PDR-016

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

November 5, 1982

Docket Nos. 50-295 and 50-304

Ms. Catherine Quigg
Research Director
Pollution and Environmental Problems, Inc.
P. O. Box 309
Palatine, IL 60067

IN RESPONSE REFER
TO FOIA-82-496

Dear Ms. Quigg:

This is in response to your letter dated October 18, 1982, in which you requested, pursuant to the Freedom of Information Act, all records on embrittlement, thermal shock, corrosion and cracking of the reactor pressure vessel at the Zion nuclear power station in Zion, Illinois, including "all records which refer or relate to these problems -- whether or not such records specifically identify that plant or company".

When Linda Robinson telephoned you on October 29 for clarification of the scope of your request with regard to the above-listed subjects, you stated that you are interested only in subject records that in fact pertain to Zion. You specifically requested records regarding a letter you believed NRC wrote to Commonwealth Edison requesting an appraisal of the Zion reactor pressure vessel and its ability to withstand thermal shock. In light of the clarification of the scope of your request, Ms. Robinson contacted Guy S. Vissing, NRC staff member working on the thermal shock generic issue, and was informed that to his knowledge NRC has not, as yet, written such a letter regarding the Zion station. Mr. Vissing noted that such inquiry is being proposed by the staff in a draft report (see item No. 13 under Category B on the enclosed appendix). This draft report, however, has not yet been submitted by the staff to the Commission, and therefore, we cannot state at this time what, if any, action will be taken in this matter.

The staff identified the records listed on the enclosed appendix as being pertinent to your request. These records have either already been made available for public inspection, or are now being made available, at the NRC Local Public Document Room (LPDR) in the Zion-Benton Public Library, 2600 Emmaus Avenue, Zion, Illinois. In order to learn when you can obtain access to these records, please contact Ms. Joan Ellington at the LPDR on (312) 872-4680.

Sincerely,

J. M. Felton

J. M. Felton, Director
Division of Rules and Records
Office of Administration

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PDR FOIA
QUIGG82-496 PDR

Enclosure: Appendix

Appendix

A. Records already in the LPDR:

1. 10/1/81 Letter to All licensees representatives by Westinghouse Owners Group (WOG) from Guy Vissing, "Summary of Meeting with the Westinghouse Owners' Group of September 18, 1981, Concerning Pressurized Thermal Shock to Reactor Pressure Vessels (RPV)". (PDR/LPDR 8110190243)

2. 10/23/81 Identical letters to Robin Berger, Renee Feuersteain, Grace Fishman, and H. M. Lipson from Wigginton regarding their concerns about safety at Zion. (PDR/LPDR 8111040446)

3. 12/30/81 Letter to Denton, NRC, from Kingsley, Chairman, WOG, Ala. Power, "Reactor Vessel Integrity", w/enclosed 12/81 report, "Summary Report on Reactor Vessel Integrity for Westinghouse Operating Plants". (PDR/LPDR 8201040111 and 8201040119)

4. 1/18/82 Letter to Denton, NRC, from Lentine, Con Ed, "Zion Station Units 1 and 2, Reactor Vessel Integrity, NRC Docket Nos. 50-295 and 50-304". (PDR/LPDR 8201290413)

5. 6/2/82 IE Bulletin No. 82-02 "Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants". (PDR/LPDR 8204210380)

6. 6/16/82 Letter to Denton from Kingsley, WOG, Ala. Power, Fuel Management to Reduce Neutron Flux", w/attached report. (PDR/LPDR 8206250139)

B. Records being placed in the LPDR:

1. 4/20/81 Letter to all Licen-ees of Operating PWR Nuclear Power Plants from Eisenhut, "Thermal Shock to Reactor Pressure Vessels (Generic Letter 81-19), w/stated enclosure:

   4/7/81 - Memo to Pawlicki from Walker, "Minutes of PWR Owner's Group Meeting with NRC on March 31, 1981". (PDR 8106250099)

2. 3/8/82 "Summary of Meeting w/Westinghouse Owners' Group, Southern California Edison Company, Carolina Power & Light Company, and Florida Power & Light Company Concerning the Pressurized Thermal Shock Issue", w/4 enclosures.

Appendix

B. Records being placed in the LPDR (Continued):

3. 3/15, 16, 18/82 Separate letters and Requests for Additional Information to Ala. Power, So. Calif., Carolina Power & Light, Fla. Power & Light, Maine Yankee, Omaha and Baltimore. (PDR 8203220528, 8203310046, 8203260294, 8203260291, 8203260063, 8204010240, and 8203260050)

4. 5/21/82 "Summary of Meeting of May 10, 1982 With Westinghouse Owners Group (WOG) Concerning the Pressurized Thermal Shock (PTS) Issue", w/3 enclosures.

5. 5/28/82 Letter to Denton from Kingsley (WOG), "Supplemental Information on Reactor Vessel Integrity" (PDR 8206070097), w/stated enclosure, "Summary of Evaluations Related to Reactor Vessel Integrity". (PDR 8206070099)

6. 6/17/82 "Summary of Meeting with PWR Industry Representatives on June 9, 1982 Concerning the Pressurized Thermal Shock (PTS) Issue", w/4 enclosures. (PDR 8206300068)

7. 6/22/82 Letter to Denton from Kingsley, "PTS Review of ERGs" (PDR 8210080294), w/enclosed report, "Review of the Emergency Response Guidelines Relative to Pressurized Thermal Shock". (PDR 8210080297)

8. 6/30/82 Memo to Distribution from Litton, "Summary of WOG/NRC Meeting on Reactor Vessel Integrity on June 22, 1982 Concerning PTS Issue". (PDR 8207300413)

9. 7/15/82 Letter to Denton from Kingsley, "Westinghouse Owners Group Activities Related to Pressurized Thermal Shock", w/attachments. (PDR 8207200166)

10. Undated "Summary of Meeting with Westinghouse Owners Group (WOG) on July 30, 1982, Concerning the Pressurized Thermal Shock (PTS) Issue", w/enclosures.

11. 8/14/81 "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)", w/enclosures. (PDR 8111060283)

12. 8/20/82 "Summary of Meeting With Westinghouse Owners Group (WOG) on August 11, 1982, Concerning the Pressurized Thermal Shock Issue", w/enclosures. (PDR 8209220317)

Appendix

B. Records being placed in the LPDR (Continued):

13. 9/2/82 Letter to Denton from Kingsley, "Westinghouse Owners Group Activities Related to Pressurized Thermal Shock", w/attachments. (PDR 8209080418)

14. 9/13/82 Draft NUREG report, "NRC Staff Evaluation of Pressurized Thermal Shock". (PDR 8210120335)

15. 9/16/82 Letter to Denton from Kingsley, "Westinghouse Owners Group Activities Related to Pressurized Thermal Shock", w/attachments. (PDR 8209210084)

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

April 20, 1981

TO ALL LICENSEES OF OPERATING PWR NUCLEAR POWER PLANTS

SUBJECT: THERMAL SHOCK TO REACTOR PRESSURE VESSELS (GENERIC LETTER 81-19)

For a number of years the NRC and industry have been studying the effects of thermal shock to reactor pressure vessels. Efforts have focused on the identification and characterization of thermal repressurization transient scenarios and their probabilities and on the development of material properties data and improved analytical tools for assessing vessel integrity. A number of on-going research programs are supportive of this effort and continue to provide information intended to quantify actual thermal/mechanical vessel behavior.

In the event of an overcooling system transient resulting in a cooldown of the reactor vessel, followed by repressurization of the pressure boundary above a critical level during the cooling period, vessel integrity could be jeopardized. The likelihood of a vessel cracking upon experiencing a thermal/repressurization transient depends upon (1) its material properties, which degrade with increased irradiation; (2) the severity of the thermal shock which is a function of the degree of mixing of primary water in the system and relatively cold water injected by the high pressure pumps making up part of the Emergency Core Cooling System; and (3) the magnitude of the pressure transient occurring during repressurization.

Earlier this year a number of analyses sponsored by the Commission research program were completed and results became available to the staff. These analyses were directed at providing a better understanding of the severity of overcooling transients which combine operational experience and expected reactor vessel material properties. In the same time frame, as a response to post-TMI requirements, the staff initiated its review of thermal/mechanical reports from licensees of Babcock and Wilcox (B&W) operating reactors intended to further evaluate the effect of high-pressure safety injection on vessel integrity for small-break loss-of-coolant accidents (Item II.K.2.13 of NUREG-0737, Clarification of TMI Action Requirements, November 1980). It was as a result of a review of these on-going efforts that the staff decided to accelerate its evaluation of possible thermal shock to reactor pressure vessels.

On March 31, 1981, the NRC staff met with the PWR Owners Group and representatives of NSSS vendors to discuss the effects of potential thermal shock to reactor pressure vessels by overcooling transients and the potential consequences of subsequent repressurization at relatively low temperature. A copy of the minutes of that meeting is enclosed for your information.

~~8106250094~~
CF

This letter is to confirm the intent of the owners groups to perform a study of this concern including developing alternatives to protect the vessel by avoiding repressurization with cold water.

As indicated in the enclosed minutes of the meeting, we expect to receive a letter report from each of the owners groups summarizing their efforts not later than May 15, 1981. We expect that you are familiar with the activities of the appropriate owners group, actively participating in their discussions, and will provide a docketed response by May 22, 1981, identifying the specific actions you propose to take for your facility.

Sincerely,

Darrell G. Eisenhut

Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation

Enclosure:
Minutes of PWR Owners Groups
Meeting with NRC on March 31, 1981

cc w/encl:
Service Lists

Xtia2

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

May 21, 1982

Dockets Nos.: All Operating Plants with Westinghouse NSSS

LICENSEE: Westinghouse Owners Group

SUBJECT: SUMMARY OF MEETING OF MAY 10, 1982 WITH WESTINGHOUSE OWNERS GROUP (WOG) CONCERNING THE PRESSURIZED THERMAL SHOCK (PTS) ISSUE

Introduction

This was a meeting with the WOG on May 10, 1982, in Bethesda, Maryland at the request of the WOG to provide the staff with an overview of a report which will be provided by the end of May. Enclosure 1 provides the list of attendees to the meeting. The WOG presentation material is provided in Enclosure 2.

Discussion

The report which will be submitted by the end of May will provide a probability assessment of transients of concern to PTS. The report will concentrate on non-DBA transients and will not consider warm prestressing in the associated fracture mechanics analyses. The report will also discuss the review of procedures as they relate to PTS.

W has developed a five step methodology leading to a PRA related to PTS. The conclusion of the study was that the likelihood of a cooldown transient can challenge the reactor vessel in less than 10^{-4} to 10^{-3} per reactor years for the lead plant at 5 EFPY from today. The total risk to the public health is in the order of 10^{-9} per reactor year. The viewgraphs (Enclosure 2) provides the step by step approach to the W methodology.

W has rereviewed the emergency response guidelines (ERG) with the considerations for PTS. This rereview will be in the report to be submitted at the end of May. The rereview will consider PTS in every step and identify additional steps necessary to address PTS.

The Functional Restoration Guidelines development will be complete in June 1982. Warm prestressing will not be considered. It will include a P/T curve and will be plant specific.

Enclosure 3 provides the specific responses to the NRC request for information dated March 16, 1982. These responses will be included in the report due at the end of May.

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

8206180019

Enclosures & cc: See next page

CF

B-4

Enclosures:
1. List of Attendees
2. WOG Presentation Material
3. Viewgraphs for Specific Responses to NRC Request for Info.

cc w/enclosures:

H. Denton/E. Case
D. Eisenhut
G. Lainas
R. Vollmer

W. Hazelton
R. Mattson
T. Speis
T. Murley
H. Thompson

D. Basdekas
L. Shao
R. Bernero

E. Igne
T. Marsh
J. Austin
J. Buzy
B. D. Liaw

D. Ziemann

C. Johnson

E. Abbott
R. Johnson
E. Goodwin
T. Novak

G. Zech
J. Roe

C. Serpan
L. Shotkin

A. Spano

T. Dunning
C. Rossi
J. Strosnider
S. J. Bhatt

Docket File
NRC PDR
L PDR
ORB#4 Rdg

JStolz
Project Manager-GVissing
Licensing Assistant-RIngram
OELD
Heltemes, AEOD
IE
SShowe (PWR) or CThayer (BWR), IE
Meeting Summary File-ORB#4
RFraley, ACRS-10
Program Support Branch

ORAB, Rm. 542
BGrimes, DEP
SSchwartz, DEP
SRamos, EPDB
FPagano, EPLB

Meeting Participants Fm. NRC:

EThrom	LLois
DWigginton	SHHanauer
JWClifford	MVagins
MVirgilio	TCox
RSenseney	FSchroeder
CMorris	RWoods
PNRandall	RWKlecker
WVJohnston	FBLitton

Enclosure 1

ATTENDANCE FOR MEETING WITH WESTINGHOUSE OWNERS GROUP CONCERNING THE PRESSURIZED THERMAL SHOCK ISSUE ON MAY 10, 1982

NRC

Guy S. Vissing
Edward D. Throm
D. L. Wigginton
J. W. Clifford
M. Virgilio
R. Senseney
C. Morris
P. N. Randall
W. V. Johnston
Lambrois Lois
S. H. Hanauer
Milt Vagins
Tom Cox
F. Schroeder
R. Woods
R. W. Klecker
F. B. Litton

Northeast Utilities

Michael F. Aherm

Doc-Search Associates

Lynn Connon

Los Alamos Nat'l. Lab.

Gordon Willcutt
James Lane

Westinghouse

Ray Sero
T. A. Meter
K. R. Balkey
D. S. Ackerson
D. G. Maire
D. R. Sharp
Bruce King
Mike Hitchler
Joan Mcadoo
Harry Julian
M. A. Weaver
J. A. Rumanlik
Gerard G. Elia
Tom Lordi

Babcock & Wilcox

Robert Borsum

Southern Co. Services

Warren M. Andrews

Duke Power Company

Gregg Swindlehurst

Carolina Power & Light Company

James J. Sheppard
David B. Waters

Power Authority of NY

James V. Brunetti

VEPCO

D. W. Lippard

National Nuclear

Lester Kormblith

Florida Power & Light Company

Vernon T. Cailson

Public Service Electric & Gas-NJ

M. O. Bandeira
Timothy N. Taylor

NUTECH

Terry Kishbaugh

PNL

L. J. Pedersen

Pacific Gas & Electric

Russ Klatt

Con. Edison of NY

Daniel M. Speyer

Enclosure 2

AGENDA

° INTRODUCTION	D. SPEYER
° PROBABILISTIC TRANSIENTS	
- OVERVIEW	R. J. SERO
- EVENT SEQUENCE TREES	D. SHARP
- FRACTURE MECHANICS CRITERIA	K. BALKEY
- SORTING RESULTS (SECONDARY DEPRESSURIZATION)	S. ACKERSON
° ERG - SET REVIEW	H. JULIAN
° NRC QUESTIONS/RESPONSE	M. WEAVER
° DISCUSSIONS/CONCLUSIONS	ALL

PROBABILISTIC TRANSIENT EVALUATION

OVERVIEW

° OBJECTIVE:

FOR REAL TRANSIENT CASES, PROVIDE AN ASSESSMENT OF THE CONTRIBUTION OF PTS TO THE "RISK" ASSOCIATED WITH PLANT OPERATION

° METHODOLOGY:

1 - CATEGORIZE THE CONDITIONS WHICH CAN CHALLENGE R.V. INTEGRITY

2 - CREATE INITIATOR TREES AND MITIGATOR TREES TO IDENTIFY THE LIKELIHOOD THAT EACH OF THESE CATEGORIES OF EVENTS CAN OCCUR

- CONSIDER

  MULTIPLE EQIPMENT FAILURES
  SENSITIVITY TO OPERATOR ACTION TIMES
  DECAY HEAT

3 - ESTABLISH A TEMPERATURE LIMIT FOR THE REACTOR VESSEL DOWNCOMER. (LEAD REPRESENTATIVE VESSELS, BEST ESTIMATE VESSELS)

4 - SORT THE TRANSIENTS FROM STEP TWO INTO TWO "BINS" - NO INITIATION AND POTENTIAL INITIATORS. SUM ONLY THE "POTENTIAL INITIATORS". RESULT REPRESENTS THE LIKELIHOOD THAT THE CATEGORY OF TRANSIENTS CONSIDERED CAN LEAD TO <u>POTENTIAL INITIATION</u>.

5 - IF A SCENARIO HAS A LIKELIHOOD $> 10^{-3}$, PERFORM LEFM ON TRANSIENTS TO DETERMINE WHICH TRANSIENT ANALYTICALLY PRODUCES A CRACK INITIATION. THIS BECOMES THE NEW <u>POTENTIAL</u> <u>INITIATOR</u> TRANSIENT AND ITS LIKELIHOOD THE LIKELIHOOD FOR THE CATEGORY OF EVENTS.

CATEGORIES

- SECONDARY DEPRESSURIZATIONS
- EXCESSIVE FEEDWATER ADDITION
- PRIMARY SYSTEM DEPRESSURIZATION WITH SUBSEQUENT RE-PRESSURIZATION
- STEAM GENERATOR TUBE RUPTURE

SPECIAL NOTES

- WOG HAS CHOSEN POTENTIAL INITIATION FOR THIS EVALUATION. NOT ARREST.
- UTILIZED PREVIOUS EVENT TREE APPROACH FROM ERG DEVELOPMENT. ESTABLISHED TECHNIQUES.
- CONSIDERED BOTH: THE PREDICTED CONSERVATIVE MATERIAL AT TODAY +5 EFPY: AND THE NRC "BEST ESTIMATE" MATERIAL.
- BETTER ESTIMATE MIXING ANALYSES CONSIDERED AS WELL AS NO MIXING.

TRANSIENT RESULTS (PRELIMINARY)

PROBABILITY OF OCCURRENCE OF A POTENTIAL INITIATION TRANSIENT BY CLASS

(OCCURRENCE PER REACTOR YEAR)

CLASS OF COOLDOWN TRANSIENT	BETTER ESTIMATE T/H WITH "BEST ESTIMATE" MATERIAL (SORT A)	BETTER ESTIMATE T/H WITH REPRESENTATIVE LEAD MATERIAL (SORT B)	"NO MIXING" REPRESENTATIVE LEAD MATERIAL (SORT C)	"NO MIXING" WITH PROCEDURE MODIFICATION (RCP TRIP) (SORT D)
SECONDARY DEPRESSURIZATION	6.0×10^{-7}	2.8×10^{-4}	2.8×10^{-4}	9.9×10^{-5}
EXCESSIVE FEEDWATER	$<1.0 \times 10^{-7}$	$<1.0 \times 10^{-7}$	4.0×10^{-5}	$<1.0 \times 10^{-7}$
LOCA	NOT AVAILABLE	1×10^{-3}	1×10^{-3}	1×10^{-3}
SGTR	$<1 \times 10^{-4}$	1×10^{-4}	1×10^{-3}	1×10^{-4}
TOTAL	NOT AVAILABLE	1×10^{-3}	2×10^{-3}	1×10^{-3}

Preliminary

PROBABILITY CONCLUSIONS

R_{PH}: $R_C, R_I, R_A, R_{CD}, R_{CR}$

R_C - RISK OF VESSEL INTEGRITY CHALLENGE

R_I - RISK OF CRACK INITIATION

R_A - RISK OF NO CRACK ARREST

R_{CD}- RISK OF CORE DAMAGE/MELT

R_{CR}- RISK OF CONTAINMENT RELEASE

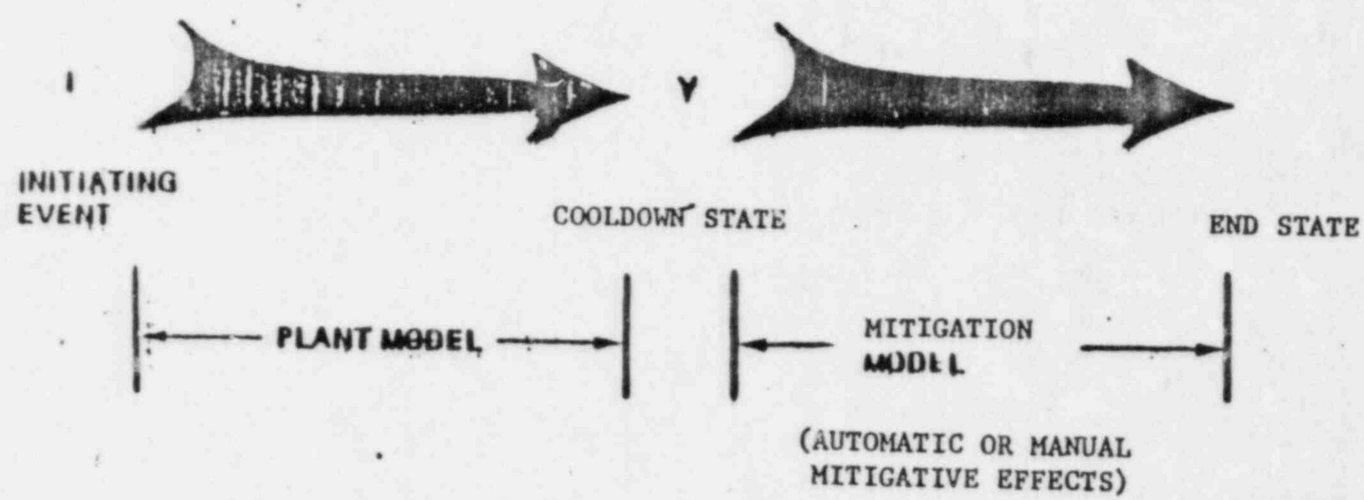
$R_{PH} = 10^{-9}$

CONCLUSIONS

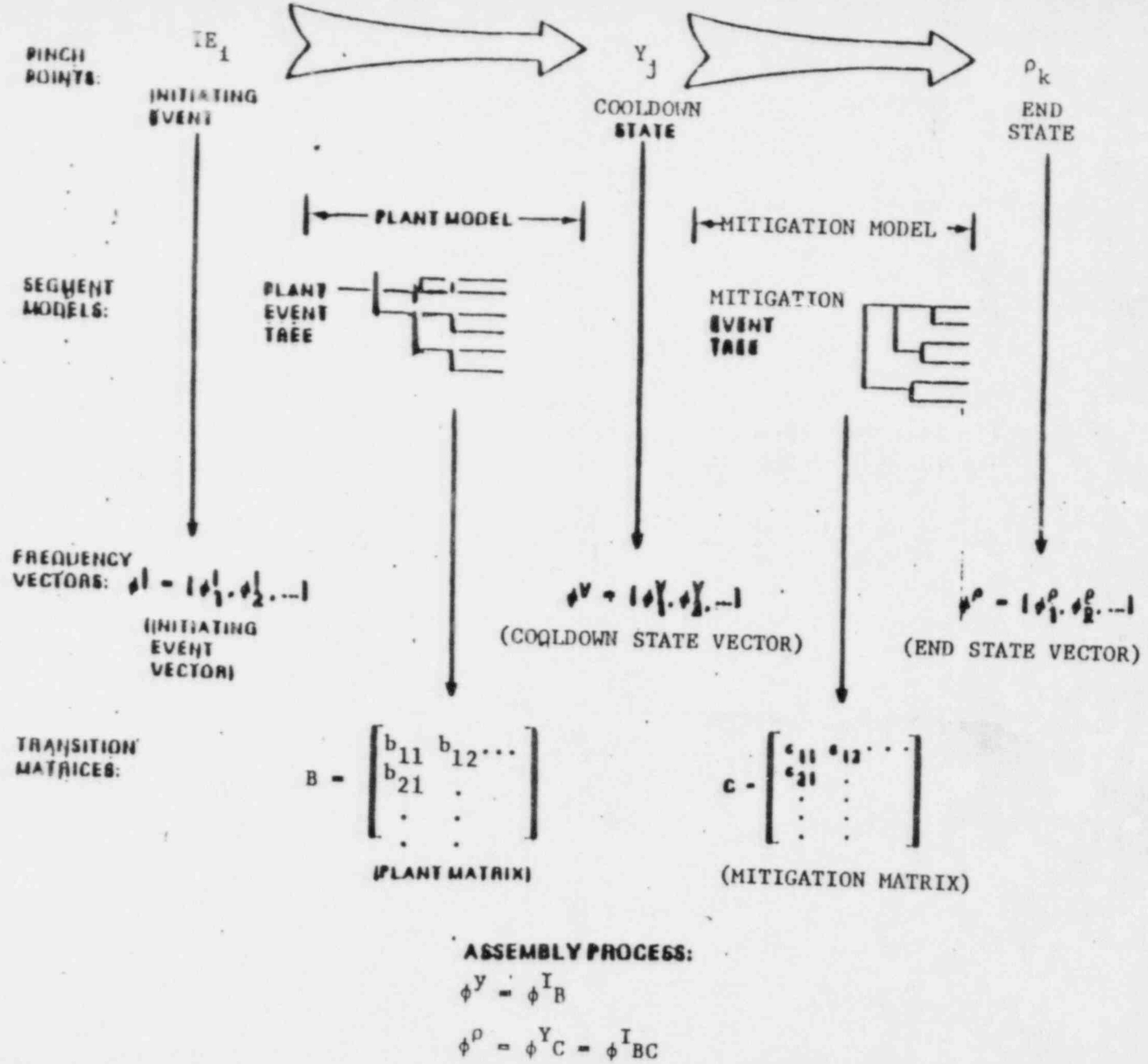
- THE LIKELIHOOD THAT A COOLDOWN TRANSIENT CAN CHALLENGE R. V. INTEGRITY(I.E., CAN CHALLENGE THROUGH POTENTIAL INITIATION) IS LESS THAN 10^{-3}-10^{-4} PER REACTOR YEAR.
- WHEN WE CONSIDER A PROBABILISTIC ASSESSMENT OF TOTAL RISK, THE RISK TO PUBLIC HEALTH IS ON THE ORDER OF 10^{-9}.
- THESE ARE ACCEPTABLE LIMITS, WHICH PERMITS IMPLEMENTATION OF A SYSTEMATIC PROGRAM FOR LONG TERM RESOLUTION OF PTS.
- AN ACCEPTABLE METHODOLOGY HAS BEEN DEVELOPED TO IDENTIFY PROCEDURE MODIFICATION, DESIGN MODIFICATION, OR TRAINING PROGRAMS WHICH CAN APPROPRIATELY IMPACT PTS.

PROBABILISTIC ASSESSMENT OF
FREQUENCY OF CHALLENGERS
TO VESSEL INTEGRITY

- OVERALL APPROACH
- CRITICAL ASSUMPTIONS
- RESULTS TO DATE



Structuring of Scenarios - Relationship of Pinch Points



Overview of the Assembly Process, Showing Relationship of Pinch Points, Frequency Vectors, Event Trees, and Transition Matrices

INITIATING EVENT FREQUENCIES

EVENT	FREQUENCY (/R-YEAR)
1. LOSS OF MAIN FEEDWATER (LOFW)	3.41
2. CLOSURE OF ONE MAIN STEAM ISOLATION VALVE (MSIV)	6.00×10^{-1}
3. LOSS OF PRIMARY FLOW (LOPF)	3.21×10^{-1}
4. CORE POWER INCREASE (POWIN)	4.77×10^{-2}
5. TURBINE TRIP (TT)	4.00
6. SPURIOUS SAFETY INJECTION ACTIVATION (SSI)	1.59×10^{-1}
7. REACTOR TRIP (RT)	4.11
8. TURBINE TRIP DUE TO LOSS OF OFFSITE POWER (TT/LOOP)	1.01×10^{-3}
9. STEAM GENERATOR TUBE RUPTURE (SGTR)	3.92×10^{-2}
10. SMALL LOCA, <1.5 IN DIAMETER (LOCA-1)	9.07×10^{-3}
11. SMALL LOCA, >1.5 IN DIAMETER (LOCA-2)	6.11×10^{-4}
12. LARGE LOCA, >6 IN DIAMETER (LOCA-3)	3.88×10^{-4}
13. EXCESSIVE MAIN FEEDWATER (EX FW)	2.50×10^{-1}
14. STEAMLINE RUPTURE INSIDE CONTAINMENT (STM BRK IN)	3.88×10^{-4}
15. STEAMLINE RUPTURE OUTSIDE CONTAINMENT (STM BRK OUT)	3.87×10^{-2}

INITIATING EVENT FREQUENCIES BY DECAY HEAT STATE

OVERALL INITIATOR FREQUENCIES FROM GENERIC PWR POPULATION.

OVERALL FREQUENCIES SEPARATED INTO DECAY HEAT SUPPORT STATE FREQUENCIES.

1. DECAY HEAT BETWEEN FULL TRIP DECAY HEAT AND 1% OF RATED POWER.

2. DECAY HEAT BETWEEN 1% AND 0.5% RATED POWER.

3. DECAY HEAT LESS THAN 0.5% RATED POWER.

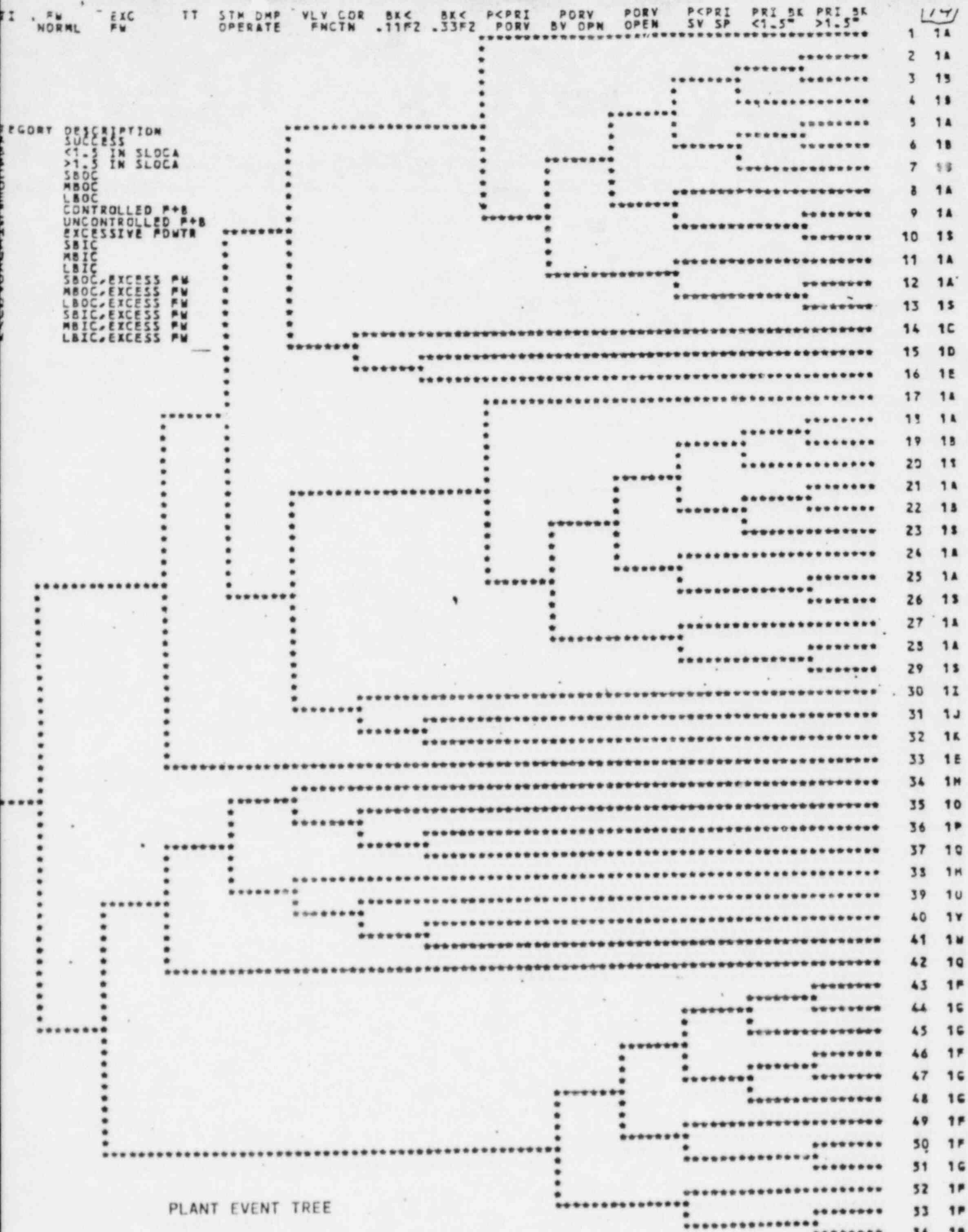
BASIS FOR ABOVE:

1. EPRI REPORT NP-2230.

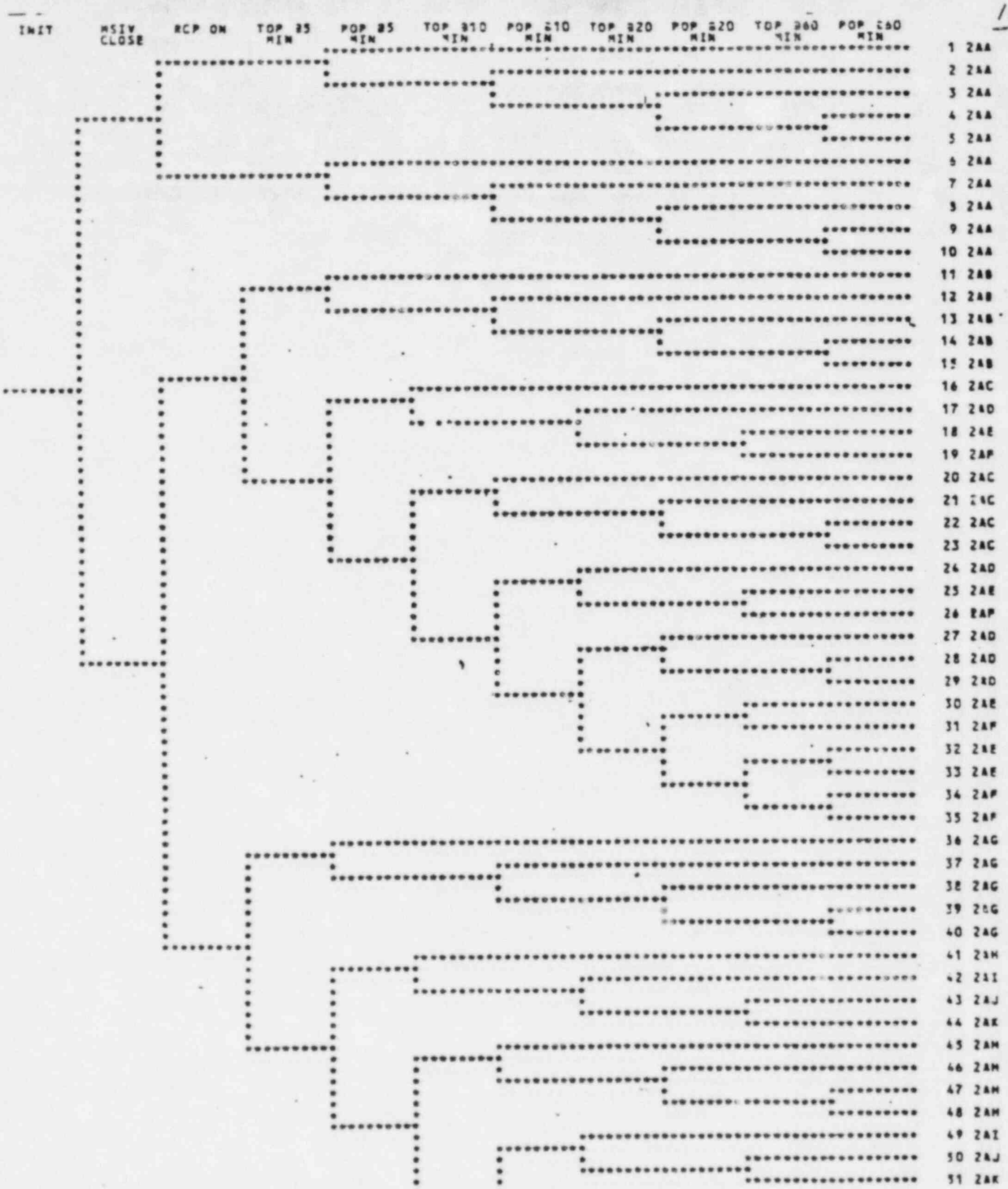
2. PLANT TRIP DATA FOR OUTAGE TIME.

DECAY HEAT (DH) SUPPORT STATE FREQUENCIES

EVENT	DH>1%	1%>DH>0.5%	DH<0.5%
1. LOFW	2.29	0.67	0.44
2. MSIV	0.42	0.11	0.07
3. LOPF	0.27	0.03	0.02
4. POWIN	0.025	0.013	0.008
5. TT	3.29	0.43	0.28
6. SSI	0.10	0.04	0.02
7. RT	3.19	0.56	0.36
8. TT/LOOP	7.76×10^{-4}	1.35×10^{-4}	8.9×10^{-5}
9. SGTR	2.94×10^{-2}	5.88×10^{-3}	3.92×10^{-3}
10. LOCA-1	6.8×10^{-3}	1.4×10^{-3}	9.1×10^{-4}
11. LOCA-2	4.6×10^{-4}	9.2×10^{-5}	6.1×10^{-5}
12. LOCA-3	2.9×10^{-4}	5.8×10^{-5}	3.9×10^{-5}
13. EX FW	2.0×10^{-1}	3.2×10^{-2}	2.0×10^{-2}
14. STM BRK IN	2.9×10^{-4}	5.8×10^{-5}	3.9×10^{-5}
15. STM BRK OUT	2.9×10^{-2}	5.8×10^{-3}	3.9×10^{-3}



PLANT EVENT TREE



CATEGORY	DESCRIPTION
2AA	MSIV CLOSE
2AB	TEMP CONTROL @ 5 MIN, W/RCP
2AC	TEMP CONTROL @ 10 MIN, W/RCP
2AD	TEMP CONTROL @ 20 MIN, W/RCP
2AE	TEMP CONTROL @ 60 MIN, W/RCP
2AF	NO TEMP CONTROL, W/RCP
2AG	TEMP CONTROL @ 5 MIN, W/O RCP
2AM	TEMP CONTROL @ 10 MIN, W/O RCP
2AI	TEMP CONTROL @ 20 MIN, W/O RCP
2AJ	TEMP CONTROL @ 60 MIN, W/O RCP
2AK	NO TEMP CONTROL, W/O RCP

MITIGATION EVENT TREE

NODE PROBABILITIES FOR PLANT EVENT TREE

EVENT	EVENT NAME	FAILURE PROBABILITY
&	REACTOR TRIP - FULL DH	.1000E+01
A	FW NORMAL	.5530E-04
B	EXCESS FW	.9040E+00
C	TURBINE TRIP	.5500E-04
D	ST DUMPS OP	.3000E-01
E	VALVES CORRECTLY FUNCTION	.8000E-02
F	BREAK <.11 FT2	.3700E-02
G	BREAK <.33 FT2	.1300E-01
H	P< PRI PORV SET	.2000E-01
I	PRI PORV BV OP	.5500E+00
J	PRI PORV OPENS	.1000E-01
K	P<PRI SV SETP	.1000E-02
L	PRI BRK <1.5IN	.1100E-04
M	PRI BRK >1.5IN	.1000E-02

NODE PROBABILITIES FOR MITIGATION EVENT TREE

EVENT	EVENT NAME	FAILURE PROBABILITY
&	DUMMY	.1000E+01
A	MSIV CLOSE	.5000E-03
B	RCP ON	.0000E+00
C	TOP @5 MIN	.3200E+00
D	POP @5 MIN	.5000E+00
E	TOP @10 MIN	.9000E-01
F	POP @10 MIN	.1900E+00
G	TOP @20 MIN	.5000E-01
H	POP @20 MIN	.1000E+00
I	TOP @60 MIN	.5000E-01
J	POP @60 MIN	.1000E+00

PROBABILITY SORTING CRITERIA

SORTING CRITERIA:

- QUANTIFY FREQUENCY OF ENDSTATE CATEGORIES.
- ASSUME CATEGORIES BELOW 10^{-6} ARE INITIATORS AND PLACE IN HOLDING BIN.
- ACCUMULATE CATEGORIES ABOVE 10^{-5} EVALUATED TO BE POTENTIAL INITIATORS.
- SUM POTENTIAL INITIATOR FREQUENCIES.
- SUM HOLDING BIN FREQUENCIES TO ASSESS THEIR RELATIVE INTEGRATED IMPACT.

End State Results For Steambreak

Decay Heat Greater Than 1%

CATEGORY	DESCRIPTION	SBOC FREQUENCY	MBOC FREQUENCY	LBOC FREQUENCY	SBIC FREQUENCY	MBIC FREQUENCY	LBIC FREQUENCY
2AA	MSIV CLOSE	.1029E+00	.2715E-03	.3331E-03	.0000E+00	.0000E+00	.0000E+00
2AB	TEMP CONTROL @ 5 MIN	.3502E-04	.9234E-07	.1150E-06	.2176E-02	.9374E-05	.1975E-08
2AC	TEMP CONTROL @ 10 MIN	.1500E-04	.3954E-07	.4926E-07	.9318E-03	.4228E-05	.9456E-09
2AD	TEMP CONTROL @ 20 MIN	.1409E-05	.3715E-08	.4628E-08	.8755E-04	.3973E-06	.7945E-10
2AE	TEMP CONTROL @ 60 MIN	.7045E-07	.1858E-09	.2314E-09	.4378E-05	.1986E-07	.3973E-11
2AF	NO TEMP CONTROL	.3708E-08	.9778E-11	.1218E-10	.2304E-06	.1045E-08	.2091E-12

Decay Heat Between 1% and 0.5%

CATEGORY	DESCRIPTION	SBOC FREQUENCY	MBOC FREQUENCY	LBOC FREQUENCY	SBIC FREQUENCY	MBIC FREQUENCY
2AA	MSIV CLOSE	.1299E-01	.1077E-04	.7558E-04	.0000E+00	.0000E+00
2AB	TEMP CONTROL @ 5 MIN	.4420E-05	.3665E-08	.2571E-07	.1156E-03	.3799E-07
2AC	TEMP CONTROL @ 10 MIN	.1893E-05	.1570E-08	.1101E-07	.4950E-04	.1623E-07
2AD	TEMP CONTROL @ 20 MIN	.1778E-06	.1475E-09	.1034E-08	.4651E-05	.1524E-08
2AE	TEMP CONTROL @ 60 MIN	.8892E-08	.7374E-11	.5172E-10	.2326E-06	.7622E-10
2AF	NO TEMP CONTROL	.4680E-09	.3881E-12	.2722E-11	.1224E-07	.4012E-11

Decay Heat Less Than 0.5%

CATEGORY	DESCRIPTION	SBOC FREQUENCY	MBOC FREQUENCY	LBOC FREQUENCY	SBIC FREQUENCY
2AA	MSIV CLOSE	.8496E-02	.6969E-05	.5092E-04	.0000E+00
2AB	TEMP CONTROL @ 5 MIN	.2890E-05	.2370E-08	.1732E-07	.5100E-04
2AC	TEMP CONTROL @ 10 MIN	.1238E-05	.1015E-08	.7418E-08	.2184E-04
2AD	TEMP CONTROL @ 20 MIN	.1163E-06	.9538E-10	.6970E-09	.2052E-05
2AE	TEMP CONTROL @ 60 MIN	.5814E-08	.4769E-11	.3485E-10	.1026E-06
2AF	NO TEMP CONTROL	.3060E-09	.2510E-12	.1834E-11	.5400E-08

FRACTURE MECHANICS SORTING CRITERIA FOR PROBABILISTIC TRANSIENT EVALUATION

FRACTURE MECHANICS EVALUATION

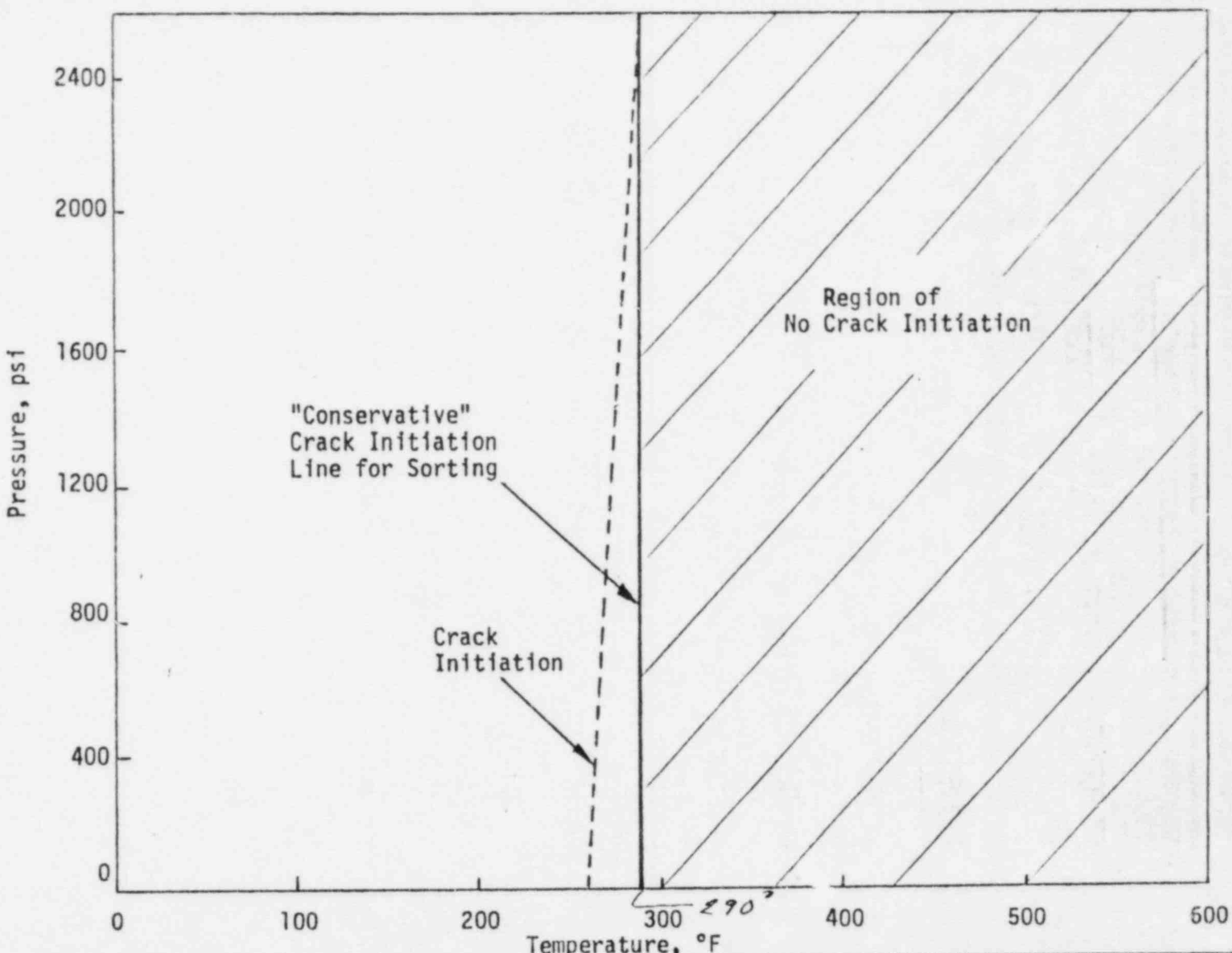
OBJECTIVE: TO DETERMINE PRESSURE-TEMPERATURE LIMITS BASED UPON CRACK INITIATION FOR SORTING PROBABILISTIC TRANSIENTS

ASSUMPTIONS:

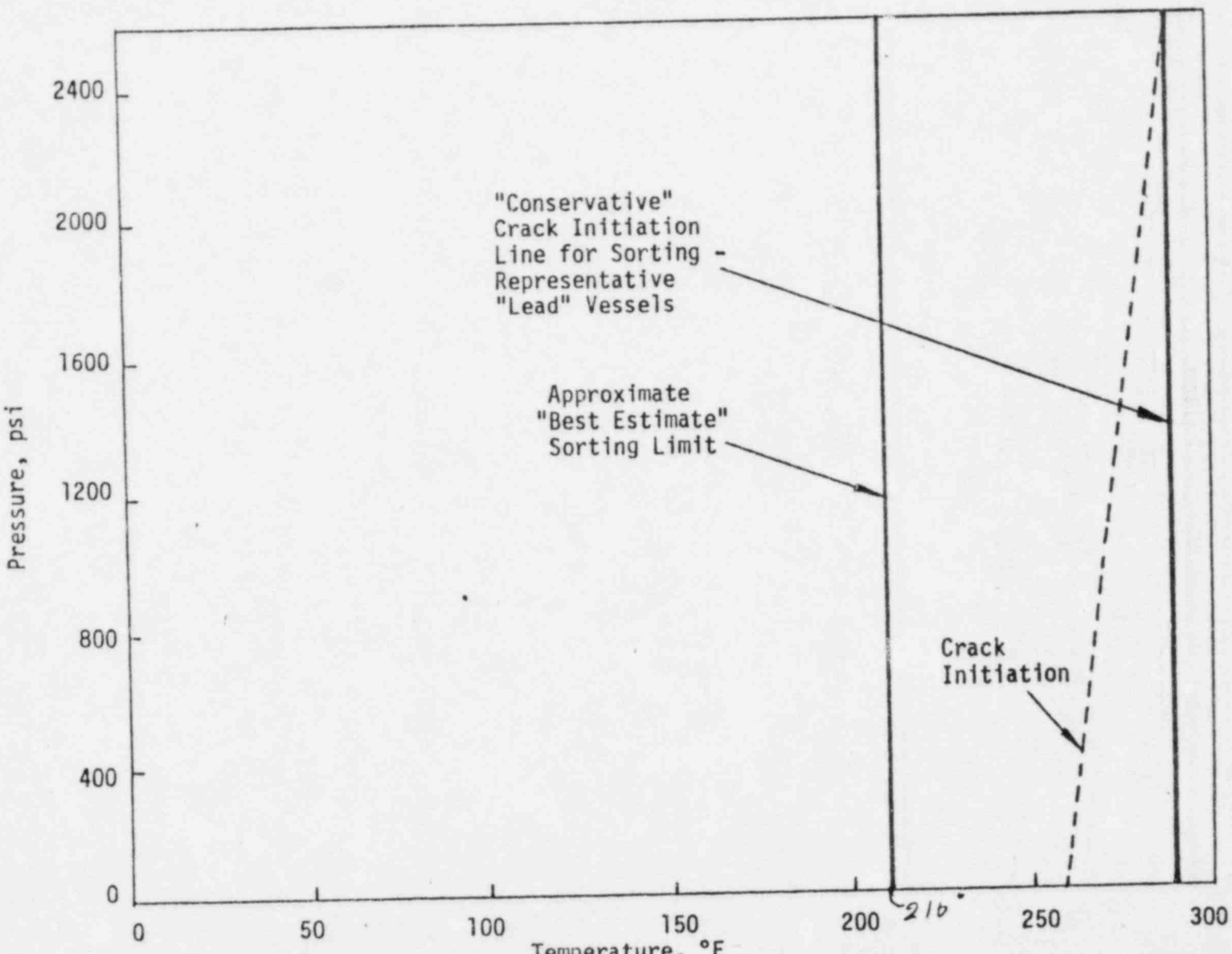
- STEP CHANGES IN TEMP. FROM 550°F
- FULL RCS FLOW
- GEOMETRY, MATERIAL PROPERTIES AND FLUENCE FOR REPRESENTATIVE "LEAD" VESSELS @ 5 EFPY FROM TODAY
- R.G. 1.99 IRRADIATION DAMAGE TREND CURVES & W LOW NICKEL TREND CURVES WHEN APPLICABLE
- EVALUATE CRACK INITIATION VALUES ≤ 25% OF VESSEL WALL THICKNESS

METHOD: DETERMINE CRITICAL PRESSURE FOR EACH THERMAL TRANSIENT

PRESSURE-TEMPERATURE LIMITS FOR SORTING PROBABILISTIC TRANSIENTS FOR REPRESENTATIVE "LEAD" VESSELS AT 5 EFPY FROM PRESENT



PRESSURE-TEMPERATURE LIMITS FOR SORTING PROBABILISTIC TRANSIENTS AT 5 EFPY FROM PRESENT



CONSERVATISMS INHERENT TO THE FRACTURE ANALYSIS SORTING CRITERIA

- LIMITING TRANSIENT - STEP CHANGE IN ΔT AND FULL RCS FLOW
- 5 EFPY (NRC TAP - 2 EFPY FOR SHORT TERM)
- NO CREDIT FOR LOW LEAKAGE CORE PATTERNS IN PLACE OR PLANNED
- R.G. 1.99 VS. VARIOUS PROPOSED REVISIONS

EFFECTS OF

SECONDARY DEPRESSURIZATIONS

ON REACTOR VESSEL INTEGRITY

PRESENTATION

OUTLINE

OPERATIONAL TRANSIENTS EFFORT - SECONDARY DEPRESSURIZATION

- RESULTS (GENERAL)
- SCENARIOS
- SORTING METHODOLOGY
- SECONDARY DEPRESSURIZATION TRANSIENT
  - ASSUMPTIONS
  - INITIAL COOLDOWN
  - FINAL RCS TEMPERATURE
- RESULTS (DETAILED)
- CONSERVATISM OF THE ANALYSIS
- MIXING CONCERN

RESULTS (GENERAL)

Sort	Sorting Temp.	Mixing	x10^{-2} Factor	Procedural Change W.R.T. RCP's	Probability of Potential Initiation
A	210	Perfect	No	No	6.0×10^{-7}
B	290	Perfect	No	No	2.8×10^{-4}
C	290	None Worst Case	No	No	2.8×10^{-4}
D	290	None Worst Case	No	Yes	9.9×10^{-5}
E	290	None Worst Case	Yes	Yes	9.9×10^{-7}
F	290	Perfect	Yes	No	2.8×10^{-6}

SORTING METHODOLOGY
(290°F)

NI BIN - Minimum RCS Temperature <u>Does Not</u> Decrease Below 290° F At Any Time During The Transient.

PI BIN - Minimum RCS Temperature <u>Does</u> Decrease Below 290° F At Some Time During The Transient.

Note That RCS Pressure Is Not A Sorting Consideration.

<u>Checks</u>

1) Initial Cooldown Final Temperature

2) Quasi - Equilibrium Temperature

3) Questionable Mixing

MAJOR SECONDARY DEPRESSURIZATION ANALYSIS ASSUMPTIONS

- 3 - LOOP PLANT
- HOT ZERO POWER
- AUX FEED SYSTEM 350 GPM MDP (2)
  700 GPM TDP (1)

  BEST ESTIMATE FLOW AS A FUNCTION OF TIME UNTIL TERMINATION
- IMMEDIATE STEAM LINE ISOLATION
- THICK METAL MODELED
- SG REVERSE HEAT TRANSFER MODELED
- IMMEDIATE SI INITIATION
- AUX FEED TEMPERATURE 40° F

FIGURE II.4.1.2-1

TYPICAL COOLDOWN ASSOCIATED WITH A .11 SQUARE FOOT EQUIVALENT STEAM BREAK ON A 3-LOOP PLANT. NO OPERATOR ACTION TO THROTTLE AUXILIARY FEED TO ANY STEAM GENERATOR

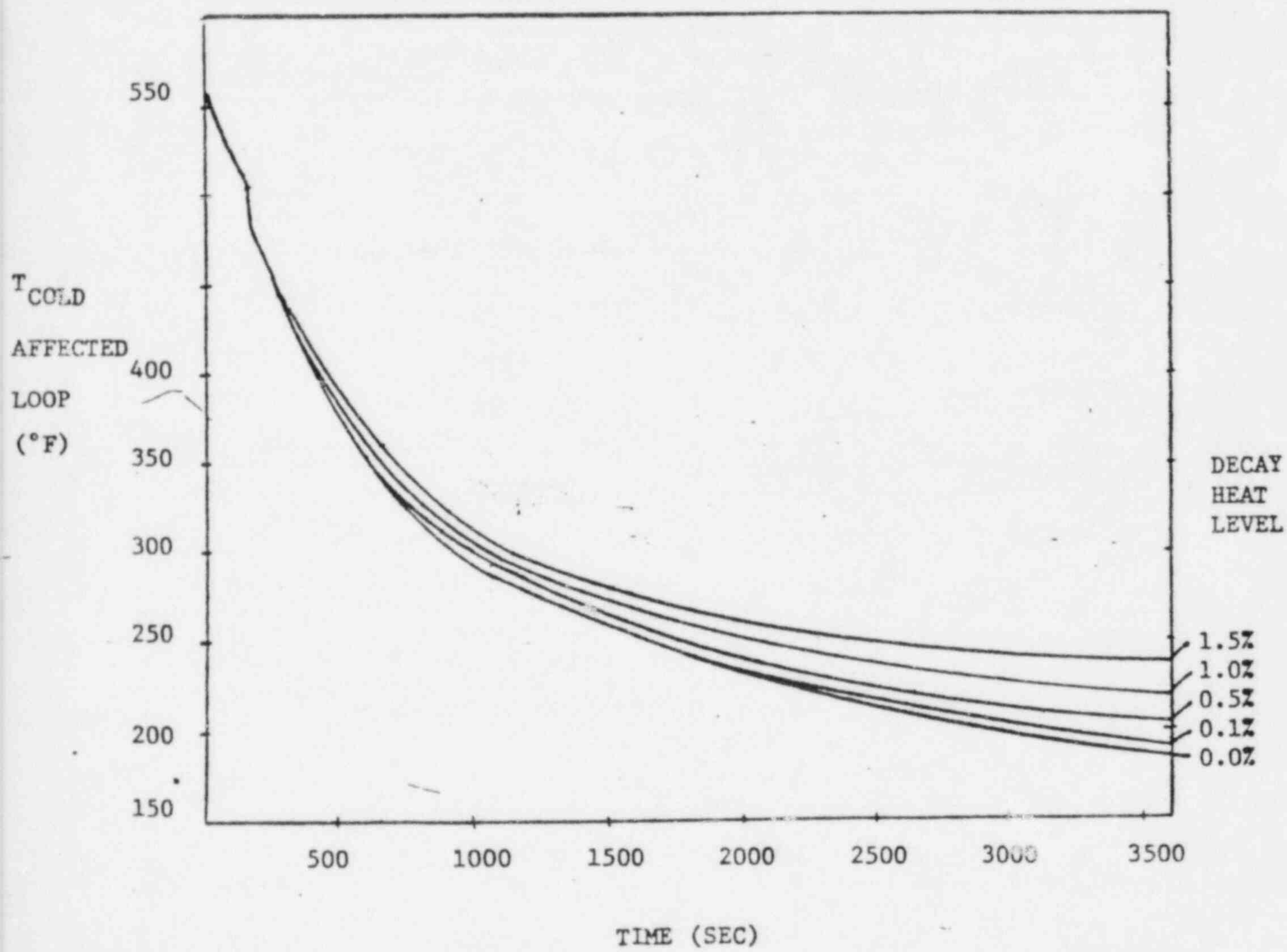
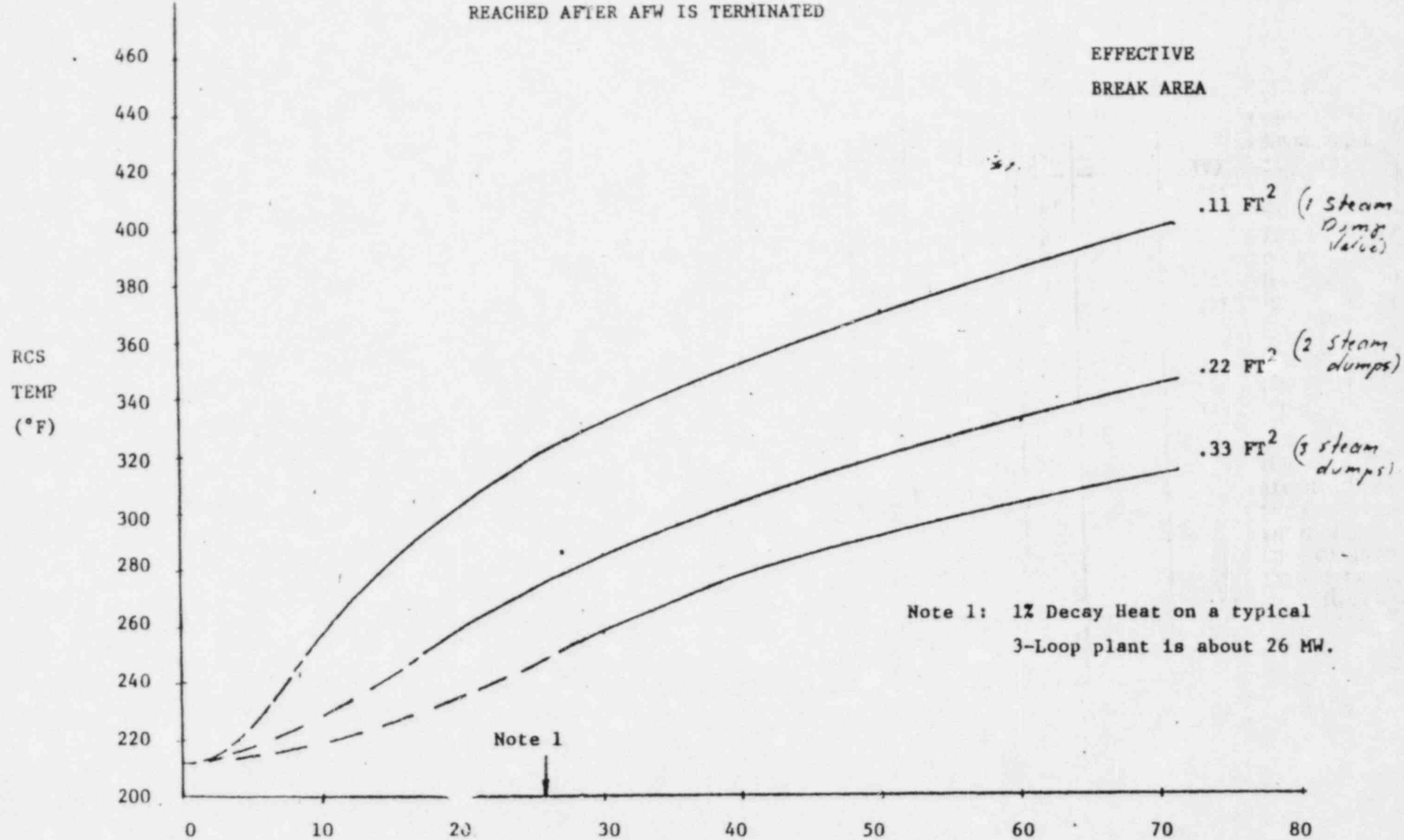


FIGURE II.4.1.2-7

QUASI - EQUILIBRIUM RCS TEMPERATURE
REACHED AFTER AFW IS TERMINATED



NET HEAT INPUT (MW)
($\dot{Q}$in net = $\dot{Q}$decay heat + $\dot{Q}$pumps - $\dot{Q}$losses)

TABLE: TABLE II.4.1.2 -1

TITLE: PROBABILITY OF CAUSING A POTENTIAL INITIATION BY CATEGORY OF SECONDARY DEPRESSURIZATION (UNITS: x10^{-4} OCCURRENCES PER REACTOR YEAR)

RCP's TRIPPED, 290°F, PERFECT MIXING

LOCATION	EQUIVALENT BREAK AREA	DECAY HEAT .GT. 1%					DECAY HEAT .GE. 1% .LE 0.5%					DECAY HEAT .LT. 0.5%					PI SUB TOTAL
		OPERATOR ACTION (1)					OPERATOR ACTION (1)					OPERATOR ACTION (1)					
		5	10	20	60	INF	5	10	20	60	INF	5	10	20	60	INF	
UPSTREAM OF MSIV's	SMALL (2)	22.0 NI	9.30 NI	.880 NI	.044 PI	.002 PI	1.20 PI	.500 PI	.047 PI	.002 PI	.000 PI	.510 PI	.220 PI	.021 PI	.001 PI	.000 PI	2.547
	MEDIUM (3)	.099 PI	.042 PI	.004 PI	.000 PI	.000 PI	.000 PI	.000 PI	.000 PI	.000 PI	.000 PI	.000 PI	.000 PI	.000 PI	.000 PI	.000 PI	.145
	LARGE (4)	.000 PI	.000 PI	.000 PI	.000 PI	.000 PI	.000 PI	.000 PI	.000 PI	.000 PI	.000 PI	.000 PI	.000 PI	.000 PI	.000 PI	.000 PI	.000
DOWNSTREAM OF MSIV's	SMALL (2)	.350 NI	.150 NI	.014 NI	.000 PI	.000 PI	.044 PI	.019 PI	.002 PI	.000 PI	.000 PI	.029 PI	.012 PI	.001 PI	.000 PI	.000 PI	.107
	MEDIUM (3)	.001 PI	.000 PI	.000 PI	.000 PI	.000 PI	.000 PI	.000 PI	.000 PI	.000 PI	.000 PI	.000 PI	.000 PI	.000 PI	.000 PI	.000 PI	.001
	LARGE (4)	.001 PI	.000 PI	.000 PI	.000 PI	.000 PI	.000 PI	.000 PI	.000 PI	.000 PI	.000 PI	.000 PI	.000 PI	.000 PI	.000 PI	.000 PI	.001
TOTAL PROBABILITY OF CAUSING A POTENTIAL INITIATION FOR ALL CATEGORIES OF SECONDARY DEPRESSURIZATION																	2.801

PI - POTENTIAL INITIATION NI - NO INITIATION

NOTES

(1) TIME TO TERMINATE UNCONTROLLED AUXILIARY FEED TO ALL STEAM GENERATORS IN MINUTES

(2) .LE. .11 SQUARE FEET (ONE VALVE)

(3) .GT. .11 AND .LE. .33 SQUARE FEET (TWO OR THREE VALVES)

(4) .GT. .33 SQUARE FEET (FOUR OR MORE VALVES)

CONSERVATISM OF THE ANALYSIS

- RCP RUNNING CASES NOT CONSIDERED
- SORTING CRITERIA ITSELF
- AUX FEED TEMPERATURE AND FLOW RATES (40°)
- NO PORV OR STEAM DUMP ISOLATION CONSIDERED
- NO BENEFIT GIVEN FOR STEAM GENERATOR DRYOUT
- PROBABILITIES ASSOCIATED WITH INITIATORS AT LOW DECAY HEAT LEVELS ARE PROBABLY TOO HIGH
- METHOD OF BINNING IS CONSERVATIVE BECAUSE OF COARSE MESH

Mixing Issue In Secondary Depressurizations

- Work To Date Indicates Loop Stagnation Only At Decay Heat Levels Below 0.5% For Small Depressurization (ie .I.E. .11 Ft^2 Area).

- When A No Mixing Sort Is Being Performed, If Good Mixing Is In Question, The Scenario Is Assigned To A Potential Initiation Bin.

CONTROL SYSTEM FAILURES

THE COOLDOWN SCENARIOS WHICH FOLLOW ARE TAKEN FROM COMPREHENSIVE CONTROL SYSTEM FAILURE ANALYSES WHICH HAVE BEEN SUBMITTED AND REVIEWED BY U. S. NRC.

TREATMENT OF CONTROL SYSTEM FAILURES IN PROBABILISTIC PTS WORK

- SPECIFIC EFFORT MADE TO INCLUDE CONTROL SYSTEM FAILURE MODES IN PROBABILISTIC PTS WORK
- INDEPENDENT, REDUNDANT PROTECTION SYSTEMS SPECIFICALLY PROTECT AGAINST SUSTAINED COOLDOWN CAUSED BY STEAM DUMP, FEEDWATER OR ROD CONTROL SYSTEMS
- NO SUSTAINED* COOLDOWN CAUSED BY STEAM DUMP, FEEDWATER OR ROD CONTROL SYSTEM IS KNOWN TO HAVE EVER OCCURED ON A W PLANT

* I. E. PROTECTION SYSTEM FAILURE TO FUNCTION AS DESIGNED TO TERMINATE COOLDOWN

CONTROL SYSTEM	CONTROL STATE PINCH POINT	INCLUDED IN PRA RESULTS ?	INDEPENDENT PROTECTION SYSTEMS DESIGNED TO PREVENT SUSTAINED COOLDOWN
1. STEAM DUMP	MECHANICAL VALVE STICKING	YES	2 TRAINS SLI
	CONTROL SIGNAL FAILURE	YES	2 TRAINS SLI, 2 TRAINS SDI (LO - LO TAVE)
2. SECONDARY PORV'S	MECHANICAL VALVE STICKING OPEN	YES	NONE
	CONTROL SIGNAL FAILURE OPEN	YES	NONE
3. ROD CONTROL	SUSTAINED INWARD MOTION OF ROD BANK(S). STEAM DUMP MAY ACTIVATE IN SOME CASES	YES	2 TRAINS REACTOR TRIP (LOW PZR PRESSURE)
4. FEEDWATER CONTROL	EXCESSIVE FEED	YES	2 TRAINS FPT/FWI (HI - HI LEVEL, OR SI OR LOW TAVE IN COINCIDENCE WITH REACTOR TRIP
	SLOW SG OVERFILL/STEAM BREAK	NO	2 TRAINS FPT/FWI (HI-HI LEVEL)
5. PRESSURIZER PRESSURE	PORV MECHANICAL STICKING OPEN	YES	NONE
	PORV CONTROL SIGNAL OPEN	YES	NONE

SDI = STEAMDUMP ISOLATION

SLI = STEAMLINE ISOLATION

FPT = FEEDPUMP TRIP

FWI = FEEDWATER ISOLATION

(RCPs Tripped, Perfect Mixing, No Combination Events Involving Other [illegible] Cooldown Types)

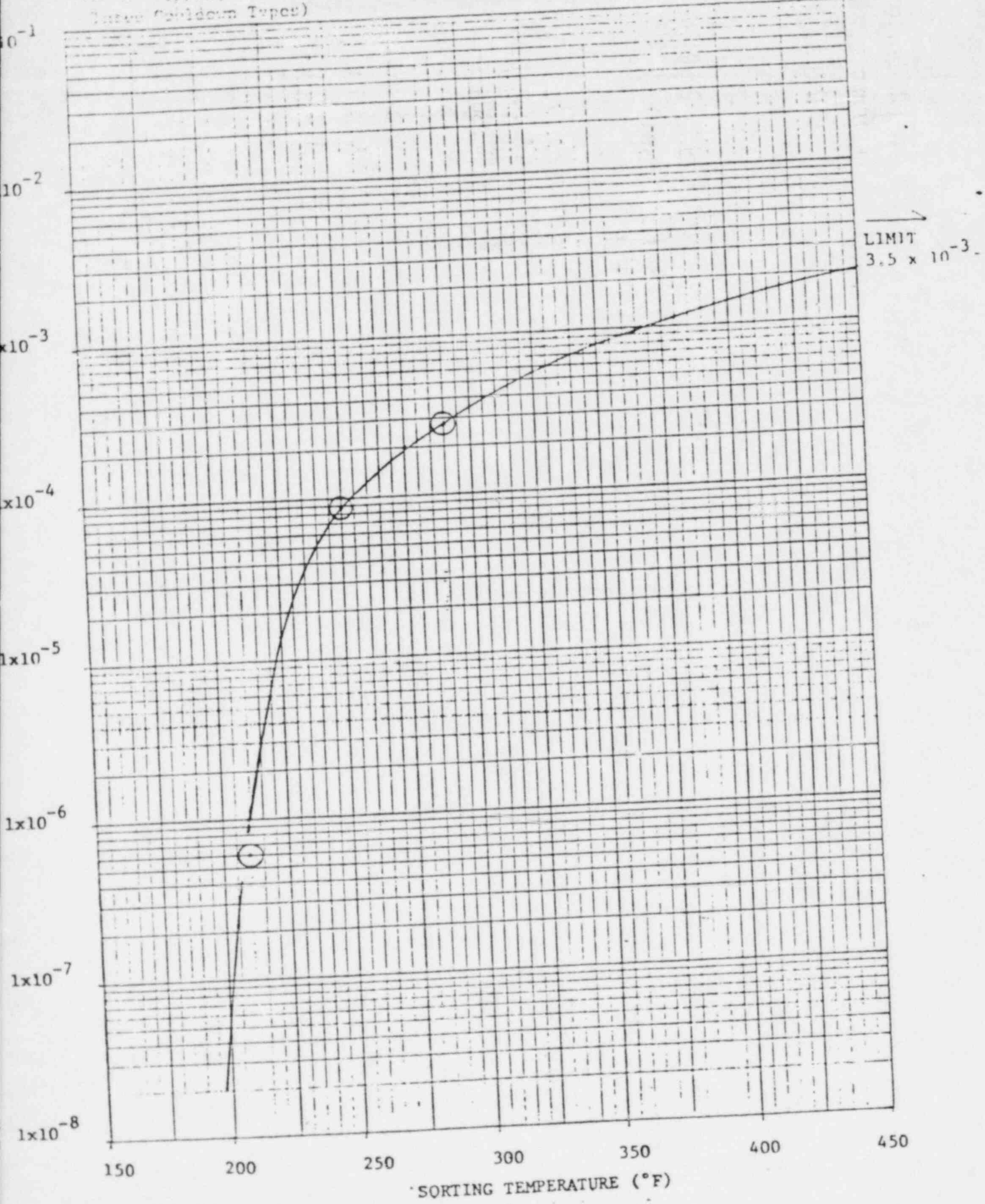


Table II.4.2.4-1 [redacted] Excessive Feedwater Addition

	AFW \ Oper Act.	Decay Heat .GE. 1.0% P=0.26 5 min	Decay Heat .GE. 1.0% P=0.11 10 min	Decay Heat .GE. 1.0% P=0.01 30 min	Decay Heat .GE. 1.0% P=0.000 60 min	1.0% .GT. Decay Heat .GE. 0.5% P=0.043 5 min	1.0% .GT. Decay Heat .GE. 0.5% P=0.018 10 min	1.0% .GT. Decay Heat .GE. 0.5% P=0.002 30 min	1.0% .GT. Decay Heat .GE. 0.5% P=0.000 60 min	Decay Heat = 0% P=0.027 5 min	Decay Heat = 0% P=0.012 10 min	Decay Heat = 0% P=0.000 30 min	Decay Heat = 0% P=0.000 60 min
MFW=125%	1000 gpm	NI	NI	NI	NI	NI	NI	NI	NI	NI	NI	NI	NI
	1500 gpm	NI	NI	NI	NI	NI	NI	NI	NI	NI	NI	NI	PI
	2000 gpm	NI	NI	NI	NI	NI	NI	NI	PI	NI	NI	NI	PI
	2500 gpm	NI	NI	NI	PI	NI	NI	NI	PI	NI	NI	NI	PI
MFW=150%	1000 gpm	NI	NI	NI	NI	NI	NI	NI	NI	NI	NI	NI	NI
	1500 gpm	NI	NI	NI	NI	NI	NI	NI	NI	NI	NI	NI	PI
	2000 gpm	NI	NI	NI	NI	NI	NI	NI	PI	NI	NI	NI	PI
	2500 gpm	NI	NI	NI	PI	NI	NI	NI	PI	NI	NI	NI	PI
MFW=175%	1000 gpm	NI	NI	NI	NI	NI	NI	NI	NI	NI	NI	NI	NI
	1500 gpm	NI	NI	NI	NI	NI	NI	NI	NI	NI	NI	NI	PI
	2000 gpm	NI	NI	NI	NI	NI	NI	NI	NI	PI	NI	NI	PI
	2500 gpm	NI	NI	NI	PI	NI	NI	NI	NI	PI	NI	NI	PI

Note: Probability values shown at top of columns are multiplied by 10000.

Total probability of vessel crack initiation due to excessive feedwater addition is less than 10^{-7} per reactor year.

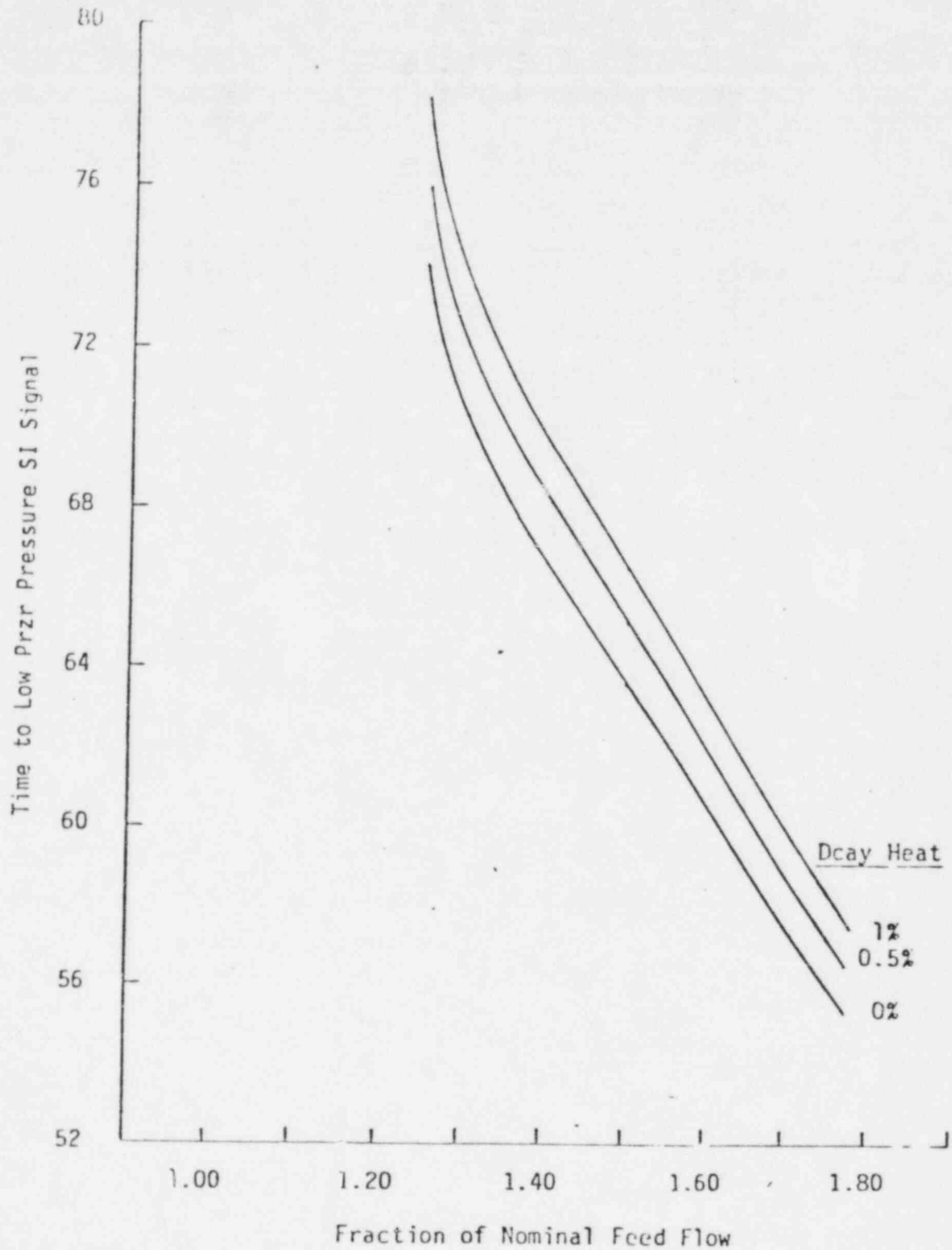


FIGURE II.4.2.1-1 TIME IN SECONDS TO LOW PRESSURIZER PRESSURE SI SIGNAL VS. MAIN FEED FLOWRATE - EXCESSIVE FEED-WATER TRANSIENT

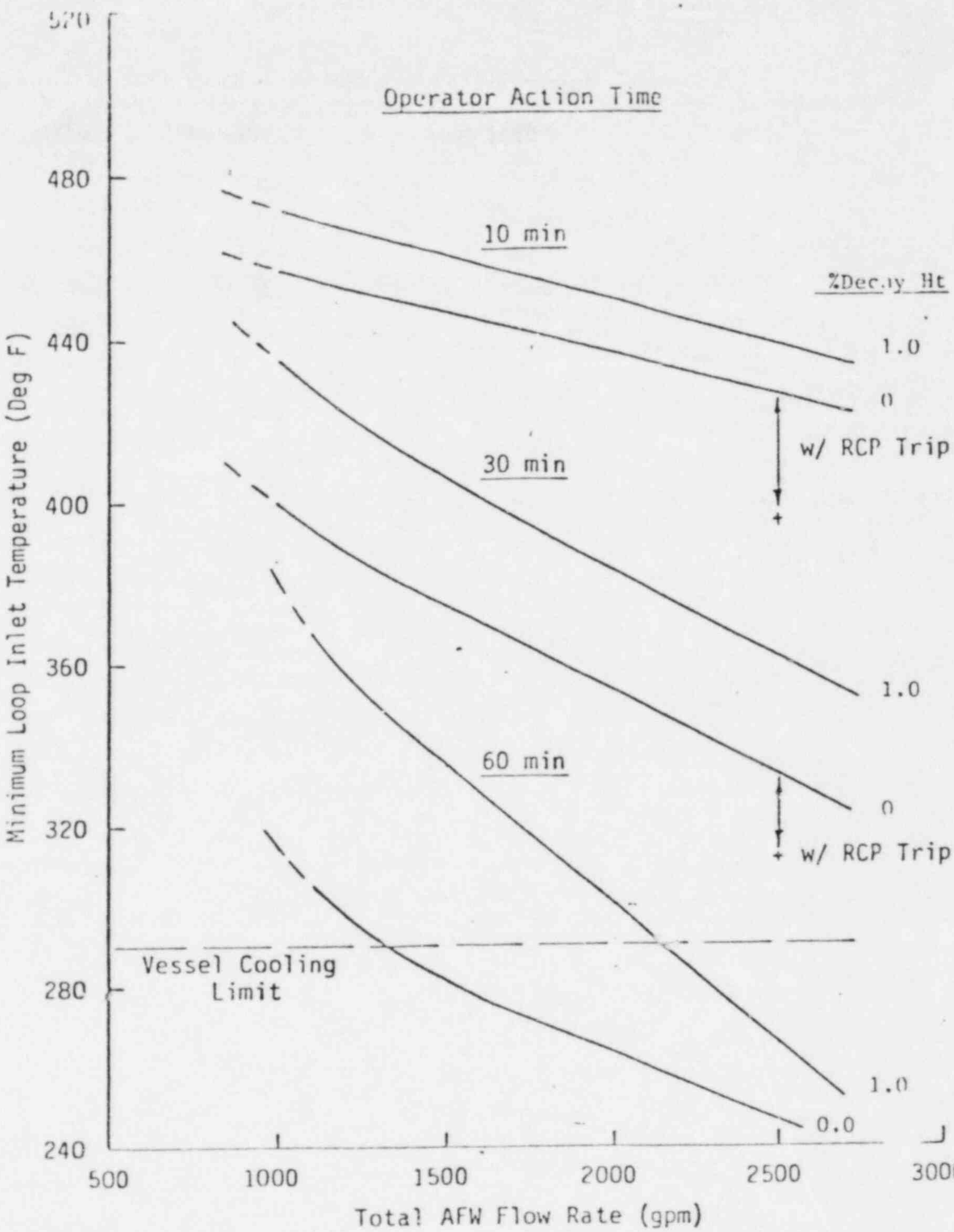


FIGURE II.4.2.1 - 3 MINIMUM LOOP INLET TEMPERATURE VS. AUXILIARY FEEDWATER FLOW - EXCESSIVE FEEDWATER TRANSIENT

REACTOR VESSEL INTEGRITY IMPACT ON PROCEDURES

* Consideration for all ERG development
* Provisions explicitly incorporated in E-2 guideline
* Based on recent RVI work all ERG will be re-reviewed
* RCS integrity function restoration guideline development
* Coverage at next WOG ERG seminar
* Generic training material development

INTENT OF REVIEW

* SHOW PRESENT STEPS ADDRESS PTS

* IDENTIFY ADDITIONAL STEPS TO ADDRESS PTS

CRITERIA OF REVIEW

* RAPID TEMPERATURE REDUCTION
* RAPID PRESSURIZATION
* CHANGE IN LOOP FLOW

METHOD OF REVIEW

* ADDRESS EACH STEP
  - COMMISSION
  - OMISSION
  - INCORRECT PERFORMANCE
* SUBCOMMITTEE REVIEW OF ANY CHANGES

FORM OF RESULTS

* JUSTIFICATION OF PRESENT STEPS

  - NO OR MINOR IMPACT

  - IMPACT OUTWEIGHED BY OTHER CONCERNS

* POSSIBLE STEP ADDITIONS, CHANGES, DELETIONS

* EMPHASIS ON OPERATOR TRAINING

PRELIMINARY

ERG PRESSURIZED THERMAL SHOCK REVIEW

EP	EXPECTED RESPONSE COMMENT	EXPECTED RESPONSE RESOLUTION	RESPONSE NOT OBTAINED COMMENT	RESPONSE NOT OBTAINED RESOLUTION
:	Unrelated item, no impact		Same as expected response	
	SI is required for core cooling.	Diagnosis of possible SI and remedial actions occur at a later time.	Same as expected response	
	AFW flow to any depressurized SG will lead to an excessive cooldown rate.			
	Diagnosis of containment status.	Event diagnosis and remedial actions are performed later. (See E-2)	Failure of MSIV's to close automatically may result in failure to terminate an uncontrolled cooldown. The potential impact is mitigated by the action of manual closure of MSIVs. Containment isolation Phase B results in loss of CCW therefore RCP's are stopped.	
	This step verifies RCS heat removal to no-load Tavg.	Excessive heat removal could be indicative of a primary or secondary break. Event diagnosis and remedial action are performed later.	Same as expected response.	
			Refers to Step 27	
	These steps check for spurious SI actuation. They have no thermal shock implication.			

1.0 In WCAP-10019, an analysis of an isolatable LOCA was conducted to study the effect of operator response time on vessel integrity to demonstrate that sufficient time exists to allow operator isolation of the PORV before the vessel is challenged. The presentation in WCAP-10019 shows that if the operator isolates the PORV in 30 minutes the vessel would not be challenged, but does not show that 30 minutes is sufficient time for the operator to take action. Provide justification that 30 minutes is sufficient time for correct operator action. Provide an evaluation of the sensitivity to 30 minute operator time.

RESPONSE:

The 30 minute operator action time was selected to be a very conservative maximum time for operator action in the case of PORV isolation. Proposed ANS Standard 58.8 recommends an operator action time of 10 minutes for this type of event. Based on Post-TMI improvements on emergency procedures and equipment, the isolation time would realistically be less than 10 minutes.

The sensitivity to the 30 minute operator action time is obtained from similar calculations originally presented in WCAP-10019. Allowable pressures for fracture were generated based on the transient temperature profile (conservatively assumed to be constant in time after operator action). These results are outlined and discussed below.

(Later)

2.0 In the WCAP-10019 Small Steam Line Break analysis, it is not stated what you assumed regarding operator action but we believe that you assumed that the operator terminates AFW and injection flow at 10 minutes. Provide an evaluation of the sensitivity of the time assumed for operator action (i.e., if the operator acts at 5 minutes, or 10 minutes, or 20 minutes later than you assumed, what are the resulting pressure/temperature transients and how does this affect the EFPY remaining?).

RESPONSE:

The existing WCAP-10019 curves are based upon actual steam break incidents which have occurred in operating plants. The lowest temperatures and maximum pressures experienced during these transients, with added margin, are bounded by the WCAP-10019 Small Steam Break curves. No single operator action time can be associated with the curves.

Please refer to the probabilistic work on secondary depressurizations included within this report for a sensitivity study of operator action time to terminate uncontrolled auxiliary feed flow.

The secondary depressurization results in Table III.2.1 and probabilistic treatment of secondary depressurizations contained within this report, do not take any credit for termination of safety injection. Thus, all results reflect worst case repressurization as far as SI is concerned. In a plant specific probabilistic analysis, operator action at various times would be considered and would provide less severe results.

3.1 Evaluate the ECC mixing in the cold leg and downcomer.

RESPONSE:

LOCA ANALYSES

The version of the NOTRUMP code assumes nodes are in thermodynamic equilibrium, therefore stratified temperature distributions within nodes are not allowed. Current fracture mechanics analyses were performed with bounding mixing assumptions; perfect or thermodynamic mixing of SI and loop flows as calculated by NOTRUMP was assumed as long as natural circulation loop flow was predicted to exist. Detailed hydrodynamic analyses, presented in WCAP-10019, have shown that as long as loop flow exists, SI and loop flows will mix in the cold leg. No mixing of SI and loop flows was assumed to occur when natural circulation loop flow was predicted to be lost. It should be noted that the code calculated mixing assumption was also evaluated by fracture mechanic analyses which showed that the results were essentially the same as or better than the no-mixing assumption. For an explanation of how NOTRUMP determines the node fluid properties, please refer to WCAP-10076.

STEAM BREAK ANALYSIS

The LOFTRAN and MARVEL codes assume nodes which are in thermodynamic equilibrium (except for the pressurizer). Non-mixing situations are handled as they were done for LOCA when it is specified that non-mixing is being considered for a particular analysis. Otherwise perfect mixing is assumed.

Note that even when RCPs are tripped, mixing is generally not a concern for higher probability small secondary depressurizations (i.e., with higher decay heat levels) because natural circulation remains strong much longer than the time that it takes for RCS pressure to reach SI shutoff head. In no decay heat, pumps tripped cases, non-affected loop flow will be stagnate, or reverse for several minutes and stagnate, at some time after about 10 minutes.

3.2 Evaluate the non-symmetrical temperature distribution in the downcomer (resulting from blowdown of a steam generator).

RESPONSE:

For small LOCA, the downcomer control volume was not divided azimuthally, only axially, therefore a non-symmetric temperature distribution in this region was not obtained. For small LOCA the most extreme situation was evaluated assuming no-mixing which yielded acceptable fracture mechanics results.

For small steam break, the downcomer temperature is assumed to be uniformly at a temperature corresponding to the coldest loop inlet temperature (generally, the affected loop). This assumption conservatively bounds the non-symmetrical temperature distribution which actually would exist in the downcomer.

3.3 Evaluate the primary-to-secondary heat transfer (and reverse heat transfer).

RESPONSE:

LOCA

The primary to secondary heat transfer is dependent upon the primary fluid pressure, temperature and flow, the tube metal material properties, and the secondary temperature, pressure and quality. For a description of how NOTRUMP evaluates the nodal heat transfer, refer to WCAP-10076.

STEAM BREAKS

The overall forward heat transfer coefficient in the LOFTRAN and MARVEL codes consists of the four major thermal resistances. The primary side convection heat transfer resistance (Dittus-Boelter correlation used), the tube metal thermal resistance, the fouling resistance, and the secondary side boiling heat transfer resistance (Jens and Lottes correlation used). The various resistances change as a function of changing fluid properties as compared to nominal conditions which are input into the code. No credit is taken for reduced heat transfer area as a function of water level for steam break analysis.

The reverse heat transfer coefficient in MARVEL is input in tabular form as a function of time. In LOFTRAN, reverse heat transfer for all resistances is identical to forward heat transfer, except that the secondary film resistance is computed based upon a natural convection correlation recommended by McAdams.

Please refer to WCAP-7635 (MARVEL Code Description) and WCAP-7878 (LOFTRAN Code Description) for additional details.

Steam generator reverse heat transfer was not accounted for in the design basis large steam break analyses contained in Table III.2.1 of WCAP-10019. It is accounted for in the probabilistic work which is currently underway.

4.1 Provide the references for the mixing computer programs.

RESPONSE:

A reference for the VARR-II code was given in WCAP-10019; CRBR-WARD-0106. This report is available from the Technical Information Center (TIC) of Oak Ridge, Tennessee.

4.2 Provide a description of any changes made in order to treat water as the working fluid.

RESPONSE:

In it's original form, VARR-II employed a polynomial curve fit to the transport properties of water as a function of temperature at atmospheric pressure. As stated on page 12 of WCAP-10019, the polynomial curve-fit equations were replaced by calls to the appropriate ASME steam table subroutines. As the steam table subroutines require both pressure and temperature inputs to evaluate thermal transport properties of water, the input to the VARR-II code was expanded to accept a pressure versus time table for use by the steam table subroutines. Pressure values at required computational times were calculated from the input table using standard interpolation methods.

4.3 Provide a description of the input data used to perform the mixing analysis. Of particular importance are any coefficients supplied by the user to treat convective terms.

RESPONSE:

Definition of the mesh dimensions for the VARR-II model used in the analyses are given in WCAP-10019, as are the assigned pressure, temperature, and flow boundary conditions. The VARR-II reference, CRBR-WARD-0106, provides recommended and required values for the turbulence models incorporated in the code. These recommended and required values were used in the analysis.

4.4 Provide a description of the limitation of the mixing model (with respect to relative velocities, fluid states, and flow regime, eg. annular, slug, stratified).

RESPONSE:

The computational scheme of the VARR-II code assumes the following:

- o The working fluid is a single-phase liquid.
- o The working fluid is incompressible.
- o The working fluid is a continuum.
- o The working fluid has no free surfaces.

A VARR-II model is a two-dimensional representation of a flow field; the model geometry defines the flow regime. The purpose of the hydraulic computation scheme in VARR-II is to evaluate bouyancy - induced flow stratification.

4.5 Provide verification of the mixing program with suitable experimental data covering a wide range of injection and coolant flow rates as might be expected for potential PTS scenarios.

RESPONSE:

There are several publications comparing both the computational models employed in the code and the results of the code itself to experimental data:

- The transport equations for time-dependent determination of turbulent energy and viscosity have been tested over a wide range of flows (J. H. Stuhmiller, "Development and Validation of a Two-Variable Turbulence Model," SAI-74-509-LJ, January 1974).

- The VARR-II reference, CRBR-WARD-0106, gives a comparison of VARR-II calculations and experimental measurements for turbulent pipe flow (J. Laufer, "The Structure of Turbulence in Fully Developed Pipe Flow," NACA R. 1174, 1954).

- VARR-II mixing predictions have been shown to be in reasonable agreement with experimental data for the case of several jets injecting into a pool (E. H. Novendstern, J. C. Reese, M. J. Budden, "Prediction of the CRBR Outlet Plenum Transient Response following a Reactor Trip," Paper 77-TH-30, Joint AIChE-ASME Heat Transfer Conference, Salt Lake City, UTHA, August 1977).

4.6 Provide a description of the method used to determine when 'non-mixing' needs to be considered to a PTS overcooling event.

RESPONSE:

For the LOCA analyses presented in WCAP-10019, 'perfect mixing' between SI and loop flows was assumed for as long as loop flow was predicted to exist. 'Perfect mixing' is defined as the mixed SI and loop flow attaining thermodynamic equilibrium temperature. When loop flow was predicted to cease (the breaking of natural circulation), it was assumed that there was no mixing between SI flow and fluid in the cold legs. The perfect mixing assumption for loop flow is a reasonable assumption that was substantiated by two-dimensional hydrodynamic analyses performed to test the assumption. The no-mixing assumption for no loop flow was established as a conservative condition that establishes a lower bound on fluid temperatures during a hypothetical PTS.

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

March 8, 1982

Dockets Nos: (All Operating Reactors with Westinghouse NSSS)

LICENSEES: All licensees represented by Westinghouse Owners' Group (WOG) and in particular Southern California Edison Company, Carolina Power & Light Company, and Florida Power & Light Company (the licensees of San Onofre 1, Robinson 2 and Turkey Point 4 respectively).

SUBJECT: SUMMARY OF MEETING WITH WESTINGHOUSE OWNERS' GROUP, SOUTHERN CALIFORNIA EDISON COMPANY, CAROLINA POWER & LIGHT COMPANY, AND FLORIDA POWER & LIGHT COMPANY CONCERNING THE PRESSURIZED THERMAL SHOCK ISSUE.

Introduction

This meeting was held in Bethesda, Maryland, at the request of the NRC staff to discuss (1) the Westinghouse generic report on Pressurized Thermal Shock (PTS), WCAP-10019, "Summary Report on Reactor Vessel Integrity for Westinghouse Operating Plants" and (2) the "150 day" responses concerning San Onofre 1, Robinson 2, and Turkey Point 4 with the WOG and the three Westinghouse NSSS owners who received our August 21, 1981 letter concerning PTS. We had previously provided the requested participants with an agenda and our concerns related to the above identified reports (Enclosure 1).

The meeting followed an agenda (Enclosure 2) which deviated from the published agenda. The attendees for the meeting are identified in Enclosure 3. Visual aides for the WOG presentations are included in Enclosure 4.

The licensee's of the Robinson 2, San Onofre 1 and Turkey Point 4 had no presentation.

Summary of Discussions

An overall emphasis was made that the staff needed an understanding of the technical basis, the sensitivities and uncertainities in the operator actions, the fracture analyses, and the thermal hydraulic, mixing and systems analyses. Therefore, additional information is necessary to aid the staff's assessment of the PTS issue for the Commission in Summer 1982. To be useful, the requested information should be received by the end of April - first week of March time frame.

The WOG discussion regarding our concerns included the following highlights:

PDR

B-2

1. General

The WOG analyses only considered the more probable events and design bases events. The reason for the approach was that it was believed that for any transient considered the final end point in the analysis would be a design basis event.

2. Emergency Guide Lines Development

a) WOG considers existing guidelines and procedures do consider PTS and are adequate.

b) The comprehensive and integrated guideline development (for NUREG-0737 Item I.C.1) will consider PTS and will be completed by September 1982. The guidelines will be reviewed by WOG with regard to PTS.

c) For at least Robinson 2, the guidelines, as they relate to PTS, have been implemented in procedures.

3. Warm Prestressing (WP) and Fracture Analysis

a) General impression was that WP was applicable, particularly when considered properly in the course of a transient.

b) Perturbations of pressure (P) and temperature (T) during the course of a transient do not necessarily negate credit for W.P. Perturbation in P stresses and T stresses tend to cancel each other.

c) Cladding is inconsequential in fracture analysis.

d) Flaw size assemed was conservative

e) The crack arrest concept is valid and is based on a good data base.

4. Thermal-Hydraulic and mixing analyses.

a) For the SBLOCA, mixing was not considered. This was considered conservative.

5. Discussion on Operation Transients

Studies of I&C systems indicate no failure (single failure?) pose a challenge to the RPV.

6. Fluence Calculations

a) The uncertainties in fluence calculations was based on observations and not a statistical data base.

b) WOG believes the actual capsule data are within the $\pm 20\%$ calculated range.

Conclusions

The WOG committed to provide the NOTRUMP code by March 31, 1982 and to provide additional information based on what was requested and what was presented and information which is available by the end of April 1982. However, the WOG did not specifically identify the information which would be provided by the end of April. The WOG requested a formal request for any remaining questions. The WOG and the representatives from San Onofre 1, Robinson 2 and Turkey Point 4 identified contacts within their organizations for the different staff reviewers.

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

Enclosures:
As Stated

MEETING NOTICE DISTRIBUTION

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Enclosure 1

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

February 18, 1982

Dockets Nos. 50-261, 50-251
and 50-206

MEMORANDUM FOR: John Stolz, Chief, Operating Reactors Branch #4, DL

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

SUBJECT: FORTHCOMING MEETING WITH WESTINGHOUSE OWNERS GROUP CAROLINA POWER & LIGHT COMPANY, FLORIDA POWER COMPANY AND SOUTHERN CALIFORNIA EDISON, CONCERNING PRESSURIZED THERMAL SHOCK ISSUE

Time & Date: 8:15am-5:00pm
Wednesday, February 24, 1982

Location: Holiday Inn, Bethesda, MD
3rd floor, Maryland Room

Purpose: To discuss the Westinghouse Generic Report followed by discussions on the plant specific "150 day" responses for Robinson 2, Turkey Point 4 and San Onofre 1. See attached agenda.

Requested Participants: NRC: H. Denton, S. Hanauer, T. Novak, W. Johnston R. Woods, R. Klecker, W. Hazelton, C. Johnson, R. Johnson, J. Clifford, L. Lois, N. Randall, E. Throm, C. Morris, C. Serpan, A. Oxfurth, M. Vagins, J. Strosnider.

Licensee: WOG-Daniel Speyer et al.
Southern California Edison
Carolina Power Company
Florida Power Company

Other: Westinghouse representatives

Guy S Vissing

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

cc: See next page

8203040422
PDR/LPDR

AGENDA FOR
MEETING WITH WESTINGHOUSE OWNERS GROUP
AND
THE WESTINGHOUSE NSSS OWNERS WHO
RECEIVED THE AUGUST 21, 1981 LETTER
CONCERNING
PRESSURIZED THERMAL SHOCK
FEBRUARY 24, 1982

I. Session Concerning Westinghouse

Generic Report WCAP-10019

1. Introduction

2. Summary of Basic Staff Concerns:
   Information is needed regarding sensitivity of remaining EFPY to principal parameters including:
   a) Operator delayed action
   b) Decrease credit for warm prestressing and uncertainities in material perperties.
   c) Decrease thermal mixing credits
   See attachment for detailed concerns

3. Owners Group Responses (Owners Group opportunity to question staff for clarifications and to discuss concerns).

4. Owners group Caucus . if necessary.

5. Identification of what more information will be provided and a schedule for submittals by WOG.

6. Concluding Remarks.

II. Session Concerning 150 Day Responses

1. Introduction

2. Staff concerns (see attachment).

3. Licensee's Responses (opportunity for licensees to question staff for clarifications and discuss concerns.
   a) Robinson 2
   b) Turkey Point 4
   c) San Onofre 1

4. Caucus if necessary

5. Identification of what more information will be provided and schedule for submittals by licensees.

6. Concluding Remarks.

CONCERNS RELATED TO OPERATOR ACTIONS

A. Applicable to Generic Report WCAP-10019

1. In WCAP-10019, an analysis of an isolatable LOCA was conducted to study the effect of operator response time on vessel integrity to demonstrate that sufficient time exists to allow operator isolation of the PORV before the vessel is challenged. The presentation in WCAP-10019 shows that if the operator isolates the PORV in 30 minutes the vessel would not be challenged, but does not show that 30 minutes is sufficient time for the operator to take action. Provide justification that 30 minutes is sufficient time for correct operator action. Provide an evaluation of the sensitivity of 30 minute operator time.

2. In the WCAP-10019 Large Steam Line Break analysis, it is assumed that the operator terminates AFW and injection flow at 10 minutes. Control of AFW does not seem to significantly affect the cooldown. Injection flow termination, however, appears to occur at a time critical to prevent significant repressurization. Provide an evaluation of the sensitivity of the time assumed for operator action (i.e., if the operator acts at 15 minutes, or 20 minutes, or 30 minutes, what are the resulting pressure/temperature transients?).

B. Applicable to Licensees 150 day Responses

3. In the San Onofre 1, the H. B. Robinson 2, and the Turkey Point 3 and 4 evaluations, the actions described do not provide the operator with clear direction for dealing with the conflicting concerns that need to be evaluated when considering the operation of HPI and charging flow as it relates to vessel integrity and maintaining core cooling. Each licensee should provide an evaluation of the need and effectiveness of procedure modification to clearly identify the concerns in the emergency operating procedures themselves, in addition to upgrading operator training.

4. Carolina Power and Light stated that a formal training program will be completed by March 31, 1982, and Southern California Edison indicated that training would be performed in February 1982. The programs described by the two licensees are adequate to address the NRC's concerns for short term action, except that the procedureal guidance provided in the training program may require emergency operating procedure modification to clearly define the conflicting concerns.

5. Florida Power and Light does not consider any procedural modifications or upgraded training programs necessary. The licensee should be advised that operator knowledge of current industry information regarding Pressurized Thermal Shock, and clearly outlined options in emergency operating procedures, could prove very valuable if an unforeseen condition challenges the vessel integrity.

CONCERNS RELATED TO FRACTURE ANALYSIS

Applicable to WCAP 100-19

A. Justification for taking credit for warm prestressing for small break LOCA and other transients. How can it be assured that pressure will not fluctuate up (if the system is repressurized) or down to such an extent that warm prestressing is negated. Both excursions up and large excursions down may negate warm prestressing. Operating experience has shown us that such excursions do in fact occur during transients.

B. Justification for Assumptions:

1. No clad contribution to K_I thermal
2. Semielliptical crack, initially

C. Describe steps in the analysis of small steam break - show curves of:

1. Metal temperature through the wall at critical times
2. K_I, K_{Ic}, and K_{Id} as functions of d/t at critical times, and
3. "Football curves" for fluence values of interest.

Applicable to WCAP 10019 and 150 day responses

- 1. Initial RT_{NDT} - Generic values for typical metal wire and flux types - could generic values be obtained?

CONCERNS RELATED TO SYSTEM ANALYSIS

A. Applicable to Generic Report WCAP-10019

1. Concerning the Thermal-Hydraulic Complete Program

Provide a description of the models used to:

1.1 Evaluate the ECC mixing in t cold leg and downcomer.

1.2 Evaluate the non-symmetrical temperature distribution in the downcomer (resulting from blowdown of a steam generator).

1.3 Evaluate the primary-to-secondary heat transfer (and reverse heat transfer).

1.4 Evaluate how voiding in the primary system and the subsequent collapse of primary system voids is treated.

1.5 Evaluate the repressurization of the primary system.

1.6 Evaluate any other thermal-hydraulic phenomena important to the PTS problem

1.7 Provide a description of the verification of the thermal-hydraulic computer program to applicable experimental data for repressurization and overcooling transients.

1.8 What additional verification is required to demonstrate that the thermal-hydraulic computer program adequately models the phenomena important to PTS evaluations.

2. Concerning the mixing models.

2.1 Provide the references for the mixing computer programs.

2.2 Provide a description of any changes made in order to treat water as the working fluid.

2.3 Provide a description of the input data used to perform the mixing analysis. Of particular importance are any coefficients supplied by the user to treat convective terms.

2.4 Provide a description of the limitation of the mixing model (with respect to relative velocities, fluid states, and flow regime, e.g. annular, slug, stratified).

2.5 Provide verification of the mixing program with suitable experimental data covering a wide range of injection and coolant flow rates, as might be expected for potential PTS scenarios.

2.6 Provide a description of the method used to determine when "non-mixing" needs to be considered to a PTS overcooling event.

B. Applicable to Generic Report and Licensee's "150 day" Responses

1. Concerning Input Data and Assumptions

1.0 Provide a description of the models or data used for:

(a) Heat sources (or sinks),
(b) Decay heat,
(c) ECC and feedwater temperatures (enthalpies) and flow rates,
(d) Primary and secondary relief capacities,
(e) Empirical correlation coefficients used for PTS evaluations,
(f) Operator Actions,
(g) Initial conditions.

1.2 Provide a list of all transients or accidents by class (for example: excessive feedwater, operating transients which result from multiple failures including control system failures and/or operator error, steam line break and small break LOCA) which could lead to inside vessel fluid temperatures of 300 F or lower. Provide any Failure Modes and Effects Analyses (FMEAs) of control systems currently available or reference any such analyses already submitted. Estimate the frequency of occurrence of these events and provide the basis for the estimates. Discuss the assumptions made regarding reactor operator actions.

For a given initiating event, potential multiple and consequences failures need to be considered to identify those transients which could lead to a PTS problem.

1.3 Identify all potential PTS events which have occurred at your facility. Include a designation of the operator actions and identify potential additional failures (including operator) which could have resulted in a more severe event.

CONCERNS RELATED TO IRRADIATION INFORMATION

A. Applicable to Generic Report WCAP-10019

1. WCAP-10019 Summary Report on Reactor Vessel Integrity for Westinghouse Operating Plants. The material covered in this review is (a) Fluence Methodology pp 22-42 and Fluence calculations pp 126-127.

   1.1 p. 26, What is the fundamental cross section set used to generate the 21-group cross sections and the zone dependent spectra? How is the P1 scattering expansion justified, particularly for Fe?

   1.2 p. 29, It is stated that "...radial power distributions applicable to long term operation are derived from a statistical analysis of calculated distributions..." Bias factors are used for the correction of observed to calculated differences. (a) How are these biases applied and how large are they? and
   (b) With the trend to low leakage loadings where will the data base be found to calculate bias factors for the outer limits?

   1.3 p. 30, To what extent is the common axial distribution applicable to 4, 3 and 2-loop plants?

   1.4 p. 30 and p. 38, An uncertainty of ±20% is given. What are the components and how are they combined to yield this overall value?

   1.5 p. 38, Data bases for support of long term core power distribution and measurements in the reactor cavity are mentioned.
   (a) What are these data bases?
   (b) Does the data bases include low leakage loading? and
   (c) (also p. 41) from what W-plants are the reactor cavity measurements?

   1.6 pp 38-41 Saturated activity of Fe-54, Ni-58, Np-237 and U-238 from ten 2-loop and eight 4-loop plant surveillance capsule are used in tables II.2-2 and II.2.3. Are these the only available data?

   1.7 p. 41, Is the uncertainty level of ±20% applicable to the single plant predictive uncertainty?

B. Applicable to Licensee's 150 Day Response

2. The licensee submittals for the W-plants are essentially identical with no plant specific details. The following questions are applicable to H. B. Robinson (HBR), Turkey Point-3 (TP), and San Onofre (SO).

   2.1 The power distributions used for fluence predictions are said to be "statistically based." What is the statistical basis for HPB, TP, SO and how was the plant specific information and data accounted for?

   2.2 The rod by rod power distributions do not reflect plant specific information. Are these distributions bounding so as to be acceptable for a conservative estimate of the fluence?

   2.3 Is the geometic information on the core plant specific with as built demensions?

   2.4 For future low leakage operation will there be a sufficient statistical base data for reliable fluence prediction?

Enclosure 2

ACTUAL AGENDA FOR
MEETING WITH WESTINGHOUSE OWNERS GROUP
AND
WESTINGHOUSE NSSS OWNERS WHO
RECEIVED THE AUGUST 21, 1982 LETTER
CONCERNING
PRESSURIZED THERMAL SHOCK
FEBRUARY 24, 1982

I. Introduction

A. Introductory Comments — Guy Vissing, NRC

B. Background and Purpose of Meeting — S. Hanauer, NRC

II. Discussion of Technical Concerns Related to WCAP 10019

A. Summary of Staff Concerns — Guy Vissing, NRC

1. General Overall Concerns — R. Woods, NRC

2. Concerns Related to Operator Actions — J. Clifford, NRC

3. Concerns Related to Fracture Analysis — R. Klecker, NRC

4. Concerns Related to Thermal-Hydraulic, Mixing and System Analysis — E. Throm, NRC

5. Concerns Related to Irradiations Information and Fluence Calculations — L. Lois, NRC

B. WOG Discussing of Staff Concerns — D. Speyer, WOG

1. WOG Development of Emergency Response Guidelines — H. Julian, Westinghouse

2. Discussion on Warm Prestressing and Fracture Analysis — W. Bamford, Westinghouse

3. Discussion of Transients — J. Rumancik, Westinghouse

4. Return to Discussion on Fracture Analysis Flow Shape, Cladding Effect Crack Arrest Criteria — W. Bamford, Westinghouse

5. Discussion on Mining SBLOCA and Typical Transients — B. Monte, T. Andreycheck, Westinghouse

6. Fluence Calculations — S. Anderson, Westinghouse

Caucus for Staff & WOG	
WOG Response	D. Speyer, WOG
NRC Response	S. Hanauer, NRC
Small Group Discussion of Specific Issue	Staff/ WOG

Enclosure 3

ATTENDANCE LIST
FOR
MEETING WITH WESTINGHOUSE OWNERS GROUP
AND
LICENSEES WHO RECEIVED THE AUGUST 21, 1981 LETTER
CONCERNING
PRESSURIZED THERMAL SHOCK
FEBRUARY 24, 1982

NAME	ORGANIZATION
Guy Vissing	NRC/DL/ORB#4
Daniel Speyer	Con Edison of NY
Ted Meyer	Westinghouse
Oliver Kingsley	Alabama Power Co.
D. Scott Ackerson	Westinghouse
J. A. Rumancik	Westinghouse
D. H. Rawlins	Westinghouse
James Sheppard	Carolina Power & Light Co.
David Waters	Carolina Power & Light Co.
Bruce King	Westinghouse
W. Parker Tomlinson	Carolina Power & Light Co.
Sam Grant	Carolina Power & Light Co.
Bindi Chexal	Electric Power Research Inst.
K. Shiratori	Mitsubishi Heavy Ind. P&H Office
Masafumi Sato	Mitsubishi Heavy Ind.
Michael F. Ahern	Northeast Utilities
Warren Andrews	Southern Co. Services
Roger Newton	Wisconsin Electric Power Col
D. L. Dill	Wisconsin Electric Power Co.
R. C. Kryten	Oak Ridge National Laboratory
Fred Mynatt	Oak Ridge National Laboratory
B. J. Short	B&W
Gtrgg B. Swindlehurst	Duke Power Co.
Tom Novak (Part time)	NRC/DL

NAME	ORGANIZATION
Earl Brown	NRC/AEOD
Carl Johnson	NRC
Fred Manning	NRC/DRA/RES
Fuat Odar	NRC/AE/RES
S. L. Anderson	Westinghouse
Betn A. King	University of Richmond
Joseph DeRoy	Power Authority of the State of NY
Timothy Taylor	Public Service Electric & Gas Co.
D. Basdekas	NRC/RES
Paul C. Paris	D&L
K. H. Cotter	Fracture Proof Design Corp.
Joel Kramer	NRC/DHFS
Tom Lorio	Westinghouse
Dennis Ziemann	NRC/DHFS
C. A. Moerke	Commonwealth Edison
A. P. Rochino	GPU Nuclear
W. H. Bamford	Westinghouse
T. Andreycheck	Westinghouse
B. S. Monty	Westinghouse
Jim Kay	Yankee Atomic Electric Co.
Ausaf Husain	Yankee Atomic Electroc Co.
Mark Stella	Westinghouse
Hugh Thompson	NRC/DHFS
Akram Zahoor	Fracture Proof Design Corp.
Brent Clayton	NRC/DHFS
N. Zuber	NRC/RES
J. A. De Mastry	Florida Power & Light Co.
Darrell Eisenhut (Part time)	NRC/D/DL
[illegible] (Part time)	[illegible]

NAME	ORGANIZATION
Vernon T. Chilson	Florida Power & Light Co.
David W. Lippard	VA Electric & Power Co.
Harry Julian	Westinghouse
Lynn Connor	Doc-Search Associates
Warren Hazelton	NRC/DE/MTEB
William Johnston	NRC/DE
Milton Vagins	NRC/DET/RES
P. Randall	NRC/DE/MTEB
S. J. Bhatt	NRC/DE/MTEB
G. G. Elia	Westinghouse
B. J. Elliot	NRC/DE/MTEB
Jack Strosnider	NRC/DET/RES
Bill Flourney	SCE
Bob Renuart	Bechtel Power Co.
Jim Clifford	NRC/PTRS/DHFS
G. N. Lauben	NRC/DSI/RSB
Ed Throm	NRC/DSI/RSB
Lambros Lois	NRC/DSI/CPB
C. Morris	NRC/DST/RRAB
A. Oxfurth	NRC/DSR/SRS
S. K. Matnavan	Florida Power & Light
S. Hanauer	NRC
Jan S. Teraszkiewicz	PASNY (IP-3) .
Jim Brunetti	PASNY (IP-3)
Roy Woods	NRC/DSR/GIB
R. Klecker	NRC/DE/MTEB
Pat Hinsberg	McGraw-Hill
Richard E. Johnson	NRC/DST/GIB
W. Rutherford	NUTECH

Enclosure 4

WESTINGHOUSE OWNERS GROUP PRESENTATIONS

DISCUSSION ON DEVELOPMENT OF

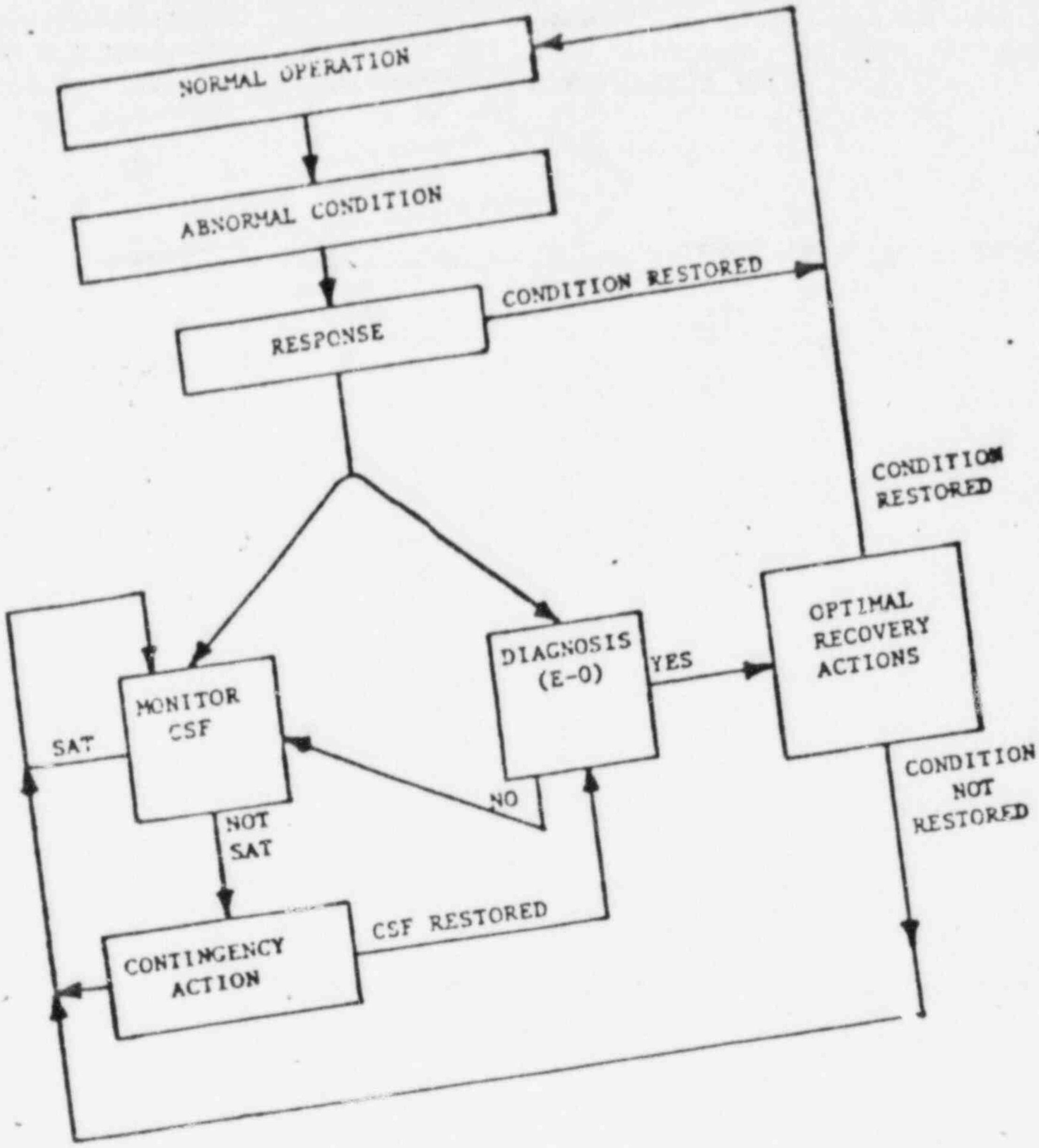
EMERGENCY RESPONSE GUIDELINES

BY

H. JULIAN , WESTINGHOUSE

REACTOR VESSEL INTEGRITY IMPACT ON PROCEDURES

* Consideration for all ERG developement
* Provisions explicitly incorporated in E-2 guideline
* Based on recent RVI work all ERG will be re-reviewed
* RCS integrity function restoration guideline developement
* Coverage at next WOG ERG seminar
* Generic training material developement



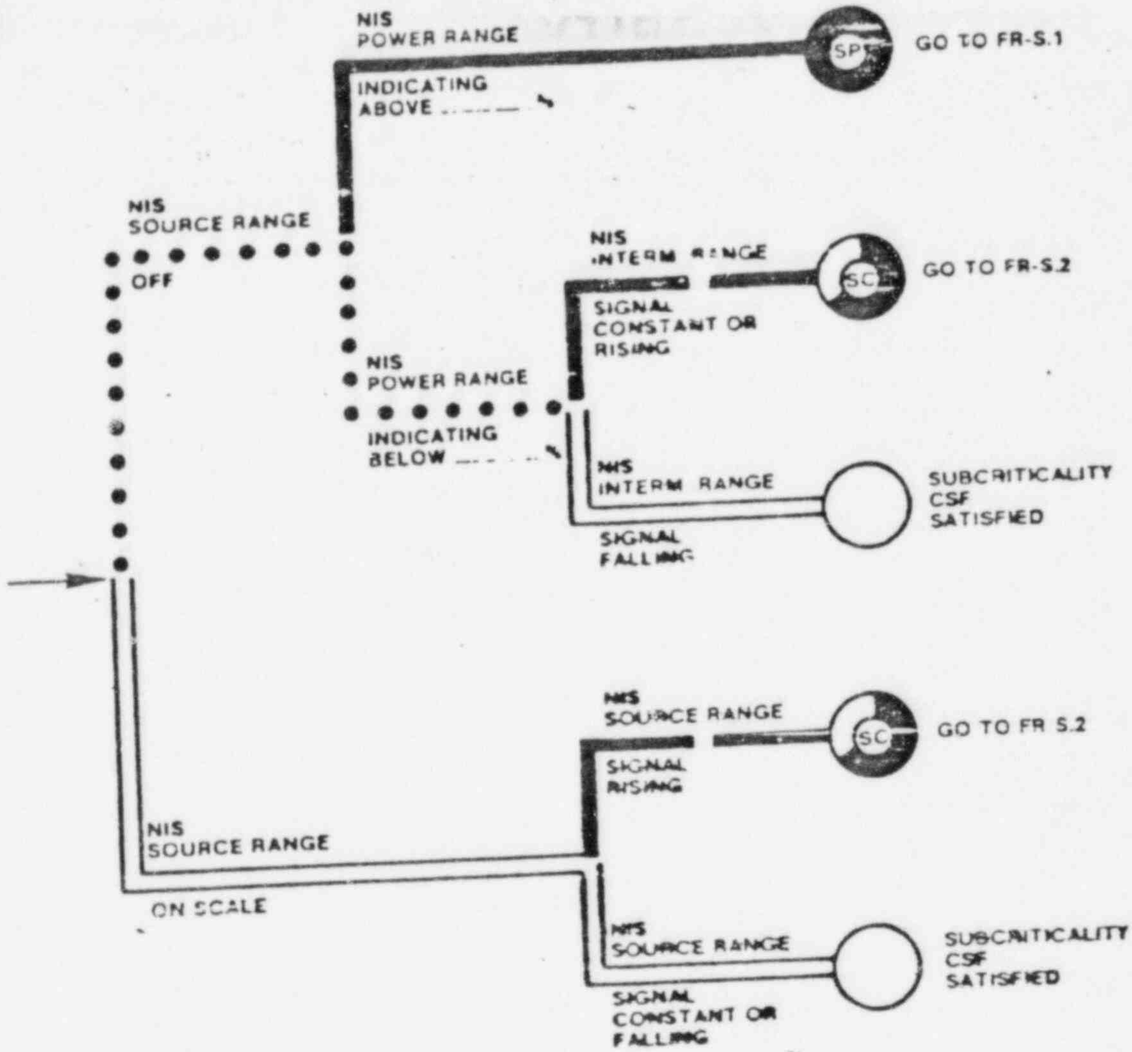
COORDINATED USE OF EMERGENCY RESPONSE GUIDELINES

FIGURE 3

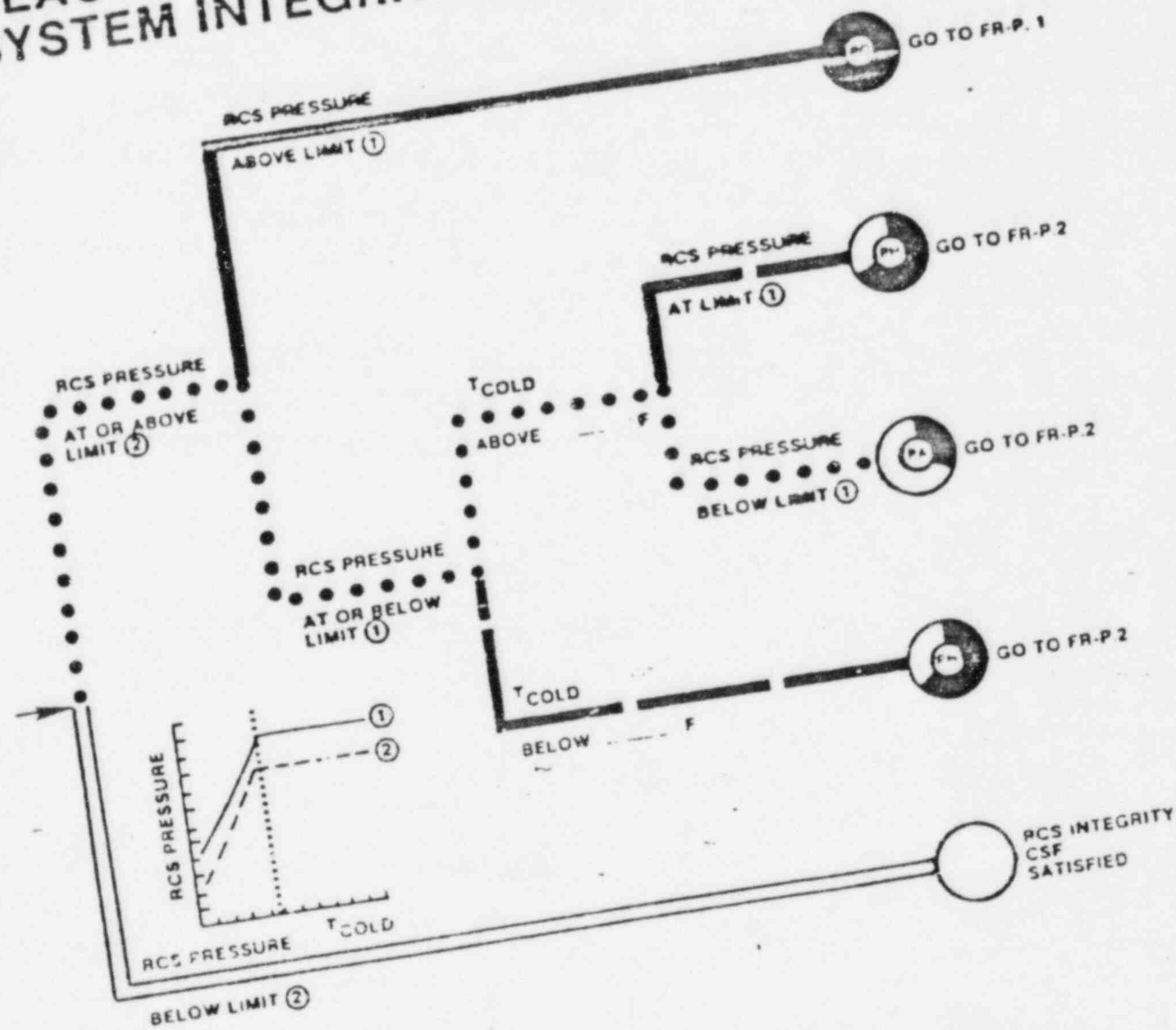
REACTOR VESSEL INTEGRITY IMPACT ON PROCEDURES

- Consideration for all ERG developement
- Provisions explicitly incorporated in E-2 guideline
- Based on recent RVI work all ERG will be re-reviewed
- RCS integrity function restoration guideline developement --
- Coverage at next WOG ERG seminar
- Generic training material developement

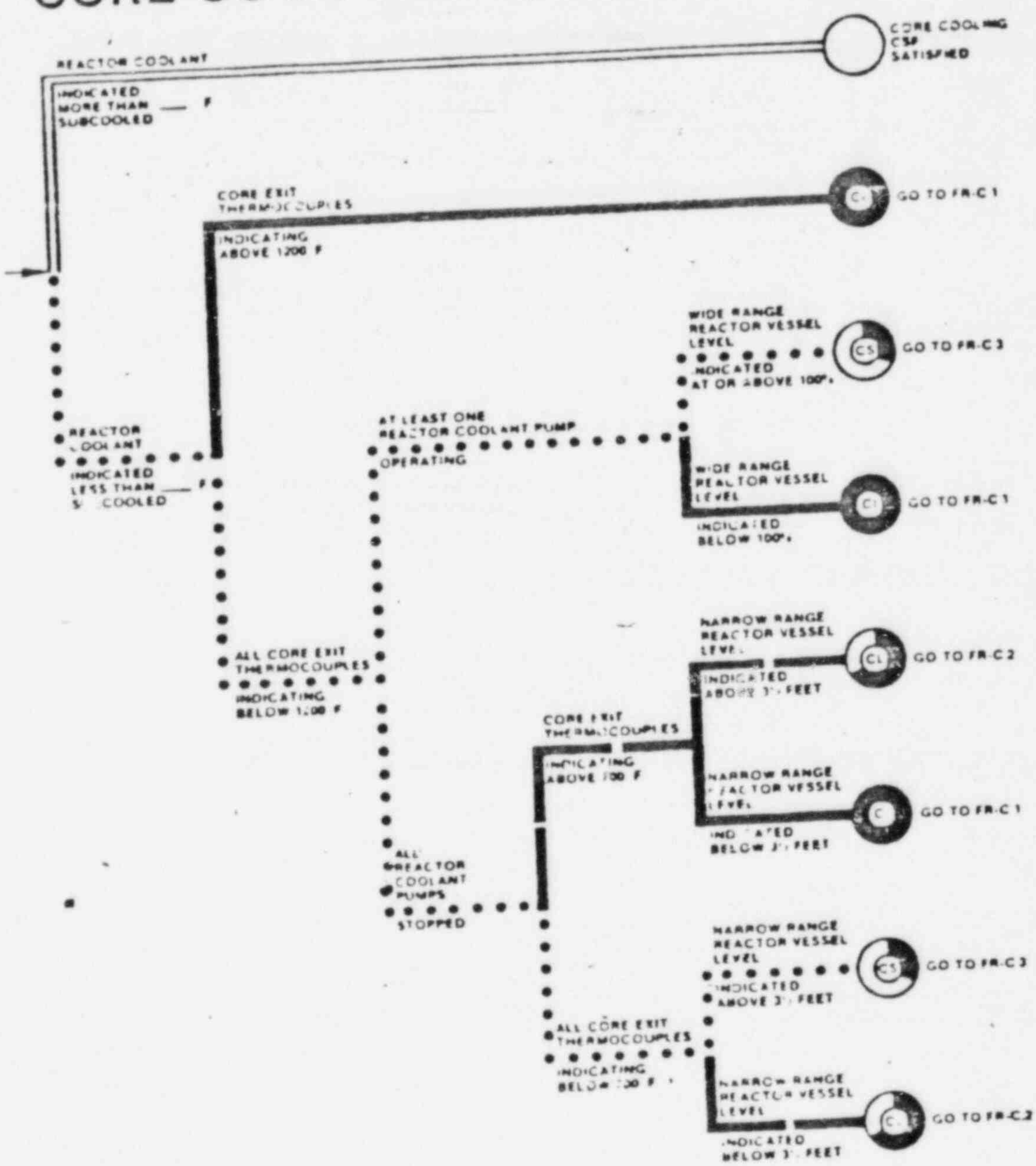
SUBCRITICALITY



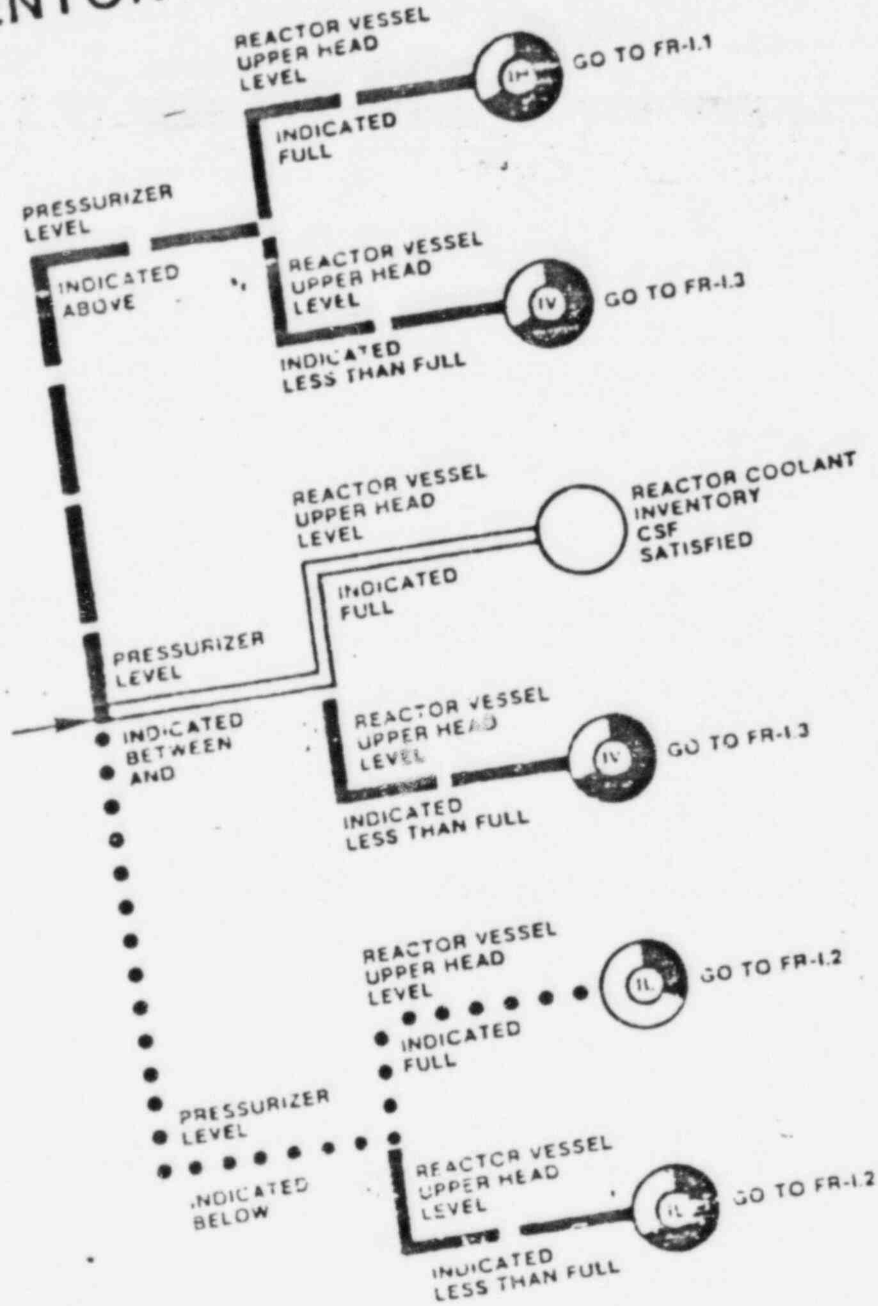
REACTOR COOLANT SYSTEM INTEGRITY



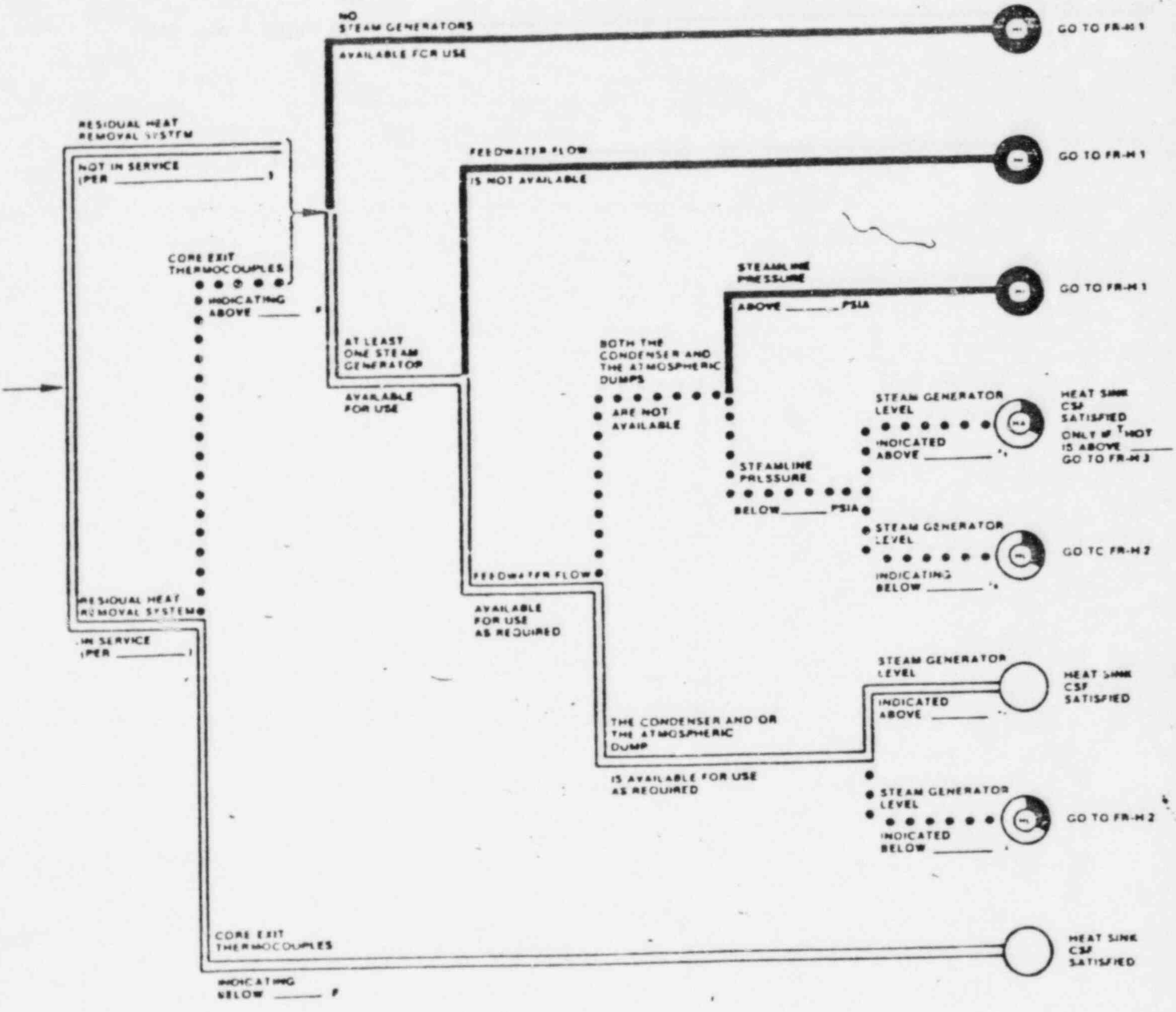
CORE COOLING



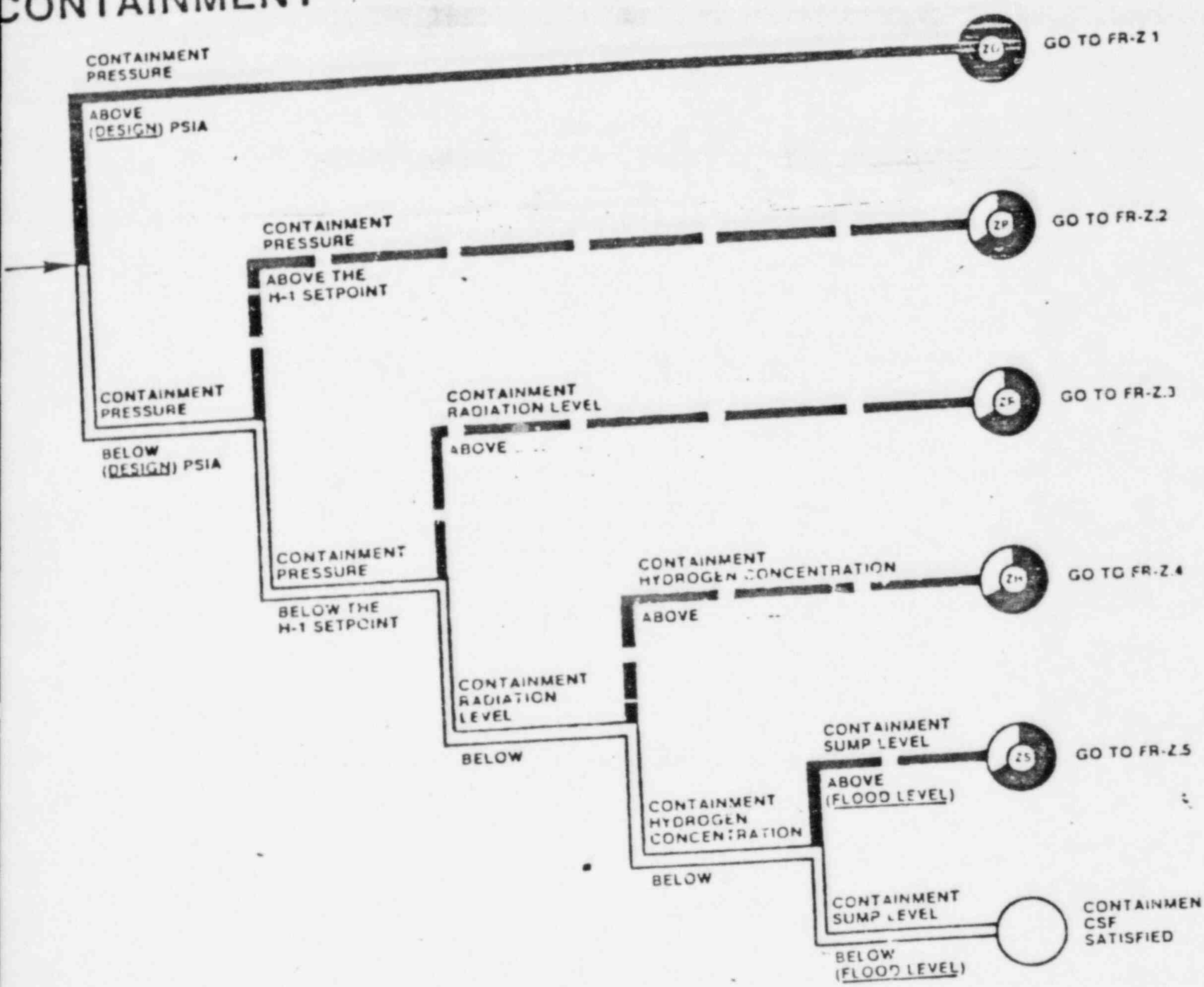
REACTOR COOLANT INVENTORY



HEAT SINK


NO STEAM GENERATORS
AVAILABLE FOR USE
GO TO FR-H 1
RESIDUAL HEAT REMOVAL SYSTEM
NOT IN SERVICE (PER ______)
FEEDWATER FLOW
IS NOT AVAILABLE
GO TO FR-H 1
CORE EXIT THERMOCOUPLES
INDICATING ABOVE ______ F
STEAMLINE PRESSURE
ABOVE ______ PSIA
GO TO FR-H 1
AT LEAST ONE STEAM GENERATOR
AVAILABLE FOR USE
BOTH THE CONDENSER AND THE ATMOSPHERIC DUMPS
ARE NOT AVAILABLE
STEAM GENERATOR LEVEL
INDICATED ABOVE ______
HEAT SINK CSF SATISFIED
ONLY IF T_{HOT} IS ABOVE ______
GO TO FR-H 3
STEAMLINE PRESSURE
BELOW ______ PSIA
STEAM GENERATOR LEVEL
INDICATING BELOW ______
GO TO FR-H 2
FEEDWATER FLOW
AVAILABLE FOR USE AS REQUIRED
RESIDUAL HEAT REMOVAL SYSTEM
IN SERVICE (PER ______)
STEAM GENERATOR LEVEL
INDICATED ABOVE ______
HEAT SINK CSF SATISFIED
THE CONDENSER AND OR THE ATMOSPHERIC DUMP
IS AVAILABLE FOR USE AS REQUIRED
STEAM GENERATOR LEVEL
INDICATED BELOW ______
GO TO FR-H 2
CORE EXIT THERMOCOUPLES
INDICATING BELOW ______ F
HEAT SINK CSF SATISFIED

CONTAINMENT



EMERGENCY RESPONSE GUIDELINE
WOG PROGRAM DEVELOPMENT SCHEDULE

To be submitted on or about October 20, 1981 to the NRC:

1. Reformatted Optimal Recovery Guidelines

E-0 Reactor Trip or Safety Injection

- ES-0.1 Reactor Trip Recovery
- ES-0.2 Natural Circulation Cooldown
- ES-0.3 SI Termination Following Spurious Safety Injection

E-1 Loss of Reactor Coolant

- ES-1.1 SI Termination Following Loss of Reactor Coolant
- ES-1.2 Post-LOCA Cooldown and Depressurization
- ES-1.3 Transfer to Cold Leg Recirculation Following Loss of Reactor Coolant
- ES-1.4 Transfer to Hot Leg Recirculation

E-2 Loss of Secondary Coolant

- ES-2.1 SI Termination Following Loss of Secondary Coolant
- ES-2.2 Transfer to Cold Leg Recirculation Following Loss of Secondary Coolant

E-3 Steam Generator Tube Rupture

- ES-3.1 SI Termination Following Steam Generator Tube Rupture
- ES-3.2 Alternate SGTR Cooldown
- ES-3.3 SGTR with Secondary Depressurization

ECA-1 Anticipated Transient Without SCRAM

ECA-2 Loss of All ac Power

- ECA-2.1 Loss of All ac Power Recovery Without SI Required
- ECA-2.2 Loss of All ac Power Recovery With SI Required

ECA-3 SGTR Contingencies

2. Reformatted Function Restoration Guidelines

- FR-C.1 Response to Inadequate Core Cooling
- FR-I.3 Response to Voids in Reactor Vessel
- FR-H.1 Response to Loss of Secondary Heat Sink

3. Complete Background Packages for All Guidelines Listed Above

4. Complete Critical Safety Function Status Tree Set with Guidance for Utilization

To be submitted by mid-1982 to NRC:

1. Remaining Functional Response Guidelines

FR-S.1	Response to Nuclear Power Generation
FR-S.2	Response to Loss of Core Shutdown
FR-P.1	Response to RCS Overpressurization
FR-P.2	Response to High RCS Pressure
FR-C.2	Response to Potential Loss of Core Cooling
FR-C.3	Response to Saturated Core Cooling Conditions
FR-I.1	Response to Pressurizer Flooding
FR-I.2	Response to Low System Inventory
FR-H.2	Response to Low Steam Generator Level
FR-H.3	Response to Loss of Normal Steam Dump Capability
FR-Z.1	Response to Containment Above Design Pressure
FR-Z.2	Response to High Containment Pressure
FR-Z.3	Response to High Containment Radiation Level
FR-Z.4	Response to High Hydrogen Concentration in Containment
FR-Z.5	Response to Containment Flooding

2. Remaining Event-related Contingency Guidelines

Loss of Cold Leg Recirculation

Multiple Steam Generator Blowdown

Secondary Side Rupture Without High Head SI

Additional SGTR Contingencies

a. Unisolable SGTR
b. Loss of RCS depressurization capability for SGTR
c. Loss of Secondary Side cooldown capability for SGTR
d. Improper SI operation during SGTR

3. Final PRA Evaluation Report

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
17	**Terminate SI:**	
	a. Go to ES-3.1, SI TERMINATION FOLLOWING STEAM GENERATOR TUBE RUPTURE	
18	**Check If Condenser Can Be Used:**	
	a. Condenser - AVAILABLE	a. IF condenser not available, THEN attempt to restore condenser. IF condenser can NOT be restored, THEN evaluate if releases from faulted steam generator will exceed 10 CFR 20 limits. IF 10 CFR 20 limits will be exceeded, THEN cooldown per ES-3.1, SGTR ALTERNATE COOLDOWN.
19	**Verify Adequate Shutdown Margin**	Borate, as necessary.

Caution *Steps 20 through 23 must be performed simultaneously to avoid loss of pressurizer level control.*

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
20	**Initiate RCS Cooldown To 350°F:**	
	a. Maintain cooldown rate - LESS THAN 50°F/HR	
	b. Dump steam from non-ruptured steam generators to condenser 1) [Enter plant specific steps]	b. Dump steam with non-ruptured steam generator PORVs.

Caution *Charging and letdown flows should be compared to determine if leakage between the RCS and ruptured steam generator is stopped.*

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
21	**Maintain Pressurizer Level In Normal Operating Range:**	
	a. Operate charging and letdown, as necessary	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5	**Verify Adequate Shutdown Margin.**	Borate as necessary.
6	**Initiate RCS Cooldown:**	
	a. Maintain cooldown rate – LESS THAN 100°F/HR	
	b. Dump steam to condenser: [Enter plant specific list]	b. Dump steam with steam generator PORVs.
	c. Maintain steam generator narrow range level – AT (1) %	c. Throttle AFW flow as necessary.

Caution *RCS subcooling must be maintained greater than (2) °F during any RCS depressurization.*

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7	**Try To Restore Pressurizer Level Above 20%:**	
	a. Maintain RCS subcooling – GREATER THAN (2) °F	a. Continue dumping steam.
	b. Reduce pressurizer pressure with normal spray	b. Use one pressurizer PORV. <u>IF</u> pressurizer PORVs <u>NOT</u> available, <u>THEN</u> use auxiliary spray.
	c. Pressurizer level – GREATER THAN 20%	c. Perform steps 9 and 10. <u>WHEN</u> level reaches 20%, <u>THEN</u> do step 8.
8	**Increase Pressurizer Temperature:**	
	a. Energize heaters	
	b. Restore temperature to 50°F above core exit TCs	
	c. Maintain temperature – GREATER THAN 50°F ABOVE CORE EXIT TCs	

(1) Enter plant specific value corresponding to no-load steam generator level including allowances for post accident transmitter errors and reference leg process errors.

(2) Enter sum of temperature and pressure measurement system errors translated into temperature using saturation tables.

Number:	Symptoms/Title:	Revision No./Date
ECA-3	SGTR CONTINGENCIES	Basic 1 Sept. 1981

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
21	**Depressurize RCS To Establish Pressurizer Level:**	
	a. Open one pressurizer PORV	a. IF no PORV available, THEN use auxiliary spray.
	b. WHEN pressurizer level is greater than 40%, THEN stop RCS depressurization	
	c. Verify PORV - CLOSED	c. Close PORV block valve.
22	**Verify Adequate Shutdown Margin.**	Borate, as necessary.
23	**Continue Cooldown To Cold Shutdown:**	
	a. Maintain cooldown rate - LESS THAN 100°F/hr	
	b. Dump steam to condenser	b. Dump steam with steam generator PORVs.
	1) (Enter plant specific steps)	
	c. Cooldown using RHR system, if in service	
24	**Check If RHR System Can Be Placed In Service:**	
	a. RCS hot leg temperatures, - LESS THAN 350°F	a. IF greater than 350°F, THEN go to Step 25.
	b. RCS pressure - LESS THAN 400 psig	b. IF greater than 400 psig, THEN go to Step 25.
	c. Place RHR system in service per (plant specific procedure)	

E-2	LOSS OF SECONDARY COOLANT (Cont.)	Basic 1 Sept. 1981

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9	**Check CST Level:**	
	a. CST level - GREATER THAN (1) %	a. IF CST level low, THEN switch to alternate AFW water supply.
10	**Check Containment Spray System:**	
	a. Spray pumps - RUNNING	a. IF pumps NOT running, THEN go to step 11.
	b. Containment pressure - LESS THAN (2) PSIA	b. IF pressure high, THEN maintain containment spray until containment pressure is reduced to normal range.
	c. Reset containment spray signal	
	d. Stop containment spray pumps and place in standby 1) [Enter plant specific steps]	
11	**Check RWST Level:**	
	a. RWST level - GREATER THAN (3)	a. IF less than (3), THEN align SI system for cold leg recirculation per ES-2.2, TRANSFER TO COLD LEG RECIRCULATION FOLLOWING LOSS OF SECONDARY COOLANT.
12	**Check RCS Hot Leg Temperatures:**	
	a. ALL RCS hot leg temperatures - GREATER THAN 350°F	a. IF any RCS hot leg temperature less than 350°F, THEN go to step 18.

(1) Enter plant specific low level setpoint.

(2) Enter plant specific value.

(3) Enter plant specific value corresponding to RWST switchover alarm in plant specific units.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
21	**Depressurize RCS To Establish Pressurizer Level:**	
	a. Open one pressurizer PORV	a. IF no PORV available, THEN use auxiliary spray.
	b. WHEN pressurizer level is greater than 40%, THEN stop RCS depressurization	
	c. Verify PORV - CLOSED	c. Close PORV block valve.
22	**Verify Adequate Shutdown Margin.**	Borate, as necessary.
23	**Continue Cooldown To Cold Shutdown:**	
	a. Maintain cooldown rate - LESS THAN 100°F/hr	
	b. Dump steam to condenser	b. Dump steam with steam generator PORVs.
	1) (Enter plant specific steps)	
	c. Cooldown using RHR system, if in service	
24	**Check If RHR System Can Be Placed In Service:**	
	a. RCS hot leg temperatures, - LESS THAN 350°F	a. IF greater than 350°F, THEN go to Step 25.
	b. RCS pressure - LESS THAN 400 psig	b. IF greater than 400 psig, THEN go to Step 25.
	c. Place RHR system in service per (plant specific procedure)	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
18	**Check If SI Can Be Terminated (RCS Hot Leg Temperature Below 350°F):**	
	a. RCS pressure - GREATER THAN 700 PSIG AND STABLE OR INCREASING	a. DO NOT TERMINATE SI. Return to step 11.
	b. Pressurizer level - GREATER THAN 20%	b. DO NOT TERMINATE SI. Return to step 11.
	c. RCS subcooling - GREATER THAN _(1)_ °F	c. DO NOT TERMINATE SI. Return to step 11.
	d. Secondary heat sink: 1) Total AFW flow to non-faulted steam generators - GREATER THAN _(2)_ GPM -OR- 2) Narrow range level in at least one non-faulted steam generator - GREATER THAN _(3)_ %	d. IF neither condition is satisfied, THEN DO NOT TERMINATE SI. Return to step 11.
19	**Terminate SI:**	
	a. Go to ES-2.1, SI TERMINATION FOLLOWING LOSS OF SECONDARY COOLANT	

— END —

(1) Enter sum of temperature and pressure measurement system errors translated into temperature using saturation tables.
(2) Enter plant specific value derived from background document to E-O.
(3) Enter plant specific value which is above top of steam generator U-tubes.

PRESSURIZER PORV ISOLATION

* ANALYSIS PERFORMED TO SHOW THAT FOR HIGHER PROBABILITY EVENT OPERATOR ACTION WOULD HAVE NO IMPACT ON THE REACTOR VESSEL INTEGRITY

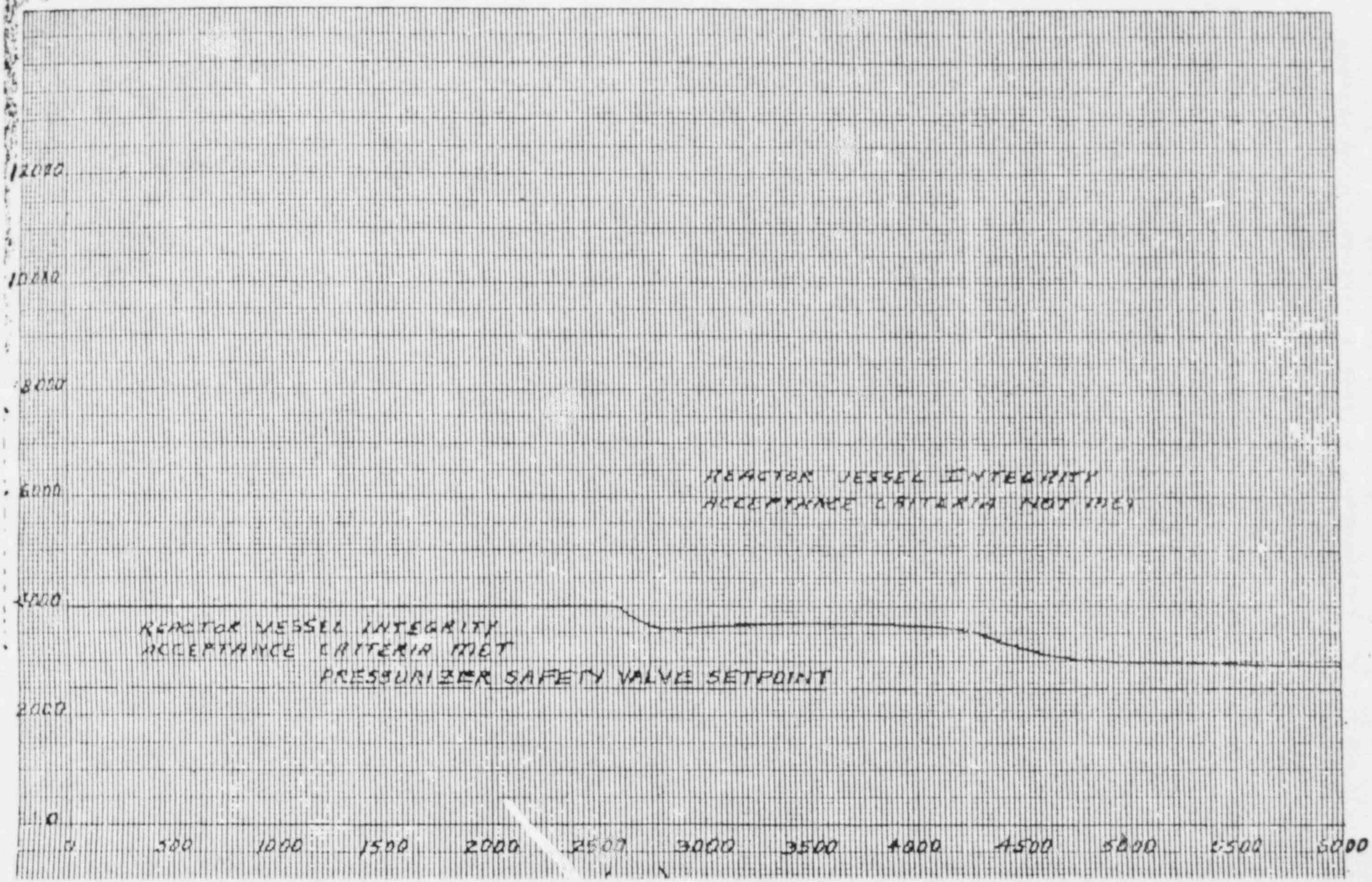
* CONSERVATIVE ANALYSIS (NOT BEST ESTIMATE)

  - WORST MATERIAL PROPERTIES
  - END OF LIFE
  - NO WARM PRESTRESSING
  - APPLICATION OF EXISTING TRANSIENT RESULTS

PRESSURIZER PORV ISOLATION (CON'D)

* PRESSURIZER PORV IS CHECKED IN E-0 FOLLOWING IMMEDIATE ACTIONS AND PRECEEDING DIAGNOSIS

* PRESSURIZER PORV IS RECHECKED EARLY IN ORG'S AND FRG'S



12000
10000
8000
6000
4000
2000
0
0
500
1000
1500
2000
2500
3000
3500
4000
4500
5000
5500
6000
REACTOR VESSEL INTEGRITY
ACCEPTANCE CRITERIA NOT MET
REACTOR VESSEL INTEGRITY
ACCEPTANCE CRITERIA MET
PRESSURIZER SAFETY VALVE SETPOINT

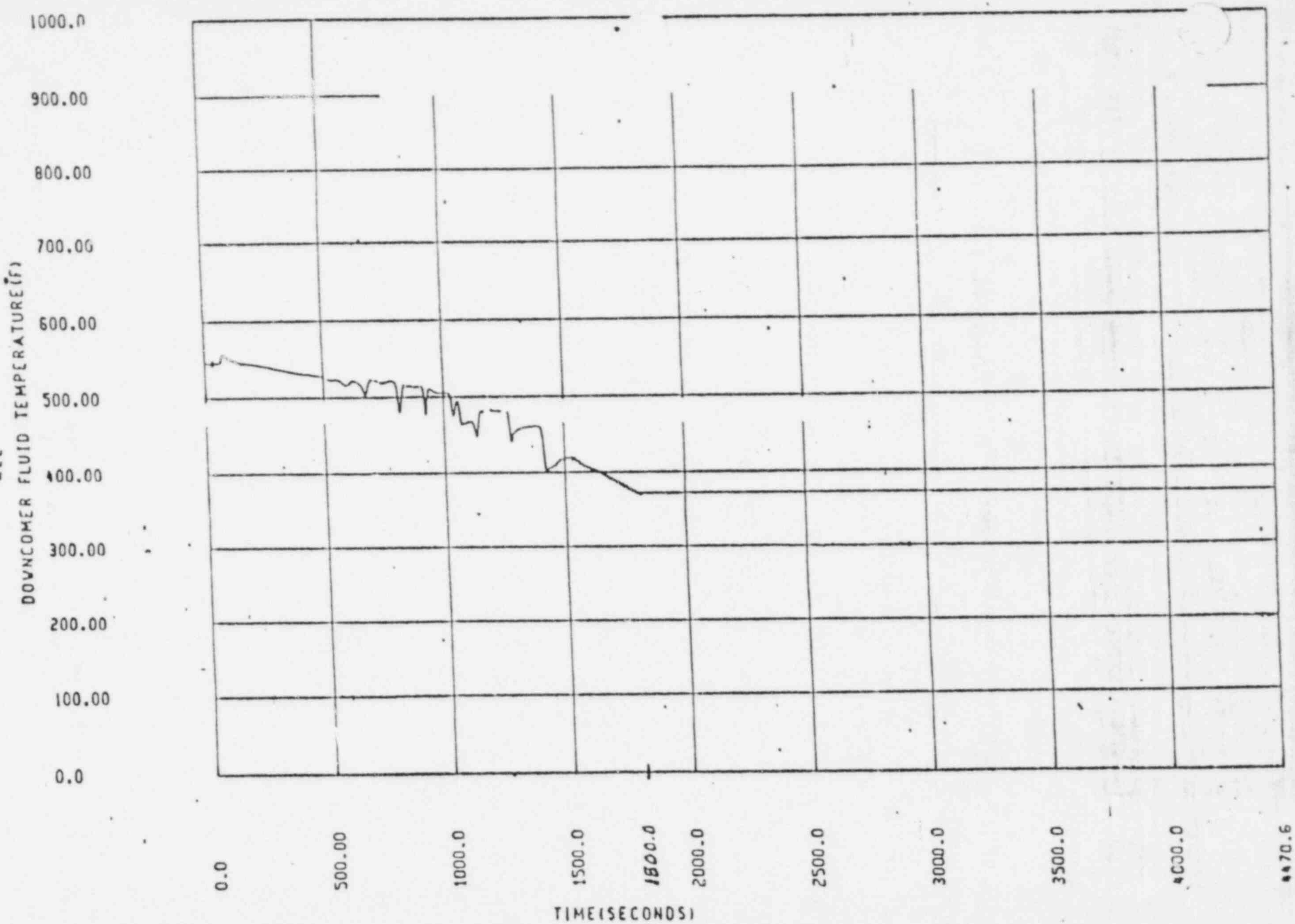


Figure III.1-33 Downcomer Fluid Temperature PORV Analysis

DOWNCOMER TEMPERATURE TRANSIENT FOR OPEN PORV ANALYSIS (PORV CLOSED AT 30 MINUTES BY OPERATOR)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3	**Check Pressurizer PORV Block Valves:** a. Power available to block valves b. Block valves – OPEN	a. Restore power to block valves. b. Open block valve unless it was closed to isolate a faulty PORV.
4	**Check Pressurizer PORVs:** a. PORVs – CLOSED	a. Manually close PORVs. <u>IF</u> any valve cannot be closed, <u>THEN</u> manually close its block valve.

Caution
- *IF any pressurizer PORV opens because of high RCS pressure, repeat step 4 after pressure drops below PORV setpoint.*
- *Seal injection flow should be maintained to all RCPs.*

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5	**Check If RCPs Should Be Stopped:** a. SI running – CHECK FOR FLOW OR PUMP BREAKER INDICATOR LIGHTS LIT • Charging/SI –OR– • High-head SI	a. DO NOT STOP RCPs. Go to step 6.
	b. RCS pressure – EQUAL TO OR LESS THAN (1) PSIG	b. DO NOT STOP RCPs. Go to step 6
	c. Stop all RCPs	

(1) Enter plant specific value derived from background document to E-0.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
27	**Check If RCS Depressurization Can Be Stopped:**	
	a. Pressurizer spray valves – CLOSED	a. Manually close valves.
	b. Pressurizer PORVs – CLOSED	b. Manually close valves. IF any valve cannot be closed, THEN manually close its block valve.

Caution *Seal injection flow should be maintained to all RCPs.*

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
28	**Check If RCPs Should Be Stopped:**	
	a. SI running – CHECK FOR FLOW OR PUMP BREAKER INDICATOR LIGHTS LIT • Charging/SI –OR– • High-head SI	a. DO NOT STOP RCPs. Go to step 29.
	b. RCS pressure – EQUAL TO OR LESS THAN (1) PSIG	b. DO NOT STOP RCPs. Go to step 29.
	c. Stop All RCPs	

(1) Enter plant specific value, derived from background document.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	NOTE *Foldout page should be open.*	
1	**Check RWST Level:** a. RWST level – SLOWLY DECREASING	a. IF RAPIDLY decreasing, THEN go to step 9.
2	**Check Containment Sump Level:** a. Containment sump level – INCREASING	a. IF NOT increasing, THEN rediagnose event, go to E-0, REACTOR TRIP OR SAFETY INJECTION, STEP 29.
	Caution *Alternate water sources for AFW pumps will be necessary if CST level is low.*	
3	**Check Steam Generator Levels:** a. Narrow range level – GREATER THAN (1) %	a. IF less than (1) %, THEN maintain full AFW flow until narrow range level is greater than (1) %.
	b. Throttle AFW flow to maintain narrow range level at (2) %.	b. IF narrow range level in one steam generator continues to increase, THEN go to E-3, STEAM GENERATOR TUBE RUPTURE.
4	**Check Pressurizer PORV Block Valves:** a. Power available to block valves	a. Restore power to block valves.
	b. Block valves – OPEN	b. Open block valve unless it was closed to isolate a faulty PORV.

(1) Enter plant specific value showing level just in the narrow range including allowances for normal channel accuracy, post-accident transmitter errors and reference leg process errors.

(2) Enter plant specific value corresponding to no-load steam generator level including allowances for post-accident transmitter errors and reference leg process errors.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	NOTE *Foldout page should be open.*	
1	**Verify Main Steamline Isolation:**	
	a. Main steamline isolation valves - CLOSED	a. Manually close valves.
	b. Main steamline isolation bypass valves - CLOSED	b. Manually close valves.
2	**Check Pressurizer PORV Block Valves:**	
	a. Power available to block valves	a. Restore power to block valves.
	b. Block valves - OPEN	b. Open block valve unless it was closed to isolate a faulty PORV.
3	**Check Pressurizer PORVs:**	
	a. PORVs - CLOSED	a. <u>IF</u> RCS pressure less than 2335 psig, <u>THEN</u> manually close PORVs. <u>IF</u> any valve cannot be closed, <u>THEN</u> manually close its block valve. <u>IF</u> block valve cannot be closed, <u>THEN</u> go to E-1, LOSS OF REACTOR COOLANT.

Caution. *If any pressurizer PORV opens because of high RCS pressure, repeat step 3 after pressure drops below PORV setpoint.*

LARGE STEAMLINE BREAK OPERATOR ACTIONS

*BETTER ESTIMATE VS CONSERVATIVE ANALYSIS

1) TEMPERATURE TRANSIENT IS MUCH LESS SEVERE
2) LACK OF SENSITIVITY TO AFW ISOLATION IS NOT REAL FOR CASE WITH DECAY HEAT
3) WITH DECAY HEAT, HOT LEG TEMPERATURE WOULD STAY ABOVE 320 DEG-F
4) FRACTURE MECHANICS RESULTS ARE ACCEPTABLE.

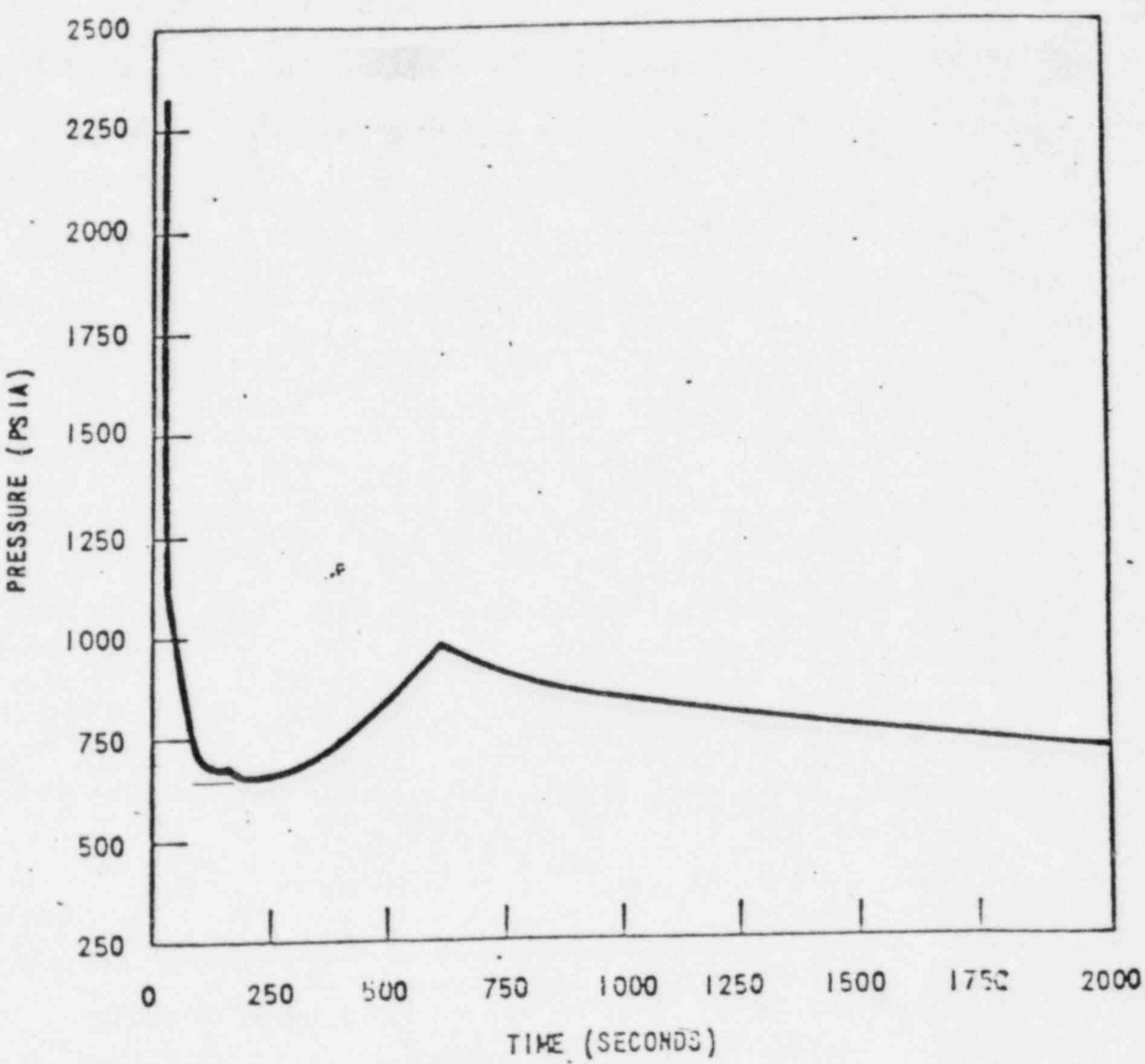
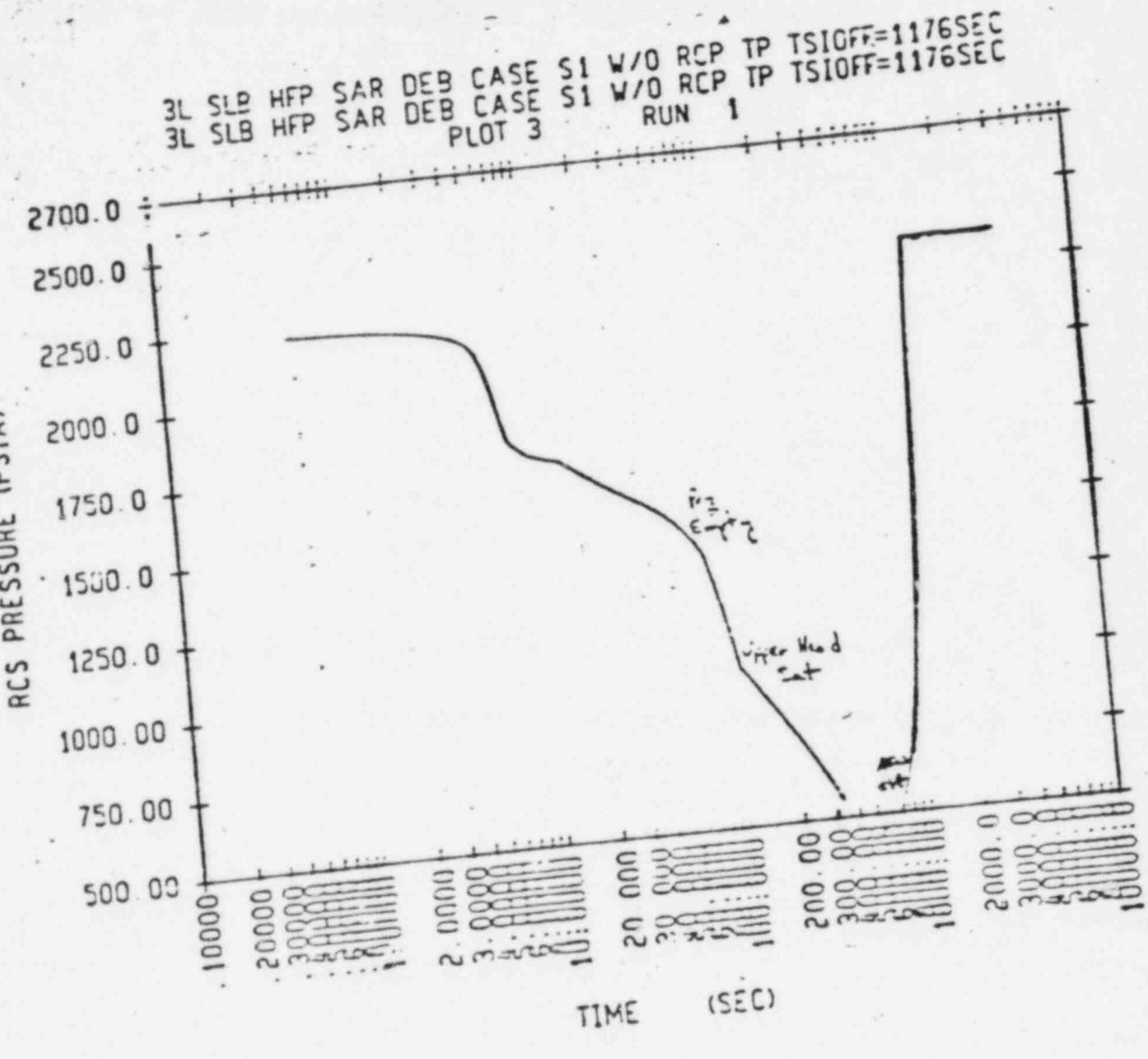


Figure III.1-27 Large Steamline Break With Reactor Coolant Pumps Running. Reactor Coolant Pressure Versus Time


3L SLB HFP SAR DEB CASE S1 W/O RCP TP TSIOFF=1176SEC
3L SLB HFP SAR DEB CASE S1 W/O RCP TP TSIOFF=1176SEC
PLOT 3 RUN 1
RCS PRESSURE (PSIA)
2700.0
2500.0
2250.0
2000.0
1750.0
1500.0
1250.0
1000.00
750.00
500.00
Upper Head Sat
TIME (SEC)

E-2

12.093-5

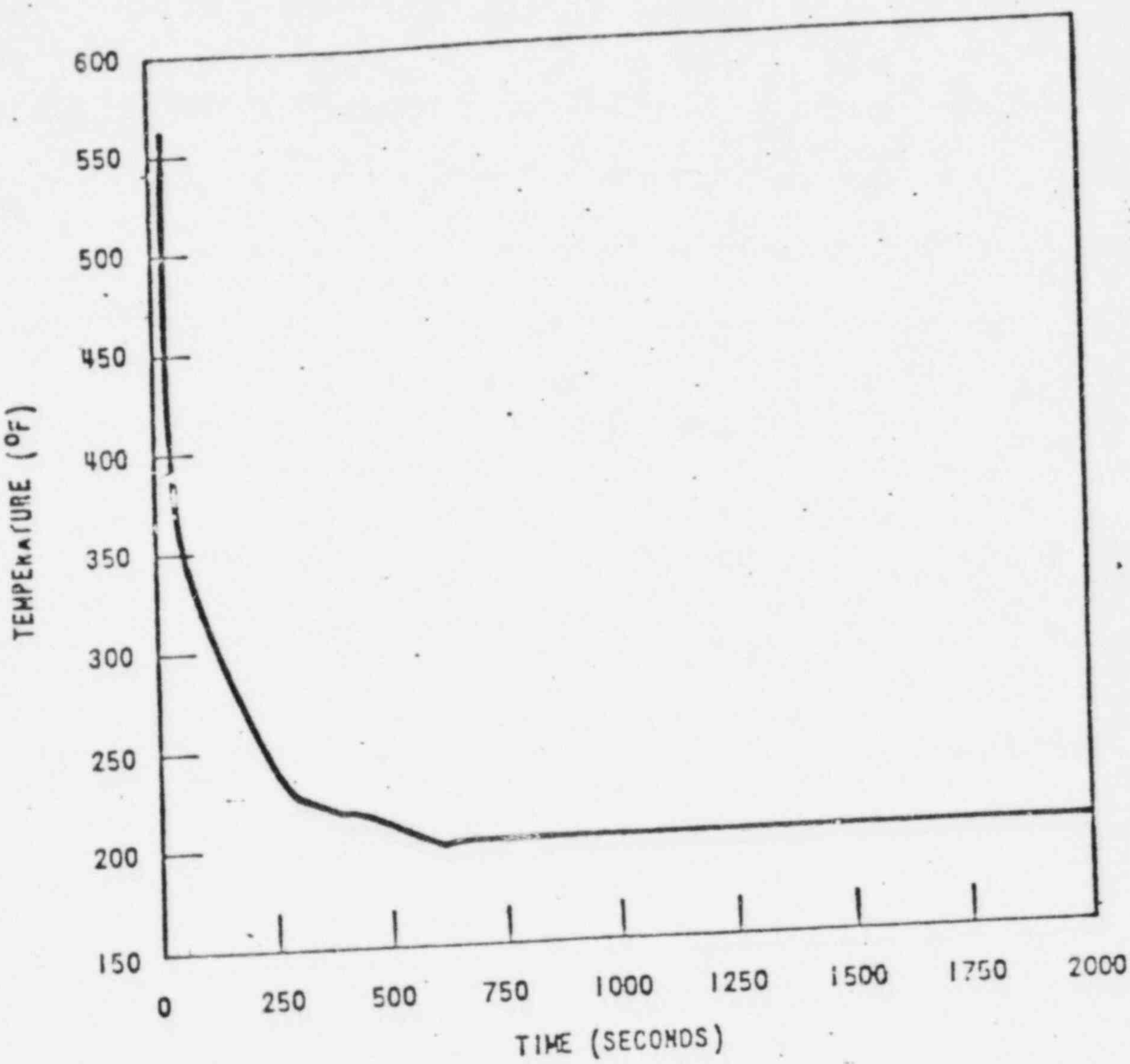
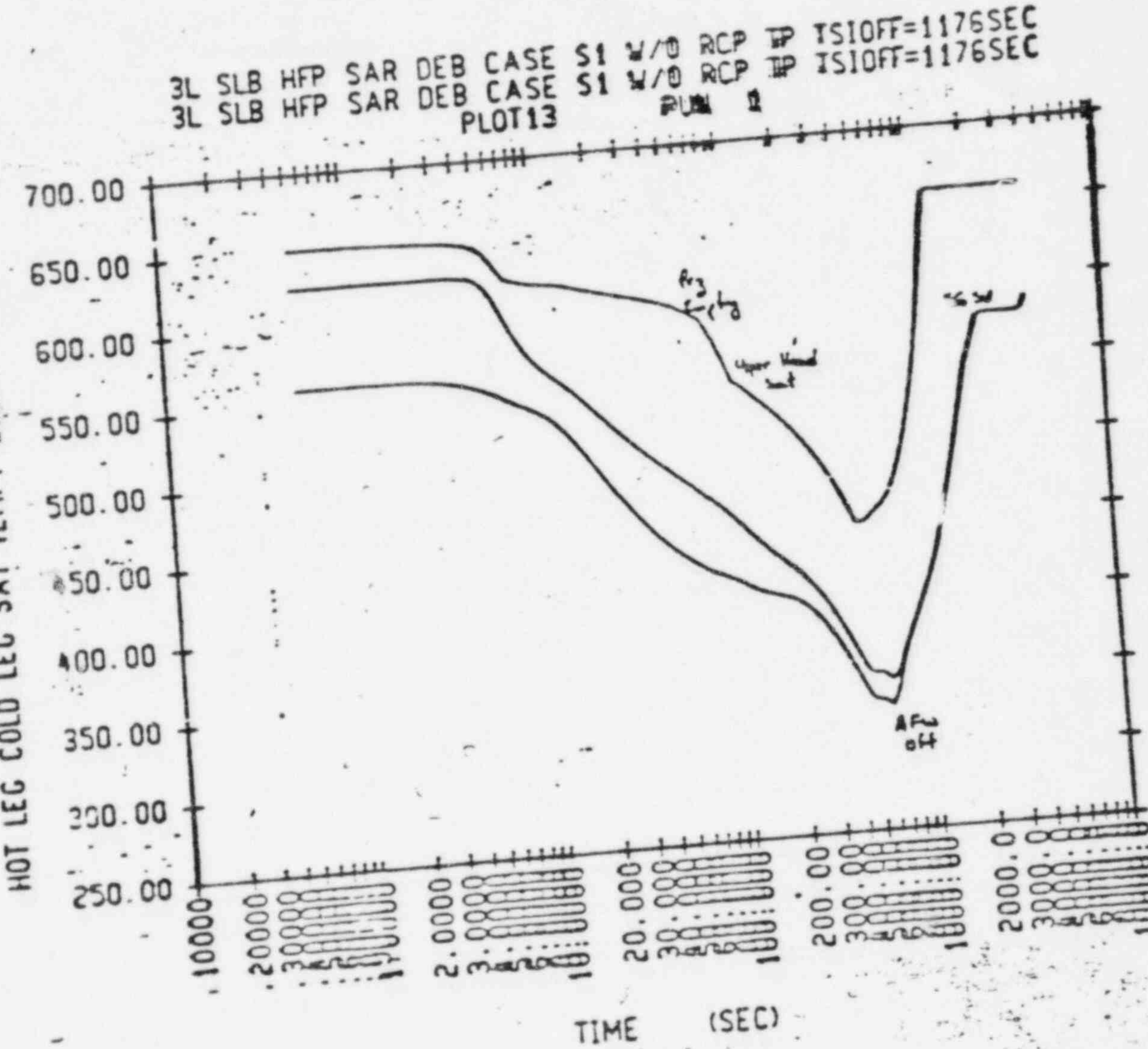


Figure III.1- 28 Large Steamline Break With Reactor Coolant Pumps Running. Cold Leg Temperature Versus Time


3L SLB HFP SAR DEB CASE S1 W/O RCP TP TSIOFF=1176SEC
3L SLB HFP SAR DEB CASE S1 W/O RCP TP TSIOFF=1176SEC
PLOT13
HOT LEG COLD LEG SAT TEMP. LOOP1 (DEG F)
700.00
650.00
600.00
550.00
500.00
450.00
400.00
350.00
300.00
250.00
TIME (SEC)

-8-

SS1

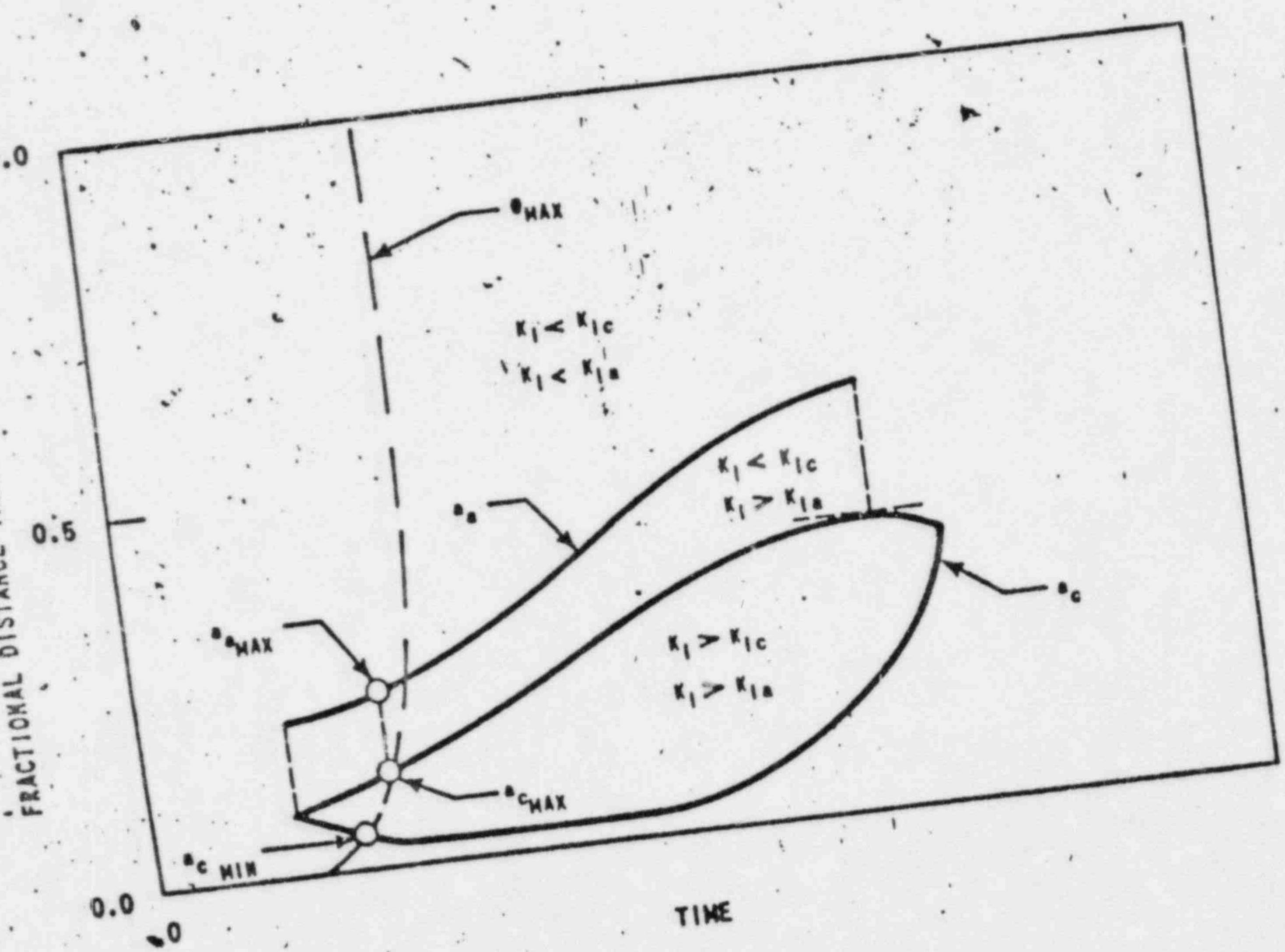
E-2

DISCUSSION ON WARM PRESTRESSING

AND FRACTURE ANALYSIS

BY

W. BAMFORD, WESTINGHOUSE


FRACTIONAL DISTANCE THROUGH WALL (a/t)
1.0
0.5
0.0
0
TIME
a_{MAX}
$K_I < K_{Ic}$
$K_I < K_{Ia}$
$K_I < K_{Ic}$
$K_I > K_{Ia}$
a_a
a_c
$K_I > K_{Ic}$
$K_I > K_{Ia}$
$a_{a_{MAX}}$
$a_{c_{MAX}}$
$a_{c\ MIN}$

12103-29

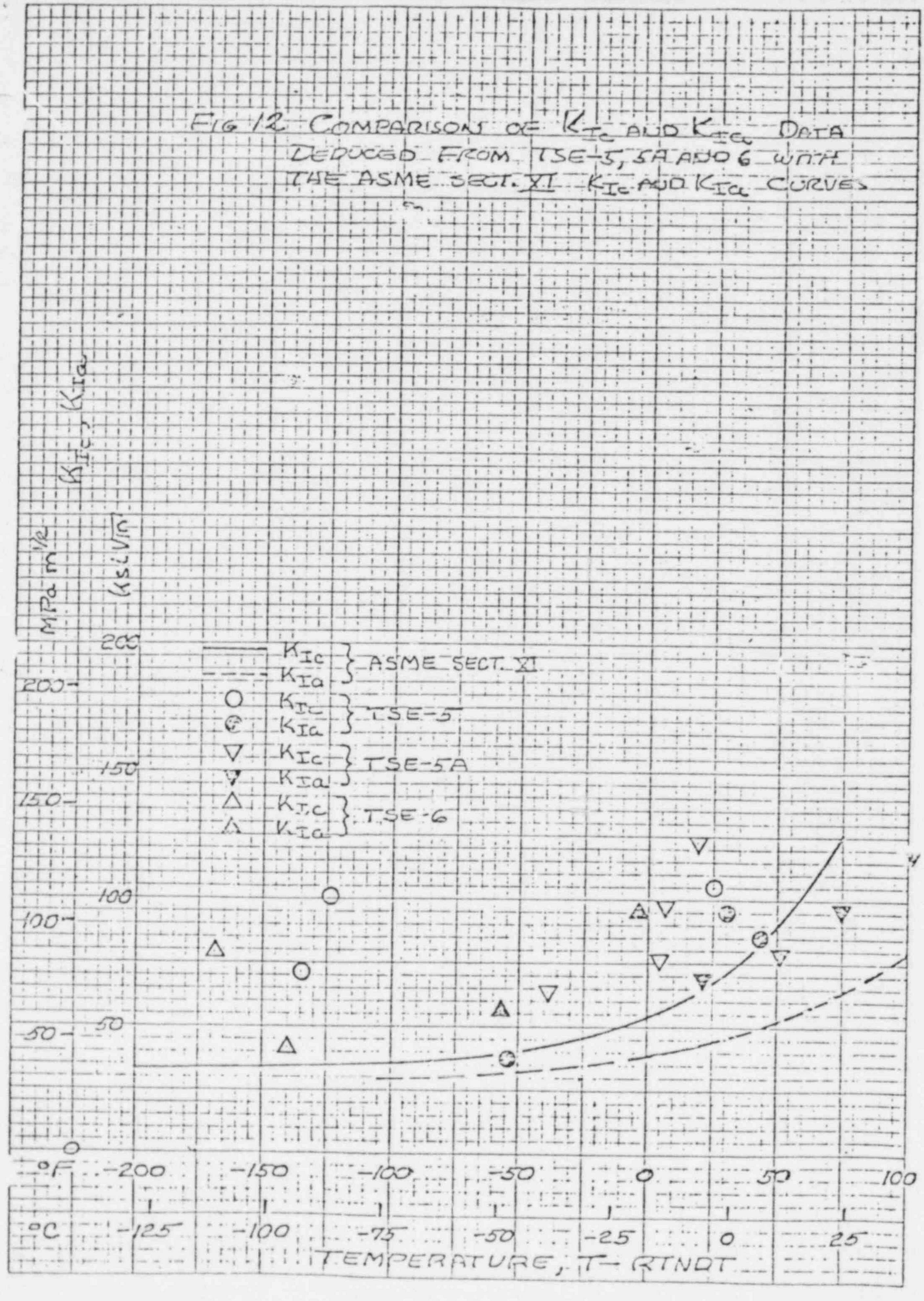


FIG 12 COMPARISONS OF K_{Ic} AND K_{Ia} DATA DEDUCED FROM TSE-5, 5A AND 6 WITH THE ASME SECT. XI K_{Ic} AND K_{Ia} CURVES
MPa m$^{1/2}$
K_{Ic}, K_{Ia}
ksi$\sqrt{in}$
K_{Ic}
K_{Ia}
ASME SECT. XI
K_{Ic}
K_{Ia}
TSE-5
K_{Ic}
K_{Ia}
TSE-5A
K_{Ic}
K_{Ia}
TSE-6
°F -200 -150 -100 -50 0 50 100
°C -125 -100 -75 -50 -25 0 25
TEMPERATURE, T - RTNDT

CRACK ARREST

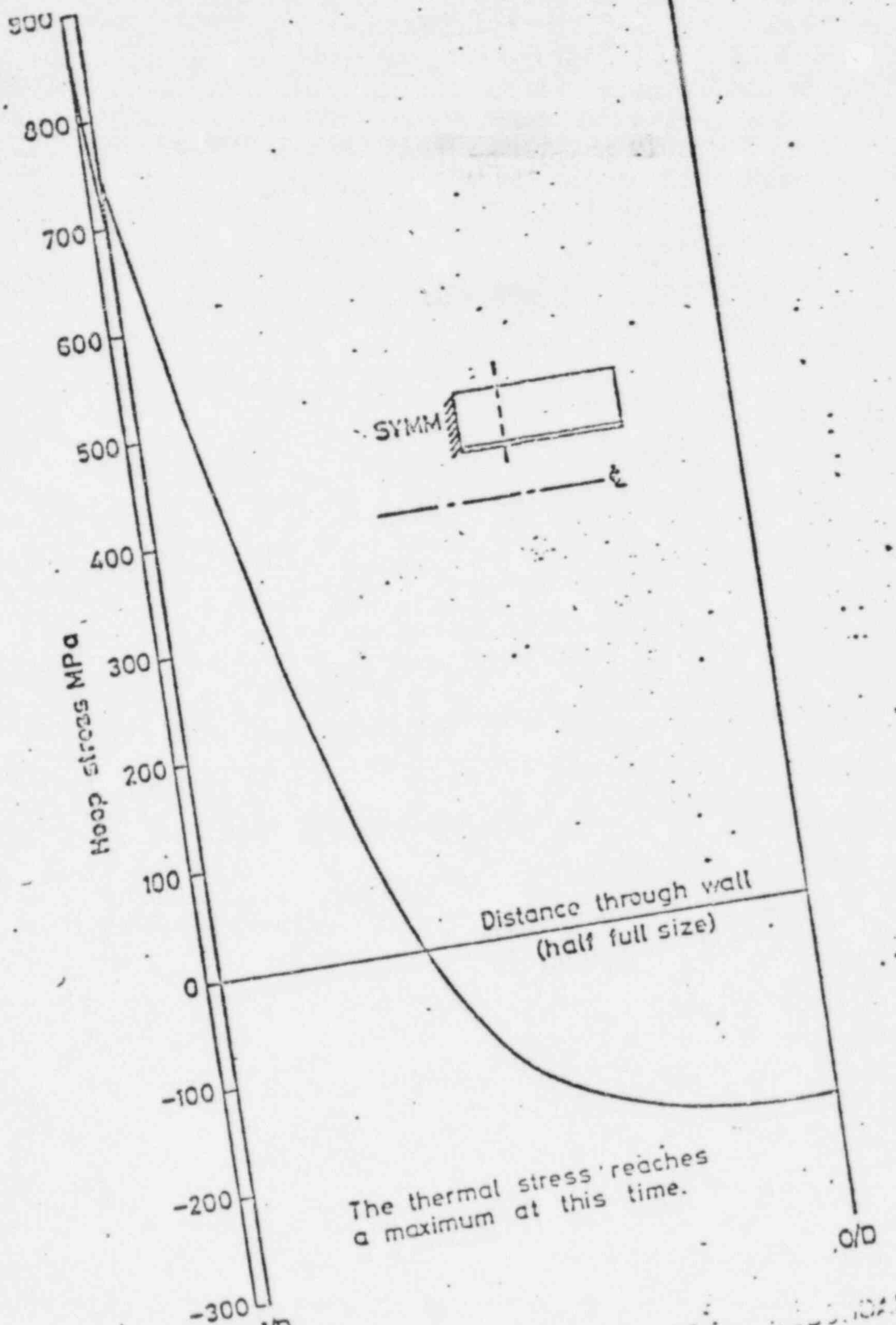
- CONCEPT
- DATA BASE FOR K_{Ia} CURVE
- CRACK ARREST CRITERIA - 75% OF WALL
- CONCLUSION

CAN GENERIC VALUES FOR RT_{NDT} BE OBTAINED FOR TYPICAL WELD WIRE - FLUX TYPES?

- o IN GENERAL, $RT_{NDT} = 0$ HAS BEEN ASSUMED WHEN DATA ARE NOT AVAILABLE. THIS IS CONSISTENT WITH THE NRC STANDARD REVIEW PLAN.

- o FOR SOME WELD WIRE FLUX COMBINATIONS IT IS POSSIBLE TO IMPLY GENERIC VALUES FOR RT_{NDT}, FOR EXAMPLE LINDE 0091 FLUX WELDS. IN THIS CASE A GENERIC RT_{NDT} WOULD BE -30F TO -60F.

- o SUCH GENERIC VALUES ARE NOT AVAILABLE FOR ALL WELD WIRE - FLUX COMBINATIONS AT THE PRESENT TIME.


900
800
700
600
500
400
300
200
100
0
-100
-200
-300
Hoop stress MPa
SYMM
℄
Distance through wall
(half full size)
The thermal stress reaches
a maximum at this time.
I/D
O/D

FLAW SHAPE AND CLADDING EFFECTS

- THE CLADDING PRODUCES A SLIGHT INCREASE IN THERMAL STRESSES IN THE IMMEDIATE VICINITY OF THE BASE METAL INTERFACE DURING A COOLDOWN EVENT. THIS CONTRIBUTION IS MORE THAN OFFSET BY IGNORING THE BENEFICIAL EFFECTS OF THE CLADDING IN KEEPING THE POSTULATED CRACK FROM EXTENDING IN LENGTH. FURTHERMORE, THE STRESS ORIGINATING HERE IS MUCH LESS THAN THE THERMAL SHOCK STRESSES (5-8%).

- THE ASSUMPTION OF A SEMI-ELLIPTIC SURFACE FLAW (AR 1:6) INITIALLY IS BASED ON A CONSERVATIVE ESTIMATE OF THE TYPE OF FLAW WHICH MIGHT EXIST. THE SHAPE OF FLAW WHICH MIGHT EXIST BASED ON A FATIGUE INITIATION IS CLOSER TO AN ASPECT RATIO OF 1:3.

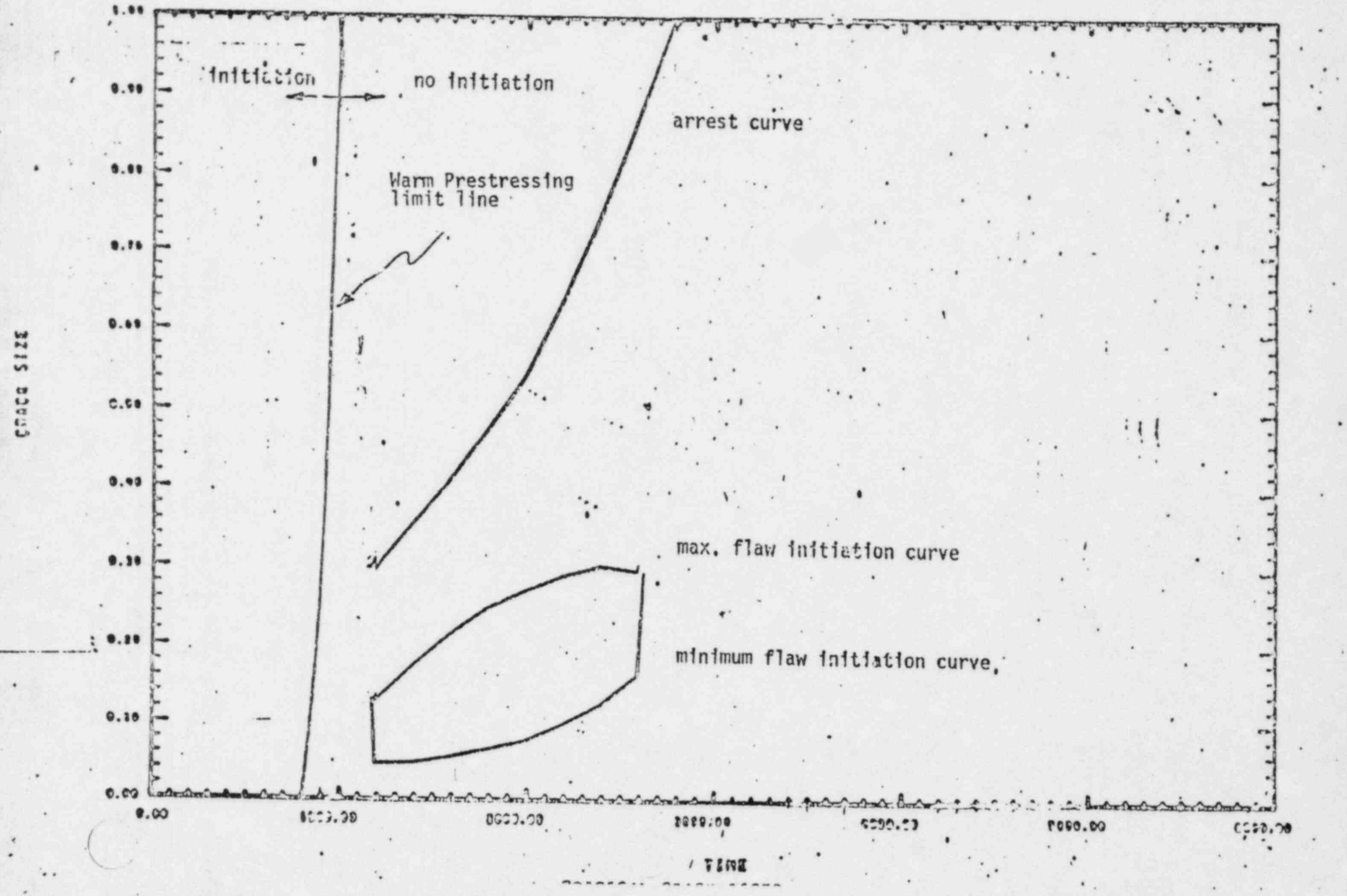
- IN ACTUALITY, THE MOST REALISTIC FLAW TO ASSUME BASED ON FIELD EXPERIENCE IS EMBEDDED, AND THIS IS CLEARLY LESS LIMITING THAN THE SURFACE FLAW CASE.

CONCLUSIONS

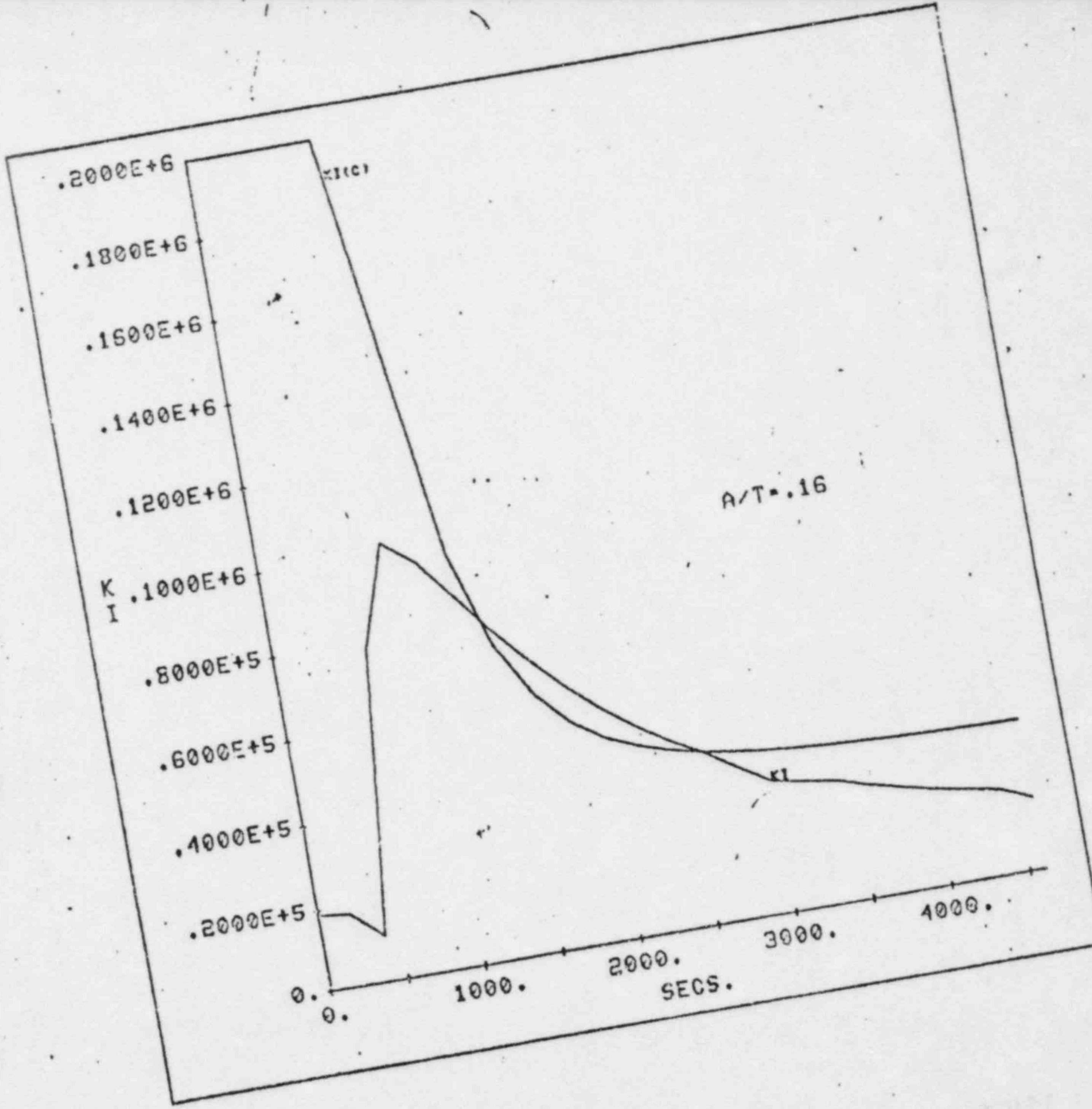
- IT IS NOW GENERALLY AGREED IN THE TECHNICAL COMMUNITY THAT WARM PRESTRESSING IS OPERABLE IN THERMAL SHOCK TRANSIENTS, WHEN THE APPLIED STRESS INTENSITY FACTOR IS DECREASING WITH TIME.

- THE WPS CONCEPT HAS BEEN DEMONSTRATED WITH BOTH MECHANICAL AND THERMAL LOADINGS, AND THEREFORE IS APPLICABLE TO PRESSURIZED THERMAL SHOCK EVALUATIONS.

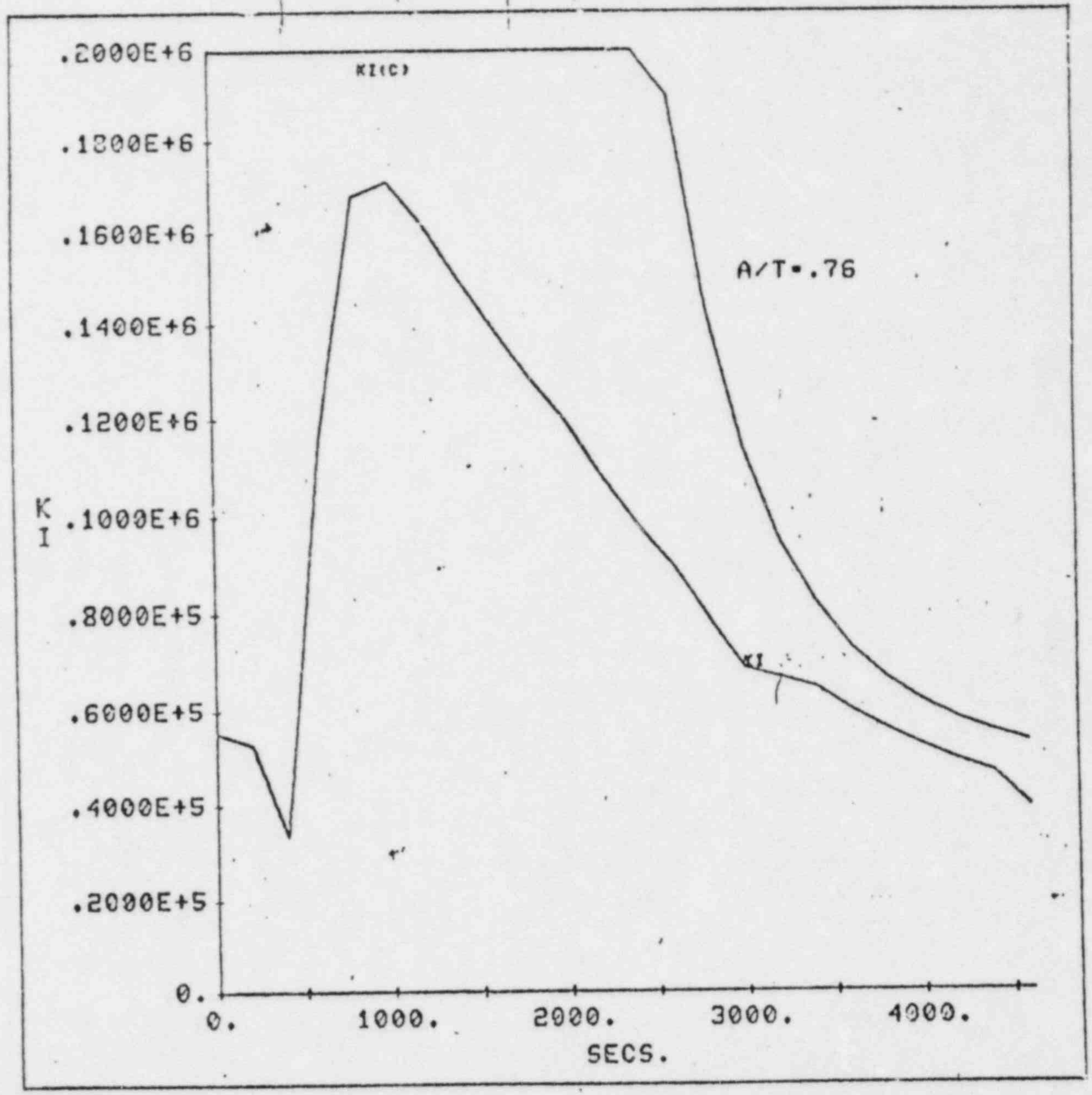
- DETERMINATION OF THE APPLICABILITY OF WPS MUST BE BASED ON THE APPLIED STRESS INTENSITY FACTOR VS TIME. PERTURBATIONS IN PRESSURE OR TEMPERATURE WILL NOT NECESSARILY INVALIDATE ITS USE, AS SHOWN BY ANALYSIS OF AN ACTUAL TRANSIENT.



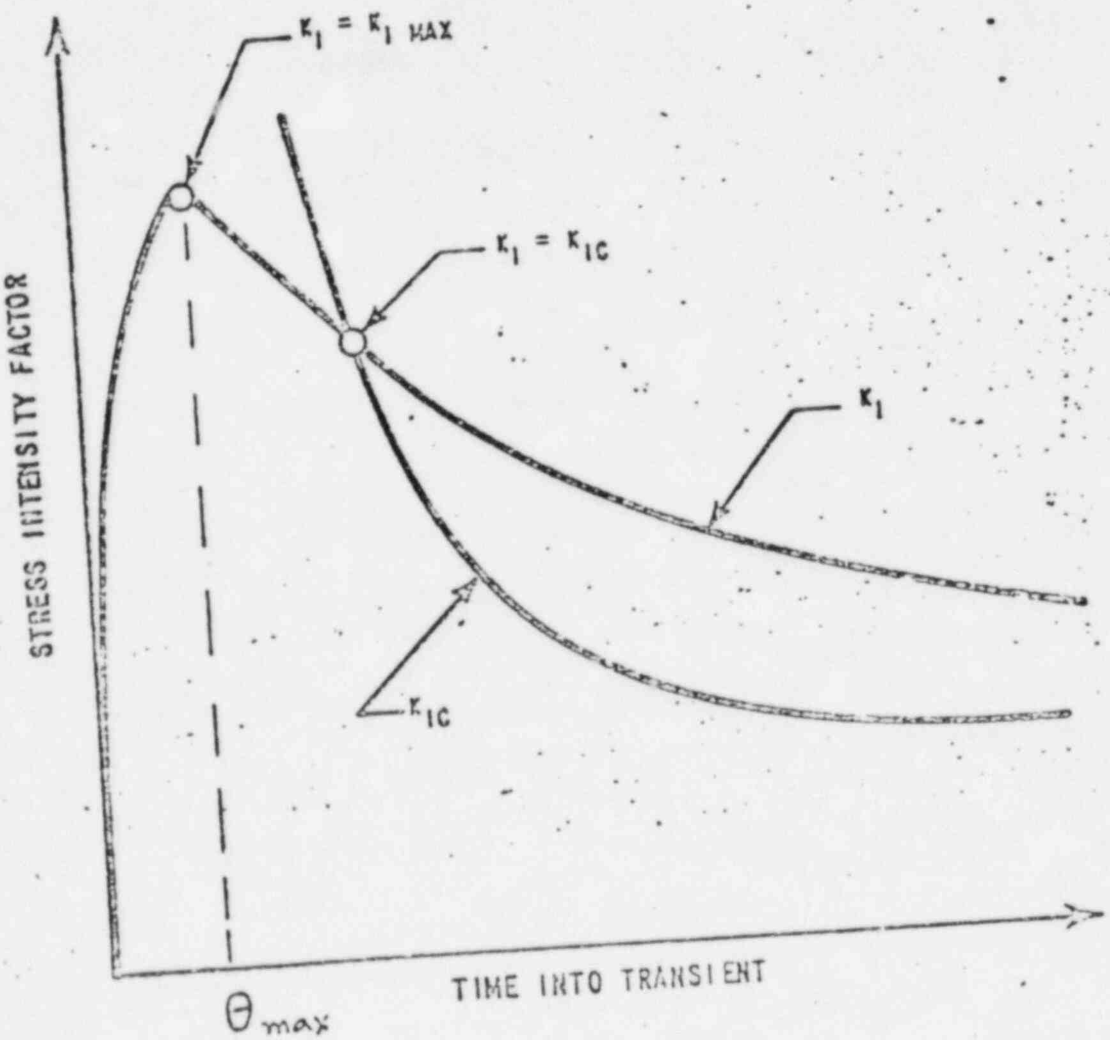
initiation
no initiation
arrest curve
Warm Prestressing limit line
max. flaw initiation curve
minimum flaw initiation curve
CRACK SIZE
TIME



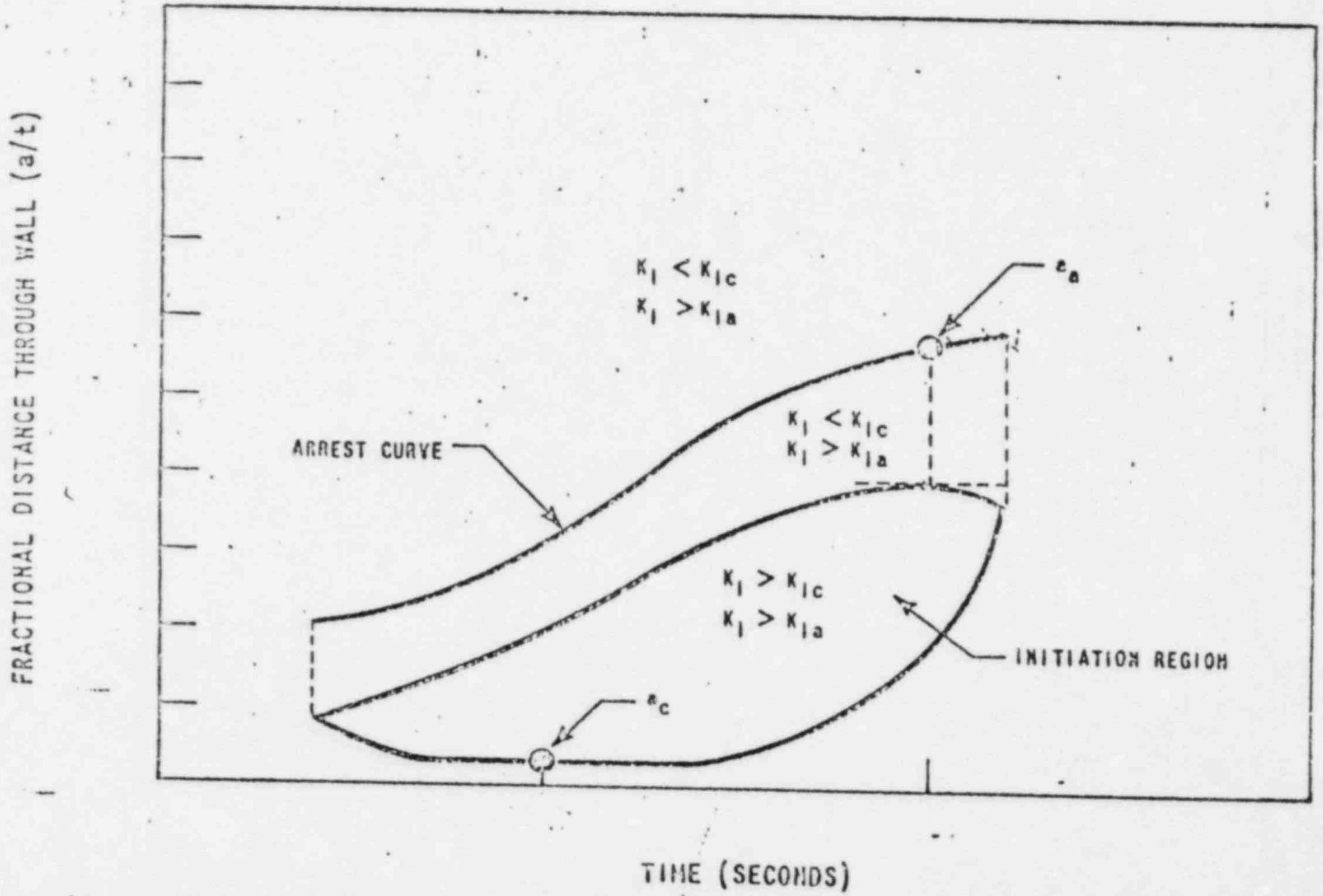
.2000E+6
.1800E+6
.1600E+6
.1400E+6
.1200E+6
K
I
.1000E+6
.8000E+5
.6000E+5
.4000E+5
.2000E+5
0.
KI(C)
KI
A/T=.16
0.
1000.
2000.
3000.
4000.
SECS.



A/T=.76
KI(C)
KI
K
I
.2000E+6
.1800E+6
.1600E+6
.1400E+6
.1200E+6
.1000E+6
.8000E+5
.6000E+5
.4000E+5
.2000E+5
0.
0.
1000.
2000.
3000.
4000.
SECS.

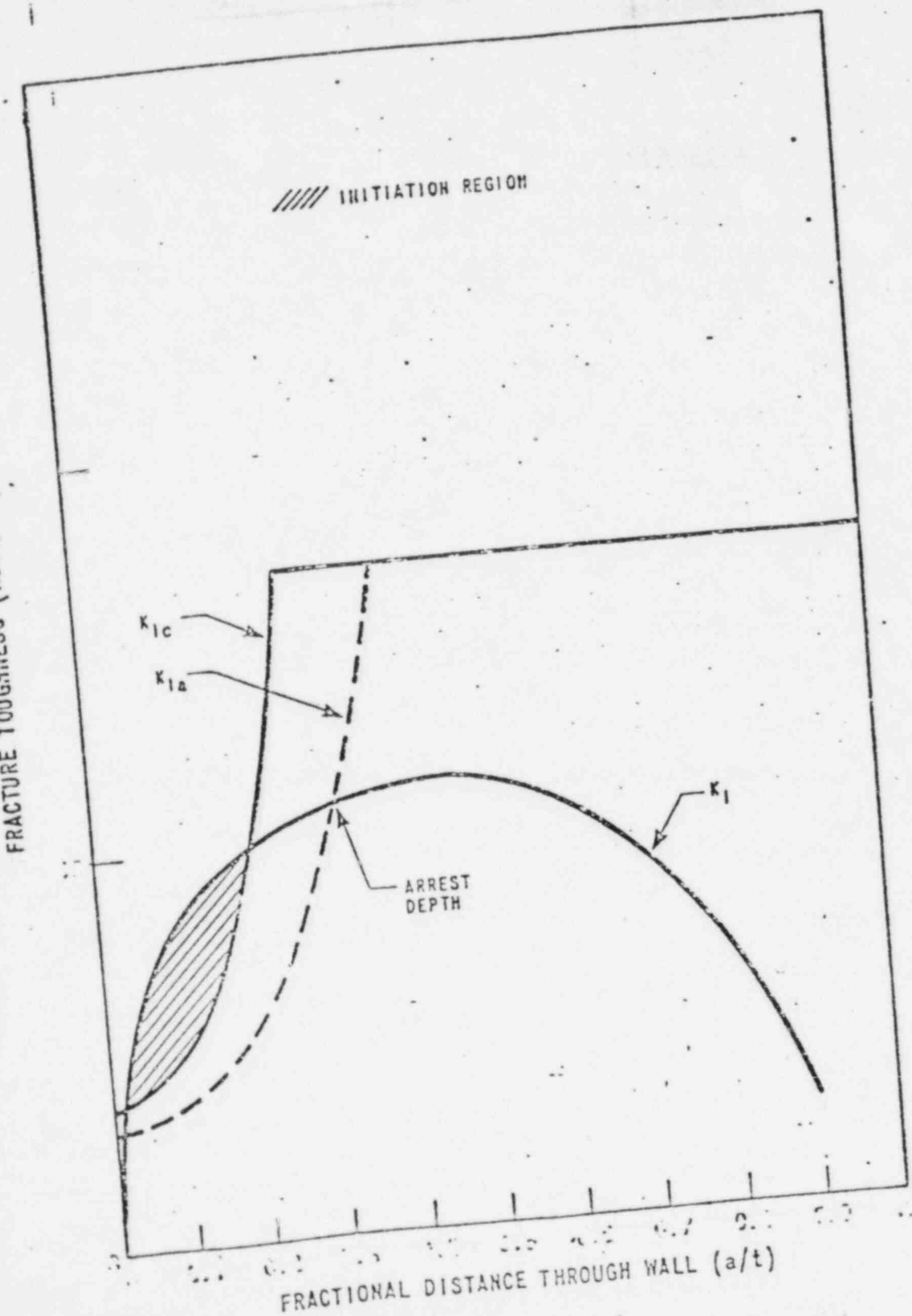


K1 = K1 MAX
K1 = K1C
K1
K1C
STRESS INTENSITY FACTOR
TIME INTO TRANSIENT
θmax

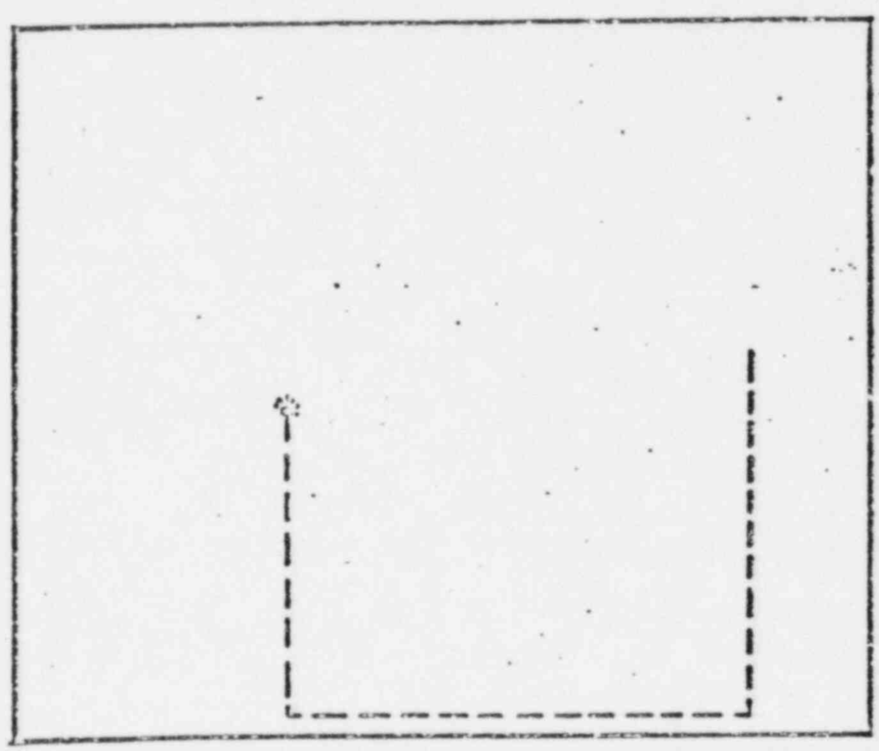


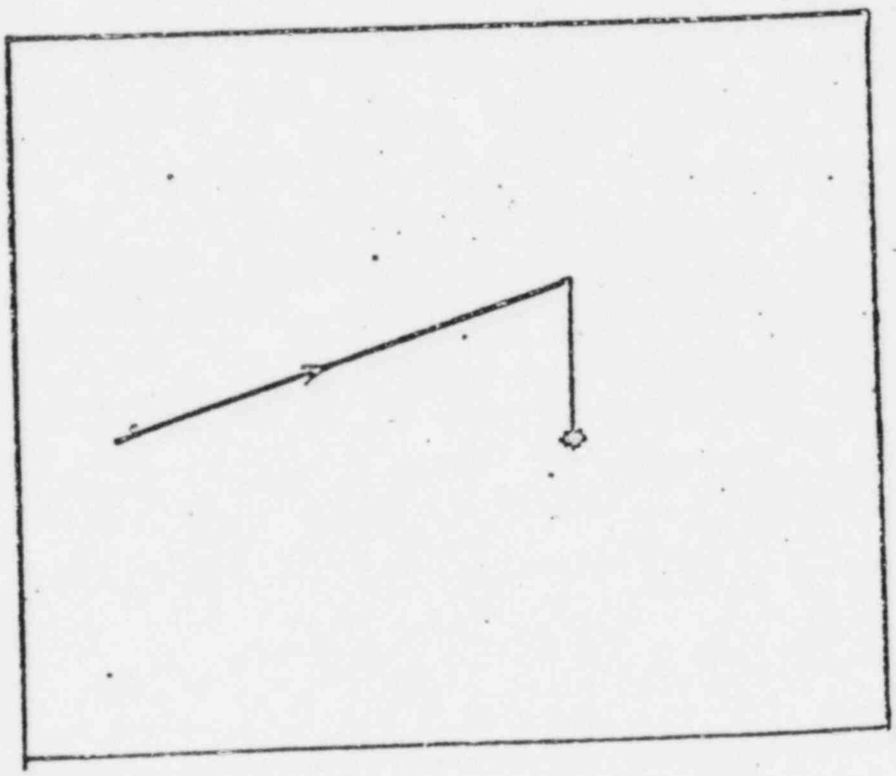
$K_I < K_{Ic}$
$K_I > K_{Ia}$
a_a
ARREST CURVE
$K_I < K_{Ic}$
$K_I > K_{Ia}$
$K_I > K_{Ic}$
$K_I > K_{Ia}$
INITIATION REGION
a_c
FRACTIONAL DISTANCE THROUGH WALL (a/t)
TIME (SECONDS)

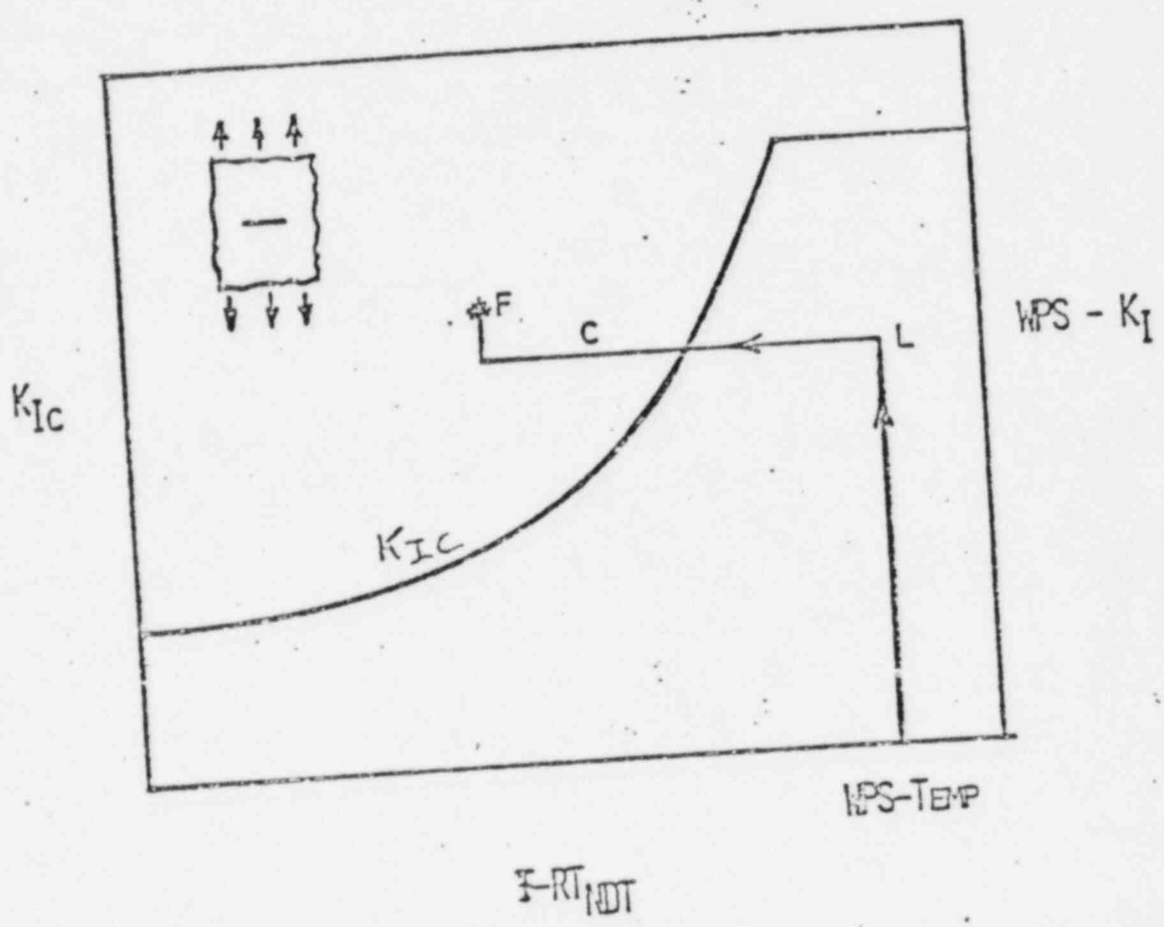
8331-11



INITIATION REGION
FRACTURE TOUGHNESS (KSI √IN)
K_{Ic}
K_{Ia}
K_I
ARREST DEPTH
FRACTIONAL DISTANCE THROUGH WALL (a/t)



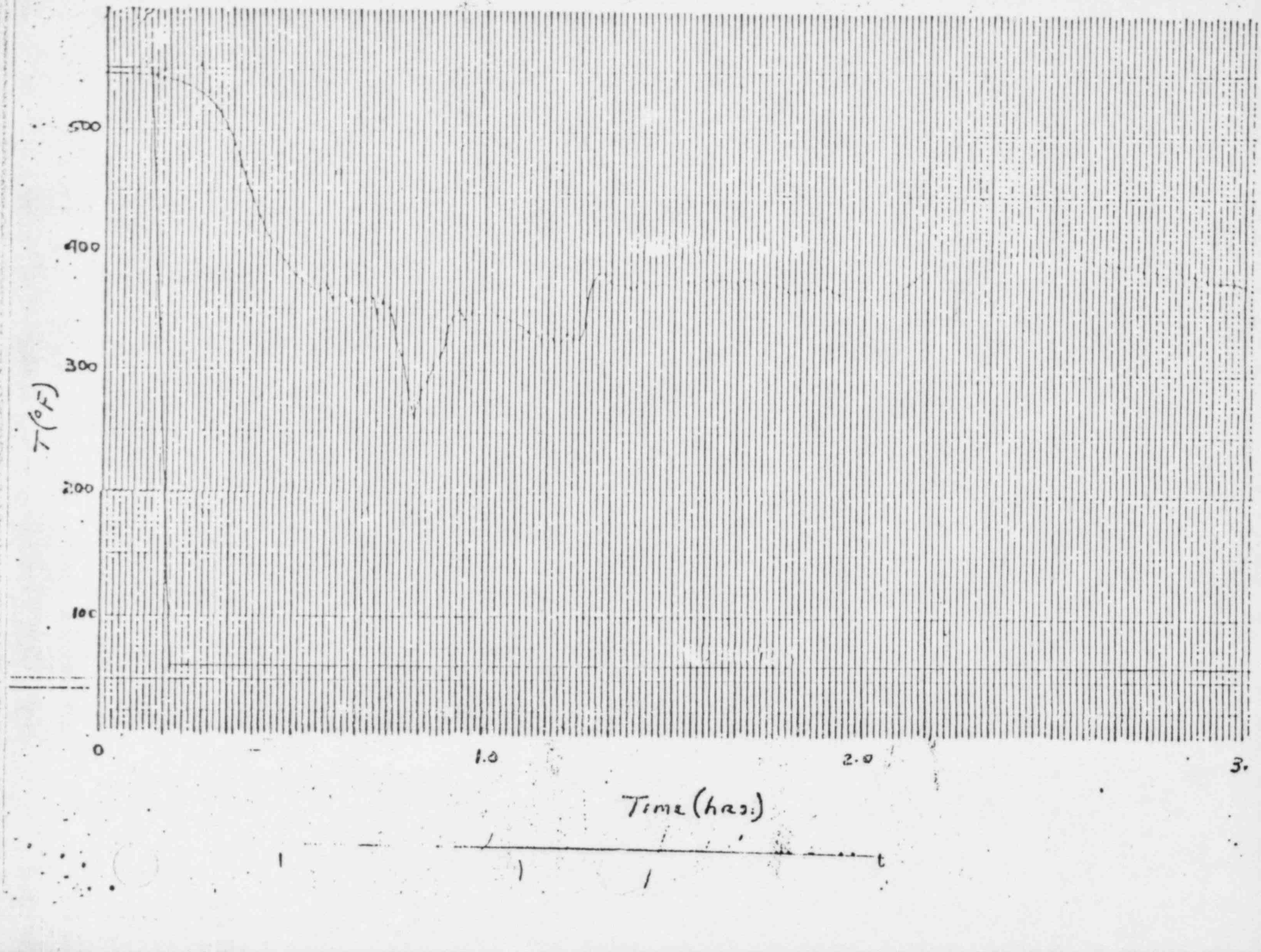




WPS - CONCEPT

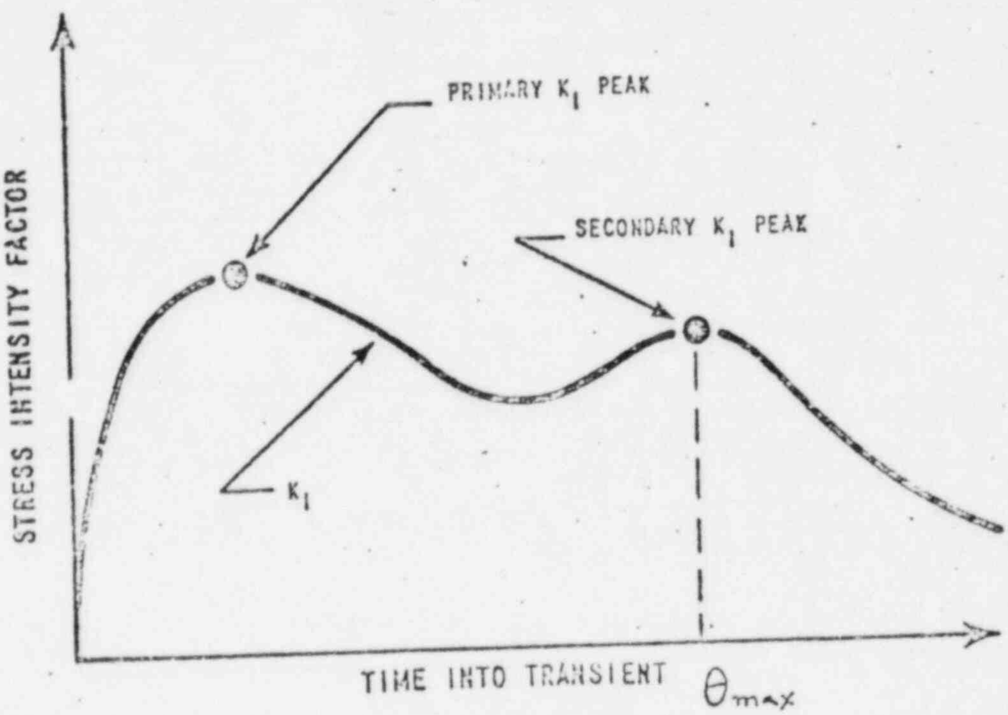
WARM PRESTRESSING

- CONCEPT
- CRITERIA USED
- DATA BASE
- EFFECTS OF TRANSIENT PERTURBATIONS
- CONCLUSIONS

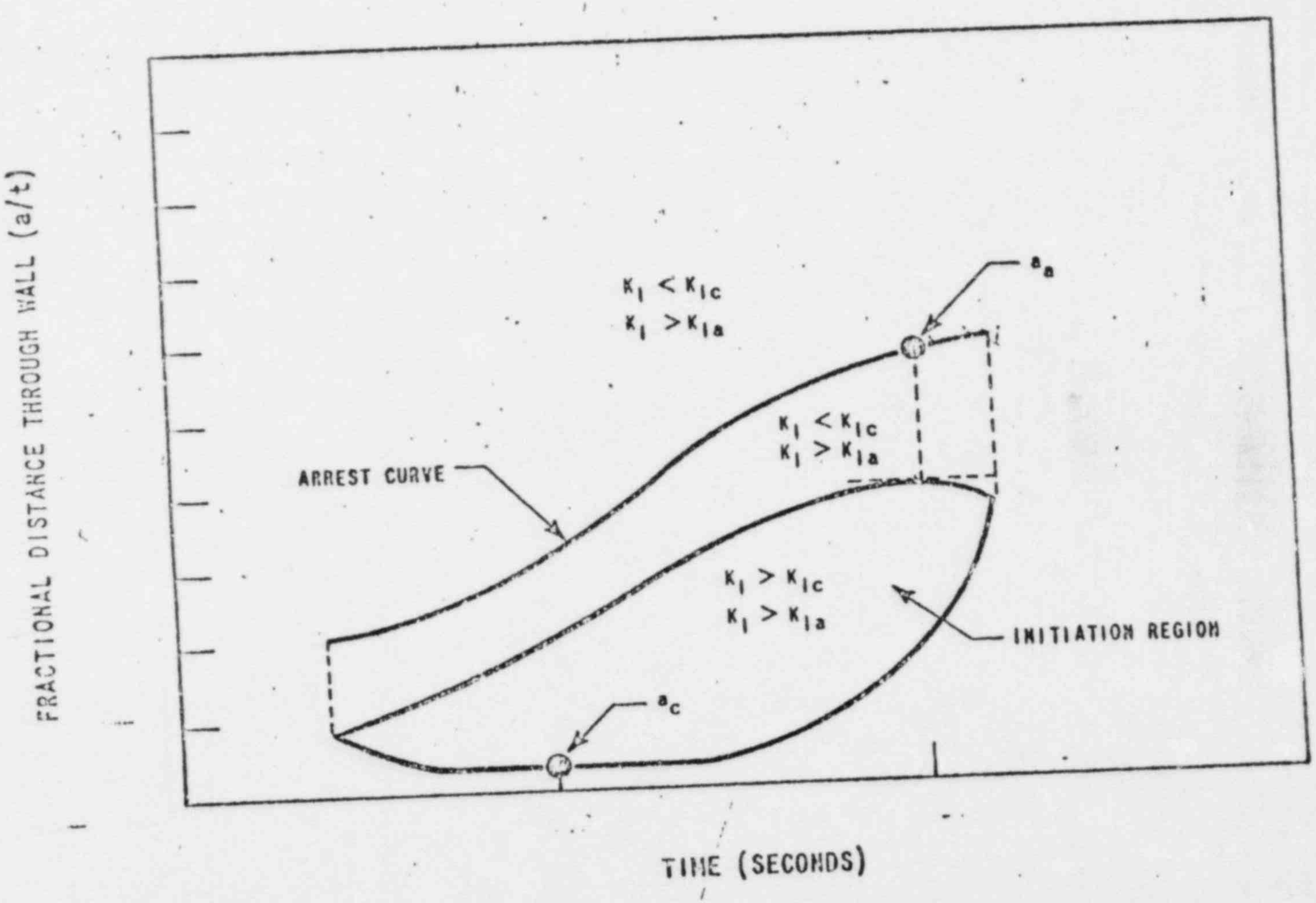


T (°F)
500
400
300
200
100
0
1.0
2.0
3.
Time (hrs.)

12103-32



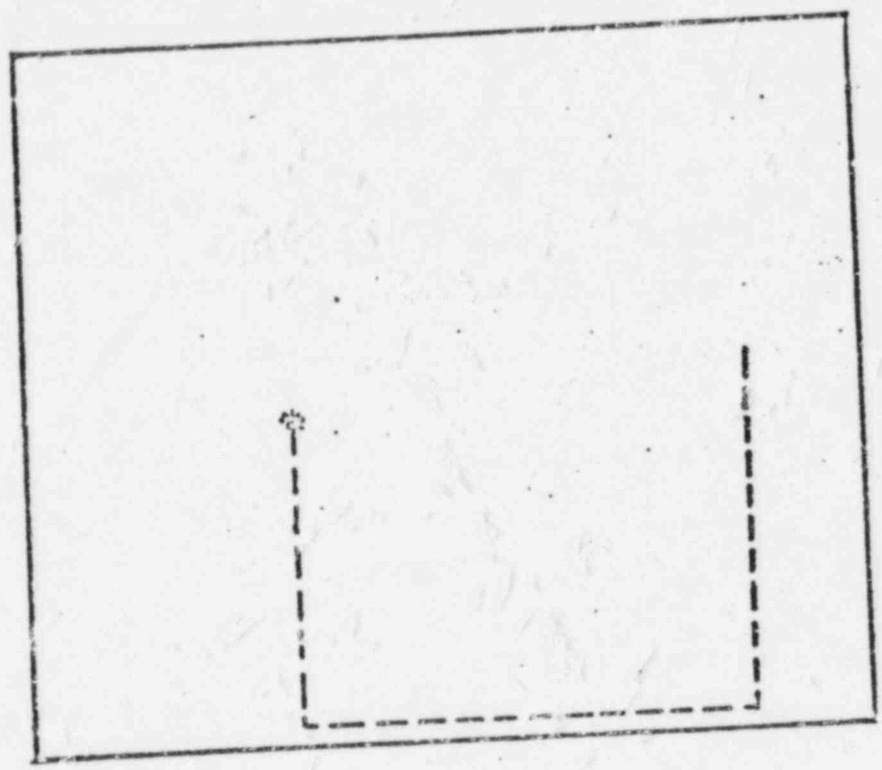
PRIMARY K_I PEAK
SECONDARY K_I PEAK
STRESS INTENSITY FACTOR
K_I
TIME INTO TRANSIENT
θ_{max}

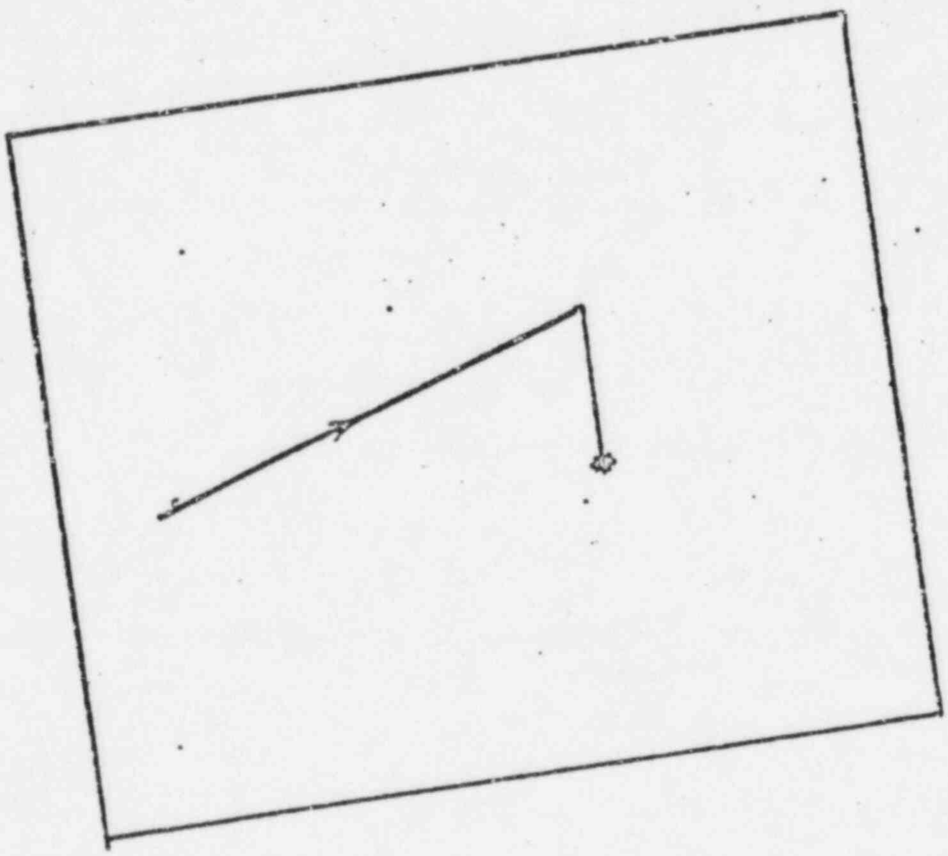


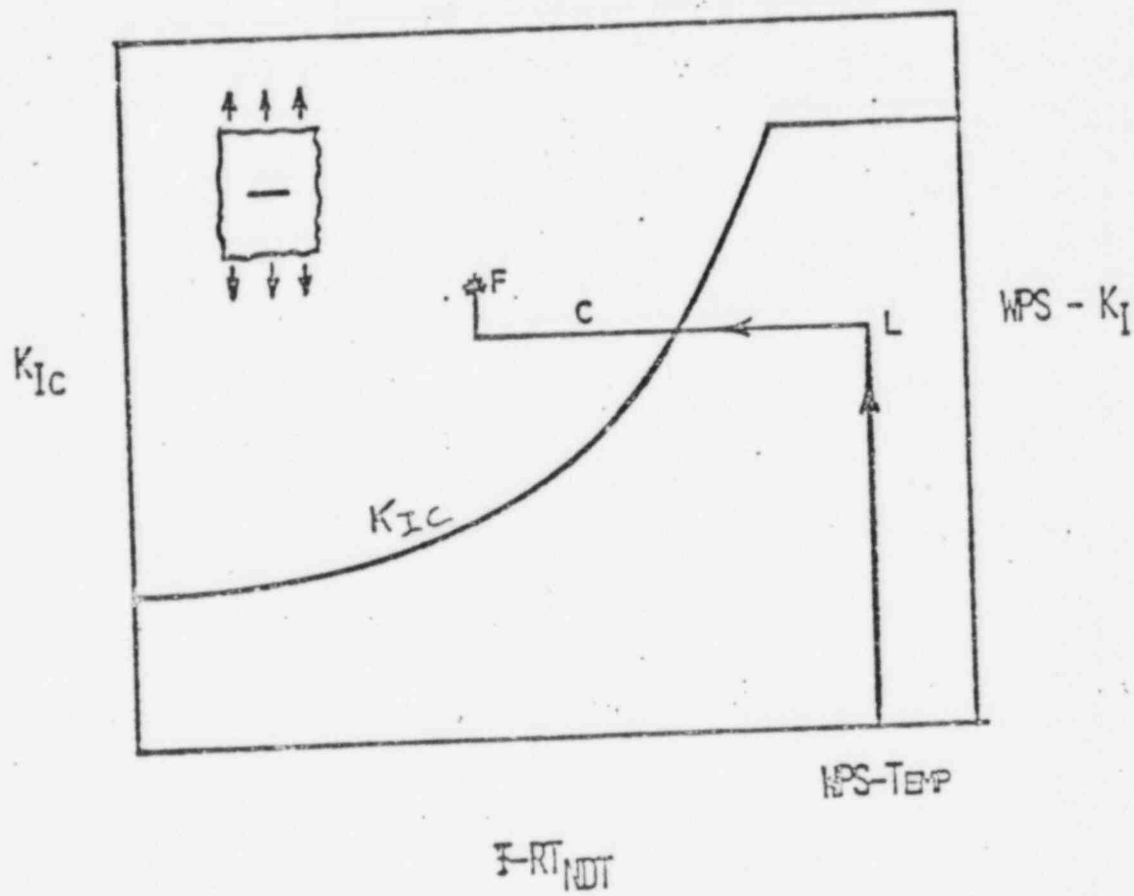
FRACTIONAL DISTANCE THROUGH WALL (a/t)
TIME (SECONDS)
$K_I < K_{Ic}$
$K_I > K_{Ia}$
a_a
$K_I < K_{Ic}$
$K_I > K_{Ia}$
ARREST CURVE
$K_I > K_{Ic}$
$K_I > K_{Ia}$
INITIATION REGION
a_c

WARM PRESTRESSING

- CONCEPT
- CRITERIA USED
- DATA BASE
- EFFECTS OF TRANSIENT PERTURBATIONS
- CONCLUSIONS

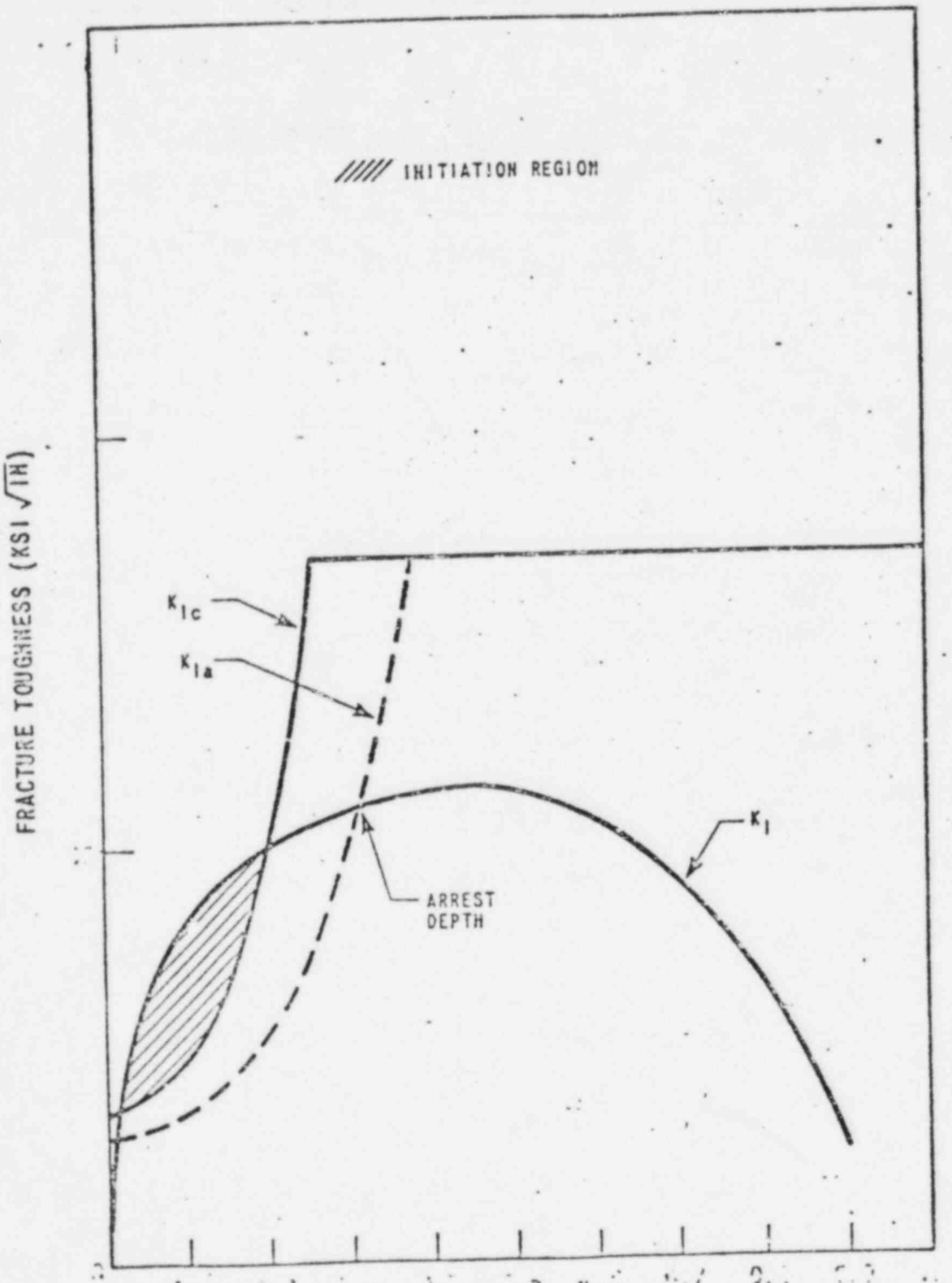




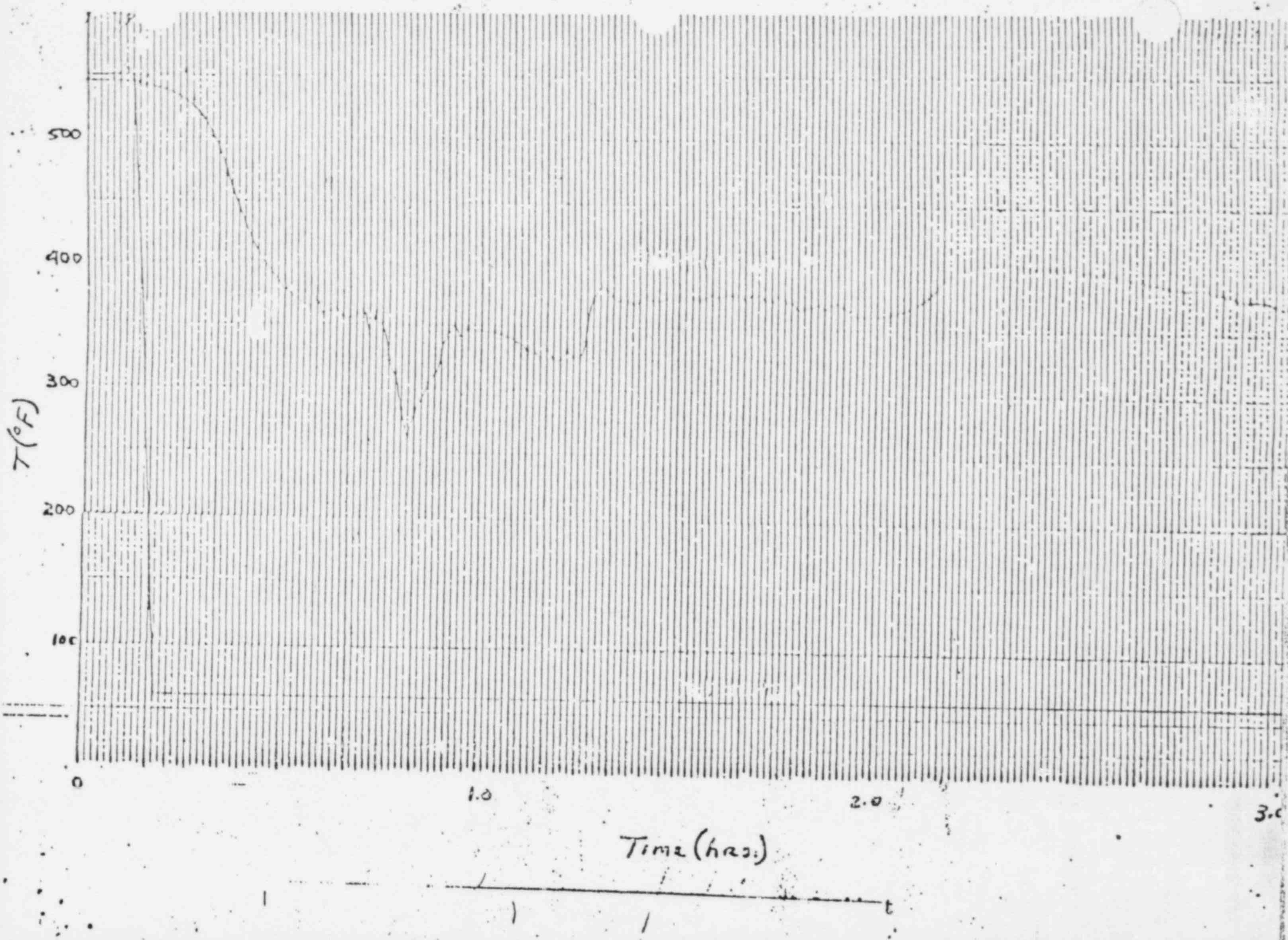


WPS - CONCEPT

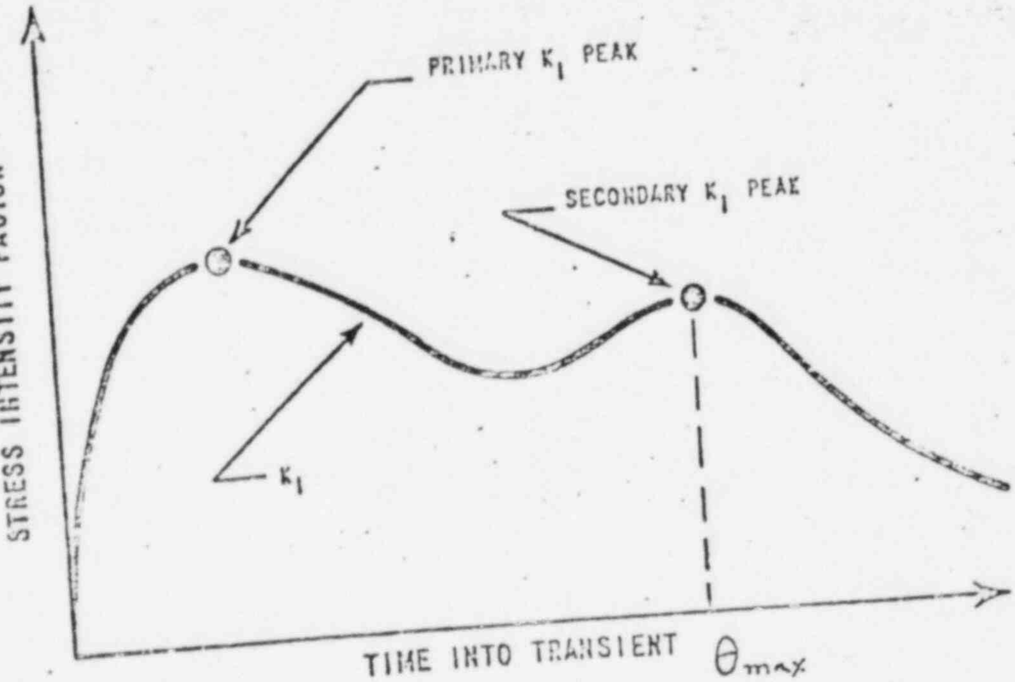
8321-11



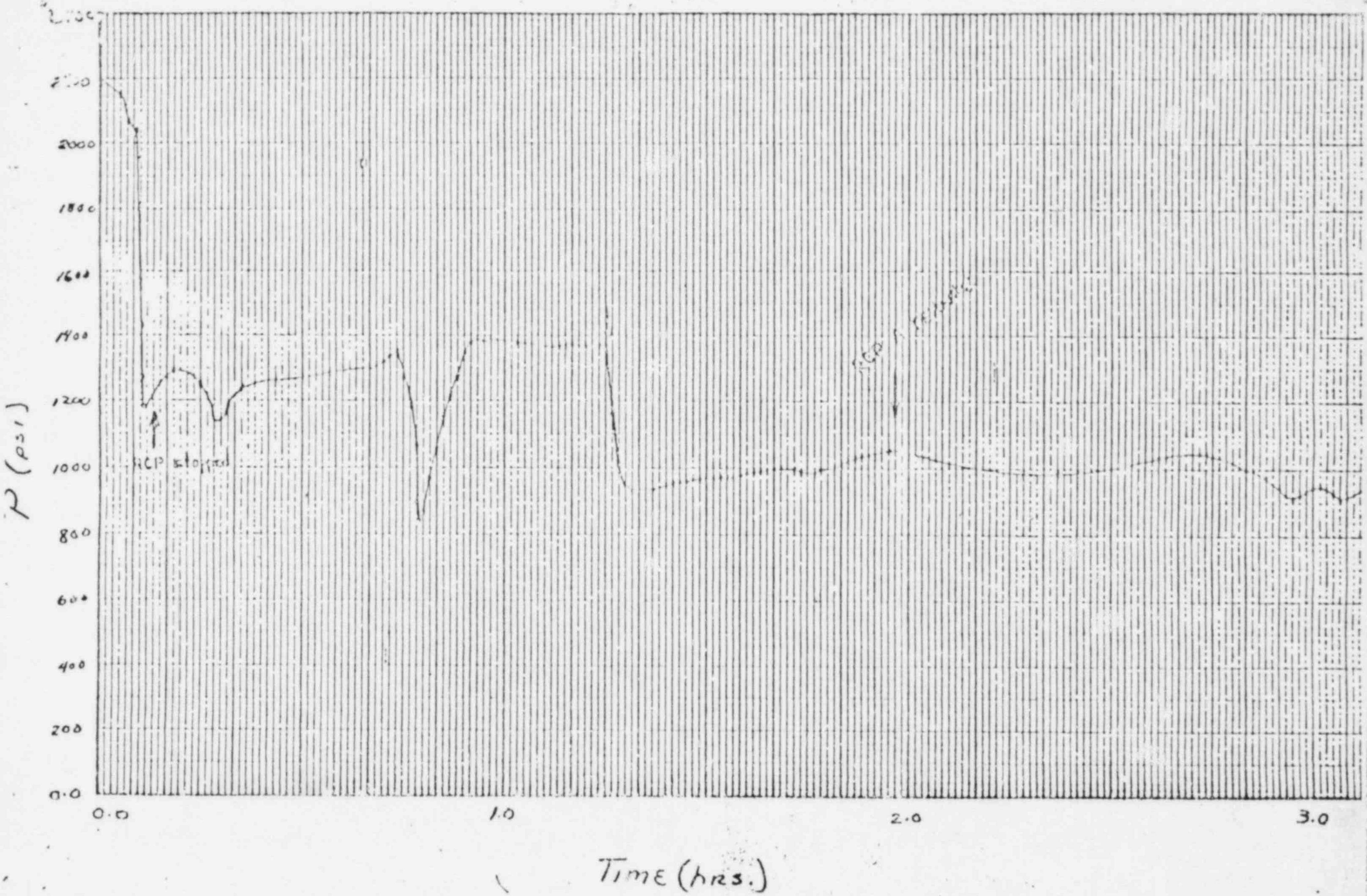
INITIATION REGION
K_{Ic}
K_{Ia}
K_I
ARREST DEPTH
FRACTURE TOUGHNESS (KSI $\sqrt{IN}$)
FRACTIONAL DISTANCE THROUGH WALL (a/t)



500
400
300
200
10C
0
1.0
2.0
3.C
T(°F)
Time (hrs.)
t



STRESS INTENSITY FACTOR
PRIMARY K_I PEAK
SECONDARY K_I PEAK
K_I
TIME INTO TRANSIENT
θ_{max}



2200
2000
1800
1600
1400
1200
1000
800
600
400
200
0.0
P (psi)
RCP stopped
0.0
1.0
2.0
3.0
Time (hrs.)

DISCUSSION ON TRANSIENTS

BY

J. RUMANICK, WESTINGHOUSE

NORMAL CONDITIONS TRANSIENTS

- RCP startup and shutdown
- Plant heatup and cooldown
- Unit loading and unloading between 0 and 15 percent of full power
- Unit loading and unloading at 5 percent of full power/minute
- Reduced temperature return to power
- Step load increase and decrease of 10 percent of full power
- Large step load decrease with steam dump
- Steady state fluctuations
- Boron concentration equalization
- Feedwater cycling
- Loop out of service
- Refueling
- Turbine roll test
- Primary side leakage test
- Secondary side leakage test

UPSET CONDITIONS TRANSIENTS

- Loss of load
- Loss of power
- Partial loss of flow
- Reactor trip from full power

  Case A – with no inadvertent cooldown
  Case B – with cooldown and no S.I.
  Case C – with cooldown and S.I.
- Inadvertent RCS depressurization
- Inadvertent startup of an inactive loop
- Control rod drop
- Inadvertent safety injection actuation
- Excessive feedwater flow

EMERGENCY CONDITIONS TRANSIENTS

- Small loss of coolant accident
- Small steam line break
- Complete loss of flow

FAULTED CONDITIONS TRANSIENTS

- Reactor coolant pipe break (large LOCA)
- Large steam line break
- Feedwater line break
- Reactor coolant pump locked rotor
- Control rod ejection
- Steam generator tube rupture

Design Transients Which Result In A Cooldown Below No Load Temperature:

- Plant Heatup And Cooldown
- Turbine Roll Test
- RX Trip With Cooldown
- RX Trip With Cooldown And SI
- Inadvertent Depressurization
- Excessive Feedwater Flow
- Small LOCA
- Small Steam Line Break
- Large LOCA
- Large Steam Line Break
- Steam Generator Tube Rupture

DESIGN TRANSIENTS WHICH RESULT IN CHALLENGES TO VESSEL INTEGRITY

- SMALL LOCA
- SMALL STEAMLINE BREAK
- LARGE LOCA
- LARGE STEAMLINE BREAK

METHODOLOGY FOR ANALYSIS

MAJOR NSSS CONTROL SYSTEMS

- REACTOR CONTROL
- STEAM DUMP CONTROL
- STEAM GENERATOR LEVEL CONTROL
- PRESSURIZER PRESSURE CONTROL
- PRESSURIZER LEVEL CONTROL

INSTRUMENT LINE / TAP FAILURE

- TABULATE COMMON TAPS
- DETERMINE SENSOR READING IF TAP BREAKS.
- EVALUATE CONSEQUENCES

Sensor Failures

- Tabulate Input To Each System
- Tabulate Each System For Which Input Used
- Postulate Failure Of Each Input Separately
- Determine Consequences

POWER SUPPLY FAILURES

- TABULATE SOURCE OF POWER FOR EACH SENSOR
- TABULATE SOURCE OF POWER FOR EACH CONTROL SYSTEM
- TABULATE FAILURE MODES ON LOSS OF POWER
- DETERMINING IMPACT OF EACH POWER SUPPLY FAILURE

FAILURES CONSIDERED

- SENSOR FAILURE (EITHER HIGH OR LOW)
- LOSS OF POWER TO SINGLE INSTRUMENT DISTRIBUTION PANEL
- BREAK IN SENSOR LINE TAP FOR COMMON INSTRUMENTS

ANALYSES CONCLUSIONS

PURPOSE OF ANALYSES

- TO SHOW THAT SENSOR FAILURE OR LOSS OF POWER TO INSTRUMENT DISTRIBUTION PANEL WILL RESULT IN AN EVENT THAT IS BOUNDED BY THE FSAR CHAPTER 15 ANALYSES.

CONCLUSIONS FROM ANALYSES

- FAILURE OF ANY SENSOR, AND LOSS OF POWER TO ANY INSTRUMENT DISTRIBUTION PANEL, WILL RESULT IN AN EVENT NO WORSE THAN AN FSAR CONDITION II EVENT (FAULT OF MODERATE FREQUENCY).

REVIEW INDICATES NO CHALLENGE
TO VESSEL INTEGRITY RESULTS
FOR THE EFFECTS IDENTIFIED IN
THE CONTROL SYSTEMS FAILURES STUDIES

DISCUSSION ON MIXING

AND SBLOCA TRANSIENTS

BY

T. ANDREYCHECK, WESTINGHOUSE

SMALL BREAK LOCA MIXING

- PERFECT MIXING ASSUMED DURING NATURAL CIRCULATION
- WITH LOOP STAGNANT, TEMPERATURE BASED ON BOTH NO MIXING AND, MIXING CALCULATED BY THERMAL-HYDRAULIC ANALYSIS

SMALL LOCA SYSTEMS ASSUMPTIONS

- REACTOR AND RCP TRIP
- SI START ON LOW PRESSURE
- NORMAL FEED TRIP
- AUX FEED INITIATION
- AUX FEED THROTTLE

SMALL LOCA GENERIC TRANSIENTS

- 2, 3, 4 LOOP PLANT TYPES
- SPECTRUM OF BREAK SIZES
- HOT LEG BREAK LOCATION
- MAXIMUM SI FLOW
- MINIMUM SI AND ACCUMULATOR TEMPERATURES

Figure III.1-11 Downcomer Mass Flow

4 Loop 2 Inch Hot Leg Break

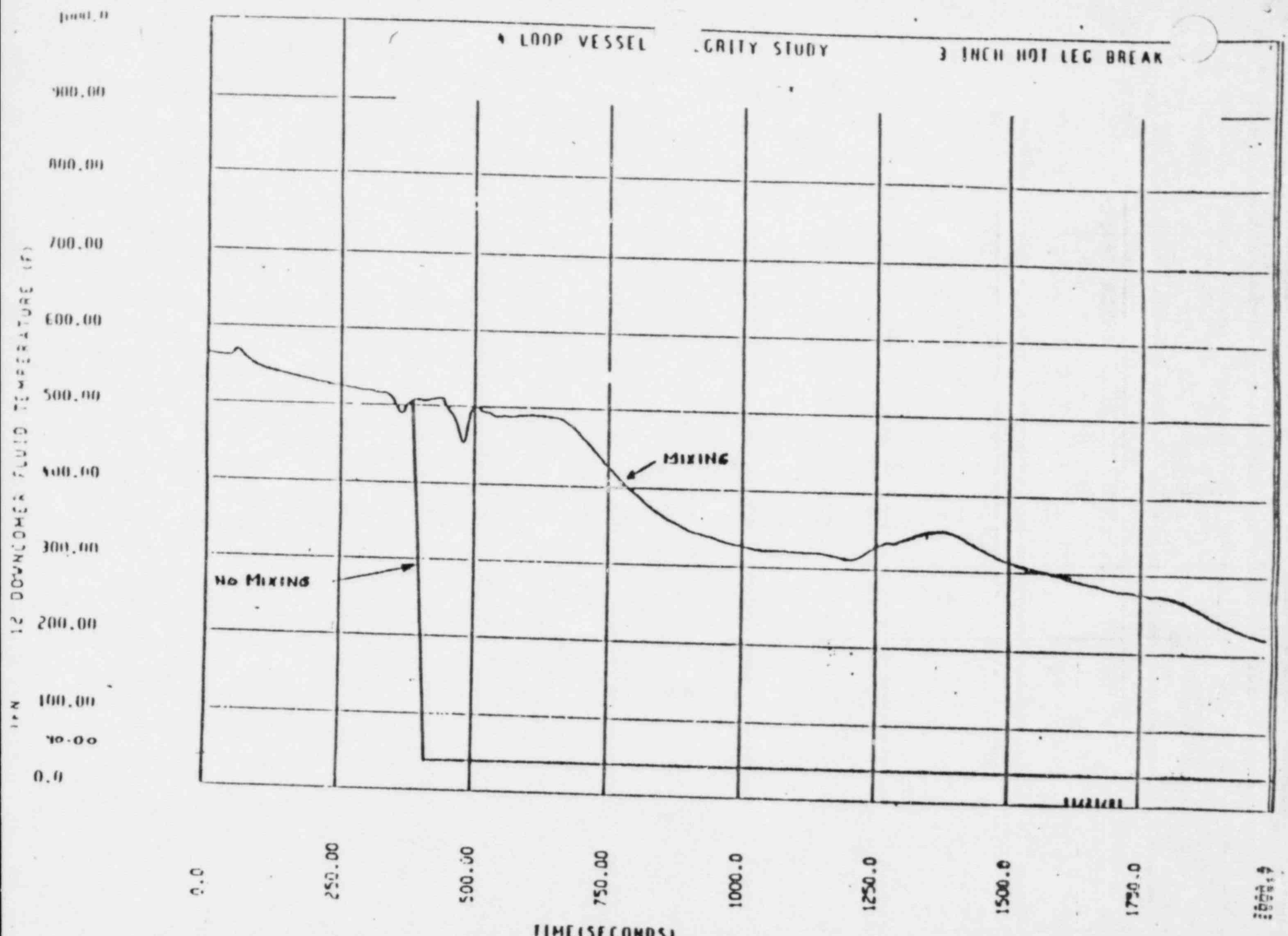


Figure III.1-4 Downcomer Fluid Temperature 4 Loop 3 Inch Hot

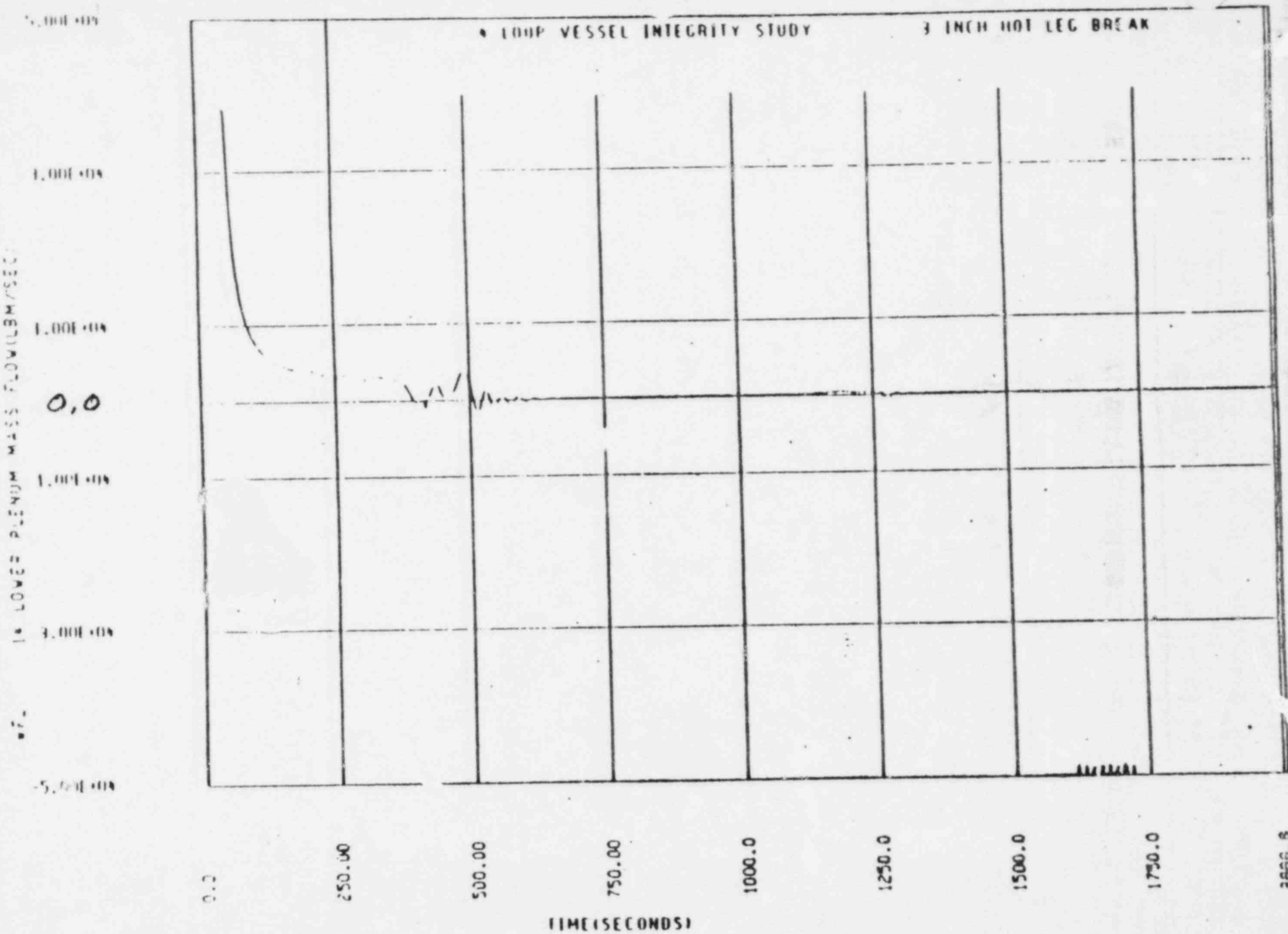


Figure III.1-10 Downcomer Mass Flow

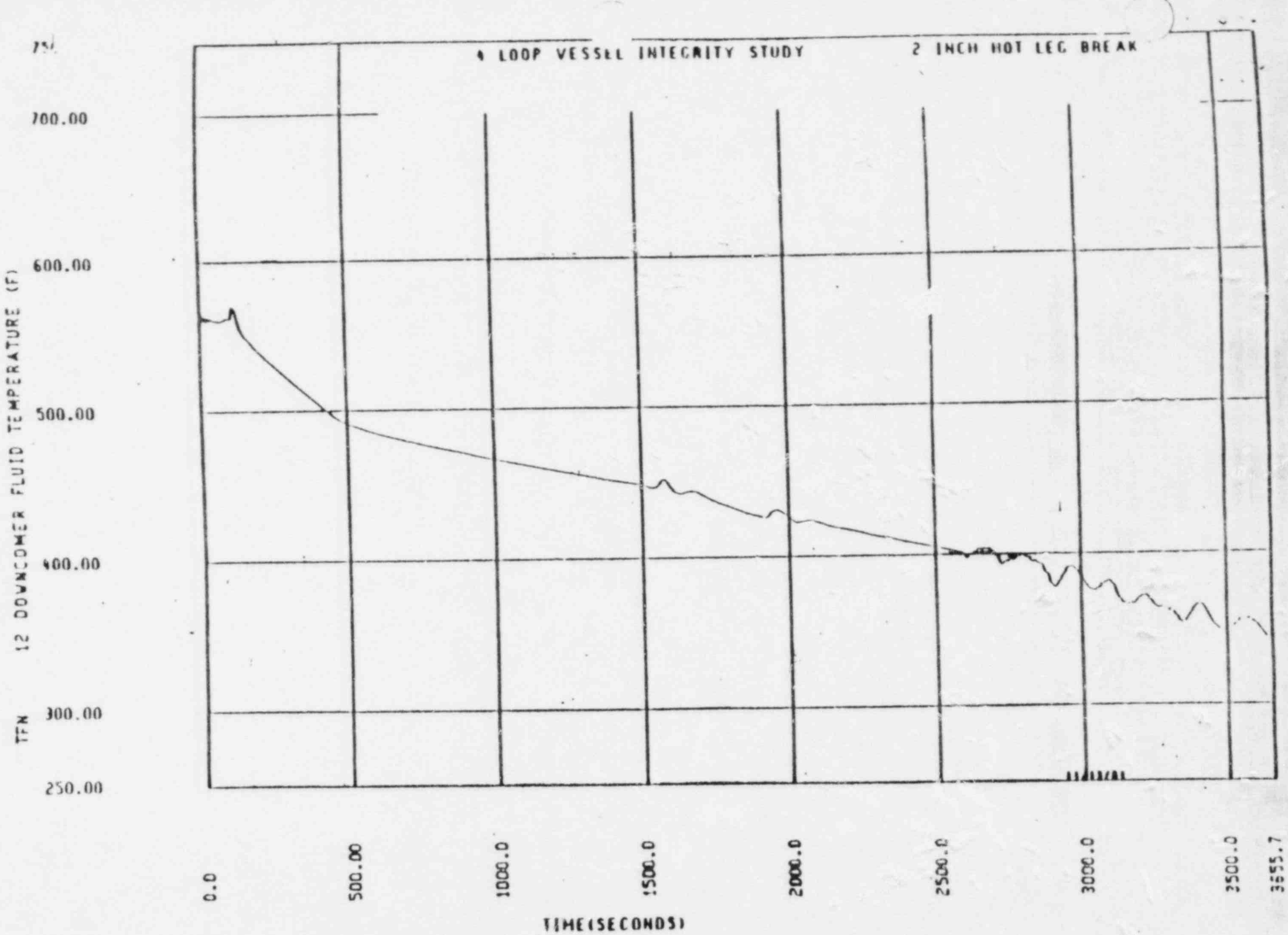


Figure III.1-5 Downcomer Fluid Temperature 4 Loop 2 Inch Hot

DISCUSSION ON FLUENCE CALCULATIONS

BY

S. ANDERSON, WESTINGHOUSE

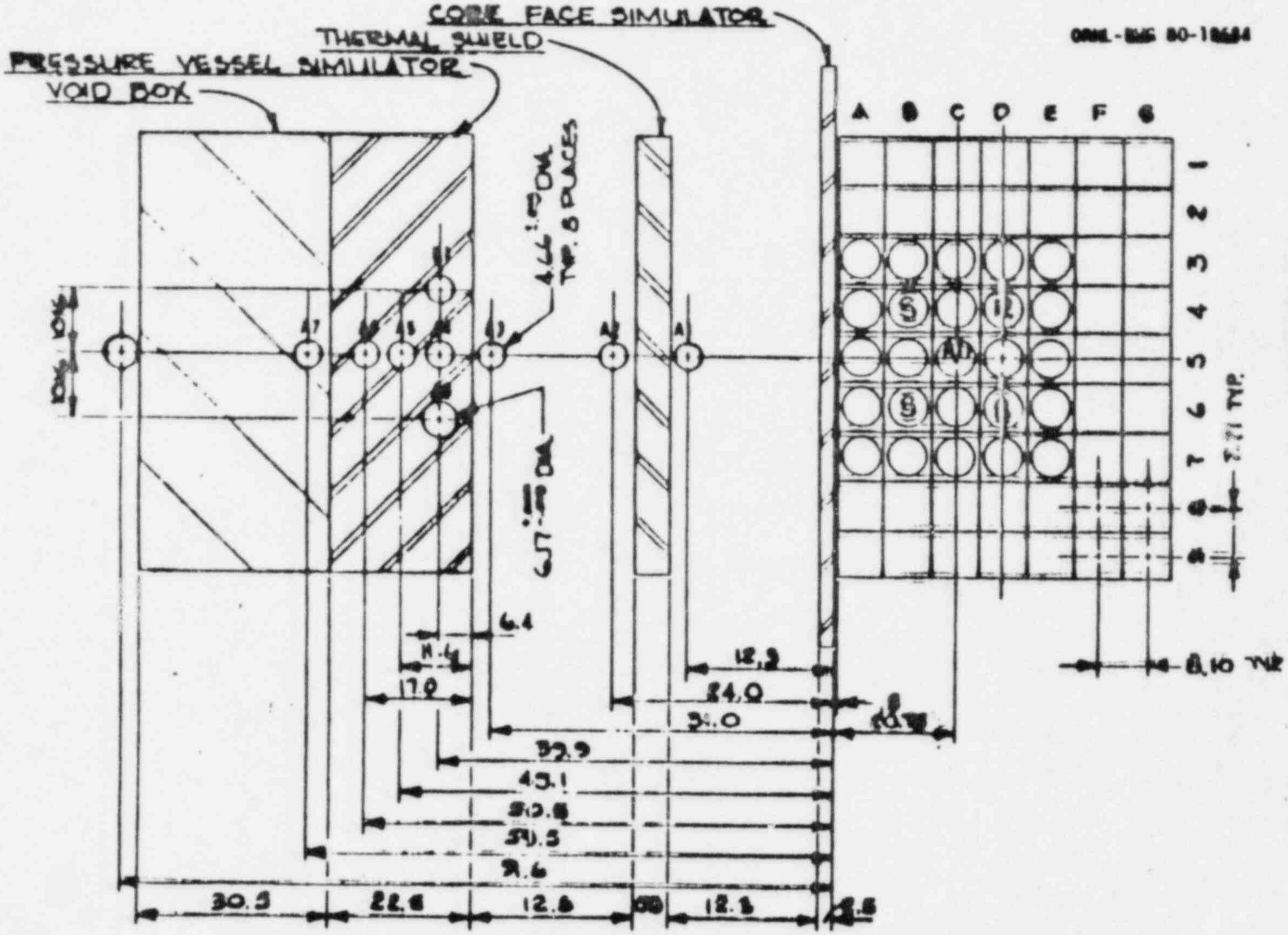


FIGURE 1.1.3. PCA Pressure Vessel Benchmark Facility - Plan View: 12/13 Configuration. (All dimensions in cm)

PCA PARTICIPANT

	WEST.	A	Q	Y
COMPUTER CODE	DOT	DOT	DOT	DOT
QUADRATURE	S_8	S_8	S_8	S_8
P_L ORDER	P_1	P_3	P_3	P_3
XSEC LIBRARY	GAM82T	VITAMIN-C	CASK	CASK

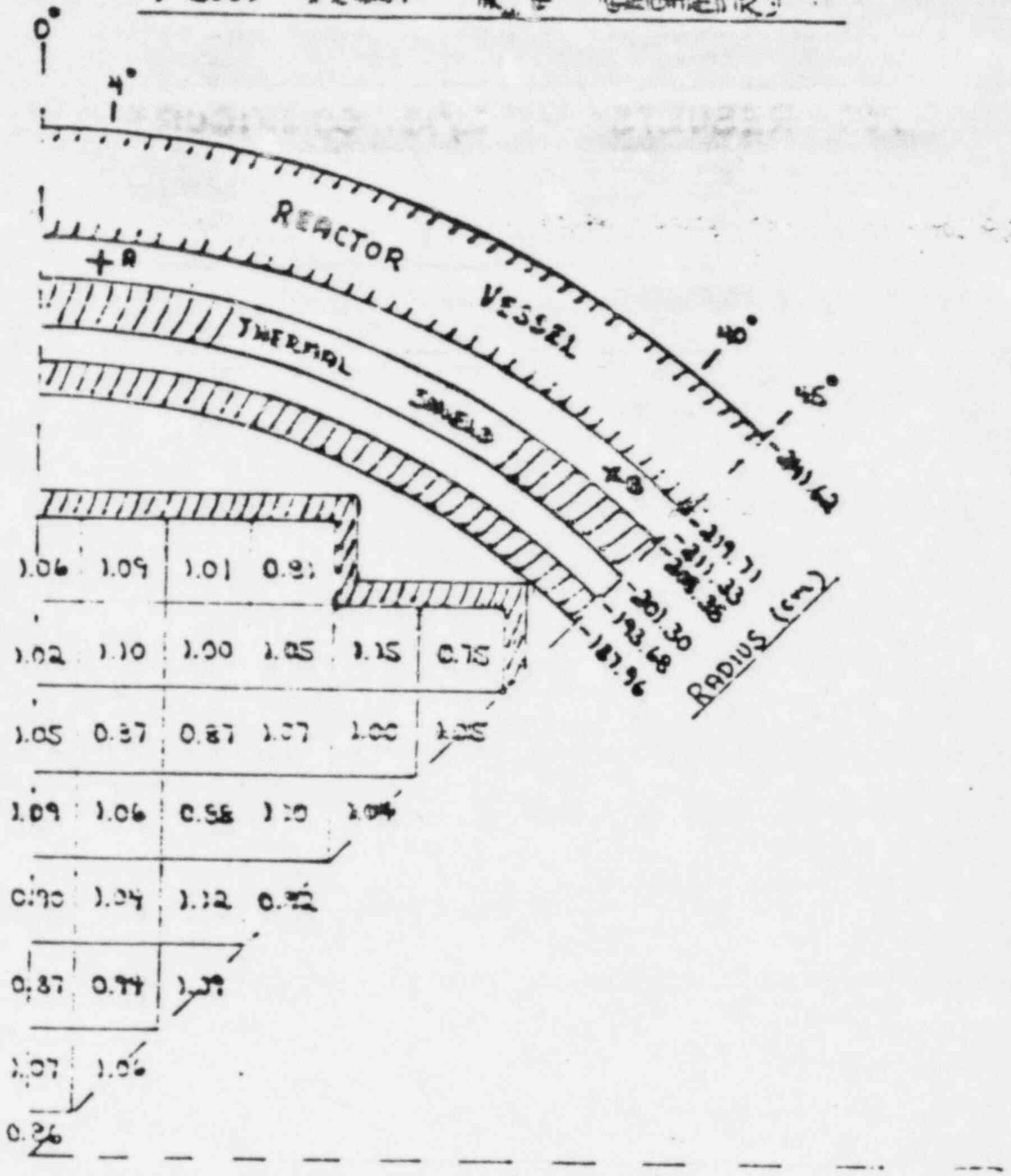
PCA RESULTS - 12/13 CONFIGURATION

LOCATION	ϕ (E > 1.0 MeV) WEST.	A	Q	Y
A_0	1.54 (-4)	1.66 (-4)		1.72 (-4)
A_1	3.68 (-6)	3.39 (-6)	3.21 (-6)	3.50 (-6)
A_2	3.88 (-7)		3.53 (-7)	4.13 (-7)
A_3	1.29 (-7)	1.21 (-7)	9.71 (-8)	1.29 (-7)
A_4	3.94 (-8)	3.96 (-8)	3.57 (-8)	4.39 (-8)
A_5	1.89 (-8)	1.84 (-8)	1.63 (-8)	2.03 (-8)
A_6	8.33 (-9)	7.88 (-9)	6.84 (-9)	8.65 (-9)

NOTE: NUMBERS IN PARENTHESES REFER TO POWERS OF TEN.

FIGURE 1

4-LOOP PLANT R-θ GEOMETRY



REV NO	REV DATE	AUTHOR DATE	CHK'D BY DATE	CHK'D BY DATE

WESTINGHOUSE FORM 56213D

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

aug 9, 1982

SUBJECT: SUMMARY OF MEETING WITH WESTINGHOUSE OWNERS GROUP (WOG) ON JULY 30, 1982, CONCERNING THE PRESSURIZED THERMAL SHOCK (PTS) ISSUE

Introduction

The meeting was held in Bethesda, Maryland on July 30, 1982, at the request of the NRC staff to discuss the subjects on the tentative agenda (Enclosure 1). The discussions followed the actual agenda (Enclosure 2). The attendees of the meeting are identified in Enclosure 3. Material used by the staff to introduce the discussions concerning the fracture mechanics analysis are included in Enclosure 4.

Discussion

The WOG discussions regarding our concerns include the following highlights in the order covered in the actual agenda, Enclosure 2:

1. SBLOCA Sizes Resulting in Stagnation Flow

The W SBLOCA analysis determined the smallest break size which would result in the maximum pressure with the min. temperature. This would be the same as loss of natural circulation or stagnation flow. W found that for pipe sizes equal to or less than 1 1/2" that natural circulation was maintained. For 4 loop plants a LOCA involving a 3" pipe size appeared to be the area of stagnation flow. For 3 loop and 2 loop plants, 2" pipe breaks are the areas of concern for stagnation flow.

2. Heat Transfer Coefficient

The heat transfer coefficient for the thermal hydraulic transient analysis comes from the code used in the analysis and is relatively low. A higher value was used in the fracture mechanics analysis which is explained in WCAP 10019 and ranges between 100 and 300.

3. Effect of Secondary System

The temperature transient which bounds the case for operator action on the secondary side is the no mixing transient. The operator action on the secondary side has some benefits and some bad effects. The present procedures have steps in the recovery phase to reduce secondary pressure. However, the review of the emergency recovery guidelines recommended that these steps be removed. W indicated that credit cannot be given for operator action because the operator would not take action until after warm prestressing has taken effect which is 30 minutes into the transient.

4. Steam Generator Tube Rupture Event

An evaluation of the steam generator tube rupture event (SGTR) was considered in the WOG May 28, 1982 submittal. The May 28 report addresses the

B/10

SGTR event with no mixing which results in a low probability number.

5. Effect of Warming RWST

The effect of warming RWST was covered in WCAP 10019. W has not considered temperature of ECCS water higher than 120°. Effect on fracture mechanics - good. Effect on ECCS - Heat transfer benefit
- Containment Cooling penalty
Overall - more benefits than penalties

6. Crack Arrest Phenomenon

The W found for all SBLOCA cases that the crack always arrested below the upper shelf. The W analysis assumed a finite flaw and considered that the crack growth would not exceed that which it would have been if they would have assumed a continuous flaw. The W analysis does consider warm prestressing. The heat transfer coefficient used was the WCAP 10019 data. It was not 300° continuous through the vessel wall.

The explanation of the method of calculating K follows: W calculated K for an elliptic flaw. Along the flaw length the largest calculated K was used in the fracture mechanics analysis. As the crack propagated the K was calculated and never allowed to be larger than if it were calculated assuming an infinite length flaw. For shallow flaws the W analysis would use the elliptic flaw. For deeper flows they assumed infinite flaw length. The W analysis did not consider clad effects. The major identifiable difference between the staff analysis and the W analysis was in the flaw shape.

7. Frequency of SBLOCA with Stagnation Flow

W determined that the probability of SBLOCA which would result in stagnation flow was $6x10^{-4}$. This figure basically comes from an adjustment of WASH1400 data — 10^{-4} as a medium adjusted to $6x10^{-4}$ as a mean for double ended breaks of 1 1/2" to 6" — The information does not come from operating history (350 reactor years of operation)(WASH1400 considered all industry experience - W adjustment considers nuclear experience only). The loss of 2 RCP seals would result in stagnation flow and would have a probability range of 10^{-5}.

The $6x10^{-4}$ figure is in the May 28 report. All incoming lines to the RCS either have orifices or check valves. Considering all the above W believes that the $6x10^{-4}$ probability could be reduced by an order of magnitude ($6x10^{-5}$) and still be conservative.

8. Other Events Resulting In Stagnation Flow

Other events which result in stagnation flow include SGTR with RCP off. The conditional probability of this event is 10^{-2}. (The probability of losing power to RCP is 10^{-3}). There are other ways for stagnation flow to occur but all are not presently known and their probabilities are not known.

9. Response After Caucus

The probability of other events which result in stagnation flow centers around events resulting in no mixing. Stagnation in a single loop is not synonymous with loss of natural circulation. For SBLOCA greater than

1 1/2" (1 1/2 to 4") W believes 6×10^{-4} is the proper probability and it may be too conservative. W believes SBLOCA produced cracks would arrest. They believe only high probability events resulting in stagnant flow effect only one loop.

10. Crack Arrest Concerns

The crack arrest phenomenon for the SBLOCA appears to be the major difference between the staff's view and the W views. W assumed an elliptic flaw. The staff assumed an infinite length flaw. Therefore, this has caused a major difference in calculating the RT_{NDT}.

Conclusions

The staff was to check their methods of fracture mechanics analysis and WOG was requested to recheck their methods and both the staff and WOG were requested to try to come to an agreement. The WOG was also requested to look further into the identification and probabilities of events which result in stagnation flow.

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

Enclosures:
1. Tentative Agenda
2. Actual Agenda
3. Attendance List
4. NRC Discussion Material

cc w/enclosures:
See next page

ORB#4:DL

MEETING SUMMARY DISTRIBUTION

Licensee: Westinghouse

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

Docket File
NRC PDR
L PDR
ORB#4 Rdg
GLainas
JStolz
Project Manager-GVissing
Licensing Assistant-RIngram
OELD
Heltemes, AEOD
IE
SShowe (PWR) or CThayer (BWR), IE
Meeting Summary File-ORB#4
RFraley, ACRS-10
Program Support Branch

ORAB, Rm. 542
BGrimes, DEP
SSchwartz, DEP
SRamos, EPDB
FPagano, EPLB

Meeting Participants Fm. NRC:

NRAnderson
MVagins
GLauben
ARubin
FSchroeder
SHanauer
TMurley
BSheron
RKlecker
RRantala
SIsrael

Enclosure 1

TENTATIVE AGENDA

FOR

MEETING WITH WESTINGHOUSE OWNERS GROUP

JULY 30, 1982

1. SBLOCAs (sizes) of concern which result in loss of natural circulation but where continued high pressure can exist to cause a PTS concern.

2. Understanding frequency and basis for frequency of SBLOCAs defined in #1.

3. Calculation details on crack arrest phenomenon for the SBLOCAs of #1.

4. Justification for credit for warm prestressing for the SBLOCAs of #1.

5. Heat transfer coefficient when there is a loss of natural circulation during SBLOCA.

6. Effect of secondary system and what operator does to secondary system on SBLOCA. What would operator be expected to do, with what probability, and does secondary depressurization make PTS more or less severe.

7. Analysis (frequency, transient, and fracture mechanics) of steam generator tube rupture event with pressurizer on the good steam generator ("Ginna" event in other SG).

8. Analysis (frequency, transient and fracture mechanics) of SBLOCA where the break is isolated early in the transient.

9. Effect of warming RWST on the SBLOCA and the fracture mechanics analysis. Include both benefit for PTS, and detriment for ECCS consideration.

Enclosure 2

ACTUAL AGENDA FOR

MEETING WITH WESTINGHOUSE OWNERS GROUP

JULY 30, 1982

1. SBLOCAs (sizes) of concern which result in loss of natural circulation but where continued high pressure can exist to cause a PTS concern.

2. Heat transfer coefficient when there is a loss of natural circulation during SBLOCA.

3. Effect of secondary system and what operator does to secondary system on SBLOCA. What would operator be expected to do, with what probability, and does secondary depressurization make PTS more or less severe.

4. Analysis (frequency, transient, and fracture mechanics) of steam generator tube rupture event with pressurizer on the good steam generator ("Ginna" event in other SG).

5. Effect of warming RWST on the SBLOCA and the fracture mechanics analysis. Include both benefit for PTS, and detriment for ECCS consideration.

6. Calculation details on crack arrest phenómenon and consideration of warm prestressing for the SBLOCAs of #1.

7. Understanding frequency and basis for frequency of SBLOCAs.

8. Other events causing stagnation flow

9. Response After Caucus

10. Crack Arrest Concerns

Enclosure 3

ATTENDANCE LIST FOR

MEETING WITH WOG CONCERNING PRESSURIZED THERMAL SHOCK
JULY 30, 1982

NRC

G. Vissing
N. R. Anderson
M. Vagins
G. Lauben
A. Rubin
F. Schroeder
S. Hanauer
T. Murley
B. Sheron
R. Klecker
R. Rantala
S. Israel

NUTECH w/FPL

J. Copeland

FP&L

J. Moaba
S. K. Matharan

Va. Elec. & Power Co.

D. W. Lippard

W

R. Sero
W. Bamford
T. Meyer
K. Balkey
B. King
H. Julian
M. J. Hitchler
D. R. Sharp
J. A. Rumancik

Con. Edison of NY

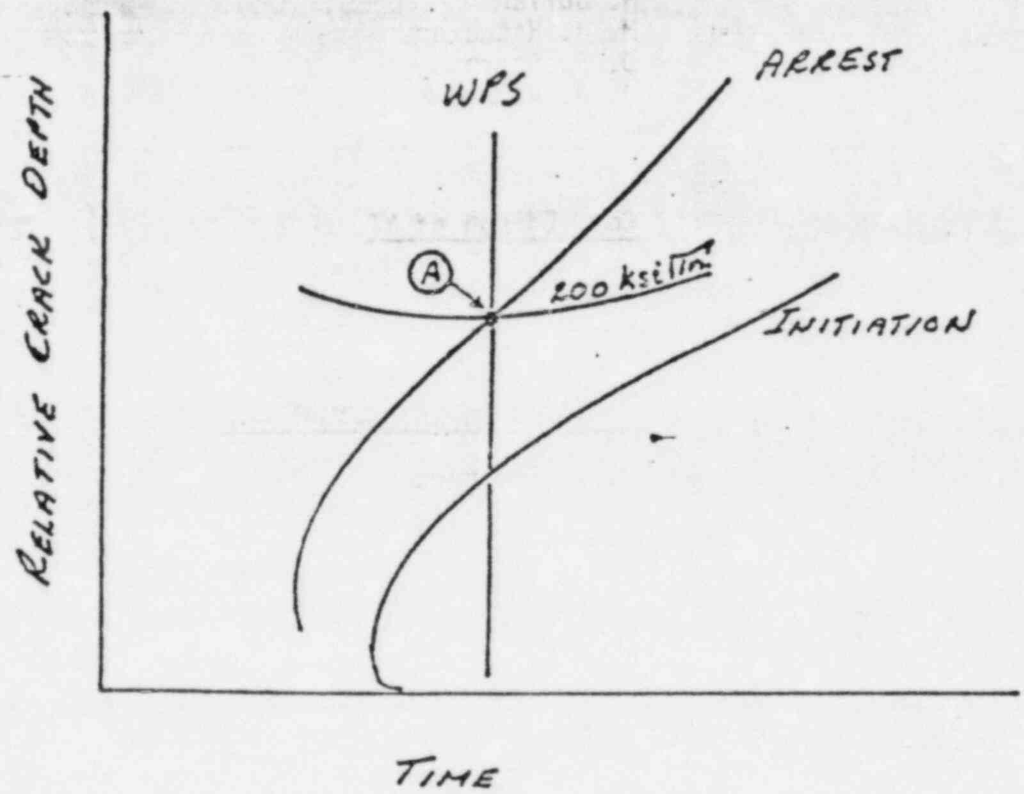
D. M. Speyer

Carolina Power & Light Co.

J. J. Sheppard

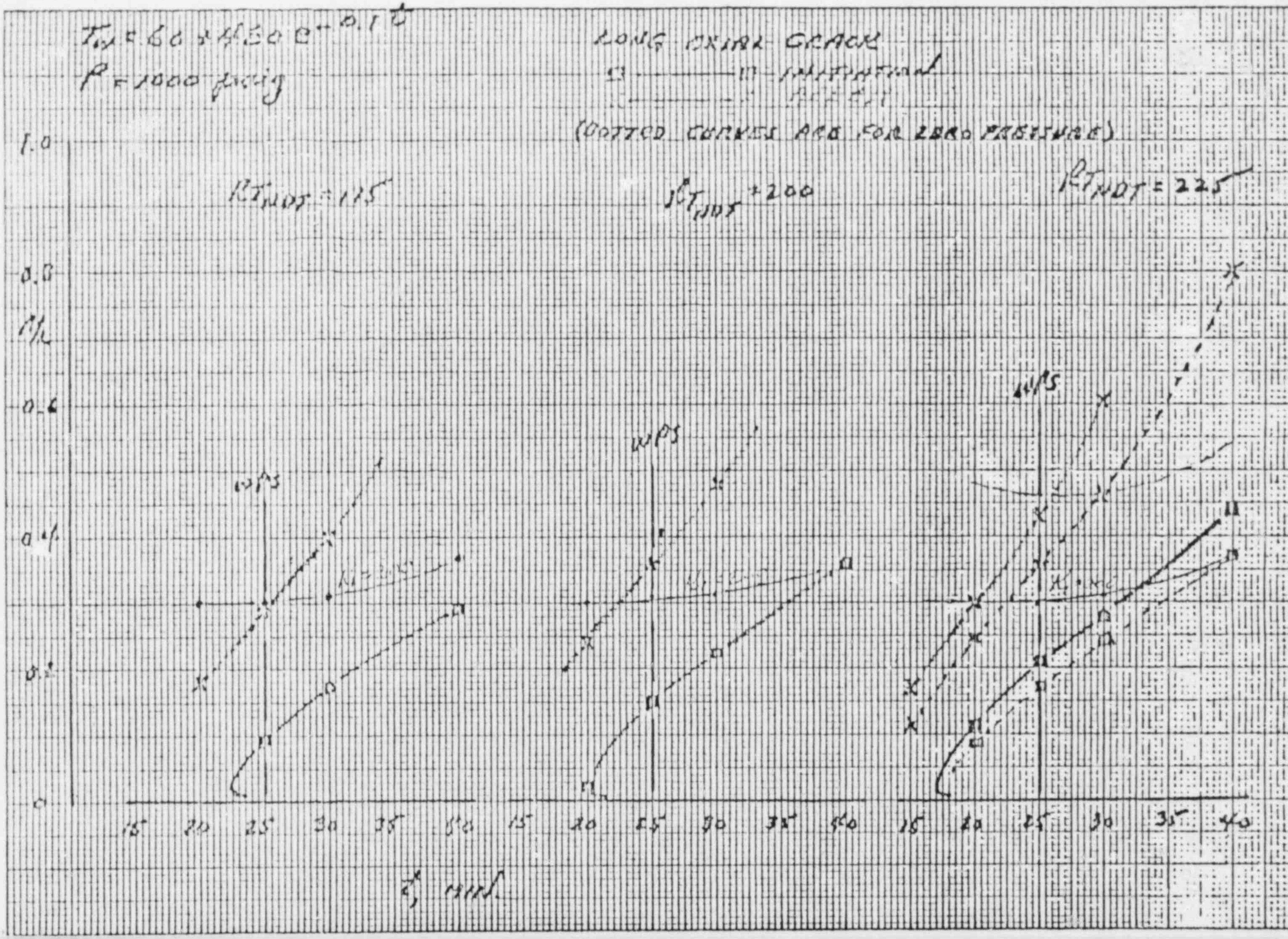
Enclosure 4

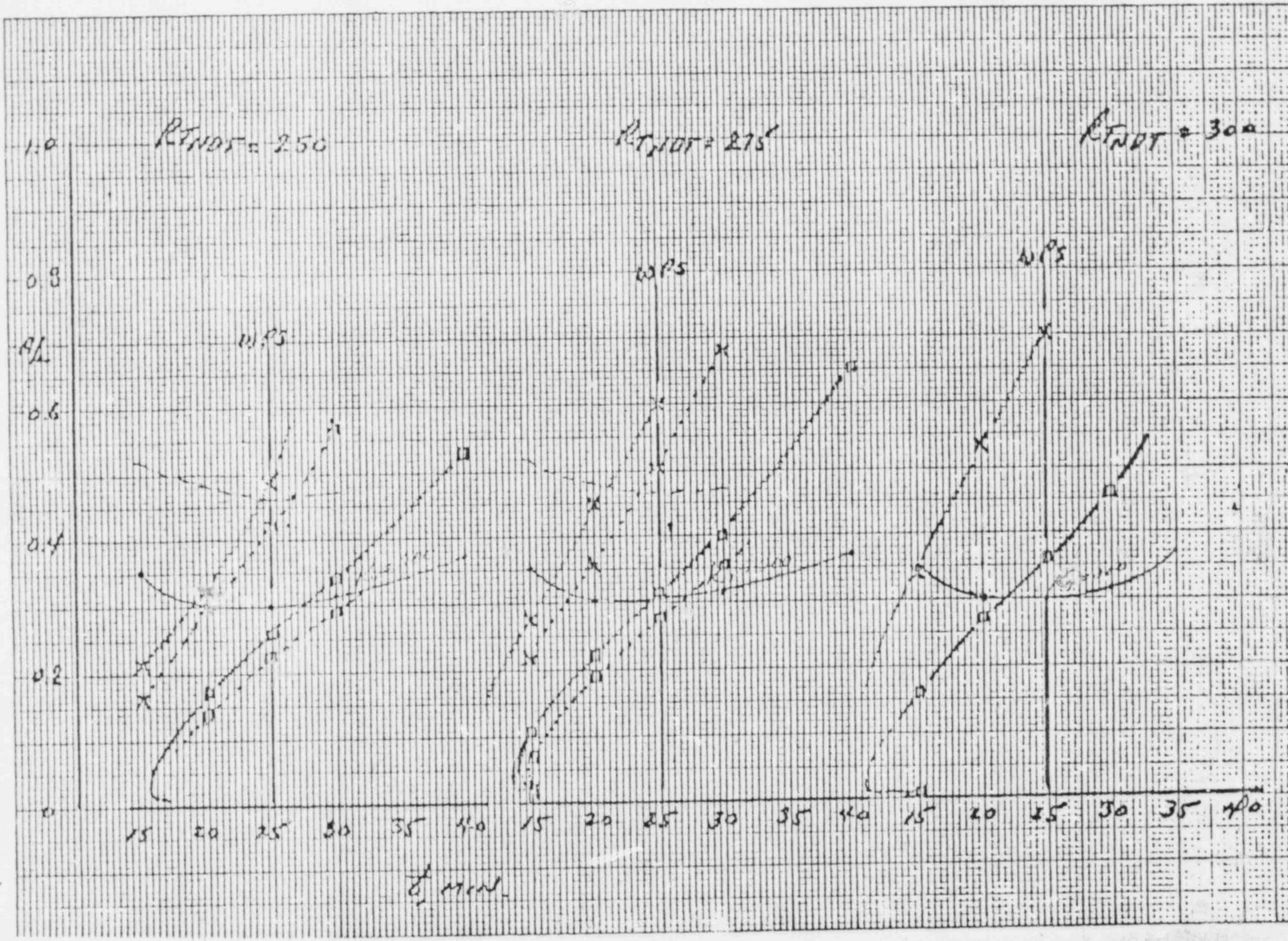
NRC CRACK ARREST MODEL



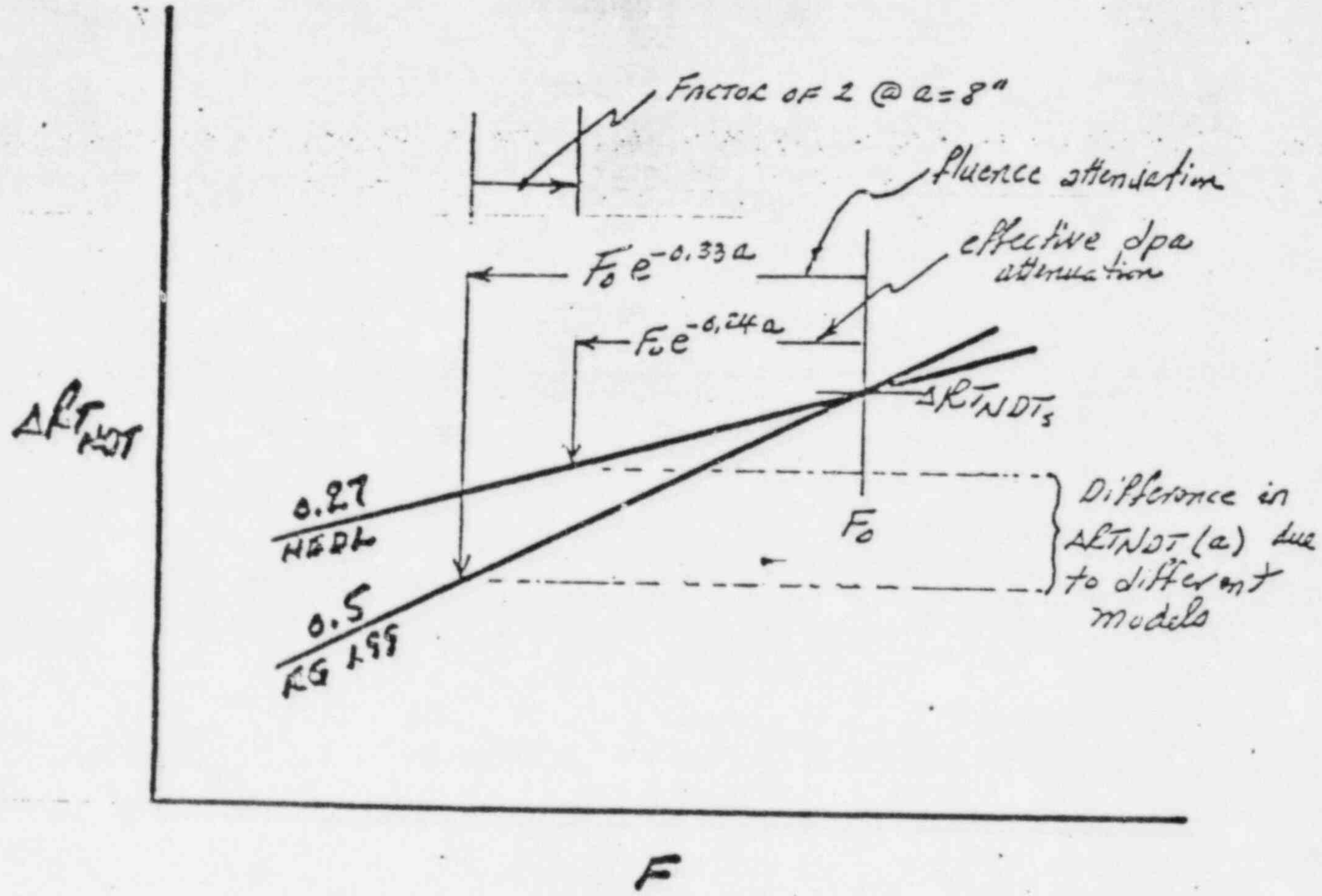
FOR A GIVEN THERMAL TRANSIENT, AND A GIVEN RT_{NDT} AT THE VESSEL INNER RADIUS, MAXIMUM PRESSURE IS DETERMINED WHEN CRACK ARREST OCCURS AT 200 ksi√in. AT THE TIME WARM PRESTRESSING OCCURS (POINT "A" IN THE ABOVE FIGURE)

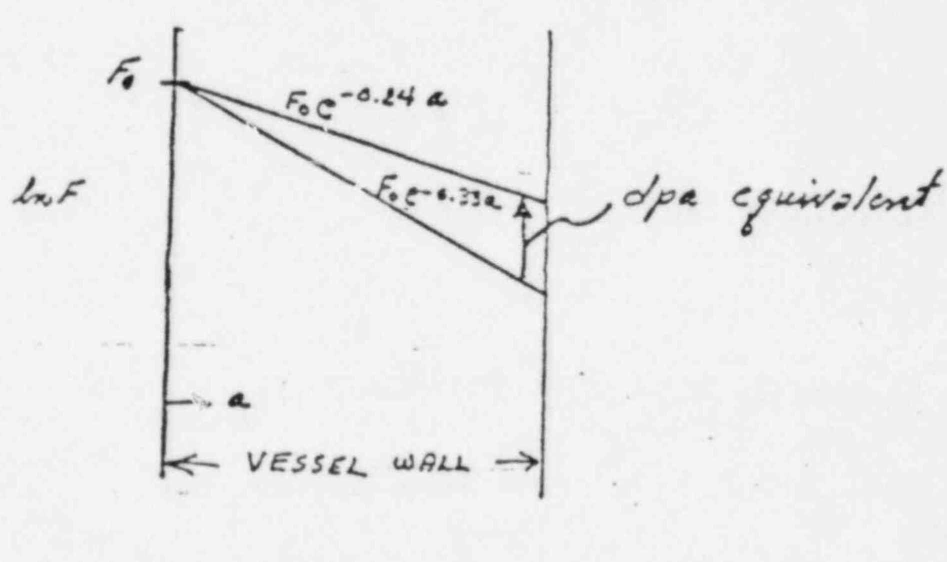
KEUFFEL & ESSER CO.

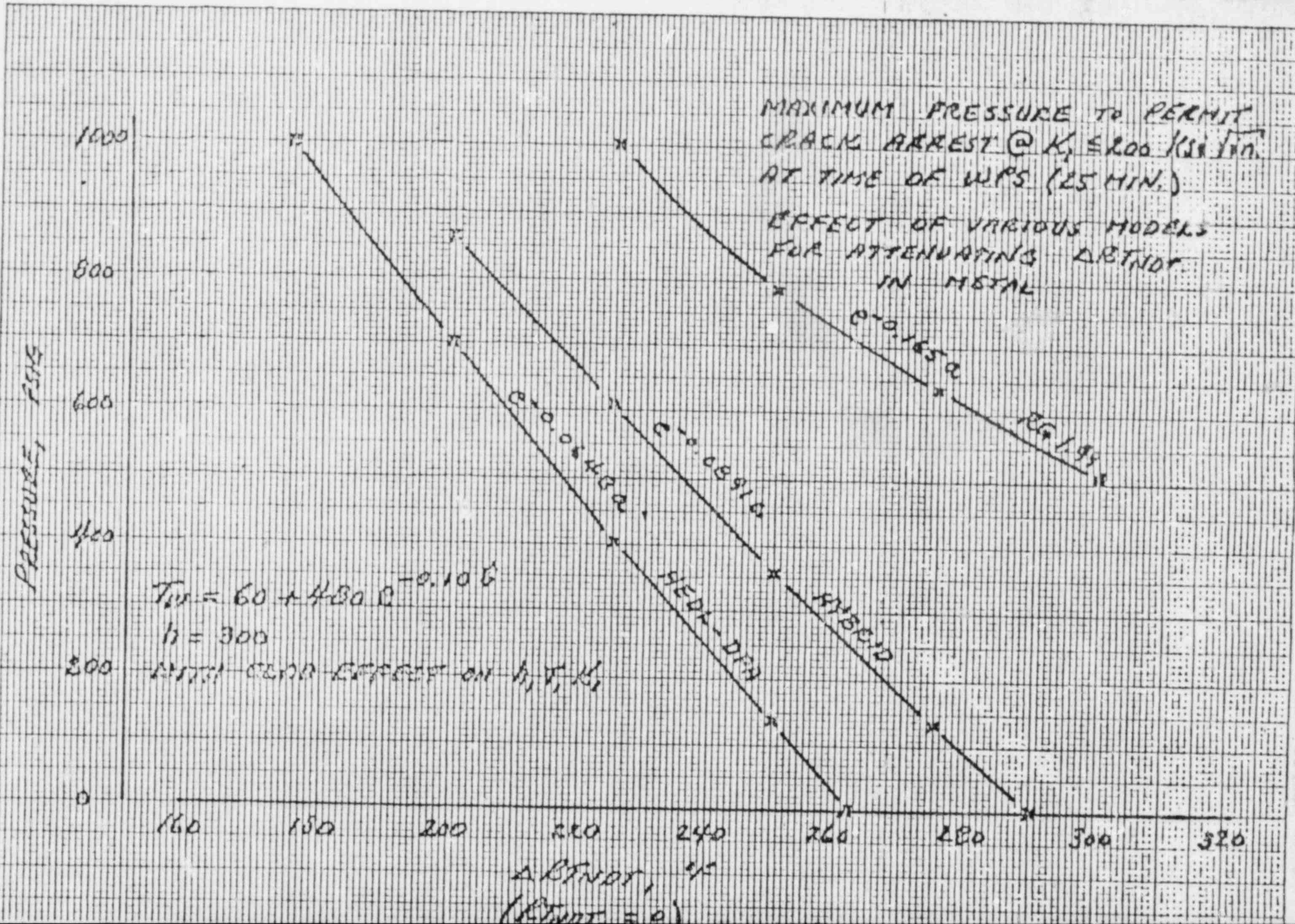


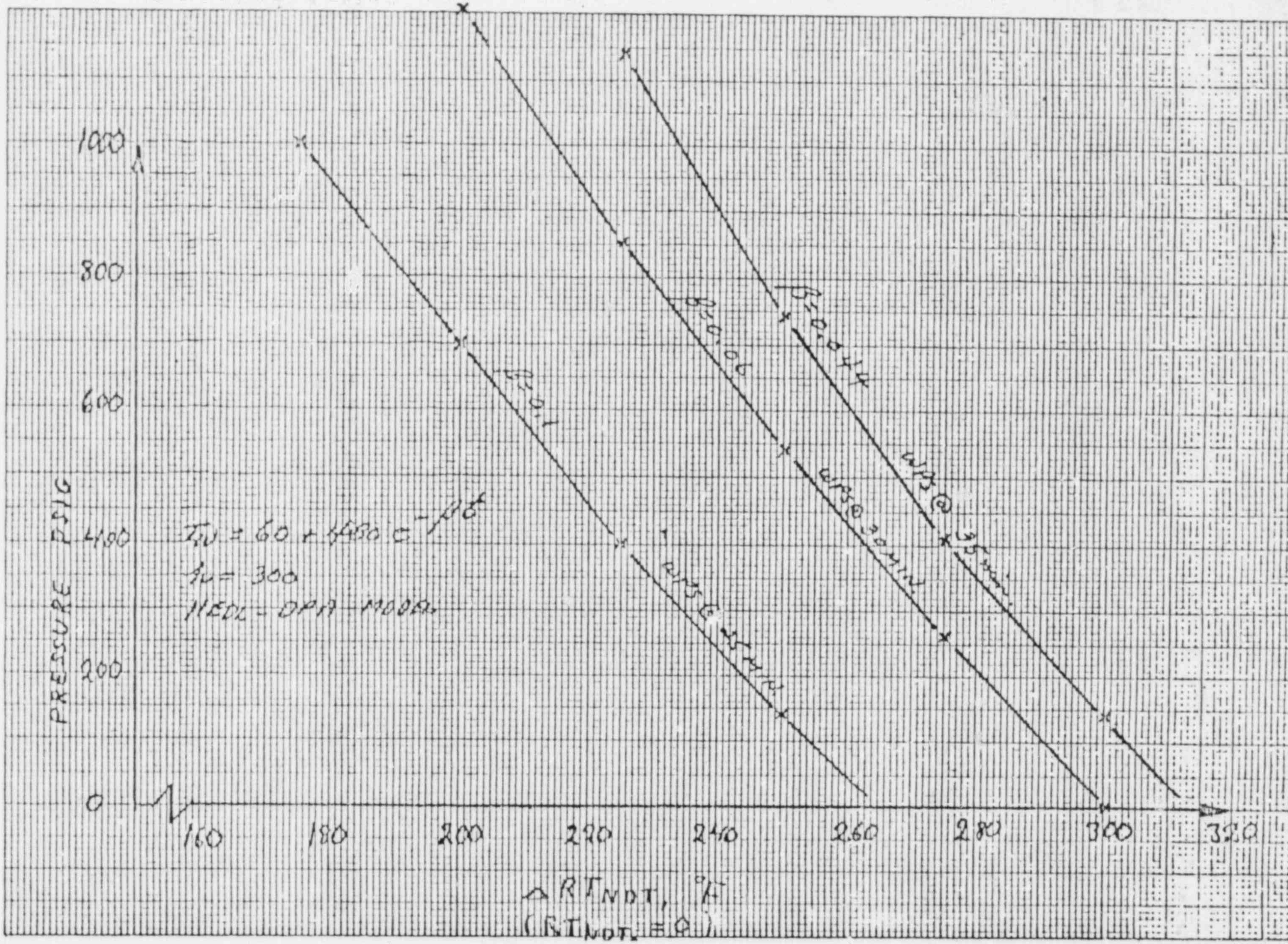

RTNDT = 250
RTNDT = 275
RTNDT = 300
WPS
WPS
WPS
a/L
1.0
0.8
0.6
0.4
0.2
0
15
20
25
30
35
40
15
20
25
30
35
40
15
20
25
30
35
40
t, min

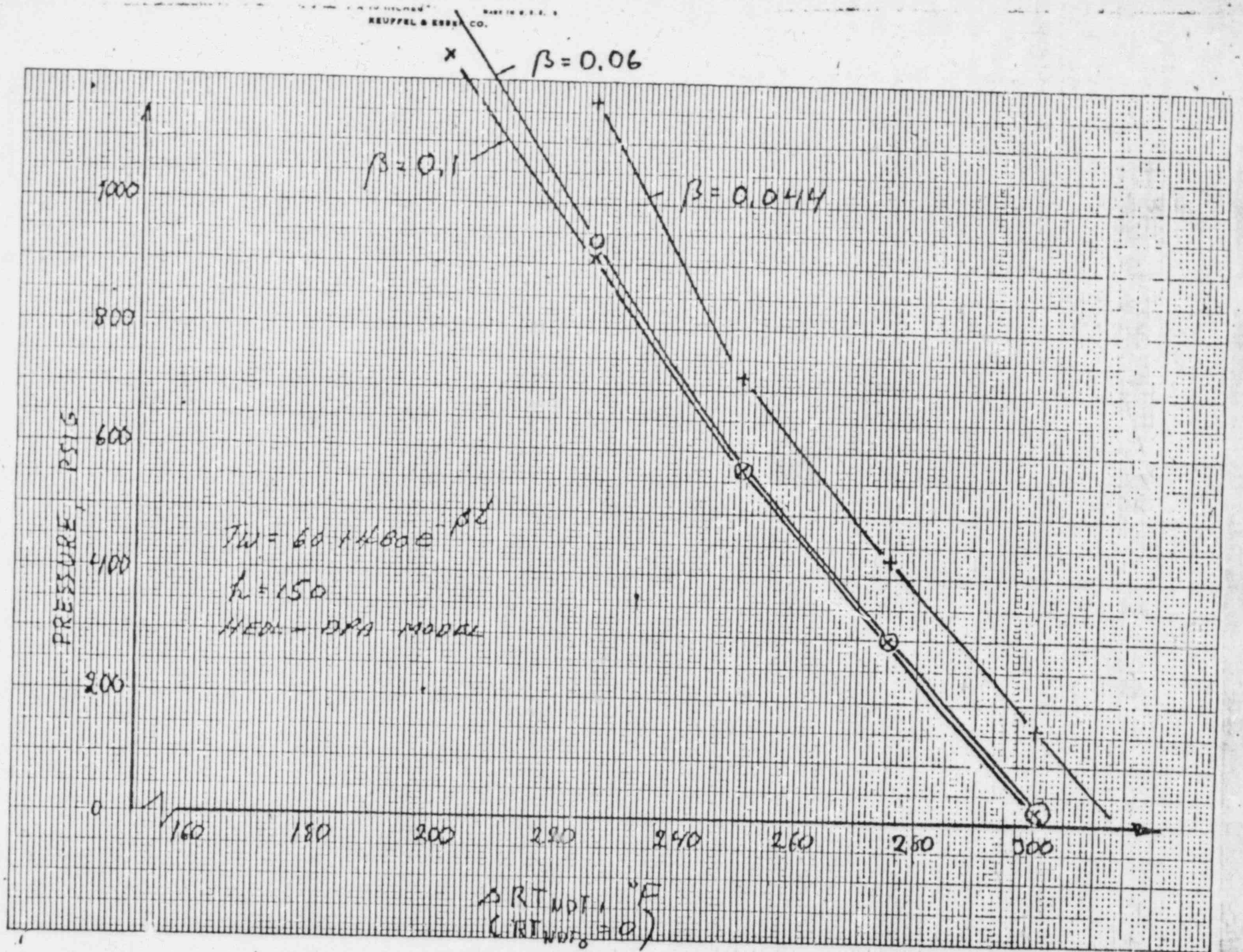
ΔRT_{NDT} ATTENUATION IN VESSEL WALL

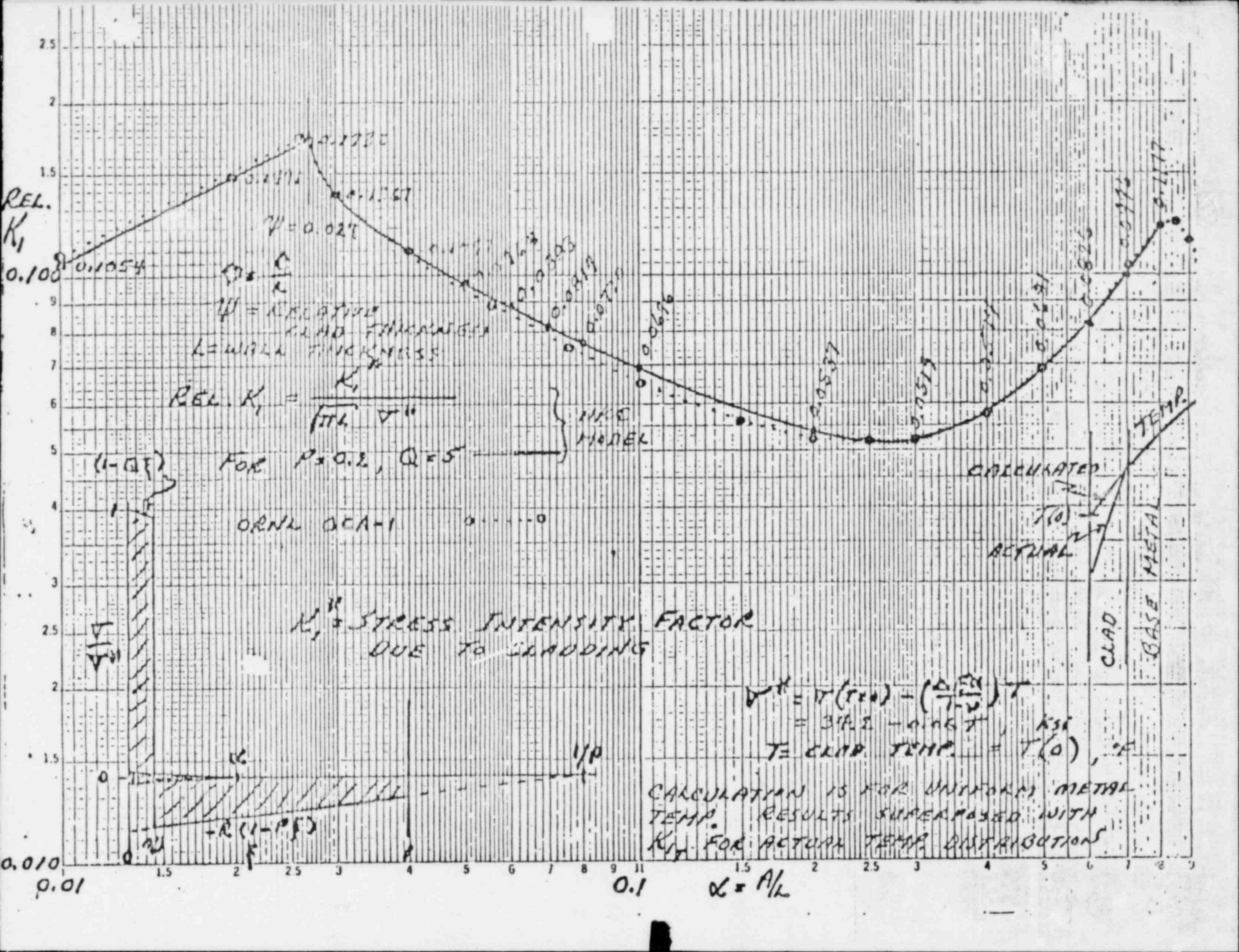





MAXIMUM PRESSURE TO PERMIT CRACK ARREST @ $K_I \leq 200$ KSI $\sqrt{in.}$ AT TIME OF WPS (25 MIN.)
EFFECT OF VARIOUS MODELS FOR ATTENUATING ΔRTNDT IN METAL
PRESSURE, PSIG
1000
800
600
400
200
0
160
180
200
220
240
260
280
300
320
ΔRTNDT, °F
(RTNDT = 0)
$e^{-0.165a}$
RG 1.99
$e^{-0.0891a}$
HYBRID
$e^{-0.0640a}$
HEDL-DPA
$T_w = 60 + 480\,e^{-0.10t}$
h = 300
WITH CLAD EFFECT ON h, T, K_I


PRESSURE PSIG
1000
800
600
400
200
0
160
180
200
220
240
260
280
300
320
ΔRTNDT, °F
(RTNDT = 0)
β=0.044
WPS @ 35 min.
β=0.06
WPS @ 30 MIN.
β=0.1
WPS @ 25 MIN.


KEUFFEL & ESSER CO.
β = 0.06
β = 0.1
β = 0.044
1000
800
600
400
200
0
PRESSURE, PSIG
Tw = 60 + 460e^-βt
h = 150
HEDL - DPA MODAL
160
180
200
220
240
260
280
300
ΔRTNDT, °F
(RTNDT0 = 0)


REL. K_I
0.100
0.010
0.01
0.1
$\alpha = A/L$
0.1054
$\psi = 0.027$
$\Omega = \frac{C}{L}$
ψ = RELATIVE CLAD THICKNESS
L = WALL THICKNESS
REL. $K_I = \frac{K_I^*}{\sqrt{\pi L}\ \sigma^*}$
FOR $P = 0.2$, $Q = 5$
ORNL OCA-1
K_I^* = STRESS INTENSITY FACTOR DUE TO CLADDING
0.0676
0.0537
0.0519
CALCULATED
$T(0)$
ACTUAL
CLAD
BASE METAL
TEMP.
0.1177
CALCULATION IS FOR UNIFORM METAL TEMP. RESULTS SUPERPOSED WITH K_{I_T} FOR ACTUAL TEMP. DISTRIBUTIONS
T = CLAD TEMP. = $T(0)$, °F
$(1-\Omega\xi)$
$-R(1-P\xi)$
1/P
$\frac{\sigma}{\sigma^*}$