DCD-016

July 8, 1982

SUBJECT: SUMMARY OF MEETING WITH COMBUSTION ENGINEERING OWNERS GROUP (CEOG) ON JUNE 23, 1982, CONCERNING THE PRESSURIZED THERMAL SHOCK (PTS) ISSUE

A meeting was held in Bethesda, MD on June 23, 1982, with the CEOG to review agreement and disagreement of concerns expressed by the staff in the June 9, 1982 meeting relating to the PTS issue. Information and recommendations were presented in the areas of RT_{NDT} predictions, deterministic and probabilistic fracture mechanics, and thermal-hydraulic transients.

Information was presented to substantiate their recommendation that the current RG 1.99 be used to predict the upper bound shift for high copper, high nickel material at fluence $>10^{19}$ nvt and that the Guthrie (HEDL) correlation be used to predict the upper bound shift for medium/low copper, high/low nickel material at fluence $<10^{19}$ nvt. The prediction should be integrated with the surveillance test data for plant specific RT_{NDT} shift.

Fracture mechanics analyses using RT_{NDT} estimates in CEN-189 show acceptable EOL toughness for all CEOG vessel welds. CEOG transient evaluation indicated less severe PTS conditions than NRC T_F 250°F, P2500 psi. Their evaluation of the Ginna and Rancho Seco events show RT_{NDT} where K_I equal K_{Ic} of 360°F (T_F 335°F) and 315°F (T_F 265°F), respectively. CEOG gave credit to warm prestressing, small crack size, negligible clad effect and heat transfer coefficient of 300 BTU/hr/ft^2/F°. CEOG expressed the opinion that ductile crack arrest would prevent loss of vessel integrity in the "worst case" CE vessels.

A probabilistic fracture mechanics assessment of overcooling events was presented by the CEOG. Five initiating events were identified which were expected to provide the greatest challenge in fault tree of about 220 scenarios with respect to PTS to vessel integrity. The events were bounded by the main steam line break. The study was presented in CEN-212.

The meeting served to continue the dialogue and obtain feedback in the areas of agreement and disagreement for resolution of the PTS issue. Enclosed are handouts of the meeting which identify specific points made by CEOG. Also, an attendance list is enclosed for record purpose.

Original signed by

F. B. Litton, Generic Issues Branch
Division of Safety Technology

8208030033 820708
PDR TOPRP EMVC-E
C PDR

Enclosures:
1. Handouts
2. Attendance List
cc w/enclosures:
See next page

OFFICE	MTEB:DE	ORB#4:DL	C-ORB#4:DL		
SURNAME	FBLitton/cab	GVissing	JStolz		
DATE	6/29/82	7/6/82	6/ /82		

ORB#4:DL

MEETING SUMMARY DISTRIBUTION

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

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RFraley, ACRS-10
Program Support Branch

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

Meeting Participants Fm. NRC:

See Enclosure 2

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

W. Hazelton

T. Murley
H. Thompson

L. Shao
R. Bernero

G. Knighton
J. Austin

B. D. Liaw

D. Ziemann

E. Abbott
R. Johnson
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G. Zech
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C. Serpan
L. Shotkin
W. Johnston
A. Spano

CEOG/NRC PTS MEETING

(6/23/82)

I. INTRODUCTION

II. RT_{NDT} SHIFT PREDICTIONS
   A. RG-1.99 AND HEDL (GUTHRIE)
   B. CEOG RECOMMENDATION

III. FRACTURE MECHANICS
   A. PARAMETER STUDY RESULTS
   B. ACCEPTANCE CRITERIA
   C. CEOG RECOMMENDATION

IV. THERMAL-HYDRAULIC TRANSIENTS
   A. CE NSSS DESIGN FEATURES
   B. PARAMETER STUDY RESULTS
   C. CEOG RECOMMENDATION

V. PROBABILITY STUDIES
   A. RESULTS
   B. COEG RECOMMENDATION

VI. SUMMARY

RT_{NDT} SHIFT PREDICTIONS

- BACKGROUND

- RG 1.99 AND GUTHRIE (HEDL) CORRELATIONS,
  ADVANTAGES AND DISADVANTAGES

- CEOG POSITION ON SHIFT PREDICTION METHODS

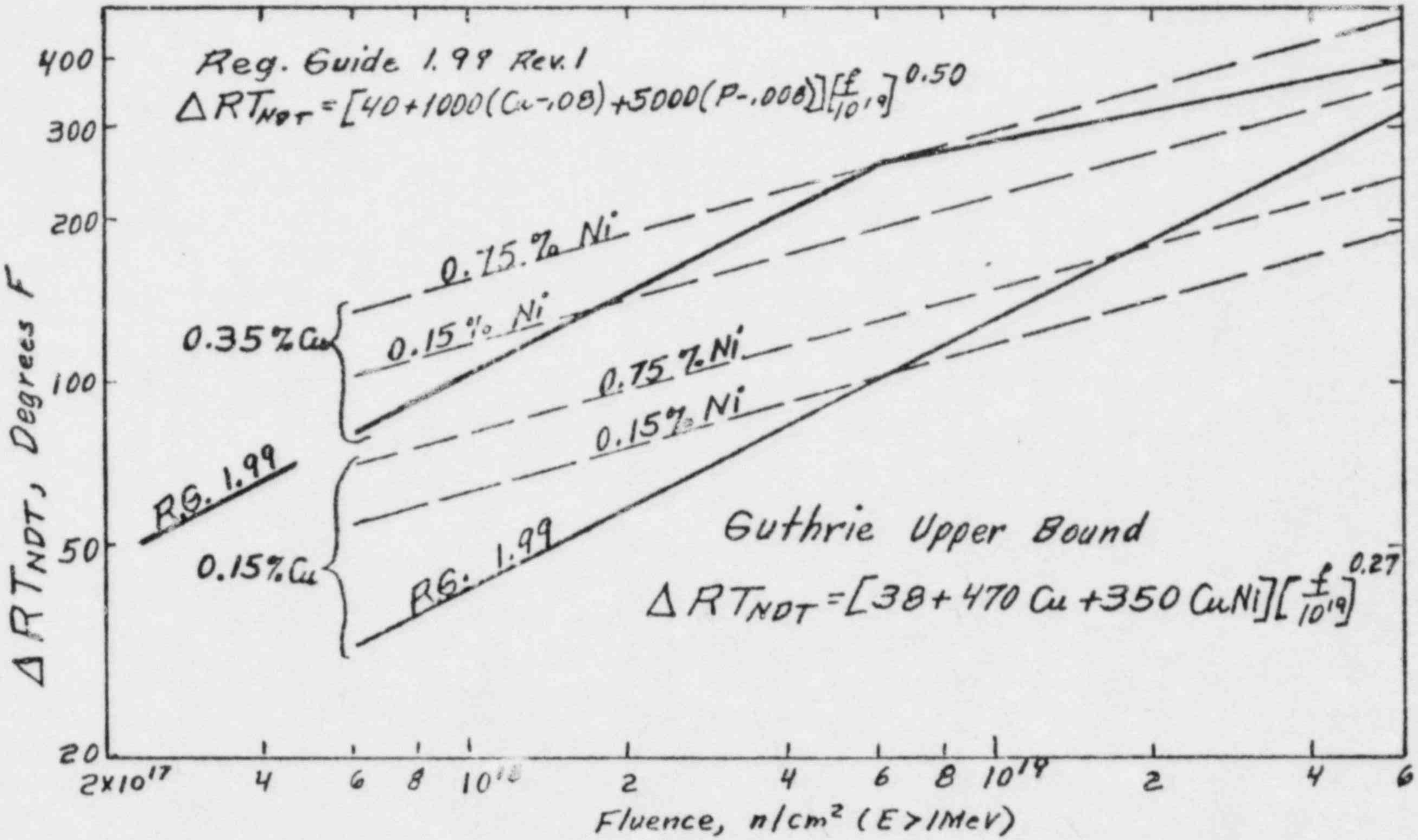
- CEOG RECOMMENDATIONS
  CURRENT ACTION FOR NRC POSITION
  NEAR TERM ACTION

- OTHER CONSIDERATIONS
  WELD METAL RT_{NDT} AND INITIATION TOUGHNESS

RT_{NDT} SHIFT PREDICTIONS

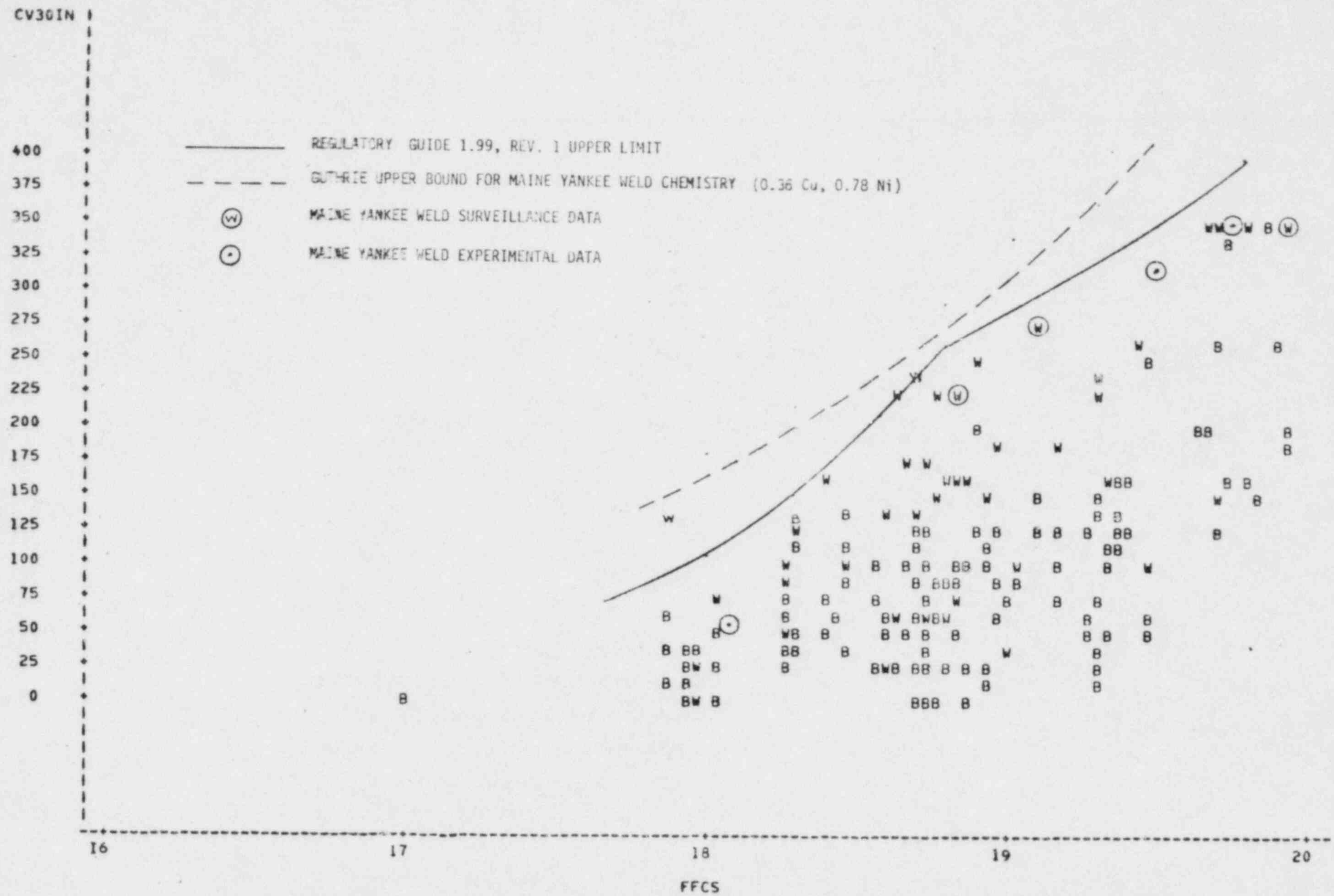
BACKGROUND

- VENDOR CURVES (1970)
  - BOUNDING CURVES
  - Cu AND FLUENCE
- REGULATORY GUIDE 1.99 (1975)
  - BOUNDING CURVES
  - Cu, PHOS AND FLUENCE
- CE WELD MODEL (1976 AND 1978)
  - BEST ESTIMATE CURVES
  - Cu PLUS Ni, Si, Mn, Mo AND Cr
  - WELDS ONLY
- REV. 1, RG. 1.99 (1977)
  - BOUNDING CURVES
  - Cu, PHOS AND FLUENCE
- MPC-6 AND ASTM (1974 - 1981)
  - STATISTICALLY DERIVED CURVES
  - Cu AND FLUENCE (PHOS UNIMPORTANT)
  - EVIDENCE FOR SATURATION
- CURRENT CONCERNS/EFFORTS
  - REDUCE RG. 1.99 CONSERVATISM AT HIGH FLUENCE
  - ACCOUNT FOR NICKEL CONTENT
  - ACCOUNT FOR DAMAGE SATURATION
  - ACCOUNT FOR PRODUCT FORM AND SPECIFICATION DIFFERENCES



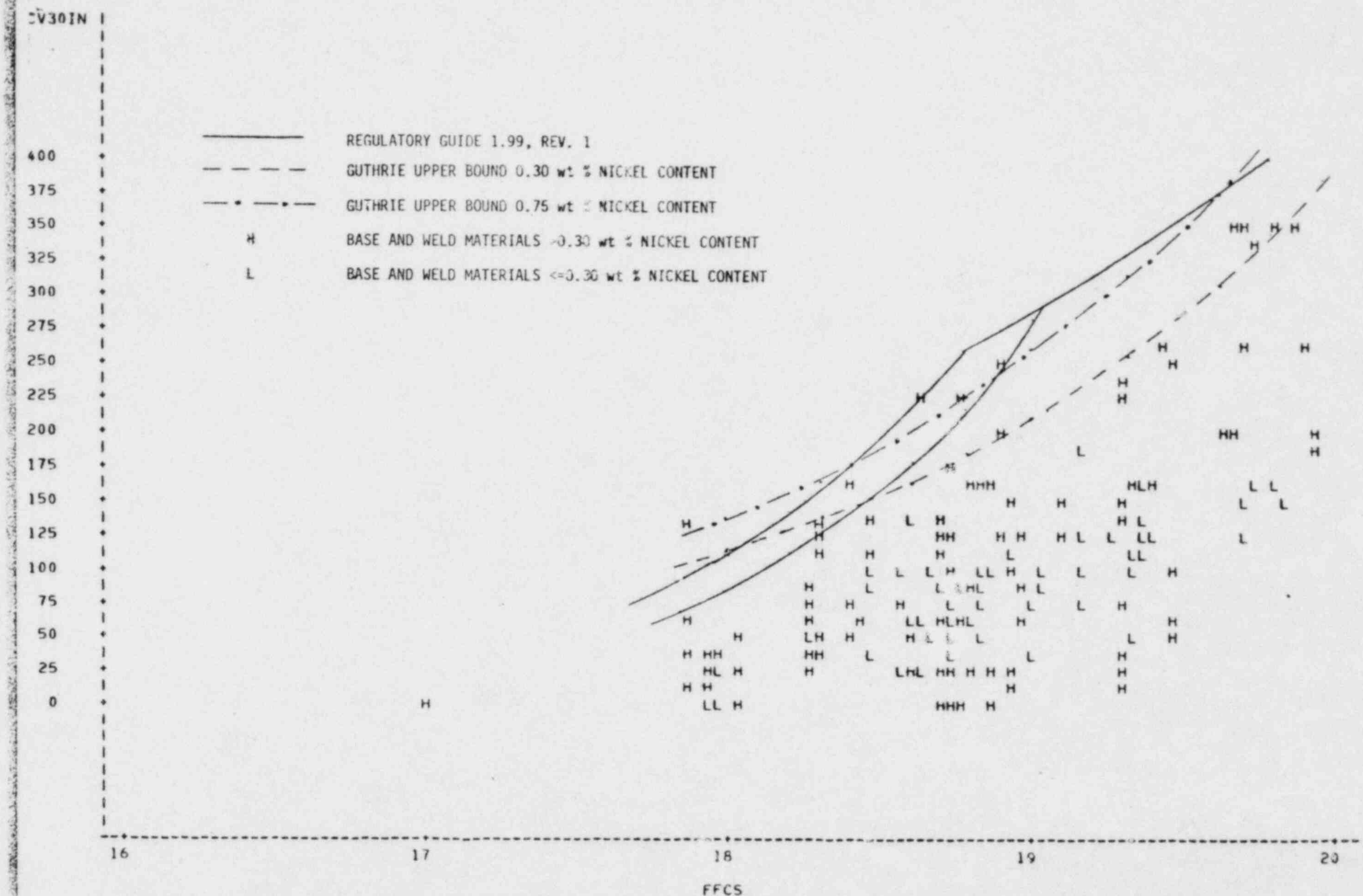
Comparison of Guthrie and Reg. Guide 1.99 Rev. 1 Formulas for the Upper Bound Values of ΔRT_{NDT}

PLOT OF CV30IN*FFCS SYMBOL IS VALUE OF PPF



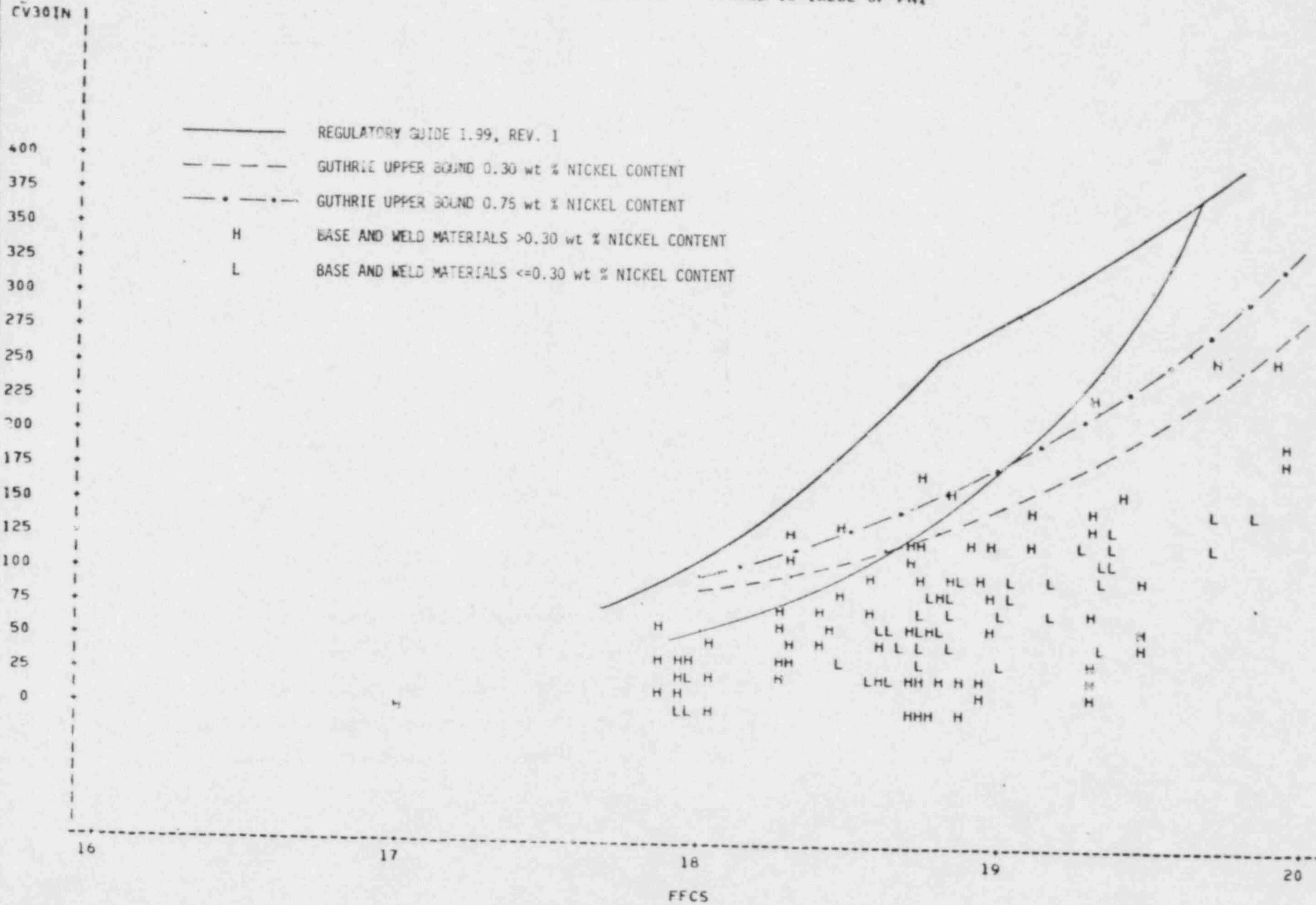
TRANSITION TEMPERATURE SHIFT VS FLUENCE 12:26 FRIDAY, JUNE 18, 1982 1
COPPER CONTENT <=0.30

PLOT OF CV30IN*FFCS SYMBOL IS VALUE OF PN$



TRANSTION TEMPERATURE SHIFT VS FLUENCE
COPPER CONTENT <=0.20

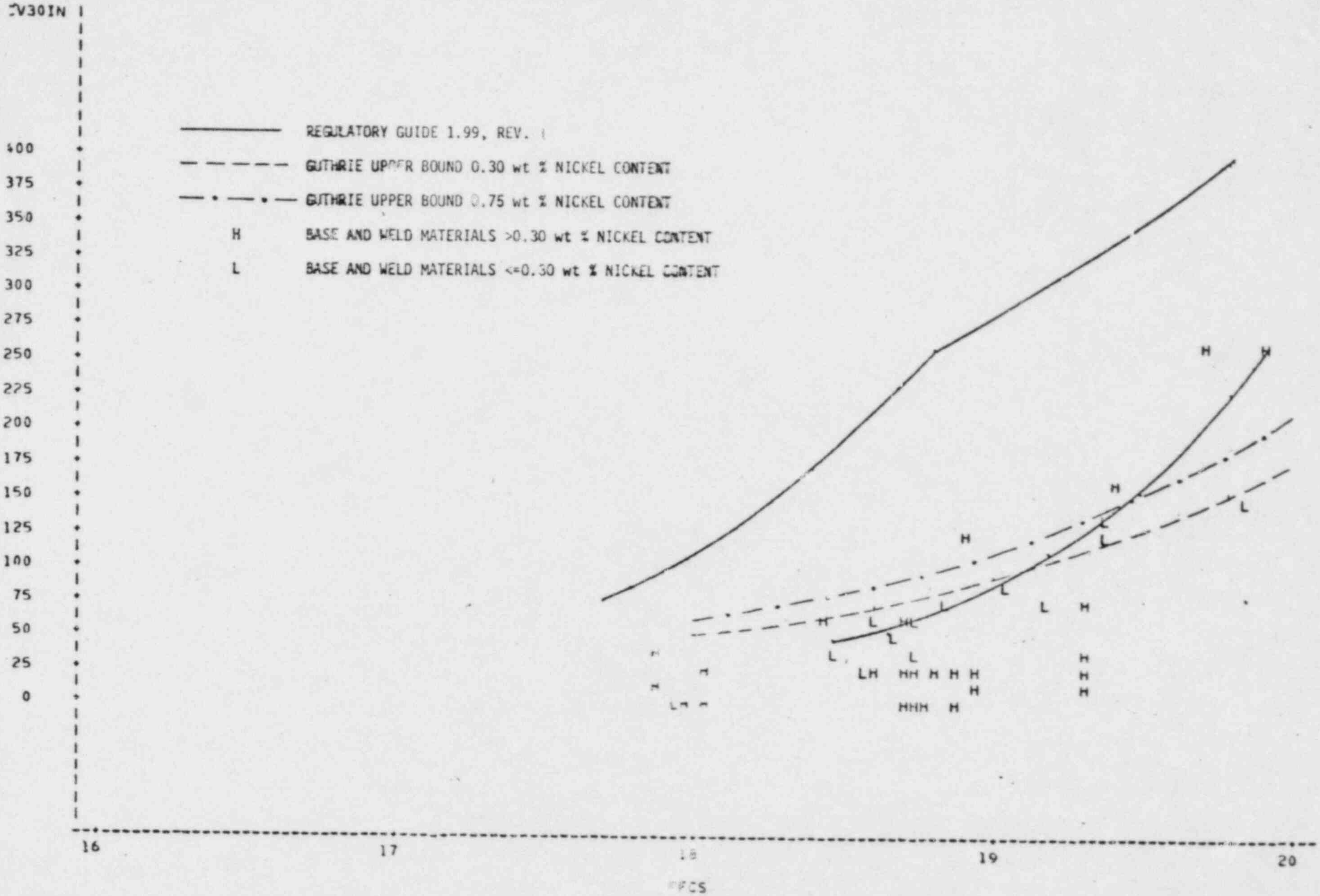
PLOT OF CV30IN*FFCS SYMBOL IS VALUE OF PNI



TRANSTION TEMPERATURE SHIFT VS FLUENCE 12:26 FRIDAY, JUNE 18, 1982 4

COPPER CONTENT <=0.10

PLOT OF CV30IN*FFCS SYMBOL IS VALUE OF PNI



RT_{NDT} SHIFT PREDICTIONS

RG 1.99 AND GUTHRIE CORRELATIONS

ADVANTAGES AND DISADVANTAGES

- RG 1.99 REV. 1
  - CONSERVATIVELY DEPICTS UPPER LIMIT BEHAVOIR FOR HIGH Cu AND Ni WELDS
  - GROSSLY CONSERVATIVE FOR HIGH Cu, LOW Ni, MATERIALS
  - OVERLY CONSERVATIVE AT HIGH FLUENCE (NO CREDIT FOR SATURATION FOR LOWER COPPER CONTENT MATERIAL)

- GUTHRIE (HEDL)
  - CREDIT GIVEN FOR SATURATION
  - CREDIT GIVEN FOR NICKEL CONTENT
  - SIGNIFICANT OVERALL IMPROVEMENT FROM RG 1.99
  - GROSSLY CONSERVATIVE FOR HIGH Cu AND Ni MATERIALS
  - SOME INCONSISTENCIES WITH OBSERVED TRENDS:
    - . DEPENDENCY ON Ni NOT CONTINUOUS
    - . DEPENDENCY ON FLUENCE VARIES

RT_{NDT} SHIFT PREDICTIONS

CEOG POSITION

Tru

- GUTHRIE (HEDL) CORRELATION IS TO PRELIMINARY AT THIS TIME AS SOLE BASIS FOR ESTABLISHING A REGULATORY POSITION

- RG 1.99 IS BEST AVAILABLE METHOD FOR DEFINING CONSERVATIVE UPPER LIMIT RADIATION BEHAVIOR

- RG 1.99 IS BEST AVAILABLE METHOD FOR DEFINING TREND OF UPPER LIMIT SHIFT WITH NEUTRON FLUENCE

- GUTHRIE CORRELATION IS REASONABLE, BUT CONSERVATIVE, METHOD FOR SHIFT PREDICTIONS ABOVE APPROXIMATELY 10^{19} N/CM^2 WITH RG. 1.99 DEFINING UPPER BOUND BEHAVIOR

RT_{NDT} SHIFT PREDICTIONS

CEOG RECOMMENDATIONS - CURRENT ACTION

- RG 1.99 FOR UPPER BOUND FOR HIGH COPPER, HIGH NICKEL
- GUTHRIE (HEDL) FOR MEDIUM/LOW COPPER, HIGH/LOW NICKEL
- CONSIDER AVAILABLE SURVEILLANCE DATA

CEOG RT_{NDT} PREDICTIONS

PLANT NAME	CURRENT RT_{NDT} (1)	EOL RT_{NDT}
FT. CALHOUN	213 (2)	310 (2)
MAINE YANKEE	168 (2)	280 (2)
PALISADES	166 (2)	325 (2)
CALVERT CLIFFS-1	151 (3)	273 (3)
CALVERT CLIFFS-2	82 (3)	164 (3)
ST. LUCIE-1	99 (3)	220 (3)
MILLSTONE-2	123 (3)	197 (3)

NOTES: (1) 12/31/81 VALUE

(2) RG-1.99

(3) HEDL (GUTHRIE) CURVES

RT_{NDT} SHIFT PREDICTIONS

CEOG RECOMMENDATIONS - NEAR TERM ACTION

- DEVELOP IMPROVED SHIFT PREDICTION METHOD(S) INCORPORATING POSITIVE FEATURES OF RG 1.99, AND GUTHRIE CORRELATIONS PLUS OBSERVED TRENDS:
  - RG 1.99 UPPER LIMIT
  - "BROKEN BACK" CURVES
  - DISCRETE NICKEL CONTENTS
  - OTHER CHEMICAL ELEMENTS (EG, Si, C, Mn, ETC)
  - PLATE VS WELD MATERIAL DIFFERENCES

RT_{NDT} SHIFT PREDICTIONS

WELD METAL RT_{NDT} AND INITIATION TOUGHNESS

- ASME CODE SECTION XI KIC CURVES ARE CONSERVATIVE FOR SUB-ARC WELDS (FIGURES)
- RT_{NDT} DATA FOR OLDER VESSELS LIMITED TO WELD QUAL RESULTS AT 10F
- GENERIC DATA FOR SUB-ARC WELDS SHOW MEAN RT_{NDT} OF -56F
- HOW DO YOU ESTIMATE VESSEL FRACTUR TOUGHNESS?
  - CONSERVATIVE RT_{NDT} PLUS CONSERVATIVE KIC CURVES GIVES ULTRA-CONSERVATIVE INITIATION TOUGHNESS ESTIMATE
  - BEST ESTIMATE RT_{NDT} REDUCES CONSERVATISM
- HOW CONSERVATIVE ARE WELD RT_{NDT} VALUES?
  - WELD RT_{NDT} TYPICALLY SET BY DROP WEIGHT NDTT
  - DROP WEIGHT TEST GIVES ARREST TOUGHNESS, NOT INITIATION TOUGHNESS

FIGURE 2

K_{IC} REFERENCE TOUGHNESS CURVE WITH SUPPORTING DATA

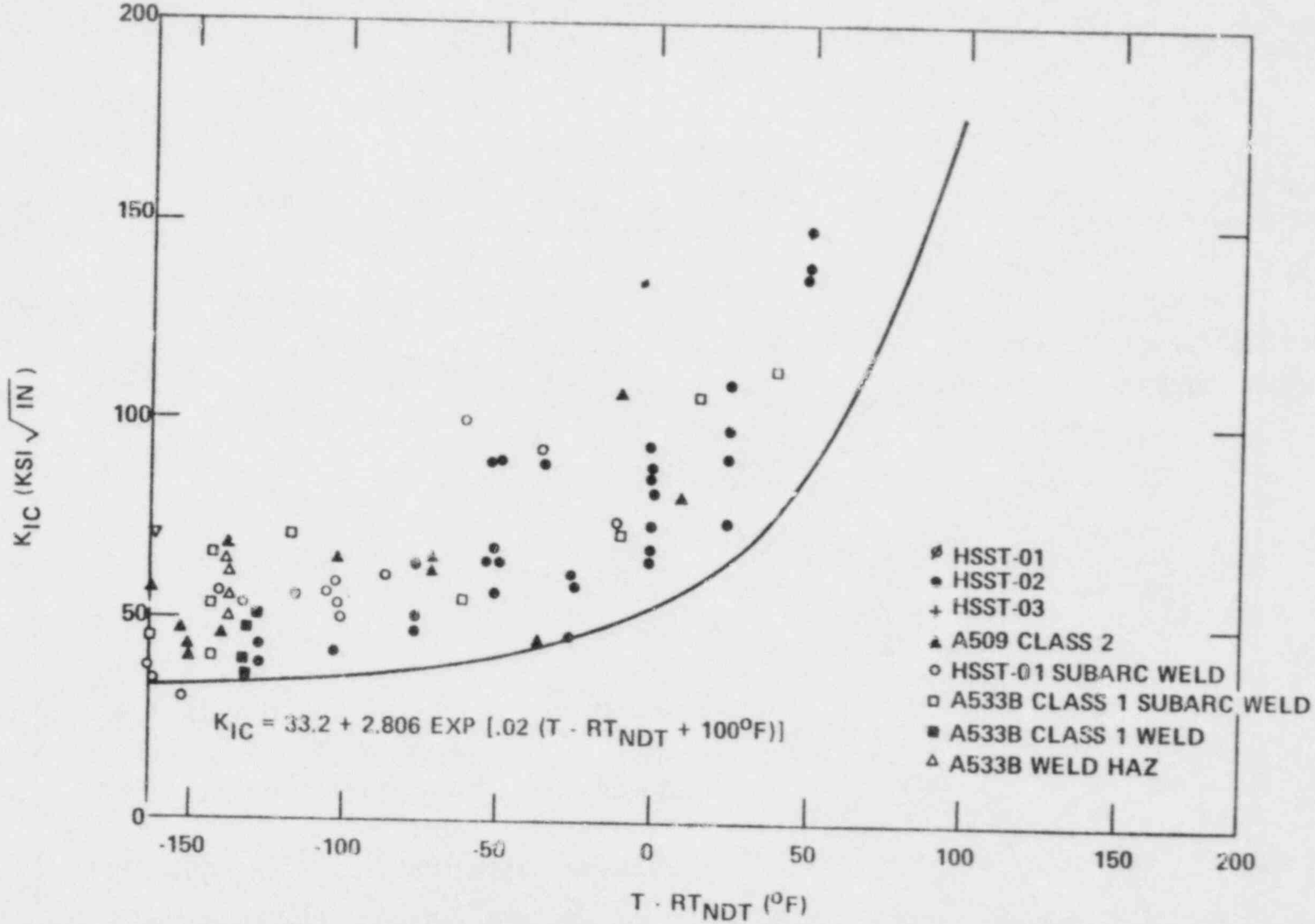
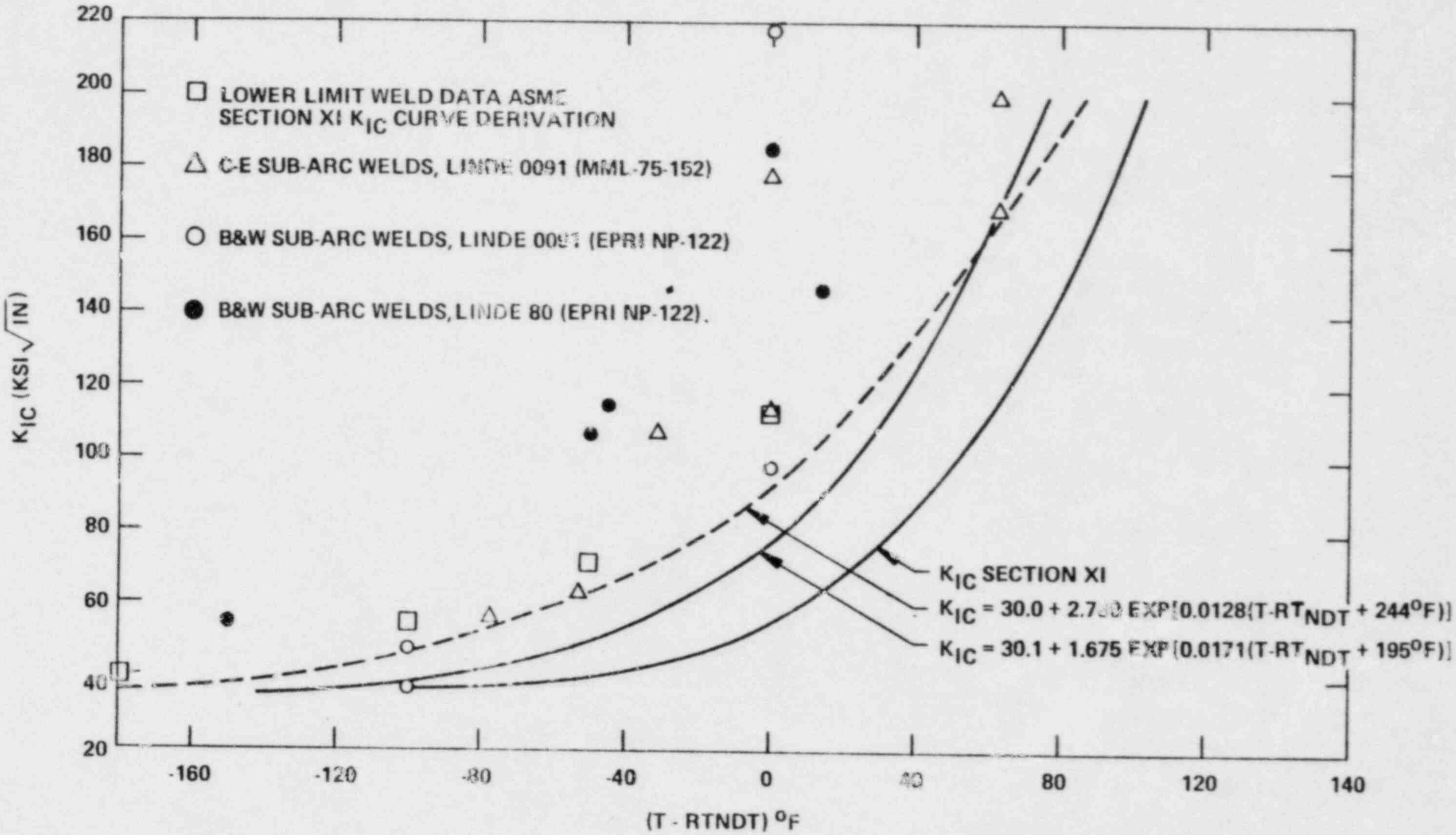


FIGURE 4

K_{IC} vs (T – RTNDT)



RT_{NDT} SHIFT PREDICTIONS

WELD METAL BEST ESTIMATE RT_{NDT} APPROACH

- ROLFE/NOVAK/BARSOM RELATIONSHIP

  $(K_{IC})^2 = 2\,E\,(CVN)^{3/2}$

  E = ELASTIC MODULUS $\approx 30.2\times10^6 - 0.46\times10^4 T$

  CVN = CHARPY ENERGY AT 10F FROM WELD QUAL RESULTS

- ASME CODE SECTION XI KIC RELATIONSHIP

  $KIC = 33.2 + 2.806\ EXP\left[0.2\ (T - RT_{NDT} + 100)\right]$

- WELD SPECIFIC KIC RELATIONSHIP (CEOG)

  $KIC = 30.1 + 1.675\ EXP\left[0.0171\ (T - RT_{NDT} + 195)\right]$

- SUMMARY OF APPROACH

  AVG CVN FROM WELD QUAL RESULTS AT 10F

  ESTIMATE KIC AT 10F USING ROLFE/NOVAK/BARSOM RELATION

  ESTIMATE RT_{NDT} FROM ASME CODE AND CEOG RELATIONSHIPS

INITIAL RT_{NDT} ESTIMATE - CE WELDS

VESSEL	AVG. CVN ENERY AT 10F (FT-LB)	R/W/B KIC AT 10F (KSI $\sqrt{IN}$)	ESTIMATED RT_{NDT} (^oF) ASME SEC. XI	CEOG/WELDS	CEN-189
FORT CALHOUN	61.0 (IS)*	169.4	- 84	- 54	- 50
	57.3 (LS)*	161.6	- 81	- 50	- 50
CALVERT CLIFFS #1	44.3 (IS)	133.2	- 69	- 36	- 30
	57.0 (LS)	161.0	- 81	- 50	- 50
CALVERT CLIFFS #2	73.0 (IS)	193.8	- 92	- 63	- 50
	131.7 (LS)	301.6	-118	- 93	- 80
MAINE YANKEE	61.0 (IS)	169.4	- 84	- 54	- 50
	57.0 (LS)	161.0	- 81	- 50	- 40
PALISADES	[illegible].7 (IS)	125.0	- 64	- 31	- 25
	53.[illegible] (LS)	153.9	- 78	- 47	- 45
MILLSTONE #2	73.0 (IS)	193.8	- 92	- 63	- 50
	73.0 (LS)	193.8	- 92	- 63	- 50
ST. LUCIE #1	67.3 (IS)	182.3	- 89	- 59	- 50
	87.0 (LS)	221.0	-100	- 72	- 50

* IS - INTERMEDIATE SHELL LONG SEAM WELD

* LS - LOWER SHELL LONG SEAM WELD

RT_{NDT} SHIFT PREDICTIONS

BEST ESTIMATE RT_{NDT} FOR WELDS

\- CONCLUSIONS -

- USE OF R/N/B RELATIONSHIP AND SECTION XI K_{IC} CURVES YIELDS VERY LOW RT_{NDT} ESTIMATES

- USE OF R/N/B PLUS WELD SPECIFIC K_{IC} CURVES YIELDS RT_{NDT} ESTIMATES CONSISTENT WITH CEN-189 RT_{NDT} ESTIMATES

- USE OF -20F RT_{NDT} FOR INITIAL WELD TOUGHNESS WILL YIELD SIGNIFICANT UNDER-ESTIMATE OF CRACK INITIATION TOUGHNESS

- CEN-189 BEST ESTIMATE INITIAL RT_{NDT} VALUES USED IN CONJUNCTION WITH ASME CODE, SECTION XI CURVES WILL YIELD CONSERVATIVE ESTIMATES OF CRACK INITIATION TOUGHNESS PROPERTIES

RECOMMENDATION

- USE CEN-189 BEST ESTIMATE INITIAL RT_{NDT} VALUES FOR EVALUATING C-E VESSEL PROPERTIES RELATIVE TO REGULATORY POSITION RT_{NDT} LIMITS

RECOMMENDED INITIAL RT_{NDT} FOR CEOG VESSEL WELDS

VESSEL	SHELL COURSE LONGITUDINAL SEAM WELD	INITIAL RT_{NDT} (°F)
FORT CALHOUN	INTERMEDIATE	- 50
	LOWER	- 50
CALVERT CLIFFS #1	INTERMEDIATE	- 30
	LOWER	- 50
CALVERT CLIFFS #2	INTERMEDIATE	- 50
	LOWER	- 80
MAINE YANKEE	INTERMEDIATE	- 50
	LOWER	- 40
PALISADES	INTERMEDIATE	- 25
	LOWER	- 45
MILLSTONE #2	INTERMEDIATE	- 50
	LOWER	- 50
ST. LUCIE #1	INTERMEDIATE	- 50
	LOWER	- 50

FRACTURE MECHANICS

1. PREVIOUS ANALYSES (CEN 189, ETC)
   SHOW ACCEPTABLE TOUGHNESS AT EOL FOR "WORST CASE"
   CE TRANSIENTS

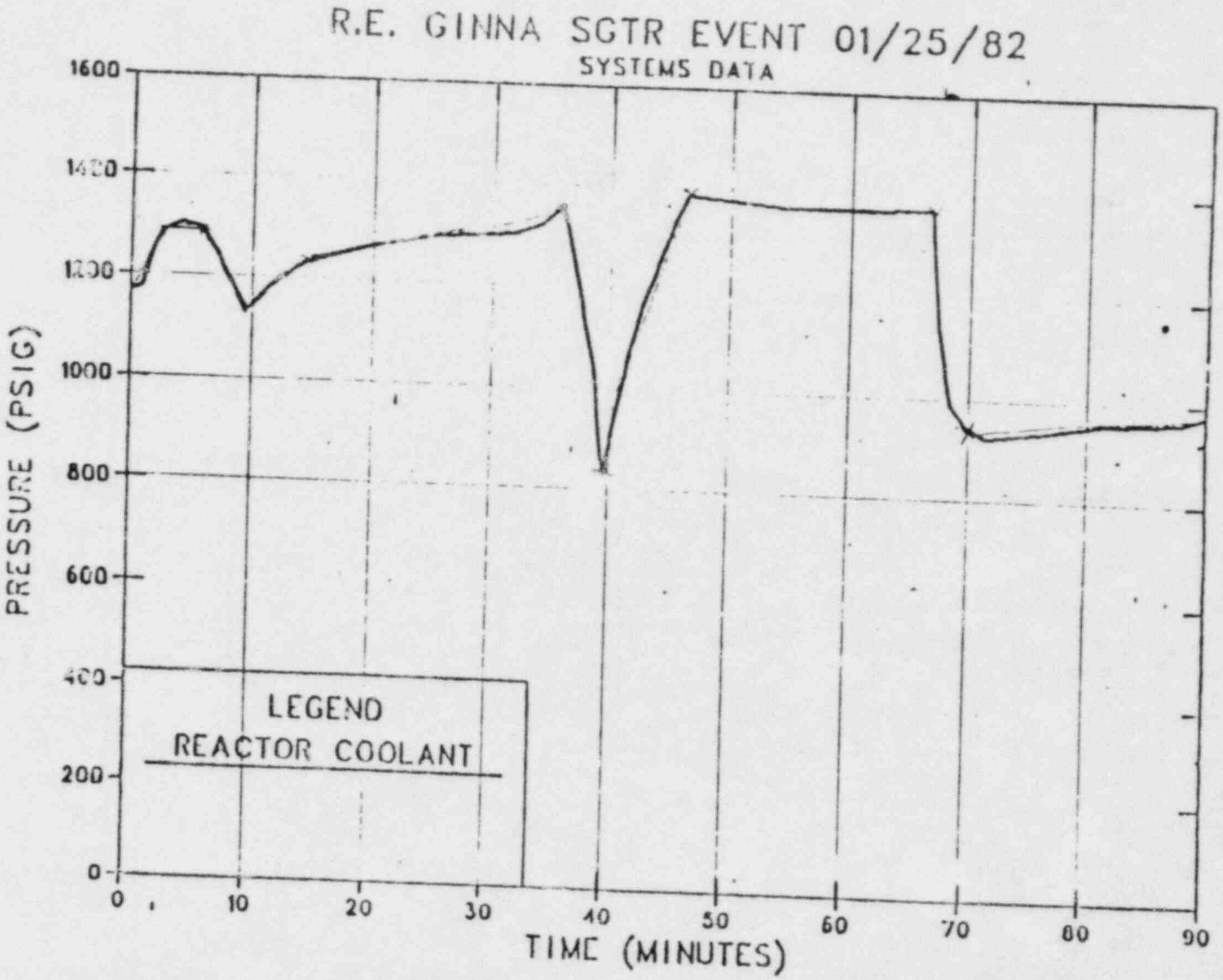
2. NRC PROPOSED SCREENING EVALUATION TRANSIENT IS
   MUCH MORE SEVERE THAN ACTUAL TRANSIENTS.

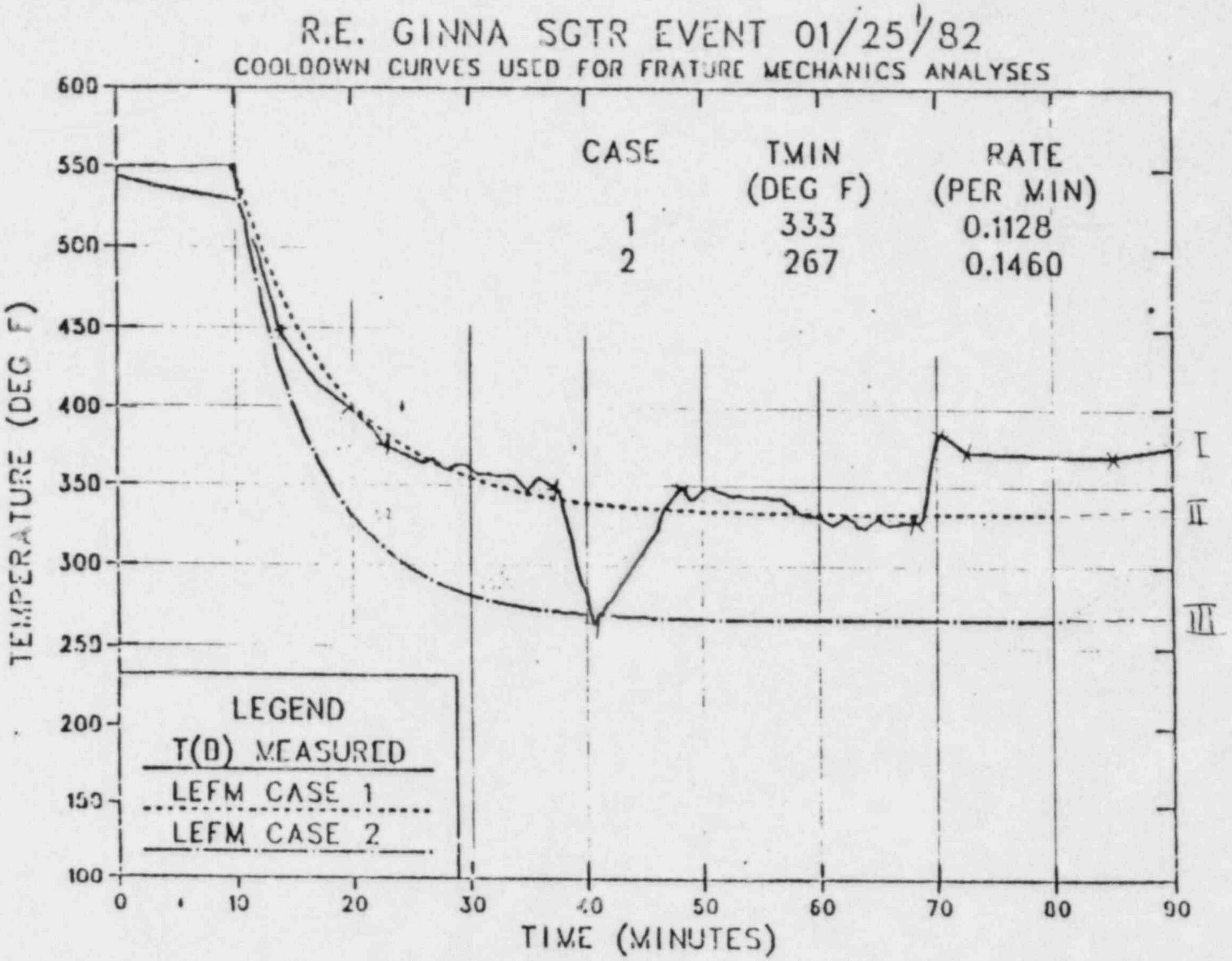
3. IF SCREENING EVALUATION IS TO BE USED IT SHOULD
   BE BASED ON APPROPRIATE ASSUMPTIONS.

TRANSIENT EVALUATIONS

TRANSIENT	RT_{NDT} @ $K_I = K_{IC}$*
GINNA	
I ACTUAL	360
II $T_F = 335$	360
III $T_F = 265$	315
RANCHO SECO	315
NRC EXPONENTIAL	
$T_F = 250$, $P = 2500$	

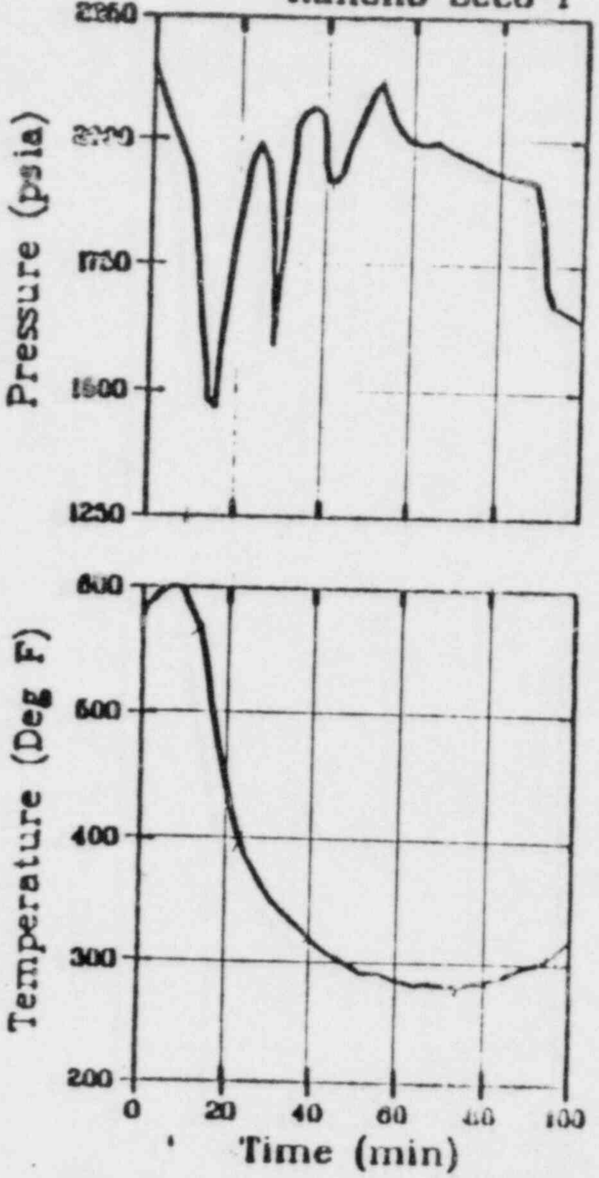
* ALL TRANSIENTS EXPERIENCE WARM PRESTRESS CONDITIONS PRIOR TO $K_I = K_{IC}$ SO THAT $K_I = K_{IC}$ DOES NOT RESULT IN CRACK INITIATION.


R.E. GINNA SGTR EVENT 01/25/82
SYSTEMS DATA
PRESSURE (PSIG)
1600
1400
1200
1000
800
600
400
200
0
TIME (MINUTES)
0
10
20
30
40
50
60
70
80
90
LEGEND
REACTOR COOLANT



R.E. GINNA SGTR EVENT 01/25/82
COOLDOWN CURVES USED FOR FRATURE MECHANICS ANALYSES
CASE
TMIN (DEG F)
RATE (PER MIN)
1
333
0.1128
2
267
0.1460
TEMPERATURE (DEG F)
TIME (MINUTES)
LEGEND
T(D) MEASURED
LEFM CASE 1
LEFM CASE 2
I
II
III

Rancho Seco 1 03/20/78 Overcooling Transient



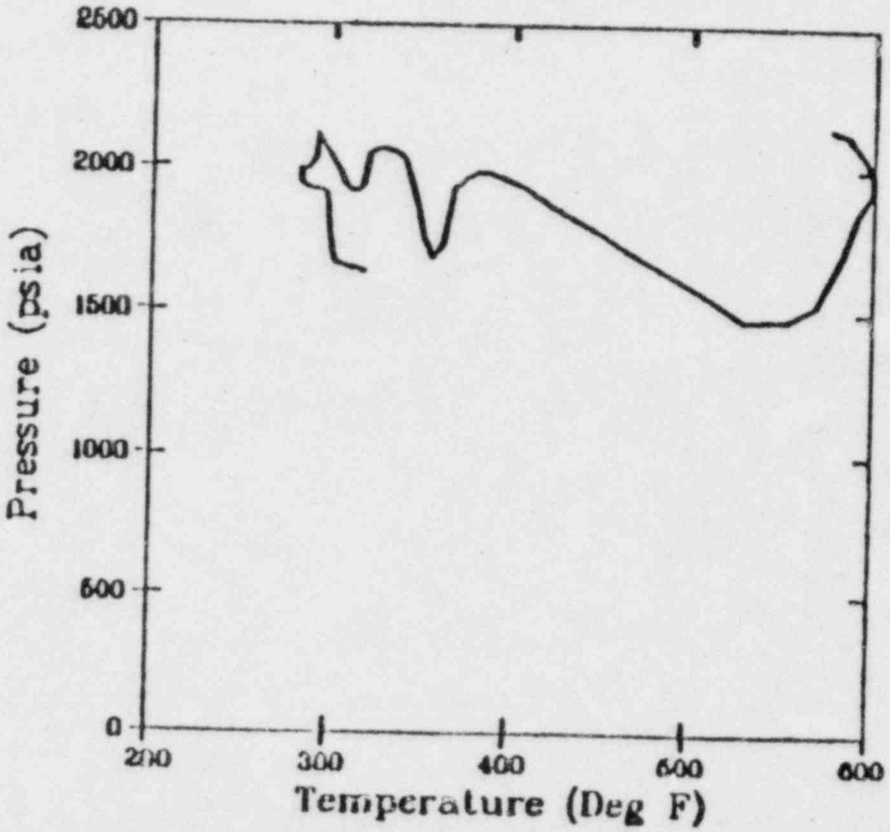
NOTES:

B&W NSSS

Loss of NNI/ICS Indicators

SG refill with MFW

Pressure Controlled with SI



FRACTURE MECHANICS
ANALYSIS ASSUMPTIONS

LE

PARAMETER	VALUE	EFFECT
1. CRACK SIZE	LONG AXIAL	SMALL
2. HEAT TRANSFER	300 BTU/HR $FT^2 {}^oF$	MODERATE
3. ELASTIC MATERIAL PROPERTIES	ASME CODE	SMALL
4. K_{IC}	ASME SECTION XI	--
5. RT_{NDT} SHIFT	REG. GUIDE 1.99 UPPER LIMIT	MODERATE
6. FLUENCE ATTENUATION	EXPONENTIAL	
7. CLADDING	STRESS FREE AT 550F	SMALL
8. LENGTH OF TRANSIENT	90 MIN	MODERATE

ANALYSIS OF NRC TRANSIENT

DURATION OF TRANSIENT (MINUTES)	RT_{NDT} AT $K_I = K_{IC}$ 6/9	RT_{NDT} AT $K_I = K_{IC}$ 6/23
30	320	300
60	290	270
90	280	260

MARGIN IN EXISTING ANALYSES

1. WARM PRESTRESSING ACTS IN REAL TRANSIENTS

2. DUCTILE CRACK ARREST PREVENTS LOSS OF REACTOR VESSEL INTEGRITY
   MARGIN AVAILABLE WILL BE QUANTIFIED BY PRESENTLY PLANNED DEVELOPMENT PROGRAMS

CONCLUSIONS

1. FRACTURE MECHANICS ANALYSES OF ACTUAL TRANSIENTS SIGNIFICANTLY DIFFERENT RESULTS FROM NRC PRESSURE TEMPERATURE ENVELOPE

2. SOME DISAGREEMENT STILL EXISTS ON ASSUMPTIONS FOR FRACTURE MECHANICS ANALYSES

Dear Earles

SUMMARY OF C-E NSSS DESIGN FEATURES

- LOW SHUTOFF HEAD HPSI PUMPS ON ALL C-E PLANTS BUT ONE 1300/1500 psi 100°F/hr
- LOW CAPACITY CHARGING PUMPS 3 pumps 44 g / 77 g
- LARGE S. G. MASS INVENTORY AT TIME OF AFW INITIATION Long cooldown
- AFW FLOW INITIATION (TIME DELAYS, AUTOMATIC ISOLATION)
- MSIS ON LOW STEAM GENERATOR PRESSURE
- SEPARATION OF SAFETY AND CONTROL SYSTEMS
- PRESSURIZER PORVs OPEN COINCIDENT WITH REACTOR TRIP ON HIGH PRESSURIZER PRESSURE
- REACTOR REGULATING SYSTEM DOES NOT AUTOMATICALLY WITHDRAW CONTROL RODS

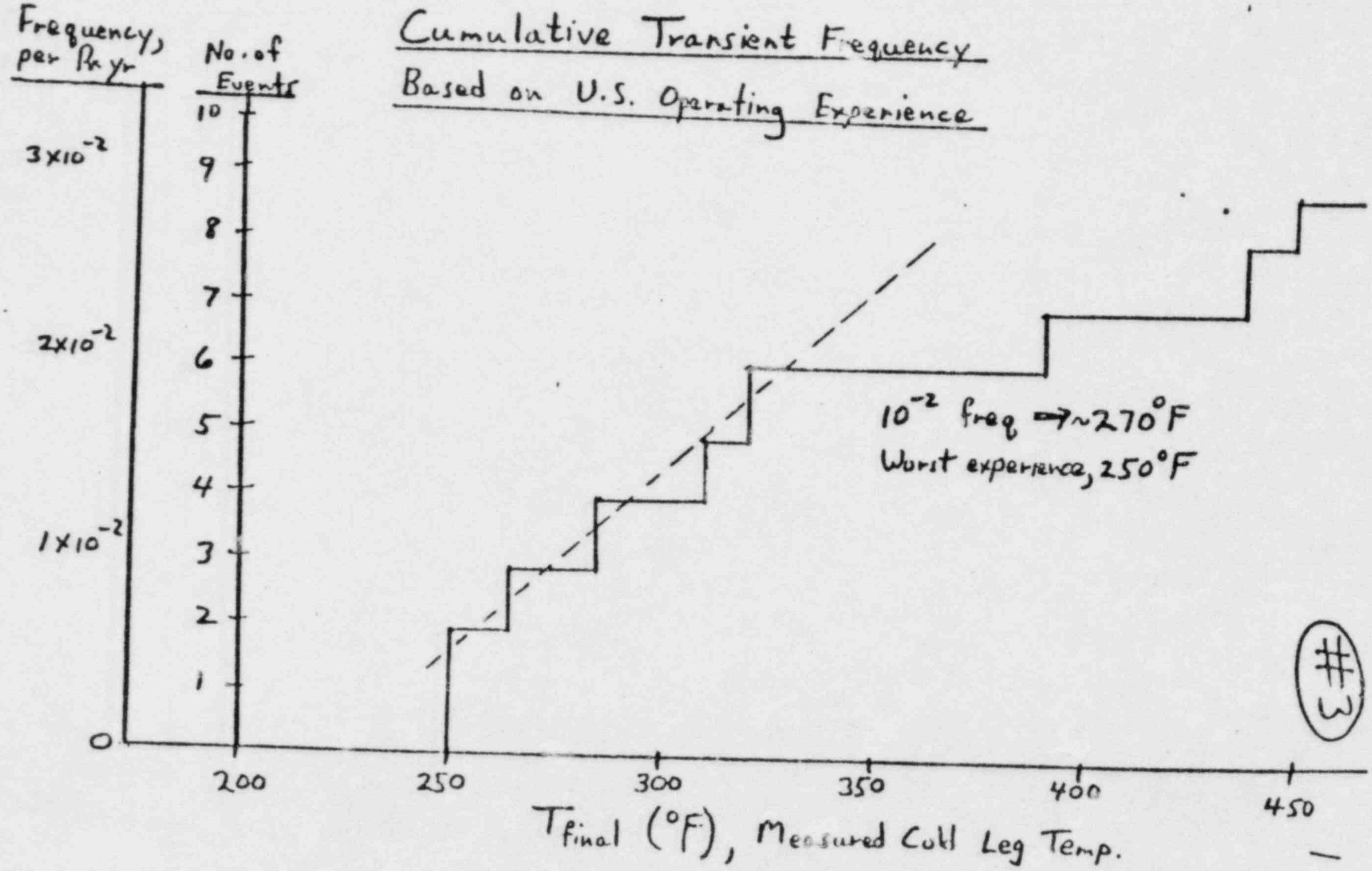
COMBUSTION ENGINEERING
PROBABILISTIC RISK ASSESSMENT
APPROACH TO
PRESSURIZED THERMAL SHOCK

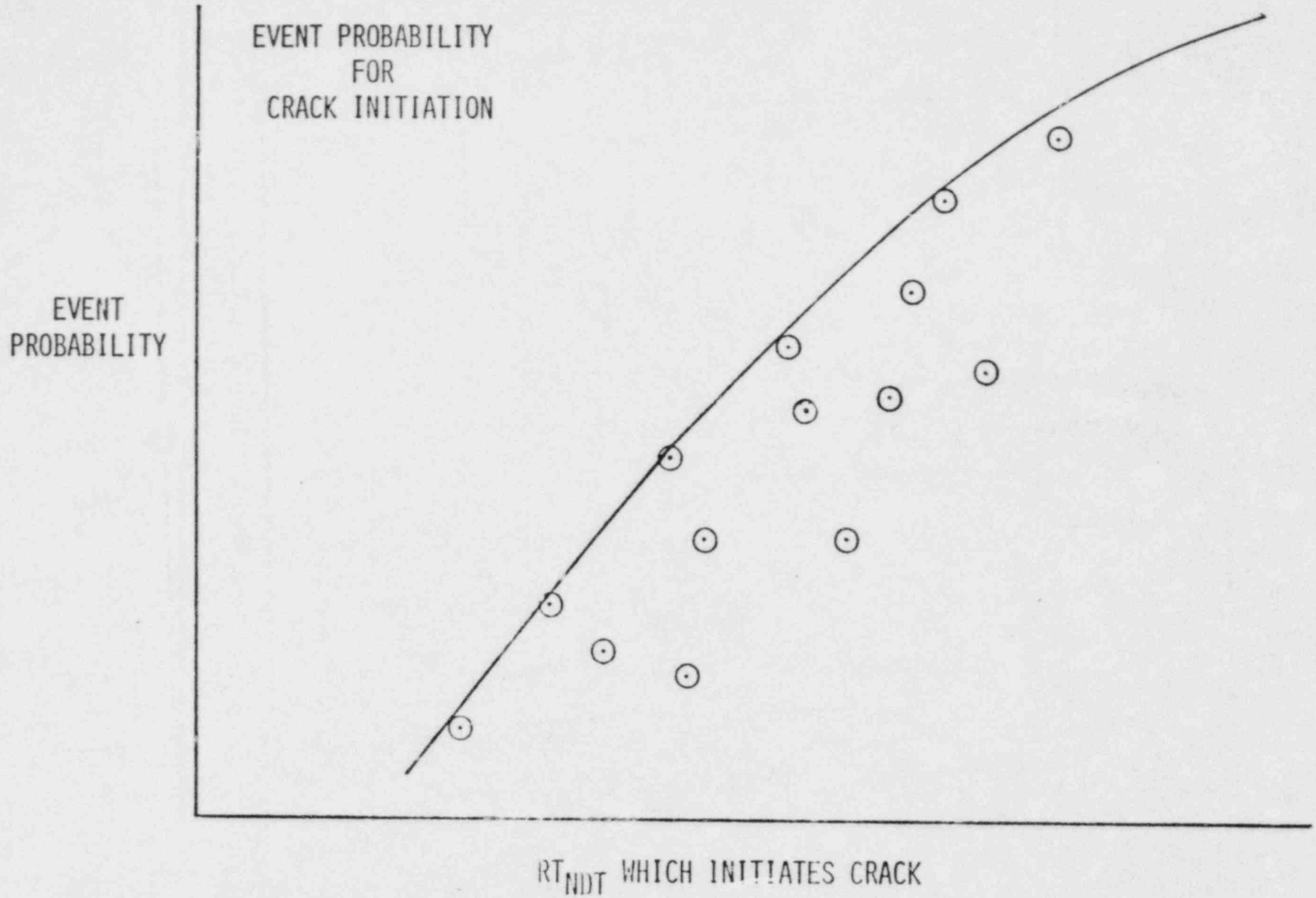
J. J. HERBST

JUNE 23, 1982

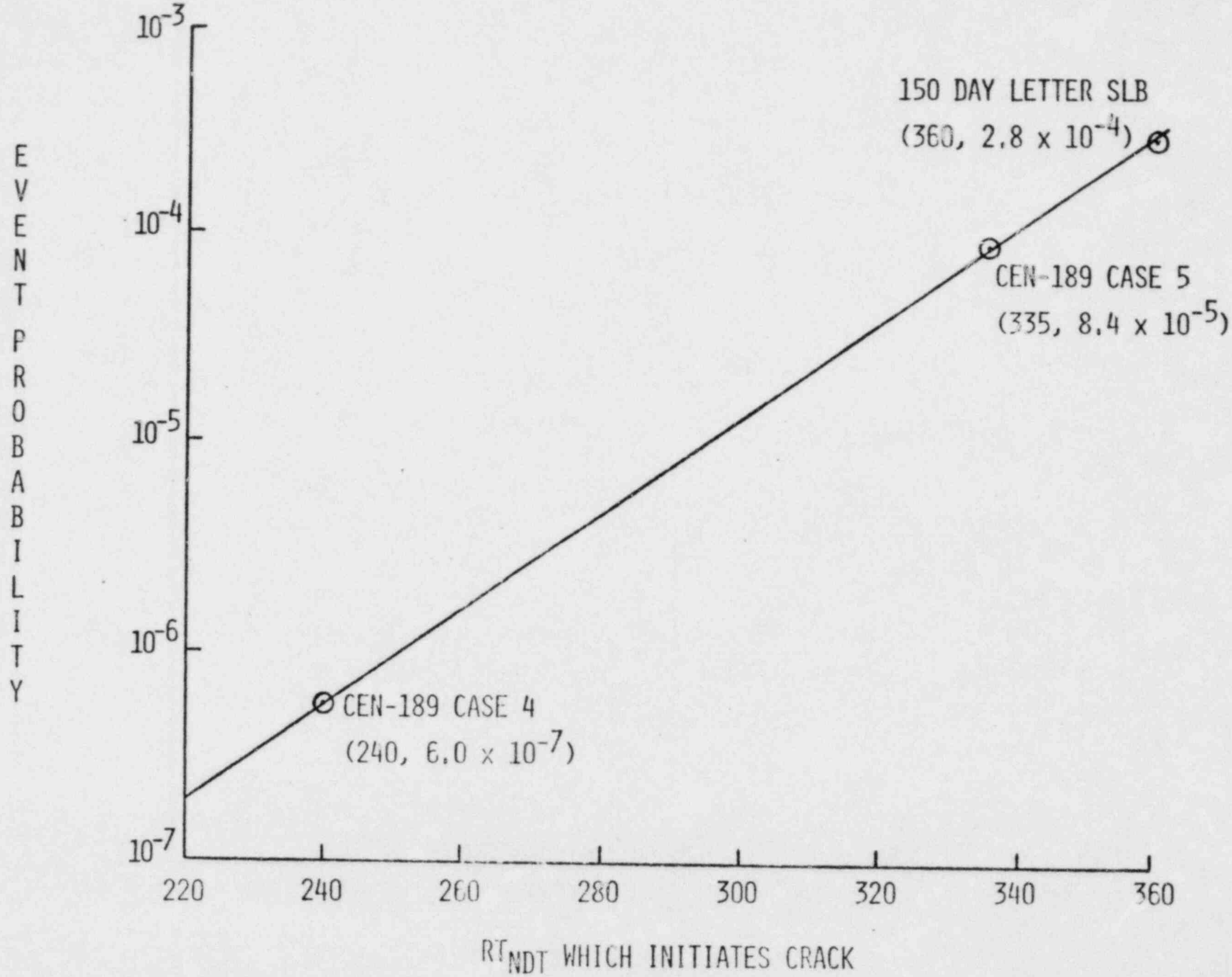
NRC PROBABILITY APPROACH FOR PTS

PROBABILITY OF CRACK INITIATION PER R/Y FOR SEVERE TRANSIENTS	=	PROBABILITY OF TRANSIENT	x	PROBABILITY OF CRACK INITIATION GIVEN TRANSIENT
$\sim 10^{-7}$	=	$\sim 10^{-2}$	x	$\sim 10^{-5}$


Cumulative Transient Frequency
Based on U.S. Operating Experience
Frequency, per Rx yr
No. of Events
3x10^-2
2x10^-2
1x10^-2
0
10
9
8
7
6
5
4
3
2
1
200
250
300
350
400
450
10^-2 freq → ~270°F
Worst experience, 250°F
T_final (°F), Measured Cold Leg Temp.
#3


EVENT PROBABILITY
FOR
CRACK INITIATION
EVENT
PROBABILITY
RT$_{NDT}$ WHICH INITIATES CRACK

$K_I = K_{IC}$



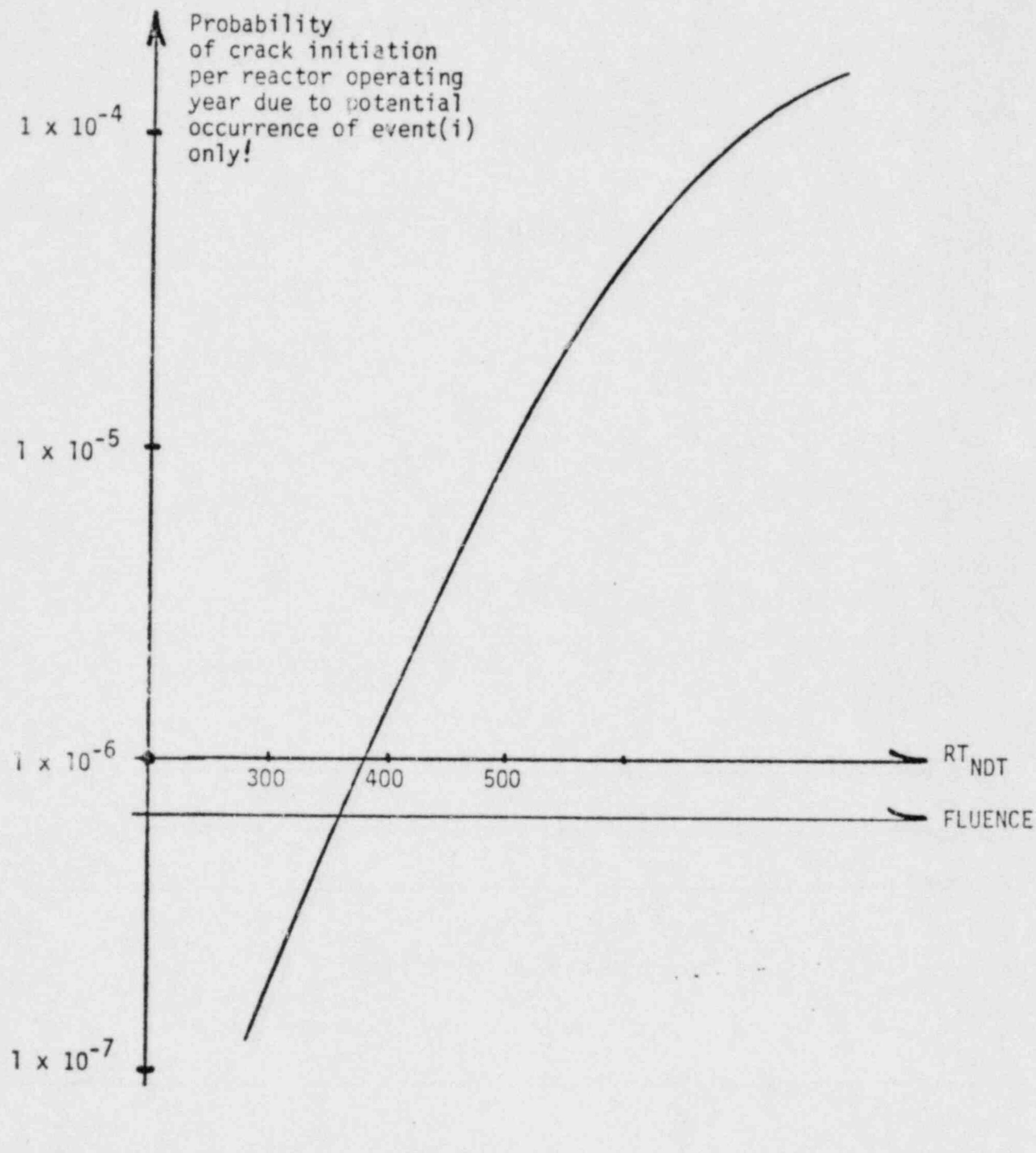
EVENT PROBABILITY
10^{-3}
10^{-4}
10^{-5}
10^{-6}
10^{-7}
150 DAY LETTER SLB
(360, 2.8×10^{-4})
CEN-189 CASE 5
(335, 8.4×10^{-5})
CEN-189 CASE 4
(240, 6.0×10^{-7})
220
240
260
280
300
320
340
360
RT_{NDT} WHICH INITIATES CRACK

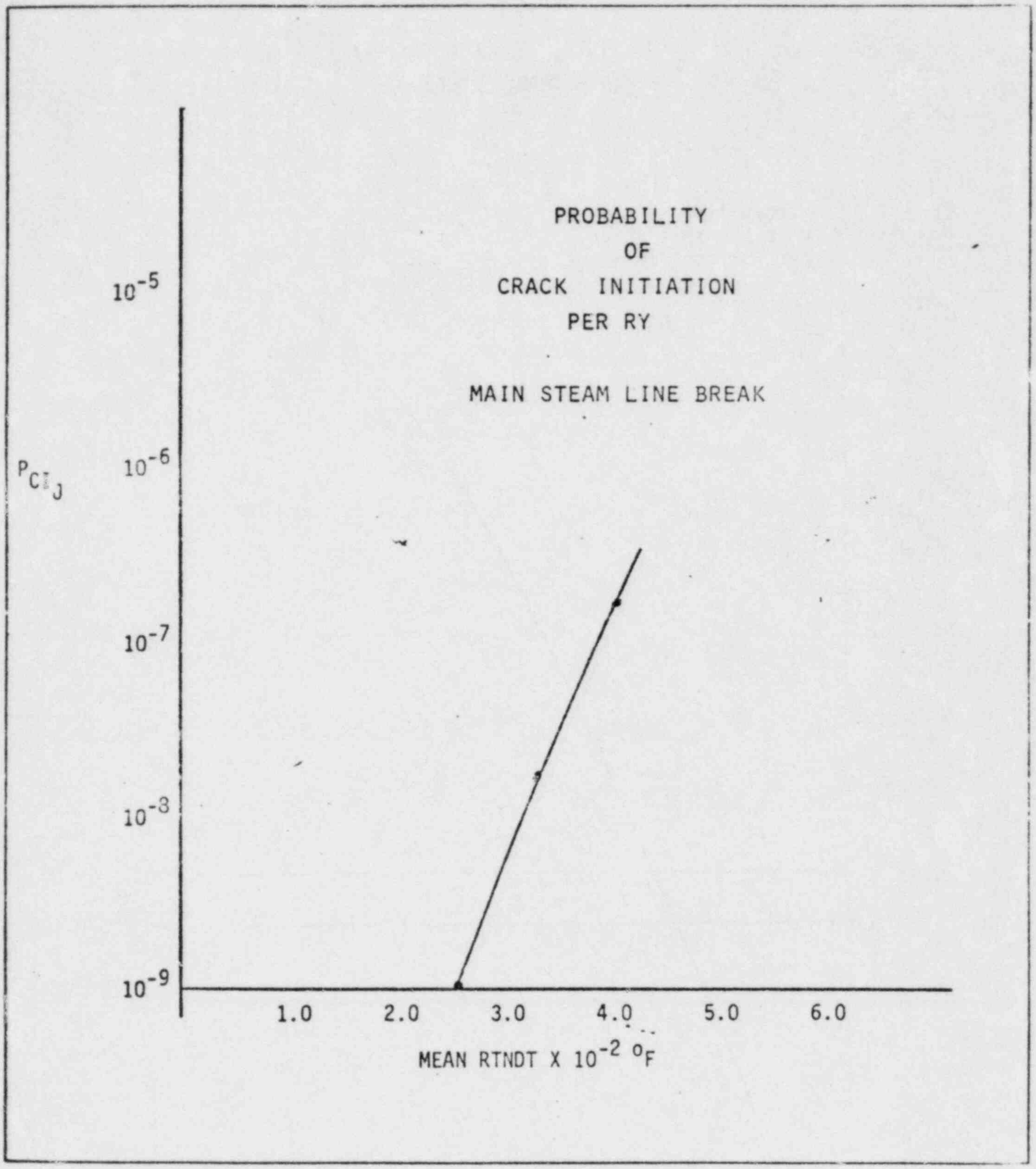
C-E RECOMMENDATION FOR PTS PRA APPROACH

$$\text{PROBABILITY OF INITIATING CRACK PER RY} = \sum_{i} \left[\text{PROBABILITY OF EVENT } i \times \text{PROBABILITY OF CRACK INITIATION GIVEN } i \right]$$

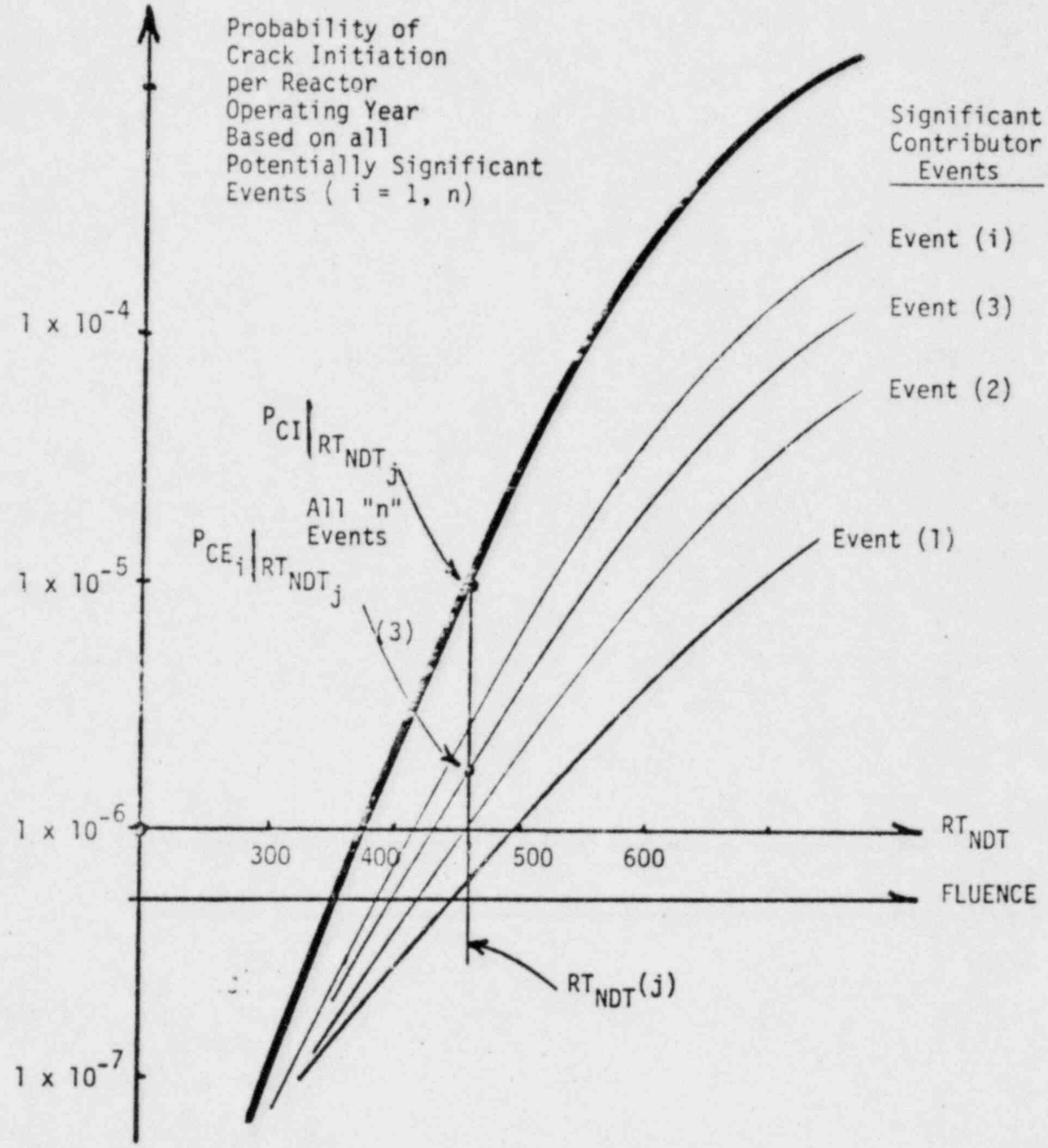
COMPARED TO SAFETY GOAL.

CRACK INITIATION PROBABILITY
(ARBITRARY SCALE)




PROBABILITY
OF
CRACK INITIATION
PER RY
MAIN STEAM LINE BREAK
P_{CI_J}
10^{-5}
10^{-6}
10^{-7}
10^{-8}
10^{-9}
1.0
2.0
3.0
4.0
5.0
6.0
MEAN RTNDT X 10^{-2} ^oF

OVERALL PROBABILITY OF CRACK INITIATION AS A FUNCTION OF RT_{NDT} OR FLUENCE
(Arbitrary Scale)



ADVANTAGES OF C-E APPROACH

- RECOGNIZES PLANT SPECIFIC DIFFERENCES
- ALLOWS CREDIT FOR DESIGN MODIFICATIONS
- DIRECT INPUT INTO SAFETY GOAL COMPARISONS
- PRESENT FRAMEWORK FOR COST/BENEFIT ANALYSIS FOR PROPOSED PTS ACTION

POTENTIAL EVENT SEQUENCES
FOR
PRESSURIZED THERMAL SHOCK

J. J. HERBST
JUNE 23, 1982

C-E PLANT MITIGATION FEATURES FOR NRC SELECTED EVENTS

OPERATIONAL EVENT	MITIGATING FEATURES OF C-E NSSS REACTORS
H. B. Robinson 2 4/28/70 Condition 2 Non-Isolable Steam Line Break During Hot Functionals	1. AFW Termination to Affected Steam Generator. 2. "Low" Head High Pressure SI Pumps. 3. "Low" Capacity Charging (Make-Up) System.
H. B. Robinson 2 11/5/72 Condition 2 Stuck Open Atmospheric Dump Valve	1. AFW Termination to Affected Steam Generator. 2. "Low" Head High Pressure SI Pumps. 3. "Low" Capacity Charging (Make-Up) System. 4. ADV Not Automatically Controlled.
Rancho Seco 3/20/78 Condition 1 Loss of Power to ICS Resulting First in Loss of and then in Excess Feedwater Flow	1. Large S. G. Inventory. 2. Independent MFW Control Systems for Each S. G. 3. "Low" Head High Pressure SI Pumps. 4. "Low" Capacity Charging (Make-Up) System. 5. Separation of Control and Safety Systems.
H. B. Robinson 2 5/1/75 Condition 1 Small Break LOCA - RCP Seal Failure	1. "Low" Head High Pressure SI Pumps. 2. RCP Restart for Boration Not Required.
Three Mile Island 2 3/28/79 Condition 1 Small Break LOCA - PORV Stuck Open	1. S. G. Inventory. 2. Turbine Bypass Valve Size. 3. "Low" Head High Pressure SI Pumps.
Crystal River 3 2/26/80 Condition 1 Loss of Power to ICS Resulting in PORV Opening, Partial Loss of Feedwater and Rod Withdrawal	1. Large S. G. Inventory. 2. "Low" Head High Pressure SI Pumps. 3. Control Rod Withdrawal Prohibit. 4. Separation of Control and Safety Systems. 5. PORVs Actuated on High RCS Pressure Trip.
R. E. Ginna 1/5/82 Condition 1 Steam Generator Tube Rupture - Stuck Open PORV	1. PORVs Not Specified for RCS Depressurization. 2. "Low" Head High Pressure SI Pumps. 3. Main Feedwater Flow Rampback to 5% on Reactor Trip. 4. MSIS Occurs on Low S. G. Pressure.

C-E POWER SYSTEMS

System Status - vs. Plant Operating States

SYSTEMS	SYSTEM RELATED PLANT OPERATING STATES			
	CONDITION 1	CONDITION 2	CONDITION 3	CONDITION 4
TURBINE	A	B	D	D
TURBINE BYPASS	C	B	A OR B	B
ATMOSPHERIC DUMP VALVES	E	B	B	B
CHARGING	A	A	A	A
LETDOWN	A	A	A	A
MAIN PRESSURIZER SPRAYS	A	A	A	A OR B
AUXILIARY PRESSURIZER SPRAYS	D	D	D	B OR D
ENGINEERED SAFETY FEATURES ACTUATION SYSTEM	C	C	C	C OR E
PORV A/ PORV B	C *	C *	C *	C
REACTOR REGULATING SYSTEM	C INSERT E WITHDRAWAL	E INSERT B WITHDRAWAL	E E	F F
MAIN FEEDWATER	A	B	D	D
MAIN FEEDWATER BYPASS	C	B	D	D
AUXILIARY FEEDWATER	C	B **	B	B
SHUTDOWN COOLING SYSTEM	F	F	F	B
MAIN STEAM ISOLATION VALVES	OPEN	OPEN	OPEN	OPEN
MAIN FEEDWATER ISOLATION VALVES	OPEN	OPEN/CLOSED	CLOSED	CLOSED

GLOSSARY:
- A - OPERATING UNDER AUTO. CONTROL
- B - OPERATING UNDER MANUAL CONTROL
- C - STANDBY - CONTROL/ACTUATION SYSTEM AUTOMATIC
- D - STANDBY - COMPLEX MANUAL PROCEDURE TO ACTUATE
- E - STANDBY - SIMPLE MANUAL ACTION TO ACTUATE
- F - OUT-OF-SERVICE

* PORV B BLOCKED

** BOTH MANUAL AND AUTOMATIC AFW ISOLATION CASES ARE EVALUATED WHILE THE SYSTEM IS UNDER MANUAL CONTROL

C-E POWER SYSTEMS

Initiating Events vs. Plant Operating States

INITIATING EVENTS	PLANT OPERATING STATE			
	1	2	3	4
1. Decrease in Feedwater Enthalpy				
a) MFW Heater System Failure	X	X		
b) AFW Delivery	X			
2. Excess Feedwater Flow				
a) MFW Flow Increases Above That for a Particular Power Level	X	X		
b) AFW Flow Increases Above That for a Particular Power Level		X	X	X
c) Reactor Power Decreases and MFW Flow Fails to Adjust	X	X		
d) Reactor Power Decreases and AFW Flow Fails to Adjust		X		
e) Turbine Trip and MFW Fails to Rampback	X			
3. Excess Steam Flow				
• Steam Flow Increases Above That Req'd for a Particular Power Level				
a) Steam Line Break	X	X	X	X
b) MFW Line Break d/s of Check Valve	X	X	X	X
c) ADV Inadvertently Opens	X	X	X	
d) TBV Inadvertently Opens	X	X	X	
e) MSSV Inadvertently Opens	X	X	X	X
f) Excess Steam Flow Through Turbine	X	X		
• Reactor Power Decreases and Steam Flow Fails to Adjust				
g) ADV is Open and Fails to Close		X	X	X
h) TBV is Open and Fails to Close		X	X	X
i) Turbine Fails to Decrease Steam Flow	X	X		
j) Reactor Trips and Turbine Fails to Trip	X	X		
k) Reactor Trips, Turbine Trips, and TBV Fails to Close After Quick Open or During Modulation	X	X		
l) Hi Pressure Transient, MSSV Opens and Fails to Close	X	X	X	

CE POWER SYSTEMS

Initiating Events vs. Plant Operating States

INITIATING EVENTS	PLANT OPERATING STATE			
	1	2	3	4
4. Large LOCA				
a. Large Pipe Break	X	X	X	X
5. Small LOCA				
a) Non-Isolable Pipe Break	X	X	X	X
b) Isolable Pipe Break (Letdown Line)	X	X	X	X
c) RCP Seal Failure	X	X	X	X
d) PORV Inadvertently Opens	X	X	X	X
e) RCS Overpressure Scenario and One PORV/PSV Fails to Reclose	X	X	X	X
f) SG Tube Rupture	X	X	X	X
6. Pressurizer Pressure Control Failures*				
a) Spurious Main Spray Actuation	X	X	X	X
b) Spurious Aux. Spray Actuation+				
c) Pressure Transient Actuates Main Spray, Spray Fails to Decrease	X	X	X	X
d) Excess Main Spray During a Controlled Depressurization or Boron Mixing	X	X	X	X
e) Excess Aux. Spray During a Controlled Depressurization				X
7. Inadvertent SIAS When Below Shutoff Head				
a) Failure to Block SIAS Setpoint				X
b) Spurious SIAS				X

\+ Spurious auxiliary spray actuation is not considered plausible due to complexity of actuation procedure.

* Pressurizer control system failures which result in over-pressure transients are included in Category 5. This category refers to RCS de-pressurization events.

Initiating Events vs. Plant Operating States

INITIATING EVENTS	PLANT OPERATING STATE			
	1	2	3	4
8. Decrease in Charging Enthalpy				
a) PLCS Failure (Max. Charging, Loss of Letdown)	X	X	X	X
9. Maximum Shutdown Cooling				X

Event Tree Branch Descriptions

EVENT CODE	DESCRIPTION
MS	MAIN STEAM LINE BREAK OUTSIDE CONTAINMENT, UPSTREAM OF MSIV B. OCCURS DURING PLANT CONDITION 2.
PV	PORV INADVERTENTLY OPENS DURING PLANT CONDITION 1.
MV	A HIGH PRESSURE TRANSIENT ON THE SECONDARY SIDE CAUSES THE MSSVS TO LIFT. ONE MSSV ON SG B FAILS TO RESEAT. OCCURS DURING PLANT CONDITION 2.
MF	REACTOR/TURBINE TRIP AND MFW FAILS TO RAMPBACK DURING PLANT CONDITION 1.
TT	THE REACTOR TRIPS AND THE TURBINE FAILS TO TRIP DURING PLANT CONDITION 1.
A	THE MSIS IS NOT GENERATED DUE TO AN ACTUATION LOGIC FAILURE.
B	MSIV A FAILS TO CLOSE AUTOMATICALLY.
C	MSIV B FAILS TO CLOSE AUTOMATICALLY.
D	THE AFW SYSTEM IS IN MANUAL AND THE OPERATOR FAILS TO ISOLATE AFW FLOW TO THE RUPTURED SG WITHIN 5 MINUTES AFTER THE INITIATING EVENT.
E	THE AFW SYSTEM IS IN MANUAL AND THE OPERATOR FAILS TO TERMINATE AFW FLOW TO THE INTACT SG WITHIN 15 MINUTES AFTER THE INITIATING EVENT.
F	THE OPERATOR FAILS TO THROTTLE HPSI FLOW WITHIN 10 MINUTES AFTER THE INITIATING EVENT.
G	AFW FLOW IS LOST TO SG A BETWEEN 10 MINUTES AND 1 HOUR AFTER THE INITIATING EVENT.
H	THE TURBINE FAILS TO TRIP ON REACTOR TRIP. MSIS GENERATED.
I	MFW FAILS TO RAMPBACK ON TURBINE TRIP.
J	ONE MFW PUMP DISCHARGE VALVE FAILS TO CLOSE AUTOMATICALLY ON HI SG LEVEL.
K	ONE MFW PUMP FAILS TO TRIP ON HI-HI SG LEVEL.
L	THE OPERATOR FAILS TO CLOSE THE PORV BLOCK VALVE WITHIN 30 MINUTES AFTER THE PORV INADVERTENTLY OPENS.
M	THE OPERATOR FAILS TO THROTTLE HPSI FLOW WITHIN 30 MINUTES TO AN HOUR AFTER THE INITIATING EVENT.
N	MFW BYPASS FLOW IS LOST BETWEEN 10 MINUTES AND 1 HOUR AFTER THE INITIATING EVENT
O	FAILURE TO AUTOMATICALLY DELIVER AFW FOLLOWING LOSS OF MFW BYPASS.
P	LOSS OF AFW FLOW FOLLOWING SAFETY INJECTION UP TO ONE HOUR AFTER SAFETY INJECTION.
Q	OPERATOR FAILS TO DECREASE AFW FLOW WITHIN 30 MINUTES AFTER THE REACTOR TRIPS.
R	OPERATOR FAILS TO ISOLATE THE LETDOWN LINE BREAK.
S	MFIV A FAILS TO CLOSE AUTOMATICALLY
T	MFIV B FAILS TO CLOSE AUTOMATICALLY
U	AFAS A FAILS TO PREVENT AFW DELIVERY TO RUPTURED SG A DUE TO AN ACTUATION LOGIC FAILURE
V	AFAS B FAILS TO PREVENT AFW DELIVERY TO RUPTURED SG B DUE TO AN ACTUATION LOGIC FAILURE.

POWER SYSTEMS

Scenario Occurrence Frequency Categorization

Condition 1 Events

CATEGORY	MODERATE	INFREQUENT	LIMITING FAULT 1	LIMITING FAULT 2	LIMITING FAULT 3	VERY LOW FREQUENCY
FREQUENCY	MAY OCCUR DURING A CALENDAR YEAR	MAY OCCUR DURING A PLANT LIFETIME	LOW PROBABILITY OF OCCURRING DURING A PLANT LIFETIME	VERY LOW PROBABILITY OF OCCURRING DURING A PLANT LIFETIME	EXCEEDINGLY LOW PROBABILITY OF OCCURRING DURING A PLANT LIFETIME	
EVENT SEQUENCES			PV MF	PV-Q PV-M PV-L MF-M MF-Q	PV-MQ PV-LQ PV-LM PV-LMQ PV-I TT TT-Q TT-M MF-QM	PV-P PV-N PV-NQ PV-MN PV-LN PV-IQ PV-IM PV-IMQ PV-IL PV-ILQ PV-ILM PV-H MF-J MF-JM MF-JQ TT-MQ TT-T TT-TQ TT-TM TT-S TT-SQ TT-SM

POWER SYSTEMS

Scenario Occurrence Frequency Categorization

Condition 2 Events - Manual AFW Isolation

CATEGORY	MODERATE	INFREQUENT	LIMITING FAULT			VERY LOW FREQUENCY
			1	2	3	
FREQUENCY	MAY OCCUR DURING A CALENDAR YEAR	MAY OCCUR DURING A PLANT LIFETIME	LOW PROBABILITY OF OCCURRING DURING A PLANT LIFETIME	VERY LOW PROBABILITY OF OCCURRING DURING A PLANT LIFETIME	EXCEEDINGLY LOW PROBABILITY OF OCCURRING DURING A PLANT LIFETIME	
EVENT SEQUENCES			MV-D MV-DF MV-DE MV-DEF	MV MV-F MV-E MV-EF MS-DEF	MS MS-E MS-F MS-EF MS-D MS-DF MS-DE	MV-DG MV-DFG MV-CD MV-CDF MV-CDE MV-CDEF MV-BD MV-BDF MV-BDE MV-BDEF MV-AD MV-ADF MV-ADE MV-ADEF

POWER SYSTEMS

Scenario Occurrence Frequency Categorization

Condition 2 Events - Automatic AFW Isolation

CATEGORY	MODERATE	INFREQUENT	LIMITING FAULT			VERY LOW FREQUENCY
			1	2	3	
FREQUENCY	MAY OCCUR DURING A CALENDAR YEAR	MAY OCCUR DURING A PLANT LIFETIME	LOW PROBABILITY OF OCCURRING DURING A PLANT LIFETIME	VERY LOW PROBABILITY OF OCCURRING DURING A PLANT LIFETIME	EXCEEDINGLY LOW PROBABILITY OF OCCURRING DURING A PLANT LIFETIME	
			MV MV-F MV-E MV-EF	MS-F MS-EF	MS MS-E	MV-BD MV-BDF MV-BDE MV-BDEF MV-AD MV-ADF MV-ADE MV-ADEF

Scenario Occurrence Frequency Categorization

Most Severe Sequences - Manual AFW Isolation for Condition 2 Sequences

CATEGORY	MODERATE	INFREQUENT	LIMITING FAULT			VERY LOW FREQUENCY
			1	2	3	
FREQUENCY	MAY OCCUR DURING A CALENDAR YEAR	MAY OCCUR DURING A PLANT LIFETIME	LOW PROBABILITY OF OCCURRING DURING A PLANT LIFETIME	VERY LOW PROBABILITY OF OCCURRING DURING A PLANT LIFETIME	EXCEEDINGLY LOW PROBABILITY OF OCCURRING DURING A PLANT LIFETIME	
CONDITION 1 EVENT SEQUENCES			PV	PV-M MF-Q	PV-MQ MF-QM	
CONDITION 2 EVENT SEQUENCES			MV-DEF	MV-EF MS-DEF	MS-DE	

Scenario Occurrence Frequency Categorization

Most Severe Sequences - Automatic AFW Isolation for Condition 2 Sequences

CATEGORY	MODERATE	INFREQUENT	LIMITING FAULT			VERY LOW FREQUENCY
			1	2	3	
FREQUENCY	MAY OCCUR DURING A CALENDAR YEAR	MAY OCCUR DURING A PLANT LIFETIME	LOW PROBABILITY OF OCCURRING DURING A PLANT LIFETIME	VERY LOW PROBABILITY OF OCCURRING DURING A PLANT LIFETIME	EXCEEDINGLY LOW PROBABILITY OF OCCURRING DURING A PLANT LIFETIME	
CONDITION 1 EVENT SEQUENCES			PV	PV-M MF-Q	PV-MQ MF-QM	
CONDITION 2 EVENT SEQUENCES			MV-EF	(MS-EF)*		

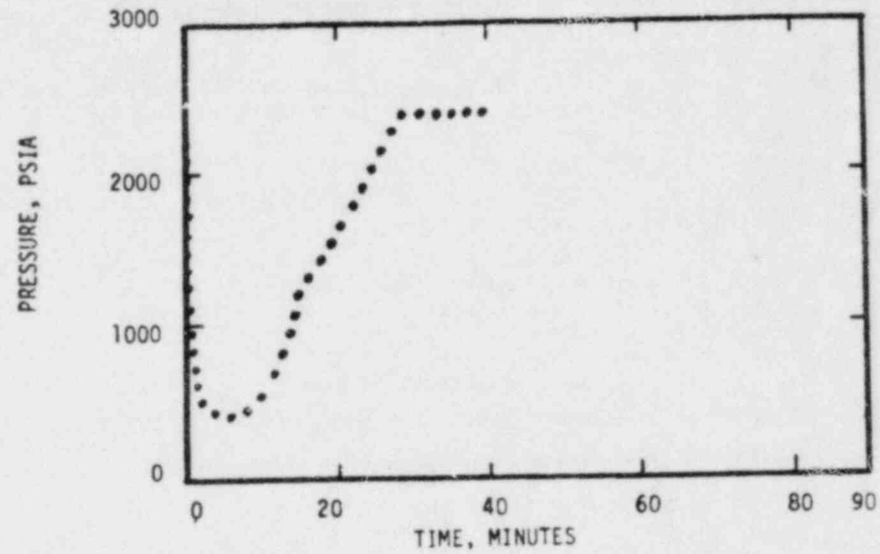
POWER SYSTEMS

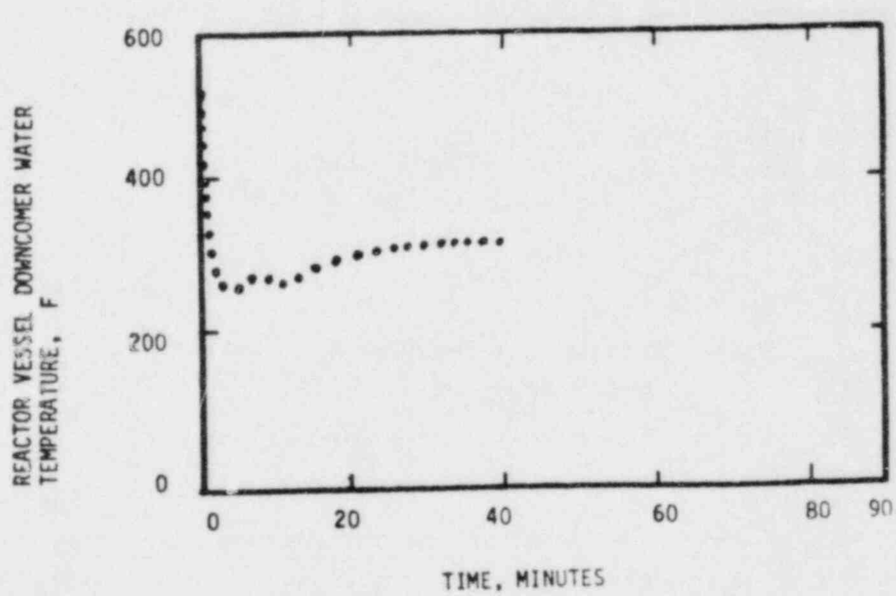
Frequency Categories of Initiating Events

INITIATING EVENTS	FREQUENCY CATEGORIES
1. Decrease in Feedwater Enthalpy	
a) MFW Heater System Failure	I
b) AFW Delivery	LF1
2. Excess Feedwater Flow	
a) MFW Flow Increases Above That for a Particular Power Level	I
b) AFW Flow Increases Above That for a Particular Power Level	LF1
c) Reactor Power Decreases and MFW Flow Fails to Adjust	I
d) Reactor Power Decreases and AFW Flow Fails to Adjust	LF1
e) Turbine Trip and MFW Fails to Rampback	M + Add. Fail. = LF1
3. Excess Steam Flow	
• Steam Flow Increases Above That Req'd for a Particular Power Level	
a) Steam Line Break	LF2
b) MFW Line Break d/s of Check Valve	LF2
c) ADV Inadvertently Opens	LF1
d) TBV Inadvertently Opens	I
e) MSSV Inadvertently Opens	LF1
f) Excess Steam Flow Through Turbine	I
• Reactor Power Decreases and Steam Flow Fails to Adjust	
g) ADV is Open and Fails to Close	LF1
h) TBV is Open and Fails to Close	I
i) Turbine Fails to Decrease Steam Flow	I
j) Reactor Trips and Turbine Fails to Trip	M + Add. Fail. = LF3
k) Reactor Trips, Turbine Trips, and TBV Fails to Close After Quick Open or During Modulation	M + Add. Fail. = I
l) Hi Pressure Transient, MSSV Opens and Fails to Close	I + Add. Fail. = LF1
4. Large LOCA	
a) Large Pipe Break	LF2
5. Small LOCA	
a) Non-Isolable Pipe Break	I
b) Isolable Pipe Break (Letdown Line)	LF1
c) RCP Seal Failure	I
d) PORV Inadvertently Opens	LF1
e) RCS Overpressure Scenario and One PORV/PSV Fails to Reclose	I + Add. Fail. = LF1
f) SG Tube Rupture	I
6. Pressurizer Pressure Control Failures	
a) Spurious Main Spray Actuation	LF1
b) Spurious Aux. Spray Actuation+	
c) Pressure Transient Actuates Main Spray, Spray Fails to Decrease	M + Add. Fail. = I
d) Excess Main Spray During a Controlled Depressurization or Boron Mixing	LF1
e) Excess Aux. Spray During a Controlled Depressurization	LF1
7. Inadvertent SIAS When Below Shutoff Head	
a) Failure to Block SIAS Setpoint	I
b) Spurious SIAS	I
8. Decrease in Charging Enthalpy	
a) PLCS Failure (Max. Charging, Loss of Letdown)	LF1
9. Maximum Shutdown Cooling	LF1
+ Spurious auxiliary spray actuation is not considered plausible due to complexity of actuation procedure.	

900 sec

REFERENCE STEAM LINE BREAK (SLB)





CEN-212

AN ASSESSMENT of the PROBABILITY of OVERCOOLING TRANSIENTS at NUCLEAR PLANTS with COMBUSTION ENGINEERING NUCLEAR STEAM SUPPLY SYSTEM

Prepared for the C-E OWNERS GROUP

NUCLEAR POWER SYSTEMS DIVISION
JUNE, 1982

8207060188

POWER SYSTEMS
COMBUSTION ENGINEERING, INC.

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TABLE OF CONTENTS

LIST OF FIGURES

LIST OF TABLES

1.0 INTRODUCTION

This report describes the results of an assessment of the probability of various over-cooling transient scenarios at nuclear plants which have Combustion Engineering (C-E) Nuclear Steam Supply Systems (NSSS). The assessment was directed specifically towards those scenarios which are expected to provide the greatest challenge to the reactor vessel with respect to the phenomenon known as Pressurized Thermal Shock (PTS).

The assessment consisted of the identification of initiating events with the potential for significant cooling of the reactor vessel, the expansion of the most severe initiators into complete scenarios using event tree analysis, the quantification of the probabilities of those scenarios and the identification of the most limiting scenarios with respect to PTS.

A total of 95 initiating events occurring in four plant operating conditions were considered for their over-cooling potential. Five of these were chosen as having the most challenging cooldowns and were evaluated further using event tree analysis. The event trees yielded approximately 220 scenarios each of which was evaluated to select the most limiting with respect to PTS.

This report takes credit for a number of design features which are considered typical among plants with a C-E designed NSSS. Several of these features are specific only to these plants. The report represents a scoping study of the general topic of over-cooling transients and should not necessarily be considered specific to any C-E plant.

There are some variations in plant response which are considered in the report. For example, as described in Section 3.0, the scenario probabilities for the events occurring in Condition 2 (See Table 2-2) were calculated twice, once with manual isolation of AFW assumed and one with automatic isolation assumed. The effect of including automatic isolation was shown to be significant in that it reduced the occurrence frequencies of several of the most challenging sequences to much less than 10^{-6}.

1.1 PLANT SPECIFIC CHARACTERISTICS

The specific characteristics of a nuclear plant, including those associated both with the NSSS design and with the balance-of-plant, strongly influence the occurrence frequency and the severity of over-cooling events at the plant. Plants which have a C-E designed NSSS contain a number of design features which tend to limit the occurrence frequencies of over-cooling events and which limit their potential for evolving into severe PTS transients. Examples of these features are provided below.

Features which limit the occurrence frequency of over-cooling events initiated on either the primary or secondary side of the plant -

1. Steam generators in the C-E NSSS design contain a large volume of secondary water. This volume of water tends to minimize the sensitivity of the C-E NSSS to minor secondary system perturbations such as feedwater oscillations, etc., that may be expected to occur on a relatively frequent basis. A large fraction of this volume of water is still in the steam generator at the low level setpoint at which the Auxiliary Feedwater System (AFW) is actuated.

2. Turbine by-pass valves are sized to minimize potential challenges to main steam safety valves, power operated relief valves (PORV's) and RCS safety valves.

3. Atmospheric Dump Valves upstream of the Main Steam Isolation Valves (MSIV's) are not normally operated in automatic.

4. The PORV's on operating C-E designed NSSS units are not normally used for manual depressurization. They are actuated automatically only coincident with a high pressure reactor trip, their function being to minimize challenges to RCS safety valves. It is part of C-E's design philosophy to maintain the integrity of the primary system boundary to

the greatest extent possible. This philosophy is reflected in the C-E Emergency Procedure Guidelines. While those guidelines do not preclude using the PORV's to depressurize the plant as a last resort, they do not recommend manual actuation of PORV's until all other methods have been attempted.

5. C-E has also maintained complete separation of control and safety grade systems in the NSSS designs. This separation tends to minimize the impact of events which may be initiated by control grade systems but which may require safety grade systems to respond.

Features which tend to limit the potential for the evolution of over-cooling events into PTS scenarios -

1. A safety grade Low Steam Generator Pressure signal causes reactor trip, main feedwater isolation and main steam system isolation. This would tend to minimize any excessive cooldown transient initiated on the secondary side downstream of the MSIV's and minimize the potential of over-cooling due to excessive feed events.

2. Prudent design philosophy has been followed by older C-E plants implementing automatic auxiliary feedwater initiation systems (Automatic initiation required by TMI Action Plan). This philosophy has led to features such as staggered AFW pump starts, time delays in AFW pump starts and automatic isolation of depressurized steam generators. These systems do not all exist on all plants and some are still in the process of being upgraded to safety grade status in accordance with the TMI action plan. All of these features, however, tend to minimize the cooldown from excessive AFW flow, particularly under conditions where an over-cooling event is already in progress.

3. The high pressure safety injection pumps on C-E designed operating plants, with one exception, are "low head" high pressure safety injection pumps. While this operating characteristic of these pumps

would not prevent complete repressurization in all rapid cooldown events, (operator action would be required to turn off the low capacity positive displacement charging pumps) these lower head pumps would give the operator significantly more time to control the plant response once he had established subcooling.

These features have been considered in the analysis described in this report, particularly in the identification of the most limiting over-cooling transients and in defining the structure of the event trees.

1.2 OPERATING EXPERIENCE

The NRC has identified seven events which it considers to be sufficiently severe over-cooling transients to qualify as potential PTS precursors. A review of C-E plant operating history failed to identify any events resulting in comparable cooldowns.

Table 1-1 illustrates some of the C-E plant design features which tend to minimize the severity of the cooldown or the repressurization associated with events of the types represented by the NRC selected transients.

2.0 PROCEDURE

The development of a complete list of potential over-cooling scenarios involves the identification of relevant event initiators and the evaluation of each one, using, for example, event tree analysis techniques to define the relevant sequences of additional failures. This is essentially the process that was used in this evaluation. However, the development of event sequences and the construction of event trees was focused on those few initiating events with the greatest potential for evolving into PTS scenarios.

The following sections describe each step in the evaluation process.

Table 1-1

C-E PLANT MITIGATION FEATURES FOR NRC SELECTED EVENTS

OPERATIONAL EVENT	MITIGATING FEATURES OF C-E NSSS REACTORS
H. B. Robinson 2 4/28/70 Condition 2 Non-Isolable Steam Line Break During Hot Functionals	1. AFW Termination to Affected Steam Generator. 2. "Low" Head High Pressure SI Pumps. 3. "Low" Capacity Charging (Make-Up) System.
H. B. Robinson 2 11/5/72 Condition 2 Stuck Open Atmospheric Dump Valve	1. AFW Termination to Affected Steam Generator. 2. "Low" Head High Pressure SI Pumps. 3. "Low" Capacity Charging (Make-Up) System. 4. ADV Not Automatically Controlled.
Rancho Seco 3/20/78 Condition 1 Loss of Power to ICS Resulting First in Loss of and then in Excess Feedwater Flow	1. Large S. G. Inventory. 2. Independent MFW Control Systems for Each S. G. 3. "Low" Head High Pressure SI Pumps. 4. "Low" Capacity Charging (Make-Up) System. 5. Separation of Control and Safety Systems.
H. B. Robinson 2 5/1/75 Condition 1 Small Break LOCA - RCP Seal Failure	1. "Low" Head High Pressure SI Pumps. 2. RCP Restart for Boration Not Required.
Three Mile Island 2 3/28/79 Condition 1 Small Break LOCA - PORV Stuck Open	1. S. G. Inventory. 2. Turbine Bypass Valve Size. 3. "Low" Head High Pressure SI Pumps.
Crystal River 3 2/26/80 Condition 1 Loss of Power to ICS Resulting in PORV Opening, Partial Loss of Feedwater and Rod Withdrawal	1. Large S. G. Inventory. 2. "Low" Head High Pressure SI Pumps. 3. Control Rod Withdrawal Prohibit. 4. Separation of Control and Safety Systems. 5. PORVs Actuated on High RCS Pressure Trip.
R. E. Ginna 1/5/82 Condition 1 Steam Generator Tube Rupture - Stuck Open PORV	1. PORVs Not Specified for RCS Depressurization. 2. "Low" Head High Pressure SI Pumps. 3. Main Feedwater Flow Rampback to 5% on Reactor Trip. 4. MSIS Occurs on Low S. G. Pressure.

2.1 INITIATING EVENT TYPES

The first step in the process was to identify the types of initiating events which may have the potential for significant cooling of the reactor vessel. Generally these are events which lead directly to a bulk cooldown of the reactor coolant, or lead to a substantial primary system depressurization with resulting safety injection. Table 2-1 presents a list of the types of initiating events that were considered.

Table 2-1

INITIATING EVENT TYPES

DECREASE IN FEEDWATER ENTHALPY
EXCESS FEEDWATER FLOW
EXCESS STEAM FLOW
LARGE LOSS OF COOLANT ACCIDENT
SMALL LOSS OF COOLANT ACCIDENT
PRESSURIZER PRESSURE CONTROL FAILURES
INADVERTENT SAFETY INJECTION ACTUATION
DECREASE IN CHARGING ENTHALPY
EXCESS SHUTDOWN COOLING

2.2 PLANT CONDITIONS

The second step was to define various plant operating states that could possibly increase the likelihood of certain over-cooling scenarios or maximize the severity of the over-cooling effects.[1] Table 2-2 defines four "plant conditions" in terms of the operational states of major systems. These plant conditions are associated with Power Operation, Start-up, Hot Standby, and Cooldown, respectively.

2.3 INITIATING EVENTS

The third step was to identify the initiating events for each event type presented in Table 1 and determine which should be considered credible during each of the plant conditions defined in Step 2. Table 2-3 identifies the applicable event/condition combinations.

2.4 EVENT SELECTION

The fourth step was to choose several of the event/condition combinations identified in the previous step for further analysis. The basis for these selections was the desire to investigate sequences with the greatest likelihood of representing PTS events, and to provide a representative cross-section of initiating event types. The event/condition combinations selected are circled on Table 2-3.

2.5 SEQUENCE OF EVENTS

The fifth step was to determine the expected sequence of events associated with each selected initiating event. This was based on the sequence of events, tables and diagrams in various FSARs (Chapter 15) and CEN-128

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1 For example, a MSLB would cause the most severe transient when SG inventory is the greatest and reactor power is the lowest. This situation is defined by Condition 2 in Table 2-2.

Table 2-2

System Status - vs. Plant Operating States

SYSTEMS	SYSTEM RELATED PLANT OPERATING STATES			
	CONDITION 1	CONDITION 2	CONDITION 3	CONDITION 4
TURBINE	A	B	D	D
TURBINE BYPASS	C	B	A OR B	B
ATMOSPHERIC DUMP VALVES	E	B	B	B
CHARGING	A	A	A	A
LETDOWN	A	A	A	A
MAIN PRESSURIZER SPRAYS	A	A	A	A OR B
AUXILIARY PRESSURIZER SPRAYS	D	D	D	B OR D
ENGINEERED SAFETY FEATURES ACTUATION SYSTEM	C	C	C	C OR E
PORV A/ PORV B	C *	C *	C *	C
REACTOR REGULATING SYSTEM	C INSERT E WITHDRAWAL	E INSERT B WITHDRAWAL	E E	F F
MAIN FEEDWATER	A	B	D	D
MAIN FEEDWATER BYPASS	C	B	D	D
AUXILIARY FEEDWATER	C	B **	B	B
SHUTDOWN COOLING SYSTEM	F	F	F	B
MAIN STEAM ISOLATION VALVES	OPEN	OPEN	OPEN	OPEN
MAIN FEEDWATER ISOLATION VALVES	OPEN	OPEN/CLOSED	CLOSED	CLOSED

GLOSSARY:
- A - OPERATING UNDER AUTO. CONTROL
- B - OPERATING UNDER MANUAL CONTROL
- C - STANDBY - CONTROL/ACTUATION SYSTEM AUTOMATIC
- D - STANDBY - COMPLEX MANUAL PROCEDURE TO ACTUATE
- E - STANDBY - SIMPLE MANUAL ACTION TO ACTUATE
- F - OUT-OF-SERVICE

* PORV B BLOCKED

** BOTH MANUAL AND AUTOMATIC AFW ISOLATION CASES ARE EVALUATED WHILE THE SYSTEM IS UNDER MANUAL CONTROL

Table 2-3

Initiating Events vs. Plant Operating States

INITIATING EVENTS	PLANT OPERATING CONDITION			
	1	2	3	4
1. Decrease in Feedwater Enthalpy				
a) MFW Heater System Failure	X	X		
b) AFW Delivery	X			
2. Excess Feedwater Flow				
a) MFW Flow Increases Above That for a Particular Power Level	X	X		
b) AFW Flow Increases Above That for a Particular Power Level		X	X	X
c) Reactor Power Decreases and MFW Flow Fails to Adjust	X	X		
d) Reactor Power Decreases and AFW Flow Fails to Adjust		X		
e) Turbine Trip and MFW Fails to Rampback	X			
3. Excess Steam Flow				
• Steam Flow Increases Above That Req'd for a Particular Power Level				
a) Steam Line Break	X	X	X	X
b) MFW Line Break d/s of Check Valve	X	X	X	X
c) ADV Inadvertently Opens	X	X	X	
d) TBV Inadvertently Opens	X	X	X	
e) MSSV Inadvertently Opens	X	X	X	X
f) Excess Steam Flow Through Turbine	X	X		
• Reactor Power Decreases and Steam Flow Fails to Adjust				
g) ADV is Open and Fails to Close		X	X	X
h) TBV is Open and Fails to Close		X	X	X
i) Turbine Fails to Decrease Steam Flow	X	X		
j) Reactor Trips and Turbine Fails to Trip	X	X		
k) Reactor Trips, Turbine Trips, and TBV Fails to Close After Quick Open or During Modulation	X	X		
l) Hi Pressure Transient, MSSV Opens and Fails to Close	X	X	X	

Table 2-3
(Cont.)
Initiating Events vs. Plant Operating States

INITIATING EVENTS	PLANT OPERATING CONDITION			
	1	2	3	4
4. Large LOCA				
a. Large Pipe Break	X	X	X	X
5. Small LOCA				
a) Non-Isolable Pipe Break	X	X	X	X
b) Isolable Pipe Break (Letdown Line)	X	X	X	X
c) RCP Seal Failure	X	X	X	X
d) PORV Inadvertently Opens	X	X	X	X
e) RCS Overpressure Scenario and One PORV/PSV Fails to Reclose	X	X	X	X
f) SG Tube Rupture	X	X	X	X
6. Pressurizer Pressure Control Failures*				
a) Spurious Main Spray Actuation	X	X	X	X
b) Spurious Aux. Spray Actuation+				
c) Pressure Transient Actuates Main Spray, Spray Fails to Decrease	X	X	X	X
d) Excess Main Spray During a Controlled Depressurization or Boron Mixing	X	X	X	X
e) Excess Aux. Spray During a Controlled Depressurization				X
7. Inadvertent SIAS When Below Shutoff Head				
a) Failure to Block SIAS Setpoint				X
b) Spurious SIAS				X

+ Spurious auxiliary spray actuation is not consi[illegible]ausible due to complexity of actuation procedure.

* Pressurizer control system failures which result in over-pressure transients are included in Category 5. This category refers to RCS de-pressurization events.

Table 2-3
(Cont.)
Initiating Events vs. Plant Operating States

INITIATING EVENTS	PLANT OPERATING CONDITION			
	1	2	3	4
8. Decrease in Charging Enthalpy				
a) PLCS Failure (Max. Charging, Loss of Letdown)	X	X	X	X
9. Maximum Shutdown Cooling				X

(Reference 1). From the expected plant responses, failure modes were hypothesized which had the potential for worsening the cooldown or increasing any subsequent repressurization.

2.6 EVENT TREES

The sixth step involved the construction of an event tree for each of the selected initiating events. The purpose of this step was to determine all logical combinations of the failure modes (or event tree branch points) defined in Step 5. These combinations represent over-cooling scenarios of varying severity. The event tree was constructed using the computer code CEETAR. CEETAR is designed to accept the failure modes and logic "rules" as input and to produce an event tree as output. The logic rules are used to eliminate those scenarios which are not possible due to the physical processes involved or which are not relevant to the PTS concern. The code also calculates the probability of each scenario and categorizes it (see Section 2.8) with respect to its occurrence probability. In this report the system interaction effects involving dependent failures (of particular interest in the Main Steam Line Break event) due to shared location, components, or auxiliary systems were not taken into account in the quantitative evaluation of the event tree branches. The event trees (along with the scenario probabilities and frequency categories) are plotted on a line-printer.

An option is also available which allows scenarios below a specified "cut-off" frequency to be eliminated.

2.7 DATA

The seventh step was to determine the initiating event frequencies and the corresponding event tree branch probabilities. Fault tree analysis was used to determine the branch point probabilities. C-E's Reliability Data System and other industry sources were used to determine the initiating event frequencies. No probabilistic credit was taken for the plant condition in which the initiating event was assumed to occur. The possibility of operator error was modelled for systems requiring manual control and credit was taken for manual backup of automatic systems provided there was adequate time during the transient to allow for operator response. Operator errors within the mitigating actions were limited to failure to respond to legitimate demand. The probabilities for operator error were obtained from Reference 3.

2.8 FREQUENCY CATEGORIZATION

This step involves the classification of each of the event sequences according to its occurrence frequency. Five frequency categories have been defined in accordance with the procedure followed in the St. Lucie 2 FSAR. These five categories are listed below:

- Moderate Frequency Event

  A Moderate Frequency event may occur during a calendar year for a particular plant. It is assumed that a Moderate frequency event has at least a 50 percent probability of occurring in any calendar year for a particular plant.

- Infrequent Event

  An Infrequent event may occur during the lifetime of a particular plant. It is assumed that an Infrequent event has less than a 50 percent probability of occurring in any calendar year, but at least a 50 percent probability of occurring in the assumed 40 year lifetime for a particular plant.

- Limiting Fault

  A Limiting Fault is not expected to occur during the lifetime of a particular plant. It is assumed that a Limiting Fault has less than a 50 percent probability of occurring in the assumed 40 year plant lifetime, but at least a 10^{-6} probability of occurring in any calendar year. This broad frequency group is divided into three subgroups to allow comparison of events with similar frequencies. These three subgroups of Limiting Faults are consistent with the acceptance guideline divisions suggested by the Standard Review Plans. The subgroups are defined below:

- Limiting Fault - 1

A Limiting Fault - 1 event has a low probability of occurring during the assumed 40 year lifetime for a particular plant.

Limiting Fault - 2

A Limiting Fault - 2 event has a very low probability of occurring in the assumed 40 year lifetime for a particular plant.

- Limiting Fault - 3

A Limiting Fault - 3 event has an exceedingly low probability of occurring in the assumed 40 year lifetime for a particular plant.

3.0 RESULTS

The following sections include the qualitative and quantitative results of the event tree/fault tree analyses for each of the following initiating events:

- o Main Steam Line Break
- o PORV Inadvertently Opens
- o MFW Fails to Rampback
- o Turbine Fails to Trip
- o Main Steam Safety Valve Fails to Reclose

A list of general assumptions concerning plant response is presented in Section 3.1. Section 3.2 provides qualitative and quantitative descriptions of each initiating event and event tree branch point. Diagrams of the event trees and corresponding scenario frequency summary tables are provided in Section 3.3 and Section 3.4 presents the scenario frequency categorization tables discussed in Section 2.8.

Since automatic isolation of auxiliary feedwater flow to a broken steam generator is a major factor in determining the severity of a cooldown it was given special consideration in the analysis. The AFW system modelled in the analysis was assumed to be designed to automatically prevent AFW flow to a broken steam generator when it is operating in automatic mode. For initiating events in Condition 2 during which the AFW system is assumed to be in manual mode two sets of tables in Section 3.4 are presented, one with automatic isolation assumed and one with manual action required.

3.1 ASSUMPTIONS

The following assumptions were made in determining the sequence of events and hypothesized failure modes. Note that these assumptions do not describe a specific plant but are representative of a "typical" C-E NSSS plant.

- o The RCPs are tripped by the operator following an SIAS on low RCS pressure.
- o The MFIVs do not close on SIAS.
- o There is one MFIV per Steam Generator.
- o The Auxiliary Feedwater Actuation System (AFAS) does not generate a signal to close the MFIVs.
- o The AFAS is designed to prevent AFW delivery to a "broken" steam generator when it is operating in the automatic mode (i.e., in Condition 1).

- No credit is taken for a MFW control valve block valve.
- When "turbine fails to trip" is included as a potential failure following the initiating event, an MSIS is assumed to occur and to successfully close the MSIVs and MFIVs.
- For the initiating event "Reactor Trip, Turbine Fails to Trip",MFW is assumed <u>not</u> to rampback.
- Allowable operator response times were based on a chronological study of the sequence of events for each initiating event.
- Each MFW pump has one associated MFW pump discharge valve.
- Isolation of AFW flow from the "broken" steam generator is accomplished using two valves, in series, in each of two lines feeding the generator.

3.2 EVENT TREE BRANCH DESCRIPTIONS AND FAILURE DATA

The event tree branch points representing failures during the plant responses are defined in Table 3-1.

The failure probabilities and initiating event frequencies used in constructing the event trees are provided in Table 3-2.

3.3 EVENT TREE RESULTS

Each of the following sections contains a list of the sequence of events or expected plant responses to each of the pre-selected initiating events.[1] Following the sequence of events is a discussion of the logic rules that were input to CEETAR along with the failure modes or event tree branch points specific to that event. Finally, a computer drawn event tree and corresponding scenario frequency summary table is presented for each initiating event.

Table 3-1

Event Tree Branch Descriptions

EVENT CODE	DESCRIPTION
MS	MAIN STEAM LINE BREAK OUTSIDE CONTAINMENT, UPSTREAM OF MSIV B. OCCURS DURING PLANT CONDITION 2.
PV	PORV INADVERTENTLY OPENS DURING PLANT CONDITION 1.
MV	A HIGH PRESSURE TRANSIENT ON THE SECONDARY SIDE CAUSES THE MSSVS TO LIFT. ONE MSSV ON SG B FAILS TO RESEAT. OCCURS DURING PLANT CONDITION 2.
MF	REACTOR/TURBINE TRIP AND MFW FAILS TO RAMPBACK DURING PLANT CONDITION 1.
TT	THE REACTOR TRIPS AND THE TURBINE FAILS TO TRIP DURING PLANT CONDITION 1.
A	THE MSIS IS NOT GENERATED DUE TO AN ACTUATION LOGIC FAILURE.
B	MSIV A FAILS TO CLOSE AUTOMATICALLY.
C	MSIV B FAILS TO CLOSE AUTOMATICALLY.
D	THE AFW SYSTEM IS IN MANUAL AND THE OPERATOR FAILS TO ISOLATE AFW FLOW TO THE RUPTURED SG WITHIN 5 MINUTES AFTER THE INITIATING EVENT.
E	THE AFW SYSTEM IS IN MANUAL AND THE OPERATOR FAILS TO TERMINATE AFW FLOW TO THE INTACT SG WITHIN 15 MINUTES AFTER THE INITIATING EVENT.
F	THE OPERATOR FAILS TO THROTTLE HPSI FLOW WITHIN 10 MINUTES AFTER THE INITIATING EVENT.
G	AFW FLOW IS LOST TO SG A BETWEEN 10 MINUTES AND 1 HOUR AFTER THE INITIATING EVENT.
H	THE TURBINE FAILS TO TRIP ON REACTOR TRIP. MSIS GENERATED.
I	MFW FAILS TO RAMPBACK ON TURBINE TRIP.
J	ONE MFW PUMP DISCHARGE VALVE FAILS TO CLOSE AUTOMATICALLY ON HI SG LEVEL.
K	ONE MFW PUMP FAILS TO TRIP ON HI-HI SG LEVEL.
L	THE OPERATOR FAILS TO CLOSE THE PORV BLOCK VALVE WITHIN 30 MINUTES AFTER THE PORV INADVERTENTLY OPENS.
M	THE OPERATOR FAILS TO THROTTLE HPSI FLOW WITHIN 30 MINUTES TO AN HOUR AFTER THE INITIATING EVENT.
N	MFW BYPASS FLOW IS LOST BETWEEN 10 MINUTES AND 1 HOUR AFTER THE INITIATING EVENT.
O	FAILURE TO AUTOMATICALLY DELIVER AFW FOLLOWING LOSS OF MFW BYPASS.
P	LOSS OF AFW FLOW FOLLOWING SAFETY INJECTION UP TO ONE HOUR AFTER SAFETY INJECTION.
Q	OPERATOR FAILS TO DECREASE AFW FLOW WITHIN 30 MINUTES AFTER THE REACTOR TRIPS.
R	OPERATOR FAILS TO ISOLATE THE LETDOWN LINE BREAK.
S	MFIV A FAILS TO CLOSE AUTOMATICALLY
T	MFIV B FAILS TO CLOSE AUTOMATICALLY
U	AFAS A FAILS TO PREVENT AFW DELIVERY TO RUPTURED SG A DUE TO AN ACTUATION LOGIC FAILURE
V	AFAS B FAILS TO PREVENT AFW DELIVERY TO RUPTURED SG B DUE TO AN ACTUATION LOGIC FAILURE.

Table 3-2

Event Tree Failure Data

EVENT CODE	BRANCH TITLE	FAILURE PROBABILITY
MS	Main Steam Line Break (U/S of MSIV B)*	4.6E-04/Yr.
PV	PORV Inadvertently Opens	8.8E-03/Yr.
MV	Hi Pressure Transient, MSSV Opens and Fails to Reclose	1.5E-02/Yr.
MF	Reactor Trip, MFW Fails to Rampback	1.7E-03/Yr.
TT	Reactor Trip, Turbine Fails to Trip	3.4E-05/Yr.
A	MSIS not Generated	9.9E-05
B	MSIV A Fails to Close	1.5E-04
C	MSIV B Fails to Close	1.5E-04
D	AFW Flow to SG B not Terminated	9.0E-01+
E	Excess AFW Flow to SG A	5.0E-01
F	HPSI Flow Delivered, not Throttled (short-term)	7.0E-01
G	Loss of AFW Flow to SG A	6.3E-05
H	Turbine Fails to Trip on Reactor Trip (MSIS Successful)	6.5E-06
I	Fail to Rampback MFW on Turbine Trip	3.3E-04
J	MFWP Discharge Vlvs. Fail to Close on Hi SG Level	5.0E-04
K	MFWP Fails to Trip on Hi-Hi SG Level	7.0E-04
L	Operator Fails to Isolate PORV	1.0E-01
M	HPSI Flow Delivered, not Throttled (long-term)	1.0E-01
N	Loss of MFW Bypass Following Safety Injection	2.5E-05
O	Fail to Deliver AFW Following Loss of MFW Bypass	2.5E-04
P	Loss of AFW Following Safety Injection	2.9E-06
Q	Excess AFW Flow	1.0E-01
R	Failure to Isolate Letdown Line Break	3.0E-02
S	MSIV A Fails to Close	3.7E-03
T	MSIV B Fails to Close	3.7E-03
U	Fail to Isolate AFW to SG A - On Actuation of AFW System	1.0E-05
V	Fail to Isolate AFW to SG B - On Actuation of AFW System	1.0E-05

* Break location is specified for modelling purposes only. No probabilistic credit is taken for break location.

\+ Value for manual isolation is 9.0E-01, for automatic isolation it is 1.7E-05.

3.3.1 Main Steam Line Break

The following plant responses are expected to occur following a Main Steam Line Break during Plant Condition 2:[1]

- o Main Steam Line Break, SG B
- o Reactor Trip (Turbine is on stand-by)
- o MSIS generated
- o MSIVs close
- o Operator terminates AFW flow to SG B[2]
- o Operator throttles HPSI
- o Operator decreases AFW flow to SG A

The following logic rules were input to CEETAR to eliminate failure modes or branch combinations which were illogical or not relevant to the PTS concern.

1. If an MSIS is not generated then MSIV A and MSIV B are not required to close, therefore, branch point A will not appear in a scenario that contains branch points B or C.

2. If AFW flow is lost to SG A then there will not be excess AFW flow to SG A, therefore, branch points E and G will not appear in the same scenario.

The Main Steam Line Break event tree appears in Figure 3-1 and the scenario frequencies are presented in Table 3-3.

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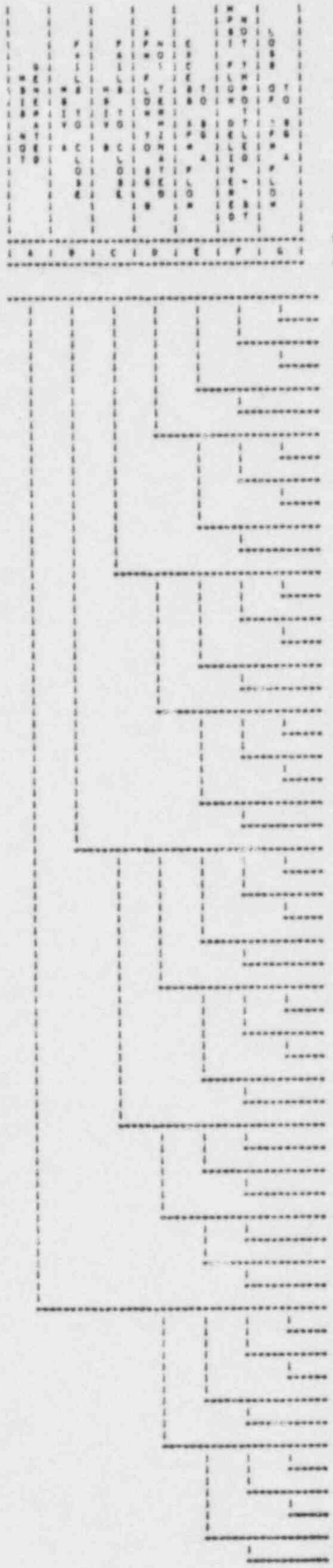
1. The following initial conditions are applicable to the two initiating events that are assumed to occur during Plant Condition 2:

   - o AFW is in use and under manual control
   - o The turbine bypass valves are open and under manual control
   - o The MSIVs are open
   - o The MFIVs are closed

2. If this event were analyzed as occurring in Condition 1, automatic termination of AFW flow to SG B would be assumed.

Figure 3-1

MSLB, UPSTREAM MSIV, OUTSIDE CONTAINMENT, CONDITION B



PATH NO.	COMBINATION CODE	PATH PROB.	FREQUENCY CATEGORY
1.	MB	6.9E-06	LF-3
2.	MB - G	4.3E-10	VLF
3.	MB - F	1.0E-05	LF-3
4.	MB - FG	1.0E-09	VLF
5.	MB - E	6.9E-06	LF-3
6.	MB - EF	1.6E-05	LF-3
7.	MB - D	6.2E-05	LF-3
8.	MB - DG	3.9E-09	VLF
9.	MB - DF	1.4E-08	LF-2
10.	MB - DFG	9.1E-09	VLF
11.	MB - DE	6.2E-05	LF-3
12.	MB - DEF	1.4E-08	LF-2
13.	MB - C	1.0E-09	VLF
14.	MB - CG	6.5E-14	VLF
15.	MB - CF	2.4E-09	VLF
16.	MB - CFG	1.5E-13	VLF
17.	MB - CE	1.0E-09	VLF
18.	MB - CEF	2.4E-09	VLF
19.	MB - CD	9.3E-09	VLF
20.	MB - CDG	5.9E-13	VLF
21.	MB - CDF	2.2E-08	VLF
22.	MB - CDFG	1.4E-12	VLF
23.	MB - CDE	9.3E-09	VLF
24.	MB - CDEF	2.2E-08	VLF
25.	MB - B	1.0E-09	VLF
26.	MB - BG	6.5E-14	VLF
27.	MB - BF	2. .E-09	VLF
28.	MB - BFG	1.5E-13	VLF
29.	MB - BE	1.0E-09	VLF
30.	MB - BEF	2.4E-09	VLF
31.	MB - BD	9.3E-09	VLF
32.	MB - BDG	5.9E-13	VLF
33.	MB - BDF	2.2E-08	VLF
34.	MB - BDFG	1.4E-12	VLF
35.	MB - BDE	9.3E-09	VLF
36.	MB - BDEF	2.2E-08	VLF
37.	MB - BC	1.6E-13	VLF
38.	MB - BCF	3.6E-13	VLF
39.	MB - BCE	1.6E-13	VLF
40.	MB - BCEF	3.6E-13	VLF
41.	MB - BCD	1.4E-12	VLF
42.	MB - BCDF	3.3E-12	VLF
43.	MB - BCDE	1.4E-12	VLF
44.	MB - BCDEF	3.3E-12	VLF
45.	MB - A	6.0E-10	VLF
46.	MB - AG	4.3E-14	VLF
47.	MB - AF	1.6E-09	VLF
48.	MB - AFG	1.0E-13	VLF
49.	MB - AE	6.0E-10	VLF
50.	MB - AEF	1.6E-09	VLF
51.	MB - AD	6.1E-09	VLF
52.	MB - ADG	3.9E-13	VLF
53.	MB - ADF	1.4E-08	VLF
54.	MB - ADFG	9.0E-13	VLF
55.	MB - ADE	6.1E-09	VLF
56.	MB - ADEF	1.4E-08	VLF

Table 3-3

SUMMARY TABLE

FOR

MSLB, UPSTREAM MSIV, OUTSIDE CONTAINMENT, CONDITION 2

BRANCH NUMBER	COMBINATION CODE	RECURRENCE FREQUENCY
1.	MS	6,9E-06
2.	MS - G	4,3E-10
3.	MS - F	1,6E-05
4.	MS - FG	1,0E-09
5.	MS - E	6,9E-06
6.	MS - EF	1,6E-05
7.	MS - D	6,2E-05
8.	MS - DG	3,9E-09
9.	MS - DF	1,4E-04
10.	MS - DFG	9,1E-09
11.	MS - DE	6,2E-05
12.	MS - DEF	1,4E-04
13.	MS - C	1,0E-09
14.	MS - CG	6,5E-14
15.	MS - CF	2,4E-09
16.	MS - CFG	1,5E-13
17.	MS - CE	1,0E-09
18.	MS - CEF	2,4E-09
19.	MS - CD	9,3E-09
20.	MS - CDG	5,9E-13
21.	MS - CDF	2,2E-08
22.	MS - CDFG	1,4E-12
23.	MS - CDE	9,3E-09
24.	MS - CDEF	2,2E-08
25.	MS - B	1,0E-09
26.	MS - BG	6,5E-14
27.	MS - BF	2,4E-09
28.	MS - BFG	1,5E-13
29.	MS - BE	1,0E-09
30.	MS - BEF	2,4E-09
31.	MS - BD	9,3E-09
32.	MS - BDG	5,9E-13
33.	MS - BDF	2,2E-08
34.	MS - BDFG	1,4E-12
35.	MS - BDE	9,3E-09
36.	MS - BDEF	2,2E-08
37.	MS - BC	1,6E-13
38.	MS - BCF	3,6E-13
39.	MS - BCE	1,6E-13
40.	MS - BCEF	3,6E-13
41.	MS - BCD	1,4E-12
42.	MS - BCDF	3,3E-12
43.	MS - BCDE	1,4E-12
44.	MS - BCDEF	3,3E-12
45.	MS - A	6,8E-10
46.	MS - AG	4,3E-14
47.	MS - AF	1,6E-09
48.	MS - AFG	1,0E-13
49.	MS - AE	6,8E-10
50.	MS - AEF	1,6E-09
51.	MS - AD	6,1E-09
52.	MS - ADG	3,9E-13
53.	MS - ADF	1,4E-08
54.	MS - ADFG	9,0E-13
55.	MS - ADE	6,1E-09
56.	MS - ADEF	1,4E-08

The event tree and summary table are shown for the case where manual isolation of AFW to a broken steam is required while the AFW system is in the manual mode. This implies that the probability of Branch D (see Table 3-2) is 9.0E-01.

3.3.2 PORV Inadvertently Opens

The following plant responses are expected to occur following an inadvertent opening of a PORV during Plant Condition 1:

- o PORV inadvertently opens
- o Reactor trip
- o Turbine trip
- o MFW rampsback to 5% flow
- o Safety injection
- o Operator isolates break
- o Operator throttles HPSI

The following logic rules were input to CEETAR to eliminate failure modes or branch combinations which were illogical or not relevant to the PTS concern.

1. If the turbine fails to trip, an MSIS will close the MFIVs which will result in loss of MFW. Therefore, branch point H will not appear in scenarios containing branch points I, J, K, N or O.

2. If MFW successfully rampsback then a HI SG level will not occur. Therefore, J and K will only appear in combination with I.

3. If the MFW pump discharge valves close on HI SG level a HI-HI SG level will not occur. Therefore, K will only appear in combination with J.

4. If MFW fails to rampback and the MFW pump discharge valves close on HI level, or the MFW pump discharge valves fail to close and the MFW

pumps trip on HI-HI level then loss of MFW will occur. Therefore, branch points N and O will not appear with the above failure combinations because MFW bypass is never established.

5. If MFW fails to rampback and the MFW pump discharge valves fail to close and the MFW pumps fail to trip then MFW flow will continue. Therefore, branch points N, O, P and Q will not appear with the above failure combination.

6. If AFW is never delivered then it will not be lost. Therefore, O will not appear with P.

7. If MFW Bypass is never lost then AFW delivery will not be required. Therefore, branch point O will not appear without branch point N.

8. If AFW is never delivered or is lost then excess AFW cannot occur. Therefore, neither O nor P will appear with Q.

The PORV Inadvertently Opens event tree appears in Figure 3-2 and the scenario frequencies are presented in Table 3-4.

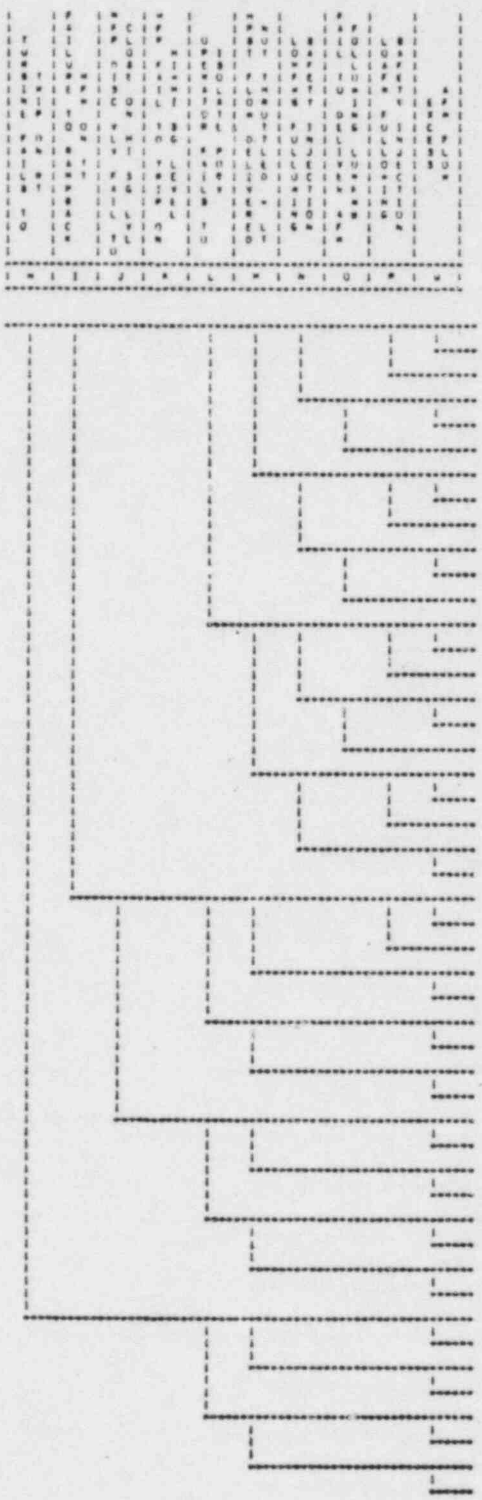
3.3.3 Reactor Trip, Turbine Trip, MFW Fails to Rampback

The following plant responses are expected to occur following a reactor trip, turbine trip, and MFW fails to rampback during Plant Condition 1:

- o Reactor trip, turbine trip, and MFW fails to rampback
- o MFW pump discharge valves close on HI SG level (MFW pumps trip on HI-HI SG level)
- o SG low level
- o AFW delivery

Figure 3-2

PORV INADVERTENTLY OPENS, CONDITION I



PATH NO.	COMBINATION CODE	PATH PROB.	FREQUENCY CATEGORY
1.	PV	6.4E-03	LF-1
2.	PV - Q	7.1E-04	LF-2
3.	PV - P	2.1E-08	VLF
4.	PV - N	1.6E-07	VLF
5.	PV - NQ	1.8E-08	VLF
6.	PV - NO	4.5E-11	VLF
7.	PV - M	7.1E-04	LF-2
8.	PV - MQ	7.9E-05	LF-3
9.	PV - MP	2.3E-09	VLF
10.	PV - MN	1.8E-08	VLF
11.	PV - MNQ	2.0E-09	VLF
12.	PV - MNO	5.0E-12	VLF
13.	PV - L	7.1E-04	LF-2
14.	PV - LQ	7.9E-05	LF-3
15.	PV - LP	2.3E-09	VLF
16.	PV - LN	1.8E-08	VLF
17.	PV - LNQ	2.0E-09	VLF
18.	PV - LNO	5.0E-12	VLF
19.	PV - LM	7.9E-05	LF-3
20.	PV - LMQ	8.8E-06	LF-3
21.	PV - LMP	2.6E-10	VLF
22.	PV - LMN	2.0E-09	VLF
23.	PV - LMNQ	2.2E-10	VLF
24.	PV - I	2.1E-06	LF-3
25.	PV - IQ	2.4E-07	VLF
26.	PV - IP	6.0E-12	VLF
27.	PV - IM	2.4E-07	VLF
28.	PV - IMQ	2.6E-08	VLF
29.	PV - IL	2.4E-07	VLF
30.	PV - ILQ	2.6E-08	VLF
31.	PV - ILM	2.6E-08	VLF
32.	PV - ILMQ	2.9E-09	VLF
33.	PV - IJ	1.1E-09	VLF
34.	PV - IJQ	1.2E-10	VLF
35.	PV - IJM	1.2E-10	VLF
36.	PV - IJMQ	1.3E-11	VLF
37.	PV - IJL	1.2E-10	VLF
38.	PV - IJLQ	1.3E-11	VLF
39.	PV - IJLM	1.3E-11	VLF
40.	PV - IJLMQ	1.5E-12	VLF
41.	PV - H	4.2E-08	VLF
42.	PV - HQ	4.6E-09	VLF
43.	PV - HM	4.6E-09	VLF
44.	PV - HMQ	5.2E-10	VLF
45.	PV - HL	4.6E-09	VLF
46.	PV - HLQ	5.2E-10	VLF
47.	PV - HLM	5.2E-10	VLF
48.	PV - HLMQ	5.7E-11	VLF

Table 3-4

SUMMARY TABLE

FOR

PORV INADVERTENTLY OPENS, CONDITION 1

BRANCH NUMBER	COMBINATION CODE	RECURRENCE FREQUENCY
1,	PV	6,4E-03
2,	PV - Q	7,1E-04
3,	PV - P	2,1E-08
4,	PV - N	1,6E-07
5,	PV - NQ	1,8E-08
6,	PV - NO	4,5E-11
7,	PV - M	7,1E-04
8,	PV - MQ	7,9E-05
9,	PV - MP	2,3E-09
10,	PV - MN	1,8E-08
11,	PV - MNQ	2,0E-09
12,	PV - MNO	5,0E-12
13,	PV - L	7,1E-04
14,	PV - LQ	7,9E-05
15,	PV - LP	2,3E-09
16,	PV - LN	1,8E-08
17,	PV - LNQ	2,0E-09
18,	PV - LNO	5,0E-12
19,	PV - LM	7,9E-05
20,	PV - LMQ	8,8E-06
21,	PV - LMP	2,6E-10
22,	PV - LMN	2,0E-09
23,	PV - LMNQ	2,2E-10
24,	PV - I	2,1E-06
25,	PV - IQ	2,4E-07
26,	PV - IP	6,8E-12
27,	PV - IM	2,4E-07
28,	PV - IMQ	2,6E-08
29,	PV - IL	2,4E-07
30,	PV - ILQ	2,6E-08
31,	PV - ILM	2,6E-08
32,	PV - ILMQ	2,9E-09
33,	PV - IJ	1,1E-09
34,	PV - IJQ	1,2E-10
35,	PV - IJM	1,2E-10
36,	PV - IJMQ	1,3E-11
37,	PV - IJL	1,2E-10
38,	PV - IJLQ	1,3E-11
39,	PV - IJLM	1,3E-11
40,	PV - IJLMQ	1,5E-12
41,	PV - H	4,2E-08
42,	PV - HQ	4,6E-09
43,	PV - HM	4,6E-09
44,	PV - HMQ	5,2E-10
45,	PV - HL	4,6E-09
46,	PV - HLQ	5,2E-10
47,	PV - HLM	5,2E-10
48,	PV - HLMQ	5,7E-11

The following logic rules were input to CEETAR to eliminate failures modes or branch combinations which were illogical or not relevant to the PTS concern.

1. If the MFW pump discharge valves fail to close and the MFW pumps fail to trip then full MFW flow will continue. Therefore, branch points P and Q will not appear with the above failure combination.

2. If the MFW pump discharge valves close on HI SG level a HI-HI SG level will not occur. Therefore, K will only appear in combination with J.

The MFW Fails to Rampback event tree appears in Figure 3-3 and the scenario frequencies are presented in Table 3-5.

3.3.4 Reactor Trip, Turbine Fails to Trip

The following plant responses are expected to occur following a reactor trip, and turbine fails to trip during Plant Condition 1:

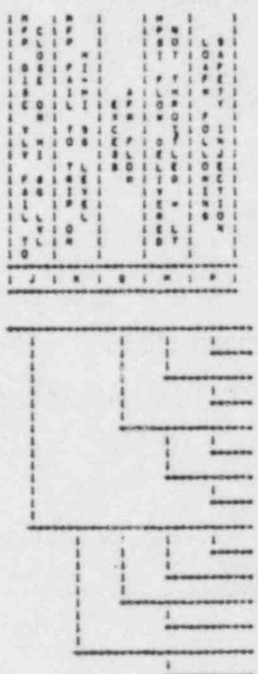
- o Turbine fails to trip
- o MSIS
- o MSIVs close
- o MFIVs close
- o Safety injection
- o AFW delivery
- o Operator throttles safety injection
- o Operator throttles AFW

The following logic rules were input to CEETAR to eliminate failure modes or branch combinations which were illogical or not relevant to the PTS concern.

1. If an MSIS is not generated then MSIV A, MSIV B, MFIV A and MFIV B are not required to close, therefore, branch point A will not appear in combination with branch points B, C, S or T.

Figure 3-3

RX/TT AND MAIN FEEDWATER FAILS TO RAMPBACK, CONDITION 1



PATH NO.	COMBINATION CODE	PATH PROB.	FREQUENCY CATEGORY
1.	MF	1.4E-03	LF-1
2.	MF - P	4.0E-09	VLF
3.	MF - M	1.5E-04	LF-2
4.	MF - MP	4.4E-10	VLF
5.	MF - G	1.5E-04	LF-2
6.	MF - GP	4.4E-10	VLF
7.	MF - GM	1.7E-05	LF-3
8.	MF - GMP	4.9E-11	VLF
9.	MF - J	6.9E-07	VLF
10.	MF - JP	2.0E-12	VLF
11.	MF - JM	7.6E-08	VLF
12.	MF - JG	7.6E-08	VLF
13.	MF - JGM	8.5E-09	VLF
14.	MF - JK	5.4E-10	VLF
15.	MF - JKM	6.0E-11	VLF

Table 3-5

SUMMARY TABLE

FOR

RT/TT AND MAIN FEEDWATER FAILS TO RAMPBACK, CONDITION 1

BRANCH NUMBER	COMBINATION CODE	RECURRENCE FREQUENCY
1.	MF	1.4E-03
2.	MF - P	4.0E-09
3.	MF - M	1.5E-04
4.	MF - MP	4.4E-10
5.	MF - Q	1.5E-04
6.	MF - QP	4.4E-10
7.	MF - QM	1.7E-05
8.	MF - QMP	4.9E-11
9.	MF - J	6.9E-07
10.	MF - JP	2.0E-12
11.	MF - JM	7.6E-08
12.	MF - JQ	7.6E-08
13.	MF - JQM	8.5E-09
14.	MF - JK	5.4E-10
15.	MF - JKM	6.0E-11

2. If either MFIV fails to close and the MFW pump discharge valves fail to close and the MFW pumps fail to trip, then MFW flow will continue. Therefore, branch points P, Q, U, and V will not appear in the above failure combinations.

3. If a SG is intact, the Auxiliary Feedwater Actuation System is not required to prevent AFW delivery to that SG. Therefore, U will only appear with B, and V will only appear with C.

4. If both MSIVs fail to close, a HI SG level will not occur. Therefore, neither J nor K will appear with B and C.

5. If MFW flow is successfully terminated by closure of the MFIVs, a HI SG level will not occur. Therefore, neither J nor K will appear unless S or T appears.

6. If the MFW pump discharge valves close on HI SG level a HI-HI SG level will not occur. Therefore, K will only appear in combination with J.

7. If AFW is lost then excess AFW cannot occur. Therefore, P will not appear with Q.

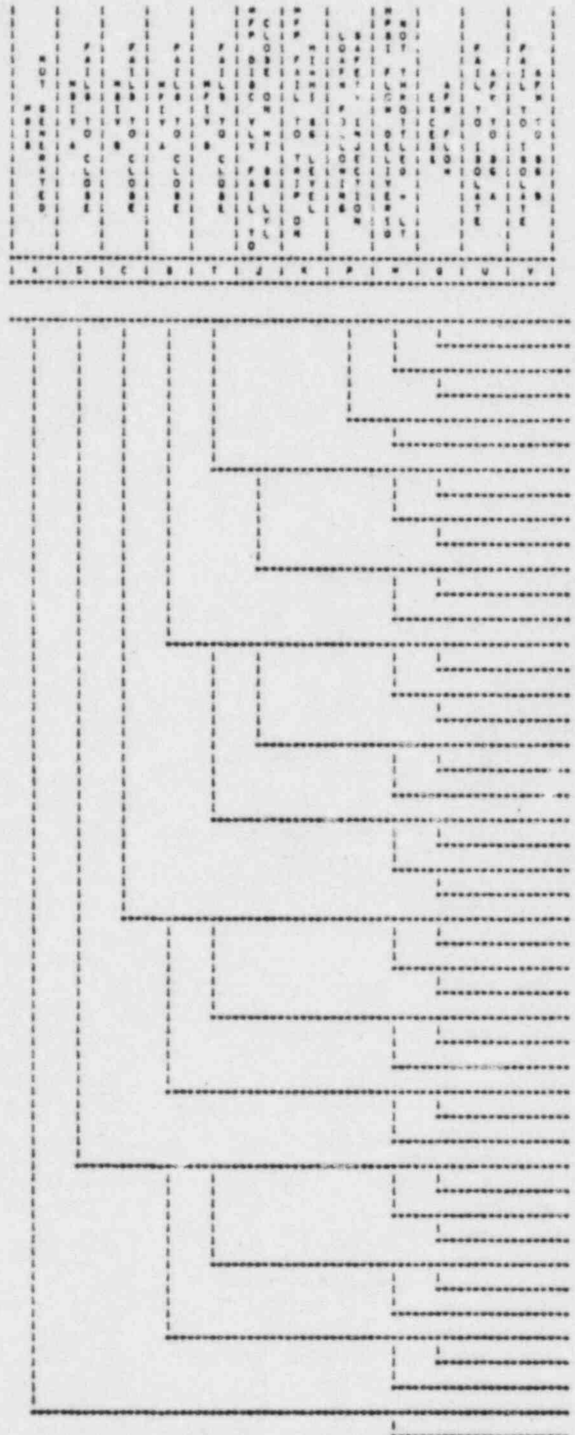
8. If the MSIS is not generated, MFW flow will continue. Therefore, Q will not appear with A.

9. The AFAS isolates a ruptured SG by noting the pressure differential between the two steam generators. U and V will not appear in the same scenario.

The Turbine Fails to Trip event tree appears in Figure 3-4 and the scenario frequencies are presented in Table 3-6.

Figure 3-4

REACTOR TRIPS AND TURBINE FAILS TO TRIP, CONDITION 1



PATH NO.	COMBINATION CODE	PATH PROB.	FREQUENCY CATEGORY
1.	TT	2.7E-05	LF-3
2.	TT - Q	5.0E-06	LF-3
3.	TT - M	1.0E-06	LF-3
4.	TT - MQ	3.4E-07	VLF
5.	TT - P	6.8E-11	VLF
6.	TT - PM	9.8E-12	VLF
7.	TT - T	1.0E-07	VLF
8.	TT - TQ	1.1E-08	VLF
9.	TT - TM	1.1E-08	VLF
10.	TT - TMQ	1.2E-09	VLF
11.	TT - TJ	5.1E-11	VLF
12.	TT - TJQ	5.6E-12	VLF
13.	TT - TJM	5.6E-12	VLF
14.	TT - D	1.0E-07	VLF
15.	TT - DQ	1.1E-08	VLF
16.	TT - DM	1.1E-08	VLF
17.	TT - DMQ	1.3E-09	VLF
18.	TT - DJ	5.1E-11	VLF
19.	TT - DJQ	5.6E-12	VLF
20.	TT - DJM	5.6E-12	VLF
21.	TT - DT	3.8E-10	VLF
22.	TT - DTQ	4.2E-11	VLF
23.	TT - DTM	4.2E-11	VLF
24.	TT - DTMQ	4.7E-12	VLF
25.	TT - C	4.1E-09	VLF
26.	TT - CQ	4.6E-10	VLF
27.	TT - CM	4.6E-10	VLF
28.	TT - CMQ	5.1E-11	VLF
29.	TT - CT	1.5E-11	VLF
30.	TT - CTQ	1.7E-12	VLF
31.	TT - CTM	1.7E-12	VLF
32.	TT - CD	1.5E-11	VLF
33.	TT - CDQ	1.7E-12	VLF
34.	TT - CDM	1.7E-12	VLF
35.	TT - B	4.1E-09	VLF
36.	TT - BQ	4.6E-10	VLF
37.	TT - BM	4.6E-10	VLF
38.	TT - BMQ	5.1E-11	VLF
39.	TT - BT	1.5E-11	VLF
40.	TT - BTQ	1.7E-12	VLF
41.	TT - BTM	1.7E-12	VLF
42.	TT - BD	1.5E-11	VLF
43.	TT - BDQ	1.7E-12	VLF
44.	TT - BDM	1.7E-12	VLF
45.	TT - A	3.0E-09	VLF
46.	TT - AM	3.4E-10	VLF

Table 3-6

SUMMARY TABLE

FOR

REACTOR TRIPS AND TURBINE FAILS TO TRIP, CONDITION 1

BRANCH NUMBER	COMBINATION CODE	RECURRENCE FREQUENCY
1.	TT	2.7E-05
2.	TT - Q	3.0E-06
3.	TT - M	3.0E-06
4.	TT - MQ	3.4E-07
5.	TT - P	8.8E-11
6.	TT - PM	9.8E-12
7.	TT - T	1.0E-07
8.	TT - TQ	1.1E-08
9.	TT - TM	1.1E-08
10.	TT - TMQ	1.3E-09
11.	TT - TJ	5.1E-11
12.	TT - TJQ	5.6E-12
13.	TT - TJM	5.6E-12
14.	TT - S	1.0E-07
15.	TT - SQ	1.1E-08
16.	TT - SM	1.1E-08
17.	TT - SMQ	1.3E-09
18.	TT - SJ	5.1E-11
19.	TT - SJQ	5.6E-12
20.	TT - SJM	5.6E-12
21.	TT - ST	3.8E-10
22.	TT - STQ	4.2E-11
23.	TT - STM	4.2E-11
24.	TT - STMQ	4.7E-12
25.	TT - C	4.1E-09
26.	TT - CQ	4.6E-10
27.	TT - CM	4.6E-10
28.	TT - CMQ	5.1E-11
29.	TT - CT	1.5E-11
30.	TT - CTQ	1.7E-12
31.	TT - CTM	1.7E-12
32.	TT - CS	1.5E-11
33.	TT - CSQ	1.7E-12
34.	TT - CSM	1.7E-12
35.	TT - B	4.1E-09
36.	TT - BQ	4.6E-10
37.	TT - BM	4.6E-10
38.	TT - BMQ	5.1E-11
39.	TT - BT	1.5E-11
40.	TT - BTQ	1.7E-12
41.	TT - BTM	1.7E-12
42.	TT - BS	1.5E-11
43.	TT - BSQ	1.7E-12
44.	TT - BSM	1.7E-12
45.	TT - A	3.0E-09
46.	TT - AM	3.4E-10

3.3.5 HI Pressure Transient, MSSVs Open and One Fails to Reclose

The following plant responses are expected to occur following the failure of an MSSV to reclose during Plant Condition 2:

- o MSSV on SG B fails to reclose
- o Reactor trip
- o MSIS generated
- o MSIVs close
- o Operator terminates flow to SG B[1]
- o Operator throttles HPSI
- o Operator decreases AFW flow to SG A

The logic rules for MSSV Fails to Reclose are identical to the logic rules for Main Steam Line Break.

The MSSV Fails to Reclose event tree appears in Figure 3-5 and the scenario frequencies are presented in Table 3-7.

The event tree and summary table are shown for the case where manual isolation of AFW to a broken steam is required while the AFW system is in the manual mode. This implies that the probability of Branch D (see Table 3-2) is 9.0E-01.

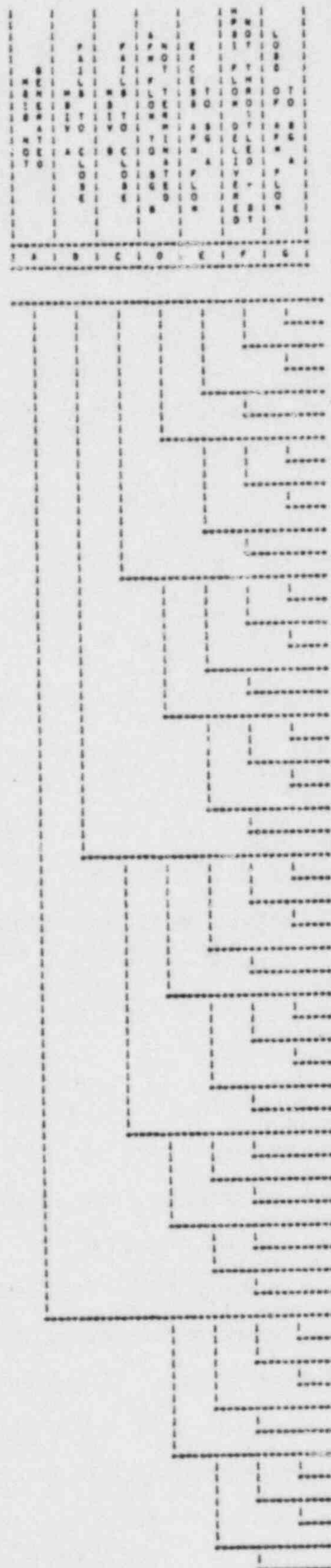
3.4 SCENARIO FREQUENCY CATEGORIZATION

The results of the frequency categorization are summarized in Tables 3-8 and 3-9. The tables contain a list of all sequences in the moderate, infrequent and limiting fault categories which were obtained from the event trees in Section 3.3. In addition, a number of sequences identified as being of Very Low Frequency (VLF) are listed on the tables. The VLF sequences fell below the range specified as Limiting Fault 3 (i.e., less than 10^{-6}/yr) but above an arbitrary cutoff frequency specified as 10^{-12} for all of the event trees except the Main Steam Line Break event

1. If this event were analyzed as occurring in Condition 1, automatic termination of AFW flow to SG B would be assumed.

Figure 3-5

HI PRESS TRANSIENT, MSSV OPENS AND FAILS TO RECLOSE, CONDITION 2



PATH NO.	COMBINATION CODE	PATH PROB.	FREQUENCY CATEGORY
1.	MV	2.2E-04	LF-2
2.	MV - G	1.4E-08	VLF
3.	MV - F	5.2E-04	LF-2
4.	MV - FG	3.3E-08	VLF
5.	MV - E	2.2E-04	LF-2
6.	MV - EF	5.2E-04	LF-2
7.	MV - D	2.0E-03	LF-1
8.	MV - DG	1.3E-07	VLF
9.	MV - DF	4.7E-03	LF-1
10.	MV - DFG	3.0E-07	VLF
11.	MV - DE	2.0E-03	LF-1
12.	MV - DEF	4.7E-03	LF-1
13.	MV - C	3.4E-08	VLF
14.	MV - CG	2.1E-12	VLF
15.	MV - CF	7.9E-08	VLF
16.	MV - CFG	5.0E-12	VLF
17.	MV - CE	3.4E-08	VLF
18.	MV - CEF	7.9E-08	VLF
19.	MV - CD	3.0E-07	VLF
20.	MV - CDG	1.9E-11	VLF
21.	MV - CDF	7.1E-07	VLF
22.	MV - CDFG	4.5E-11	VLF
23.	MV - CDE	3.0E-07	VLF
24.	MV - CDEF	7.1E-07	VLF
25.	MV - B	3.4E-08	VLF
26.	MV - BG	2.1E-12	VLF
27.	MV - BF	7.9E-08	VLF
28.	MV - BFG	5.0E-12	VLF
29.	MV - BE	3.4E-08	VLF
30.	MV - BEF	7.9E-08	VLF
31.	MV - BD	3.0E-07	VLF
32.	MV - BDG	1.9E-11	VLF
33.	MV - BDF	7.1E-07	VLF
34.	MV - BDFG	4.5E-11	VLF
35.	MV - BDE	3.0E-07	VLF
36.	MV - BDEF	7.1E-07	VLF
37.	MV - BC	5.1E-12	VLF
38.	MV - BCF	1.2E-11	VLF
39.	MV - BCE	5.1E-12	VLF
40.	MV - BCEF	1.2E-11	VLF
41.	MV - BCD	4.6E-11	VLF
42.	MV - BCDF	1.1E-10	VLF
43.	MV - BCDE	4.6E-11	VLF
44.	MV - BCDEF	1.1E-10	VLF
45.	MV - A	2.2E-08	VLF
46.	MV - AG	1.4E-12	VLF
47.	MV - AF	5.2E-08	VLF
48.	MV - AFG	3.3E-12	VLF
49.	MV - AE	2.2E-08	VLF
50.	MV - AEF	5.2E-08	VLF
51.	MV - AD	2.0E-07	VLF
52.	MV - ADG	1.3E-11	VLF
53.	MV - ADF	4.7E-07	VLF
54.	MV - ADFG	2.9E-11	VLF
55.	MV - ADE	2.0E-07	VLF
56.	MV - ADEF	4.7E-07	VLF

Table 3-7

SUMMARY TABLE

FOR

HI PRESS TRANSIENT, MSSV OPENS AND FAILS TO RECLOSE, CONDITION 2

BRANCH NUMBER	COMBINATION CODE	RECURRENCE FREQUENCY
1.	MV	2.2E-04
2.	MV - G	1.4E-08
3.	MV - F	5.2E-04
4.	MV - FG	3.3E-08
5.	MV - E	2.2E-04
6.	MV - EF	5.2E-04
7.	MV - D	2.0E-03
8.	MV - DG	1.3E-07
9.	MV - DF	4.7E-03
10.	MV - DFG	3.0E-07
11.	MV - DE	2.0E-03
12.	MV - DEF	4.7E-03
13.	MV - C	3.4E-08
14.	MV - CG	2.1E-12
15.	MV - CF	7.9E-08
16.	MV - CFG	5.0E-12
17.	MV - CE	3.4E-08
18.	MV - CEF	7.9E-08
19.	MV - CD	3.0E-07
20.	MV - CDG	1.9E-11
21.	MV - CDF	7.1E-07
22.	MV - CDFG	4.5E-11
23.	MV - CDE	3.0E-07
24.	MV - CDEF	7.1E-07
25.	MV - B	3.4E-08
26.	MV - BG	2.1E-12
27.	MV - BF	7.9E-08
28.	MV - BFG	5.0E-12
29.	MV - BE	3.4E-08
30.	MV - BEF	7.9E-08
31.	MV - BD	3.0E-07
32.	MV - BDG	1.9E-11
33.	MV - BDF	7.1E-07
34.	MV - BDFG	4.5E-11
35.	MV - BDE	3.0E-07
36.	MV - BDEF	7.1E-07
37.	MV - BC	5.1E-12
38.	MV - BCF	1.2E-11
39.	MV - BCE	5.1E-12
40.	MV - BCEF	1.2E-11
41.	MV - BCD	4.6E-11
42.	MV - BCDF	1.1E-10
43.	MV - BCDE	4.6E-11
44.	MV - BCDEF	1.1E-10
45.	MV - A	2.2E-08
46.	MV - AG	1.4E-12
47.	MV - AF	5.2E-08
48.	MV - AFG	3.3E-12
49.	MV - AE	2.2E-08
50.	MV - AEF	5.2E-08
51.	MV - AD	2.0E-07
52.	MV - ADG	1.3E-11
53.	MV - ADF	4.7E-07
54.	MV - ADFG	2.9E-11
55.	MV - ADE	2.0E-07
56.	MV - ADEF	4.7E-07

tree which used a cut-off of 10^{-14}. Since a large number of sequences fell in this range, only those with a frequency greater than 10^{-7} were listed.

Table 3-8 contains sequences in which the initiating event occurs in Condition 1 (see Table 3). Table 3-9 contains those which occur in Condition 2.

Table 3-10 provides a list of what are considered to be the most severe PTS scenarios of the sequences presented in Tables 3-8 and 3-9.

As described in Section 3.0 the scenario probabilities for the Condition 2 events (i.e., MV and MS) were calculated twice, once with manual AFW isolation assumed and once with automatic isolation assumed. This changed the probability of the "D" branch from 0.9 to 1.7×10^{-5}, respectively. The effect of this change can be seen by comparing Tables 3-9 and 3-10 with Tables 3-11 and 3-12. The first two tables were constructed assuming manual isolation. Tables 3-11 and 3-12 show the effect on these scenarios of changing to automatic isolation. The most significant effect is the elimination of the severe scenarios which contain the manual isolation branch (D) due to their greatly reduced frequency.

Table 3-12 was constructed by selecting events from Table 3-10 which remained after the probability of Branch D (Manual Auxiliary Feedwater Isolation) was reduced to represent the failure probability of automatic isolation. This eliminated all MS cases from Table 3-10. The highest probability MS case was selected from Table 3-11 and shown in Table 3-12 for illustrative purposes.

Table 3-13 presents the frequency categories for each of the initiating events listed in Table 3-3. This table demonstrates that a full range of Anticipated Operating Occurrences (AOO) and accidents was considered.

Table 3-8

Scenario Occurrence Frequency Categorization
Condition 1 Events

CATEGORY	MODERATE	INFREQUENT	LIMITING FAULT			VERY LOW FREQUENCY
			1	2	3	
FREQUENCY	MAY OCCUR DURING A CALENDAR YEAR	MAY OCCUR DURING A PLANT LIFETIME	LOW PROBABILITY OF OCCURRING DURING A PLANT LIFETIME	VERY LOW PROBABILITY OF OCCURRING DURING A PLANT LIFETIME	EXCEEDINGLY LOW PROBABILITY OF OCCURRING DURING A PLANT LIFETIME	
EVENT SEQUENCES			PV MF	PV-Q PV-M PV-L MF-M MF-Q	PV-MQ PV-LQ PV-LM PV-LMQ PV-I TT TT-Q TT-M MF-QM	PV-P PV-N PV-NQ PV-MN PV-LN PV-IQ PV-IM PV-IMQ PV-IL PV-ILQ PV-ILM PV-H MF-J MF-JM MF-JQ TT-MQ TT-T TT-TQ TT-TM TT-S TT-SQ TT-SM

Table 3-9

Scenario Occurrence Frequency Categorization
Condition 2 Events - Manual AFW Isolation

CATEGORY	MODERATE	INFREQUENT	LIMITING FAULT			VERY LOW FREQUENCY
			1	2	3	
FREQUENCY	MAY OCCUR DURING A CALENDAR YEAR	MAY OCCUR DURING A PLANT LIFETIME	LOW PROBABILITY OF OCCURRING DURING A PLANT LIFETIME	VERY LOW PROBABILITY OF OCCURRING DURING A PLANT LIFETIME	EXCEEDINGLY LOW PROBABILITY OF OCCURRING DURING A PLANT LIFETIME	
EVENT SEQUENCES			MV-D MV-DF MV-DE MV-DEF	MV MV-F MV-E MV-EF MS-DEF	MS MS-E MS-F MS-EF MS-D MS-DF MS-DE	MV-DG MV-DFG MV-CD MV-CDF MV-CDE MV-CDEF MV-BD MV-BDF MV-BDE MV-BDEF MV-AD MV-ADF MV-ADE MV-ADEF

Table 3-10

Scenario Occurrence Frequency Categorization

Most Severe Sequences - Manual AFW Isolation for Condition 2 Sequences

CATEGORY	MODERATE	INFREQUENT	LIMITING FAULT			VERY LOW FREQUENCY
			1	2	3	
FREQUENCY	MAY OCCUR DURING A CALENDAR YEAR	MAY OCCUR DURING A PLANT LIFETIME	LOW PROBABILITY OF OCCURRING DURING A PLANT LIFETIME	VERY LOW PROBABILITY OF OCCURRING DURING A PLANT LIFETIME	EXCEEDINGLY LOW PROBABILITY OF OCCURRING DURING A PLANT LIFETIME	
CONDITION 1 EVENT SEQUENCES			PV	PV-M MF-Q	PV-MQ MF-QM	
CONDITION 2 EVENT SEQUENCES			MV-DEF	MV-EF MS-DEF	MS-DE	

Table 3-11

Scenario Occurrence Frequency Categorization

Condition 2 Events - Automatic AFW Isolation

CATEGORY	MODERATE	INFREQUENT	LIMITING FAULT			VERY LOW FREQUENCY
			1	2	3	
FREQUENCY	MAY OCCUR DURING A CALENDAR YEAR	MAY OCCUR DURING A PLANT LIFETIME	LOW PROBABILITY OF OCCURRING DURING A PLANT LIFETIME	VERY LOW PROBABILITY OF OCCURRING DURING A PLANT LIFETIME	EXCEEDINGLY LOW PROBABILITY OF OCCURRING DURING A PLANT LIFETIME	
			MV MV-F MV-E MV-EF	MS-F MS-EF	MS MS-E	MV-BD MV-BDF MV-BDE MV-BDEF MV-AD MV-ADF MV-ADE MV-ADEF

Table 3-12

Scenario Occurrence Frequency Categorization

Most Severe Sequences - Automatic AFW Isolation for Condition 2 Sequences
(As Selected from Table 3-10)

CATEGORY	MODERATE	INFREQUENT	LIMITING FAULT 1	LIMITING FAULT 2	LIMITING FAULT 3	VERY LOW FREQUENCY
FREQUENCY	MAY OCCUR DURING A CALENDAR YEAR	MAY OCCUR DURING A PLANT LIFETIME	LOW PROBABILITY OF OCCURRING DURING A PLANT LIFETIME	VERY LOW PROBABILITY OF OCCURRING DURING A PLANT LIFETIME	EXCEEDINGLY LOW PROBABILITY OF OCCURRING DURING A PLANT LIFETIME	
CONDITION 1 EVENT SEQUENCES			PV	PV-M MF-Q	PV-MQ MF-QM	
CONDITION 2 EVENT SEQUENCES			MV-EF	(MS-EF)*		

* SELECTED FROM TABLE 3-11

Table 3-13

Frequency Categories of Initiating Events

INITIATING EVENTS	FREQUENCY CATEGORIES
1. Decrease in Feedwater Enthalpy	
a) MFW Heater System Failure	I
b) AFW Delivery	LF1
2. Excess Feedwater Flow	
a) MFW Flow Increases Above That for a Particular Power Level	I
b) AFW Flow Increases Above That for a Particular Power Level	LF1
c) Reactor Power Decreases and MFW Flow Fails to Adjust	I
d) Reactor Power Decreases and AFW Flow Fails to Adjust	LF1
e) Turbine Trip and MFW Fails to Rampback	M + Add. Fail. = LF1
3. Excess Steam Flow	
• Steam Flow Increases Above That Req'd for a Particular Power Level	
a) Steam Line Break	LF2
b) MFW Line Break d/s of Check Valve	LF2
c) ADV Inadvertently Opens	LF1
d) TBV Inadvertently Opens	I
e) MSSV Inadvertently Opens	LF1
f) Excess Steam Flow Through Turbine	I
• Reactor Power Decreases and Steam Flow Fails to Adjust	
g) ADV is Open and Fails to Close	LF1
h) TBV is Open and Fails to Close	I
i) Turbine Fails to Decrease Steam Flow	I
j) Reactor Trips and Turbine Fails to Trip	M + Add. Fail. = LF3
k) Reactor Trips, Turbine Trips, and TBV Fails to Close After Quick Open or During Modulation	M + Add. Fail. = I
l) Hi Pressure Transient, MSSV Opens and Fails to Close	I + Add. Fail. = LF1
4. Large LOCA	
a) Large Pipe Break	LF2
5. Small LOCA	
a) Non-Isolable Pipe Break	I
b) Isolable Pipe Break (Letdown Line)	LF1
c) RCP Seal Failure	I
d) PORV Inadvertently Opens	LF1
e) RCS Overpressure Scenario and One PORV/PSV Fails to Reclose	I + Add. Fail. = LF1
f) SG Tube Rupture	I
6. Pressurizer Pressure Control Failures	
a) Spurious Main Spray Actuation	LF1
b) Spurious Aux. Spray Actuation+	
c) Pressure Transient Actuates Main Spray, Spray Fails to Decrease	M + Add. Fail. = I
d) Excess Main Spray During a Controlled Depressurization or Boron Mixing	LF1
e) Excess Aux. Spray During a Controlled Depressurization	LF1
7. Inadvertent SIAS When Below Shutoff Head	
a) Failure to Block SIAS Setpoint	I
b) Spurious SIAS	I
8. Decrease in Charging Enthalpy	
a) PLCS Failure (Max. Charging, Loss of Letdown)	LF1
9. Maximum Shutdown Cooling	LF1
+ Spurious auxiliary spray actuation is not considered plausible due to complexity of actuation procedure.	

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6. IEEE STD500-1977, IEEE Guide to the Collection and Presentation of Electrical, Electronic, and Sensing Component Reliability for Nuclear Power Generating Stations.

7. NPRDS 1977 Annual Reports of Cumulative System and Component Reliability, September, 1978.

8. St. Lucie - 2 FSAR.

9. San Onofre (SONGS) FSAR.

10. Calvert Cliffs FSAR.

Enclosure 2

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