U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No.: 50-423/88-03

Docket No.: 50-423

License No.: NPF-49

Licensee: Northeast Nuclear Energy Company
P. O. Box 270
Hartford, Connecticut 06101-0270

Facility: Millstone Nuclear Power Station, Unit 3

Meeting Location: NRC Region I, King of Prussia, Pennsylvania

Meeting Date: March 8, 1988

Approved by: Ed C McCabe, Jr 3/14/88
E. C. McCabe, Chief, Reactor Projects Section 1B Date

Meeting Summary: The Enforcement Conference was held to discuss the findings of Inspection Report 50-423/88-03. The topics discussed at the meeting dealt with the root cause analysis and corrective actions for an overpressure transient that occurred at low temperature without the required overpressure protection systems being in service. Additional topics were discussed, including generic implications of the event and predicted plant response without operator action.

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DETAILS

1. Participants

   Northeast Nuclear Energy Company (NNECo)

   W. Romberg, Vice President, Nuclear Operations
   S. Scace, Station Superintendent
   C. Clement, Millstone 3 Superintendent
   G. Van Noordennen, Supervisor, Generation Facilities Licensing
   J. Ely, Supervisor, Component Engineering
   J. Harris, Acting Operations Supervisor
   R. Joshi, Licensing

   U.S. Nuclear Regulatory Commission

   W. Kane, Director, Division of Reactor Projects
   L. Bettenhausen, Chief, Projects Branch No. 1
   E. McCabe, Chief, Reactor Projects Section 1B
   W. Raymond, Senior Resident Inspector
   S. Barber, Resident Inspector
   V. Pritchett, Reactor Engineer, Reactor Projects Section 1B
   R. Ferguson, Millstone 3 Project Manager, NRR
   J. Strosnider, Chief, Materials and Processes Section
   J. Gutierrez, Regional Counsel
   D. Holody, Enforcement Coordinator

2. Purpose

   The Enforcement Conference was initiated by NRC Region I to discuss the January 19, 1988 overpressure event without the required overpressure protection systems operable. Northeast Nuclear Energy Company was requested to identify and describe the event's root cause; broader implications; generic concerns; lessons learned; and, both their short and long term corrective actions. To ensure that the licensee was aware of the specific NRC concerns, Attachment 1 to this report had previously been provided to the licensee during the February 18, 1988 exit meeting for Special Inspection 50-423/88-03.

3. Discussion

   NRC management made a few brief opening remarks. NNECo then made a presentation based on the outline included as Attachment 2. NNECo agreed that the events and technical issues described in the NRC January 18-29, 1988 special inspection report were accurate.

During the RHR isolation event on January 19, a potentially significant event existed when the operating charging pump continued to run after the established RHR letdown path was secured. Operator actions at the time were prompt and appropriate. A relief valve on the letdown line downstream of the orifices with a setpoint of 600 psig was capable of relieving RCS pressure during the RHR isolation. The engineering evaluation discussed above demonstrates that adequate margins remained to the reactor vessel limits. Although the RHR isolation and RCS pressure increase event on January 19 is a significant event that highlighted weaknesses in operating procedures, operator and technician knowledge, drawings, and controls, the actual safety impact of the event was minimal.

One item that warrants further licensee and NRC staff consideration because of the potential significance of the isolation event is the question of "what would the outcome of the transient have been if the operators had not acted to mitigate the pressure increase." This question was presented to the licensee for consideration in his evaluation of the event.

8.0 Summary of Findings

Listed below are some issues to be discussed by the NRC and the licensee at an enforcement conference.

-- Evaluation of what the January 19th pressure transient outcome would have been assuming no operator action.

-- Adequacy of drawings and of training personnel in their use, especially in interrelationships between drawings from different suppliers (e.g., architect-engineer and NSSS vendor drawings).

-- Assuring that work affecting SSPS and other safety-related equipment is performed by personnel having appropriate training, qualifications, and experience.

-- Adequacy of controls allowing work on turbine bypass control circuits without tags for pulling fuses, and the duty senior control room operator on January 19th approving work without reviewing the applicable AWO.

-- Adequacy and utility of Technical Specification curves prescribing pressure-temperature limits.

-- Adequacy of procedures and training on the COPS design and interface with the SSPS.

-- Use of alarm response procedures in post-transient reviews.

-- Appropriateness of provisions that allow up to 31 days to elapse between surveilling a system for operability and placing it in operation, and assuring adequacy of supporting equipment configuration in the interim and afterwards.

-- Adequacy of the surveillance used to determine COPS operability.

-- Lack of positive indication of COPS arming.

-- Development of a specific low temperature overpressure protection procedure, or providing additional guidance in existing procedures to assure that operators are properly aware of low pressure overprotection features' status and that they follow-up quickly, appropriately, and fully to transients and to losses of overpressure protection.

-- Assumption by the operators that COPS was the available overpressure system when, in fact, it was not operable. This resulted in the violation of TS 3.4.9.3 which requires at least one overpressure system to be available at all times when less than 350 degress F (VIO 88-03-02).

-- The failure to notify the NRC about the overpressure transient via the Emergency Notification System (ENS) within four hours (VIO 88-03-03).

ATTACHMENT 2

Docket No. 50-423

NRC ENFORCEMENT CONFERENCE

Millstone Unit 3

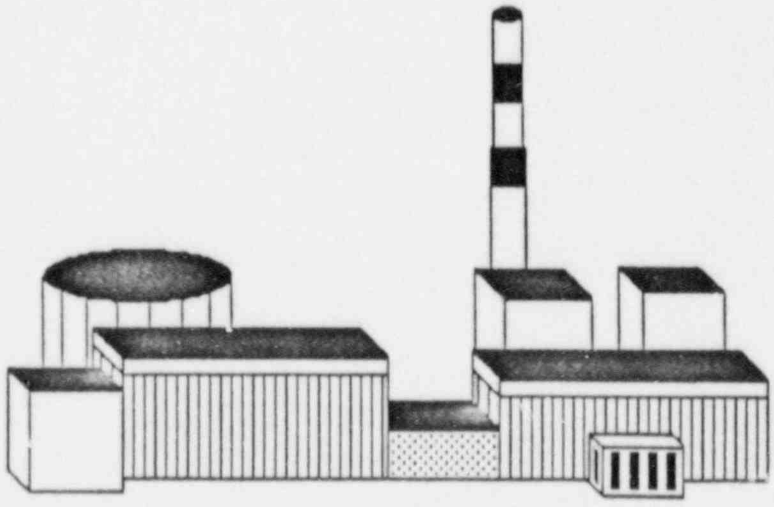
Cold Overpressure Protection System
Fails to Operate During a Pressure
Transient

(Incident Date: January 19,1988)

Presented By:

Northeast Nuclear Energy Company

March 8, 1988



NRC ENFORCEMENT CONFERENCE

- EVENT SUMMARY
- PLANT CONDITIONS, JANUARY 16,1988
- ARMING COPS, JANUARY 16, 1988
- PRESSURE TRANSIENT, JANUARY 19,1988
- POST TRANSIENT REVIEW
- SAFETY SIGNIFICANCE
- ANALYSIS - RHR ISOLATION
- CORRECTIVE ACTION - RHR
- ANALYSIS - ARMING COPS
- CORRECTIVE ACTION - COPS
- REPORTABILITY
- GENERIC IMPLICATIONS
- CONCERNS
- SUMMARY

EVENT SUMMARY

On January 19, 1988 with the plant solid in cold shutdown, temperature 135 °F, pressure 350 psia, a pressure transient occurred which challenged the COLD Overpressure Protection System (COPS). COPS failed to operate when required. The pressure transient was mitigated by manual operator action.

PLANT CONDITIONS, JANUARY 16,1988

- COLD SHUT DOWN, 135 °F, 350 psia
- RCS solid
- Loops B & C filled and vented
- Loops A & D drained and isolated for RCP locking cap repair
- Reactor Coolant Pump C running
- Residual Heat Removal (RHR) A & B operable
- Overpressure protection provided by RHR relief valves (440 psia)

ARMING COPS, JANUARY 16, 1988

- Technical Specification surveillances reviewed
  - Protection set Analog Channel Operational Test (31 day)
  - Reactor Protection Channel Calibration (18 month)
  - PORV block valve open (72 Hour)
- Verified support systems operable by discussion with I&C tech
- Armed COPS by placing the PORV switches to ARM position
- Overpressure protection provided by COPS
- RHR train A tagged for system outage

PRESSURE TRANSIENT, JANUARY 19,1988

- I&C calibrating the Main Steam Dump Valves
- P-12 Interlock must be disabeled to stroke the steam dumps
- I&C tech A researched the fuse to be pulled to disable P-12
- I&C Tech B reviewed and concured with the fuse pull
- I&C tech A discussed the fuse pull with the SCO
- Fuse 61 pulled in the reactor protection system
  - RHR suction valve starts to close
  - RHR low flow alarm
- Operator action to mitigate the pressure transient
  - RHR pump stopped
  - RCS pressure increasing
  - Stopped the charging pump
  - Increased letdown flow
  - Reestablished RHR

POST TRANSIENT REVIEW

- The fuse pulled not only supplied power to the steam...
- Deenergization of the relays closed the RHR suction valves...
- Investigation showed COPS setpoints were attained but the...
- The failure of COPS to operate was do to the lack of input from...

SAFETY SIGNIFICANCE

- COPS failure challenged overpressure limits per 10 CFR 50...
- Actual Peak pressure was 526 psia.
- Actual low temperature was 134 °F.
- COPS setpoint curves are conservative. They include...
- Appendix G limits not exceeded when conservatisms removed.

ANALYSIS - RHR ISOLATION

- Electrical Schematic for steam dump does not show branch loads.
- SSPS vendor drawing showing all fuse load is complex.
- I&C Technician A not qualified in SSPS.
- I&C Technician A did request assistance, but not formally.
- I&C Technician B did not fully review the consequences of the...
- I&C Technician A did not request assistance from foreman.
- The procedure for setting steam dump positioners was a generic...
- ROOT CAUSE
  - Incorrect use of drawings by a technician not qualified in...

CORRECTIVE ACTION - RHR

- Memo issued to discuss proper use of drawings.
- Tail board discussion of incident with all departments.
- Training to incorporate lessons learned - April 15, 1988.
- Policy set to ensure only qualified personnel work on SSPS.
- Procedure for setting steam dump positioners - May 15, 1988.
- Conduct of Maintenance procedure to inlcude correct line of...
- Train additional I&C personnel for SSPS qualification.

ANALYSIS - ARMING COPS

- GOP to arm COPS was not adequate for plant condition.
- Surveilances to arm COPS did not ensure pre-requisites.
- Relationship between SSPS and COPS not fully understood.
- No direct indication that COPS is armed.
- Support system reviews for arming COPS were inadequate.

CORRECTIVE ACTION - COPS

- Procedure for arming COPS - complete.
- Annunciator response for SSPS General Warning - complete.
- Review feasability of providing direct COPS armed indication -...
- Operator Training will be revised to re-emphasize the...
- Tag SSPS test switch to SS when COPS is armed.

REPORTABILITY

- Initially classified as not reportable.
- Reclassified as reportable per 10 CFR 50.73(a)(2)(vii).
- Should have been reportable via ENS per 10 CFR 50.72.
- Resident was initially informed on January 19, 1988.
- Resident was kept informed during investigation.

GENERIC IMPLICATIONS

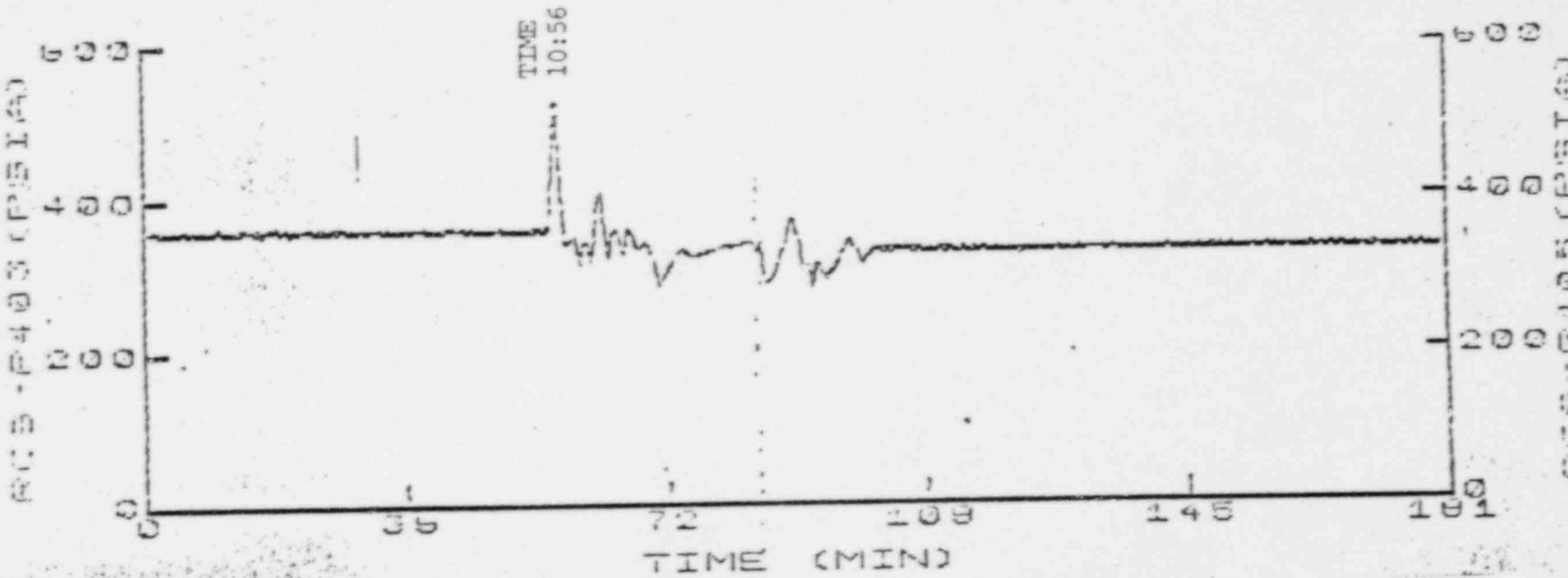
- CY has mechanical relief valves for cold overpressure.
- MP2 - COPS is a single setpoint for PORV's.
- Use of drawings for pulling fuses at MP1, MP2 and CY.
- Qualified personnel only to remove fuses in I&C systems with...
- Review other potential procedures for fuse pulls.

CONCERNS

- Pressure transient outcome assuming no operator action.
- Adequacy of drawings and of training personnel in their use.
- Work on safety-related equipment performed by trained...
- Adequacy of controls of work.
- Adequacy of curves prescribing pressure-temperature limits.
- Procedures/training on the COPS design/interface with SSPS.
- Surveillance interval/equipment configuration control.
- Are procedures inadequate to insure COPS operability?...
- Development of a specific low temperature overpressure...
- Use of alarm response procedures in post-transient reviews.
- Lack of positive indication of COPS arming.
- The failure to notify the NRC within four hours.

SUMMARY

- COPS not fully operable for 63 hours January 16 - 19.
- RCS was protected against overpressure until RHR suction valve...
- Max RCS pressure 526 psia.
- Corrective Action extensive and prompt.



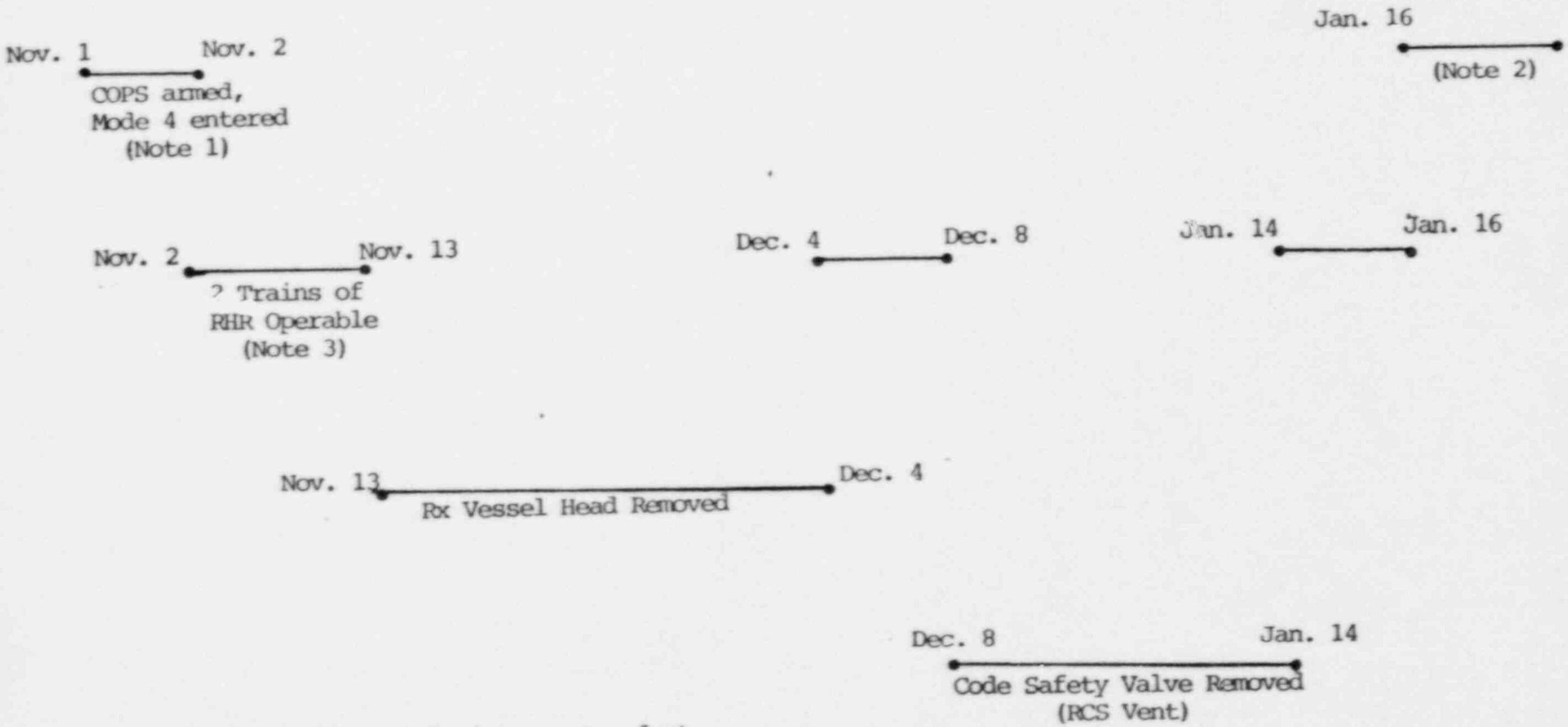
MAXIMUM
PRESSURE
ON PT403 =
526 PSIA

MAXIMUM
PRESSURE
ON PT405 =
524 PSIA

RCS PRESSURE TRANSIENT ON 1/19/88

FIGURE 1

OVERVIEW OF COLD OVERPRESSURE PROTECTION DURING THE MILLSTONE UNIT 3 OUTAGE

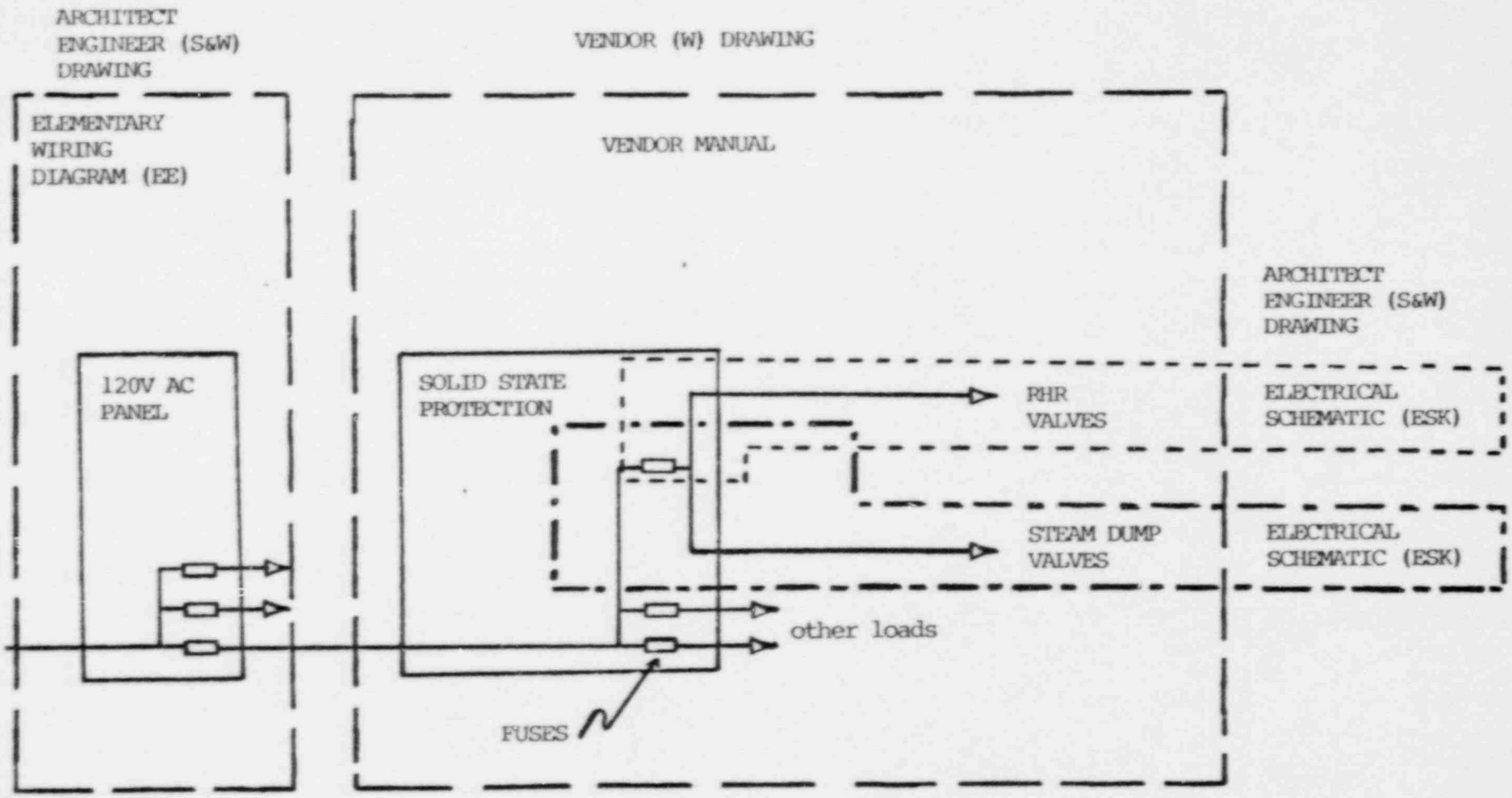


Notes: 1) On Nov. 1, 1987, mode 4 was entered at 1730; COPS had been armed at 1655. But, there is an SS log entry that states "SSPS restored" at 1910. Therefore, COPS may not have been available for 3 hours and 40 minutes.

2) Time period of the events concerning loss of RHR, SSPS, and COPS.

3) 3 loops isolated (after both trains of RHR available).

FIGURE 2



DRAWING INTERFACES

FIGURE 3

STEAM DUMP ELECTRO-PNEUMATIC CONTROL (TYPICAL OF NINE)

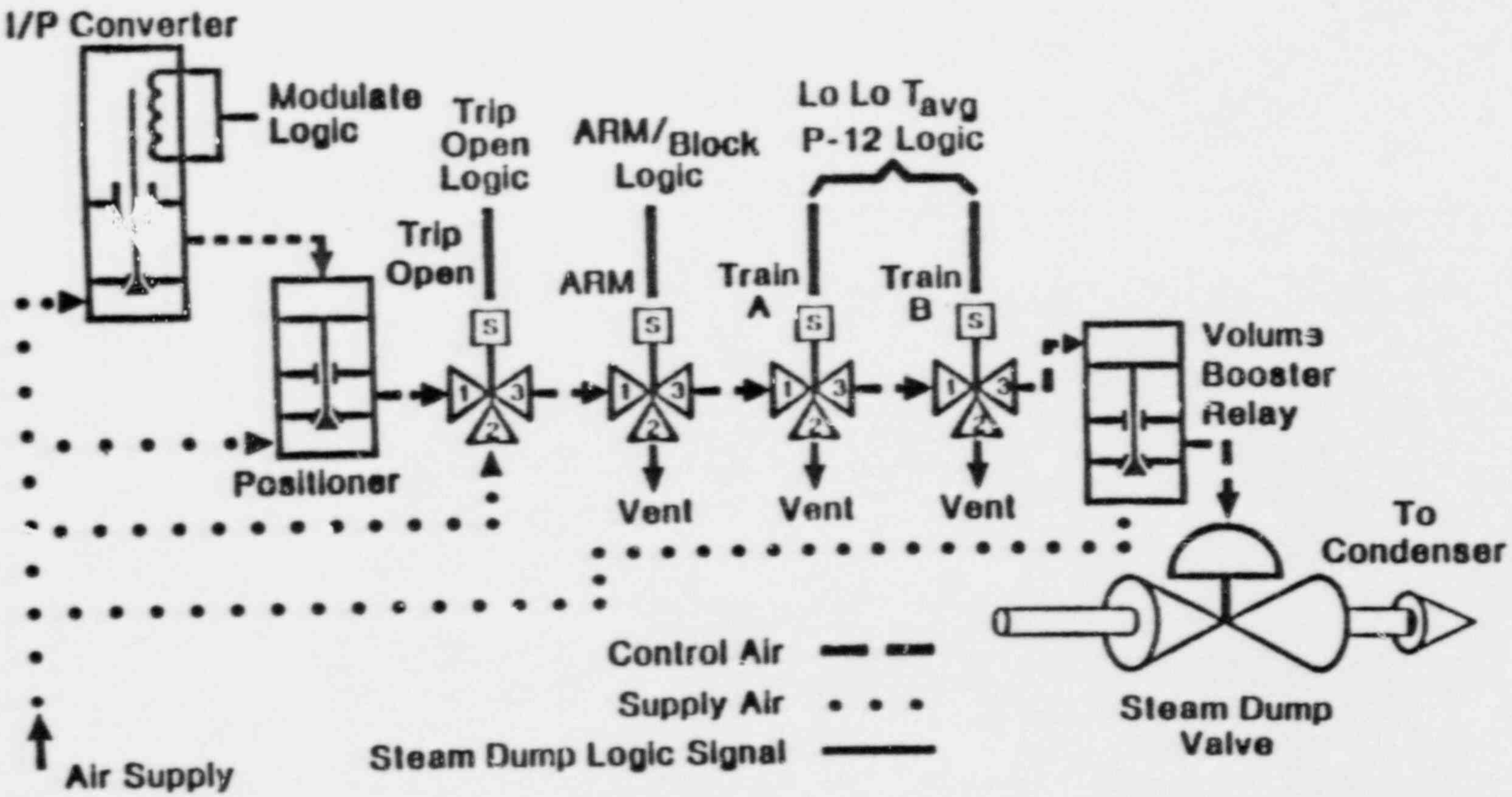


FIGURE 4

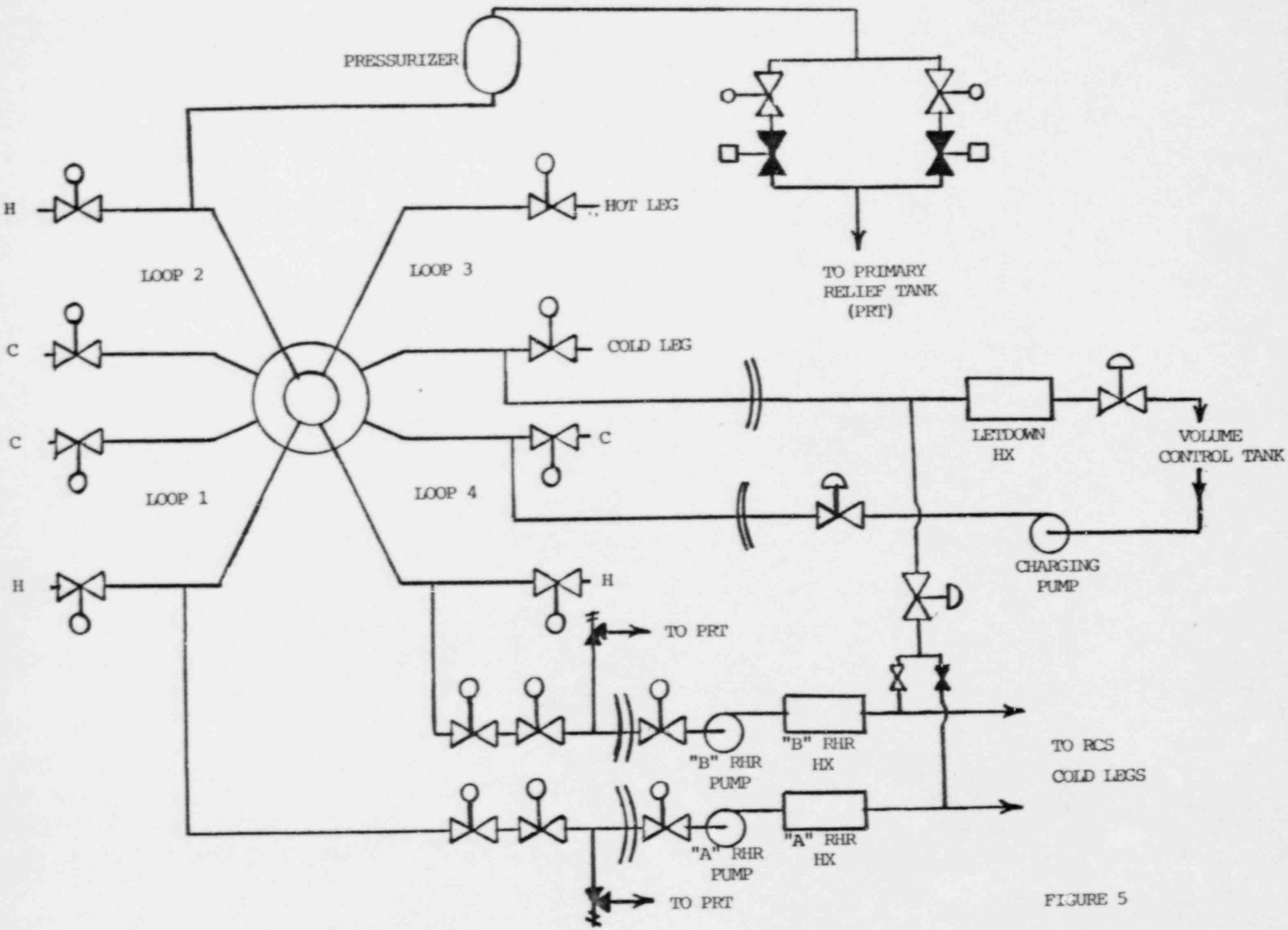

PRESSURIZER
H
HOT LEG
LOOP 2
LOOP 3
TO PRIMARY
RELIEF TANK
(PRT)
C
COLD LEG
C
C
LETDOWN
HX
VOLUME
CONTROL TANK
LOOP 1
LOOP 4
CHARGING
PUMP
H
H
TO PRT
"B" RHR
HX
"B" RHR
PUMP
TO RCS
COLD LEGS
"A" RHR
PUMP
"A" RHR
HX
TO PRT

FIGURE 5

COPS FUNCTIONAL DIAGRAM

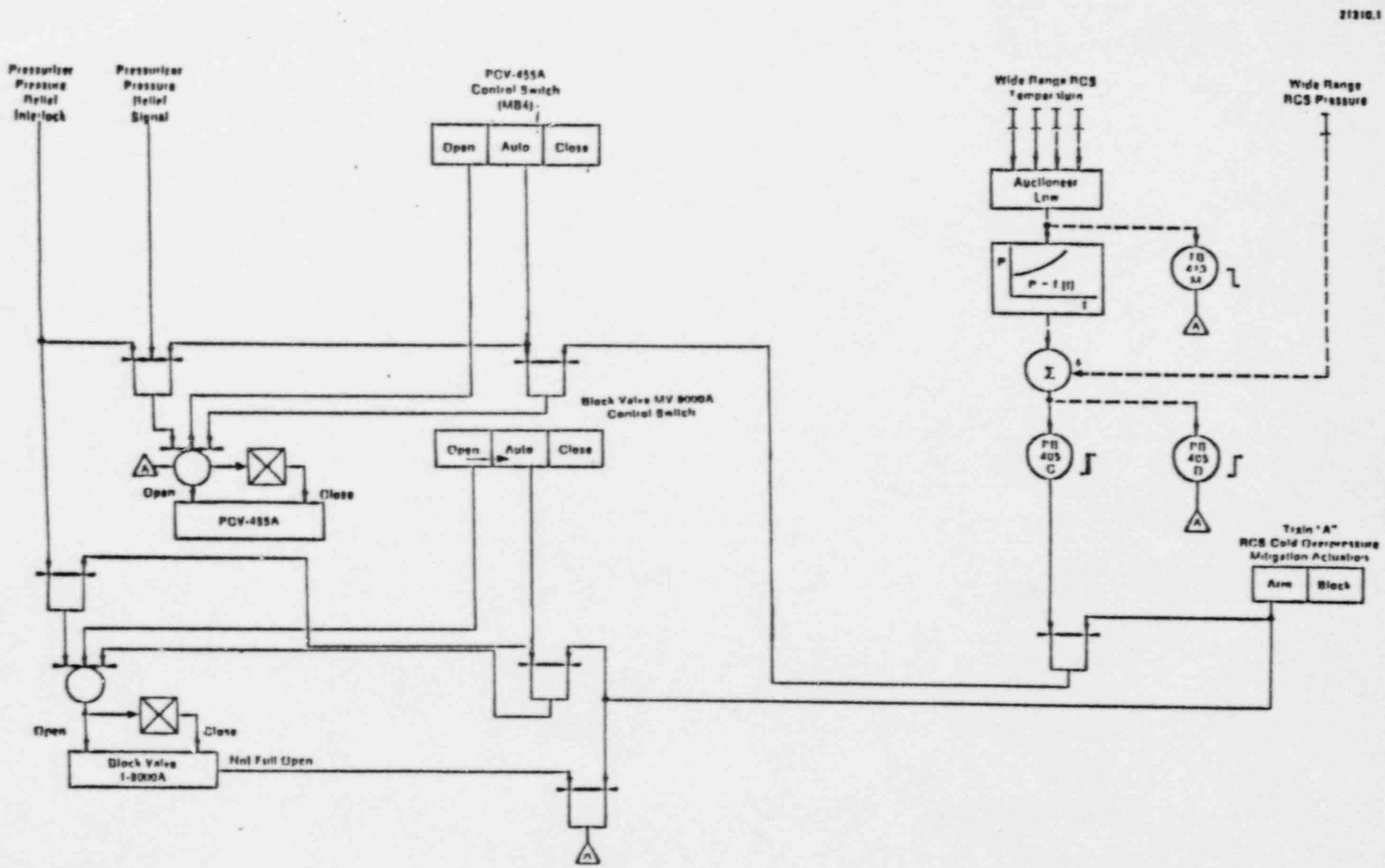


FIGURE 6

NORTHEAST NUCLEAR ENERGY COMPANY

STRUCTURAL INTEGRITY ISSUES FROM THE JANUARY 19 MILLSTONE 3 PRESSURE TRANSIENT

- o ADEQUACY AND UTILITY OF TECHNICAL SPECIFICATION CURVES:

  - -- BACKGROUND--DEPRIVATION AND MARGINS.

- o EVALUATION OF THE JANUARY 19 PRESSURE TRANSIENT ASSUMING NO OPERATOR ACTION:

  - -- USE OF ASME SECTION XI, APPENDIX E VERSUS "OPERATIONAL" TECHNICAL SPECIFICATION CURVES.

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MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL : PLATE METAL
COPPER CONTENT : CONSERVATIVELY ASSUMED TO BE 0.10 WT%
PHOSPHORUS CONTENT : 0.010 WT%
RT_{NDT} INITIAL : 60°F
RT_{NDT} AFTER 10 EFPY : 1/4T, 122°F
3/4T, 101°F

CURVE APPLICABLE FOR HEATUP RATES UP TO 60°F/HR FOR THE SERVICE PERIOD UP TO 10 EFPY AND CONTAINS MARGINS OF 10°F AND 60 PSIG FOR POSSIBLE INSTRUMENT ERRORS

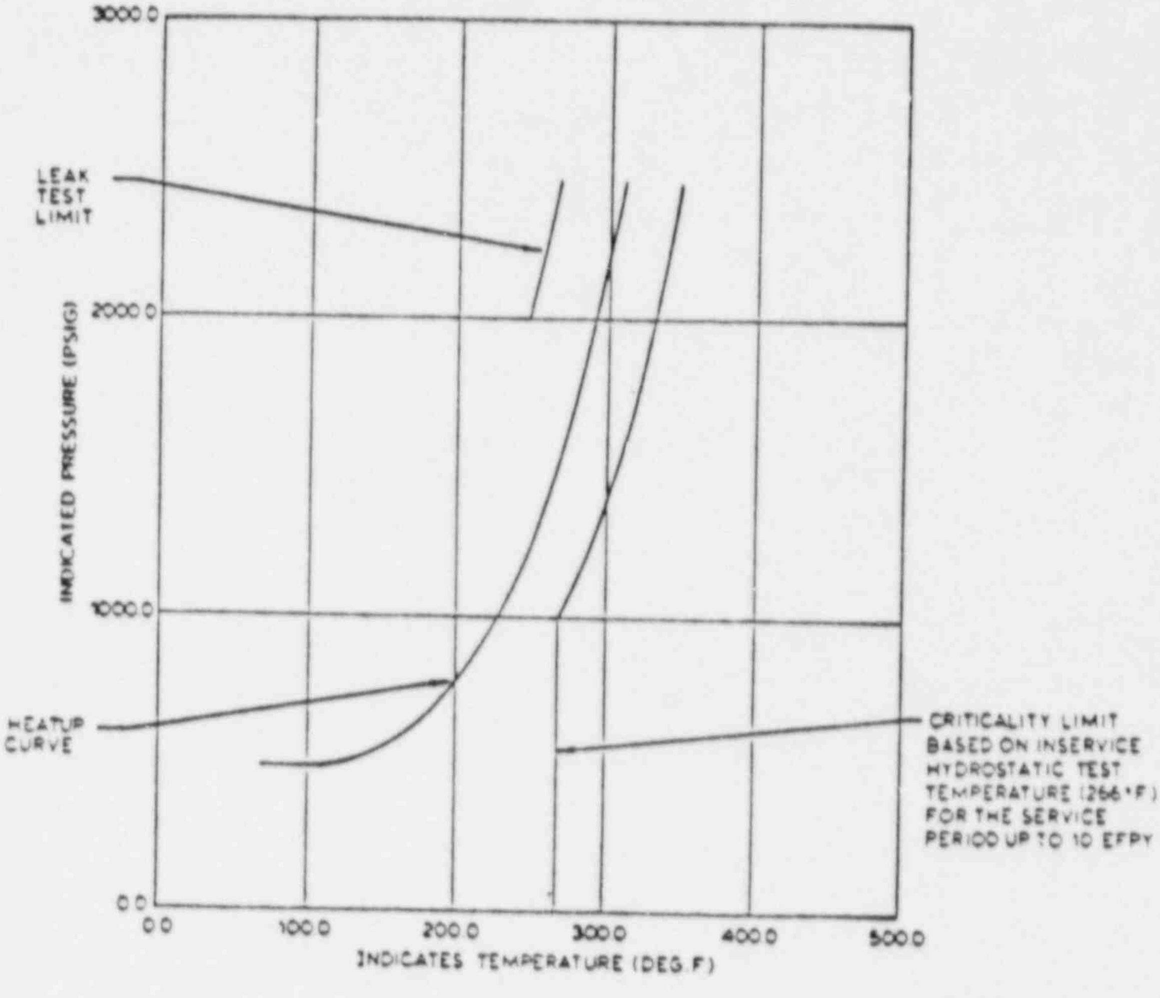


FIGURE 3.4-2

REACTOR COOLANT SYSTEM HEATUP LIMITATIONS - APPLICABLE UP TO 10 EFPY

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL	PLATE METAL
COPPER CONTENT	CONSERVATIVELY ASSUMED TO BE 0.10 WT %
PHOSPHORUS CONTENT	0.010 WT %
RT_{NDT} INITIAL	60°F
RT_{NDT} AFTER 10 EFPY	1/4 T, 122°F 3/4 T, 101°F

CURVE APPLICABLE FOR COOLDOWN RATES UP TO 100°F/HR FOR THE SERVICE PERIOD UP TO 10 EFPY AND CONTAINS MARGINS OF 10°F AND 60 PSIG FOR POSSIBLE INSTRUMENT ERRORS

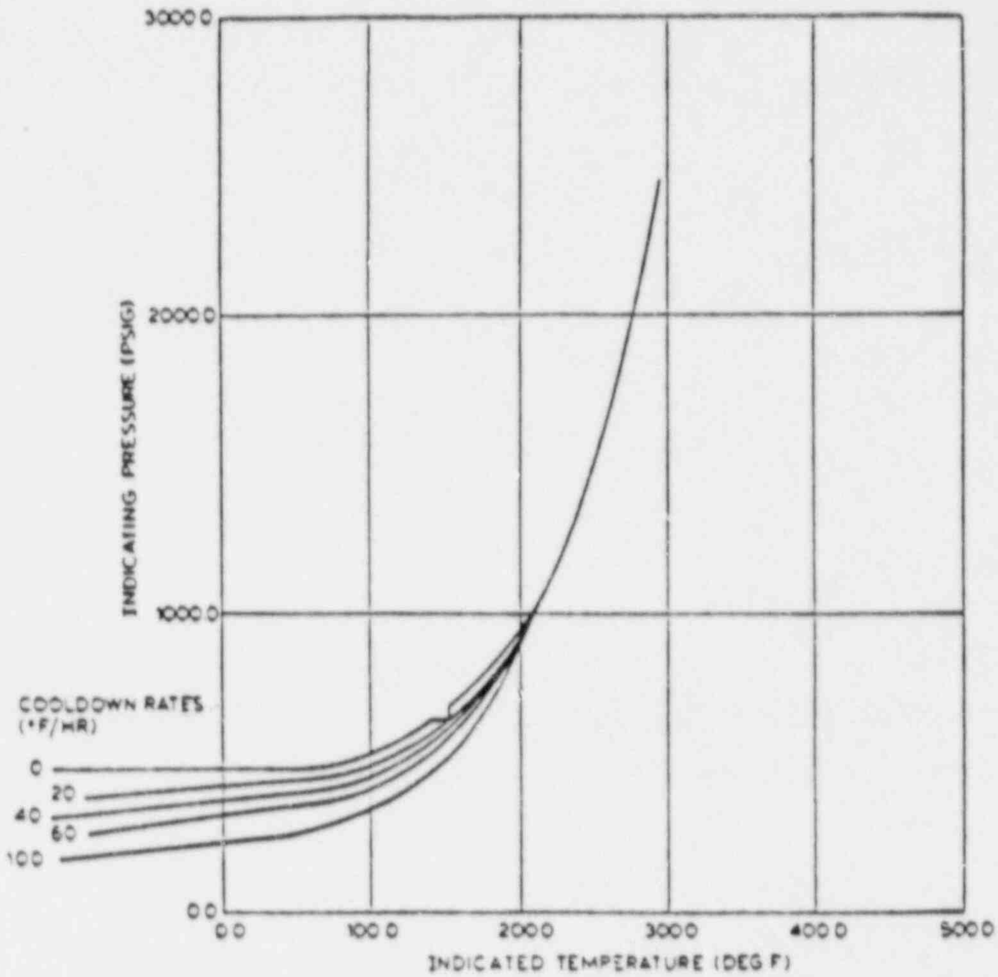
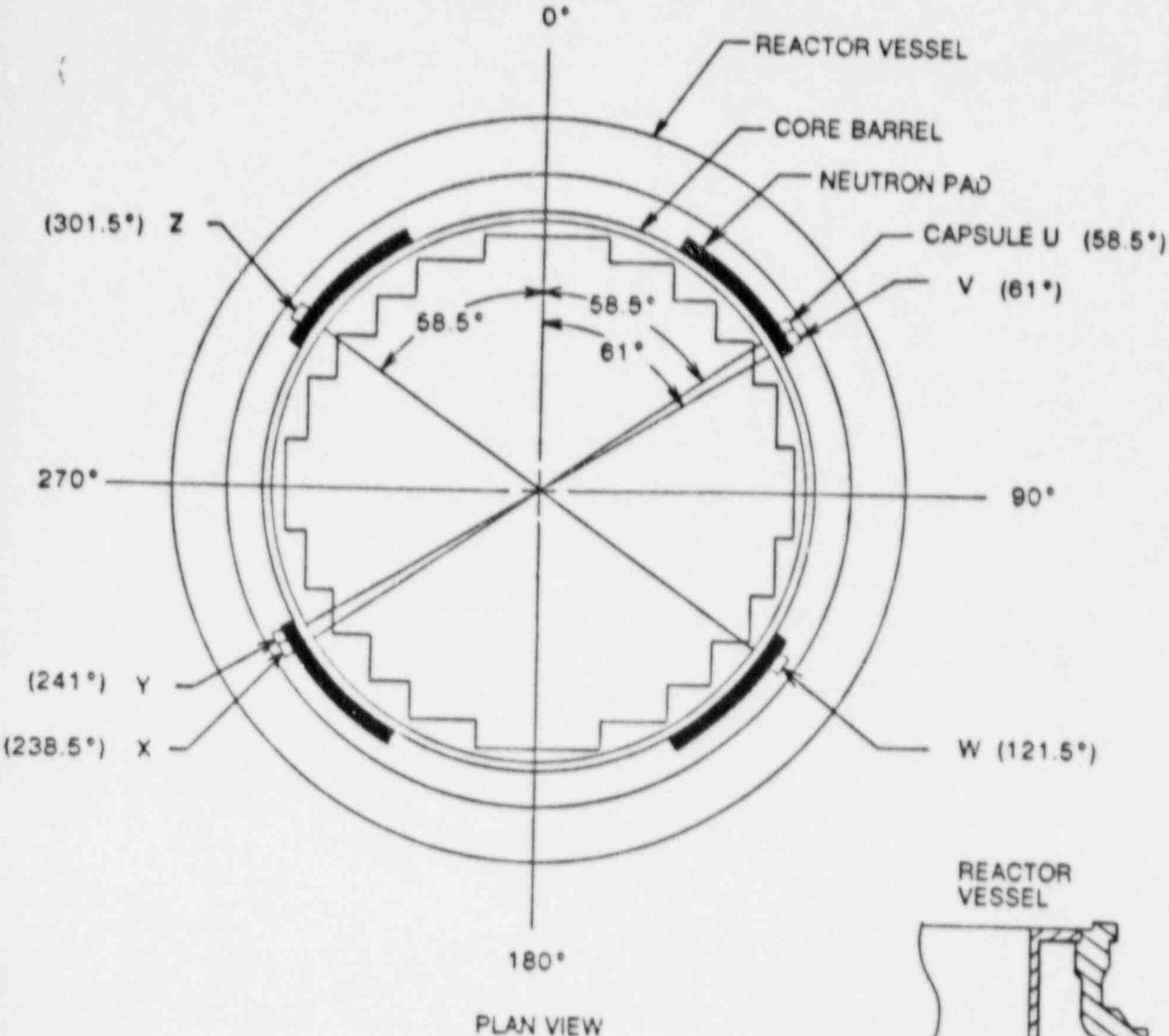


FIGURE 3.4-3

REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS - APPLICABLE UP TO 10 EFPY



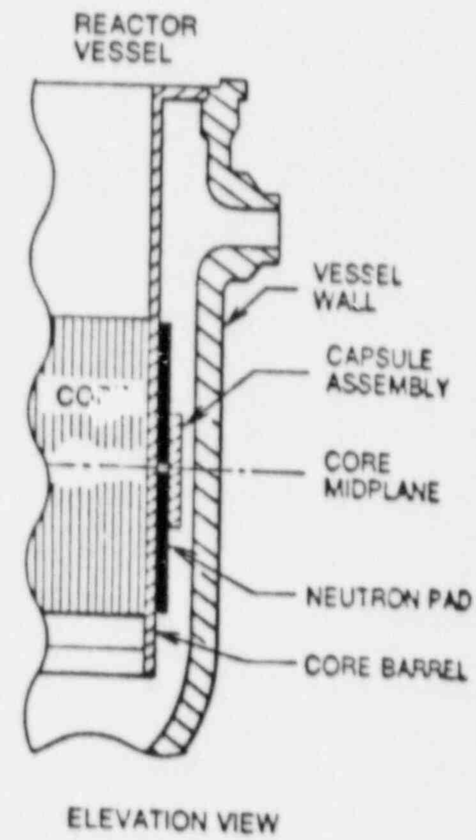


FIGURE 1-1. LOCATION OF THE IRRADIATION TEST CAPSULES IN THE MILLSTONE UNIT NO. 3 REACTOR VESSEL

SECTION 4
POSTIRRADIATION TESTING

4-1. CAPSULE REMOVAL

The first capsule (Capsule U) should be removed at the end of the first core cycle (1st refueling) as shown in Table 4-1. Subsequent capsules should be removed at 5, 9, and 15 EFPY (Effective Full Power Years) as indicated. Each specimen capsule, removed after exposure, will be transferred to a postirradiation test facility for disassembly and testing of all the specimens.

TABLE 4-1
SURVEILLANCE CAPSULE REMOVAL SCHEDULE

Capsule Identification	Orientation of Capsules[a]	Lead Factor[b]	Removal Time	Expected Capsule Fluence (n/cm^2)
U	58.5°	4.00	1st Refueling	3.6×10^{18}
Y	241 °	3.69	5 EFPY	1.3×10^{19}[c]
V	61 °	3.69	9 EFPY	2.4×10^{19}[d]
X	238.5°	4.00	15 EFPY	4.3×10^{19}
W	121.5°	4.00	Stand-By	———
Z	301.5°	4.00	Stand-By	———

a. Reference Irradiation Capsule Assembly Drawing, Figure 2-4.

b. The factor by which the capsule fluence leads the vessels maximum inner wall fluence.

c. Approximate Fluence at ¼-wall thickness at End-of-Life.

d. Approximate Fluence at vessel inner wall at End-of-Life.

TABLE A-4

T_{NDT}, RT_{NDT} AND UPPER SHELF ENERGY FOR THE MILLSTONE UNIT NO. 3 REACTOR PRESSURE VESSEL CORE REGION SHELL PLATES AND WELD METAL

Material	T_{NDT} [a] [b]		RT_{NDT}		Upper Shelf[a] [c] Energy	
	(°C)	(°F)	(°C)	(°F)	(J)	(ft lb)
Intermediate Shell Plates:						
B9805-1	−40	−40	16	60	126	93
B9805-2	−51	−60	−12	10	122	90
B9805-3	−40	−40	−18	0	145	107
Lower Shell Plates:						
B9820-1	−46	−50	−12	10	104	77
B9820-2	−34	−30	4	40	103	76
B9820-3	−34	−30	− 7	20	108.5	80

a. Data obtained from Combustion Engineering, Inc. Reactor Vessel Material Certification Reports.

b. Drop weight data obtained from the transverse material properties (normal to the major working direction).

c. From impact data obtained from the transverse material properties (normal to the major working direction).

Material	T_{NDT} [d]		RT_{NDT}		Upper Shelf[d] Energy	
	(°C)	(°F)	(°C)	(°F)	(J)	(ft lb)
Intermediate and Lower Shell Longitudinal Weld Seams and Closing Girth Weld Seam (Weld Wire Heat No. 4P6052, Linde 0091 Flux, Lot No. 0145)	−46	−50	−46	−50	271	200

d. Data obtained from Combustion Engineering, Inc. Wire/Flux Weld Deposit Material Certification Test No. 1332.

TABLE 3-5

SUMMARY OF MILLSTONE UNIT NO. 3 REACTOR PRESSURE VESSEL IMPACT TEST RESULTS FOR INTERMEDIATE SHELL PLATE B9805-1 AND CORE REGION WELD AND HEAT-AFFECTED-Z[illegible] MATERIAL

Material	Upper Shelf Energy (USE)		41-J (30-ft lb) Index Temp		68-J (50-ft lb) Index Temp		0.89 mm (35 mils) Index Temp	
	(J)	(ft lb)	(°C)	(°F)	(°C)	(°F)	(°C)	(°F)
Plate B9805-1 (Longitudinal Orientation)	180	133	−15	5	2	35	− 1	30
Plate B9805-1 (Transverse Orientation)	151	111	−15	5	−16	60	2	35
Weld Metal	194	143	−37	− 35	−26	−15	−29	− 20
Heat Affected Zone	193	142	−90	−130	−71	−95	−76	−105

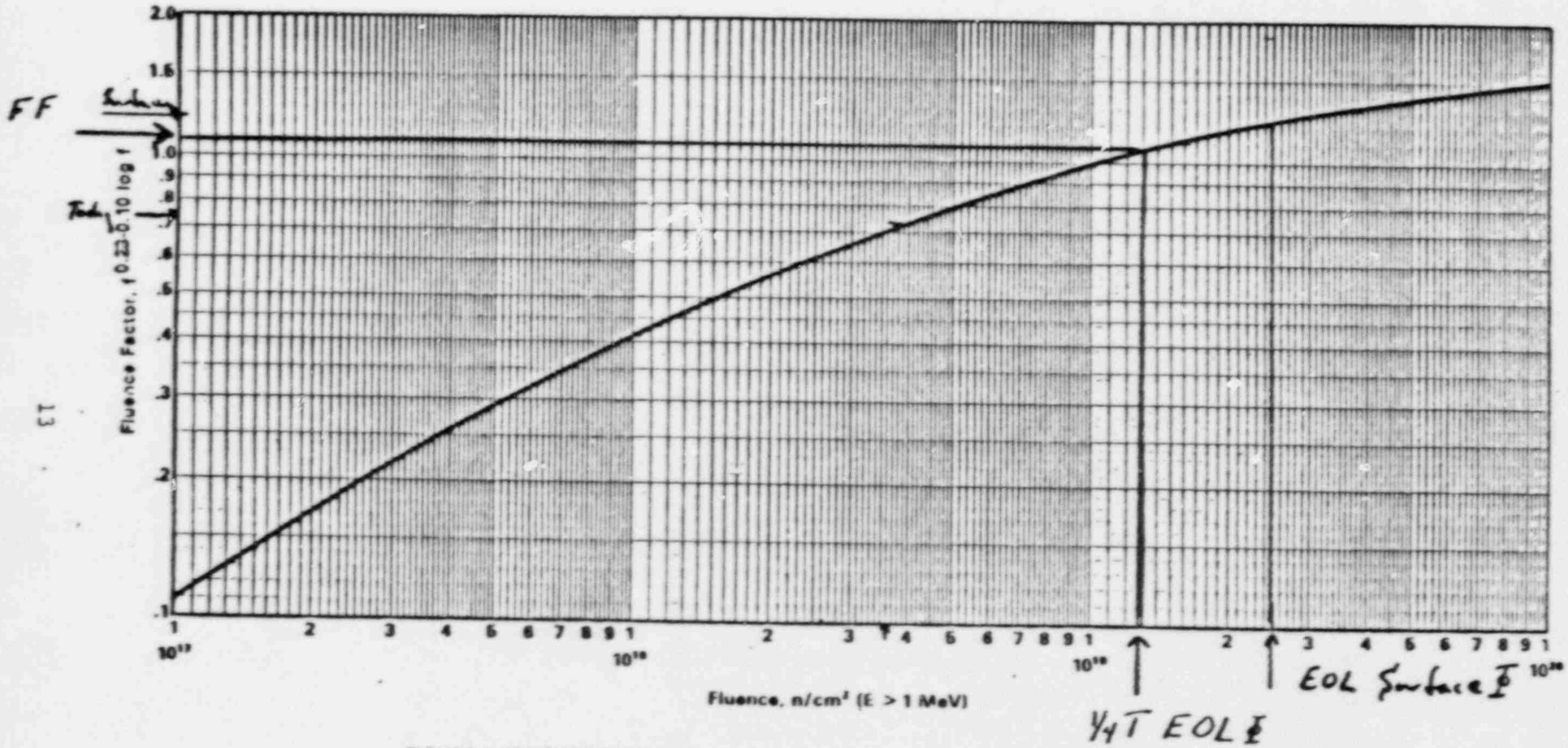


FIGURE 1 FLUENCE FACTOR FOR USE IN EQUATION 2, THE EXPRESSION FOR ΔRT_{NDT}

Weld - 1/4T EOL $RT_{NDT} = -50 + (24)(1.1) + 28 = 4.4°F$

RM $= 60 + 31\,(1.1) + 17 = 111°F$

NORTHEAST UTILITIES SERVICE COMPANY
NUCLEAR ENGINEERING AND OPERATIONS GROUP
GENERATION MECHANICAL ENGINEERING

ASME SECTION III CLASS 2 AND 3
AND ANSI B31.1.0 PIPING ANALYSIS

PROJECT ASSIGNMENT: 84-003

CALCULATION NUMBER: 84-003-371 GP

PLANT: CONNECTICUT YANKEE

TITLE: PRESSURE TEMPERATURE LIMIT CURVES FOR 22 EFPY

QA CATEGORY 1

REVISION 0 [signature] 5/16/85

PREPARED BY	DATE	REVIEWED BY	DATE
N.F. AZEVEDO	4/24/85	[signature]	5/16/85
REVIEW METHOD FULL REVIEW		APPROVED BY [signature]	5/16/85

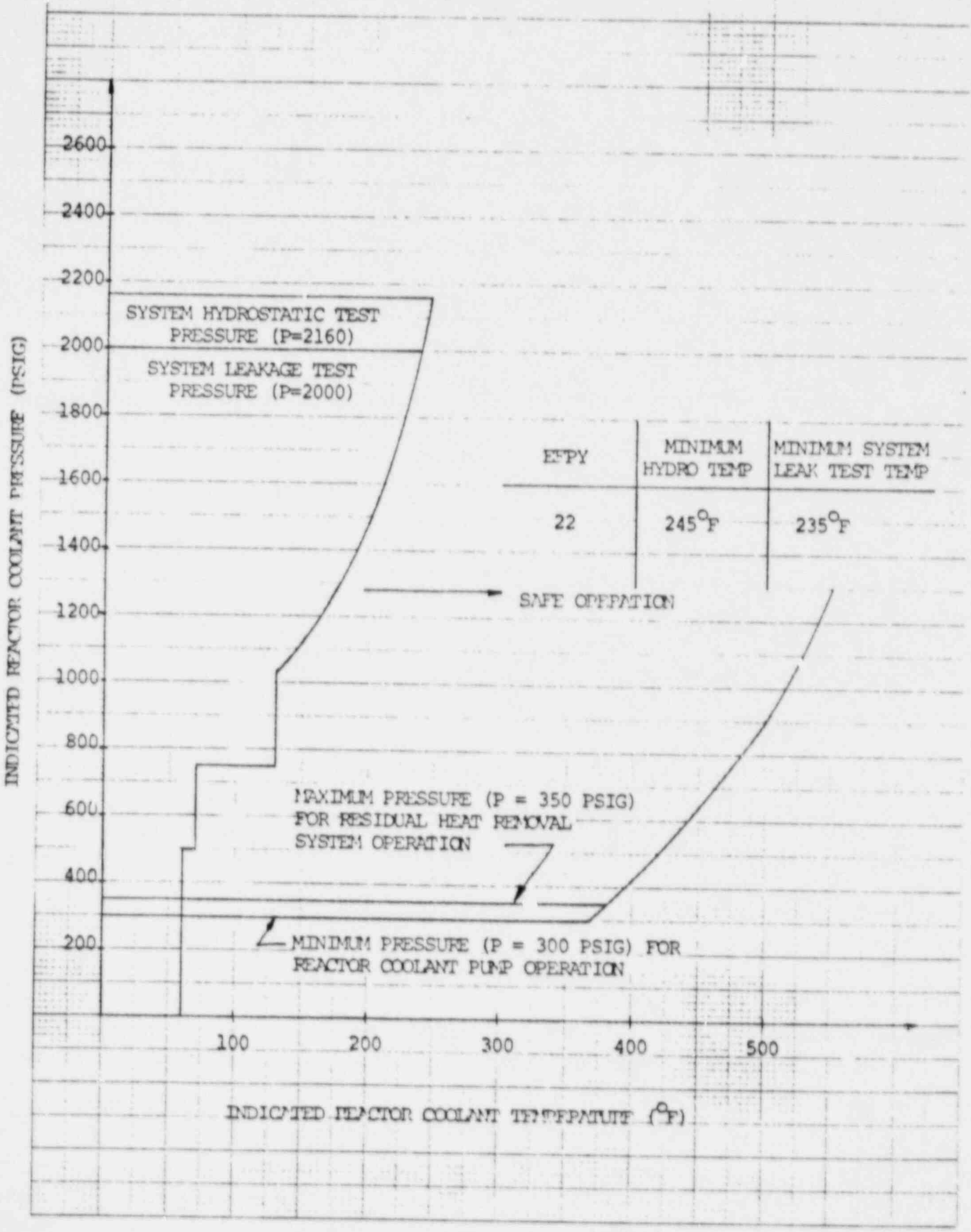
REVISION 1

PREPARED BY	DATE	REVIEWED BY	DATE
REVIEW METHOD		APPROVED BY	DATE

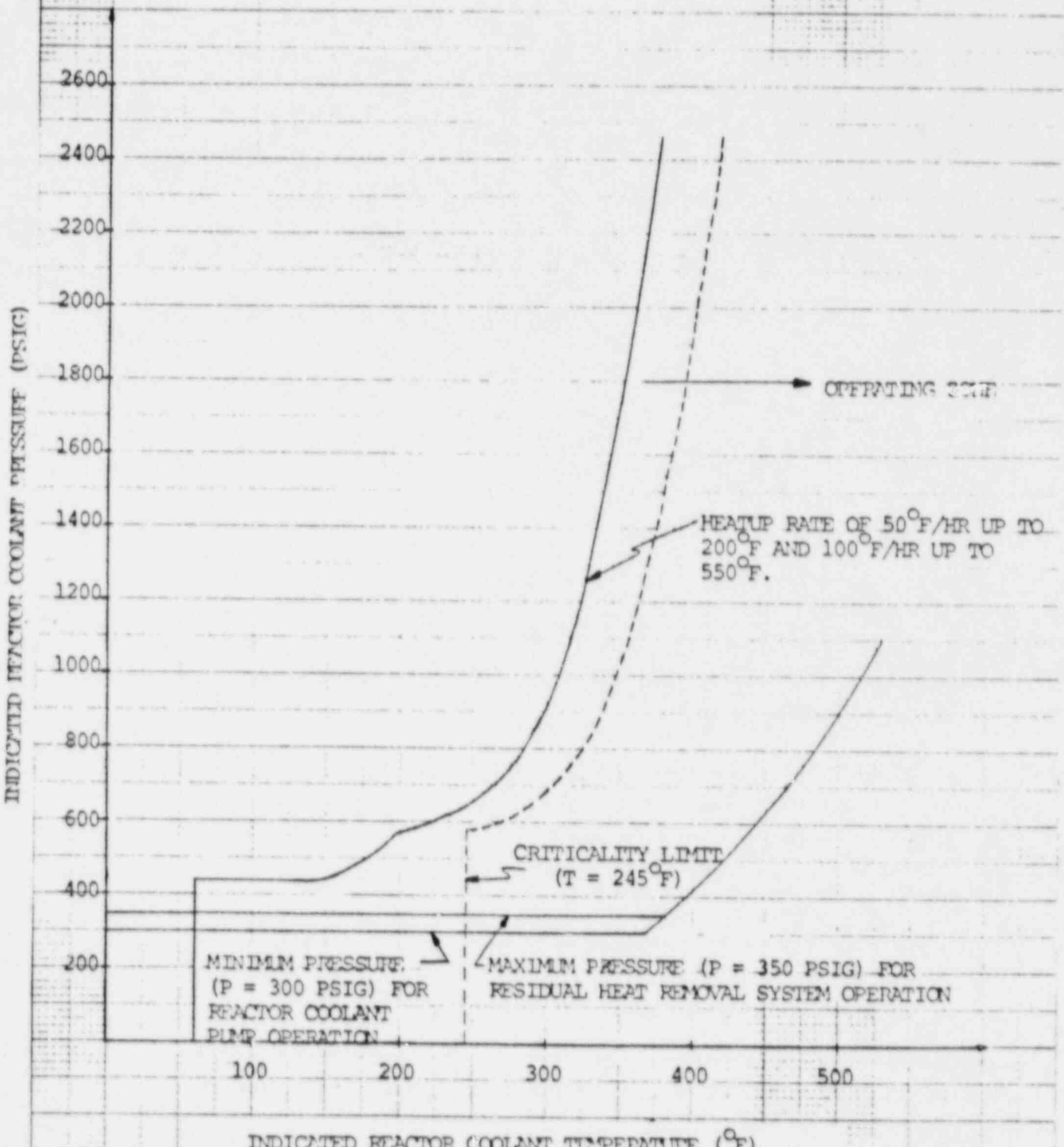
REVISION 2

PREPARED BY	DATE	REVIEWED BY	DATE
REVIEW METHOD		APPROVED BY	DATE

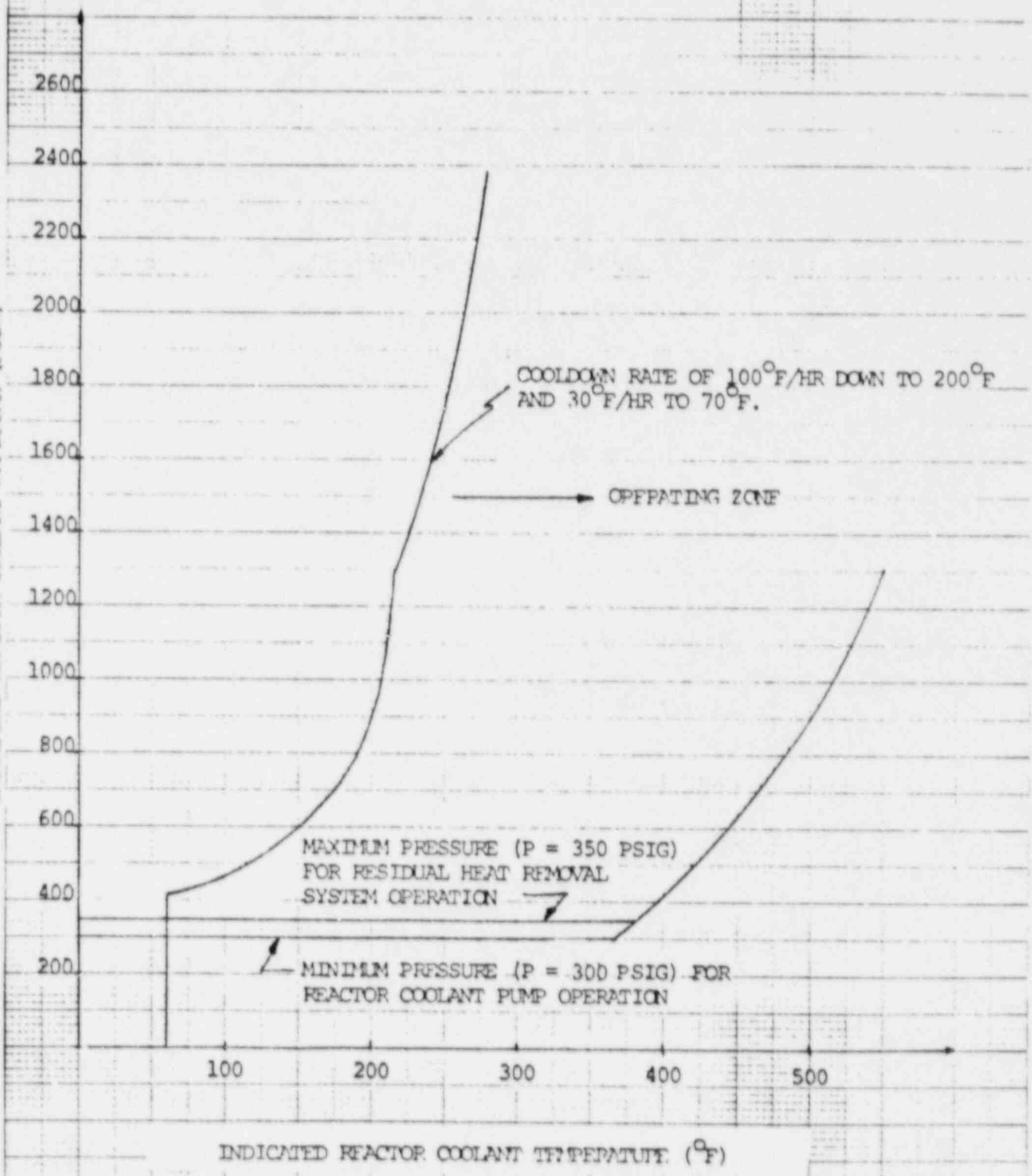
CONNECTICUT YANKEE LIMIT CURVE FOR HYDROSTATIC AND LEAK TESTING
APPLICABLE FOR 22.0 EFFECTIVE FULL POWER YEARS.
(T_{ERROR} = 10°F, P_{ERROR} = 60 PSIG)



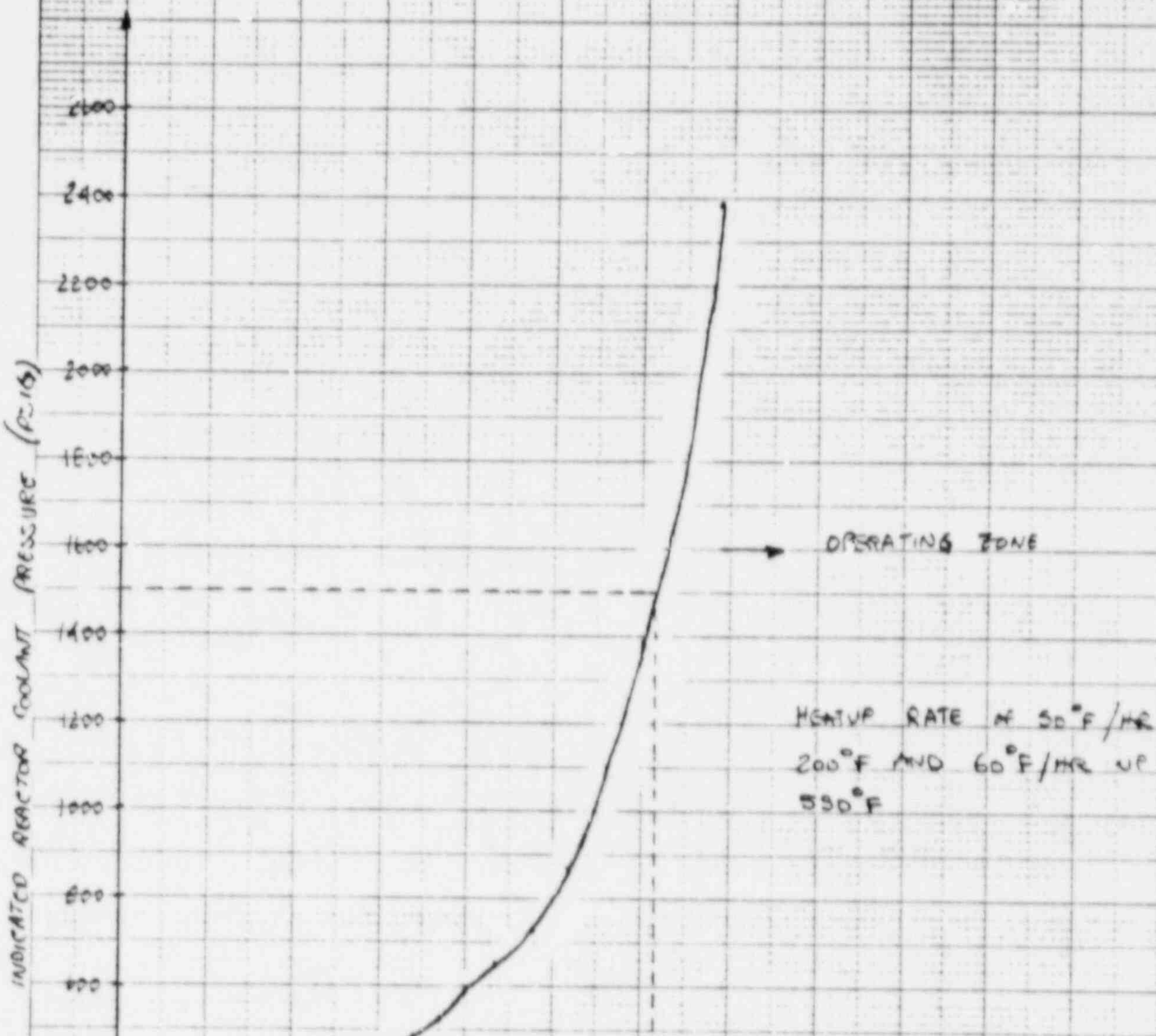
CONNECTICUT YANKEE REACTOR COOLANT SYSTEM HEATUP LIMITATIONS FOR 22.0 EFFECTIVE FULL POWER YEARS.
($T_{ERROR} = 10^{\circ}F$, $P_{ERROR} = 60$ PSIG)



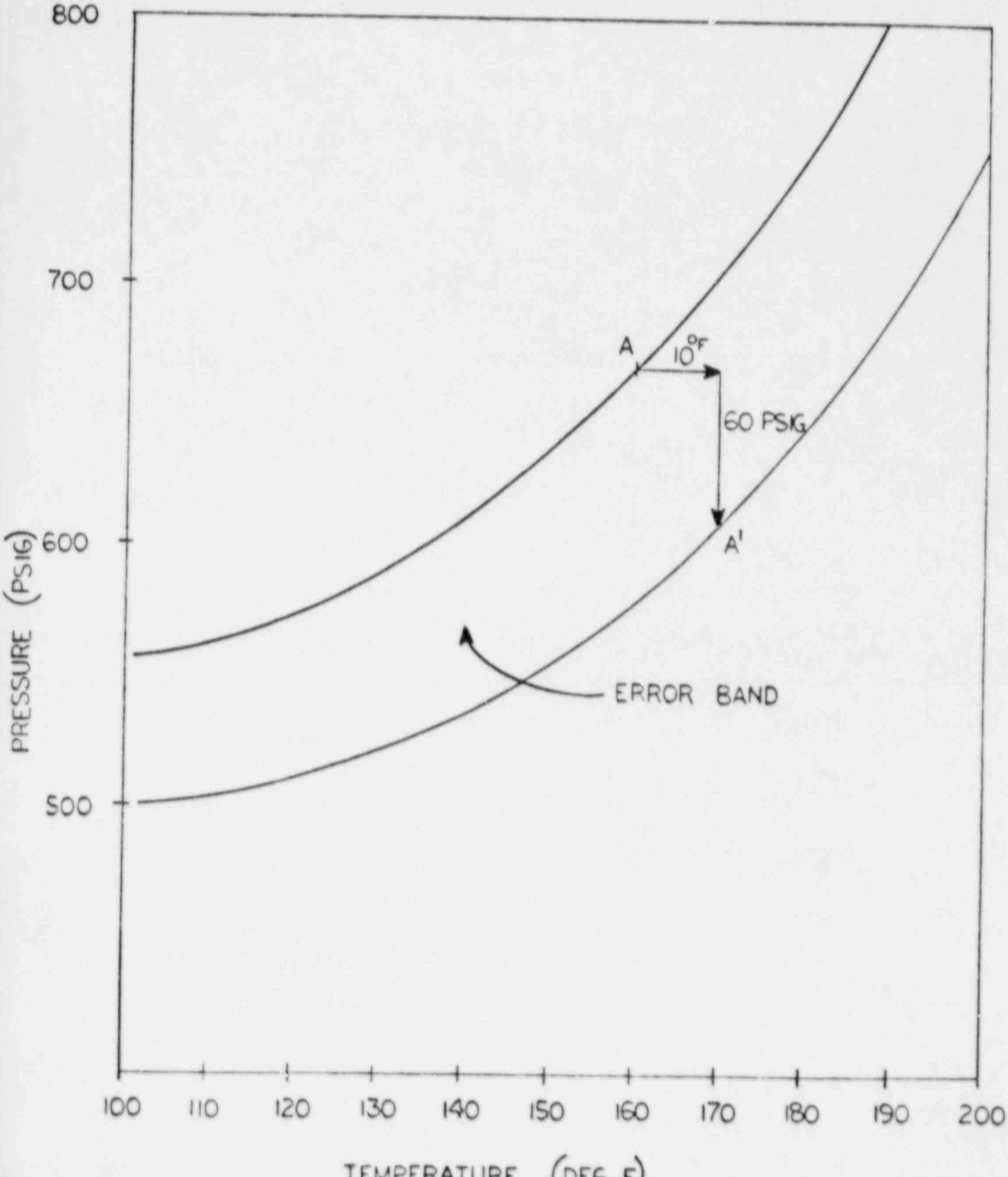
[illegible] RANGE REACTOR COOLANT SYSTEM COOLDOWN
LIMITATIONS FOR 22.0 EFFECTIVE FULL POWER YEARS.
(T_{ERROR} = 10°F, P_{ERROR} = 60 PSIG)



CONNECTICUT YANKEE REACTOR COOLANT SYSTEM HEATUP
LIMITATIONS TO 12.0 EFFECTIVE FULL POWER YEARS
(T_{ERROR} = 10°F, P_{ERROR} = 60 PSIG)



PT LIMIT CURVE INSTRUMENT ERROR INCORPORATION



EVALUATION OF JANUARY 19 TRANSIENT
ASSUMING <u>NO OPERATOR ACTION</u>

- Use Section XI, Appendix E, Evaluation of Unanticipated Operating Events.

- Section III, Appendix G Contains Assumed Boundary Conditions:

  - 1/4 T Flaw, Aspect Ratio of 1/6.

  - "Bounded" Heat-Up or Cooldown Ramp Rates.

  - K_{IR} Curve.

  - Fluence Through the End of an Operating Period.

- Transient Conditions, Even Design Basis Accidents From Chapter 15, are <u>Not</u> Considered or Consistent With the P-T Limit Curves.

ARTICLE E-1000
INTRODUCTION

E-1100 SCOPE

This Appendix provides acceptance criteria and guidance for performing an engineering evaluation of the effects of an out-of-limit condition on the structural integrity of the reactor vessel beltline region. Showing compliance with the criteria in either E-1200 or E-1300 assures that the beltline region has adequate structural integrity for the unit to return to service.

E-1200 ACCEPTANCE CRITERIA[1,2]

Adequate structural integrity of the reactor vessel beltline region is assured if the following applicable criterion is satisfied throughout the event:

(a) For isothermal pressure transients (i.e., $\Delta T_c/\Delta t < 10°F/hr$), the maximum pressure does not exceed the allowable values of Table E-1 at any value of $T_c - RT_{NDT}$.

(b) For pressurized thermal transients (i.e., $\Delta T_c/\Delta t > 10°F/hr$), the maximum pressure does not exceed the design pressure and $T_c - RT_{NDT}$ is not less than 55°F.

If compliance with the above applicable criterion is not shown, adequate structural integrity can be assured by satisfying the guidelines and criteria specified in E-1300.

TABLE E-1
MAXIMUM ALLOWABLE PRESSURE AS A FUNCTION OF $T_c - RT_{NDT}$ FOR ISOTHERMAL PRESSURE TRANSIENTS ($\Delta T_c/\Delta t < 10°F/hr$)
For design pressures greater than 2400 psig

$T_c - RT_{NDT}$ (°F)	Maximum Allowable Pressure (psig)
+25 and greater	1.1 × Design
+15	2400
+10	2250
0	2000
−10	1750
−25	1500
−50	1200
−75	1000
−105	850
−130	800
−200	750

GENERAL NOTE: Linear interpolation is permitted.

E-1300 EVALUATION BY ANALYSIS

(a) Adequate structural integrity of the reactor vessel beltline region is assured if it can be shown by analysis using the input of Table E-2 that the following criterion is met throughout the event:

$$1.4\,(K_{Im} + K_{It}) + K_{Ir} \leq K_{Ic}$$

where

K_{Im} = stress intensity factor due to membrane stress
K_{It} = stress intensity factor due to thermal stress
K_{Ir} = stress intensity factor due to residual stress
K_{Ic} = fracture toughness per Article A-9000

(b) If compliance with the above criterion cannot be shown, additional analyses or other actions shall be taken to assure that acceptable margins of safety will be maintained during subsequent operation.

[1] T_c is the bulk reactor coolant temperature, and $\Delta T_c/\Delta t$ is the maximum variation of temperature T_c in any one hour period.

[2] RT_{NDT} is the highest adjusted reference temperature (for weld or base material) at the inside surface of the reactor vessel as determined by Regulatory Guide 1.99 Rev. 2.

TABLE E-2
EVALUATION INPUT FOR PLANT AND EVENT SPECIFIC LINEAR ELASTIC FRACTURE MECHANICS ANALYSIS

Variable	Value
Pressure	Event pressure time history
Temperature	Event temperature time history
Heat transfer	Event/plant specific flow/mixing conditions
Crack type	Semi-elliptical surface flaw
Minimum initiation crack size	$0.0 < a \leq 1.0$ in. [Note (1)]
Crack orientation	Longitudinal
K_a / K_I location	Surface and maximum depth
Clad effects	Clad to be considered in the thermal, stress, and fracture mechanics analyses [Note (2)]
Transition toughness	K_a per Article A-9000
Upper shelf toughness	(In course of preparation)
Fluence	Fluence at the time of the transient
Shift curve	Regulatory Guide 1.99 Rev. 2
Residual stress	Appropriate distribution for the fabrication process, or linear distribution with +10ksi at the inside surface and −10ksi at the outside surface

NOTES:

(1) a = the maximum crack depth in the base metal

(2) The stresses due to the difference between the base metal and cladding thermal expansion coefficients need not be considered in the isothermal pressure transient evaluation (i.e., $\Delta T_c / \Delta t < 10°F/hr$).

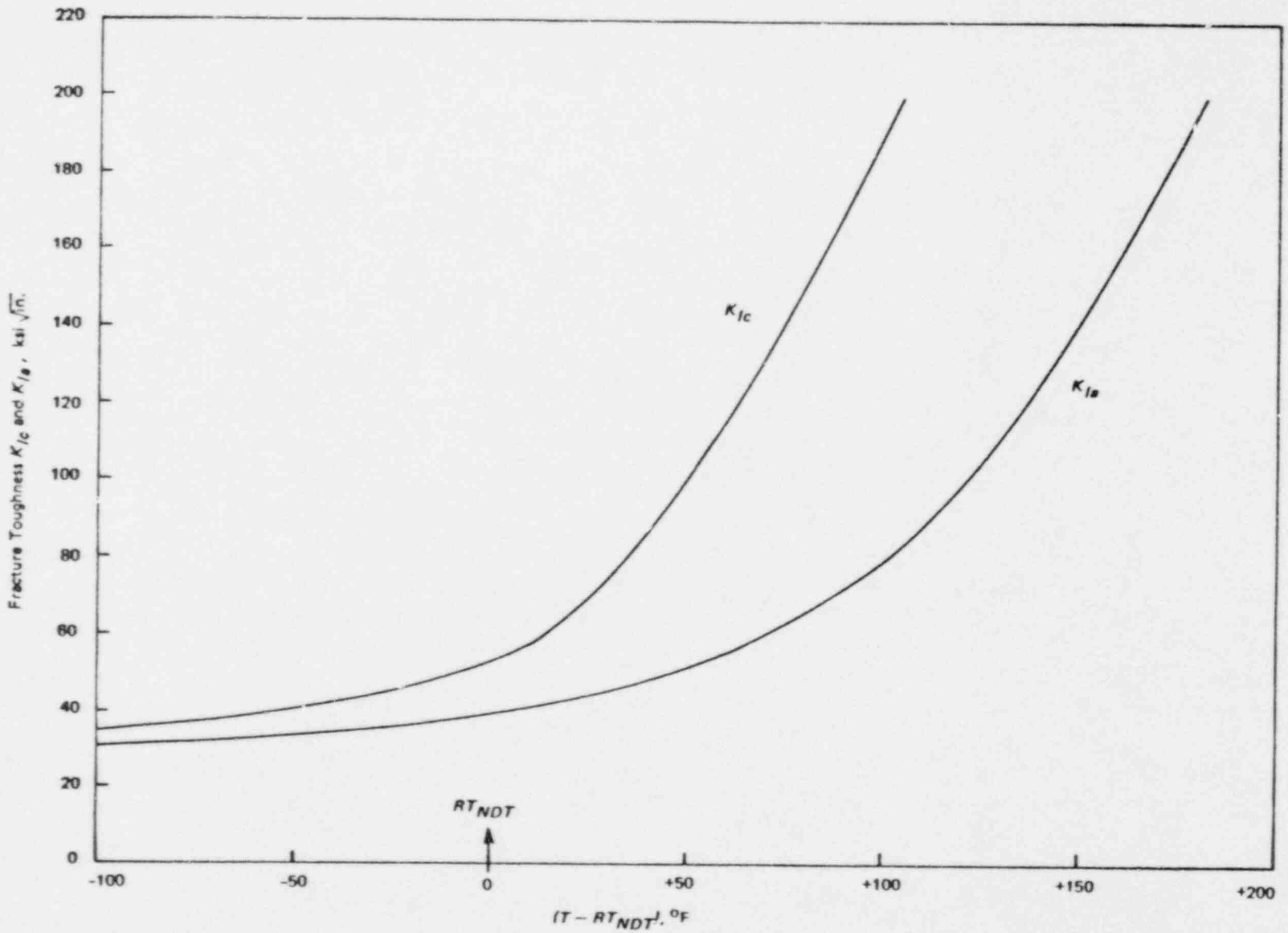


FIG. A-4200-1 LOWER BOUND K_{Ia} AND K_{Ic} TEST DATA FOR SA-533 GRADE B CLASS 1, SA-508 CLASS 2, AND SA-508 CLASS 3 STEELS

ANALYSIS--APPENDIX E

- o Current Adjusted $RT_{NDT} \sim 90^{o}F$:

  -- Original 60^{o} and $30^{o}F$ Shift Measured in Surveillance Capsule.

  -- Conservative Due to Capsule Lead Factor ~ 4.0.

- o $T_C - RT_{NDT} = 134-90 = 44^{o}F$.

- o Table E-1, Allowable Pressure = 1.1 x Design @ $T_C - RT_{NDT} = 2$.

- o Transient Pressure Would Be Limited to 2485 psi by Code Safeties.

- o $P_{ALLOW} > P_{TRANSIENT}$ - Okay.

CONCLUSIONS

- Pressure Transient of January 19 Has No Impact on Millstone 3 Structural Integrity.

- Chemistry and Fluence Limit Range of Isothermal Transients of Concern ~ $150^{\circ}F$ or Less.

- Operational Limits Have Significant Safety Margins to <u>Preclude</u> Component Damage.