Florida Power
CORPORATION

September 28, 1988
3F0988-20

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Subject: Crystal River Unit 3
Docket No. 50-302
Operating License No. DPR-72
Emergency Feedwater

Dear Sir:

Attached are Florida Power Corporations responses to the questions provided in your August 30, 1988 correspondence. The preliminary assessments on thermal stratification and water hammer in the Emergency Feedwater system are discussed herein. Final analysis on these two issues is being completed at this time and will be provided to the NRC by October 12, 1988.

If you have any questions, please contact this office.

Rolf C. Widell

Rolf C. Widell, Director
Nuclear Operations Site Support

REF:RCW

Attachment

xc: Regional Administrator, Region II

Senior Resident Inspector

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Responses to Emergency Feedwater Questions of August 30, 1988

Question # 1:

The concrete expansion anchors of pipe restraint EFH-126 were found to be pulled partially loose from their structural attachment. Piping analysis performed for piping outside containment were modeled to reflect the as-found condition based on the visual inspections. You have not provided the results of stress analysis that demonstrate the thermal conditions in the piping systems that would have led to this failure. In connection with this, no information has been provided to determine whether the piping attached to this restraint may have been significantly overstressed prior to the failure of EFH-126.

Response to Question # 1:

EFH-126;

The piping outside the Reactor Building has been analyzed to determine the stresses in the piping system that would have been present during the elevated temperature events had the piping been fully restrained (all supports completely operational). The results of this analysis indicate that the actual piping stress, including dead weight and thermal loading, were within the allowable stress criteria as defined in the B31.1 piping code. Furthermore, the support loads resulting from this analysis have been compared to the allowables. All supports met the short term operability requirements by having a Factor of Safety greater than 2.0. The Factor of Safety for EFH-126 was 2.48. The support member stresses under this loading condition were within the B31.1 allowables. The maximum stress in EFH-126 would have been 12,663 psi. On the basis of this analysis, it can be concluded that the loads resulting from the elevated temperature did not contribute to the failure of EFH-126. In addition, thermal stratification does not appear to have contributed to this failure since loads under this postulated condition would have been in the opposite direction of those which would pull the anchor from the ceiling (refer to Response to Question 2).

The damage to EFH-126 may have been the result of an un-analyzed system transient. Since the cause of this damage and the associated loading cannot be determined, a detailed inspection of the support/pipe welds of EFH-126 will be performed during the October outage. If the inspection identifies defects or deficiencies, the proper corrective action will be performed prior to returning the system to operable status.

Question # 2:

Thermal stratification in piping systems is known to result from low flow conditions. Thermal stratification can cause fatigue damage and large bending stresses. You should address the possibility that thermal stratification existed in the piping system as a result of the backleakage and line cooling, and address any damage to the piping that may have resulted.

Response to Question # 2:

STRATIFICATION;

Thermal stratification may occur in piping systems where low flow conditions exist. This is primarily due to the absence of sufficient mixing to promote a homogeneous solution. Typically, stratified vertical runs are not of concern since the piping can easily tolerate the radial growths. However, horizontal runs can see large bending stresses under stratified conditions and it is these sections of piping that FPC has evaluated.

Although the leak from EFV-18 was one of low flow (0.2-0.7 gpm measured), the stratification conditions in the pipe regions outside the reactor building are not considered to be significant since they were full of water. There is sufficient mixing created by the short, alternating vertical and horizontal runs outside the reactor building near the leak and injection point to preclude thermal stratification. This is reinforced by measured delta T's of less than 40°F at the penetration during the early stages of a very slow cooling operation from a very hot initial condition of approximately 480°F. The geometry of the piping in this region could permit the cool trickle flow through FWV-43 to stratify in the low point trap, but the vertical drop and rise sections initiated sufficient mixing to limit the delta T to a minimal value (see figure 1). System walkdowns with the line hot and during this cooling operation identified no visible signs of distress in the piping, supports, or anchors.

The piping geometry inside the reactor building however, is such that steam and relatively cool water could co-exist in horizontal runs as the line is being filled or evacuated. This condition involves a stratified steam layer on top of water with a surface temperature at or near saturation (due to a layer of condensed steam) and lower water temperatures on the bottom of the line.

Three flow conditions pictorially represented in figures 2, 3 and 4 will be discussed from a stratification perspective along with the potential effects.

Response to Question # 2 continued:

Figure 2 depicts the anticipated thermal phases of the piping near the steam generator due to the 0.2 -0.7 gpm bonnet leak from EFV-18. Due to the upward thermal growth of the steam generator at temperature, the horizontal runs contain gentle downward slopes which could create localized traps for steam and water. Because of this geometry, as the water and piping heat up, counterflow conditions could permit cool water to flow under hotter, steam blanketed water which could result in thermal stratification for extremely short durations (figure 2).

Figures 3 and 4 represent two flow rates that were used at CR-3 for cooling the line. A preliminary analysis performed by Babcock & Wilcox indicates back leakage greater than 0.09 gpm will eventually void the piping of water from the steam generator to the reactor building penetration. Based on local temperature readings of approximately 480°F outside the Reactor Building and the observed higher leak rates, it is believed the voided condition existed when the cooling operations were initiated. Figure 3 represents the 50 gpm (for 5 minutes) cooling approach used seven times during 1988. Figure 4 represents the virtually continuous 0.5-1.0 gpm line filling and cooling approach induced by the current temporary injection system.

In both fill cases, the first 100 gallons of water introduced are relatively hot. Additional heat is absorbed by this water from the vertical pipe wall as it ascends inside the building (see figure 1). During the 50 gpm fill, it will take the entire 250 gallons to fill the piping with the water never reaching the steam generator shell. A subsequent temperature profile could result in a stratified condition at the penetration end of the long horizontal run of the "B" steam generator which will quickly abate as the water obtains heat.

The 0.5-1.0 gpm fill will have the same initial conditions as the 50 gpm fill. The hot water will slowly fill the vertical run and the short horizontal run at elevation 115'-0" with little stratification due to the high temperature of the incoming water. The hot water will then lay in the bottom of the long horizontal run until its level is sufficient to cascade over into the trap section near the steam generator (see figure 4). As cool water then flows into the horizontal piping, there will be some degree of stratification at this end of the line until the line is full and begins to cool due to the loss of condensing steam energy. The end product will be a virtually stagnant line with conditions similar to figure 2A.

Response to Question # 2 continued:

It is reasonable to assume that some degree of stratification could exist in portions of all of the horizontal runs at some point in time under the previously discussed flow conditions. The locations subjected to the severest conditions are expected to be at the extreme ends on either side of the 140' long horizontal run to the "B" steam generator inside the reactor building. A typical configuration is being modeled to determine what theoretical stresses may have existed in these segments. The model will utilize 300-400°F delta T's. These are the absolute bounding conditions and are not expected to actually exist in transient conditions. The preliminary report on the model results is due from Gilbert Commonwealth Inc. on October 3, 1988.

Both ends of this piping run will be inspected for external damage at the local supports and piping during the October shutdown. Fatigue damage in these areas is not expected since the number of subsequent evacuation and fill cycles were few and the refill water was reasonably hot. Since the analyzed loads generated by stratification are empirical in nature and the model being utilized is unproven, the inspection will be the primary determination of whether a problem did indeed exist. The piping model will be used as a reference tool to help decide where to focus the inspection. Keeping the piping run full of water will preclude reoccurrence, therefore back leakage should be limited to maintain a solid piping system. The amount of back leakage permissible is determined by the heat transfer characteristics of the piping system. Preliminary analysis indicates that with the piping system in its current insulated condition the system can tolerate approximately 15 ml/min of inventory loss and remain full beyond the trap at the steam generator. Final back leakage analysis is due from Babcock & Wilcox on September 29, 1988 and will be factored into the determination of the leak rate criteria for FWV-43 and 44.

Question # 3:

In response to CAL item number 4, you identified a number of reasons why you believe that the potential for water hammer was considered unaffected by the backleakage. These reasons have been reviewed, and the staff feels that FPC should provide more substantially supported arguments to resolve the question of whether a water hammer could or did occur.

Response to Question # 3:

WATERHAMMER;

Considering that the "B" EF line was evacuated of water and was steam filled back to penetration 109, it is conceivable that a waterhammer event was possible under certain conditions. Classic waterhammer (single phase water) on a fast fill transient is being evaluated as well as condensation induced waterhammer (two phase counterflow). The following conditions are under study and are considered to envelope the worst case waterhammer loadings:

1. Fast fill transient - a voided line with both EFP's starting, which results in the maximum possible flow (>1000 gpm) of water through the system. This will cover all waterhammer loads for flows in excess of 160 gpm.

2. 50 gpm fill - An Operating Procedure (OP-450) describes this technique to prevent steam binding in the EF pumps. This procedure was employed seven times to cool the line and seat FWV-43. There is a potential for two phase waterhammer (see figure 5) when the line is greater than 45% full of water.

3. 125 gpm fill - this is the flow produced when EFIC is actuated and when the level in the steam generator is above the desired water level set point. Two phase waterhammer is possible, per figure 5, at all fill depths.

4. The other extreme is the low flow injection system currently being employed to maintain inventory in the line. This is not considered to be a waterhammer concern because the flow rates are below 1 gpm (see figure 5).

Conditions 1 and 3 did not occur at CR-3 during the high temperature condition but are being evaluated because the possibility of this type of flow introduction did exist. Condition 2 did occur seven times. Two phase type waterhammer can only happen when filling horizontal lines and downward flowing vertical lines where steam can be trapped between water slugs. This type of geometry exists in the long horizontal run that terminates in the downturn to the steam generator trap (see figure 1). Gilbert Commonwealth, Inc. analyses shows that only flow rates below 160 gpm could exhibit some degree of two phase waterhammer in a six inch line with the conditions described in Figure 5. Flows in excess of 160 gpm simulate full piping bounded by the single phase fill conditions.

Response to Question # 3 continued:

Preliminary assessment indicates waterhammer did not occur during the 50 gpm cooling cycles, however an inspection will be performed in the vicinity where the waterhammer would have most likely been initiated. Inspections will be performed on the horizontal run near the "B" steam generator and will include EFH-25A which is the only restraint utilizing concrete anchors and the most susceptible to damage. Formal analytical data for cases 1, 2 & 3 are due from Gilbert Commonwealth, Inc. on October 10, 1988.

The waterhammer concern, similar to the problems with thermal stratification, is alleviated by maintaining the system full. Leakage rates that will prevent evacuating the system are due in the Babcock & Wilcox report on September 29, 1988 and will be factored into the leak rate criteria for FWV-43 and 44.

Question # 4:

In your July 14, 1988 letter responding to the CAL, you indicated that FPC was evaluating test procedures and acceptance criteria for check valves FWV-43 and 44. FWV-43 has been identified as having a trend of poor performance from the standpoint of backleakage. Based on the fact that even a small backleakage through FWV-43 or 44 can result in overheating of the EFW system, you should indicate why these valves are not categorized as ASME Section XI A/C valves and tested accordingly beginning with the post-maintenance testing during the October shutdown.

Response to Question #4:

TESTING;

To date, FWV-43 and 44 have been classified as Category "C" valves per ASME Section XI (applicable edition), Article IWV-2200. However, our recent operating experience has shown that EFW system operability is very sensitive to seat leakage past these valves. Therefore, FPC has reclassified FWV-43 and 44 as valves for which seat leakage is limited to a specific maximum amount. This will make FWV-43 and 44 subject to the testing requirements of Category "A" and "C" as explained in Article IWV-2200. The leakage rate will be developed in accordance with ASME Section XI, 1983ED with summer 1983 Addenda, Article IWV-3426. This testing requirement will be implemented beginning with the post maintenance testing during the October 1988 shutdown.

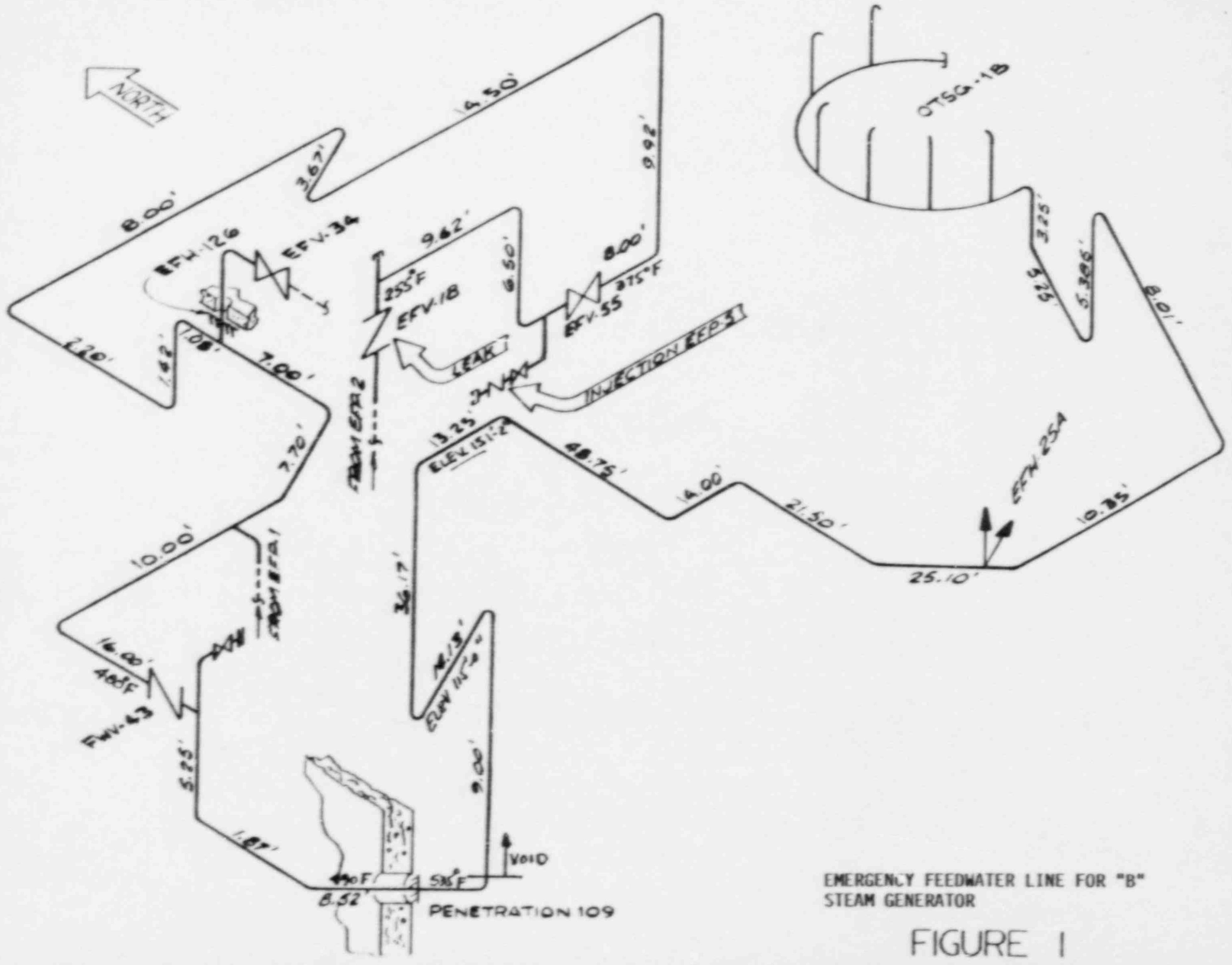
Question # 5:

You should discuss your reasons for increasing the design temperature of the EFW piping and the implications of doing so.

Response to Question # 5:

TEMPERATURE;

A detailed review of the normal operating conditions experienced by the EFW piping inside the Reactor Building has determined that portions of the EFW piping near the steam generator is subject to higher temperatures than was assumed during original design (110°F). The design temperature of the EFW piping outside the containment is being increased to 150°F to provide additional margin above the original design of 110°F. In order to increase the design temperature, a re-analysis of both trains of EFW piping is presently in progress. Preliminary results indicate that the design temperatures can be increased without exceeding the applicable code stresses for the piping and support systems. Upon completion of the analysis, the appropriate documents will be updated to reflect the new design parameters.



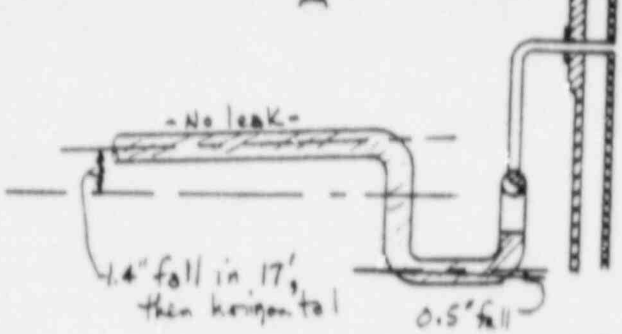
EMERGENCY FEEDWATER LINE FOR "B" STEAM GENERATOR

FIGURE 1

FIGURE 2

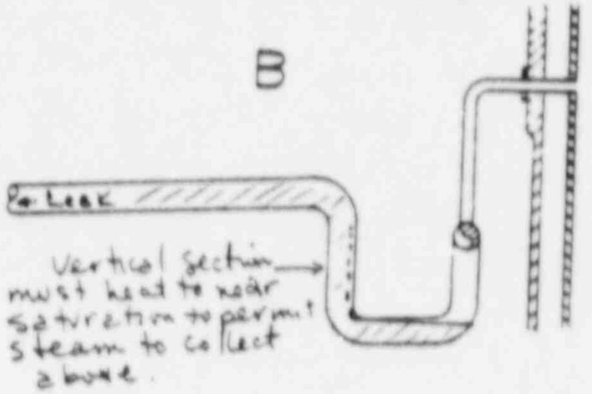
Phases of 0.2 - 0.7 gpm leak

A



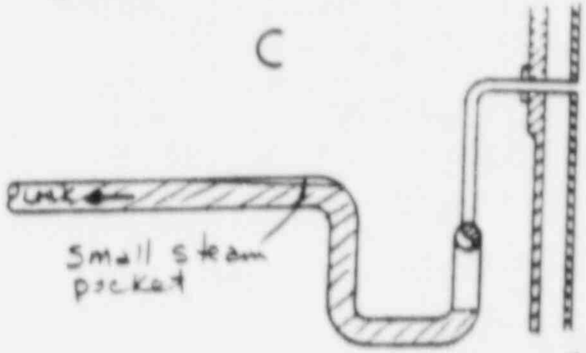
A. Steady state - no leakage, solid water system up to near ring header. Ring header and raisers (6) are full of steam.

B



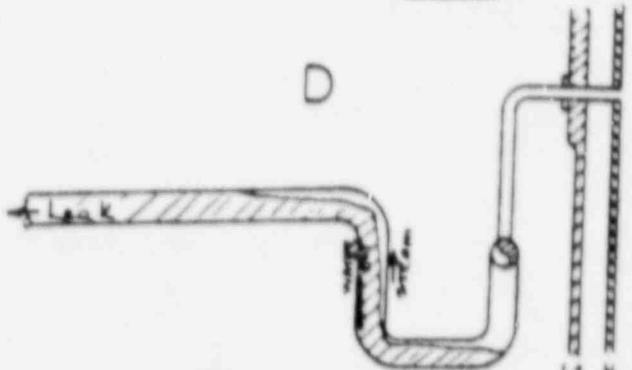
B. Leak causes water level to drop since leakrate is faster than condensing steam rate. Steam bubbles eventually reach vertical section and condense while heating vertical run. Some stratification exists in lower horizontal run but spring can support system permits pipe movement until the water heats to saturation.

C



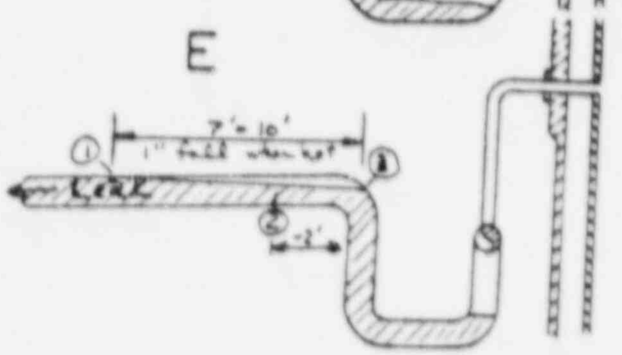
C. A steam pocket exists at 535 degrees F until it gives up its heat of vaporization to piping and nearby cool water. The steam supply is cut off until the pressure lowers.

D



D. Condensation of bubble in "C" permits water to push from generator side up into horizontal run where pocket was. This, in turn permits hot water to return to lower section as pressures equalize with result looking like "C", but a larger pocket. Iterations continue between "C" & "D".

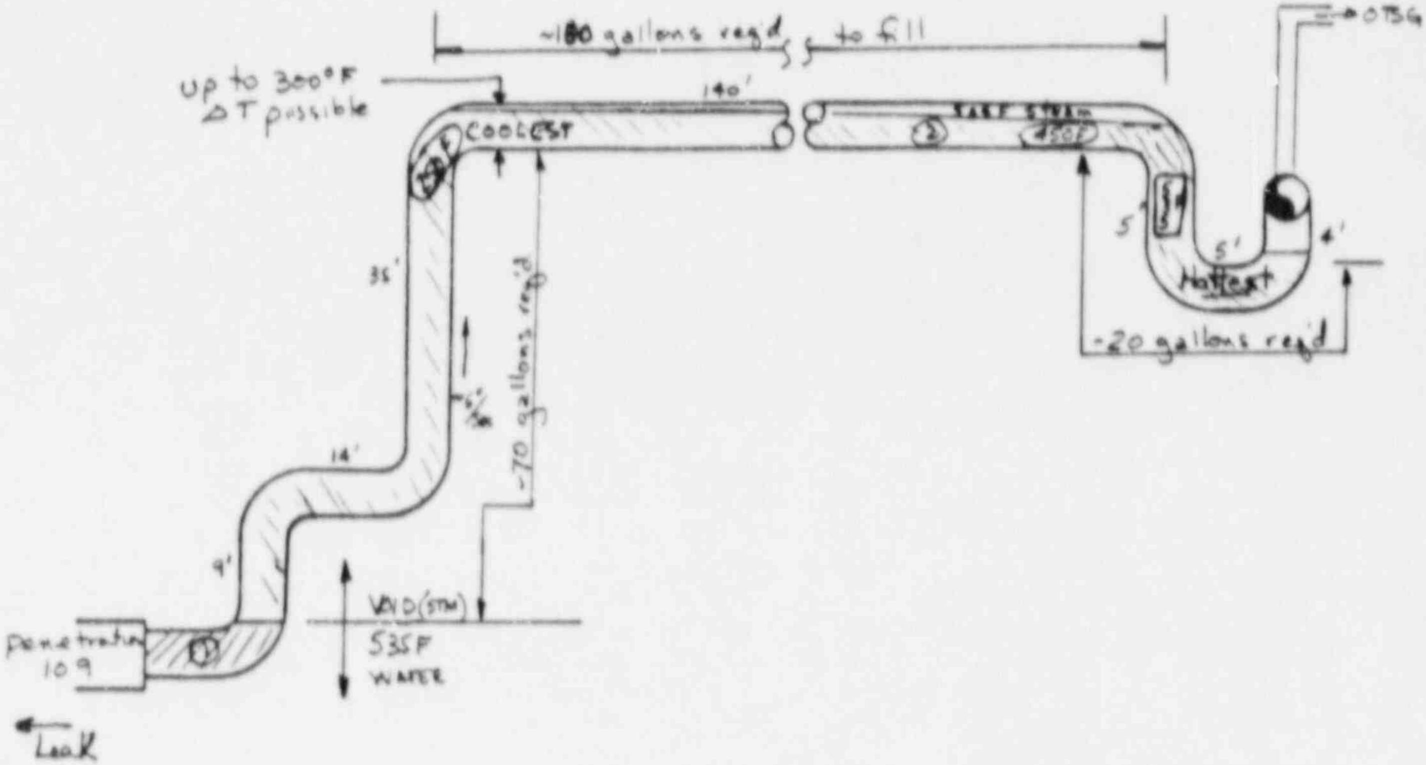
E



E. Probable worst case stratification in the system induced by EFV-18 leak. Maximum delta T is expected at point (1) with somewhat lower delta T at point (2) due to heated water, and little or no delta T at point (3).

The remaining 138' of piping is horizontal and the next steam pocket size will run the entire piping length creating a large steam/water interface. This will result in faster condensation, hence faster water heating and somewhat lower delta T's.

50 GPM FILL SEQUENCE

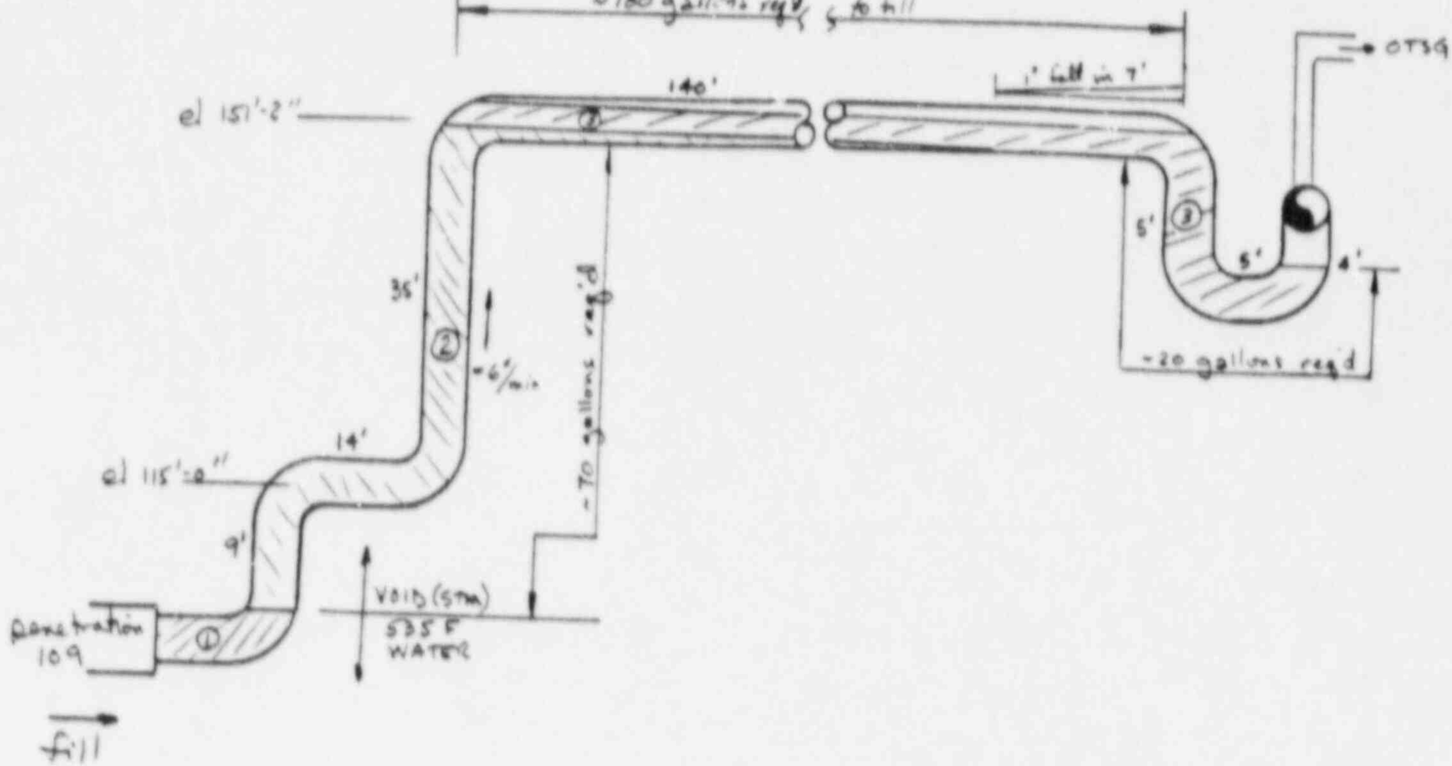


1. After sufficient time to void the line due to the leak, the piping run inside containment is virtually voided.

2. A 250 gallon injection at 50 gpm will not fill the line entirely. The 1st 100 gallons introduced to the horizontal run are hot and should fill the pipe halfway. The next 80 gallons would be cooler and the resultant thermal gradient in the pipe run would probably be on the order of 250 F to 450 F as shown above. This is very similar to the later phases of the leak described in Figure 2, since the leak continues.

FIGURE 4

0.5 - 1.0 GPM FILL SEQUENCE

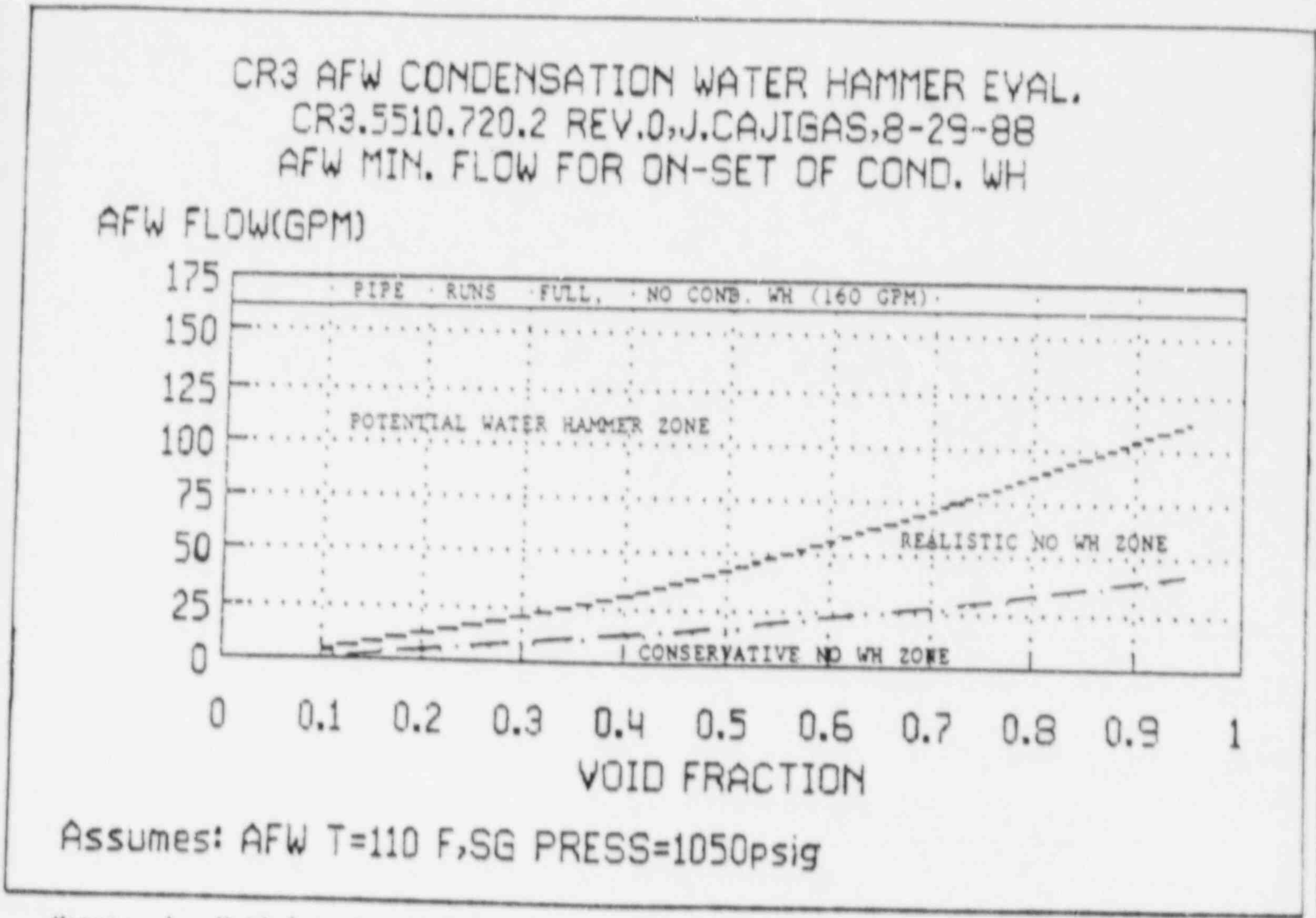


1. The piping run inside containment was essentially void of water due to leak @ EF7-18.

2. The vertical run is slowly filled until it reaches the horizontal run @ 151'-2" where 1" of depth is trapped (133' section) due to thermal growth up at the generator end (~20 gallons trapped). This is very hot water.

3. The next 80 gallons is also hot water (but gradually cooler) which fills the trap. Not much condensation occurs until the line is at least 1/2 full because of the temperature of the incoming water.

4. Eventually cool water will flow under the hot water unti the line is full creating a stratified condition at the incoming end. (Somewhat lower down the line as the water temperature increases toward the generator.)



Notes:

1. Void fraction is defined as aereal fraction of pipe occupied by vapor.
2. Colder AFW will move curve downward, i.e., increase possibility of condensation water hammer.
3. Lower OTSG pressure will move curve upward, i.e., decrease possibility of condensation water hammer.