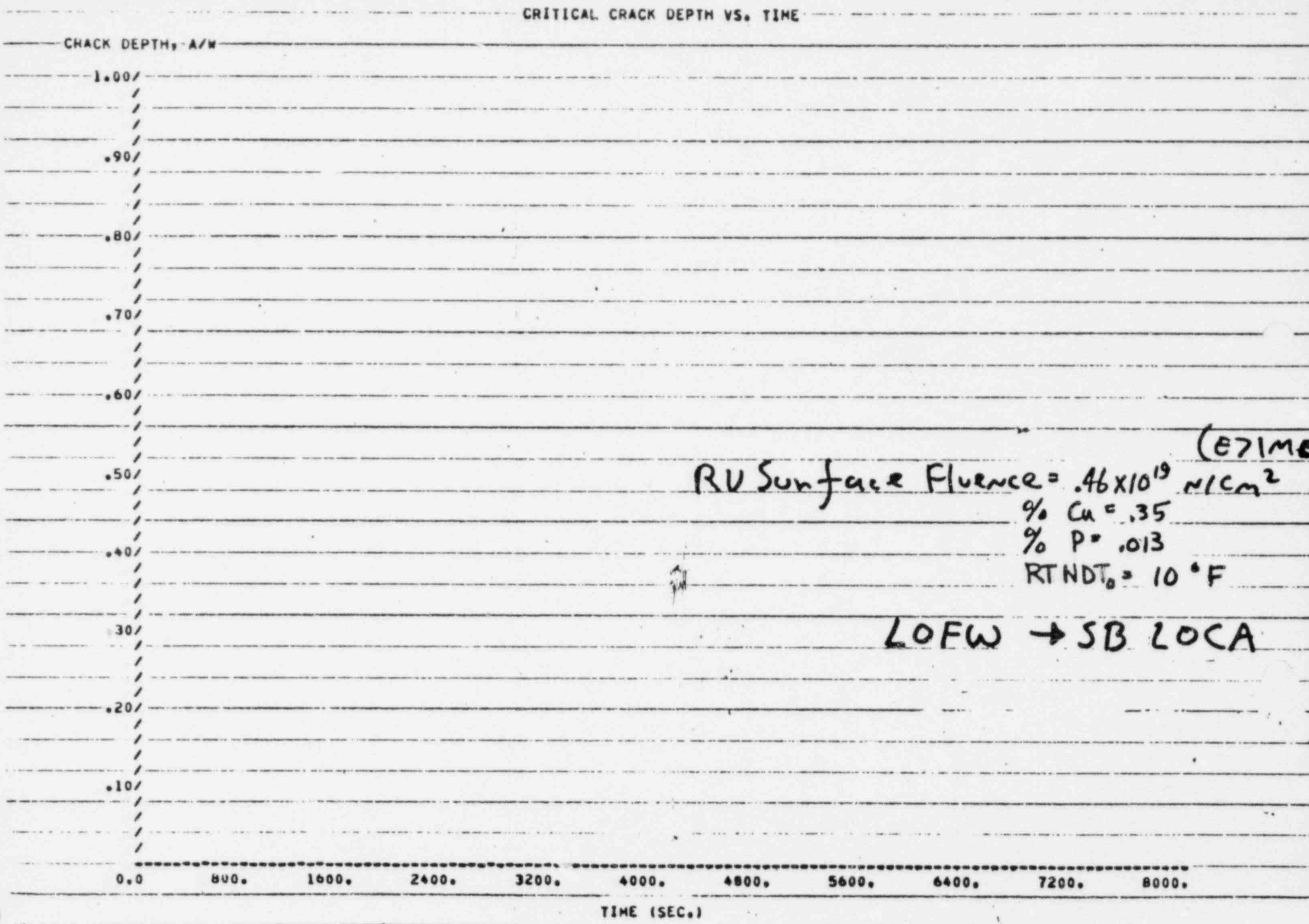


CONSERVATIVE ASSUMPTIONS
IN
FRACTURE EVALUATION

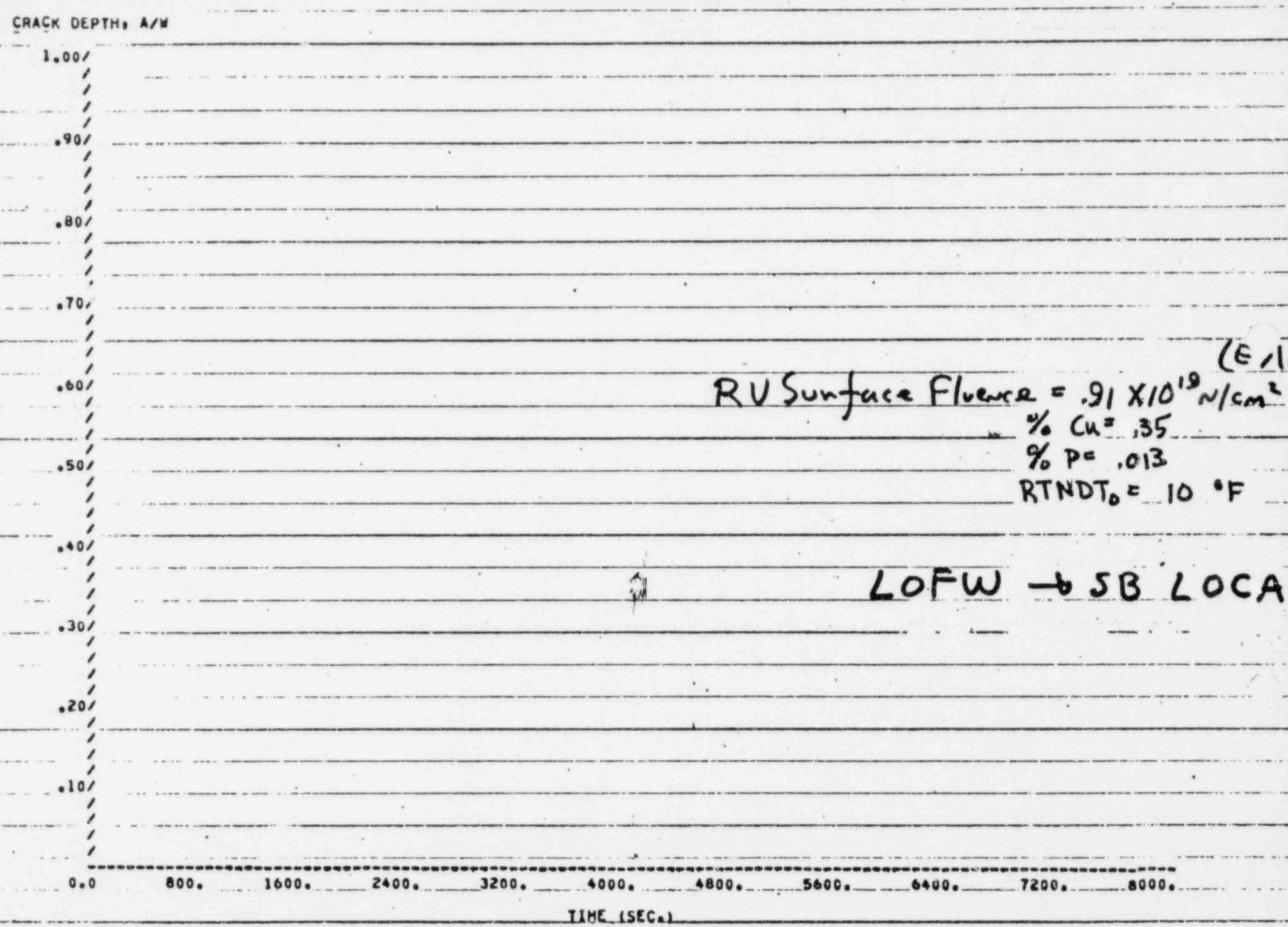
1. LONG AXIAL CRACKS
2. WELD WITH HIGHEST NDT
3. AXIAL PEAK FLUENCE
4. NO FLUID HEATING FROM WALL
5. FULL PRESSURE IN CRACK
6. DIFFERENTIAL THERMAL EXPANSION OF CLAD
7. REG. GUIDE 1.99 NDT SHIFT
8. LINEARIZED FLUENCE ATTENUATION

REFERENCE
OVERCOOLING TRANSIENT
LOFW ——▶ SB LOCA



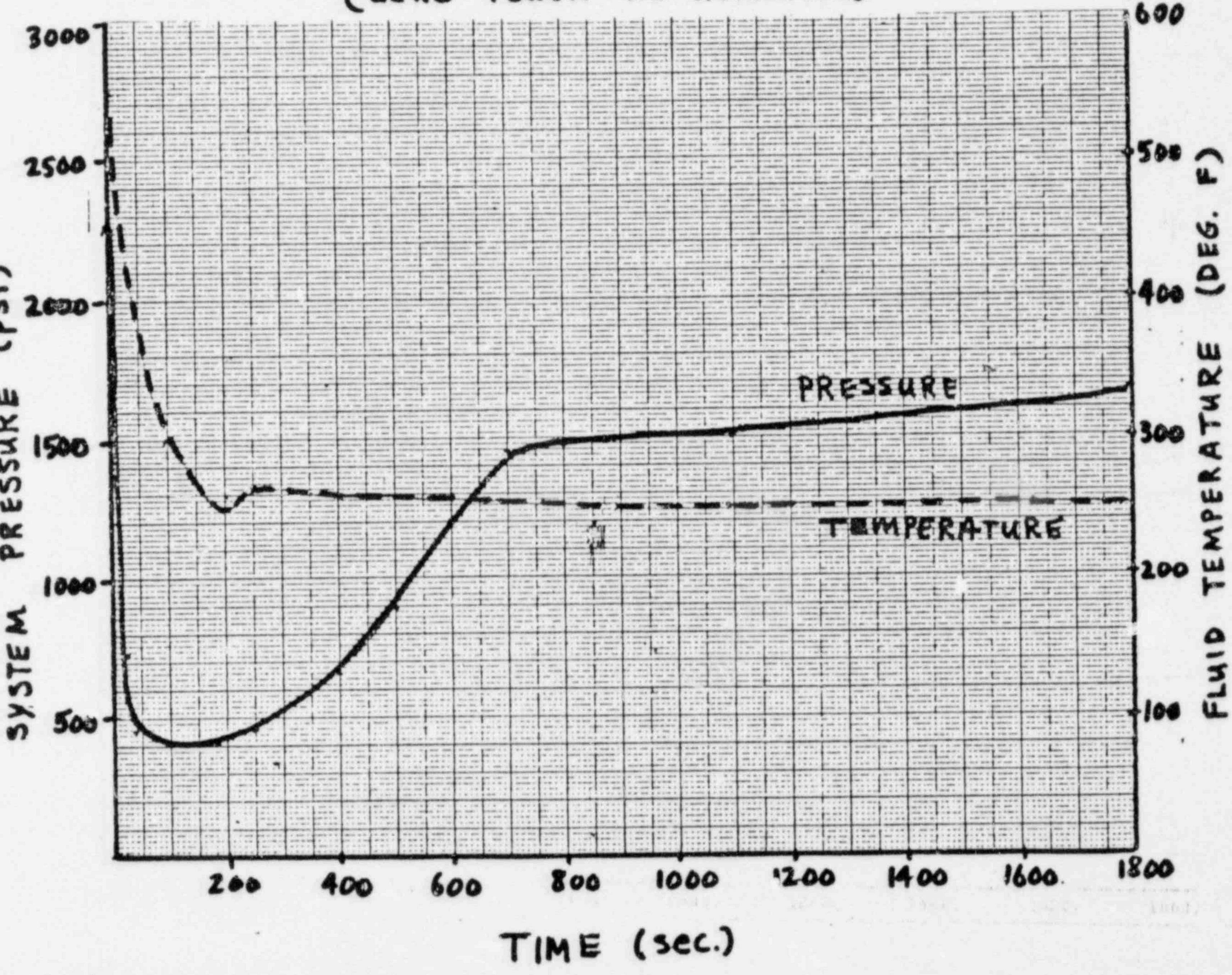

CRITICAL CRACK DEPTH VS. TIME
CRACK DEPTH, A/W
1.00
.90
.80
.70
.60
.50
.40
.30
.20
.10
0.0
800.
1600.
2400.
3200.
4000.
4800.
5600.
6400.
7200.
8000.
TIME (SEC.)
(E71ME
RV Surface Fluence = .46X10^19 n/cm^2
% Cu = .35
% P = .013
RTNDT0 = 10 °F
LOFW → SB LOCA

CRITICAL CRACK DEPTH VS. TIME

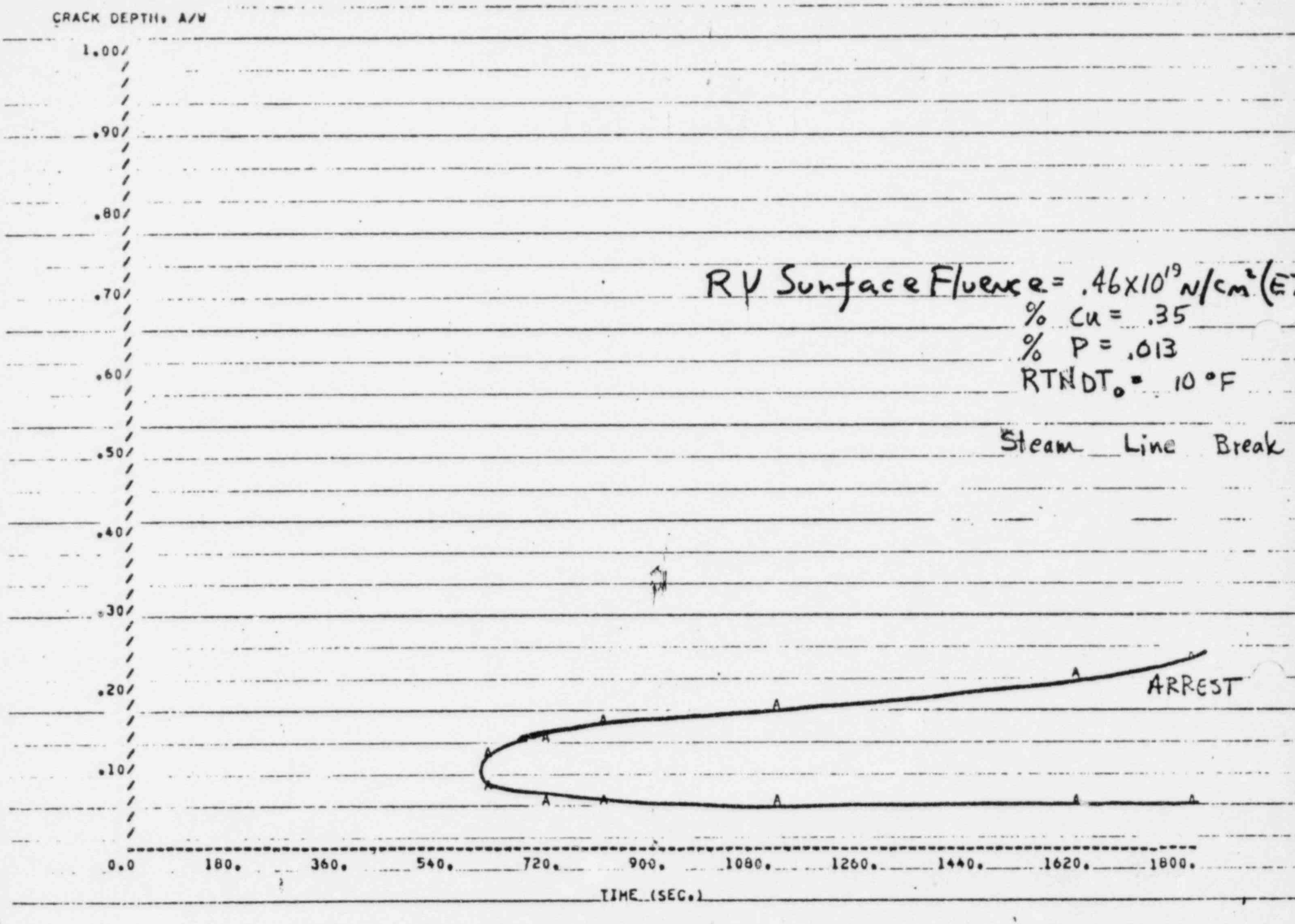


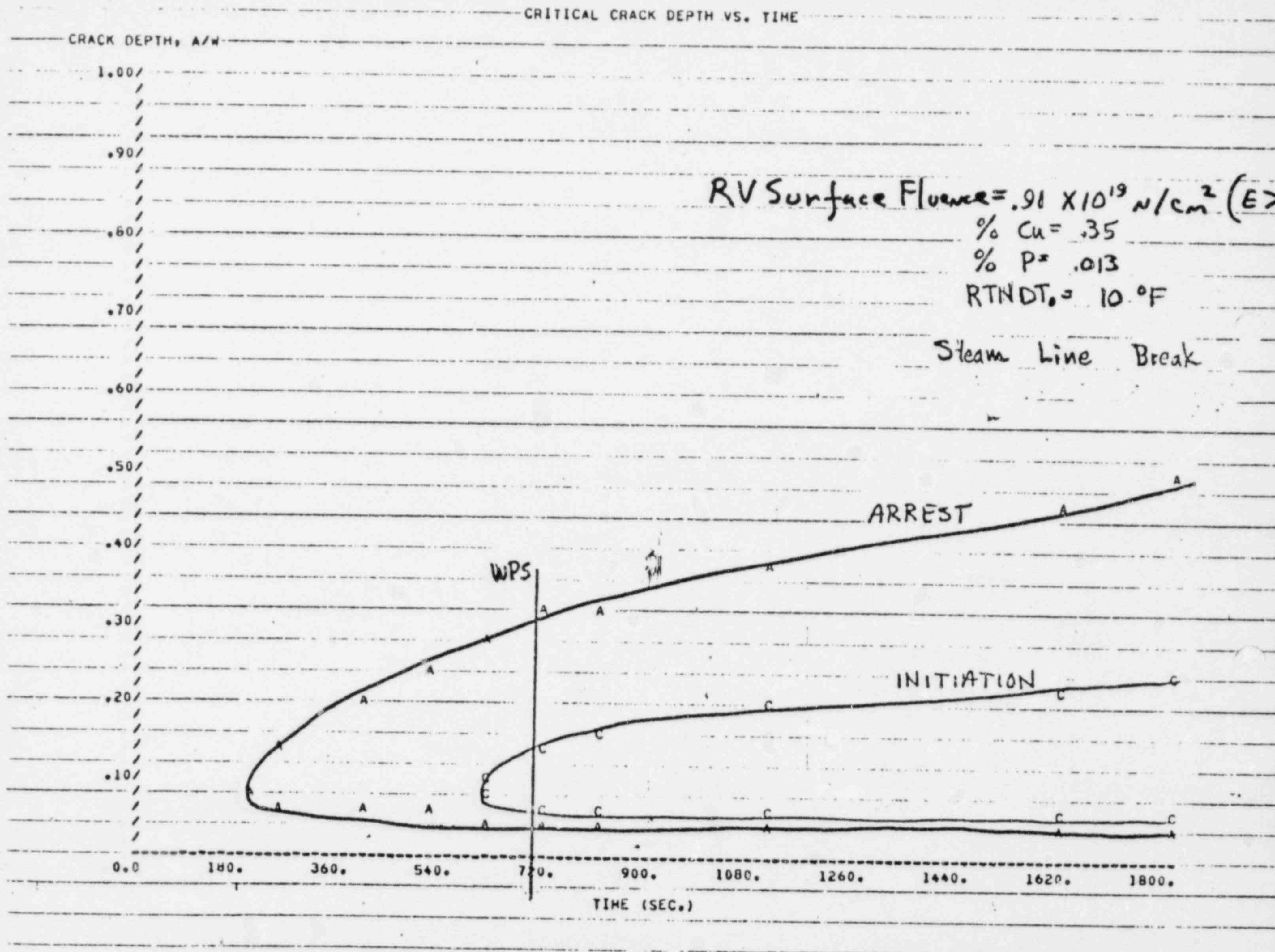
STEAM LINE BREAK TRANSIENT

(ZERO POWER AC AVAILABLE)



CRITICAL CRACK DEPTH VS. TIME




CRITICAL CRACK DEPTH VS. TIME
CRACK DEPTH, A/W
1.00
.90
.60
.70
.60
.50
.40
.30
.20
.10
0.0
180.
360.
540.
720.
900.
1080.
1260.
1440.
1620.
1800.
TIME (SEC.)
RV Surface Fluence = .91 X10^19 n/cm^2 (E>
% Cu = .35
% P = .013
RTNDT₀ = 10 °F
Steam Line Break
ARREST
INITIATION
WPS

STEAM LINE BREAK TRANSIENT No. 2

(ZERO POWER AC AVAILABLE)