PSNH

Public Service of New Hampshire

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April 15, 1986
SBN- 1004
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-955), dated February 28, 1986, "Level Measurement Error (SER Outstanding Issue No. 10)", J. DeVincentis to V. S. Noonan

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

Dear Sir:

During our telephone conversation on April 2, 1986, the staff requested additional information relating to Reference (b), our latest submittal on the above-referenced issue. Attachment A, the revised response to RAI 420.23, contains our response to these requests. We have included a commitment to upgrade the high containment pressure alarms that alert the operators to abnormal conditions inside the containment that could be the result of a feedwater line break and have provided an outline of the alarm response procedure that will be used by the operators to respond to the high containment pressure alarm.

Attachment B, revised FSAR Section 15.2.8, documents our reliance on operator action and safety functions actuated by high containment pressure to limit the level measurement error resulting from a feedwater line break inside containment. This revision will be incorporated into the FSAR by a future amendment.

This revised response to RAI 420.23 and the information provided in Reference (b) complete our response to the above-referenced SER outstanding

B001
1/1

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

John DeVincentis, Director
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Enclosures

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SBN-

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
4/86

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. <u>Reference Leg Boiling</u>

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. <u>Process Density Changes</u>

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

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

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

B. <u>Effects on Safety-Related Level Setpoints</u>

1. <u>Steam Generator Low-Low Level Reactor Trip and Emergency Feedwater Initiation</u>

These functions are provided for protection in the event of a loss of feedwater including those caused by a Feedwater Line Break (FWLB). 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.

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 heatup 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 for water 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^{o}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 x 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 are sensitivity studies of the containment temperature at the time the containment pressure reaches the maximum Hi-1 setpoint (6.8 psig setpoint was used) 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 so that the ambient temperature approaches the saturation temperature for water. 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.

We have determined the critical break size that results in the highest reference leg temperature 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 within 15 minutes 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. Verify high containment pressure conditions by observing the narrow range (COP-PI-1787) and accident monitoring (SI-PI-934 and 935) instrumentation.

2. Isolate containment ventilation (COP).

3. Commence a plant shutdown if containment pressure is above 15.7 psia (COP-PI) and increasing rapidly, or is above 2 psig (SI-PI). Note that a gradual pressure increase caused by ambient temperature increase during a plant startup is not cause for a plant shutdown. The applicable Technical Specification will apply.

4. Monitor containment pressure, temperature, humidity, and sump level.

5. Trip the plant if containment pressure exceeds 3 psig (SI-PI-934 or 935) or the RVLIS reference leg temperature exceeds 180°F (RC-TE-1313 or 1323, indicated on RC-XX-7315-1 or 4). Redundant instrumentation will be used to verify the presence of extreme conditions that warrant a plant trip.

The RVLIS temperature was selected since it is an environmentally and seismically qualified indication, and will respond to containment temperature in the same way as the steam generator reference legs. The trip criteria were selected to ensure that the analysis assumptions are met. By isolating containment ventilation early in the transient, the pressure buildup will be faster and the containment temperatures will be farther from saturation temperatures thereby providing additional assurance that the analysis assumptions are met.

Alarm Response Procedures, implementing this procedural outline, will be available prior to core load.

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 would 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 (less than 5 minutes at 100% power).

At lower power levels the Feedwater System may supply the critical 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 are presently available to alert the operator to abnormal conditions inside the containment resulting from 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 $\pm$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 will perform frequent monitoring of the Control Room instrumentation when the VAS is not available.

To increase the reliability of our alarms we will implement the following changes prior to core load to provide redundant and diverse alarms:

1. Provide diverse alarms from COP-PT-1787 by adding an annunciator alarm from bistable COP-PDYY-1787 and generating a VAS alarm from the analog input (A3250).

2. Generate a high containment pressure alarm (about 2.0 psig) from SI-PT-936 analog input (A0502). The alarm will be set as low as is feasible without causing nuisance alarms (channel has a span of 0 to 60 psig).

From the above, it can be seen that the critical break size is not a probable event and, depending on power level, results in a low-low level trip within 15 minutes of the event or is small enough that manual action to protect the core may not be required. Since this event does not have significant impact on the safety of the plant, we consider that the upgraded alarms are sufficiently reliable to alert the operator in case manual action is required to terminate an event. The need for a manual reactor trip is determined by observing a limited number of instruments mounted on the front of the MCB and is performed using controls in the same vicinity. Therefore, the required evaluation and necessary actions can be performed within 15 minutes of the high containment alarm.

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.

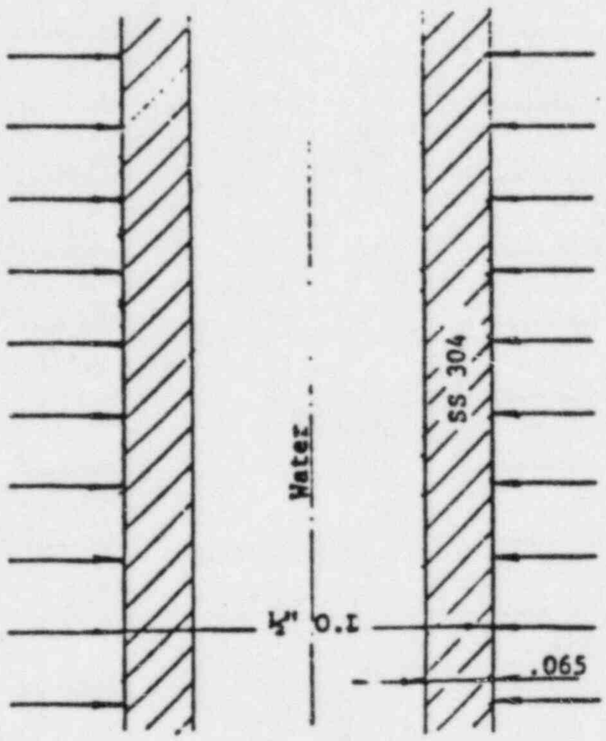

Figure 1 Schematic of Heat Transfer Model for Reference Leg Thermal Response

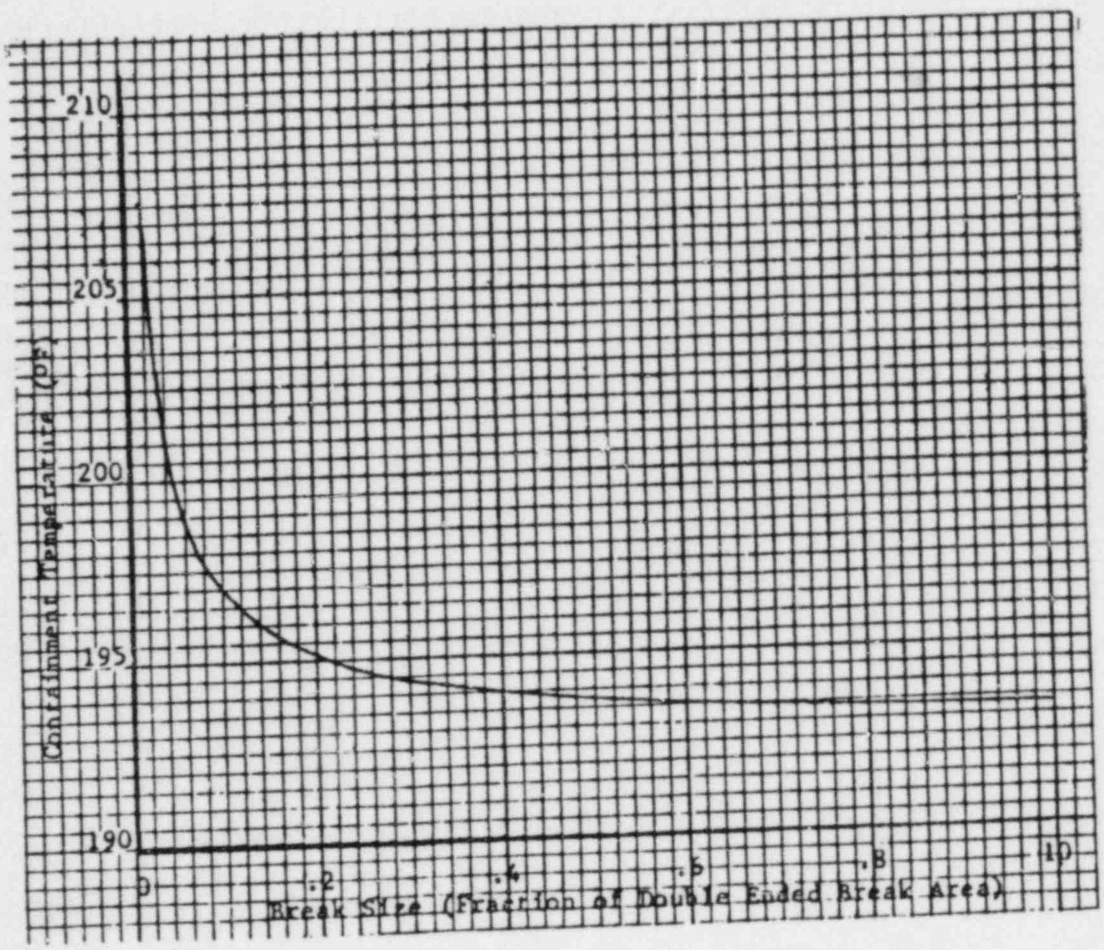

FIGURE 2

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

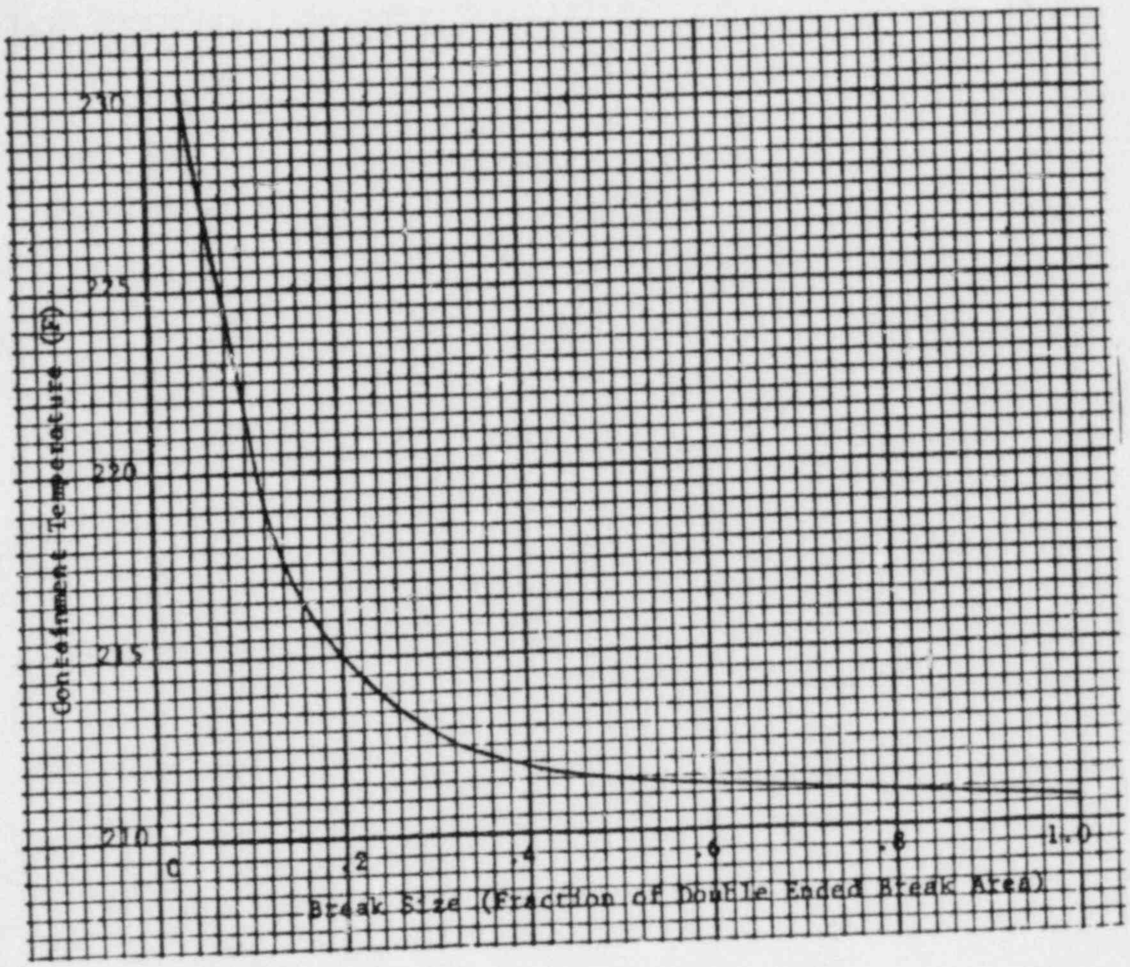

FIGURE 3

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

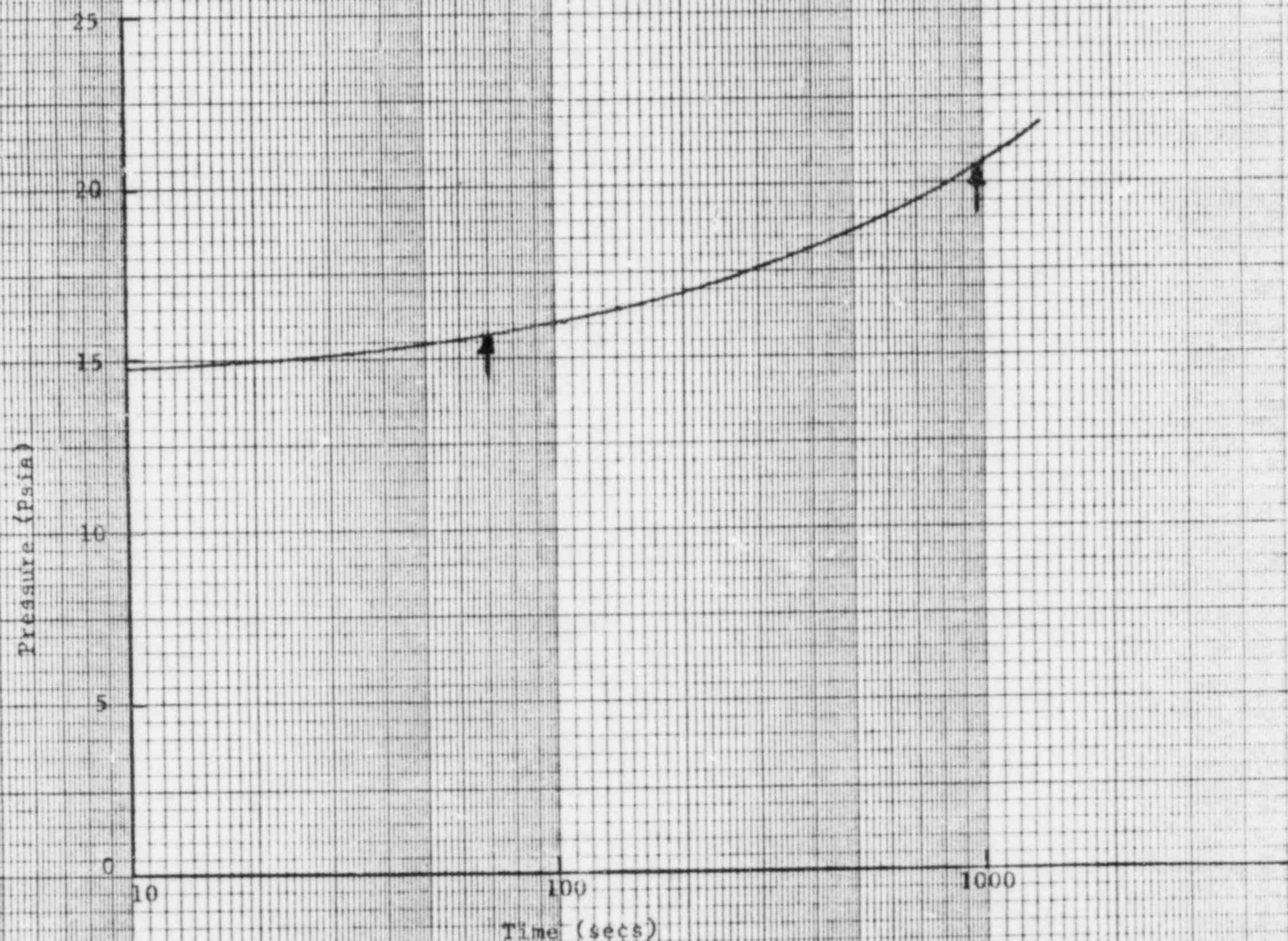

FIGURE 4 Containment Pressure Response Following a D.029DE FWLB

Rev. 2/86

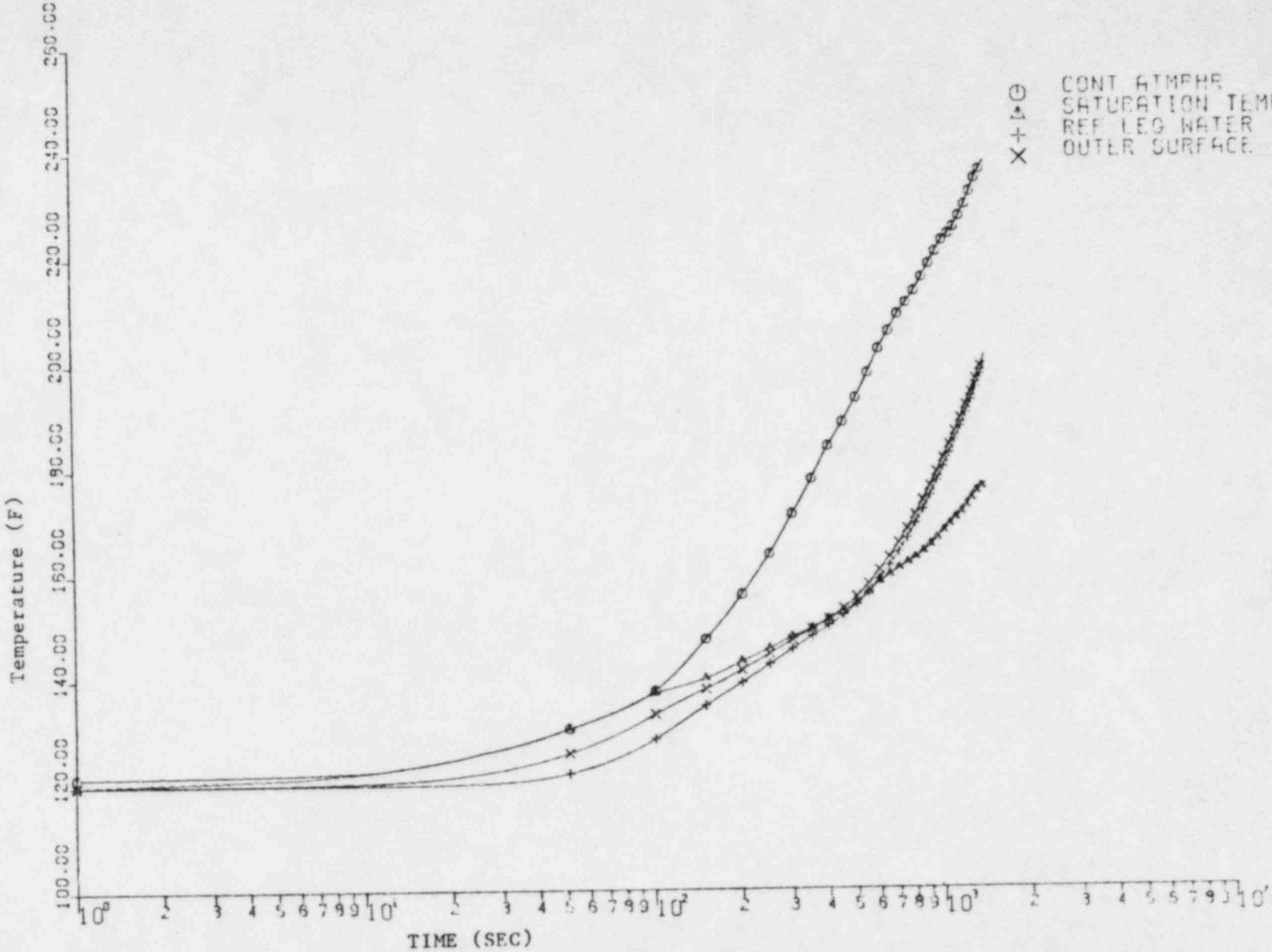

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FIGURE 5: Ref. Leg Heat-up Following a 0.029DE Feedwater Line Break in Containment Rev. 2/86

ATTACHMENT B

Revised FSAR Section 15.2.8

Seabrook Station

2. Initial reactor coolant average temperature is 5.8°F above the nominal value, and the initial pressurizer pressure is 30 psi above its nominal value.

3. No credit is taken for the pressurizer power operated relief valves or pressurizer spray.

4. Initial pressurizer level is at the nominal programmed value plus 2 percent (error); initial steam generator water level is at the nominal value plus 5 percent in the faulted steam generator, and at the nominal value minus 5 percent in the intact steam generators.

5. No credit is taken for the high pressurizer pressure reactor trip.

6. Main feedwater to all steam generators is assumed to stop at the time the break occurs (all main feedwater spills out through the break).

7. The worst possible break area is assumed. This maximizes the blowdown discharge rate following the time of trip, which maximizes the resultant heatup of the reactor coolant.

8. A conservative feedwater line break discharge quality is assumed prior to the time the reactor trip occurs, thereby minimizing the water inventory of the intact loops in order to reduce any long term cooling benefit. After the trip occurs, a saturated liquid discharge is assumed until all the water inventory is discharged 56
from the affected steam generator. This minimizes the heat removal capability of the affected steam generator.

(SEE TABLE 15.0-4)

9. Reactor trip is assumed to be initiated when the low-low steam generator level trip setpoint minus 10 percent of narrow range span in the ruptured steam generator is reached. INSERT A

10. The emergency feedwater system is actuated by the low-low steam generator water level signal. The emergency feedwater system is assumed to supply a total of at least 470 gpm to two unaffected steam generators, including allowance for possible spillage through the main feedwater line break. A 60 second delay was assumed following the low-low level signal to allow time for startup of the standby diesel generators and the emergency feedwater pumps. An additional 382 seconds was assumed before the feedwater lines were purged and the relatively cold (120°F 56
maximum) emergency feedwater entered the unaffected steam generators.

11. No credit is taken for heat energy deposited in RCS metal during the RCS heatup.

Insert A

A separate analysis has been performed to support the reference leg heatup environmental allowance (EA) used in the low-low level setpoint calculation. This separate analysis indicates that the worst case EA results from a relatively small FWLB. Credit in this separate analysis was taken for limited operator action 15 minutes after receipt of a high containment pressure alarm and for the safety functions initiated by high containment pressure.

12. No credit is taken for charging or letdown.

13. Steam generator heat transfer area is assumed to decrease as the shell side liquid inventory decreases.

14. Conservative core residual heat generation is assumed based upon long term operation at the initial power level preceding the trip.

15. No credit is taken for the following potential protection logic signals to mitigate the consequences of the accident:

    (a) High pressurizer pressure.

    (b) Overtemperature ΔT.

    (c) High pressurizer level.

    ~~(d) High containment pressure.~~

Receipt of a low-low steam generator water level signal in at least one steam generator starts both the motor driven emergency feedwater pump and turbine driven emergency feedwater pump, which in turn initiate emergency feedwater flow to the steam generators. Similarly, receipt of a low steam line pressure signal in at least one steam line initiates a steam line isolation signal which closes all main steam line isolation valves. This signal also gives a safety injection signal which initiates flow of cold borated water into the RCS. The amount of safety injection flow is a function of RCS pressure.

Emergency operating procedures following a feedwater system pipe rupture require the following actions to be taken by the reactor operator:

1. Isolate feedwater flow spilling from the ruptured feedwater line and align the system so that the level in the intact steam generators is recovered.

2. Stop high head safety injection charging pumps if a) wide range reactor coolant pressure is greater than 2000 psig, and is stable or increasing, b) pressurizer water level is greater than 50 percent of span, and c) steam generator narrow range level indication exists in at least one steam generator.

Isolation feedwater flow through the break allows additional emergency feedwater flow to be diverted to the intact steam generators.

Safety injection is assumed to be terminated by the operator 30 minutes after reactor trip.

Subsequent to recovery of level in the intact steam generators, the high head safety injection pumps will be turned off and plant operating procedures will be followed in cooling the plant to hot shutdown conditions. 53