RV THERMAL SHOCK WITH REPRESSURIZATION

OVERVIEW OF SCOPING ANALYSIS METHODS

AND SOME PRELIMINARY RESULTS

D. J. AYRES

JULY 30 1981

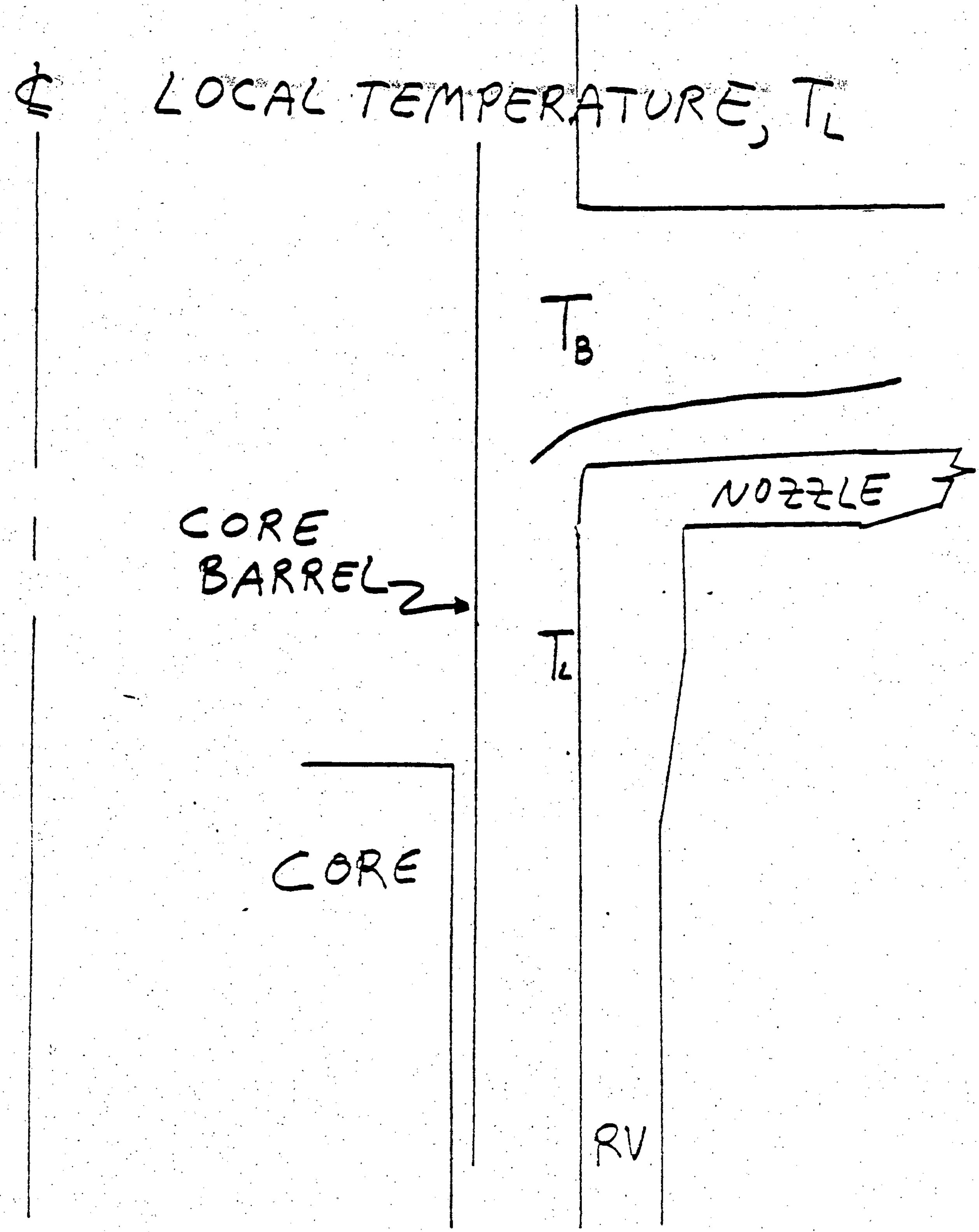
PRESSURIZED THERMAL SHOCK

TRANSIENTS ANALYSES

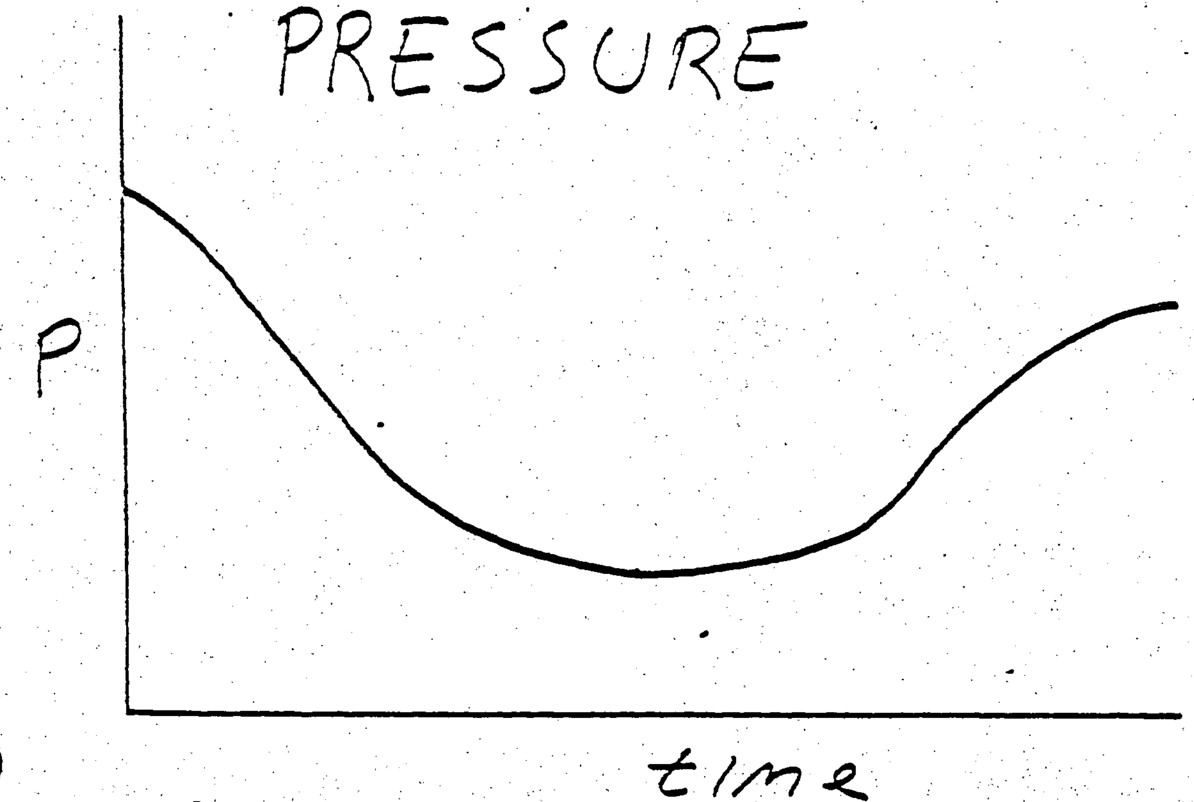
MATERIAL PROPERTIES

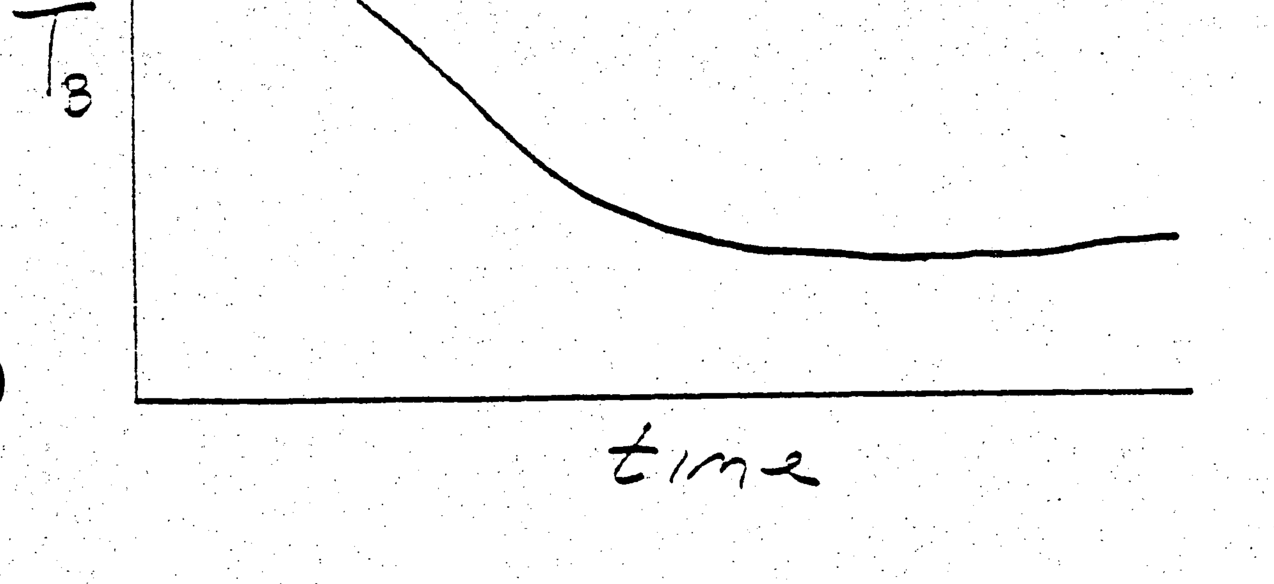
FLUENCE DISTRIBUTIONS

VESSEL INTEGRITY EVALUATIONS

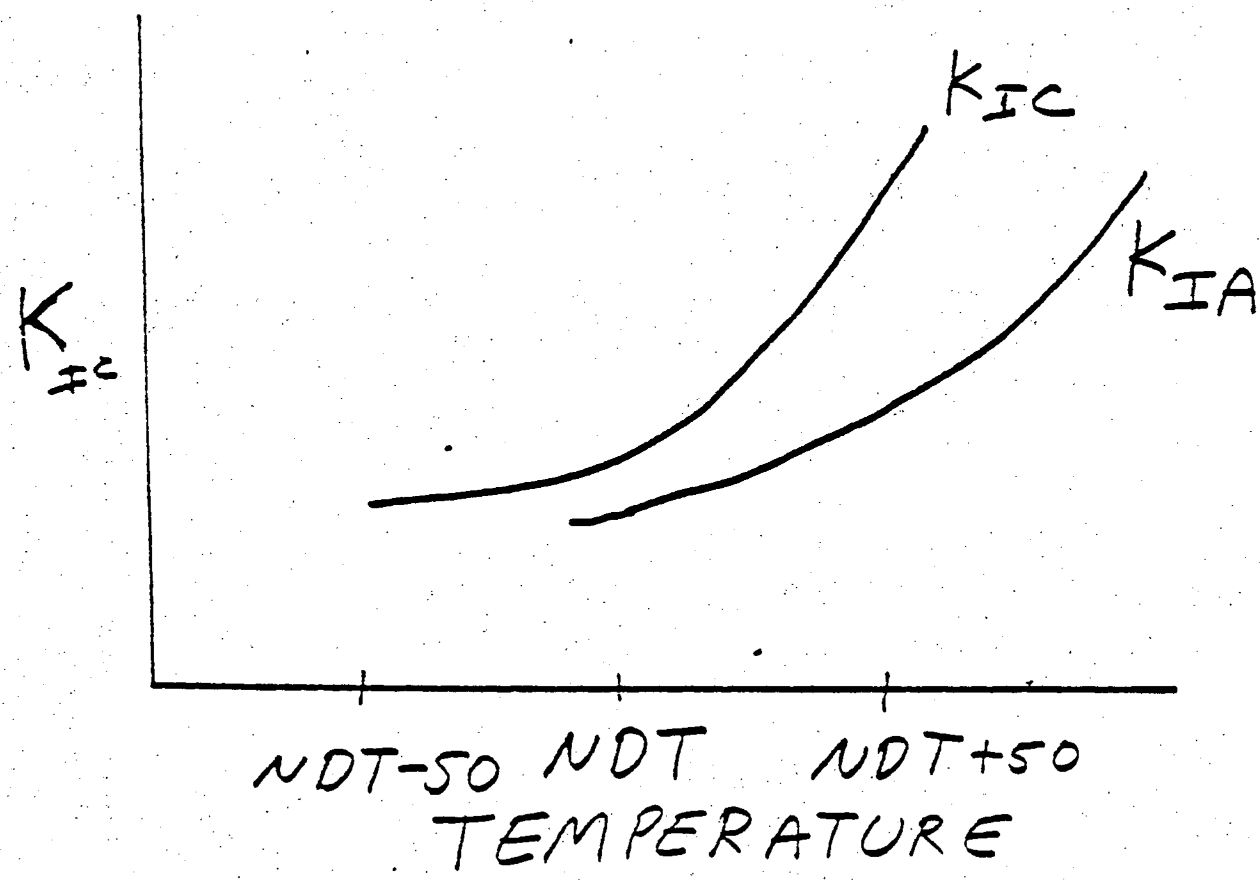


℄
LOCAL TEMPERATURE, T_L
T_B
NOZZLE
CORE
BARREL
T_L
CORE
RV

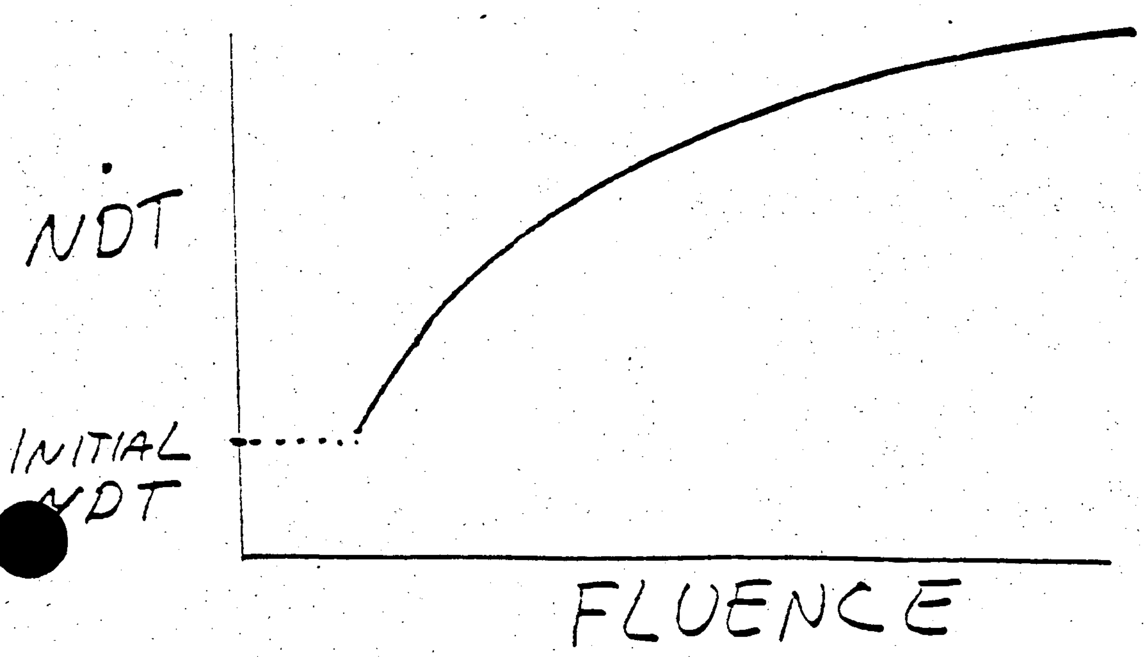

PRESSURE
P
time


BULK TEMPERATURE
T_B
time

TOUGHNESS

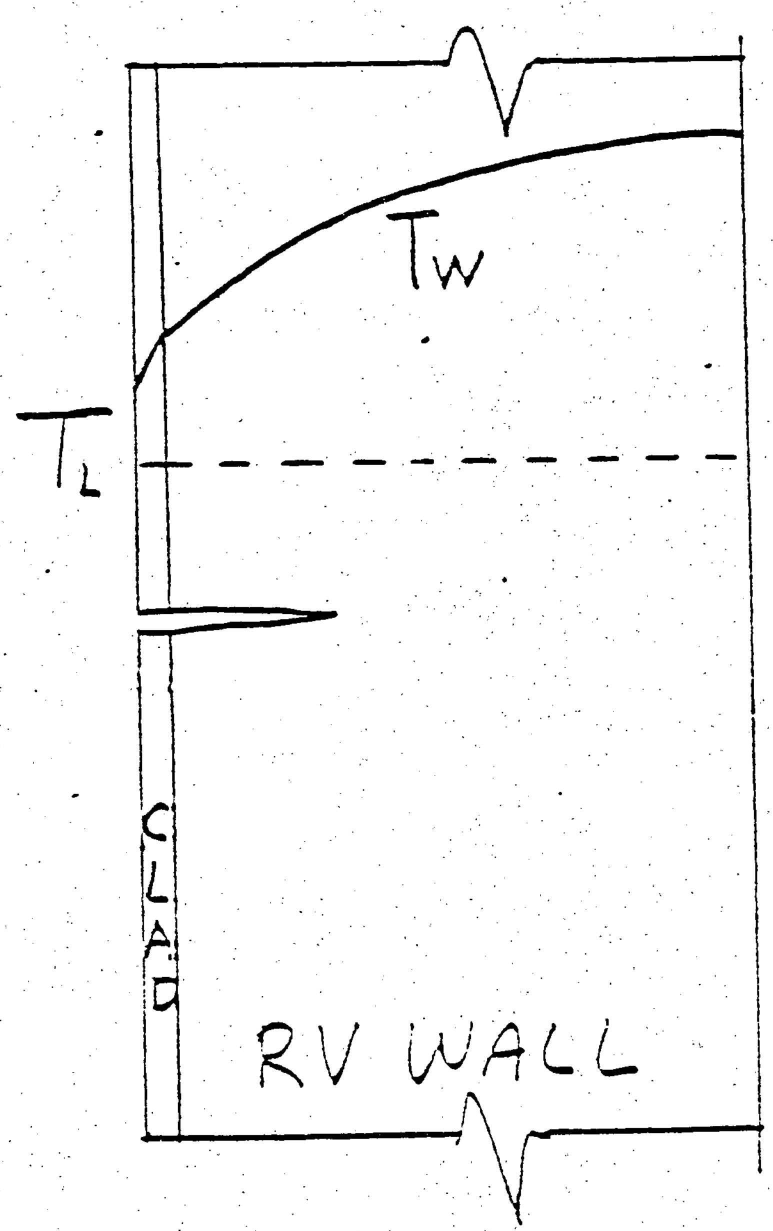


K_{IC}
K_{IA}
K_{IC}
NDT-50
NDT
NDT+50
TEMPERATURE



NDT
INITIAL NDT
FLUENCE

LOCAL HEAT TRANSFER



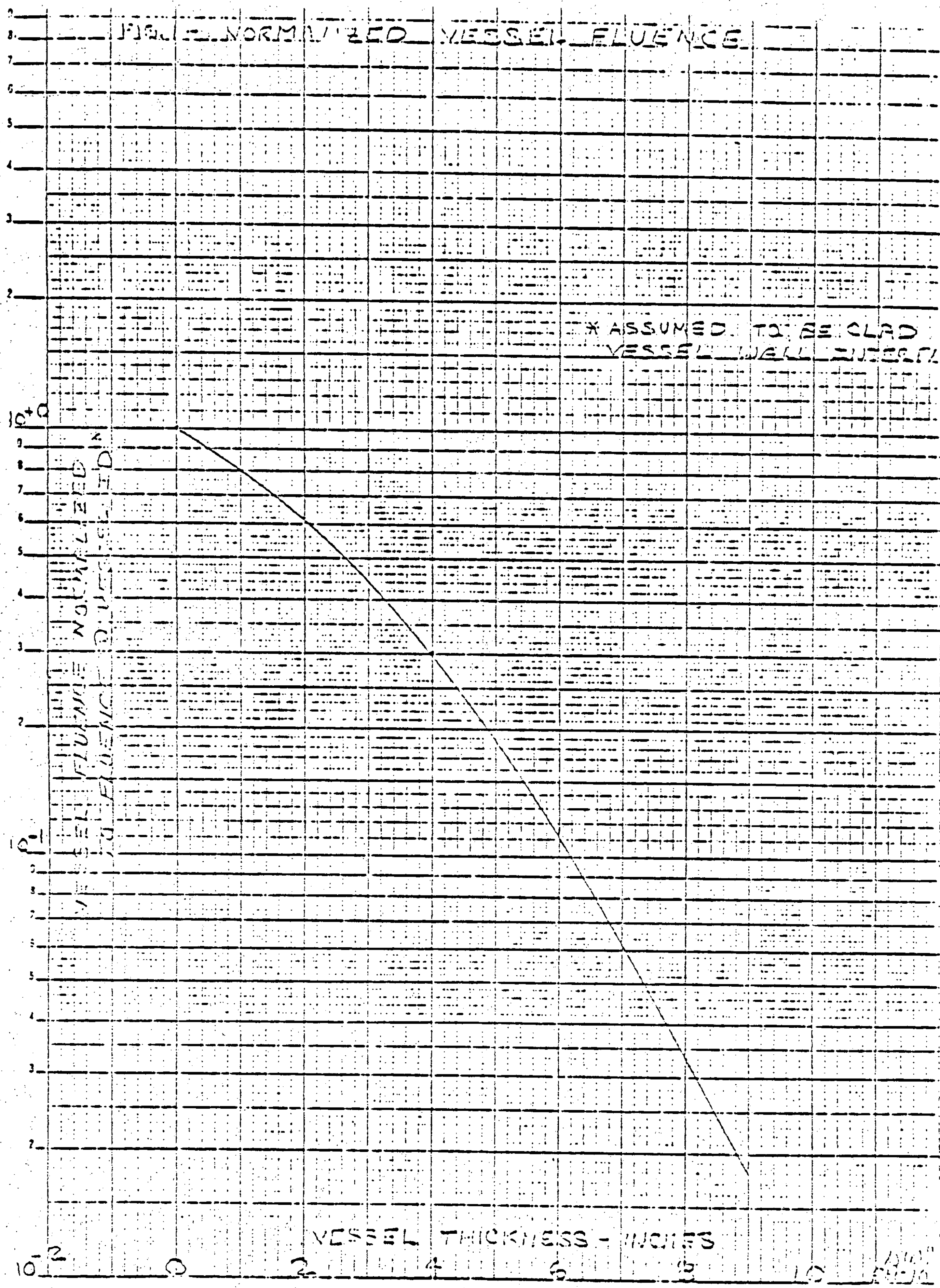
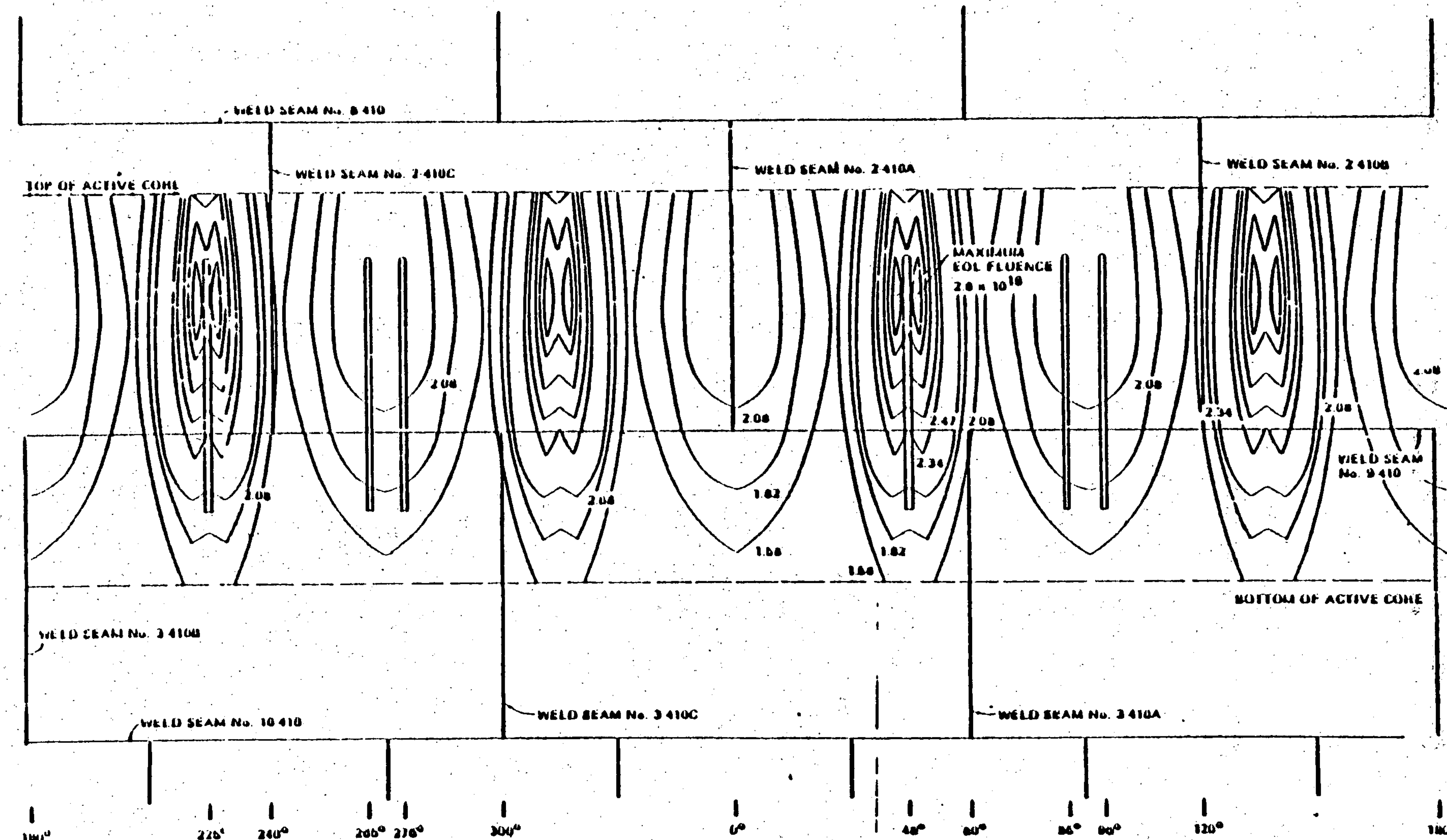
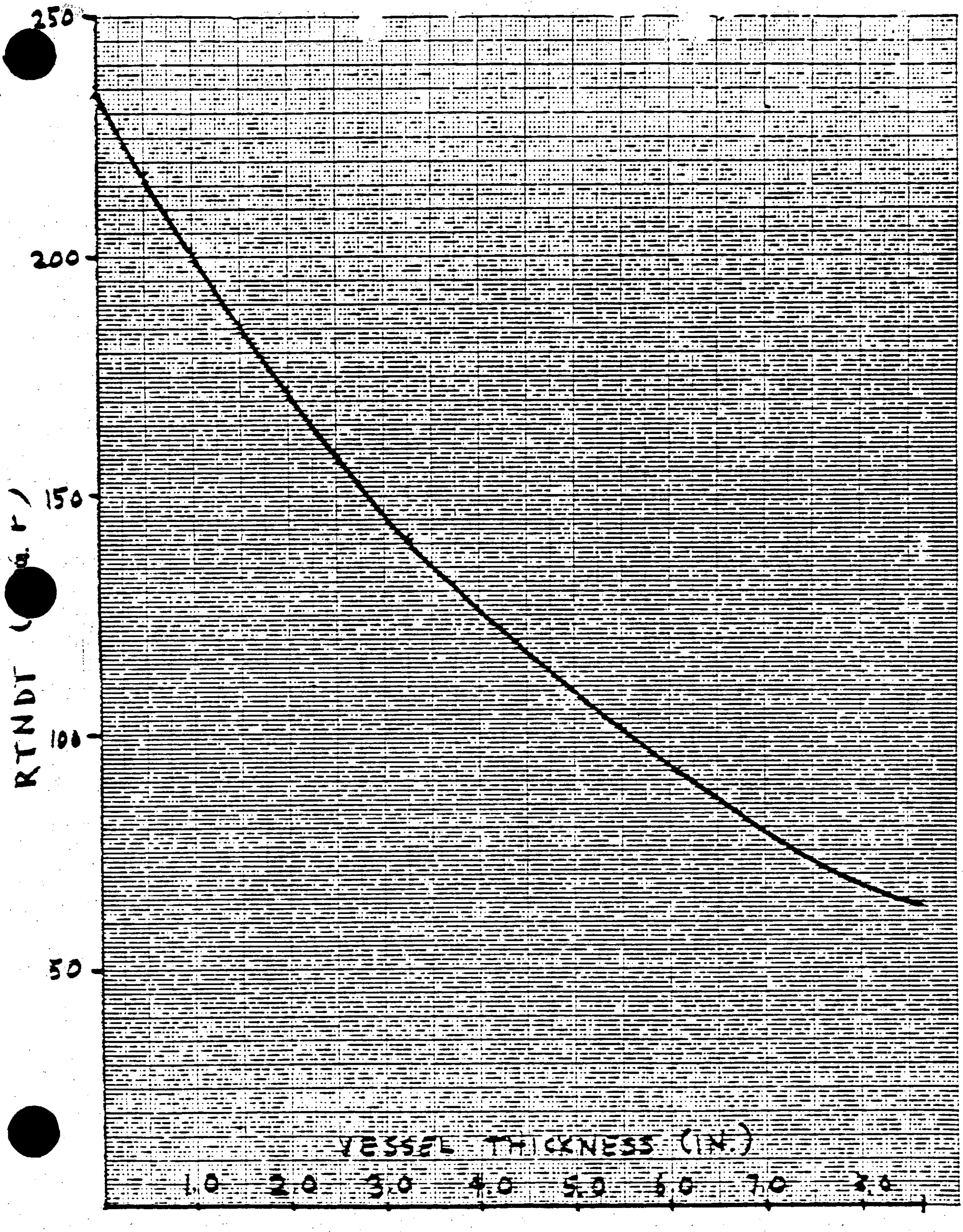


FIG. 1 NORMALIZED VESSEL FLUENCE
* ASSUMED TO BE CLAD
VESSEL WALL INTERN.
VESSEL FLUENCE NORMALIZED
TO FLUENCE ON INSIDE
VESSEL THICKNESS - INCHES
0
2
4
6
8
10
10^{+0}
10^{-1}
10^{-2}

SUPERIMPOSED FLUENCE AND WELD MAPS



LINES OF ISOFLUENCE, E>1MEV ($\times 10^{18}$ n/cm^2)



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

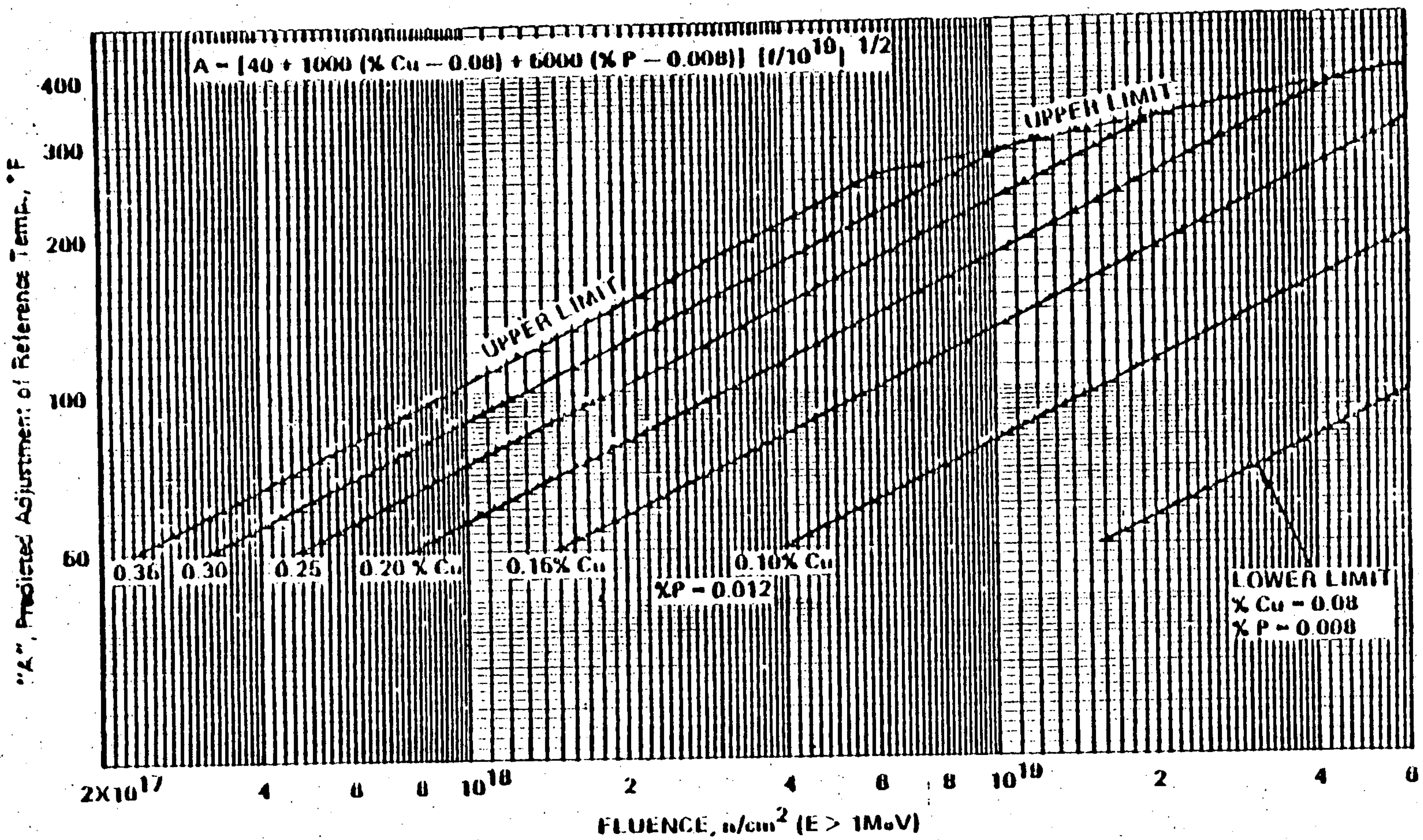
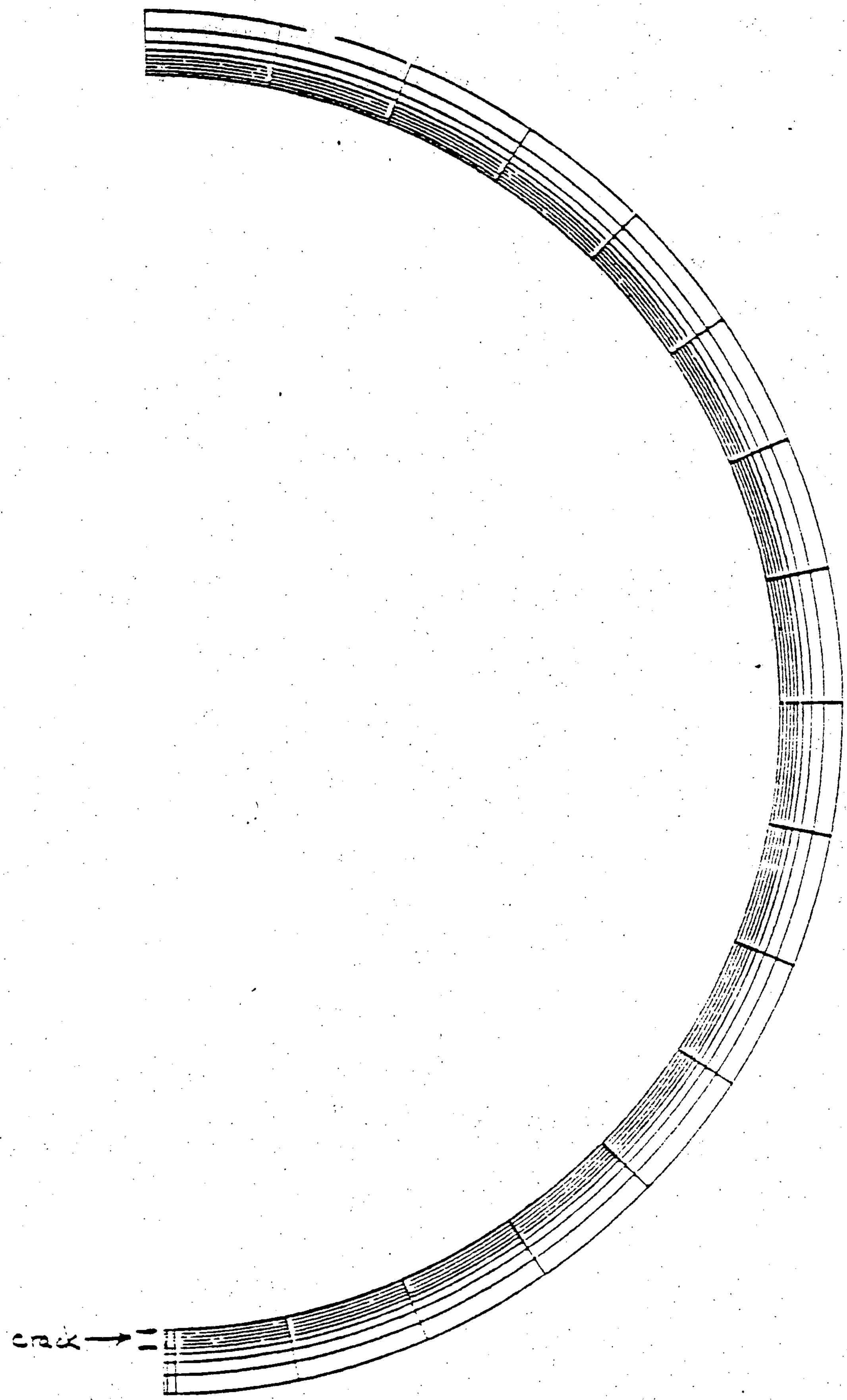
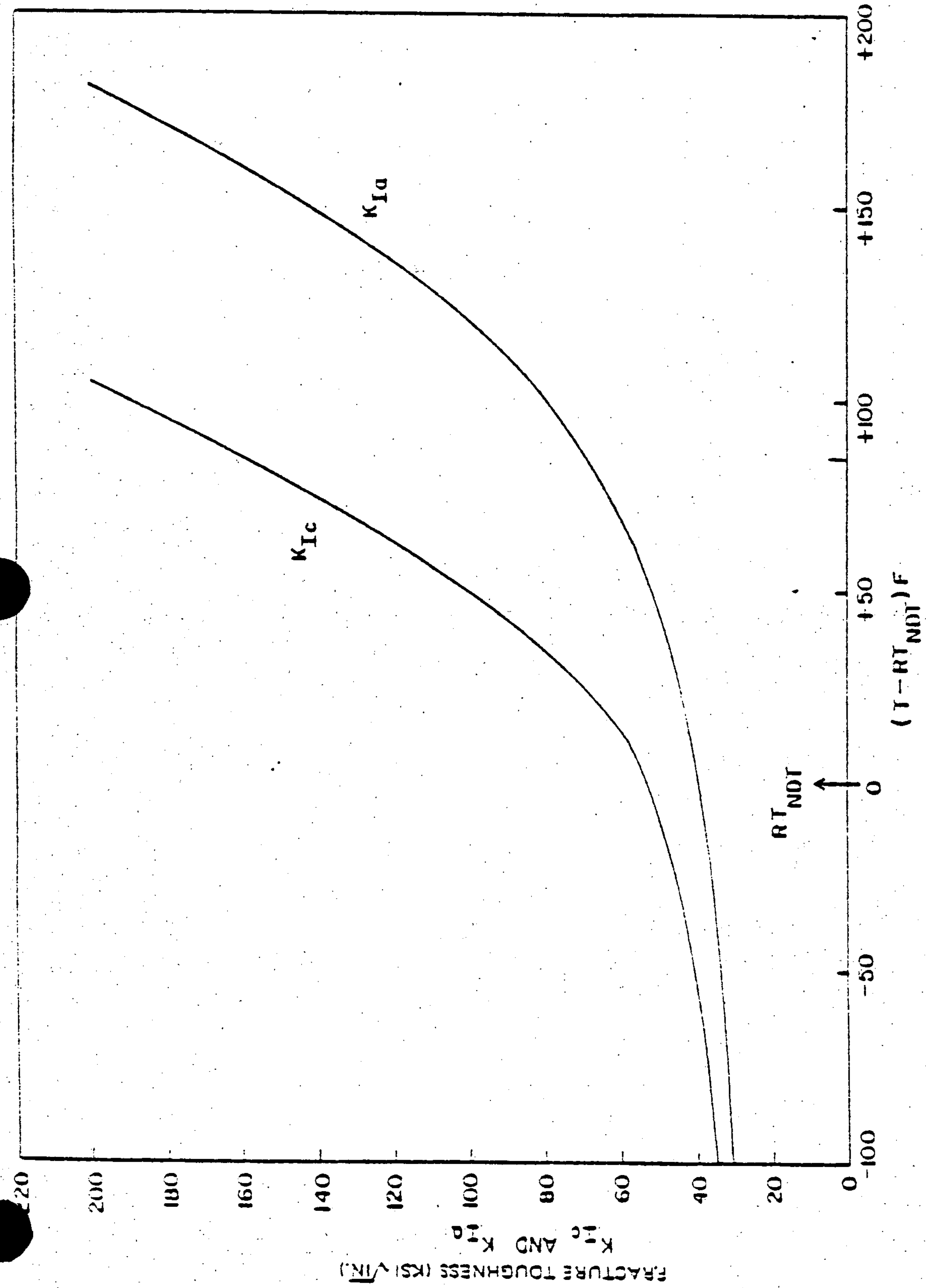


Figure 1 Predicted Adjustment of Reference Temperature, "A", as a Function of Fluence and Copper Content.
For Copper and Phosphorus Contents Other Than Those Plotted, Use the Expression for "A" Given on the Figure.

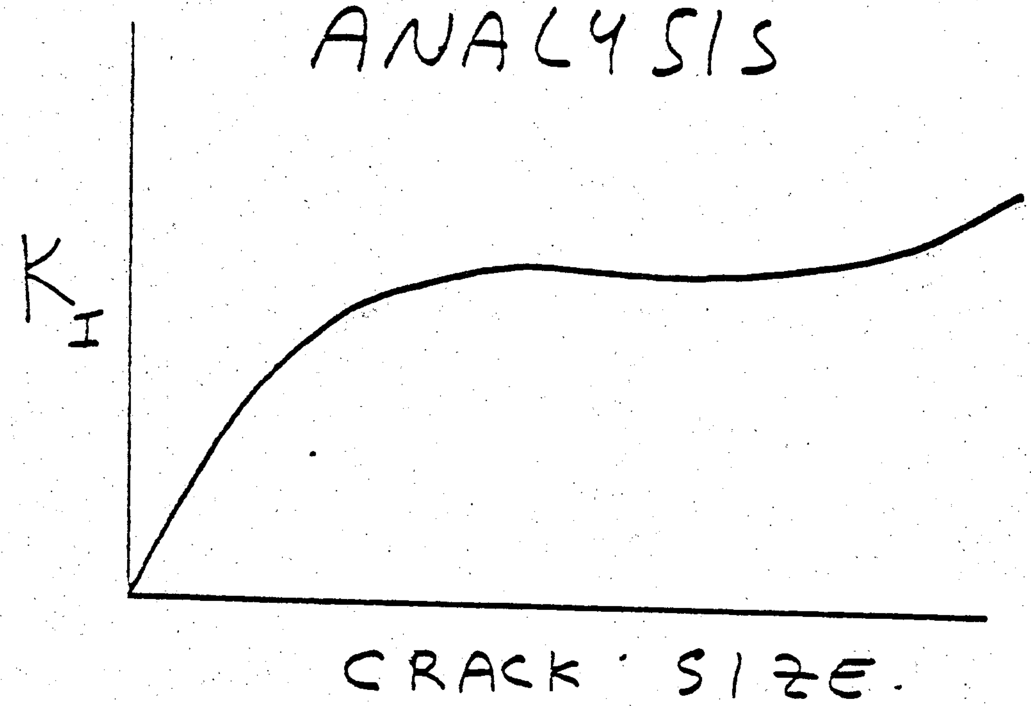


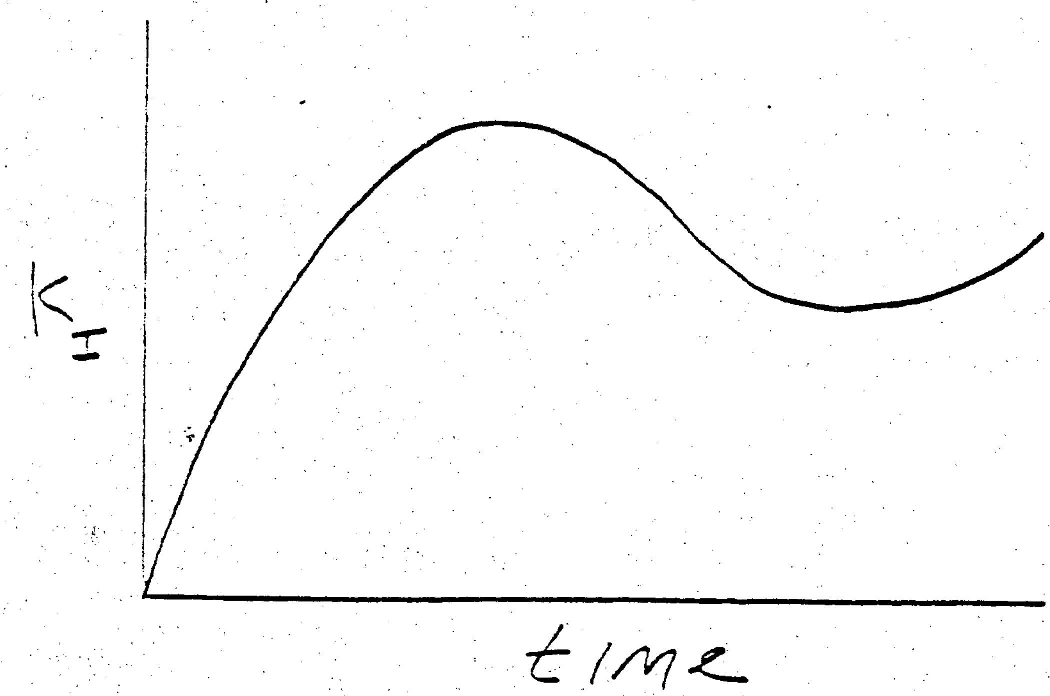
REACTOR VESSEL WITH AXIAL CRACK

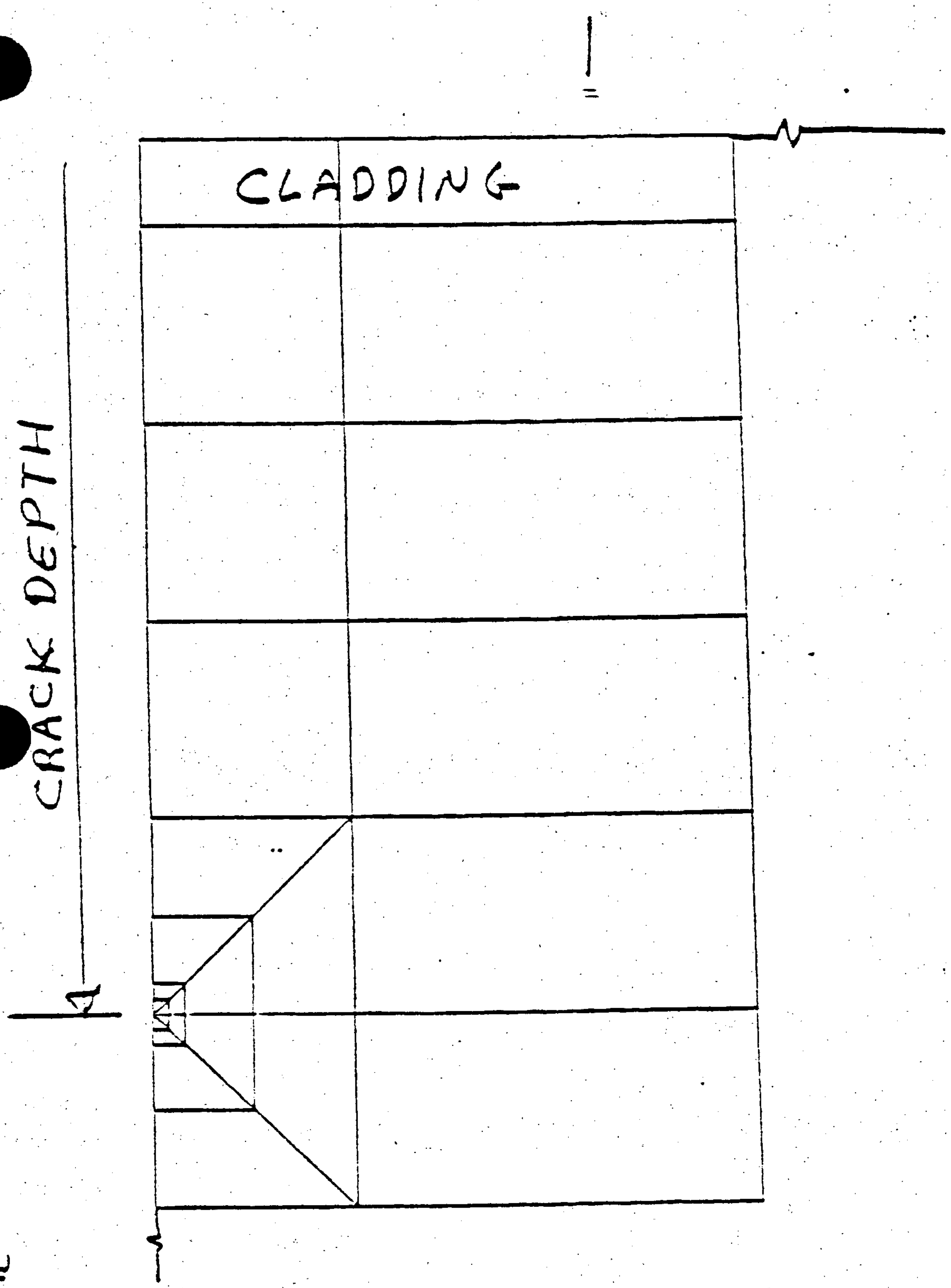


K_{Ia}
K_{Ic}
RT_{NDT}
$(T - RT_{NDT})$ F
-100
-50
0
+50
+100
+150
+200
FRACTURE TOUGHNESS (KSI $\sqrt{IN.}$)
K_{Ic} AND K_{Ia}
0
20
40
60
80
100
120
140
160
180
200
220

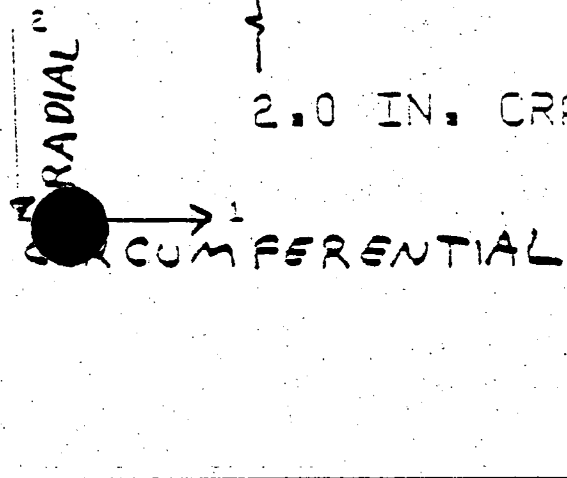
FRACTURE MECHANICS ANALYSIS



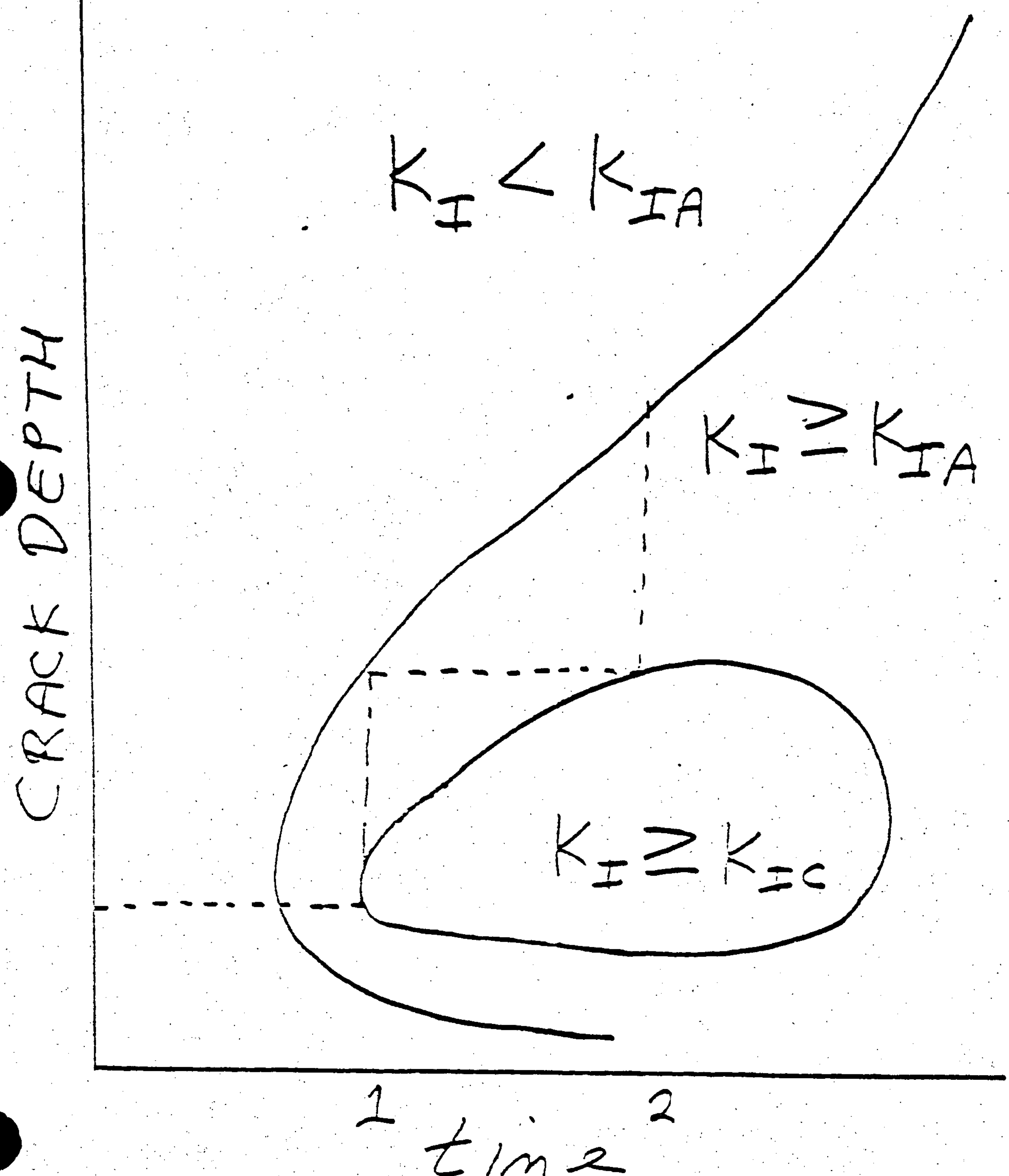




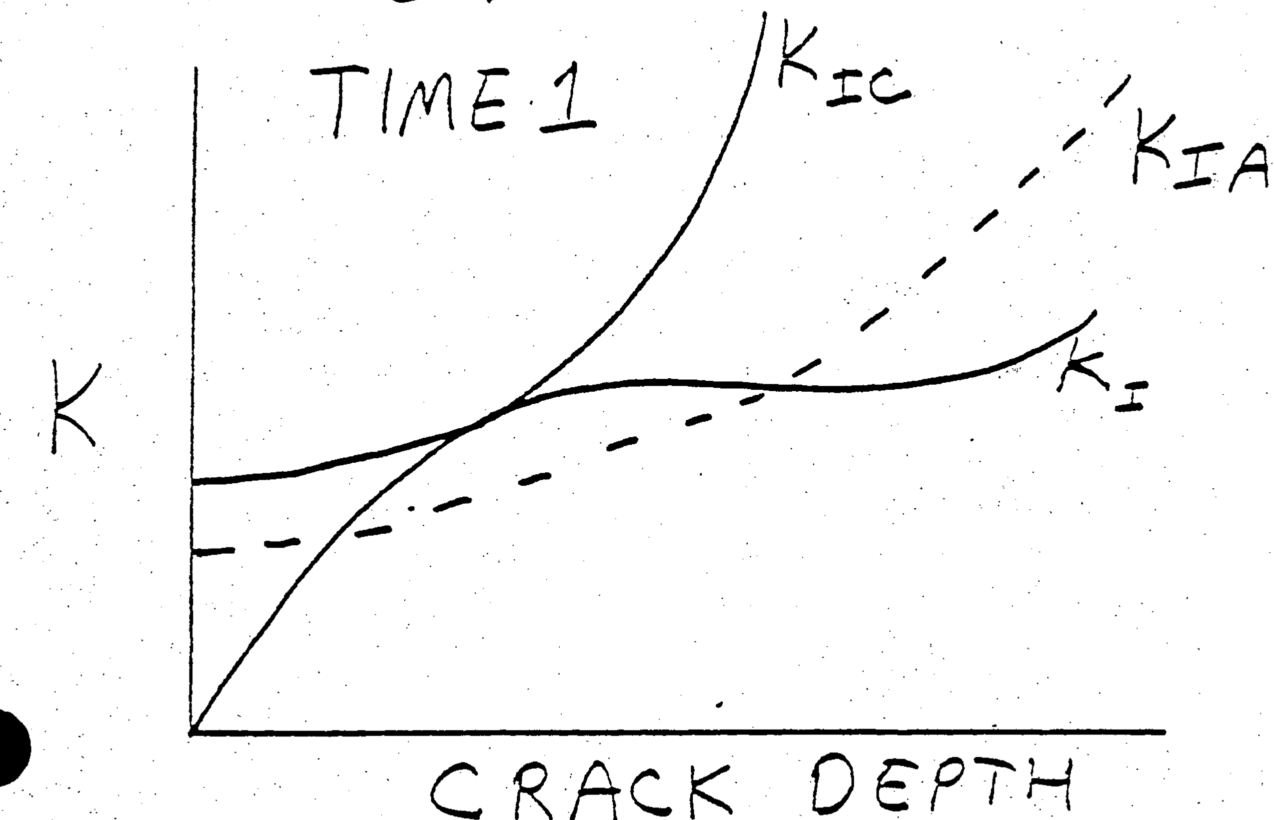
2.0 IN. CRACK AT I.D. OF REACTOR VESSEL

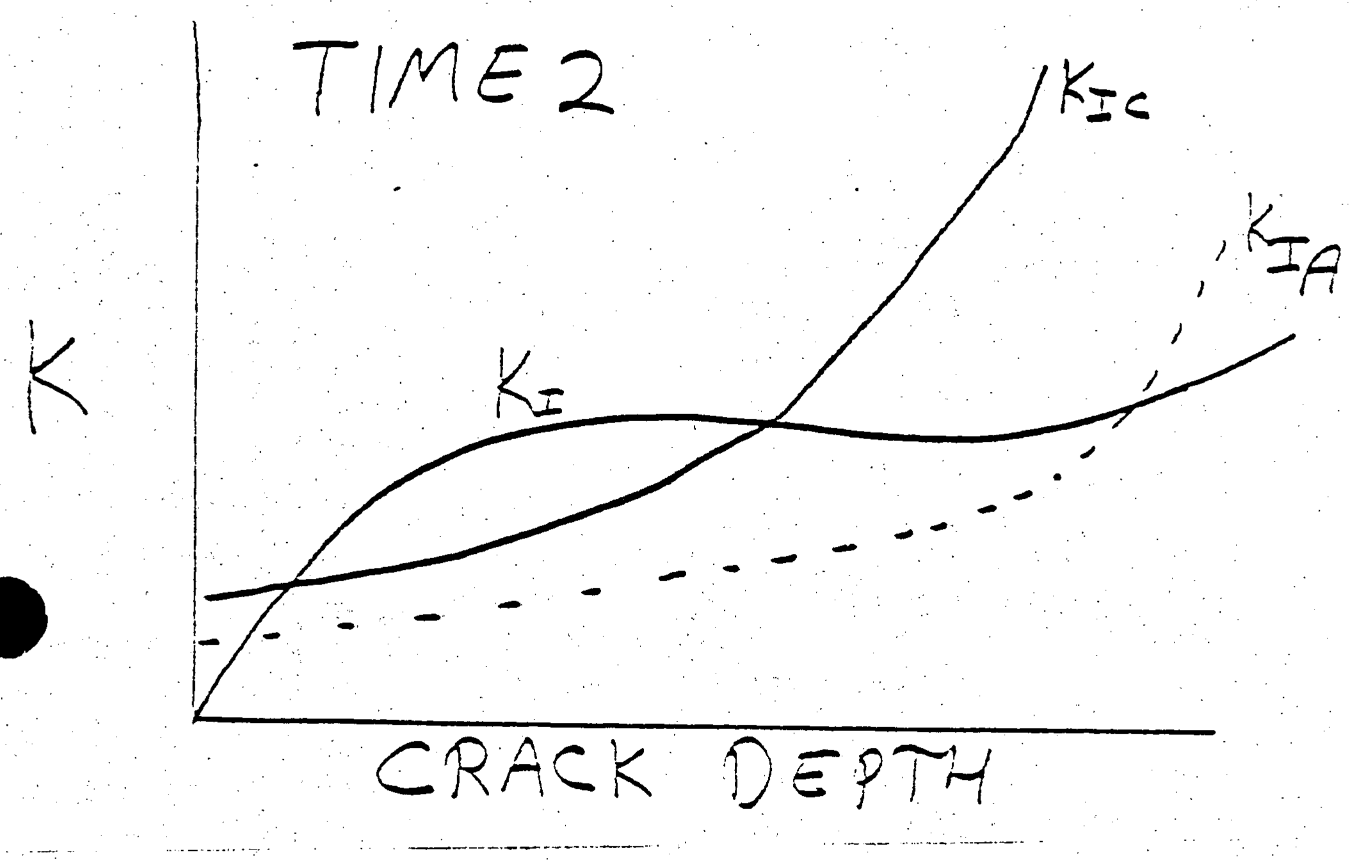


EVALUATION SUMMARY



EVALUATION





CONSERVATIVE ASSUMPTIONS
IN
FRACTURE EVALUATION

1. LONG AXIAL CRACKS

2. WELD WITH HIGHEST NDT

3. AXIAL PEAK FLUENCE

4. NO FLUID HEATING FROM WALL

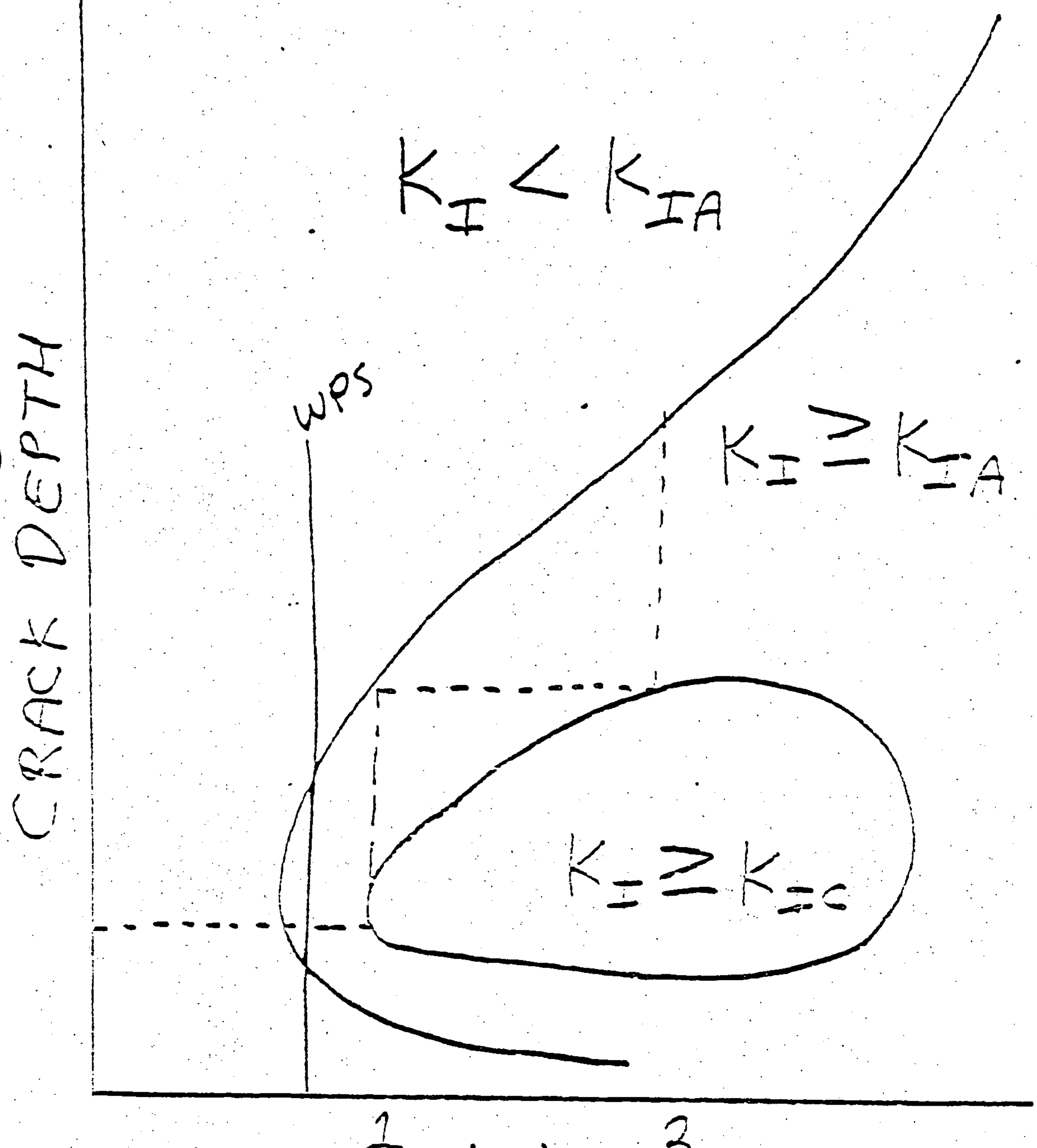
5. FULL PRESSURE IN CRACK

6. DIFFERENTIAL THERMAL EXPANSION OF CLAD

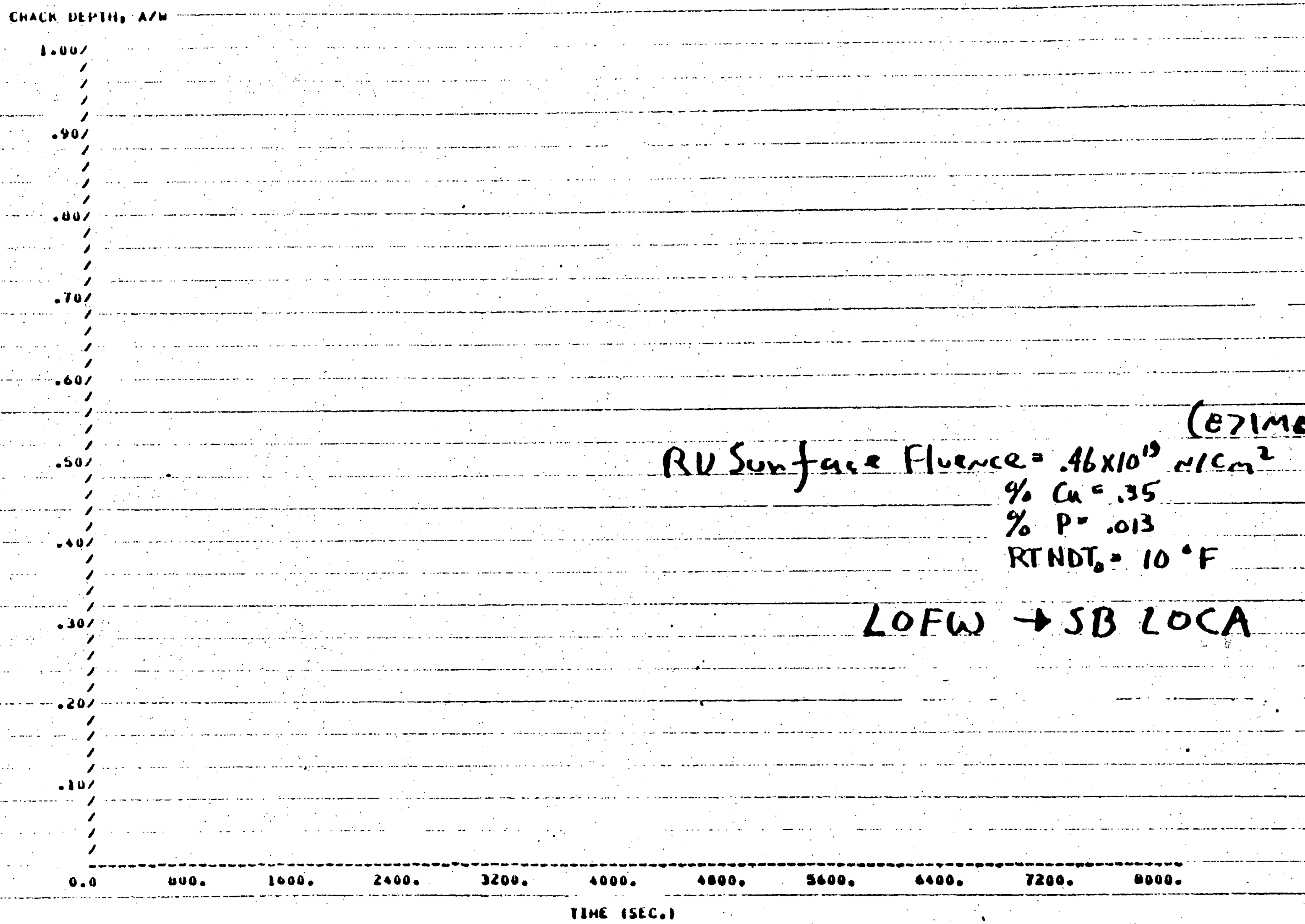
7. REG. GUIDE 1.99 NDT SHIFT

8. LINEARIZED FLUENCE ATTENUATION

EVALUATION SUMMARY

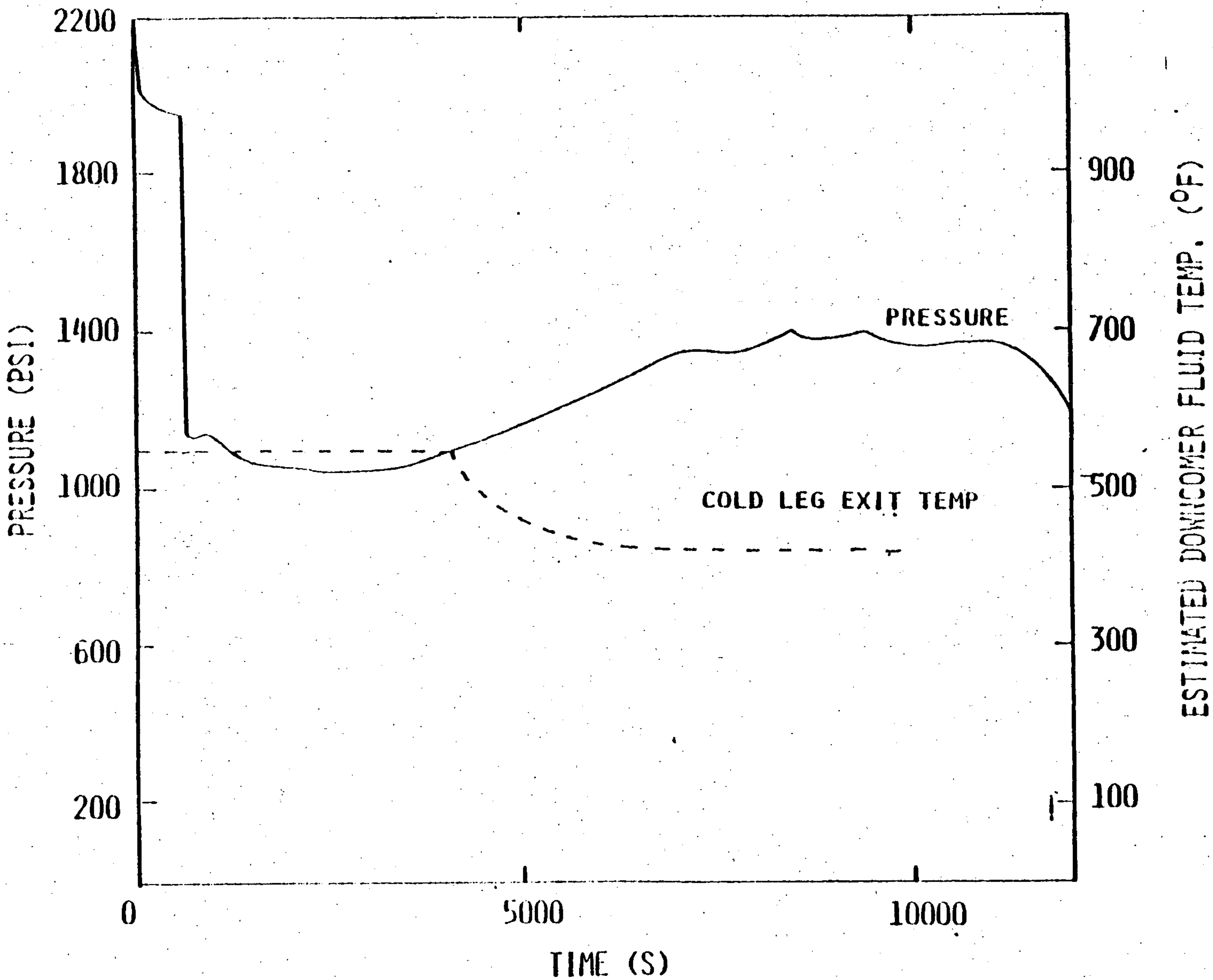


CRITICAL CRACK DEPTH VS. TIME



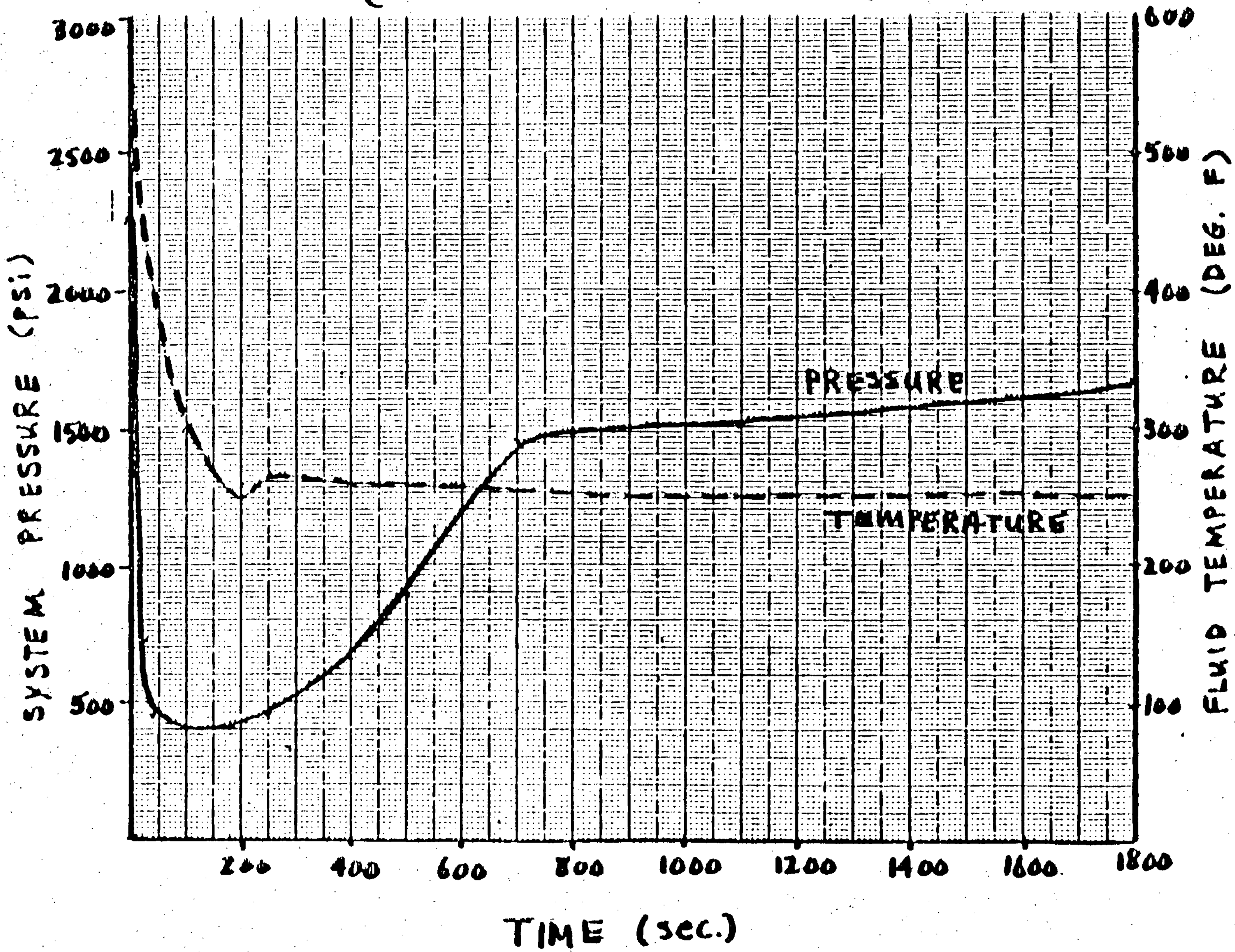
RE[illegible]NCE
OVERCOOLING TRANSIENT

LOFW → SB LOCA

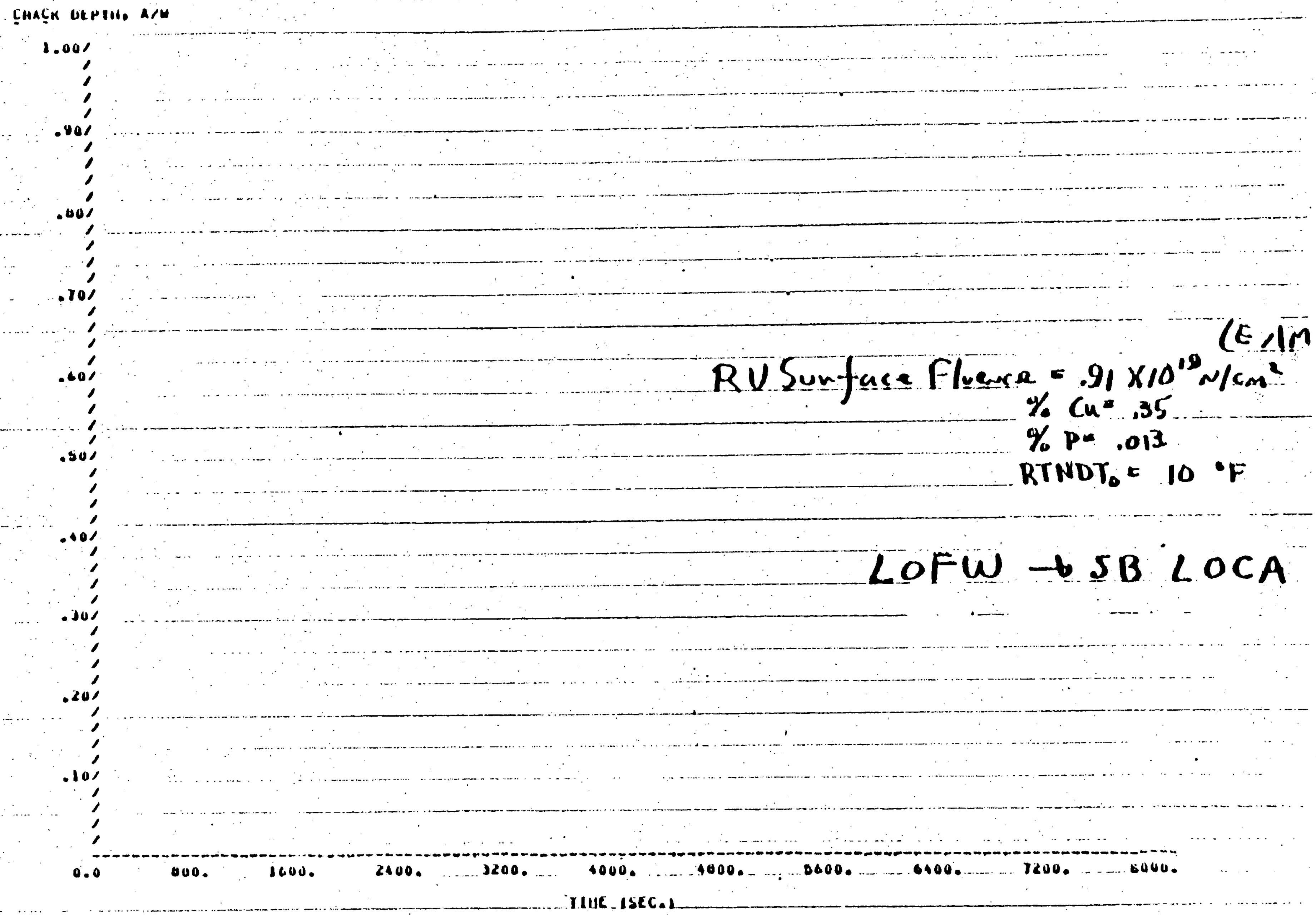


STEAM LINE BREAK TRANSIENT

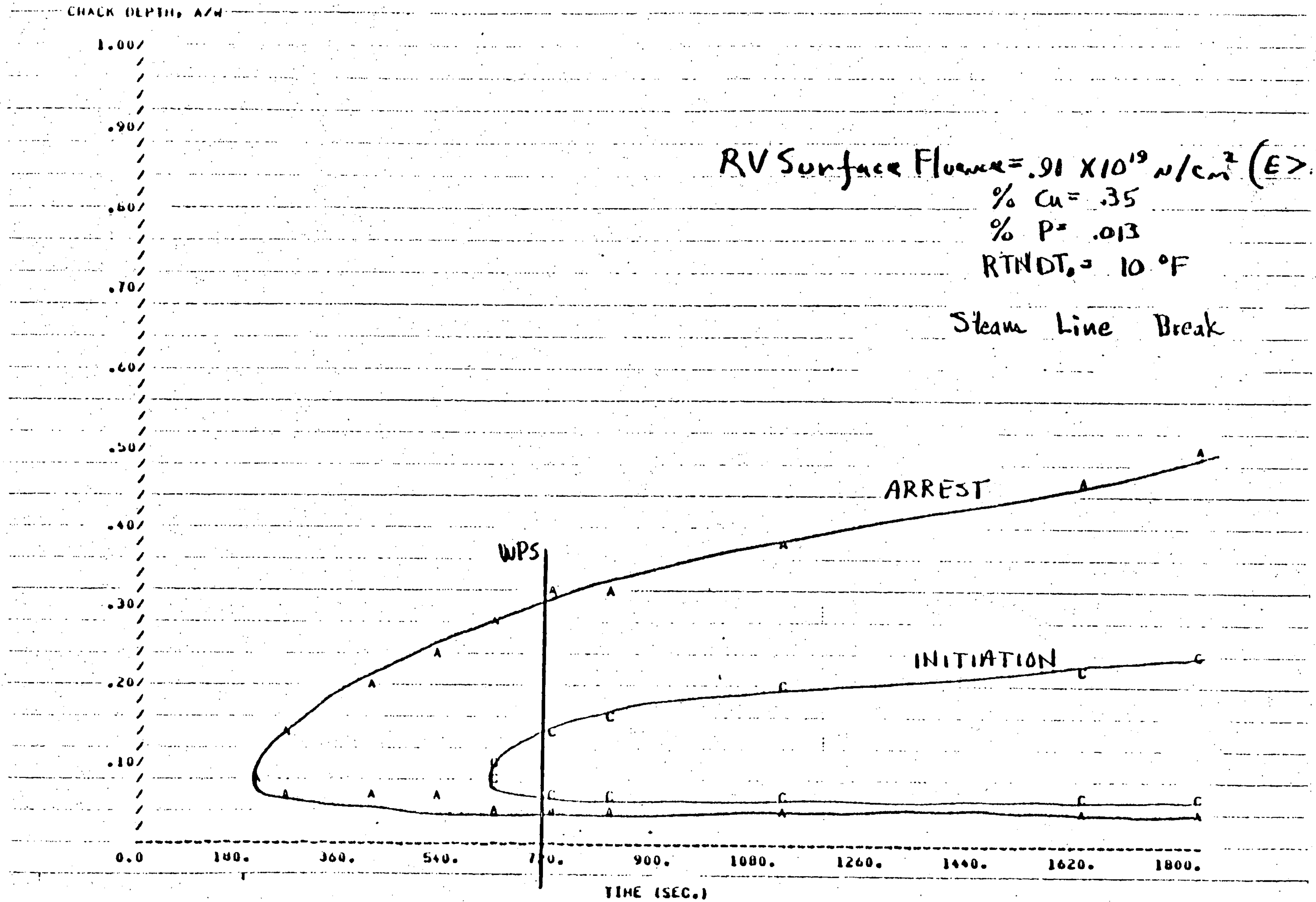
(ZERO POWER AC AVAILABLE)

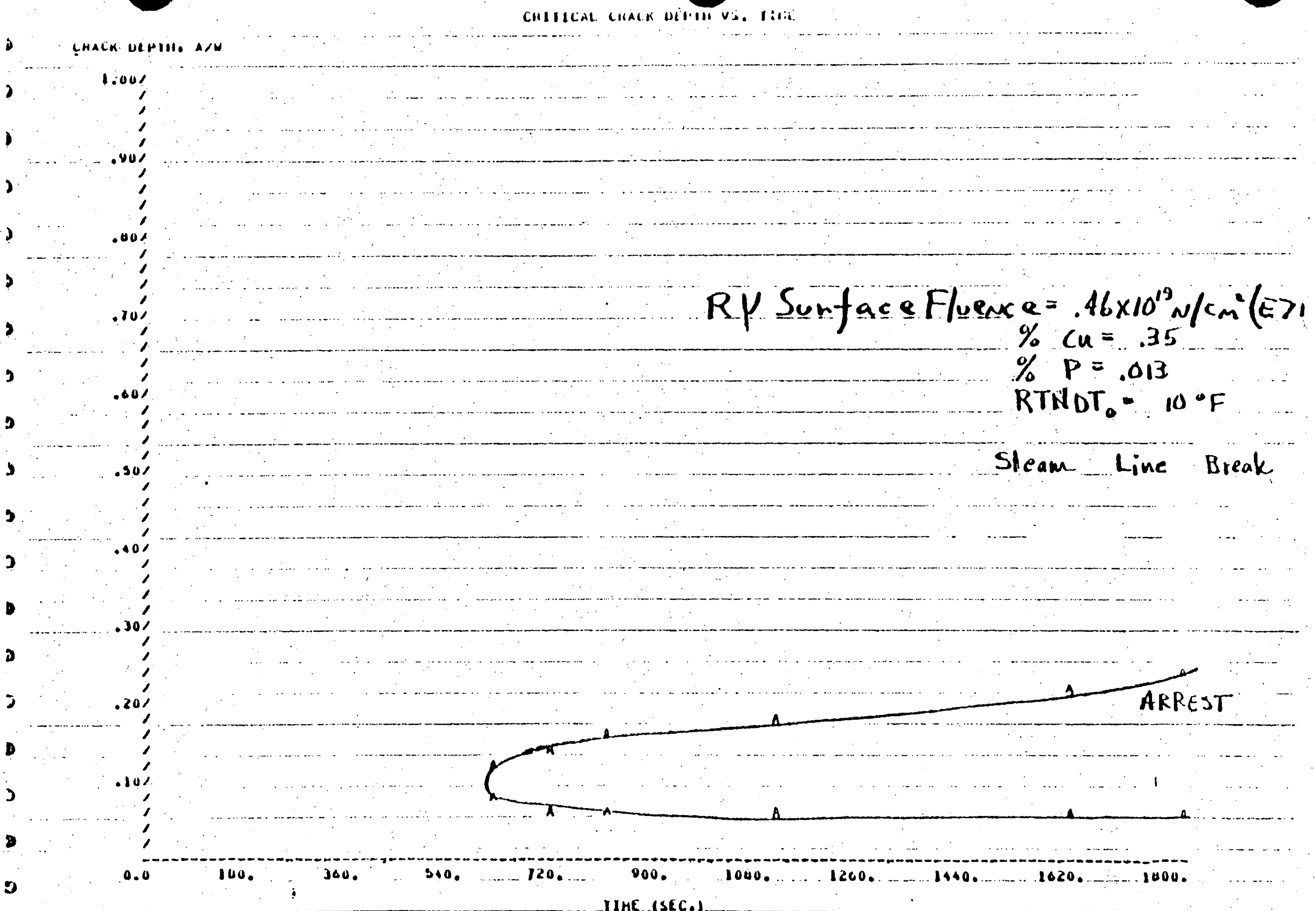


CRITICAL CRACK DE[illegible] VS. TIME

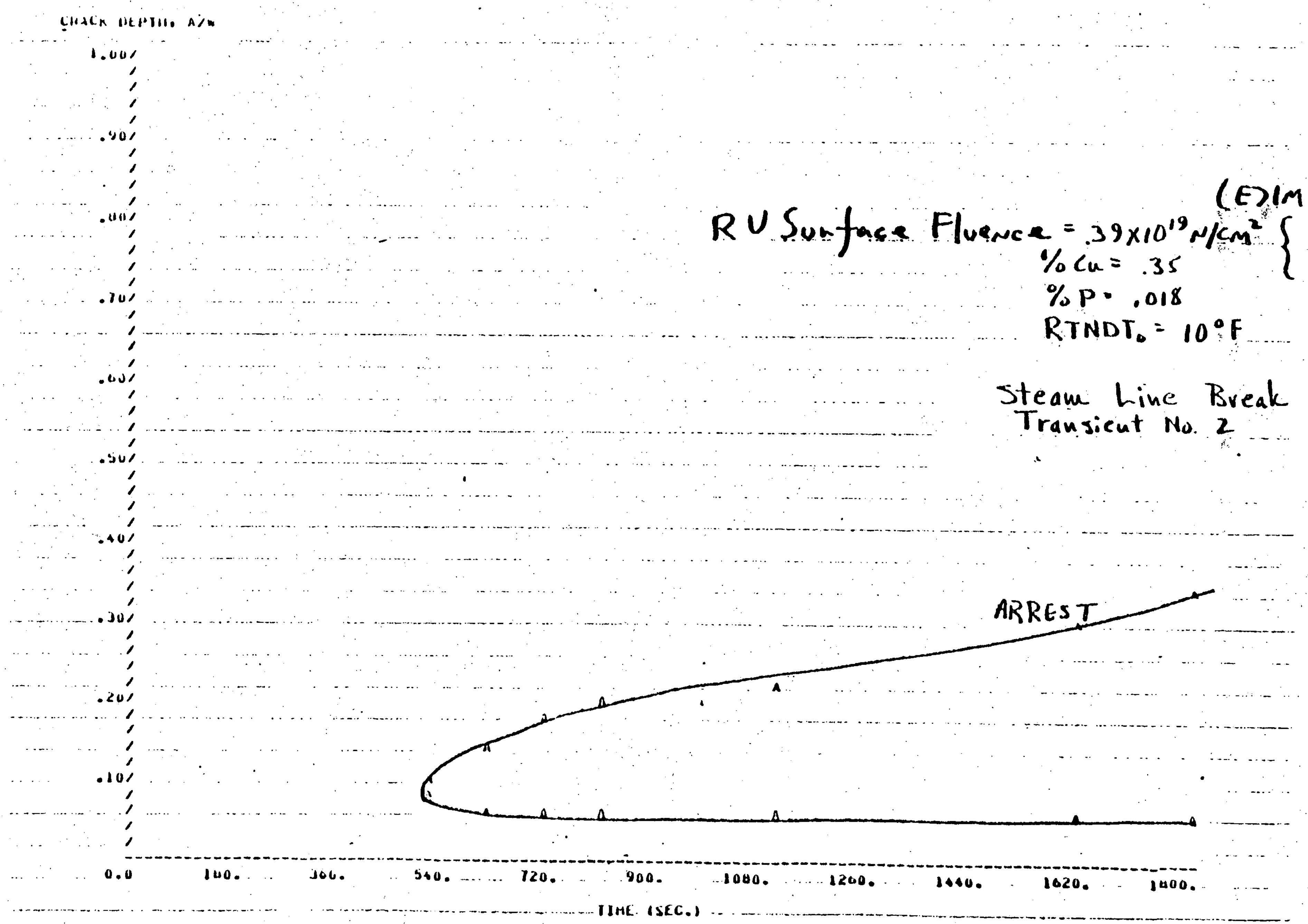


CRITICAL CRACK DEPTH VS. TIME



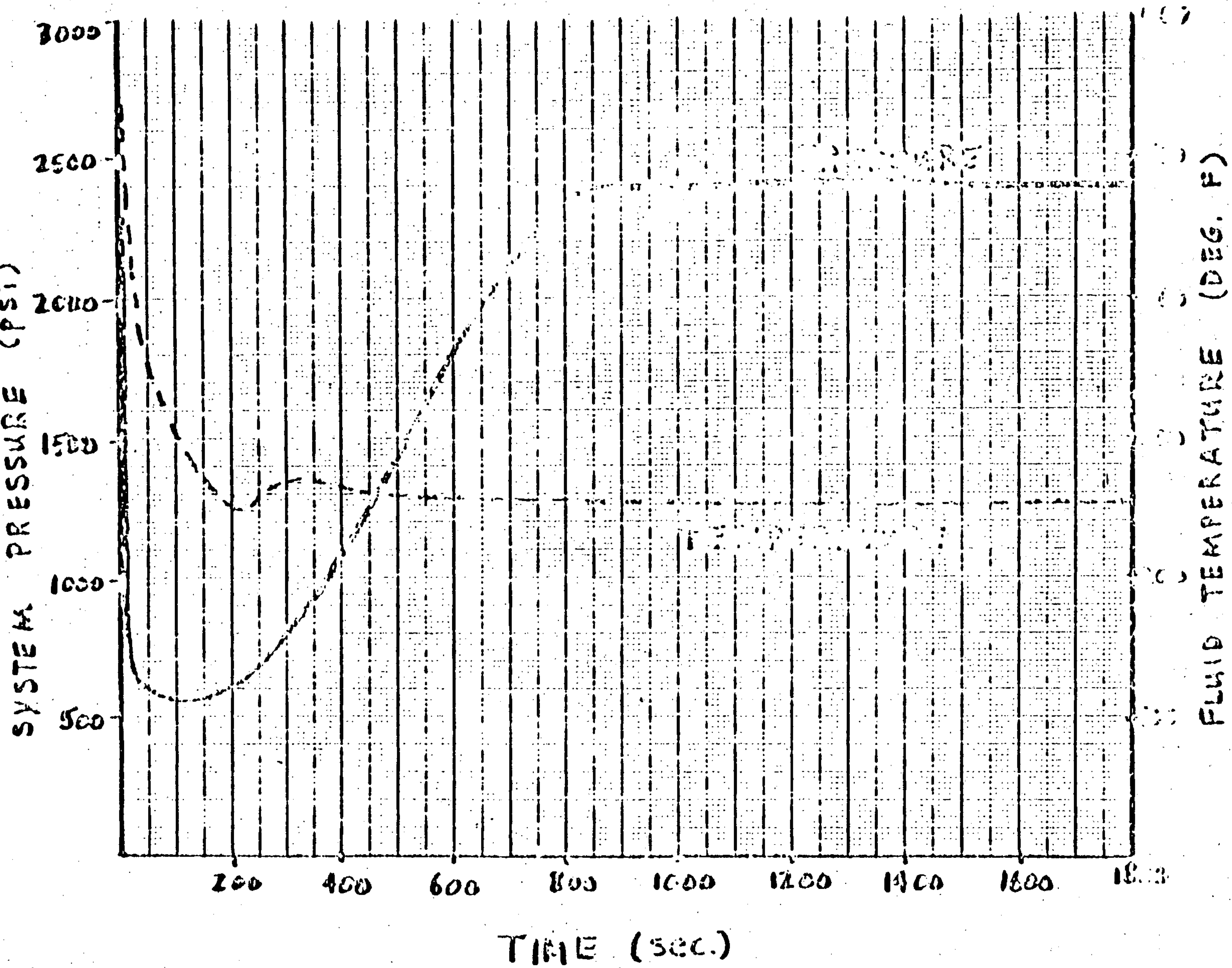

CRITICAL CRACK DEPTH VS. TIME
CRACK DEPTH, A/W
1.00
.90
.80
.70
.60
.50
.40
.30
.20
.10
0.0
180.
360.
540.
720.
900.
1080.
1260.
1440.
1620.
1800.
TIME (SEC.)
RV Surface Fluence = .46x10^19 n/cm² (E>1
% Cu = .35
% P = .013
RTNDT₀ = 10 °F
Steam Line Break
ARREST

CRITICAL CRACK DEPTH VS. TIME

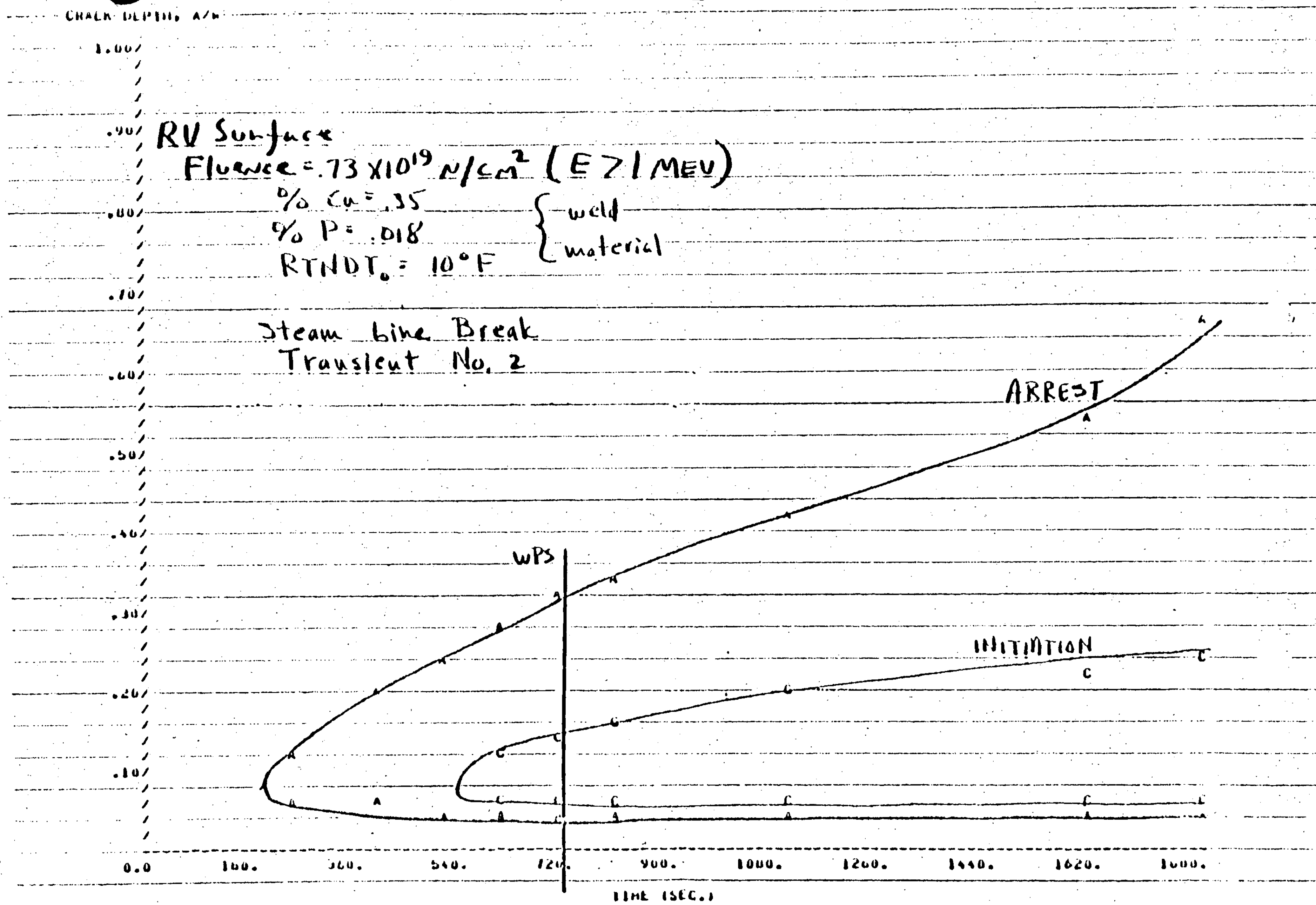


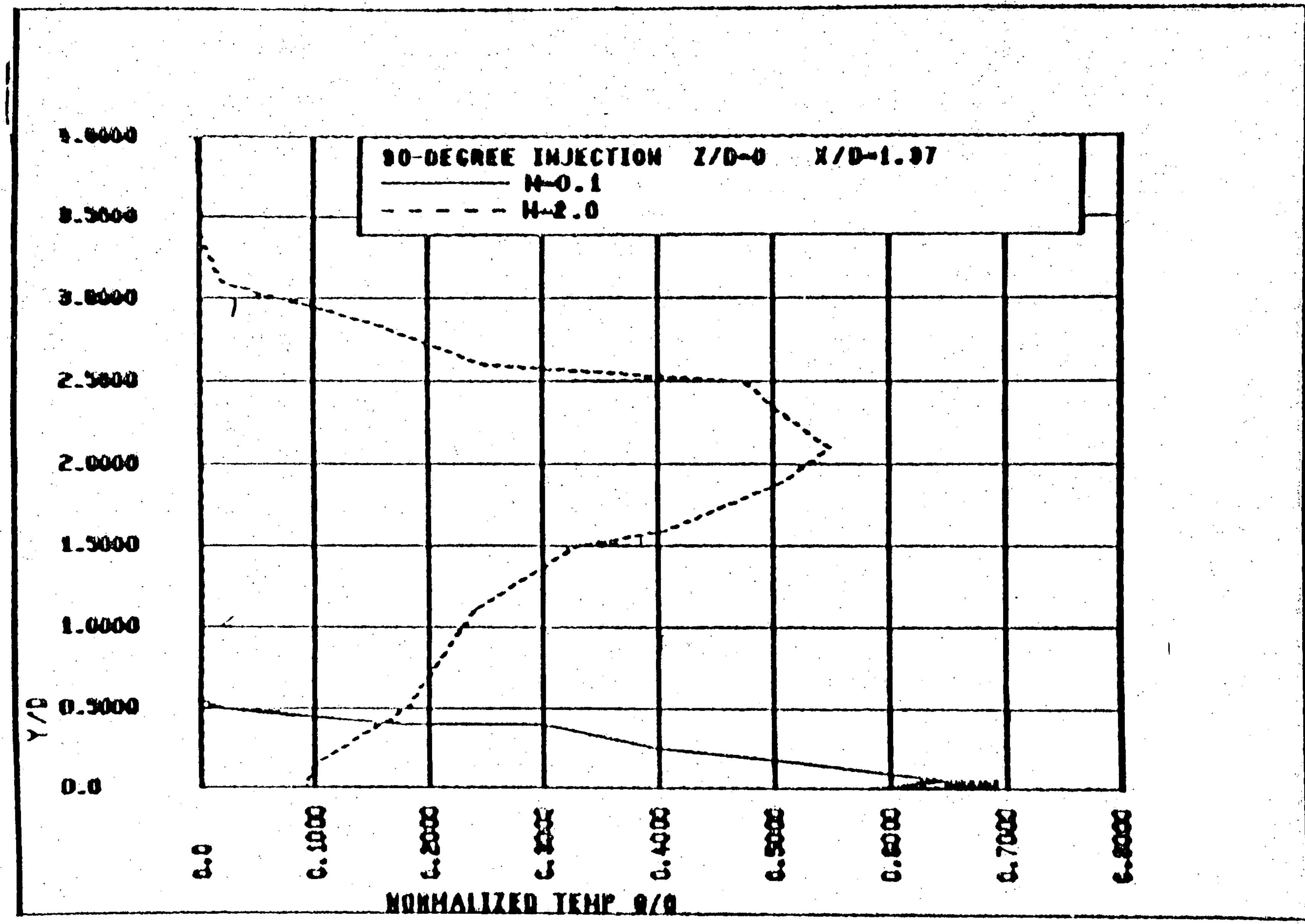
STEAM LINE BREAK TRANSIENT No. 2

(ZERO POWER AC AVAILABLE)



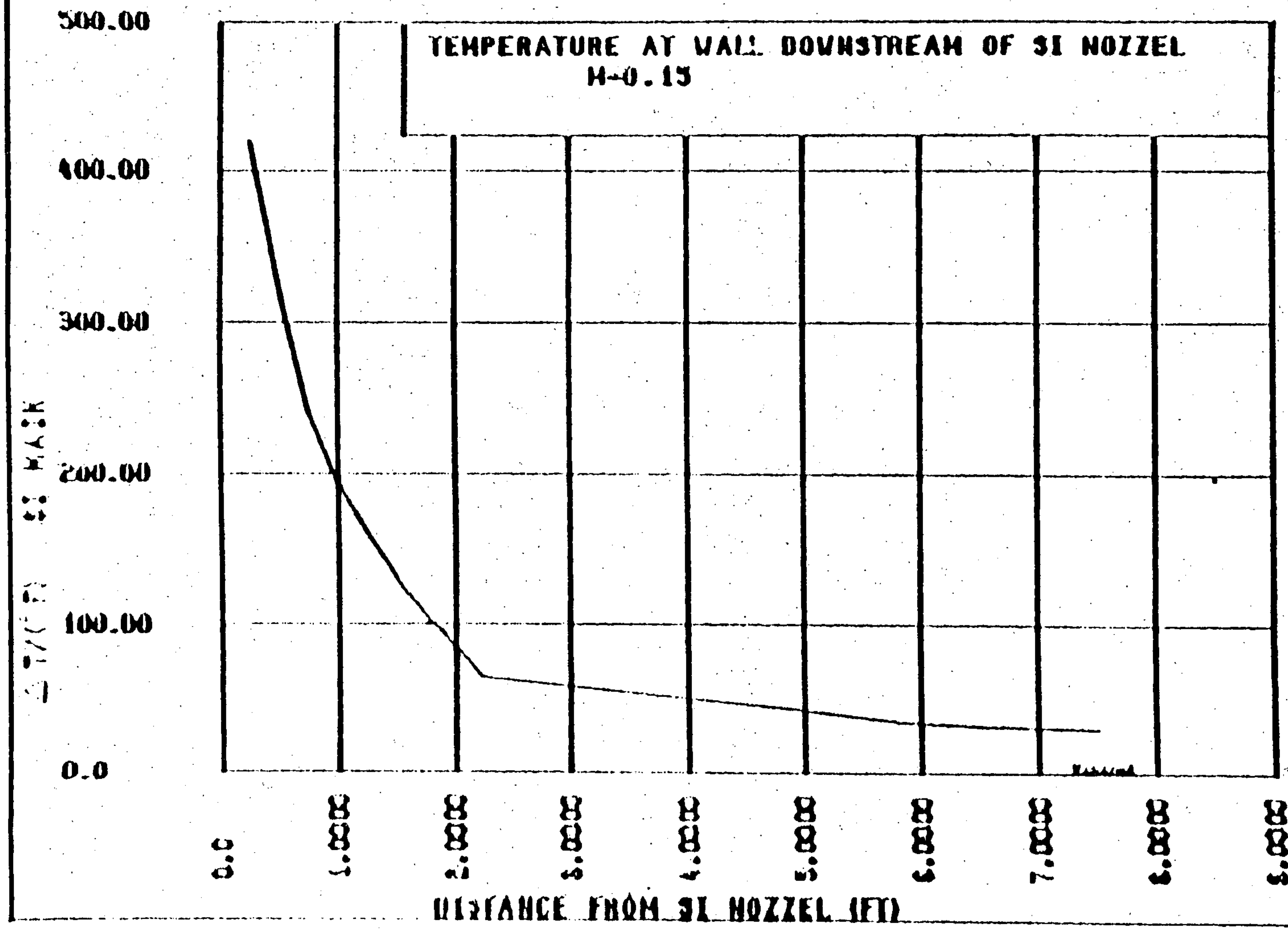
CRITICAL CRACK VS. TIME





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

FRAME 01 F


TEMPERATURE AT WALL DOWNSTREAM OF SI NOZZEL
H-0.15
500.00
400.00
300.00
200.00
100.00
0.0
0.0
DISTANCE FROM SI NOZZEL (FT)

ENCLOSURE 2

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

August 21, 1981

Docket No. 50-289

Mr. Henry D. Hukill, Vice President
and Director - TMI-1
Metropolitan Edison Company
P. O. Box 480
Middletown, Pennsylvania 17057

Dear Mr. Hukill:

SUBJECT: PRESSURIZED THERMAL SHOCK TO REACTOR PRESSURE VESSELS

We have reviewed the PWR Owners' Groups responses of May 15, 1981 and the licensees' responses of May 22, 1981 to our letter dated April 20, 1981 concerning the subject issue. The EPRI work which bears on the issue was included in the licensees' responses. On the basis of our independent review, of the plants where neutron irradiation has significantly reduced the fracture toughness of the reactor pressure vessels (RPVs), all plants could survive a severe overcooling event for at least another year of full power operation. However, we believe that additional action should be taken now to resolve the long-term problems.

This belief is based upon our analyses which indicate that reductions in fracture toughness for some RPVs are approaching levels of concern. It is also based in part on the fact that any proposed corrective action must allow adequate lead time for planning, review, approval, procurement and installation. These conclusions were recently discussed with the PWR Owners Groups on July 28-30, 1981. At those meetings, the Owners Groups reviewed the programs underway at the three PWR vendors which are designed to scope the magnitude and applicability of the generic problem and to be completed by late 1981. The three programs appeared to contain the necessary elements for resolution of the problem on a generic basis and the NRC plans to make full use of the reports due by the end of the year. While the vendors and Owners Groups are to be commended and encouraged in addressing the generic issue, there is also a need for plant-specific information for your plant.

Based on current vessel reference temperature and/or system characteristics, we have identified Ft. Calhoun, Robinson 2, San Onofre 1, Maine Yankee, Oconee 1, Turkey Point 4, Calvert Cliffs 1 and Three Mile Island 1 as plants from which we require additional information at this time.

The staff has used the time-dependent pressure and temperature data from the March 20, 1978 Rancho Seco transient as a starting point for our evaluation of this issue because: (1) it is the most severe overcooling event experienced to data in an operating plant; (2) it is a real, as

Dupe 8109140225

opposed to a postulated, event; and (3) it was severe enough that it could challenge the RPV when combined with physically reasonable values of irradiated fracture toughness and initial crack size. In future reviews the staff plans to use the steam line break accident or other appropriate transient/accident in order to estimate minimum operational times available before plant modifications are required.

Using calculated RPV steel mechanical properties, credible initial flaw sizes, reasonable thermal-hydraulic parameters, and a simplified pressure-temperature transient similar to that observed during the Rancho Seco event, the staff has concluded that all operating plants could safely survive such an event at the present time and for at least an additional year of full power operation. However, because of the required lead times for future actions, the margins in time for long term operation are not large, and there is considerable uncertainty in the probability that similar or more severe transients may occur. It is clear that positive action must be initiated soon for those plants with significantly high transition temperatures. As indicated above, several such plants have been selected by the staff, based on estimates of the current reference temperature for the nil ductility transition (RT_{NDT}) of the RPVs.

The need to initiate further action at this time is emphasized by the recognition that implementation of any proposed fixes or remedial actions must allow for adequate lead time. Because long-term solutions may require a year or more, you should explore short-term approaches as well. Although clear, concise instructions should be provided to operators to reduce the likelihood of repressurization during overcooling transients, the NRC staff believes that reliance on operator actions to prevent repressurization during an overcooling transient will be very difficult to justify as an acceptable long-term solution to the problem.

In accordance with 10 CFR 50.54(f) of the Commission's regulations, you are requested to submit written statements, signed under oath or affirmation, to enable the Commission to determine whether or not your license should be modified, suspended or revoked. Specifically, you are requested to submit the following information to the NRC within 60 days from the date of this letter:

(1) Provide the RT_{NDT} values of the critical welds and plates (or forgings) in your vessel for:
   (a) initial (as-built) conditions and location (e.g., 1/4 T) and
   (b) current conditions (include fluence level) at the RPV inside carbon steel surface.

(2) At what rate is RT_{NDT} increasing for these welds and plate material?

(3) What value of RT_{NDT} for the critical welds and plate material do you consider appropriate as a limit for continued operation?

(4) What is the basis for your proposed limit?

(5) Provide a listing of operator actions which are required for your plant to prevent pressurized thermal shock and to ensure vessel integrity. Include a description of the circumstances in which these operator actions are required to be taken. Included in this summary should be the specific pressure, temperature and level values for: a) high pressure injection (HPI) termination criteria presently used at your facility, b) HPI throttling criteria and instruction presently used at your facility and c) criteria for throttling feedwater presently used at your facility. For each required operator action, give the information available to the operator and the time available for his decision and the required action. State how each required operator action is incorporated in plant operating procedures and in training and requalification training programs.

You are also requested to submit a plan for Three Mile Island, Unit No. 1 to the NRC within 150 days of the date of this letter that will define actions and schedules for resolution of this issue and analyses supporting continued operation. We request that you include consideration and evaluation of the following possible actions:

(1) reduction of further neutron radiation damage at the beltline by replacement of outer fuel assemblies with dummy assemblies or other fuel management changes;

(2) reduction of the thermal shock severity by increasing the ECC water temperature;

(3) recovery of RPV toughness by in-place annealing (include the basis for demonstrating that your plant meets the requirements in 10 CFR 50 Appendix G IV C);

(4) design of a control system to mitigate the initial thermal shock and control repressurization.

For these, as well as for any other alternative approaches, provide implementation schedules that would assure continuance of adequate safety margins.

In the interest of efficient evaluation of your submittal, we request that you include with the above plan, a response to the enclosed request for additional information.

Due to the nature of this review, and the past review effort that has been expended, we consider the above schedules to be reasonable; however, inform us within 30 days if you anticipate conflicts with previous commitments with either submittal and a basis for any delay. We also expect participation by the appropriate PWR Owners Group and NSSS vendors in developing solutions to the problem.

Sincerely,

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

Enclosure:
Request for Additional
Information

cc w/enclosure:
See next page

cc w/enclosure(s):

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Washington, D. C. 20555

* Atomic Safety and Licensing Board Panel
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Enclosure

REQUEST FOR ADDITIONAL INFORMATION

1. Geometry

Geometrical description including design and as-built (when available) dimensions of the core, assemblies, shroud/baffle, thermal shield, downcomer, vessel, cavity, and surrounding shield and/or support structure.

2. Material Description

Region-wise material composition and material isotopic number densities (atoms/barn-cm) for the core, near-core regions and RPV, suitable for neutron transport calculations.

3. Neutron Source

Present and expected EOL:

a) Assembly-wise and core power history (EFPY).
b) Rod-wise and core power history (EFPY) for peripheral assemblies.
c) Core average axial power history distribution.

4. Vessel Fluence

a) Description of available calculations of the vessel fluence including fluence values, locations, and corresponding power histories (EFPY), including 1/4T, 1/2T and 3/4T through the RPV.
b) Description of available capsule-inferred vessel fluences including fluence values, locations, and corresponding power histories (EFPY).

5 . Surveillance Capsules

a) Capsule materials, radial and axial dimensions and locations.
b) Capsule fluence measurements, together with the accumulated power history (EFPY) and a description of the lead factors used to extrapolate the measurements to the peak wall fluence location.

6. Vessel Welds

Axial and azimuthal locations of vessel weld-seams with respect to the core. Overlay of current fluence map with weld locations. Identify the critical welds, vertical and circumferential, and give the weld wire heat numbers. Give weld chemistry for the critical welds. For each weld wire heat number, report the estimated mean copper content, the range and the standard deviation, based on all the reported measurements for that weld wire heat. The welds may be surveillance weldments for your vessel or others, nozzle dropouts that contain a weld, weld metal qualification data, or archive material. In the absence of any information, assume that copper content is at its upper limit (0.35 percent when using R.G. 1.99, Rev. 1) and that the nickel content is high.

7. Systems Analysis

a) Provide a list of 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. Provide the analysis of the most limiting transient or accident with regard to vessel thermal shock considerations. Estimate the frequency of occurrence of this event and provide the basis for this estimate. Discuss the assumptions made regarding reactor operator actions.

b) Identify the computer programs used to calculate the limiting transient or accident. Indicate the degree to which the computer programs used have been verified and any other additional verification required to demonstrate that the computer program models adequately treat the identified important physical models (i.e., ECC mixing, heat transfer, and repressurization).

Dupe 8109220006

PRESSURIZED THERMAL SHOCK OF PRESSURE VESSELS

COMMISSION BRIEFING

T.E. MURLEY
SEPTEMBER 15, 1981

PRESSURIZED THERMAL SHOCK OF PRESSURE VESSELS
COMMISSION BRIEFING

I. SUMMARY OF THE PROBLEM

II. DISCUSSIONS WITH INDUSTRY

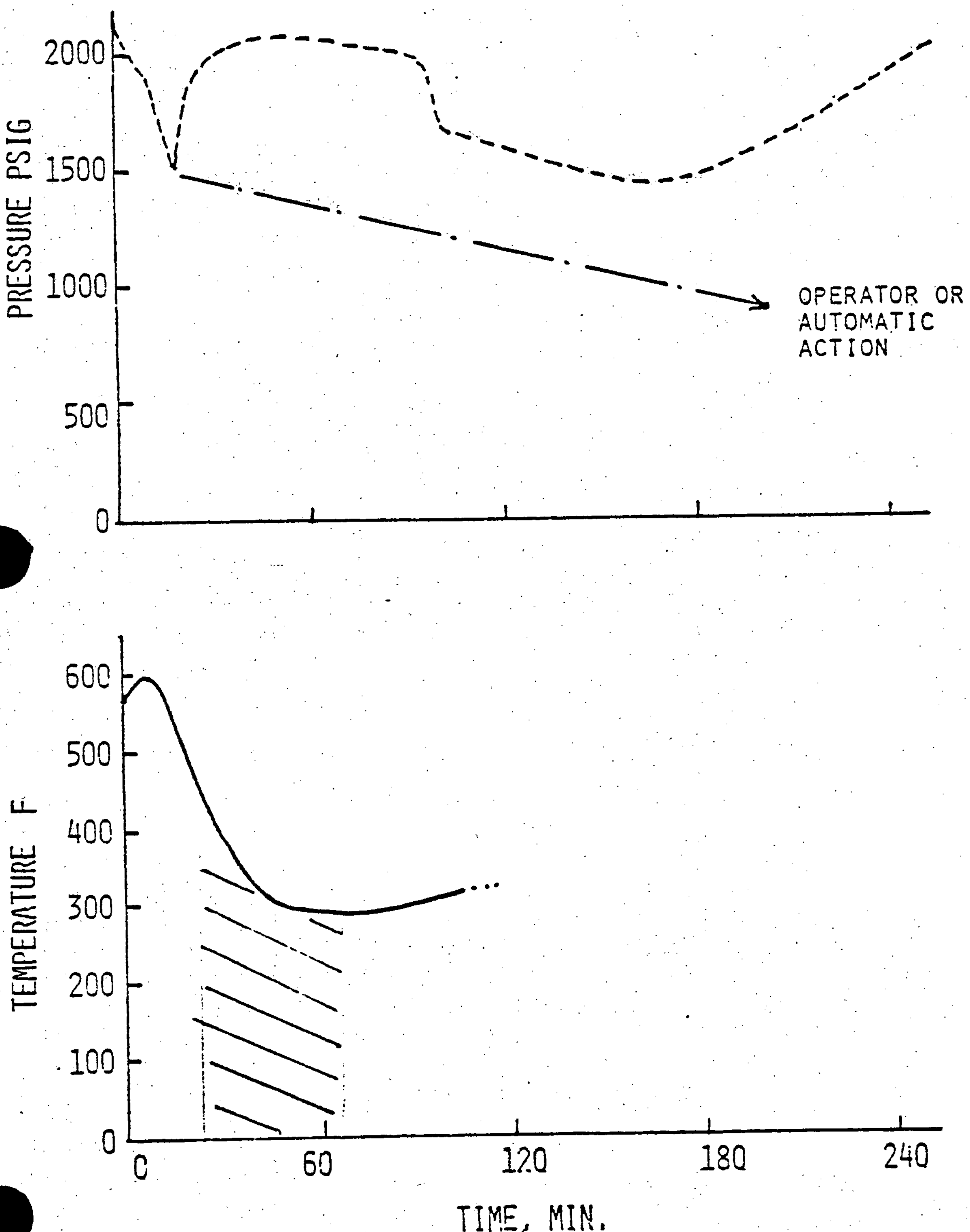
III. RECENT INFORMATION

IV. POSSIBLE REGULATORY LIMITS

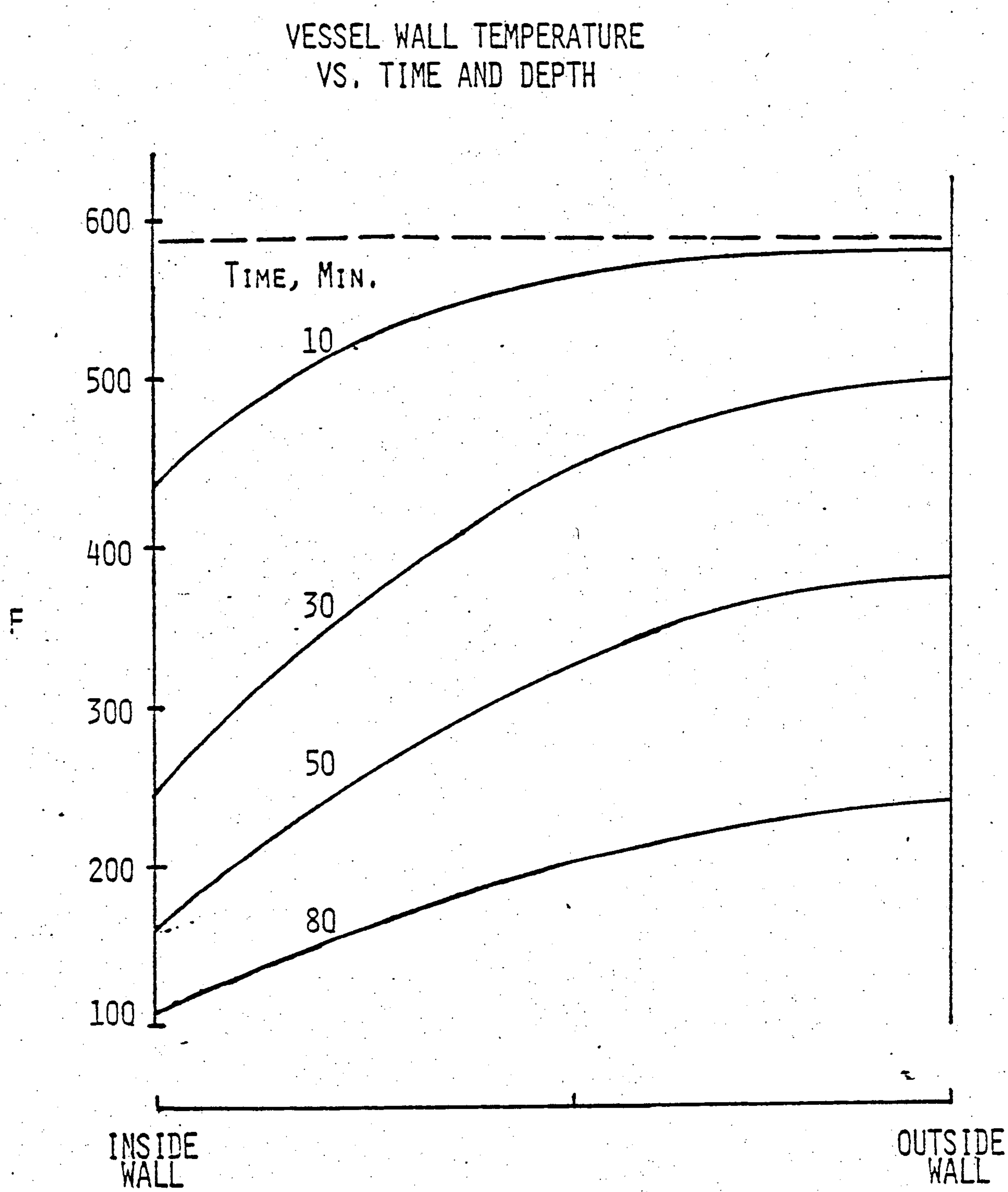
V. STAFF CONCLUSIONS

I. SUMMARY OF THE PROBLEM

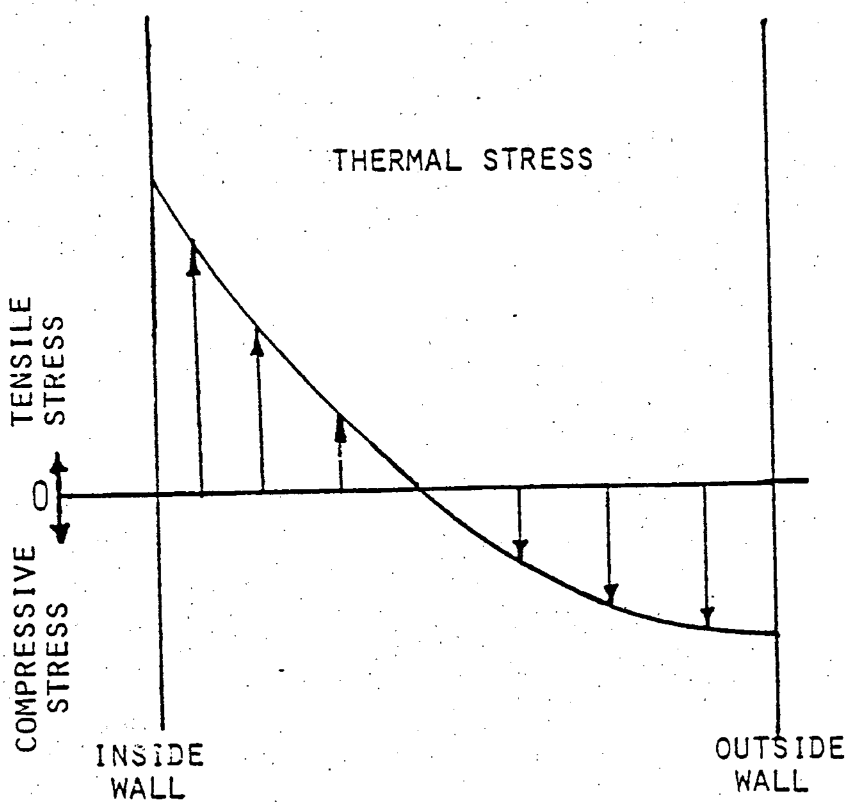
RANCHO SECO 3-20-78 TRANSIENT

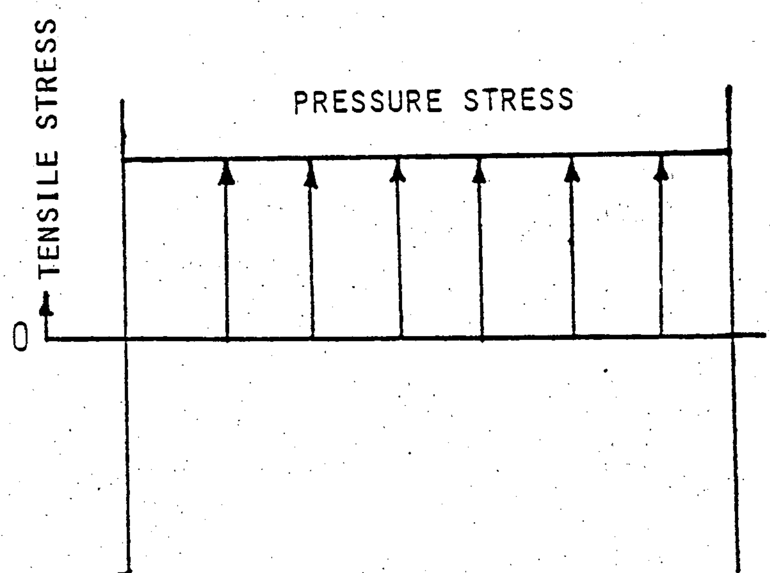


B1

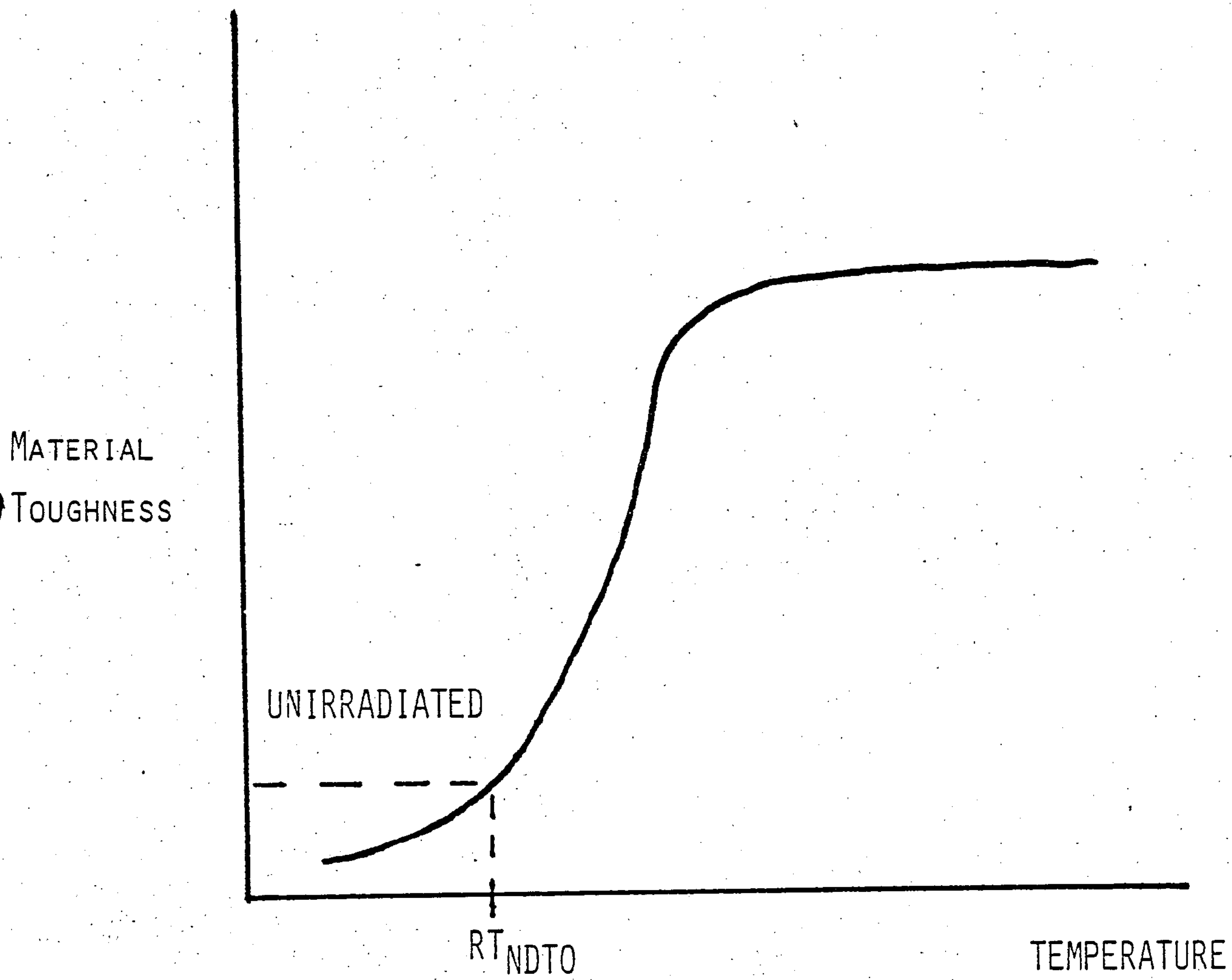

VESSEL WALL TEMPERATURE
VS. TIME AND DEPTH
600
500
400
300
200
100
F
TIME, MIN.
10
30
50
80
INSIDE WALL
OUTSIDE WALL

STRESS DISTRIBUTION

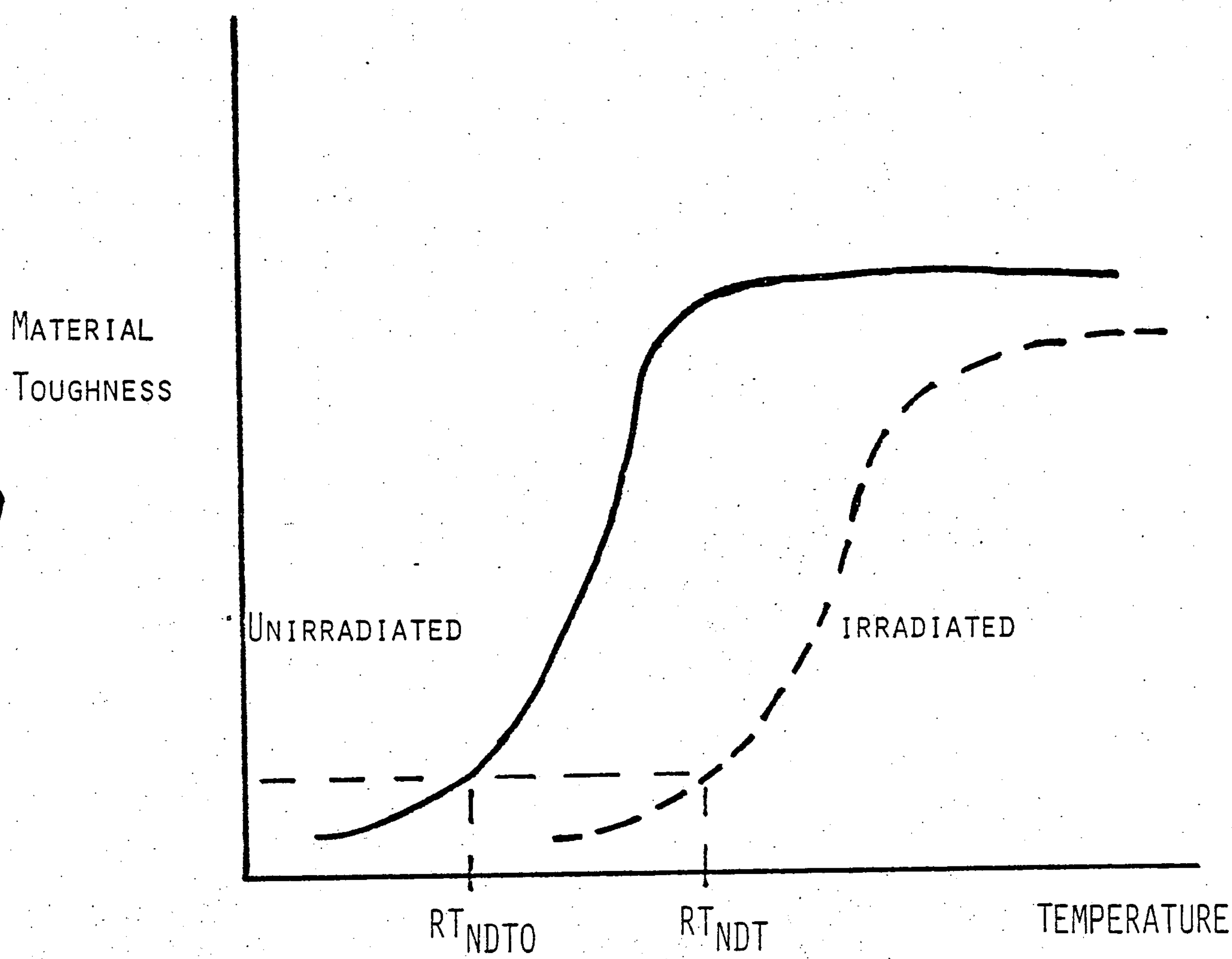


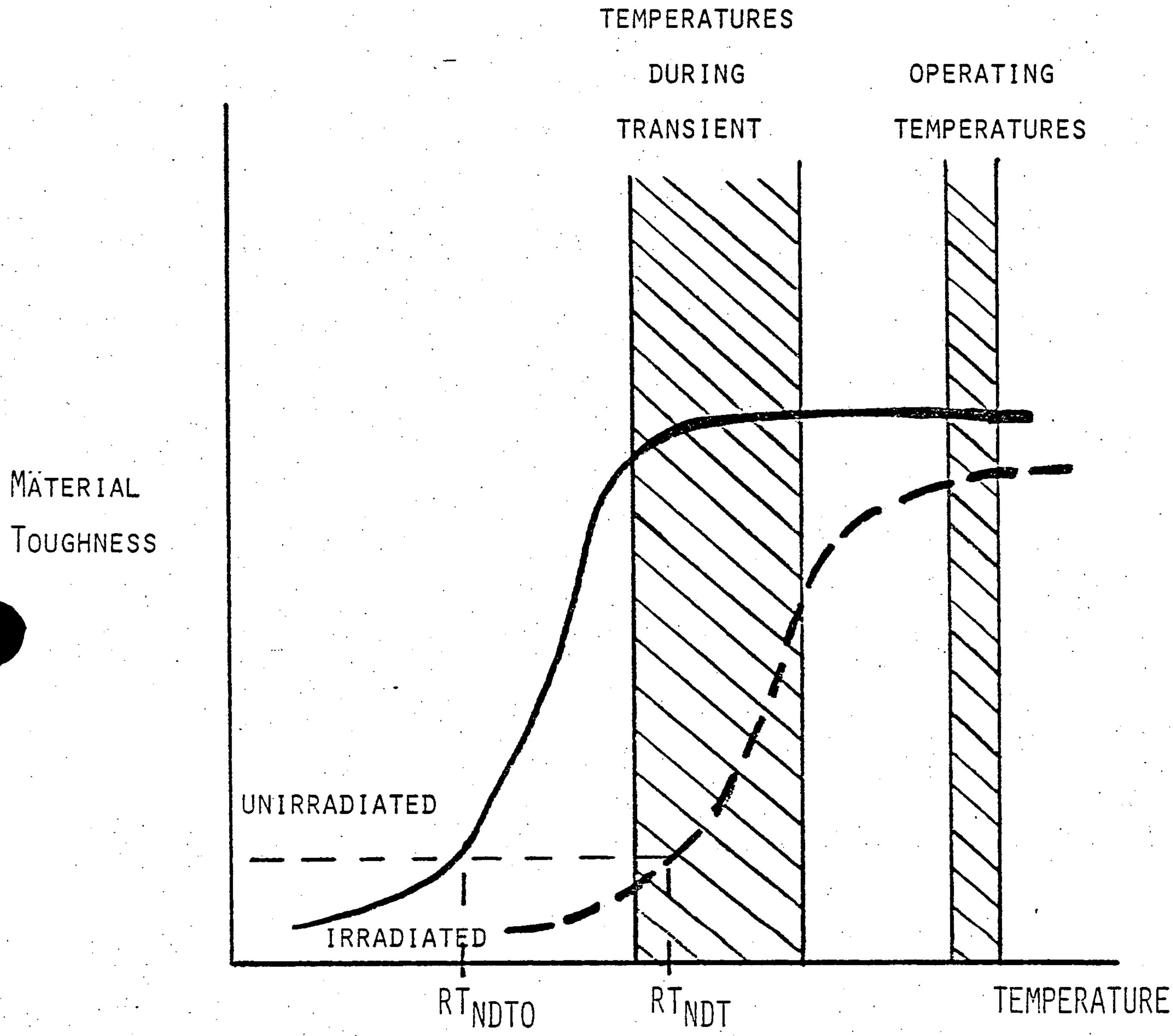


MATERIAL TOUGHNESS VS. TEMPERATURE



NEUTRON-RADIATION-INDUCED
TOUGHNESS CHANGES




TEMPERATURES
DURING
TRANSIENT
OPERATING
TEMPERATURES
MATERIAL
TOUGHNESS
UNIRRADIATED
IRRADIATED
RT_{NDTO}
RT_{NDT}
TEMPERATURE

II. DISCUSSIONS WITH INDUSTRY

Discussions with Industry Groups

-- March 31, 1981, meeting with PWR Regulatory Response Groups

-- April 29, 1981, Progress briefing from PWR Owners Groups

-- May 15, 1981, Reports from Owners Groups

-- May 30, 1981, Reports from PWR licensees

-- July 28-30, 1981, Meetings with Owners Groups

-- August 21, 1981, Letters to 8 licensees

-- Schedule for meetings on technical issues
  -- W Owners Group - September 18, 1981
  -- B&W Owners Group - September 22, 1981
  -- CE Owners Group - October 7, 1981

PRELIMINARY STAFF CONCLUSIONS PRESENTED TO OWNERS GROUPS

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

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

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

Industry Views -- B&W Plants

-- Bounding transient is small break LOCA

-- Preliminary analysis using realistic assumptions showed Oconee-1 vessel could sustain the SBLOCA after 32 EFPY

-- Analysis assumes operator action to prevent repressurization

-- Plant specific analyses planned

Oconee 1	-	December 31, 1981
Rancho Seco	-	March 1, 1982
Other plants	-	later in 1982

Industry Views -- Westinghouse Plants

-- Bounding transient is plant specific (large LOCA, small LOCA or steam line break)

-- All W plants can safely sustain severe thermal shock transient, including repressurization, to beyond January 1983

-- Generic analysis scheduled for completion December 31, 1981

Industry Views -- CE Plants

-- Bounding transient is a steam line break

-- Most limiting plant can safely sustain the most severe overcooling transient for an additional 5 EFPY

-- Analysis assumes no operator action beyond those prescribed by procedures

-- Generic analysis scheduled for completion December 31, 1981

Reference Temperature for the Reactor Vessels
Owned by the Eight Recipients of the Aug. 21, 1981
Letter on Pressurized Thermal Shock

Plant	NSSS Vendor	Vessel Fabricator	Reference Temperature, RT_{NDT}, deg. F.	
			Cir. Welds	Long. Welds
Oconee 1	B&W	B&W	150	170
Three Mile Is. 1	B&W	B&W	180	160
Fort Calhoun	CE	CE	280	280
Maine Yankee	CE	CE	240	240
Calvert Cliffs 1	CE	CE	230	230
Robinson 2	West.	CE	290	290
San Onofre 1	West.	CE	270	270
Turkey Point 4	West.	B&W	290	Forgings

Summary of Information Requested From 8 Licensees

60 Days

-- RT_{NDT} values of the critical welds and plates

-- Rate of increase of RT_{NDT}

-- Appropriate limit on RT_{NDT} for continued operation

-- Operator actions to prevent pressurized thermal shock

150 Days

-- Plan for resolving the issue

-- Evaluation of possible corrective actions

-- Implementation schedules

-- Detailed information for staff evaluations

III. RECENT INFORMATION

Recent Information

-- Oconee-1 in-service inspection showing indications of small cracks under the cladding

-- Overcooling transients at Borssele and San Onofre

-- Sensitivity calculations by staff

- Rancho Seco transient as a benchmark
- Nominal material properties
- Very conservative parameters and material properties

LOVIISA 1 EXPERIENCE

- -- MONITORING PROGRAM SHOWED WELD EMBRITTLEMENT FASTER THAN EXPECTED
- -- RT_{NDT} INCREASED FROM 45°F TO 182°F
- -- 36 OUTER FUEL ELEMENTS WERE REPLACED BY DUMMY ELEMENTS
- -- NEUTRON FLUX AT VESSEL WALL REDUCED TO 1/3 ORIGINAL VALUE
- -- TEMPERATURE OF ECC WATER FROM HPI RAISED FROM 68°F TO 113-131°F

Borselle Event - March 2, 1981

- 400 MWe, 2 loop, 2 S-G, KWU design
- During maintenance, some electrical connections affecting secondary safety valves were interchanged
- During startup (not critical) operator tested power to a pilot valve, causing S-G safety valve to open
- Valve did not reclose (apparent independent failure)
- One S-G boiled dry, operators isolated second S-G, turned on charging pumps to regain pressurizer level
- Primary temperature decreased from 446°F to 284°F in 20 min.
- Primary pressure dropped to 2130 psig, then increased to 2350 psig by charging pumps

SAN ONOFRE FEEDWATER TRANSIENT - SPET. 3, 1981

- POWER SUPPLY FAILURE CAUSED PARTIAL CONTROL SYSTEM FAILURE
- OSCILLATIONS OBSERVED IN FLOW AND LEVEL OF STEAM GENERATORS
- OPERATORS TRIPPED THE REACTOR MANUALLY
- NATURAL CIRCULATION WAS ESTABLISHED IN ACCORDANCE WITH PROCEDURES
- AFTER 8 MINUTES SAFETY INJECTION OCCURRED DUE TO LOW PRESSURE
- OPERATORS IMPLEMENTED THE LOCA PROCEDURE UNTIL IT WAS ESTABLISHED THAT A LOCA HAD NOT OCCURRED
- COLD LEG TEMPERATURE DROPPED FROM $532^{\circ}F$ TO $462^{\circ}F$ IN 10 MINUTES
- NO DATA ON PRESSURE WAS RECORDED

IV. POSSIBLE REGULATORY LIMITS

POSSIBLE CORRECTIVE ACTIONS

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

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

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

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

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

WHAT SHOULD BE THE REGULATORY LIMIT?

WHAT SHOULD BE THE DESIGN BASIS TRANSIENTS?

IS RT_{NDT} THE PROPER CRITERION?

IF SO:

WHAT SHOULD BE THE LIMIT ON RT_{NDT}?

WHAT SHOULD THE EVALUATION MODEL FOR CALCULATING RT_{NDT}?

HOW SHOULD UNCERTAINTIES BE TREATED?

SHOULD ONE DIFFERENTIATE BETWEEN CIRCUMFERENTIAL AND LOGITUDINAL WELDS?

Rate of Increase of RT_{NDT}

Plant	Current RT_{NDT}	Additional EFPY to Reach RT_{NDT} = 300°F
Ft. Calhoun	280°F	2.0 EFPY
Robinson 2	290°F	1.4 EFPY
San Onofre 1	270°F	2.7 EFPY
Turkey Point 4	280°F	2.2 EFPY
Maine Yankee	240°F	7.4 EFPY
Calvert Cliffs 1	230°F	3.4 EFPY
TMI 1	160°F	10.4 EFPY
Oconee 1	170°F	13.3 EFPY

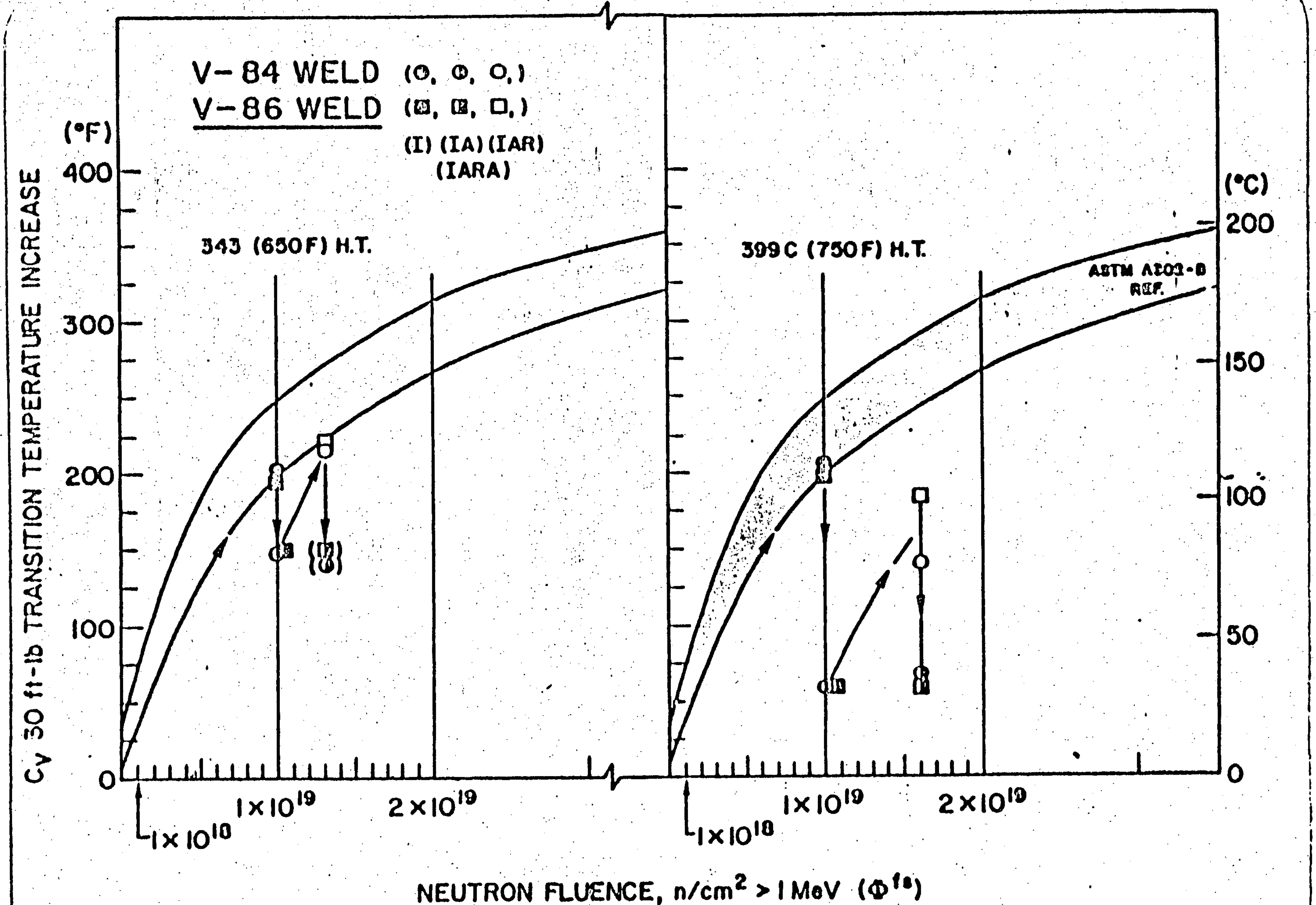
NUMBER OF PLANTS WITH RT_{NDT} ABOVE 200°F

RANGE OF RT_{NDT} (°F)

	200-220	220-240	240-260	260-280	280-300
VESSELS HAVING LONGITUDINAL WELDS	2	2	1	1	2
VESSELS HAVING CIRCUMFERENTIAL WELDS	2	1	1	0	2

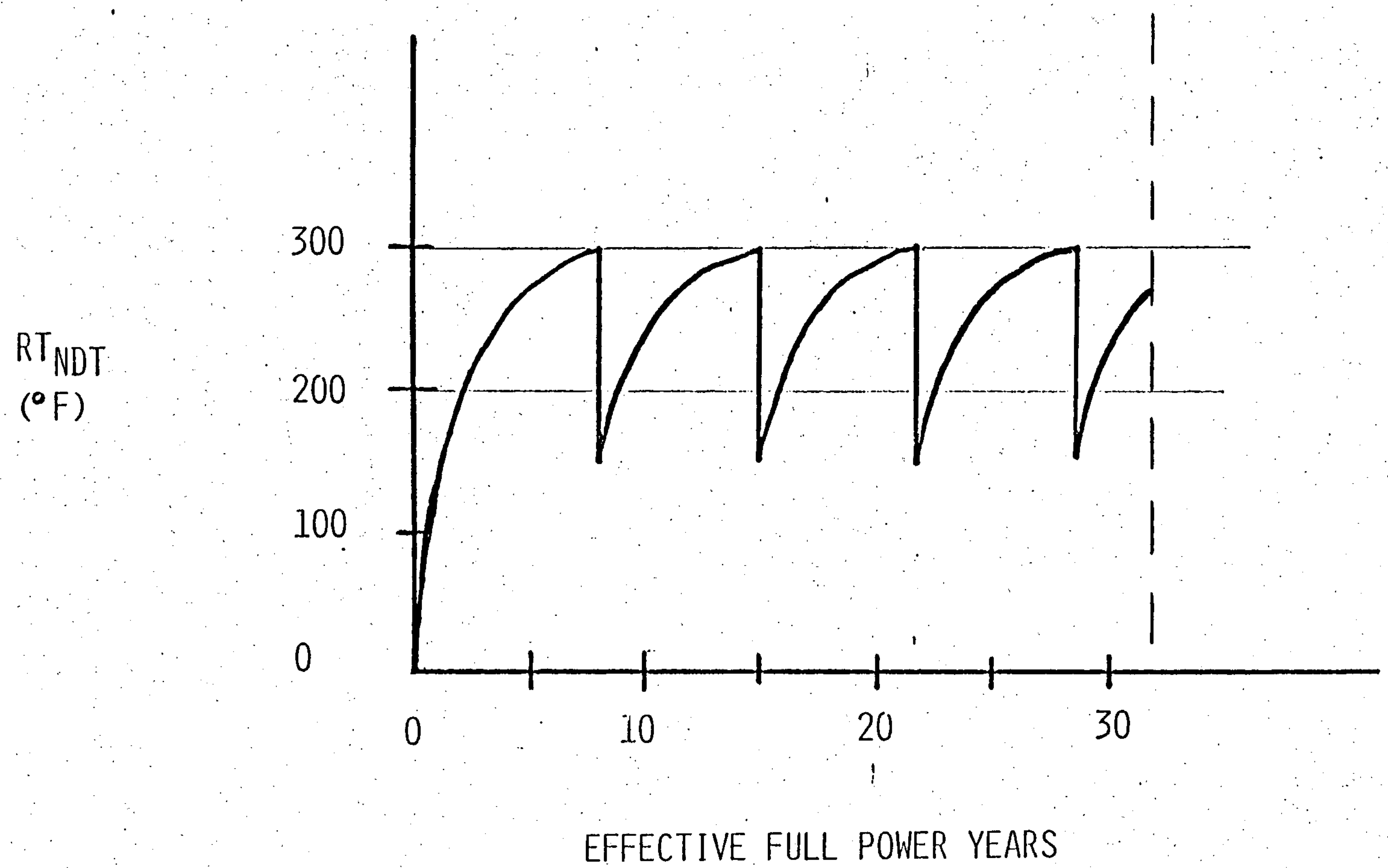
10 CFR 50 APPENDIX G PARAGRAPH IV

"REACTOR VESSELS FOR WHICH THE PREDICTED VALUE OF ADJUSTED REFERENCE TEMPERATURE EXCEEDS 200^{o}F SHALL BE DESIGNED TO PERMIT A THERMAL ANNEALING TREATMENT TO RECOVER MATERIAL TOUGHNESS PROPERTIES OF FERRITIC MATERIALS OF THE REACTOR VESSEL BELTING."



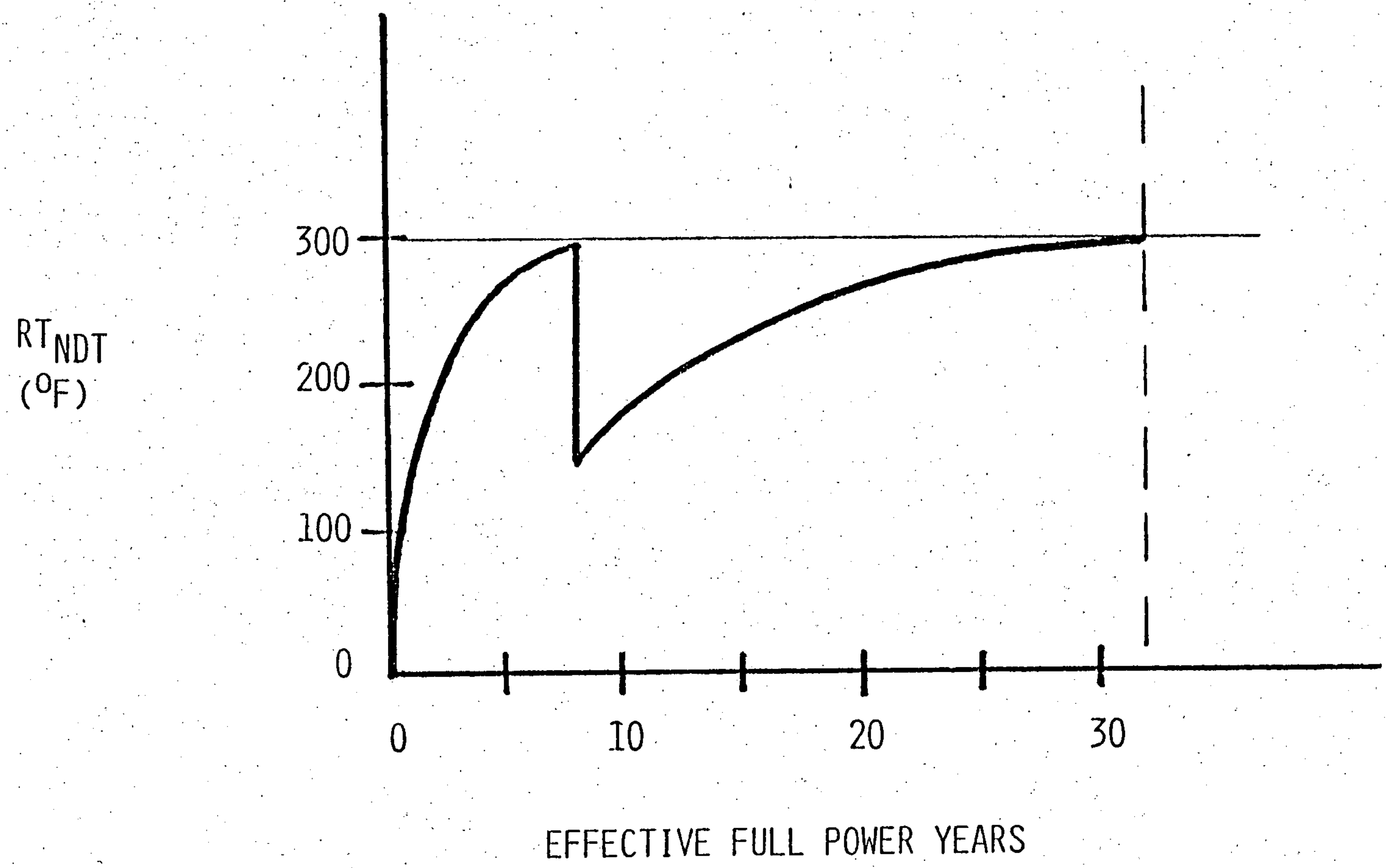
V-84 WELD (O, O, O,)
V-86 WELD (□, □, □,)
(I) (IA) (IAR)
(IARA)
(°F)
(°C)
343 (650F) H.T.
399C (750F) H.T.
ASTM A302-B
REF.
Cv 30 ft-lb TRANSITION TEMPERATURE INCREASE
0
100
200
300
400
0
50
100
150
200
1×10^18
1×10^19
2×10^19
NEUTRON FLUENCE, n/cm² > 1 MeV (Φ^fs)

EXAMPLE OF ANNEALING TO MEET RT_{NDT} LIMITS



EXAMPLE OF ANNEALING TO MEET RT_{NDT} LIMITS

FOLLOWED BY FUEL REMOVAL TO REDUCE FLUX BY FACTOR OF 3



V. STAFF CONCLUSIONS

STAFF CONCLUSIONS

-- PRESSURIZED THERMAL SHOCK IS A SAFETY CONCERN FOR OLDER PWR VESSELS HAVING HIGH COPPER CONTENT MATERIAL.

-- IMMEDIATE CORRECTIVE ACTION IS NOT NECESSARY, BUT SUBSTANTIAL WORK IS NEEDED BY THE INDUSTRY DURING COMING YEAR

-- SOME CORRECTIVE ACTIONS WILL LIKELY BE REQUIRED FOR SOME PLANTS WITHIN A YEAR FROM NOW

-- BASES FOR STAFF CONCLUSIONS THAT NO CORRECTIVE ACTIONS ARE REQUIRED NOW:

A) CHANCES OF A SEVERE OVERCOOLING TRANSIENT IN OLDER PWR PLANT DURING COMING YEAR IS ABOUT 1 IN 100

B) EVEN IF A TRANSIENT AS SEVERE AS RANCHO SECO WERE TO OCCUR, VESSEL FAILURE WOULD NOT BE PREDICTED USING NOMINAL MATERIAL PROPERTIES

C) ANALYSES INCLUDE A NUMBER OF CONSERVATISMS

Staff Actions

-- Independent Analyses of Transients, fluence and fracture mechanics calculations

-- Review licensee analyses

- 60 day replies
- 150 day replies
- Generic reports

-- Generic Resolution by staff (draft NUREG by Summer 1982)

- Possible operator guidelines
- Possible limits on RT_{NDT}

Reference Temperature, RT_{NDT} For PWR Operating Plants At Vessel Inside Surface As of May 1, 1981

<u>Definition</u>:

RT_{NDT} = Reference Temperature Nil-Ductility Transition

RT_{NDT} = Initial RT_{NDT} + ΔRT_{NDT}

Initial RT_{NDT} is determined by following ASME Code procedures, ¶ NB-2331

ΔRT_{NDT} is the temperature shift in the Charpy transition caused by irradiation. Values used in the attached tables are from Reg. Guide 1.99, Rev. 1.

BABCOCK AND WILCOX PLANTS

Plant	Vessel Fabricator	Effective Full Power Years	Fluence Per EFPY		Percent Copper		Reference Temperature, RTNDT Deg. Fahr.		
			Cir. Welds $\times 10^{18}$ n/cm^2	Long. Welds $\times 10^{18}$ n/cm^2	Cir. Welds	Long. Welds	Plate	Cir. Welds	Long. Welds
Oconee 1	B&W	4.89	0.54	0.46	0.26	0.31		150	170
Three Mile Is. 1	B&W	3.45	0.60	0.60	0.35	0.31		180	160
Rancho Seco	B&W	3.25	0.66	0.58	0.31	0.31		160	150
Arkansas 1	B&W	3.85	0.61	0.45	0.31	0.31		170	150
Crystal River 3	B&W	2.18	0.58	0.55	0.35	0.31		150	130
Oconee 2	B&W	4.26	0.61	Forgings	0.35	Forgings		200	Forgings
Oconee 3	B&W	4.3	0.61	Forgings	0.24	Forgings		140	Forgings
Davis Besse	B&W	1.26	0.66	Forgings	0.24	Forgings		60	Forgings

COMBUSTION ENGINEERING PLANTS

Plant	Vessel Fabricator	Effective Full Power Years	Fluence Per EFPY Cir. Welds $x10^{18} n/cm^2$	Fluence Per EFPY Long. Welds $x10^{18} n/cm^2$	Percent Copper Cir. Welds	Percent Copper Long. Welds	Reference Temperature, RT_{NDT} Deg. Fahr. Plate	Reference Temperature, RT_{NDT} Deg. Fahr. Cir. Welds	Reference Temperature, RT_{NDT} Deg. Fahr. Long. Welds
Fort Calhoun	CE	4.76	1.72	1.72	0.35	0.35		280	280
Maine Yankee	CE	5.50	0.85	0.85	0.36	0.36		240	240
Calvert Cliffs 1	CE	4.10	1.47	1.47	0.30	0.30		230	230
Calvert Cliffs 2	CE	3.03	1.47	1.47	0.23	0.30		150	200
Palisades	CE	3.92	1.16	1.16	0.25	0.25		190	190
Millstone 2	CE	3.37	0.56	0.56	0.37	0.37		160	160
St. Lucie	CE	3.31	0.63	0.63	0.31	0.30		140	140
Arkansas 2	CE	0.69							

WESTINGHOUSE PLANTS

Plant	Vessel Fabricator	Effective Full Power Years	Fluence Per EFPY		Percent Copper		Reference Temperature, RT_{NDT} Deg. Fahr.		
			Cir. Welds $x10^{18} n/cm^2$	Long. Welds $x10^{18} n/cm^2$	Cir. Welds	Long. Welds	Plate	Cir. Welds	Long. Welds
Robinson 2	CE	6.80	1.72	1.59	0.34	0.34		290	290
San Onofre	CE	8.82	1.74	1.74	0.19	0.19		270	270
Haddam Neck	CE	10.42	1.31	1.09	0.22	0.22		230	220
Point Beach 1	B&W	7.69	1.24	0.91	0.24	0.24		220	210
Surry 2	B&W/RDM	4.34	1.56	0.34	0.19	0.31		140	110
Zion 2	B&W	4.16	0.63	0.20	0.26	0.35		120	100
Zion 1	B&W	4.32	0.63	0.20	0.35	0.31		170	90
Farley	CE	1.86	1.69	0.38	0.24	0.27		120	70
Surry 1	B&W	4.51	1.56	0.34	0.25	0.18		190	60
Beaver Valley	CE	1.33	1.69	0.25	0.37	0.36		160	60
Cook 1	CE	4.15	0.63	0.34	0.40	0.13		170	50
Cook 2	CB&I	2.00							
Sequoyah 1	RDM	0.20							
Yankee Rowe	B&W	14.14	0.78 Plate		0.20 Plate		200	No weld data	
Indian Point 2	CE	3.97	0.50 Plate		0.25 Plate		140	No weld data	

WESTINGHOUSE PLANTS

Plant	Vessel Fabricator	Effective Full Power Years	Fluence Per EFPY Cir. Welds $x10^{18} n/cm^2$	Fluence Per EFPY Long. Welds $x10^{18} n/cm^2$	Percent Copper Cir. Welds	Percent Copper Long. Welds	Reference Temperature, RTNDT Deg. Fahr. Plate	Reference Temperature, RTNDT Deg. Fahr. Cir. Welds	Reference Temperature, RTNDT Deg. Fahr. Long. Welds
Indian Point 3	CE	2.67	0.56 Plate		0.24 Plate		120	Plate is controlling	
Salem 1	CE	1.76	1.16 Plate		0.24 Plate		90	Plate is controlling	
Trojan	CB&I	2.57	0.69 Plate		0.15 Plate		60	Plate is controlling	
Turkey Point 3	B&W	5.67	1.97		0.31			290	Forgings
Turkey Point 4	B&W	5.16	1.97		0.30			280	Forgings
Point Beach 2	B&W/CE	6.96	1.24		0.25			240	Forgings
Ginna	B&W	7.66	1.16		0.25			220	Forgings
Kewaunee	CE	5.32	1.34		0.20			150	Forgings
Prairie Is. 2	SFAC	5.11	1.34		0.19			140	Forgings
Prairie Is. 1	SFAC	5.36	1.34		0.14			100	Forgings
North Anna 1	RDM	1.90	1.78		0.14			110	Forgings
North Anna 2	RDM	0.35	1.78		0.13			90	Forgings

Letter No. 0824-22-81 # 8110010040

OAK RIDGE NATIONAL LABORATORY

OPERATED BY

UNION CARBIDE CORPORATION

NUCLEAR DIVISION

POST OFFICE BOX X

OAK RIDGE, TENNESSEE 37830

August 24, 1981

Mr. Warren Hazelton
Engineering Branch
Division of Operating Reactors
Mail Stop 440
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Hazelton:

I visited the Oconee nuclear reactor site on July 21—23, 1981 and again on August 10—12, 1981. The purpose of these two visits was to observe and evaluate the ten-year in-service ultrasonic inspection of the welds in the unit 1 reactor vessel. The pressurized-water reactor is owned and operated by the Duke Power Company. The in-service ultrasonic inspections were performed by Babcock and Wilcox (B&W) personnel. A preliminary meeting discussing these inspections was held in Bethesda on March 24, 1981. My comments on that meeting were presented to you in Letter No. 0401-16-81, dated April 1, 1981. The following is a summary of the two visits to Oconee.

On July 21—23, 1981, John Gleim, John Gieske, and I visited the Oconee reactor site. On the first day (July 21) we attended the radiation/safety course (7:00 am to 3:00 pm) and obtained body counts. On the second day (July 22) we picked up dosimeters and security badges and met with Nick Economos (NRC representative from Region II). A copy of the Babcock and Wilcox proposed ultrasonic test procedure and a copy of the published issue of the NRC Regulatory Guide 1.150 (June 1981) were made available by Mr. Economos. We reviewed these documents briefly and discussed their compatibility. An unofficial report was received about noon that water had leaked into the electronics of the B&W Automated Reactor Inspection System (ARIS) as it was lowered into the reactor vessel and that the system would require some repair. After lunch we suited up and went into the reactor to examine the scanning fixture and the inspection site. We found the ARIS fixture isolated with no one around to discuss its status. When we got out of containment (about 4:00 pm) we learned that the ARIS fixture had been more severely damaged than earlier estimated and that the repair would take about five days. A meeting was arranged with B&W personnel to

discuss the ultrasonic test procedure, but the meeting was not scheduled until 8:30 am the following day. We were allowed to make copies of the B&W test procedure (treating them as proprietary information). We each took a copy of the test procedure back to the motel and studied it individually for a couple of hours, then met during dinner to discuss the procedure as a group. This did not accomplish much because the procedure is written in general language and few details are given discussing how some of the steps are to be performed. In summary, one cannot determine merely by reading the test procedure whether or not the ultrasonic inspection of the welds in the reactor vessel will be performed in accordance with Regulatory Guide 1.150.

On Wednesday, July 23, we met with Jim Brackett (Duke Power) and Mike Hacker and Bert Chism (B&W) to discuss and clarify the ultrasonic test procedure, and the following is a summary of that discussion:

We learned that B&W planned to inspect the near-surface region using shear waves and a refracted angle of 70°. One-half-inch diameter transducers are used and they mount two transducers diagonally side by side to increase the area inspected on a single scan. The ARIS scanning arm has been modified to accommodate five transducers (it was originally designed to operate with three). The sensitivity of the 70° inspection was determined at the B&W laboratories using notches and side-drilled holes. We were told they planned to calibrate for the inspection using a T/4 hole (side-drilled hole located at a depth of one-fourth of the sample thickness). The sensitivity of the 70° inspection was established in the laboratory using corner notches. Since corner notches typically reflect more sound energy, the sensitivity to notches located in the center of the specimen would normally be reduced. No information was available as to how much the sensitivity would be reduced. This may be an item that needs to be investigated at our laboratory. Most of the remaining details discussed were routine and the general appearance was that the ultrasonic inspection would be carried out in accordance with Regulatory Guide 1.150. We left Oconee with plans to return and witness actual testing when the ARIS fixture was repaired.

On Monday, August 10, Nick Economos and I returned to the Oconee plant site. After completing the routine details of body counts, dosimeters, and security badges, we met with Jim Brackett, Mike Hacker, and Bob Micheleski (B&W) to discuss the status of testing to date. We learned that the inspection of the three beltline welds had been completed. On B&W inspection sequence number 4 approximately 16 indications were obtained from one of the beltline welds during the 70° inspection. Most of the indications occurred in a 60° segment on the circumference of the weld, and most occurred at the top of the

weld. Preliminary indications showed the depths of the indications to fall in the range of about 0.3 to 0.9 in. with lengths of 0.1 to 0.75 in. The amplitudes of the signals varied from 50 to 150% of the distance amplitude correction (DAC) curve. A couple of the signals were also observed at the 60° inspection angle. No other significant indications had been obtained. We learned that B&W planned to reinspect these indications for confirmation and to obtain additional measurements. We asked to be present during these reexaminations.

There is no good way to compare these data with previous baseline ultrasonic inspection results. The 70° refracted angle inspection has never been performed before and the baseline inspection was performed using contact ultrasonic techniques whereas the in-service inspection was performed using immersion techniques. (This seems to be typical of most reactor sites that I visit. The so-called baseline inspection is usually performed by a different method than the in-service inspection.)

Nick Economos and I spent the afternoon in the B&W trailer that houses the ultrasonic instrumentation, video displays, and computer control system for the ARIS text fixture. The trailer was located in the security area, but outside of containment. We observed the calibration of the ultrasonic test system and learned more about the test procedure.

The ultrasonic inspection was performed as follows: the ARIS system was calibrated manually. The calibration blocks were mounted on the ARIS fixture and were physically located in the reactor pool. The computer was used to placed the transducers in the general vicinity of the calibration blocks, then the transducer arm was manipulated manually using push-button motorized controls to obtain peak signals from the reflectors in the calibration blocks. The scanning and detection portions of the inspection were performed automatically using computer control. A rectangular zig-zag scanning pattern (similar to a sewing machine pattern) was employed. The scanning pattern was performed in both transverse and longitudinal directions on each side of the weld. Scanning was performed at a gain 6 dB higher than the calibration setting. When indications were obtained the computer was stopped and the reflectors were evaluated manually by obtaining peak signals using the push-button motorized controls.

The calibration for the 70° inspection was performed in a manner slightly different from that discussed at our previous meeting on July 23. The beltline weld where the indications were obtained is approximately 9 in. thick. In this area the DAC curve was produced as follows: the reflected signal from a 0.2-in.-deep EDM (corner) notch located just below the cladding was adjusted to 80% full-scale height. A straight line was drawn from the left edge of the screen to this point. The signal from a T/4 hole (in the 9-in.-thick calibration block) occurred further in time and had an amplitude of about 40–50%.

A straight line was drawn from the 80% signal peak to this peak. The resulting lines produced the DAC curve. The signal from the 0.2-in.-deep notch and the T/4 hole were sharp and distinct and both had good signal-to-noise ratios.

The 0, 45, 60, and 70° (two transducers) inspections were performed simultaneously. The output signals from the 0, 45, 60, and one of the 70° transducers were fed to a Krautkramer KB-600 ultrasonic instrument and were displayed as four separate traces on a single oscilloscope. The fifth transducer (70° angle) was not displayed visually but was connected to an audible alarm module. A single audible alarm system was available for the four channels on the KB-6000 but it was continuously triggered by standing signals (multiples) that could not be eliminated, therefore it was disabled. To summarize: four transducers were displayed visually with no audible alarm and one transducer was connected to an audible alarm but was not displayed visually. During the performance of the ultrasonic examination one man monitored all five transducers and the technicians worked 12-h shifts. The technicians did not have to observe the oscilloscope continuously during the 12-h shifts but the time still seems a little long to me. This particular area (possible technician fatigue) is not covered in the Regulatory Guide.

On the following day we observed more testing and went into the containment area to observe the test fixture. (The ARIS system performs quite well when it is working.) Around 5:00 pm we learned that B&W planned to reinspect the 70° indications that night and we left word to call us at the motel. We received the call about 10:30 pm (does this sound familiar) and we were at the site by 11:30 pm.

The reinspection and evaluation process progressed rather slowly because of the additional requirements to evaluate each indication at 50 and 20% of DAC. The main reason for the increased inspection time, in my opinion, was not the requirement for additional data but was due to the fact that the technicians performing the evaluation were not familiar with 20% DAC or with the new Regulatory Guide. Also the data form had not been modified to have specific slots available to record the required information. The reinspection therefore was a training exercise as well as an evaluation process. Based on my observations for this inspection the evaluation time to obtain the additional data required by the new Regulatory Guide should increase the old evaluation time by a factor of 1.5 to 2.0 per indication. This should amount to roughly 30 min per indication at most. Compared to the overall inspection time (approximately 21 days) this is almost insignificant.

I requested and was allowed to observed the oscilloscope while one of the indications obtained at 70° was scanned in the normal manner. The indication on the screen was distinct and readily detectable but it only occurred on two or three scans and it could have easily been missed if the operator did not apply his full attention to the scope.

It is my opinion that individual audible alarms should be provided for each recorded channel during an ultrasonic inspection. The audible alarms are even more important when multiple channels are being observed simultaneously.

Compared with contemporary inspections that I have observed, the ultrasonic inspection of the welds in the unit 1 reactor vessel at Oconee were conducted in a competent manner. Test sensitivity and repeatability both appeared to be good. The addition of the 70° angle inspection provided coverage for the near-surface area and additional data required by the Regulatory Guide were recorded. I got the impression that B&W personnel were tryinng to follow the new Regulatory Guide and I know of no cases where they did not comply in the performance of the examination. The technicians could have been more familiar with the Regulatory Guide but competent personnel were on hand to direct them. I would suggest the use of individual audible alarms as well as video displays for each channel of data recorded but this is not required by either ASME Code or the Regulatory Guide. I will also mention that we were treated cordially by both Duke and B&W personnel. Our questions were answered and we were allowed to freely visit the inspection areas.

I am including a copy of the preliminary test results for inspection sequence number 4. No other significant indications were obtained during my visit.

Sincerely yours,

J. H. Smith

J. H. Smith
Nondestructive Testing Group
Metals and Ceramics Division

JHS:jlb

Enclosure

cc/enc: N. Economos, NRC
J. Gieske, Sandia
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M. R. Hum, NRC
A. L. Lotts
R. W. McClung
G. M. Slaughter
J. H. Smith/File

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EXAMINATION SUMMARY

SEQUENCE NO.: 4
FIGURE NO.: B1.1.3

NUMBER OF RECORDABLE INDICATIONS			
0	45	60	70
9	0	3	16

NUMBER OF INDICATIONS EXCEEDING SECTION XI			
0	45	60	70
0	0	0	0

COMMENTS:

1. All of the 0 degree indications are laminar reflectors of acceptable size.
2. All of the 60 and 70 degree indications are reflectors of acceptable size.
3. No limited examination was encountered on this weld.

Preliminary

FOR INFORMATION ONLY

FIGURE NO: 61.13

	SUR a	SUB 2a	l	a/l	a/t %	ALLOWABLE a/t %	EXCEEDS SECTION XI YES	EXCEEDS SECTION XI NO	EXCEEDS NOMOGRAPH YES	EXCEEDS NOMOGRAPH NO
00	[illegible]	.50	.448	.50	2.77	6.5		✓		✓
01		.375	1.492	.12	2.08	2.72		✓		✓
02		.375	.746	.25	2.08	3.7		✓		✓
600	.225		.597	.37	2.5	3.5		✓		✓
601		.15	.298	.25	0.83	3.7		✓		✓
602	.225		.298	.50	2.5	3.5		✓		✓
603		.10	.10	.50	0.55	6.5		✓		✓
604	.30		.298	.50	3.3	3.5		✓		✓
605		.15	.298	.25	0.83	3.7		✓		✓
606		.10	.10	.50	0.55	6.5		✓		✓
607	.225		.298	.50	2.5	3.5		✓		✓
608		.30	.30	.50	1.67	6.5		✓		✓
609		.10	.10	.50	1.1	6.5		✓		✓
61[illegible]		.10	.10	.50	0.55	6.5		✓		✓
[illegible]	.187		1.64	.11	2.08	2.24		✓		✓
12	.225		.149	.50	2.5	3.5		✓		✓
13	.15		1.343	.11	1.66	2.24		✓		✓
14		.15	.298	.25	0.83	3.7		✓		✓
15		.15	.448	.16	0.83	2.76		✓		✓

The above are preliminary results pending final review.

M. I. Hacker
Howard Stoppelmann

FOR INFORMATION ONLY

Babcock & Wilcox

EXAMINATION SUMMARY

SEQUENCE NO.: 4
FIGURE NO.: 5.1.1.3

NUMBER OF RECORDABLE INDICATIONS			
0	45	60	70
9	0	3	16

NUMBER OF INDICATIONS EXCEEDING SECTION XI			
0	45	60	70
0	0	0	0

COMMENTS:

(1) All of the 0 degree indications are laminar reflectors of acceptable size.

(2) All of the 60 and 70 degree indications are reflectors of acceptable size.

(3) No limited examination was encountered on this weld.

[illegible]

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FIGURE NO: 6.1.5

[illegible]	SUR a	SUB 2a	l	a/l	a/t %	ALLOWABLE a/t %	EXCEEDS SECTION XI YES	EXCEEDS SECTION XI NO	EXCEEDS NOMOGRAPH YES	EXCEEDS NOMOGRAPH NO
0	[illegible]	.50	.448	.50	2.77	6.5		✓		✓
01		.375	1.492	.12	2.08	2.72		✓		✓
02		.375	.746	.25	2.08	3.7		✓		✓
00	.225		.597	.37	2.5	3.5		✓		✓
01		.15	.298	.25	0.83	3.7		✓		✓
02	.225		.298	.50	2.5	3.5		✓		✓
03		.10	.10	.50	0.55	6.5		✓		✓
04	.30		.298	.50	3.3	3.5		✓		✓
05		.15	.298	.25	0.83	3.7		✓		✓
06		.10	.10	.50	0.55	6.5		✓		✓
07	.225		.298	.50	2.5	3.5		✓		✓
08		.30	.30	.50	1.67	6.5		✓		✓
09		.10	.10	.50	1.1	6.5		✓		✓
010		.10	.10	.50	0.55	6.5		✓		✓
011	.187		1.64	.11	2.08	2.24		✓		✓
12	.225		.149	.50	2.5	3.5		✓		✓
13	.15		1.343	.11	1.66	2.24		✓		✓
14		.15	.298	.25	0.83	3.7		✓		✓
15		.15	.448	.16	0.83	2.76		✓		✓

the above are preliminary [illegible] pending final review.

[illegible]
Howard [illegible]

	%Cu	INITIAL RT_{NDT}	FLUENCE WHERE CRACK INITIATION COULD OCCUR (10^{19} neut/cm²)	Current FLUENCE ([illegible] neut/cm²)	FLUENCE/EFPY (10^{19} neut/cm²)	No. OF EFPY OF CONTINUED OPERATION	JOB #
...EE 1	.31	30	1.0×10^{19} .75 no arrest	0.225	0.046		(4H)38 4x
TMI 1	.31	30	1.0×10^{19} "	0.207	0.060		(4H 38 4x
RANCHO SECO	.31	30	1.0×10^{19} "	0.189	0.058		(4H)38 4x
ARKANSAS 1	.31	30	1.0×10^{19} "	0.173	0.045		(4H)38 4x
CRYSTAL RIVER 3	.31	30	1.0×10^{19} "	0.119	0.055		(4H)38 4x
OCONEE 2 *	F .35	30	1.8×10^{19} 1-1.25 a 5.5	0.259	0.061		4044
...CONEE 3 *	F .24	30	2.2×10^{19} 1.0 a 3.5"	0.262	0.061		46 4.
...AVIS BESSE *	F .24	30	2.2×10^{19} 1.0 a 3.5"	0.083	0.066		46 45

* RING FORGINGS

mean K_{Ic}

	% Cu	INI. R_{NDT}	[illegible]NCE WHERE [illegible] INITIATION COULD OCCUR (10^{19} neut/cm²)		Current FLUENCE ([illegible] neut/cm²)	FLUENCE/EFPY (10^{19} neut/cm²)	No. OF EFPY OF CONTINUED OPERATION	J[illegible]
S[illegible] OFRE	.19	60	2.2×10^{19}	.5-1.0 no arrest	1.535	0.174		4T 4V
ROBINSON 2	.34	0	1.7×10^{19}	1.0 no arrest	1.081	0.159		4U 4W
TURKEY PT. 3*	.31	0	2.9×10^{19}	1.0 a 4.5"	1.116	0.197		4R [illegible]
POINT BEACH 1	.24	0	2.1×10^{19}	.5-.75 no arrest	0.699	0.091		4L 4V
SURRY 1	.18	0	3.8×10^{19}	.5-.75 no arrest	0.153	0.034		4M 4W
ZION 1	.31	30	1.0×10^{19}	.75 no arrest	0.086	0.020		4O 4Y
IP-2 **	.25	60	1.3×10^{19}	.5-1.0 no arrest	0.199	0.050		4R 4-7
YANKEE ROWE **	.20	10	2.9×10^{19}	.5-.75 no arrest	1.103	0.078		4T 45

* RING FORGINGS
** PLATE DATA

mean K_{Ic}

	%Cu	INITIAL RT_{NDT}	FLUENCE WHERE CRACK INITIATION COULD OCCUR (10^{19} neut/cm^2)	Current FLUENCE (10^{19} neut/cm^2)	FLUENCE/EFPY (10^{19} neut/cm^2)	No. OF EFPY [illegible] CONTINUED OPS [illegible]	JOB #
FT. [illegible]HOUN	.35	30	.9 X 10^{19} 1.25" no arrest	0.819	0.172		44
MAINE YANKEE	.36	10	1.4 X 10^{19} 1.0 no arrest +	0.468	0.085		46
[illegible]LISADES	.25	0	2.0X10^{19} .75-1.0 no arrest	0.455	0.116		8R41
CALVERT CLIFFS I	.30			0.603	0.147		

mean K_{Ic}

PLANT	%Cu	IN. RT_{NDT}	[illegible]NCE WHERE INITIATION COULD OCCUR (10^{19} neut/cm²)	Current FLUENCE (10^{19} neut/cm²)	FLUENCE/EFPY (10^{19} neut/cm²)	No. OF EFPY OF CONTINUED OPERATION	[illegible]
SAN ONOFRE	.19	60	1.6×10^{19} .75-1.0 no arrest	1.535	0.174		5F
ROBINSON 2	.34	0	0.9×10^{19} 1.0-1.25 no arrest	1.081	0.159		✓ 5G
TURKEY PT. 3*	.31	0	1.70×10^{19} r-1.25 a 3.25	1.116	0.197	1.	✓ (~~X~~) 5W
POINT BEACH 1	.24	0	1.7×10^{19} .5-1.0 no arrest	0.699	0.091		✓ 5I
SURRY 1	.18	0	3.0×10^{19} .5-1.0 no arrest	0.153	0.034		✓ 5J
ZION 1	.31	30	0.8×10^{19} .5-.75 no arrest	0.086	0.020		✓ 5L
IP-2 **	.25	60	0.9×10^{19} .75-1 no arrest	0.199	0.050		63 5M
YANKEE ROWE**	.20	10	2.2×10^{19} .5- a-3	1.103	0.078		7B 5H

* RING FORGINGS

** PLATE DATA

CF [illegible] 10-1-86

R. [illegible]

Code K_{Ic}

PLANT	%Cu	INITIAL RT_{NDT}	FLUENCE WHERE CRACK INITIATION COULD OCCUR (10^{19} neut/cm²)	Current FLUENCE (10^{19} neut/cm²)	FLUENCE/EFPY (10^{19} neut/cm²)	No. OF EFPY OF CONTINUED OPERATION	JOB #
*FT. CALHOUN	.35	30	$.6 \times 10^{19}$ .75 a 3.5	0.819	0.172		7D 5Q
MAINE YANKEE	.36	10	0.7×10^{19} .75 no arrest	0.468	0.085		7N 5R
PALISADES	.25	0	1.5×10^{19} .5-.75 a 3.5"	0.455	0.116		7U 5T
CALVERT CLIFFS 1	.30			0.603	0.147		

Code K_{Ic}

PLANT	%Cu	INITIAL RT_{NDT}	FLUENCE WHERE CRACK INITIATION COULD OCCUR (10^{19} neut/cm²)	Current FLUENCE (10^{19} neut/cm²)	FLUENCE/EFPY (10^{19} neut/cm²)	No. OF EFPY OF CONTINUED OPERATION	JOB #
OCONEE 1	.31	30	0.8×10^{19} .5-1.0 no arrest	0.225	0.046		5149
TMI 1	.31	30	0.8×10^{19} "	0.207	0.060		5149
RANCHO SECO	.31	30	0.8×10^{19} "	0.189	0.058		5149
ARKANSAS 1	.31	30	0.8×10^{19} "	0.173	0.045		5149
CRYSTAL RIVER 3	.31	30	0.8×10^{19} "	0.119	0.055		5149
OCONEE 2 * F	.35	30	0.8×10^{19} 1.0 a 2.75	0.259	0.061		7550
OCONEE 3 * F	.24	30	1.7×10^{19} 1/4-1.5 a 1.75	0.262	0.061		5V
DAVIS BESSE * F	.24	30	1.7×10^{19}	0.083	0.066		5V

* RING FORGINGS

Code K_{Ic}