PSNH

SEABROOK STATION
Engineering Office

Public Service of New Hampshire

New Hampshire Yankee Division

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T.F. B7.1.2

United States Nuclear Regulatory Commission
Washington, DC 20555

Attention: Mr. Vincent S. Noonan, Project Director
PWR Project Directorate No. 5

References:
(a) Construction Permits CPPR-135 and CPPR-136, Docket Nos. 50-443 and 50-444
(b) PSNH Letter (SBN-513), "Open Items Response (SER Section 7.3.2.8; Instrumentation and Controls System Branch)," J. DeVincentis to G. W. Knighton, dated May 31, 1983
(c) PSNH Letter (SBN-916), "Level Measurement Error (SER Outstanding Issue No. 10)," J. DeVincentis to V. S. Noonan, dated December 31, 1985
(d) PSNH Letter (SBN-945), "Meeting Notes, Instrument Action and Control System Branch," J. DeVincentis to V. S. Noonan, dated February 14, 1986

Subject: Level Measurement Error (SER Outstanding Issue No. 10)

Dear Sir:

In Reference (d), we indicated that the response to RAI 420.23 would be revised to provide additional information requested by the staff during our telecon of January 29, 1986. Accordingly, enclosed please find, as Attachment A, the revised response to RAI 420.23. This revised response includes the results of the reference leg heatup analysis.

One of the major contributors to reactor trips is low-low steam generator level conditions caused by feedwater control problems at low power levels. By limiting the level measurement error to the maximum expected prior to safety injection actuation on high containment pressure, we are maximizing the operation margin. The wider margin will provide the operator with more time to correct for minor feedwater control problems. We expect that this will reduce the probability of unnecessary reactor trips and resultant challenges to safety systems.

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This revised response to RAI 420.23 and the information provided in Reference (c) complete our response to the above referenced SER outstanding issue. Therefore, we request that the resolution of this issue be reflected in the next supplement to Seabrook Station's SER.

Very truly yours,

John DeVincentis, Director
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Enclosures

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ATTACHMENT A

Revised Response to RAI 420.23

Seabrook Station

420.23 (7.2) Describe how the effects of high temperatures in reference legs of steam generator and pressurizer water level measuring instruments subsequent to high energy breaks are evaluated and compensated for in determining setpoints. Identify and describe any modifications planned or taken in response to IEB 79-21. Also, describe the level measurement errors due to environmental temperature effects on other level instruments using reference legs.

RESPONSE: 5/83 2/86 The error in dp level measurement systems due to changes in fluid densities is:

$$E = \frac{R(\Delta\rho_R - \Delta\rho_g) - L(\Delta\rho_f - \Delta\rho_g)}{S(\rho_{f,cal} - \rho_{g,cal})} \times 100$$

where:

- E = Error in % span
- R = Height of reference leg water level above the variable leg tap
- S = Span (distance between taps)
- L = Water level above the variable tap
- ρg_{cal} = Vapor calibration density
- ρf_{cal} = Process fluid calibration density
- $\Delta\rho_R$ = Change in reference leg density from the calibration value
- $\Delta\rho_g$ = Change in vapor density from $\rho_{g,cal}$
- $\Delta\rho_f$ = Change in process fluid density from $\rho_{f,cal}$

Note: $\Delta\rho = \rho_{cal} - \rho_{accident}$

This error determination assumes that the reference leg and variable leg below the variable tap are at the same temperature and produce counteracting errors.

A. Effects of Post-Accident Conditions on Indicated Level

1. Reference Leg Heatup

If the process conditions are not affected by the accident, the error due to reference leg heatup is:

$$E = \frac{R\,\Delta\rho_R}{S(\rho_{f,cal} - \rho_{g,cal})}$$

A decrease in reference leg density will increase the indicated level.

2. Reference Leg Boiling

Depressurization to a pressure lower than the saturation pressure of a hot reference leg will eject water from the reference leg. This will greatly reduce the reference leg average density and will result in a sudden, large increase in indicated level. The level error cannot be determined as there is no way of knowing the amount of water ejected. As reference leg boiling could only occur on a faulted component after a large steam line break inside containment or a LOCA, the operators will be instructed to disregard the level instruments on the faulted component if such a depressurization has occurred. Reference leg flashing on intact steam generators is prevented by the automatic closure of the Main Steam Isolation Valves (MSIV) after a main steam or feed line break that causes system depressurization.

3. Process Density Changes

If the containment conditions are not affected by the accident, the error due to process density changes is:

$$E = \frac{-R\Delta\rho_g - L(\Delta\rho_f - \Delta\rho_g)}{S(\rho_{f,cal} - \rho_{g,cal})} \times 100$$

A decrease in system pressure and temperature will increase the indicated level.

B. Effects on Safety-Related Level Setpoints

1. Steam Generator Low-Low Level Reactor Trip and Emergency Feedwater Initiation

These functions are provided for protection in the event of a loss of feedwater including those caused by a Feedwater Line Break (FWLB). Those events due to problems outside of the containment will not result in a harsh environment that affects the level measuring system.

The analysis in FSAR Section 15.2.8 shows that the steam generator low-low level trip will provide adequate protection for all FWLBs regardless of break location. The Solid State Protection System (SSPS) setpoint includes an environmental allowance to compensate for errors introduced by the level sensor or reference leg heatup when exposed to the harsh conditions caused by the FWLB inside containment. A FWLB inside containment will increase containment pressure and, depending on the size of the FWLB, may actuate safety injection on high containment pressure (Hi-1).

Safety injection actuation performs the safety functions provided by the steam generator low-low level signal. Since the low-low level signal is not required after Hi-1

safety injection actuation, we are limiting the reference leg heatup error to the maximum calculated prior to Hi-1 actuation. It should be noted that taking credit for Hi-1 safety injection actuation, to limit level measurement error, does not require a revision of the FSAR Section 15.2.8 analysis since it is still conservative/ bounding. The setpoint calculations will provide adequate documentation.

Main steam line breaks were not considered as steam generator level is not relied on to provide protection.

Analyses to determine the maximum level error due to reference leg heating were performed with the following considerations/assumptions:

a. Containment response was analyzed with the CONTRAST-S[1] computer program using passive heat sink data given in FSAR Tables 6.2-3 and 6.2-4. Conservatively high heat transfer coefficients have been used for the passive heat sinks to delay the containment pressure rise. This will purge more noncondensibles prior to the Hi-1 containment pressure signal with a resulting temperature increase as saturation temperature is approached.

b. Breaks are double-ended and the break effluents from both sides do not mix. This conservatively envelopes all types of breaks and maximizes containment temperature by maximizing the steam enthalpy.

c. The mass and energy release rates are calculated assuming choked flow through the break.

d. Containment initial conditions are 14.7 psia, 120°F, 100% RH.

e. Hi-1 operates prior to 5.8 psig (20.5 psia) which includes an allowance for instrumentation uncertainties. All functions actuated by Hi-1 occur.

f. Pressure buildup is delayed and air is purged from containment by venting through the containment on-line purge system. A discharge coefficient of 1.0 was used. The pressure downstream of the purge valve is conservatively assumed to be 13.5 psia.

g. Containment free volume is 2.84×10^6 ft^3 (approximately 5% higher than that used in the DBA analysis).

h. The reference leg response is calculated following guidelines prescribed in NUREG-0588[2] using the HESITET[3] computer program. Figure 1 is the schematic of the reference leg model.

i. Containment high pressure alarm is actuated at 15.7 psia (1.0 psig).

j. All events that reach the Hi-1 setpoint 15 or more minutes after the high pressure alarm are manually terminated within 15 minutes of the alarm.

Results of Analyses

Figures 2 and 3 show containment temperature at the time the containment pressure reaches the maximum Hi-1 setpoint as a function of break size at 100% and 0% power levels, respectively. Smaller breaks result in higher containment temperature because more air is vented through the containment on-line purge system. Also, the containment temperatures are higher at lower power levels because there is less flashing of the feedwater. The addition of low enthalpy steam from the flashed feedwater tends to reduce the containment temperature as it mixes with the high enthalpy steam coming from the steam generator.

The event, due to small FWLBs, is manually terminated by the operator within 15 minutes of receipt of the high containment pressure alarm. Operator actions that would be taken in response to alarms indicating an abnormal condition inside the containment are:

1. Evaluate containment conditions by observing values and changes in:

    a. Containment pressure (including accident monitoring instruments listed in FSAR Section 7.5).

    b. Containment sump level (see FSAR 5.2.5).

    c. Containment temperature and humidity.

    d. Qualified RVLIS reference leg temperature.

2. If rapidly changing containment conditions indicate a significant event, the operators will initiate a controlled plant shutdown. The speed at which plant shutdown is accomplished will depend on the rate that containment pressure is approaching the Hi-1 setpoint.

3. If containment conditions are changing slowly or are relatively stable, there may be attempts to identify and isolate the source of the leakage without a plant shutdown. This will not be done if personnel safety is compromised or Technical Specifications are violated.

    Since all evaluations and manual actions are performed from the Control Room, we are assured that the required actions can be performed in the assumed 15-minute operator response time.

We have determined that the most severe event, i.e., highest reference leg temperature, occurs for a small break at low power when the Hi-1 setpoint is reached 15 minutes after the high containment pressure alarm is actuated. Smaller breaks would result in higher reference leg temperatures when Hi-1 is reached, but will be terminated by operator action, since they will take more than 15 minutes.

The most severe event is a 0.029 DE break at 0% power level. Figures 4 and 5 are histories of the containment pressure and temperature and reference leg water temperature. The maximum reference leg water temperature is 180°F at 970 seconds after the break. The high pressure alarm is actuated at 70 seconds. If the high pressure alarm is increased to 2.0 psig (16.7 psia) the maximum temperature would be 185°F.

The increase in reference leg water temperature to 185°F results in an error of 3.1%. This reference leg error has been included in the calculated value for the low-low level trip setpoint of 17.0%.

This setpoint is very conservative as the channel statistical error allowance includes a transmitter environmental allowance considering temperature and radiation effects associated with a DBA. Initial investigations for the Westinghouse Owners Group (WOG) Trip Reduction and Assessment Program (TRAP) indicate that the transmitter environmental allowance should be significantly reduced for a FWLB prior to Hi-1.

The critical break size of 0.029 DE has an area of approximately 0.031 ft^2. This is equivalent to a pipe size somewhere between 4 and 5 inches. The feedwater P&ID, Drawing 9763-F-805003 (typical) shows that the only feedwater lines inside containment are 16 inches or larger. In NUREG/CR-4305, comments on the leak-before-break concept for nuclear power plant piping systems, it was concluded that leak-before-break is the most probable failure if the piping system is protected from overpressure, overtemperature, and reduced wall thickness. The critical size break in the 16-inch pipe is not probable since the feedwater piping inside containment is protected from all these failure initiating events. In addition, the pipe is manufactured from normalized A-106B steel to enhance its fracture toughness to further minimize the probability of unstable flaw growth that could lead to catastrophic failure.

Breaks in connections to the steam generator shell below the normal water level are also considered a FWLB. The largest connection is the 2-inch blowdown connection. This is smaller than the critical break size and would always pass saturated water during the event. The containment

temperatures were maximized in the analysis by assuming saturated steam discharge when the feed ring in the steam generator was uncovered.

It is estimated that the feedwater flow that will be lost out the 0.029 DE break will be about 1,000,000 lb/hr. The feedwater flow to each steam generator is about 3,800,000 lb/hr at 100% power. At high power levels there is little excess feedwater capacity and a critical size break will result in a rapid decrease in level. The low-low level setpoint will be reached within 15 minutes. At low power levels the Feedwater System will probably supply the break and maintain level if the main feedwater pumps are operating. If the emergency or startup feedwater pumps are operating, power is restricted to below 10%. Initial results from the WOG TRAP analyses indicate that protection, other than prevention of steam generator dryout, is not required for loss of feed to one steam generator at low power.

The most accurate alarm for alerting the operators to abnormal conditions within the containment is the high containment pressure alarm at 15.7 psia. Input to the alarm is provided by COP-PT-1787 that has a range of 12 to 18 psia. Normal containment pressure is 15.2 psia with a Technical Specification limit of 16.2 psia.

Other alarms that could alert the operator to abnormal conditions inside the containment or a small feedwater line break are:

1. Containment leakage monitoring described in FSAR 5.2.5.

2. Feed/steam mismatch alarm (set at 700,000 lb/hr).

3. Steam generator level deviation (set at programmed level ±5%).

4. Steam generator low-level alarm (set at 30%).

The steam generator level deviation alarm is provided on both the annunciator and the Video Alarm System (VAS). The other alarms are provided on the VAS. Failure of the VAS actuates an alarm on the annunciator. The operators will augment the Control Room personnel and perform frequent monitoring of the Control Room instrumentation when the VAS is not available.

From the above, it can be seen that the critical break size is not a probable event and is small enough that protective action may not be required. Since this event does not have significant impact on the safety of the plant, we consider that the alarms provided are sufficiently reliable to alert the operator in case manual action is required to terminate the event.

2. Steam Generator High-High Level Trip

The steam generator high-high level trip provides protection for excess feedwater flow events. These events do not cause a harsh environment; therefore, the trip setpoint will not be changed.

3. Pressurizer High-High Level Trip

The pressurizer high-high level trip provides protection for increase in reactor coolant inventory events. These events do not cause a harsh environment; therefore, the trip setpoints will not be changed.

C. Effects on Accident Monitoring Instrumentation

The Seabrook Emergency Operating Procedures will follow the guidance of the Westinghouse Owners Group Emergency Response Guidelines. These guidelines specifically require that the instrumentation errors due to environmental effects and system pressure changes be considered. Appropriate error calculations (see Paragraph A.1 and A.3) will be performed and incorporated in the emergency operating procedures when the guidelines are finalized and the as-built dimensions are obtained.

REFERENCES

1. "Predictions of Containment Pressure-Temperature Transients Using CONTRAST-S MOD 1 - A Digital Computer Program", UEC-TR-006-SUP, June 1979.

2. NUREG-0588, "Interim Staff Position of Environmental Qualification of Safety-Related Equipment", 1979.

3. "HESITET - A Digital Computer Program to Analyze Temperature Transients in Containment Passive Heat Sinks and Equipment", UEC-NU-509, October 1979.

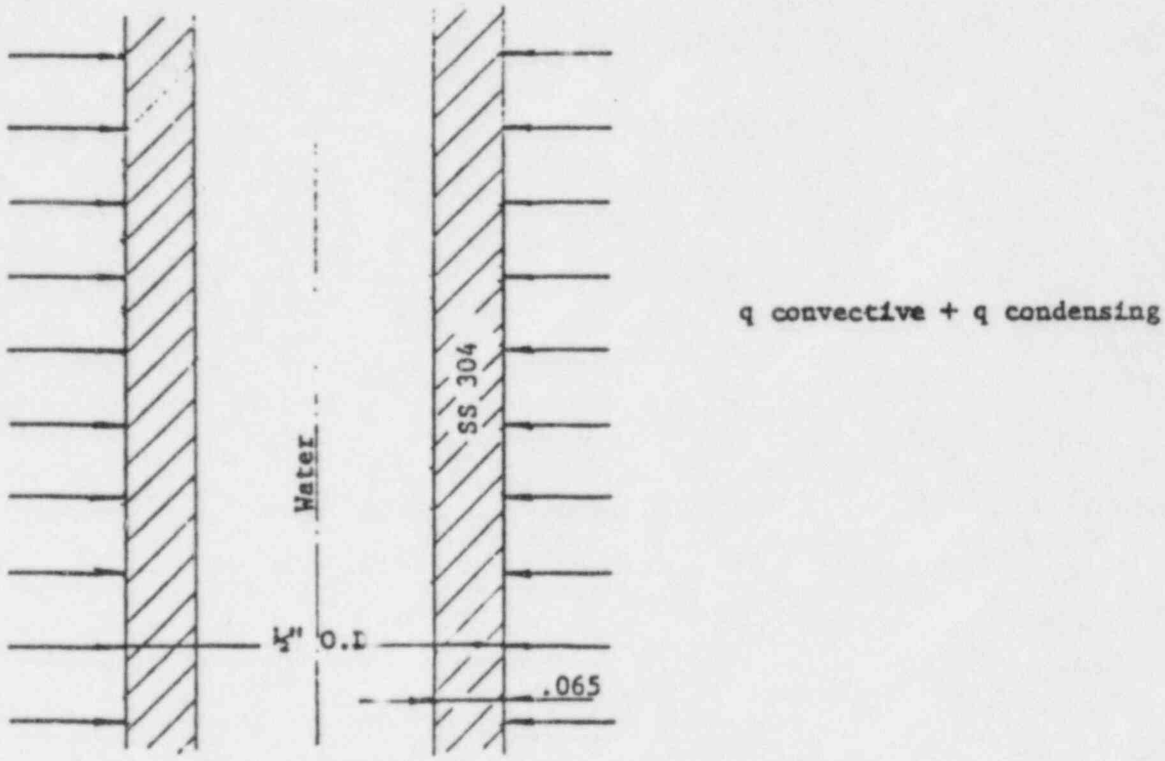

Figure 1 Schematic of Heat Transfer Model for Reference Leg Thermal Response

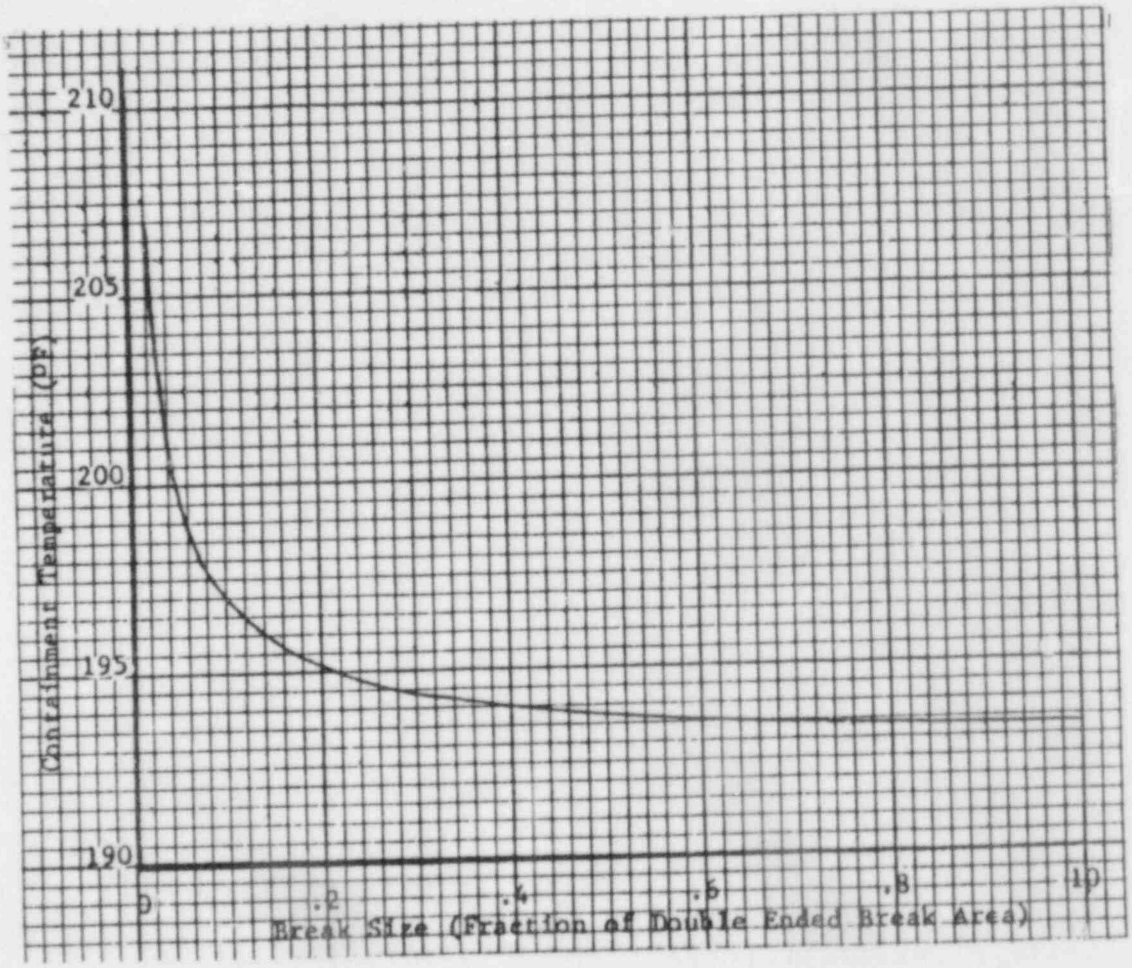

FIGURE 2

Containment Temperature At Time Of
High 1 Trip vs. Break Size (100% Power Level)

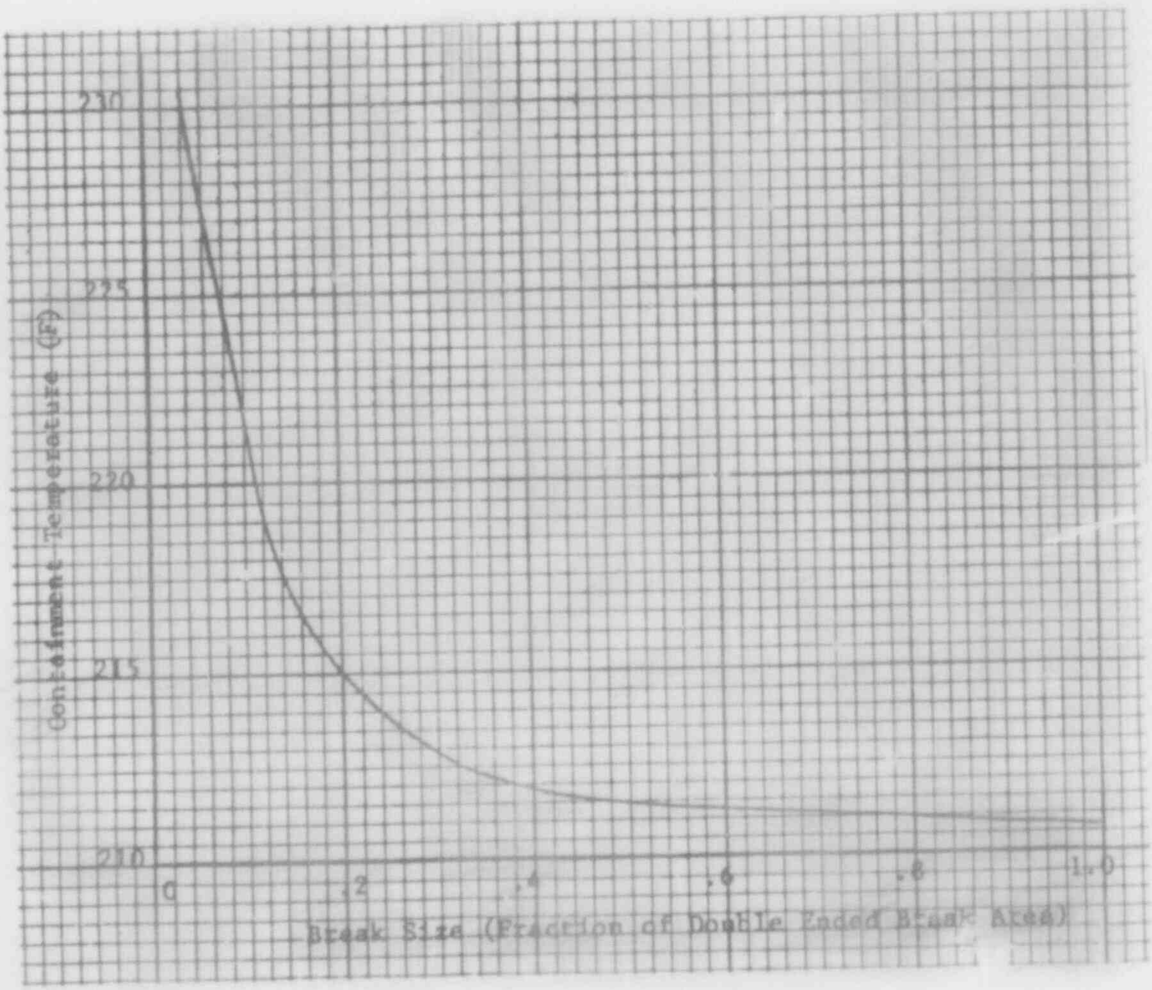

FIGURE 3

Containment Temperature At Time Of
High 1 Trip vs. Break Size (0% Power Level)

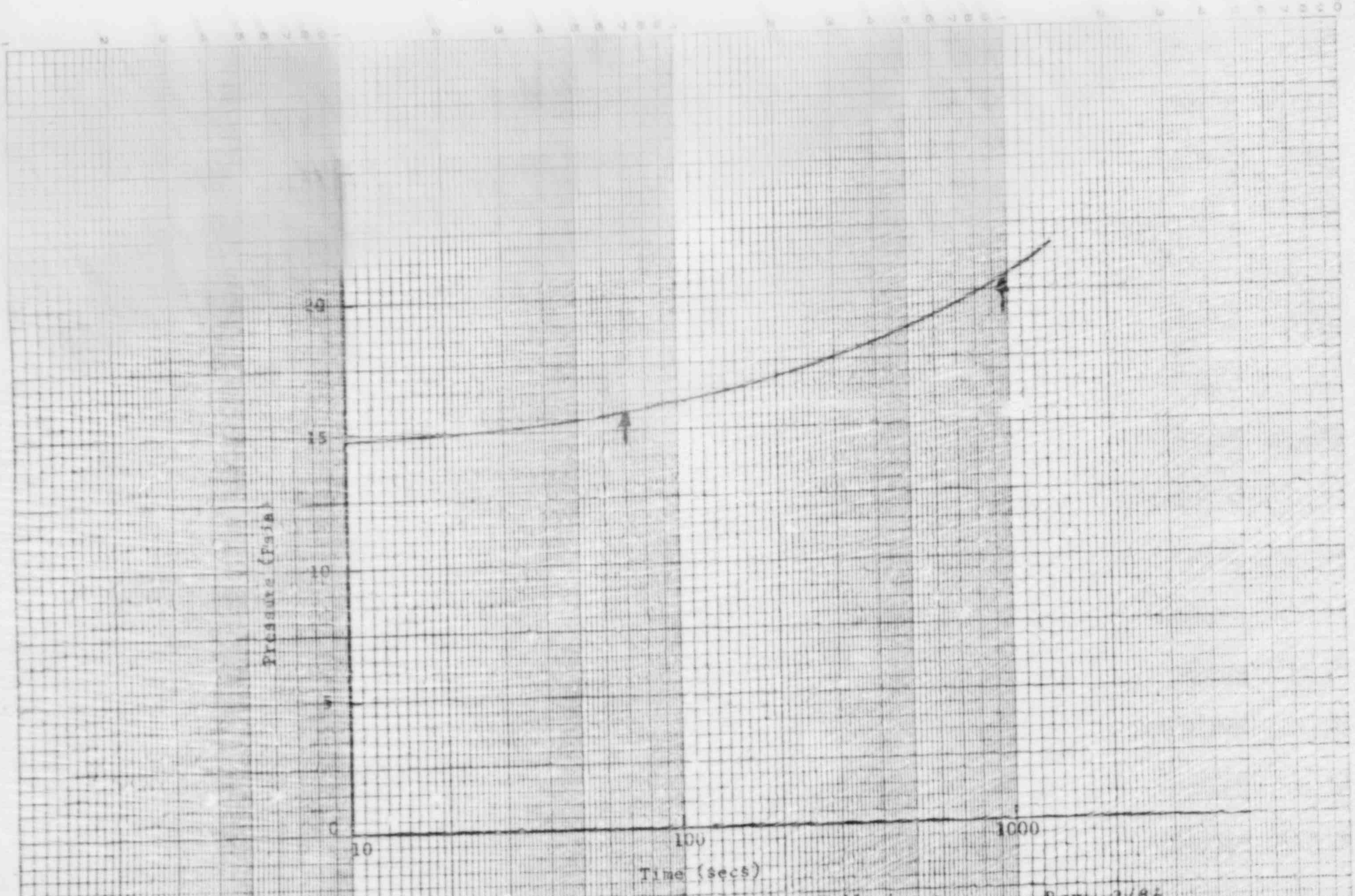

FIGURE 4 Containment Pressure Response Following a 0.029DE FWLB

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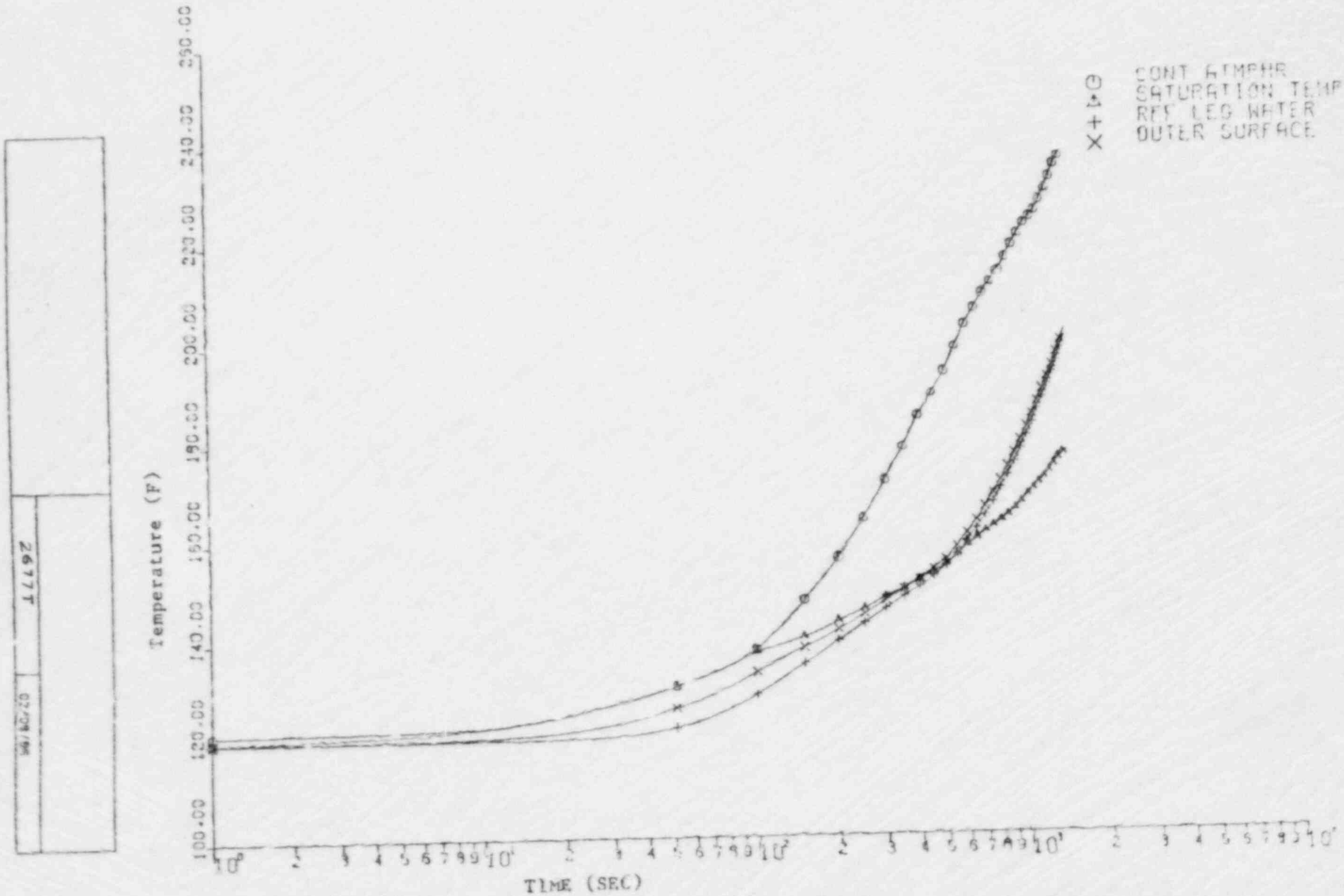

FIGURE 5: Ref. Leg Heat-up Following a 0.029DE Feedwater Line Break in Containment Rev. 2/86