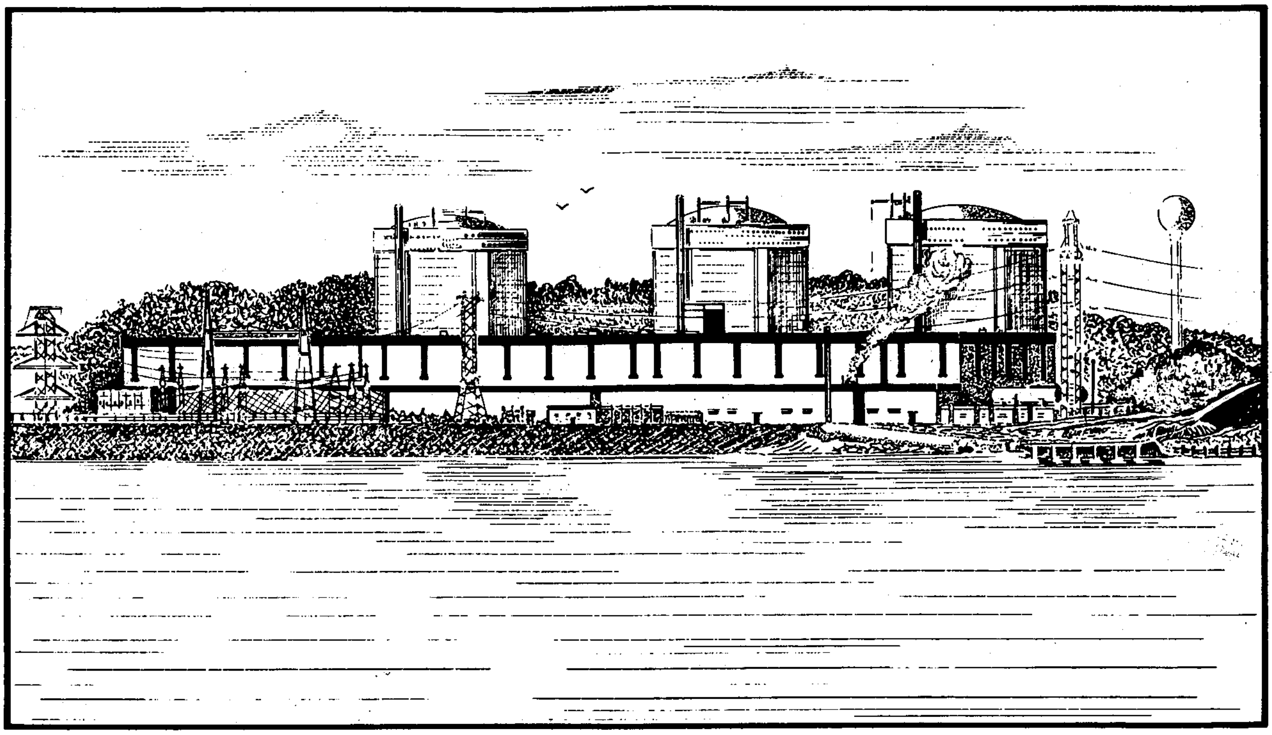
Oconee Nuclear Station
Unit 3



Abnormal Transient Operating Guidelines
(ATOG)

Part II - Volume 2
Discussion Of Selected Transients

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BABCOCK & WILCOX
NUCLEAR POWER GENERATION DIVISION

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D·R·A·F·T

- ATOG -

ABNORMAL TRANSIENT

OPERATING GUIDELINES

PART II

VOLUME 2

DISCUSSION OF SELECTED TRANSIENTS

ATOG GUIDELINES

PART II

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DISCUSSION OF SELECTED TRANSIENTS

INTRODUCTION

To develop the ATOG approach to integrated plant transient control six initiating events were studied. The results of these studies are given as examples:

1. Excessive Feedwater
2. Loss of Feedwater
3. Steam Generator Tube Leaks
4. Loss of A/C Power
5. Small Steam Line Break
6. Loss of Coolant Accident

These events show how the procedures of Part I are applied to specific transients and they amplify the general guidance given by the first volume of Part II. After reviewing these individual transients in detail, the operator should be able to see that regardless of (1) the initial event, or; (2) whether or not he can immediately identify the initial event or, (3) how many additional failures occur, he can keep the core and the plant in a safe condition by following Part I of these guidelines.

Each appendix discusses the plant designed response to its initiating event and, if available, gives an example of an actual transient. The discussion points out what operator actions are required and how Part I directs the operator to those actions.

Next is a review of the initial event compounded with other equipment failures. These are generally broken down into the failures of the fundamental methods of heat transfer control that are important to that event. Again the appropriate operator actions for that particular sequence of failures are given. Part I is referenced so that the operator can see how the basic procedure covers many multiple failures.

Each appendix contains a logic diagram which is a summary sheet for the transient being discussed. It is a simplified event tree which has been modified to show how correct operator actions will influence the outcome of the transient. The central vertical block diagram is the initiating event without additional failures. The failure paths branch out to the right and the left. The details on the diagram show identifying symptoms, including P-T, the corrective actions, and the limits to be considered for each additional failure that might occur. References are made to the appropriate parts of Part I and Part II. Nearly every kind of plant condition is covered either in the initiating event path or in the branches. Even if the initiating event is different from the examples, the principles to be used are the same, and therefore these examples illustrate how various other plant conditions can be handled.

Plots of various parameter trends are also provided. The times can only be approximate because they vary with such things as decay heat, initial power level and the size of the leak involved. However, they will give the operator a feel for the timing involved in these transients.

The operator should give special attention to the sections on LOCA and steam generator tube rupture because very detailed information has been prepared for these two events.

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In general, main feedwater overfeed can happen in three ways:

1. A failure of the Feedwater Control System to run back after reactor trip.
2. An operator error of feedwater control while in manual.
3. Equipment failure when the plant is in automatic operation.

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Excessive feedwater can occur at any time the main feedwater system is in operation. The plant may be tripped or at power. The steam generators will fill at different rates depending on what the plant power level is when the high flow begins. The rate of fill of the generator will be greater when the reactor is at low power (or tripped) than at high power. The overcooling effects on the reactor coolant system will be greater at low power. The reasons are that at low power less core heat exists to boil off the additional feedwater and the feedwater system (valves and pumps) has a lot of capacity left to overmatch the low reactor power. At full load, the valves and pumps are near full capacity and cannot open much more to increase feed flow.

Because the effects of excessive feedwater are different across the power range and because it can be caused by different failures, the rate of the Reactor Coolant System response will be different depending on what has happened. But all excessive feedwater additions will look similar.

The P-T curve and sequence of events shown in Figure A-1 depict a typical main feedwater transient that is terminated by the ICS before water enters the steam lines or the pressurizer is drained. The transient shown is also applicable if terminated early by the operator.

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The P-T curve and sequence of events shown in Figure A-2 depict an excessive main feedwater transient that is not terminated before water enters the steamlines or before the pressurizer is drained. The transient is initiated by a reactor trip from 100% power with a failure of feedwater to runback on the A steam generator.

Several important points should be noted regarding this transient:

- The affected steam generator can fill very rapidly, in this case three minutes after the reactor trip. Thus, if the failure causing the excessive feedwater condition is not corrected by the ICS the operator has little time available to prevent spillage into the steam lines. The high OTSG level trip of the MFW pumps provides backup to the ICS runback of MFW. However, due to the rapid nature of this transient and the potentially severe consequences the operator should not rely on this trip. For this case the automatic trip was assumed to fail.
- The operator is required to trip the RC pumps and raise OTSG levels to 95% on the operate range if the subcooling margin is lost. The additional cooling worsens the transient.

APPENDIX A

EXCESSIVE MAIN FEEDWATER

1.0 GENERAL TRANSIENT DESCRIPTION

Excessive main feedwater is a failure to control secondary inventory. It is an overcooling transient that results in too much primary to secondary heat transfer.

Excessive main feedwater is defined as the sustained addition of more water to the steam generator than can be boiled off by the available core heat to make superheated steam. This mismatch between heat source and heat sink will cause the steam generator level to rise and will cool the reactor coolant down. The severity and rapidity of the transient will vary with the size of the mismatch. Under worst case conditions (i.e., maximum mismatch and failure of the automatic MFW pump trip on high SG level) the excessive main feedwater flow must be terminated within two minutes to prevent water spillage into the steam lines. Thus, this is a transient that may require fast operator action.

As the reactor coolant temperature decreases, the RCS water volume will shrink, dropping pressurizer level. This in turn causes RCS pressure to drop. In the secondary side, while at power, the excessive feedwater will cause a loss of superheat and may cause a slight reduction of steam pressure. A reactor trip may occur

on low pressure or high flux. If a trip occurs and the excessive feedwater continues, the mismatch will be much larger (less core heat) and the steam generator fill rate and RCS cooldown rate will increase. If the shrinkage of the RCS water volume is sufficient to drain the pressurizer, the RCS will rapidly approach saturation conditions and ES will occur. A loss of subcooling margin will require that the RC pumps be tripped and MFW be directed to the upper nozzles or EFW be started. MFW or EFW will automatically raise SG levels to 50% and this additional FW flow will make the overcooling of the primary system worse. However, FW flow can be throttled to obtain a gradual increase in SG level and thereby limit the overcooling. If the loss of subcooling margin is caused only by the overcooling it will be temporary. When the subcooling margin is restored HPI can be throttled and RC pumps can be restarted.

If the excessive addition of feedwater to the steam generator is not stopped, water will spill into the steam lines. The ability of the steam system to maintain its integrity with water spillage is not known; therefore it is very important that the excessive feedwater transient is terminated before spillage occurs. In addition, it is highly desirable to stop the RCS cooldown before the pressurizer is drained and ES actuates. This will significantly reduce the magnitude of the transient and number of operator actions required, as well as limit challenges of protection systems and allow for quicker recovery to stable plant conditions.

However, the operator should throttle FW flow to obtain a gradual level increase. (See Part II. Section I.E. "Best Methods for Equipment Operation".)

- Excessive MFW flow to OTSG A was allowed to continue for almost three minutes after the steam generator was full. Thus a significant quantity of water was spilled into the steam lines with the potential for severe consequences. Since the consequences are not known, this discussion does not describe those effects.
- Even though the operator started HPI early in the transient, the HPI flow was not sufficient to overcome the shrinkage due to the cooldown and the pressurizer was drained.
- Once the overcooling transient has been terminated, the RCS will reheat and the water volume will swell. Since a large quantity of cold HPI water was added to the RCS, the operator must act to prevent the pressurizer from going solid. This will be discussed in more detail later.

Actual Plant Excessive Feedwater

On 3/18/77, an excessive feedwater transient occurred at an operating plant. The transient was not serious and the plant ended in a good condition because the operator recognized what was going on and took control quickly and the excessive feedwater was terminated automatically by the ICS after reactor trip. The steam generator did not fill and spill water into the steam lines; that is the most important

limit for this transient. The operator also turned HPI on and prevented the pressurizer from draining and that is another important limit. Because the cooldown was stopped early in this transient, the operator did not have to take drastic action to prevent filling the pressurizer solid due to reactor coolant reheating. After trip, the HPI flow was throttled.

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The plant was operating at 75% in a power escalation sequence. Because of the power escalation sequence, the overpower trip setpoint was set at 85%. The overfeeding started because main feedwater pump "A" failed and went to full speed.

The plant data shows main feedwater abruptly increasing to full flow on generator "A" and within about 30 to 45 seconds, its affects appear in other signals: "A" generator level goes up, T_{av} drops (because of the increased heat transfer), pressurizer level drops (because of shrinkage due to lowered T_{av}), and RC pressure lowers (because of the lowered pressurizer level). When T_{av} began to drop, the ICS pulled rods to try and keep temperature steady. The operator sensed the change in plant conditions almost immediately and quickly diagnosed the problem and took the appropriate action. He tried to cut back feed flow. He did this by manually reducing the feedwater demand; this did not work (more about this later).

He then put rod control into manual to try to raise T_{av}, and he also started HPI to stabilize the pressurizer level. The reactor then tripped at 85% power because of the low overpower trip setting. When the reactor tripped, the ICS switched into track and controlled the feedwater on level; because of the high steam generator level, the "A" main feedwater valve closed stopping the transient (the "A" start-up flow also closed down) and the "B" generator startup and main valves controlled to maintain the OTSG level after trip. When the plant tripped, the large inventory of cold water in the "A" generator cooled down the reactor coolant considerably (illustrating the effect on core heat removal due to the boil off of the extra inventory in the generator). The reactor steam generator heat transfer interplay is shown well by this example. After the trip, the reactor pressure increased and the pressurizer level increased; this is largely due to HPI.

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A look at steam pressure shows a very slight reduction (before trip) because of the cooling and condensing effect of the excessive feed-water on the steam in the generator. This illustrates the magnitude of steam pressure loss to be expected because of excesive feedwater; a greater loss would indicate an additional failure that the operator would have to correct. The reason "A" pressure is lower than "B" after trip is mostly because of one mis-set safety valve; the water in the generator would have some pressure reduction effect, but it

would be small. The lower steam pressure did have some effect on overcooling, but because it was only about 100 psi low, the effect was small.

This steam pressure loss was mostly an inconvenience; if it had been greater (about 200 psi), the pressurizer could have drained.

When the transient was over, the "A" steam generator was about half full. It took about two minutes to increase the level about 200 inches; most of the filling took place before trip when a high reactor power (75% to 85%) was available to boil the water off. The steam generator level increase was about 150" during this time (or about 3/4 of the total increase). If a main feedwater failure had occurred after trip, the rate of level rise would have been must faster. Excessive feedwater is probably the one transient the operator must react to faster than any other transient. The steam generator can fill and water can spill into the steam lines in as little as 3 to 4 minutes (after trip). The operator must act fast to stop feedwater and the equipment he chooses to use is important. He should understand that the equipment he elects to use may be the component that failed causing excessive feedwater and therefore may not respond. Thus he must be prepared to switch to an alternate device if necessary to terminate feed flow.

When the operator tried to correct this transient, he made the right choice of action: <u>cut back feedwater</u>. But the equipment he used did not respond. He used the ICS feedwater demand to try to run back the feed pump and valves. A post trip review showed that the feedpump controller had failed. It was essentially dead, and no signal would have made it respond. Excessive main feedwater is a complex accident which can be caused by about 20 different equipment failures (operator error with the feedwater control in manual can also happen). The ICS can have several failures. The accident can be too fast to try to figure out what failure has occurred. Therefore, the best way to correct a very fast overfill is to use the direct controls to trip both main feedwater pumps. Adequate time will be available to regain MFW or EFW to at least one SG.

Tripping both MFW pumps is the fastest method to stop excessive MFW flow and should prevent water entering the steam lines for even the most severe MFW transient. However, for much slower fill rates, or if a pump should fail to trip, the operator should isolate feedwater to the SG with the high, increasing level by closing the MFW control and isolation valves.

The following figures show actual plant data from the above transient. Note the large disparity between levels and feedwater flowrates for the two steam generators. This magnitude of mismatch should, and did, facilitate rapid recognition and response by the operator.

Figure A-3. ACTUAL PLANT RESPONSE DURING AN EXCESSIVE MAIN FEEDWATER TRANSIENT (RCS AND FEED FLOW PARAMETERS)

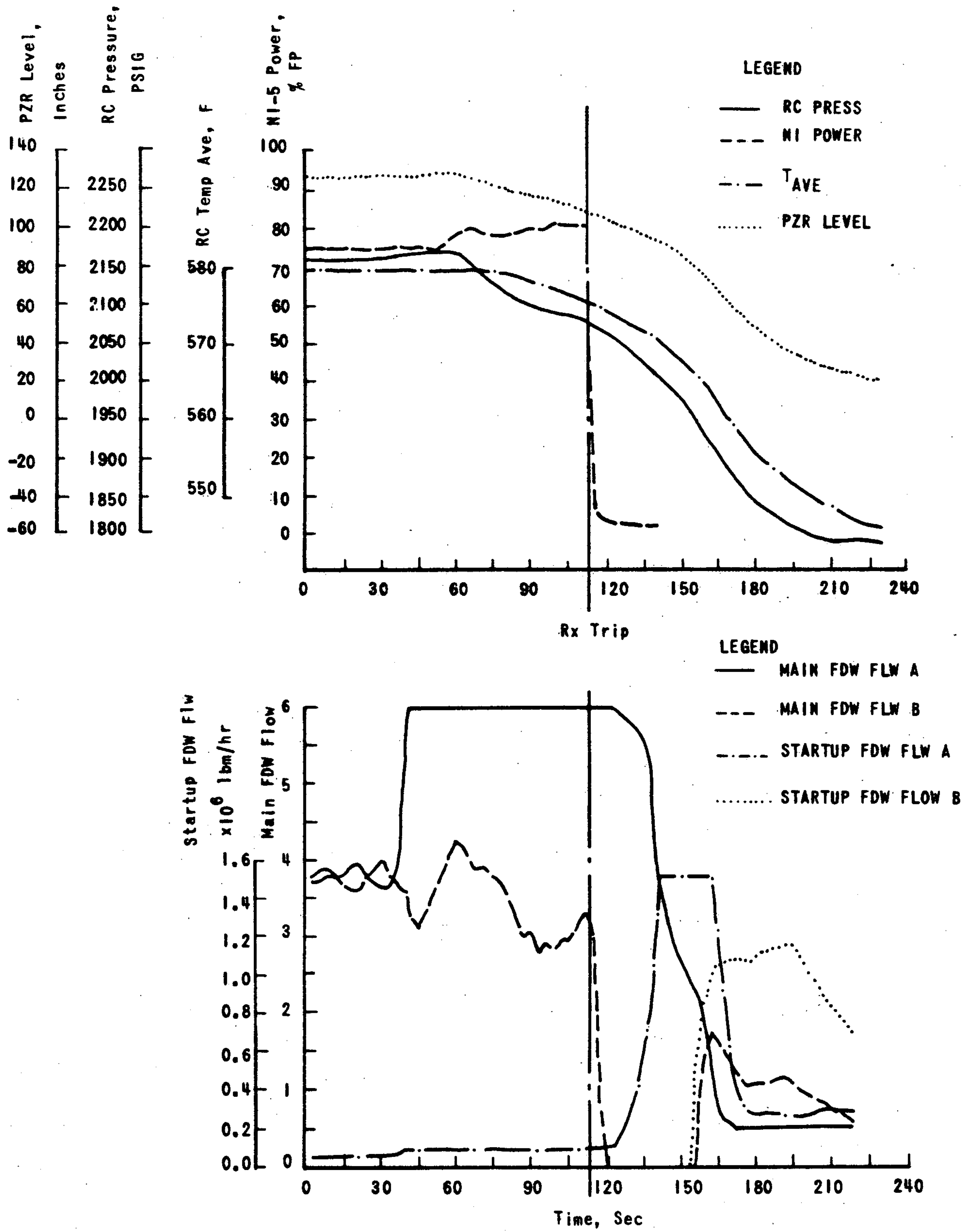
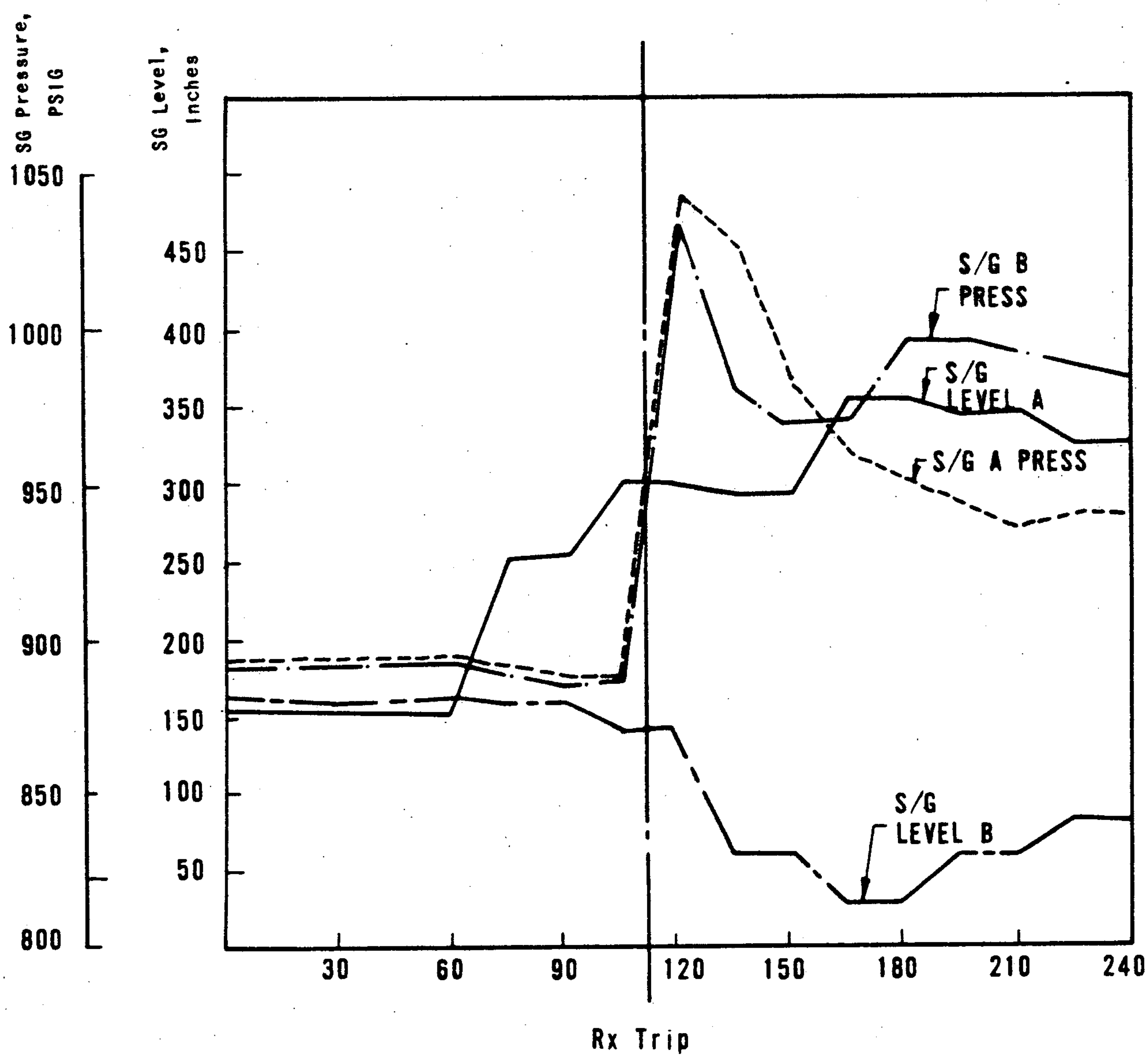


Figure A-4. ACTUAL PLANT RESPONSE DURING AN EXCESSIVE MAIN FEEDWATER TRANSIENT (STEAM GENERATOR PARAMETERS)



2.0 OPERATOR ACTIONS SUMMARY

Immediate Actions

- Attempt to close feedline isolation valves if the overfill is slow and the affected generator is obvious.
- Trip running MFW pumps to stop fast overfills.
- Start EFW and verify operation; control EFW to limit overcooling.
- Start HPI if pressurizer level is less than 80" and RCS pressure is decreasing.
- Follow remainder of Part I, Section III.C.

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Identifying Symptoms

- Excessive feedwater is an overcooling transient as shown below:

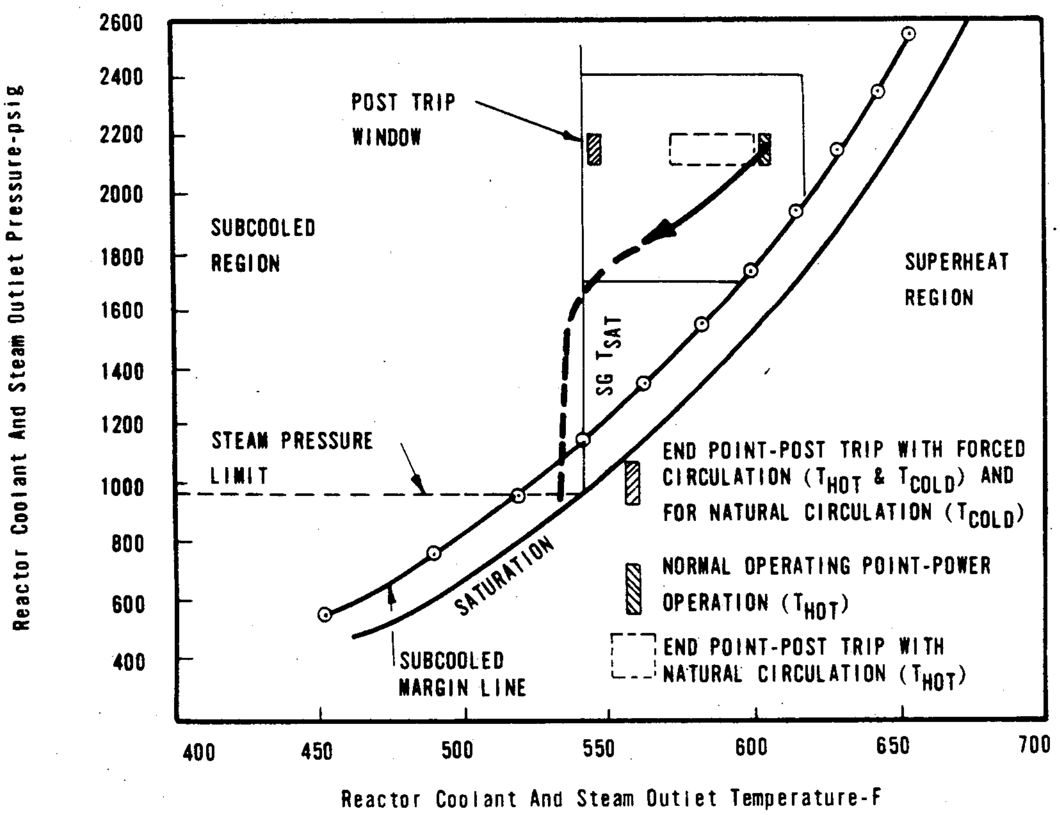

FIGURE A-5

- Other identifying symptoms to distinguish excessive feedwater from other overcooling transients are:
  - High steam generator level
  - High main feedwater flowrate

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NOTE: Rapid excessive feedwater transients, e.g., large MFW flowrate after reactor trip, will result in the pressurizer being in a near-drained condition by the time T_h/pressure exceed the post-trip window. Drainage and RCS saturation will occur very quickly after the post-trip window boundaries are exceeded (dotted path in Figure A-5) and water will enter the steam lines. Therefore, it is very important that the operator recognize the overcooling transient before the window boundaries are exceeded by checking MFW flowrates and SG levels. He should discover that excessive feedwater is in progress in Step 5.0 of Part I, Section II, "Vital System Status Verification".

The previous section discussed three of the many possible examples for excessive main feedwater transients. However, it can be seen from that discussion that the primary transient of concern is the rapid filling of a steam generator that is not automatically terminated early by the ICS run-back or high level trip of the MFW pumps. Such a transient requires rapid response by the operator. In addition, the operator must exercise caution

whenever feedwater is in manual control to prevent large feedwater mismatches from developing. Historically, oversights while in manual control have been a significant contributor to the frequency of excessive feedwater transients.

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This section will discuss how operator actions in accordance with Part I will terminate the overcooling transient and provide recovery to stable shutdown conditions. The assumed transient will be the same as that used for the second transient discussed in the previous section. A reactor trip occurs (for whatever reasons) and the MFP's run back to the low speed stop. However, the main feedwater valve for the "A" steam generator sticks partially open and thus allows a continuous excessive feedwater flow to that generator on the order of 8,000 gpm. This transient was selected because the ICS will not correct the excessive feedwater addition (assuming failure of the high level trip) and the "B" MFP will provide flow to the "A" steam generator longer if uncorrected since water in the "A" side steam lines will not affect operation of the "B" MFP turbine.

After performing the immediate actions of Part I, Section I, the operator will verify vital system status in accordance with Section II. Step 5.0 of Section II requires the operator to verify that feedwater has runback. He should check steam generator levels and feedwater flow rates and note that level and flow for the OTSG "A" are high. The corrective action noted is to trip the running feedwater pumps and start and verify proper operation of the EFW system. These actions will terminate the excessive feedwater and return the plant to stable conditions.

However, for illustrative purposes it is assumed the operator fails to note the abnormal conditions at this time. Step 12.0 of Section II requires the operator to verify that primary to secondary heat transfer is not excessive. The operator should note by RCS response on the P-T curve that an overcooling transient is in progress (see Section I.B of Part II) and therefore primary to secondary heat transfer is excessive. He does not need to concern himself at this time whether the overcooling is due to excessive feedwater, loss of steam pressure control, or loss of feedwater heaters. The procedure directs him to Section III.C of Part I.

Step 1.0 of III.C requires the operator to start HPI if pressurizer level goes below 80" and RCS pressure is decreasing. While full HPI flow will not maintain pressurizer level during this magnitude of shrinkage due to overcooling, it will slow down the rate of pressurizer level decrease and provide additional time for the operator to terminate the overcooling transient before the pressurizer drains (see Figure A-6). If RB pressure and temperature are normal, which will be the case for this particular transient, the operator is directed to Step 5.0 of Section III.C.

Steps 5.0, 5.2, 5.3, 5.4, 5.5, and 5.6 of III.C effectively isolate the steam generators from most failures that would cause overcooling and indeed, by isolating MFW in Steps 5.3 and 5.4, this particular overcooling transient will be terminated.

OTSG "A" level is high therefore the operator will perform the actions under Step 6.0. Maintaining RCS temperature at the present value by lowering the TBS setpoint will prevent filling the pressurizer solid due to RCS reheating and swell and is especially important since RCS inventory has been increased due to HPI. However, HPI is still in progress and must be throttled or stopped when the Subcooling Rule is satisfied. Establishing EFW flow to maintain OTSG levels will restore stable primary to secondary heat transfer.

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If the operator follows the guidelines in an expeditious manner and performs the actions such that MFW is isolated within two to three minutes following reactor trip, he will probably prevent water entering the steam lines and drainage of the pressurizer. In fact, the RCS will probably stay within the post trip window and recovery to stable shutdown conditions can be quickly achieved. If, however, the cooldown is allowed to continue to the point of pressurizer drainage, the RCS will rapidly approach saturation conditions and further operator actions and precautions are in order.

When the subcooling margin is lost the operator will trip the RC pumps, verify MFW transfers to the upper nozzles, and begin to raise SG levels. For this particular transient these actions will worsen the overcooling by removing pump heat input and spraying FW into the steam space (decreasing the heat source and increasing the heat sink). However, the operator should immediately throttle FW flow to obtain a steady, gradual increase in SG level thus minimizing the overcooling due to FW. (See "Best Methods for Equipment Operation" in Part II, Section I.E).

With FW throttled steam pressure should not decrease significantly. However, by this time the operator should recognize the overcooling transient. He will follow Section III.C of Part I and, in Steps 5.3 and 5.4, terminate the excessive MFW.

With the overcooling transient terminated, HPI flow will overcome the shrinkage of the RCS and rapidly recover RCS pressure and pressurizer level. The operator must again respond to prevent HPI refill, and reactor coolant swelling due to reheating, from filling the pressurizer. He should perform the following actions:

- Lower the TBS setpoint to a value near the corresponding saturation pressure for the existing cold leg temperature. This will limit RCS heatup and thus limit the resultant swell of the RCS inventory.
- Throttle HPI as soon as the subcooling margin is regained. This action will reduce the injection rate and allow a more gradual, stable recovery of pressurizer level. Throttling should be accomplished by using one HPI pump (preferably the normal makeup pump) and one injection line (preferably the normal makeup nozzle with the thermal sleeve).
- When pressurizer level returns on-scale low (with the RCS above the subcooled margin) and is increasing the operator should terminate HPI and realign for normal makeup/letdown operation.

- If desired, the operator can return the plant to normal post-trip conditions by gradually increasing the TBS setpoint and regulating RCS pressure with makeup and pressurizer heaters. He should also restart RCP's once the subcooling margin is restored.

The excessive feedwater transients discussed herein all involve a reactor trip. If a feedwater excursion occurs while at power that does not result in an automatic reactor trip, the operator should attempt to locate the failure causing excessive feedwater and correct it while at power if possible. A manual reactor trip at this time would result in a much larger mismatch between heat source and heat sink and thus make the transient more severe.

3.0 EXCESSIVE MAIN FEEDWATER WITH OTHER PLANT FAILURES

Introduction

The previous section described excessive main feedwater in general, but did not discuss other failures that might also happen at the same time. This section will show what symptoms to look for when other equipment fails and will show what steps the operator should take to restore the heat transfer from the core to the steam generators. The event that was chosen for simulation starts with the reactor at 100% power; a failure in feedwater system allows main feed to run away in one generator; automatic ICS corrective action to control main feedwater does not happen; the plant is tripped automatically on high flux or low pressure, and excessive feedwater continues. All the data that is shown starts from the time of plant trip.

Remember that all feedwater transients will not start from high power. They may look different than the examples used. The reason for these examples is to provide understanding, so close study of the effects is required.

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Branch Discussion

Figure A-7 has separate failure branches for loss of reactor inventory control (high and low), loss of secondary inventory control (high and low), and loss of secondary pressure control. Significant failures in RC pressure control, such as those due to overcooling or excessive HPI, are adequately covered by these branches; therefore separate branches specifically for loss of RC pressure control are not shown. Minor failures, such as loss of pressurizer heaters, are discussed at the end of the main transient path. This section will discuss each of these additional failure branches and illustrate how operator actions in accordance with the procedures in Part I and with the "Best Methods for Equipment Operation" in Part II, Section I.E, will restore proper control of the parameter in question.

These branches are structured to address the particular function failure in question even though the excessive feedwater transient may still be in progress. However, since some additional failures result in further overcooling of the RCS, actions to correct such failures may also correct the excessive feedwater condition. In any case, it should be understood that Figure A-7 and this discussion

are provided as tools to promote familiarity with expected plant responses. Another valuable tool to facilitate operator recognition and identification of overcooling transients is Figure 22, "Overcooling Diagnosis Chart," in Part II, Section I.C. The operator should become familiar with this chart.

Figure A-6 is provided to show key distinguishing parameters for excessive main feedwater that have a time dependency important to the operator in identifying both the type and severity of transient. The parameter plots show typical responses to a large excessive main feedwater transient. Arrows, where used, show the effect of other failures and operator actions on the time relationship.

One item of particular note on Figure A-6 is the effect of large excessive feedwater transients on steam pressure in the <u>unaffected</u> generator. The pressure is reduced because the primary system has been cooled so rapidly that the unaffected SG becomes, temporarily, a heat source and loses heat (and thus pressure) to the primary system.

<u>Loss of Reactor Inventory Control (High)</u>

A loss of reactor inventory control (high) exists whenever makeup or HPI flows are excessive causing the pressurizer to fill and overpressurizing the RCS for the existing plant conditions. Severe excessive feedwater transients involving large mismatches between

feed flow and primary heat input will result in RCS cooldowns and shrinkages that cannot be compensated for by full HPI flow. Thus, while the excessive feedwater transient is in progress, pressurizer level will continue to drop, although full HPI flow will slow the rate of level drop. This can be seen in Figure A-6 where manual initiation of HPI early in the transients (before automatic initiation by ES) shifts the curves for RCS pressure and pressurizer level to the right, i.e., more time is available before pressurizer drainage occurs. However, for smaller feedwater transients, and when the excessive feedwater has been terminated, full HPI flow will overwhelm the coolant shrinkage and result in a rapid increase in pressurizer level and RCS pressure. Rapid operator response will be required to prevent a solid pressurizer and RCS overpressurization.

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The operator should perform the following actions to restore proper RCS inventory and pressure control:

1) Throttle HPI as soon as the subcooled margin is restored and RCS pressure is increasing. Throttling should be accomplished using one HPI pump (preferably the normal makeup pump) and one injection line (preferably the one utilizing the normal makeup nozzle with the thermal sleeve).
2) If the RCS is reheating and thus swelling, lower the TBS setpoint to a value near the corresponding saturation pressure for the existing cold leg temperature. This will stop the

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RCS heatup and swell. If desired, the operator can then gradually increase the setpoint to allow a gradual heatup while controlling pressurizer level.

3) When pressurizer level returns on-scale low (with the RCS above the subcooled margin) and is increasing, the operator should terminate HPI and realign for normal makeup/letdown operation.

NOTE: Throughout Part I the operator is required to throttle HPI as soon as the subcooling margin is restored and to reduce TBS setpoints to maintain RC temperature. Thus, adhering to these guidelines will prevent a loss of RCS inventory control. This is discussed in more detail in Part II, Section I.E., "Best Methods for Equipment Operation."

Loss of Reactor Inventory Control (Low)

A loss of reactor inventory control (low) exists whenever makeup or HPI flow is insufficient to overcome a primary leak rate or the coolant contraction rate, resulting in drainage of the pressurizer. As stated previously, full HPI flow will be insufficient to maintain pressurizer level during severe excessive main feedwater transients, but will rapidly refill and repressurize the RCS once the overcooling is terminated.

Too little makeup or HPI flow, while undesirable, is not a major concern for this particular transient. If the overcooling is terminated before the pressurizer empties, the RCS will reheat and

the resultant swell will restore pressurizer level. If the overcooling continues, ES will actuate and HPI will initiate. It is extremely unlikely that at least one HPI pump will not start; however, should that occur the RCS will lose subcooling margin. The operator will trip the RC pumps.

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Control of FW to attain and maintain 95% level on the operate range will provide adequate core cooling while the problem with HPI is being corrected. The operator should throttle the FW flowrate to obtain gradual SG level increases and limit further overcooling.

Following the actions specified in III.A of Part I will restore primary system inventory control and subcooled margin.

Loss of Secondary Inventory Control (High)

A loss of secondary inventory control (high) exists whenever significantly more feedwater (main or emergency) is being injected into one or both steam generators than is required by existing plant conditions. It is an overcooling transient and is very similar to the main initiating event covered in this section (excessive main feedwater). However, there are basic differences in definition and plant response.

Excessive main feedwater was defined in Section 1.0 of this Appendix as basically supplying more feedwater than could be boiled off to

make superheated steam. The definition for excessive emergency feedwater (or excessive MFW after reactor trip) must differ slightly in that 1) the steam generator is at saturation conditions and 2) more importantly, whenever the natural circulation setpoint is in effect EFW (or MFW) will provide more flow than can be boiled off in order to raise SG levels to the appropriate setpoint. However, the rate at which FW builds SG levels can be excessive and overcool the primary system.

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In addition, excessive emergency feedwater will cause depressurization of the affected SG to a larger extent than excessive main feedwater. This is due primarily to the increased condensing action introduced by spraying colder EFW in near the top of the tube bundle (into the steam space).

Thus, even when the EFW (or MFW) system performs as designed, it can cause overcooling of the primary system, particularly when achieving the natural circulation level setpoint (20 ft) with low decay heat. Therefore, Section I.E of Part II states that the operator should throttle FW to obtain a gradual increase in SG levels and to maintain SG pressures. This will minimize the overcooling effects on the primary system. This action should only be required when the natural circulation setpoint is in effect since EFW or MFW cannot cause significant overcooling while attaining the low level setpoint for forced circulation. It should also be noted that automatic

level control of EFW flow will only occur if the motor-driven pumps receive an automatic start signal. Manual initiation of EFW will require manual flow control.

Should EFW or MFW flow control fail, the operator should recognize the overcooling as well as high FW flow and SG level higher than the appropriate setpoint. Following the actions in Part I, Section III.C for excessive primary to secondary heat transfer will terminate the runaway FW. Step 5.0 of III.C requires the operator to close the EFW regulating valves and steps 5.3 and 5.4 isolate MFW. The operator should not restore FW to the generator with high level until the failure causing excessive FW has been identified and corrected. Restoration of FW to the "good" generator will provide DH removal. The operator should align the EFW or MFW system to allow feeding of the good generator with both EFW or both MFW pumps for reliability while FW is being corrected and restored to the affected SG.

Figure A-6 shows the impact of excessive main feedwater overcooling compounded by overcooling due to excessive EFW. The curves for RCS pressure and pressurizer level will shift to the left, i.e., pressure reduction and drainage of the pressurizer will occur faster.

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Loss of Secondary Inventory Control (Low)

A loss of secondary inventory control (low) exists whenever too little feedwater is being supplied to the steam generators resulting in too little primary to secondary heat transfer and overheating of the RCS. This is an unlikely event since the initial condition was excessive main feedwater with too much primary to secondary heat transfer. In any case, should a total loss of both main and emergency feedwater subsequently occur, the operator will have more time available for corrective actions due to initial SG inventory increase caused by the excessive main feedwater transient. A detailed discussion is provided in Appendix B, "Loss of Main Feedwater".

Loss of Steam Pressure Control

A loss of steam pressure control exists whenever one or both steam generators undergo a pressure reduction significantly below the TBS reseat setpoint. It is an overcooling transient and will look similar (on the P-T curve) to an excessive feedwater transient. It can be caused by excessive main or emergency feedwater (discussed earlier) or by an unplanned steam flow through stuck open valves or a pipe break. Improper EFW flow control (or MFW flow control through the upper nozzles) will also result in a SG pressure reduction.

The operator will isolate both SG's (by isolating all FW, common steam extraction lines, and closing TBV's) and then monitor their respective levels and pressures. If both SG's stabilize, indicating a steam leakage path downstream of one of the steam extraction line isolation valves or in the TBV's, he can restore FW to both. If only one SG stabilizes he will restore FW to that SG for DH removal and allow the broken SG to boil dry. In the highly unlikely event that neither SG stabilizes, the operator must pick one for DH removal while trying to locate the leakage path. One SG may be broken and the other may have a leaking MSRV.

It should be noted that the overcooling caused by the excessive feedwater coupled with the overcooling due to loss of secondary pressure control may be too rapid and too severe to prevent pressurizer drainage and saturation of the RCS. Figure A-6 shows the impact of excessive feedwater overcooling compounded by overcooling due to loss of steam pressure control. The curves for RCS pressure and pressurizer level will shift to the left, i.e., pressure reduction and drainage of the pressurizer will occur faster. When the overcooling transient is terminated, the operator must react to prevent overpressurization of the RCS and possible violation of NDT limits.

Figure A-7 EXCESSIVE FEEDWATER LOGIC DIAGRAM

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APPENDIX B

LOSS OF MAIN FEEDWATER

1.0 GENERAL TRANSIENT DESCRIPTION

Loss of main feedwater is a failure to control secondary inventory. Loss of Main Feedwater (LOFW) is not by itself a severe transient. The plant was designed to respond automatically to this event so that the important plant parameters, like RC pressure and temperature, will stay within acceptable limits. The recent addition of an anticipatory reactor trip has helped reduce the severity of this transient even more.

However, when main feedwater is lost, some important backup systems like the Emergency Feedwater System are called into play. If these backup systems fail to function correctly, a much more severe transient may start. The important plant parameters may go outside their limits and the operator will have to step in to control them. So even though loss of main feedwater is not by itself a bad transient, it can be the starting point for some severe abnormal transient.

This section will discuss what can cause a loss of feedwater and how the plant will behave.

Causes for a Loss of Main Feedwater

The most common cause for a loss of main feedwater is a trip of the main feedwater pumps. This can be caused, for example, by low pump suction pressure which, in turn, could have a number of causes including a loss of the condensate pumps. Main feedwater pump trip is significant because it always results in an anticipatory reactor trip. The anticipatory trip helps reduce the severity of the transient because it quickly drops the production of heat in the reactor core. Without the anticipatory trip, the core will continue to produce heat for up to 8 seconds after a LOFW before the reactor trips on high RC pressure.

LOFW can also be caused by inadvertent closure of the main feedwater control valves, the main feedwater isolation valves, or by a feedwater line break. If these failures occur, they may not result in an anticipatory trip. However, they are much less likely than a main feedwater pump trip. So for the remainder of this discussion, we will assume that the LOFW has been caused by a main feedwater pump trip and that an anticipatory reactor trip also occurs.

Plant Behavior Following a LOFW

For the first few seconds following a LOFW, RC pressure, pressurizer level and steam pressure all "spike" upwards. This happens because within a second after the main feedwater pumps trip, the reactor and

turbine also trip, and the effects of the turbine trip are seen more quickly than the reactor trip. The turbine trip causes steam pressure to increase and reduces primary to secondary heat removal. The reactor coolant gets hotter and expands slightly causing an increase in pressure and pressurizer level. On the P-T diagram below, this initial spike is shown (greatly exaggerated) to indicate its direction. In actual practice the spike is so small and so quick that the operator probably won't see it. (Note: Only for the case where there is no anticipatory trip does the spike become significant. In that case RC pressure will rise to 2300 psig before causing a reactor trip).

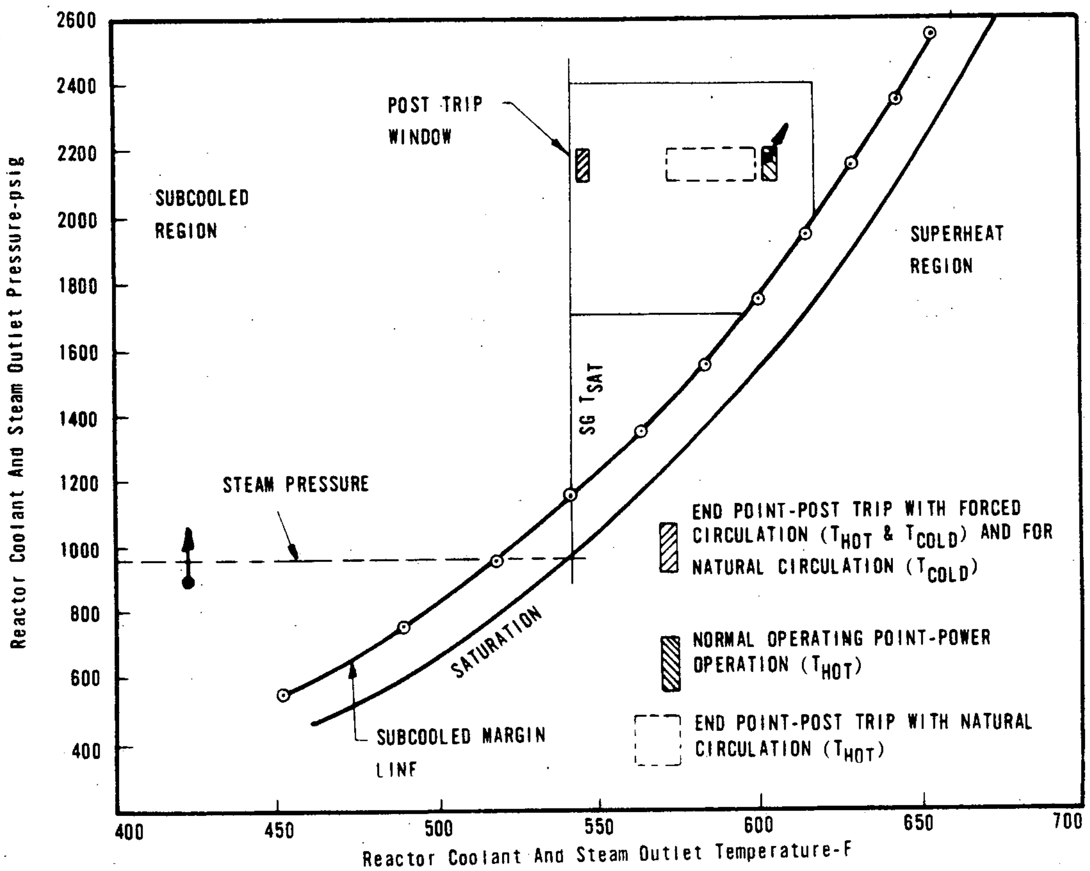

FIGURE B-1

Quickly the effects of the reactor trip become dominant. The loss of power generation in the core ends the overheating spike. As the steam generator inventory is boiled away, heat is removed from the secondary side faster than it is being supplied by decay heat in the core and the RCS cools. This part of the transient takes much longer and will probably be the first system behavior observed by the operator. The following P-T diagram shows the trend that he will see. Reactor coolant pressure, temperature and pressurizer level will fall as a result of the RCS cooling down. Pressurizer heaters will come on and makeup flow will automatically increase.

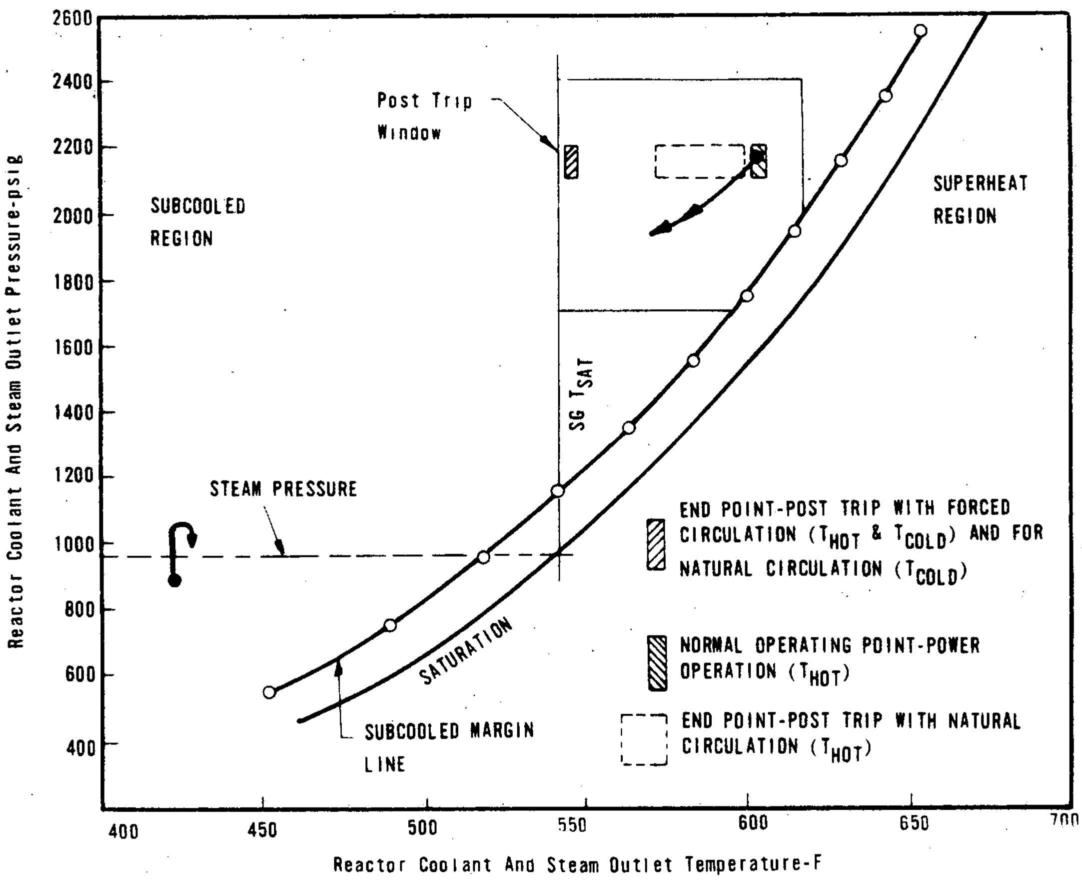

FIGURE B-2

Steam generator levels in both generators will continue to drop as the inventory is boiled away. At the same time, steam pressure will peak as the main steam safety valves lift. After the excess energy has been vented, steam pressure drops and these valves close. The turbine bypass valves will control steam pressure near the 1010 psig setpoint.

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Because the reactor trip and turbine trip occur so quickly after the main feedwater pumps trip, the transient up to this point will look almost identical to a normal reactor trip. For positive identification of a LOFW, the operator must rely on other indications such as the main feedwater pumps have tripped or the emergency feedwater pumps have been turned on. Main feedwater flow rate is a fast indication which also should be checked.

The emergency feedwater pumps are actuated early in the transient, but no EFW flow will occur until the steam generator levels have dropped to the low level setpoint. When this occurs, the EFW flow will be automatically controlled to maintain a nearly constant level and the heat removed from the steam generator will closely match the core decay heat. When this happens, RC temperature will remain nearly constant. Pressurizer level will slowly increase in response to the additional makeup flow, and operation of the pressurizer heaters will cause RC pressure to go up. This represents the final

stage of the LOFW transient as the system approaches its desired post-trip conditions. The P-T diagram below shows the expected system behavior during this period.

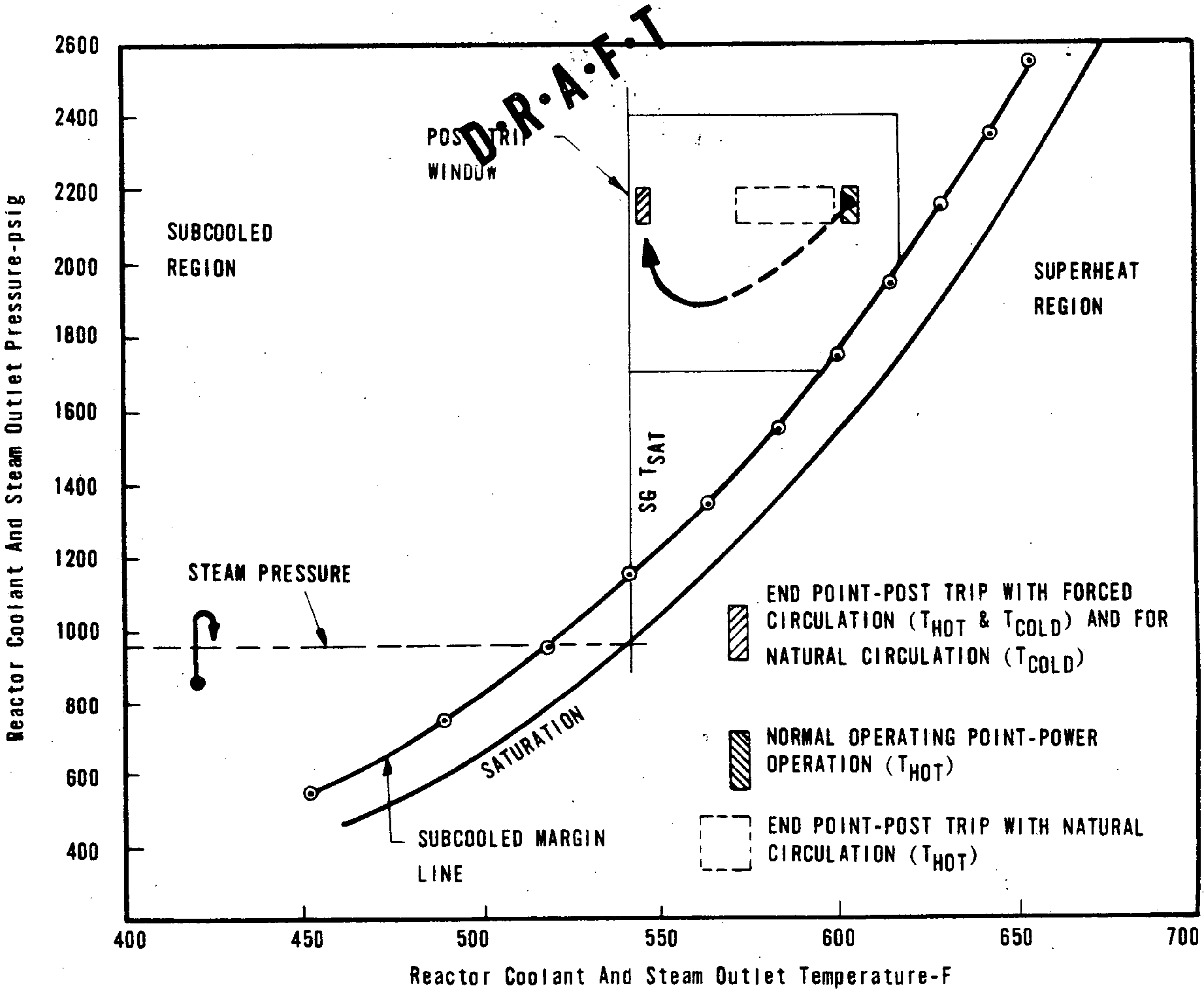

FIGURE B-3

2.0 OPERATOR ACTIONS SUMMARY

Immediate Actions

- Confirm emergency feedwater system is providing feedwater flow to the steam generators. If not, start EFW pumps and check EFW valve alignment. Control EFW flow rate to prevent overcooling. (EFW throttling should only be necessary when all RC pumps are off and the higher level setpoint for natural circulation is in effect.)

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Identifying Symptoms

- Main Feedwater Flow Rate
- Main Feedwater Pumps Tripped Alarm
- EFW controlling OTSG levels

NOTE: The RC pressure and temperature behavior is almost identical to a "normal" reactor trip.

If the plant responds normally, there is no need for the operator to take any immediate actions other than his normal post-trip response (Part I, Sections I and II). If the loss of main feedwater is compounded by other failures, Sections I and II of Part I will help the operator identify these situations and take the appropriate actions.

3.0 LOSS OF MAIN FEEDWATER WITH OTHER PLANT FAILURES

Introduction

The previous section describes the loss of main feedwater without additional plant failures. The plant is designed to automatically

handle the simple loss of main feedwater events without immediate operator action (although to limit overcooling effects when in natural circulation the operator should control EFW when it starts). However, a number of other plant failures can also occur at the same time which will compound the LOFW event and increase the complexity of the transient. These compound events require operator recognition and corrective action to mitigate the transient. This section will show what symptoms to look for when other equipment fails and will show what steps the operator should take to restore the heat transfer from the core to the steam generators.

There are three significant failures which may compound the loss of main feedwater event. These are:

- Loss of Secondary Inventory Control (Low)
- Loss of Secondary Inventory Control (High)
- Loss of Secondary Pressure Control

These events are shown on the Loss of Feedwater Logic Diagram (Figure B-5) and are discussed separately below.

Loss of Secondary Inventory Control (Low)

Loss of main feedwater is already a loss of secondary inventory control. If the LOFW is compounded by a failure of the EFW system, the steam generators will dry out and a loss of heat transfer will result.

The operator can recognize the total loss of feedwater by the lack of both MFW and EFW flow indication on the flowmeters and by the low steam generator level which will be decreasing below 25 inches on the startup range. Secondary steam pressure will decrease below the post-trip setpoint of 1010 psig once the steam generator dries out and can no longer produce enough steam to hold the setpoint pressure. Also, by following the steps in Part I, Sections I and II, the operator will identify that inadequate primary to secondary heat transfer exists at Step 11.0 of Part I, Section II. This step directs him to Section III.B of Part I where instructions for establishing HPI cooling, if necessary to protect the reactor core, and for restoring proper primary to secondary heat transfer are given. Use of HPI cooling protects the core while providing the operator the time necessary to correct EFW and/or MFW problems and restore feedwater to at least one steam generator. HPI cooling should be started when the loss of heat transfer is noted (T_{cold} decouples from steam generator saturation temperature and steam pressure drops). Also at that time all but one RC pump should be tripped; if the subcooling margin is lost the remaining RC pump should be stopped. After HPI cooling has been established, the operator should continue attempts to feed at least one SG from any available source.

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Figure B-4 is provided to show how RC hot leg temperature and RC pressure typically change as a function of time. On a loss of main feedwater with no EFW both RC temperature and pressure respond initially like a normal LOFW trip but then within minutes both parameters increase due to the inadequate primary-to-secondary heat transfer.

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Loss of Secondary Inventory Control (High)

The loss of main feedwater may be compounded by a failure in the EFW system resulting in EFW overfeed. This is an overcooling event because too much primary to secondary heat removal occurs; however, it is different from the excessive feedwater event of Appendix A because main feedwater has tripped off and cannot be causing the overfeed. Therefore, this transient will be much slower.

Excessive emergency feedwater can be recognized by the following symptoms:

- High steam generator level in one or both generators
- Continuous EFW feed flow indication above the correct level setpoint in one or both generators
- Falling steam generator pressure in the "bad" (overfed) OTSG

By following the instructions in Part I, Sections I and II, the operator will identify the excessive primary to secondary heat transfer at Step 12.0 of Section II. This step directs him to Part I, Section III C. Following those actions will terminate the excessive EFW. A more detailed discussion of Loss of Secondary Inventory Control (High) is provided in Appendix A.

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Figure B-4 shows the behavior of RC hot leg temperature and RC pressure versus time for the LOFW event compounded by excessive EFW. Steam pressure will decrease in both OTSG's and RC cold leg temperature will follow steam generator T_{sat} throughout the transient.

Loss of Secondary Pressure Control

A loss of main feedwater event may also be compounded by a loss of secondary pressure control. The loss of secondary pressure control can be caused by such things as a turbine bypass valve or a main steam relief valve failing in an open position. Any of these initiating causes will result in an overcooling transient.

Symptoms of a loss of secondary pressure control caused by steam leakage through an open safety or turbine bypass valve are: (1) rapidly falling steam generator pressure in both OTSG's with the "bad" generator pressure falling more rapidly; After a few minutes the "bad" generator pressure may be as much as 200 psi lower than the good generator pressure, and (2) low steam generator level in the "bad" generator and/or higher EFW flow to the "bad" generator.

The operator should follow the instructions in Part I, Sections I and II. Step 12.0 in Section II will identify the overcooling event and direct him to Section III C which provides the explicit instructions for identifying and dealing with the steam leakage.

These steps include isolating all feedwater to the "bad" OTSG by closing the control or isolation valves and verifying the TBV's are closed, and if the leak cannot be isolated then boiling the "bad" OTSG dry.

With a loss of secondary pressure control caused by a stuck open turbine bypass valve or main steam safety valve, the reactor coolant hot leg temperature and pressure decrease very rapidly. The time dependence of these parameters for a stuck open MSRV is shown in Figure B-4.

Figure B-4. TIME RELATIONSHIP OF LOSS OF MAIN FEEDWATER WITH FAILURES

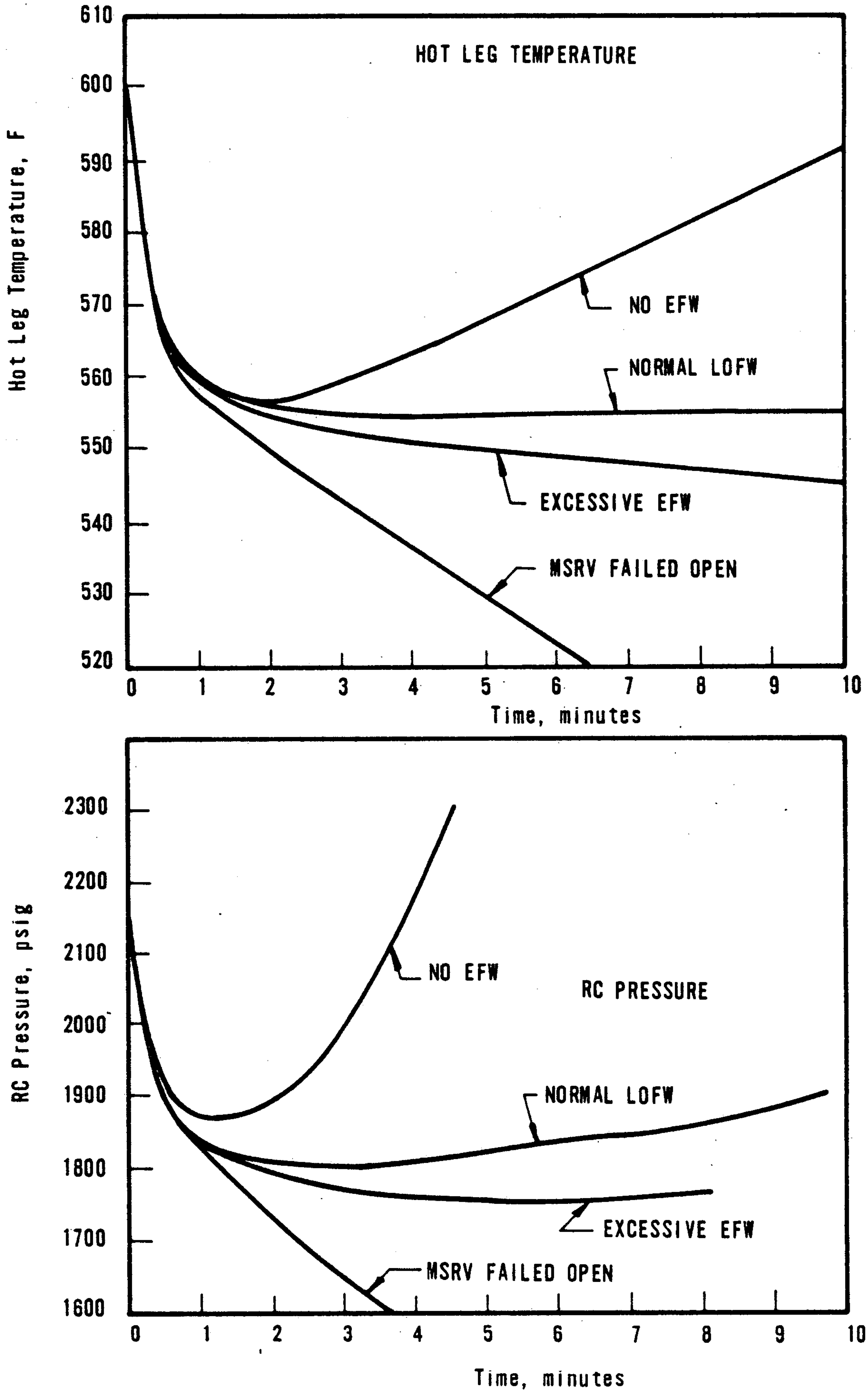
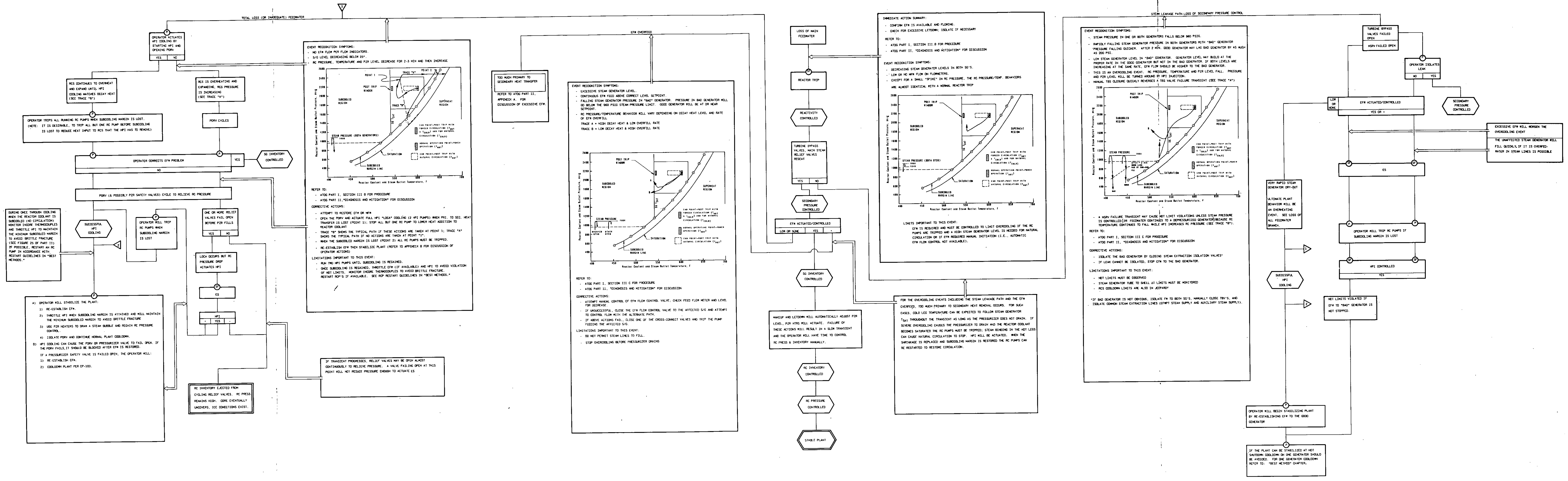


Figure B-5 LOSS OF FEEDWATER LOGIC DIAGRAM



APPENDIX C

STEAM GENERATOR TUBE RUPTURE (SGTR)

1.0 INTRODUCTION

A SGTR is a loss-of-coolant accident (LOCA) that allows reactor coolant to leak into the secondary side of the once through stèam generator (OTSG) where it is released into the steam plant. A SGTR is a serious accident; it contaminates the secondary plant and can lead to significant offsite doses if steam from the affected steam generator(s) is released to the environment. It can have the complications associated with a normal small break LOCA (see Appendix F).

A SGTR is a loss of integrity of the steam generator tubes. It can be a small leak of one tube or failures of more than one tube. SG tube failure can be caused by corrosion (bad water chemistry), excessive thermal or hydraulic loadings during severe plant transients, or mechanical wear due to foreign objects in the primary or secondary system. Tube failure can occur by itself or it can be combined with another severe plant transient.

The leak rate during a SGTR can be small (a few gpm) or quite large (several hundred gpm). Some of the major factors which influence the leak rate of reactor coolant into the steam generators are:

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1. The number of defective tubes,
2. The size (break area) of the tube failure(s),
3. The pressure and temperature conditions in the primary and secondary systems, and
4. The location (elevation) of the tube failure in the steam generator tube bundle.

On B&W plants, SGTR's have been limited to small leaks with leak rates less than 20 gpm. However, larger leak rates can occur. For a complete severance of one SG tube, a leak rate of approximately 400 gpm at normal steam pressure and temperature conditions would be expected. SGTR events with high leak rates have occurred at commercial nuclear plants.

The leak from a failed tube cannot be isolated and reactor coolant will continue to be lost until the plant is completely cooled and depressurized and the primary system loops have been drained. Hot shutdown is not a safe end condition for a SGTR.

Maintaining an RCS subcooling margin will prevent the equalization of the primary and affected OTSG pressures. The primary to secondary elevation difference will also prevent termination of the tube leak until the RCS is drained. Therefore, a rapid plant cooldown and depressurization is required for accident mitigation. Higher RCS pressures result in higher tube leak flows and ultimately higher offsite doses, so early detection and diagnosis of this accident is

very important. The RCS must be at cold shutdown before the BWST is depleted because recirculation from the sump is not possible, all injection water is lost through the steam lines. Therefore, the RCS cooldown and depressurization must be initiated as soon as safely possible. Since a tube leak is a small break LOCA, the general procedures for LOCA correction must be also followed (see Appendix F). Some modifications to the LOCA rules are required for this unique accident: 1) the leak rate through the tube will increase as subcooling margin increases, therefore a minimum subcooling margin should be maintained, and 2) delays in cooldown and depressurization must be avoided.

Delays of cooldown and depressurization can happen if failures occur in plant systems in addition to the tube leak. If possible, the additional failures should be corrected before final cooldown and depressurization to cold conditions. However, to minimize delays in the RCS cooldown and depressurization, the failures may have to be corrected during the cooldown.

This appendix will show characteristics of and corrective actions for large and small tube leaks, with and without additional failures.

2.0 GENERAL OPERATOR ACTIONS

The most important stages in the treatment of steam generator tube leaks are:

1. Diagnose that a SGTR is in progress and identify which steam generator has the leak.

2. Bring the plant to a stable, zero power condition by:
   - Performing a plant runback, or
   - Applying the appropriate Abnormal Transient Operating Guidelines if a reactor trip occurs during another accident.
3. Rapidly cool down and depressurize the RCS so that the RCS pressure is below the main steam safety valve setpoint.
4. Once the RCS temperature is approximately 500F (540F for small leaks), isolate feedwater to the generator with the tube leak. Isolate all steam extraction lines from the steam generator with the tube leak except the turbine bypas line. Leak sizes of 5-20 gpm are small and will probably not fill the generator before complete plant cooldown if RCPs are running. Slightly larger leaks can be accommodated depending on the time it takes to completely cool down. Isolation of the generator will help prevent contamination of the secondary plant and the cleanup efforts associated with it. For all large leaks and small leaks without RCP operation, secondary plant contamination will occur because the generator with the leak will have to be steamed periodically to prevent filling. The operator should make every effort to limit the degree of contamination in the balance of

plant without inhibiting mitigation of the SGTR event. This includes isolation of the auxiliary steam supply and the steam supplies to the MFW and EFW pump turbines and the second stage reheaters from the affected generator.

5. If additional component failures have occurred which will prevent a) plant depressurization with the spray or PORV, or b) prevent the use of both generators, then repair or bypass the failures if possible.

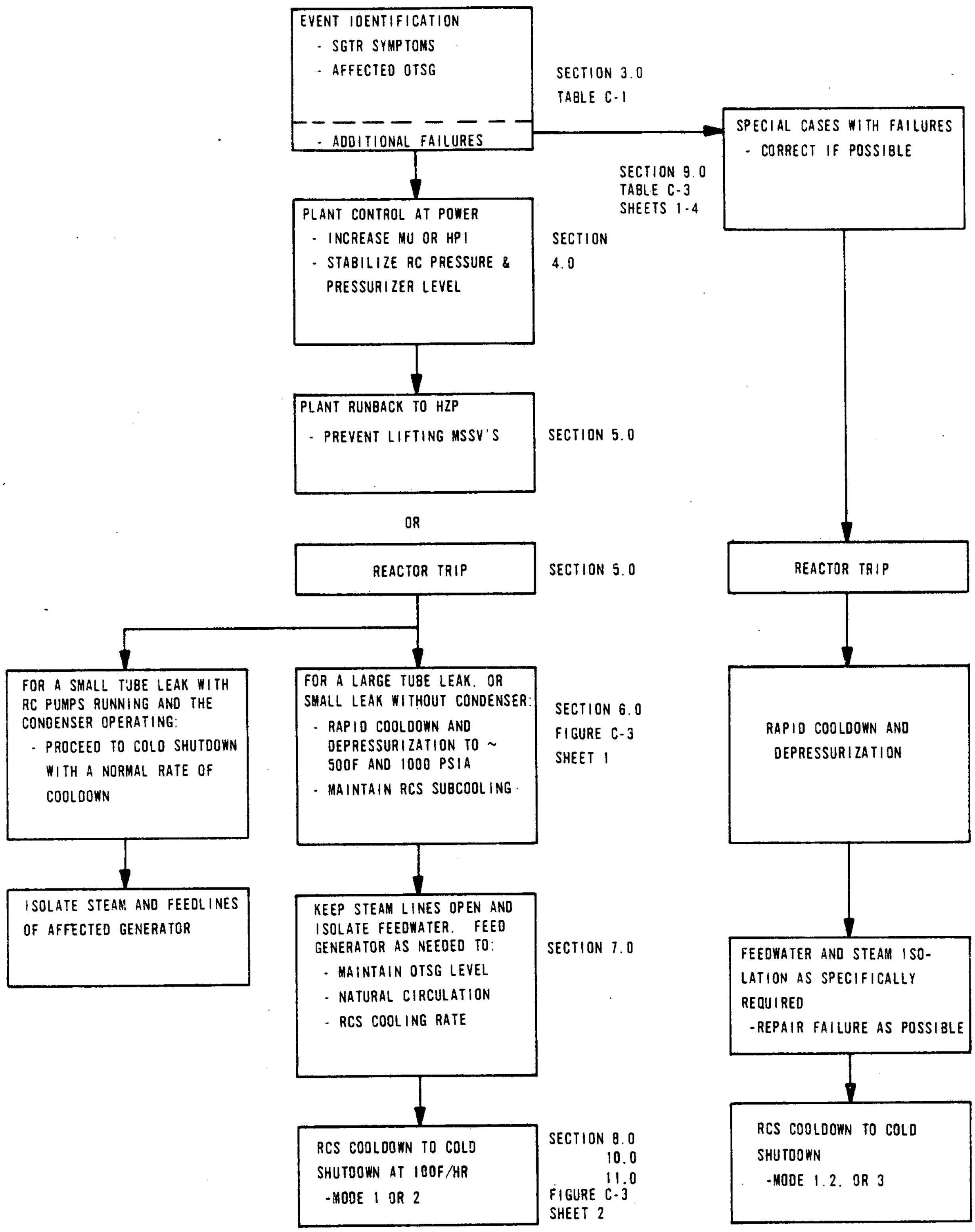
6. Cooldown and depresurize the plant to cold shutdown conditions while maintaining the reactor coolant subcooled and minimizing the offsite releases.

These steps are shown in the block diagram (Figure C-1). The text follows the outline of the diagram. The first part of the text shows actions for tube leaks without failures; actions to take if other failures occur with the tube rupture are discussed in Section 9.0.

During these stages, the tube leak must be treated to correct for the LOCA and also to limit the radioactive steam release. Since LOCA's are discussed at length separately (Appendix F), the following discussion will be focused on treatment of the tube rupture as a unique accident and will only discuss the LOCA as it related to plant control.

Figure C-3, which is at the end of this appendix, summarizes the general operator actions for a SGTR. Sheet 1 of Figure C-3 summarizes those actions required to identify the accident and to bring the plant to a stable zero power condition and Sheet 2 addresses plant cooldown. Table C-3 outlines actions to minimize offsite releases for SGTR's with additional equipment failures. These figures should be studied as a supplement to the follow-on sections.

Figure C-1. STEAM GENERATOR TUBE LEAKS-OPERATOR ACTION OUTLINE



3.0 EVENT DETECTION

The first stage for correction of steam generator tube leaks is prompt detection and determining which generator has the leak. It is mandatory to know that a tube rupture has occurred and it is very important to know which generator is affected. Table C-1 summarizes the ways a SGTR and the affected steam generator can be detected. Secondary plant radiation levels are the best and most timely indicators of a SGTR. The other SGTR symptoms (LOCA or asymmetric high water level in one generator) are best used as back up methods in case the plant's radiation detection equipment is inoperative. Figure C-2 shows the P-T diagram and a sequence of events for a large tube leak without operator intervention prior to trip; as expected, the tube leak looks like a small break LOCA (see Appendix F).

Table C-1. WAYS TO DETECT A SGTR

1. Abnormal Radiation Level in Steam Plant • High readings and alarm of condenser radiation monitor • High readings and alarm of main steam line radiation monitors.	The secondary plant radiation monitors give the best indication of a primary to secondary leak. They are effective when the SGTR is the only accident or when a SGTR occurs along with another accident. Both the steam line and condenser radiation monitor will indicate a SGTR; but, the individual steam line monitors will show which steam generator is leaking. Typically, a condenser radiation alarm and an alarm on one of the steam line monitors will sound if a SGTR occurs. NOTE: A local frisk of the steam lines using portable equipment can also be done to find the affected SG.
2. LOCA Symptoms • Decreasing RCS Pressure • Decreasing Pressurizer Level • High MU FLow (or HPI initiation) • Low Letdown Storage Tank Level	The LOCA symptoms depend on the size of the SGTR (leak rate) and may not show up immediately if the leak rate is within the capacity of the normal MU system. These symptoms are best utilized as confirmatory indicators of a SGTR. NOTE: A SGTR can be distinguished from a normal small break LOCA in that normal RB conditions (Pressure, temperature, and radiation) should exist unless additional equipment failures occur.
3. Asymmetric SG Conditions • Increasing water level with zero feedwater addition.	Once the plant is tripped, asymmetric OTSG water levels may develop. When feedwater is stopped and water level is at or above the appropriate control setpoint, the water level in the affected SG may continue to rise because of the primary to secondary leakage. This method of detection should be used only as a backup because it is effective only for very large leak rates. For small leaks asymmetric water level conditions will not develop; the reactor coolant will simply boil off as if it were normal feedwater. The differences in indicated feedwater flows between the unaffected and affected OTSG will not be significant enough to detect.
• High activity levels or boron concentrations in the secondary water inventory.	The affected SG can also be identified by drawing a SG water sample. The affected SG will contain some activity and boron due to the presence of reactor coolant. This method of detection should be used only as a last resort because it is time consuming and requires action outside the control room. Coolant sampling becomes less effective as the transient evolves and the reactor coolant is mixed with the secondary coolant.

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4.0 PLANT CONTROL AT POWER FOLLOWING A SGTR

The second stage for correcting a SGTR is to stabilize reactor coolant (RC) pressure and pressurizer level so that the plant may be run back without tripping. Tube leaks will result in decreases in RC pressure and pressurizer level. The makeup (MU) system will automatically increase MU flow to stabilize pressurizer level and the pressurizer heaters will come on to restore RC pressure. Letdown storage tank (LDST) level will drop, but must not be allowed to drain. Action must be taken to supply the LDST with a water supply with a boron concentration equal to or greater than RCS boron concentration. Larger tube leaks or a complete rupture of a tube will require starting a second makeup pump or manual initiation of HPI. Action to start HPI should be taken if pressurizer level cannot be maintained with MU. Shift HPI pump suction to BWST if unable to maintain LDST level. If the leak rate is greater than the MU or HPI flow rate, pressurizer level and RC pressure will drop, the heaters will turn off on the low level interlock, and a reactor trip will occur. A complete severence of one SG tube (approximately 400 gpm) will result in a reactor trip in approximately 10 minutes if the operator does not initiate HPI (refer to Figure C-2). For tube ruptures with a flow greater than approximately 700 gpm (more than one tube ruptured), the rate of RCS depressurization will be too rapid to prevent a reactor trip, even with initiation of full HPI. Letdown should be secured as rapidly as possible for all SGTR's.

Once RC pressure and pressurizer level have been stabilized ("normal" plant conditions) by any of the above actions, the third stage of actions (Section 5.0) should be followed. By achieving RC pressure and inventory control, the plant may be manually run back. If a reactor trip does occur, the minimum RC subcooling margin and pressurizer level must be re-established. Actions to be taken are discussed in Section 5.0.

5.0 PLANT CONTROL TO HOT ZERO POWER (See Sheet 1 of Figure C-3)

The third stage for correcting a SGTR is to bring the plant to hot shutdown (unless the leak is large enough to depressurize the plant to a low pressure and automatically trip the plant). Depending on the leak size and other factors, the plant may be runback or it may trip. It is preferable to run back as much as possible, even if a reactor trip is imminent, to avoid or limit lifting the steam safeties and releasing radioactive steam to the environment. Tube leaks can occur alone or other plant failures can occur at the same time. If other plant failures occur at the same time, it may be difficult to establish hot shutdown and then proceed with cooldown and depressurization. Generally, but not always, the best course to follow, if possible, is to fix the other failures while taking actions to cool the plant down. This section will show how to establish hot shutdown and prepare for cooldown with tube leaks only. Treatment of other failures will be given in a later portion of the text (Section 9.0).

Action should be initiated to stablize pressurizer level and RCS pressure while conducting a plant runback (5% per minute) to low power without tripping. RCS inventory should be closely watched during the plant runback and subsequent cooldown.

Upon reaching a low power level where the turbine bypass capacity is sufficient to avoid lifting the steam safeties, the plant can be tripped as follows:

1. Place TBV's in manual and open to unload turbine.
2. Unload turbine generators and trip the turbine.
3. Trip reactor
4. Place TBV's back in automatic with appropriate pressure setpoints or control header pressure in manual (at operator discretion).

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If pressurizer level has not been maintained and a reactor trip should occur, there is a good chance that the pressurizer will drain, ES will actuate on low RCS pressure, and the reactor coolant subcooled margin will be lost. If this happens, the operator's actions are to:

1. Ensure full HPI flow.
2. Trip the RC pumps immediately following the loss of subcooled margin.
3. Ensure that MFW flow is diverted through the upper nozzles and is controlled.
4. Throttle HPI once the reactor coolant subcooled margin is regained and maintain pressurizer level at 100 inches or greater.

5. Bring the plant to a hot stable condition so that a plant cooldown may be initiated.

The status of the plant is now at hot zero power. The next step is to begin a rapid cooldown and depressurization.

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6.0 RAPID COOLDOWN AND DEPRESSURIZATION

Once the plant has been brought to a hot zero power condition by a runback or has been stabilized following a reactor trip, a rapid cooldown and depressurization of the RCS should be started while maintaining the reactor coolant with a minimum subcooled margin.

NOTE: THE RAPID COOLDOWN ONLY APPLIES TO LARGE LEAKS OR CONDITIONS WHEN THE CONDENSER IS NOT OPERATING. IT DOES NOT APPLY TO SMALL LEAKS WITH THE CONDENSER AND RC PUMPS OPERATING, WHERE A NEAR NORMAL COOLDOWN SHOULD BE USED.

This action is required to minimize the offsite dose because:

1. By reducing RCS pressure, the primary to secondary leakage will decrease and
2. By decreasing RCS temperature and steam pressure, radioactive steam release through the steam safety valves is less likely.

The objective of the rapid cooldown is to bring the RCS subcooled temperature to a value that corresponds to a saturation pressure which is below the steam safety valve setpoint. This action will prevent inadvertent lifting of the steam valves and will limit atmospheric radiation releases. The initial rapid cooldown and depressurization should bring the RCS pressure and temperature to about 1000 psia and about 500F. The rapid cooldown rate should be about 100 F/Hr.

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The plant cooldown and depressurization should be continued to cold shutdown conditions because the primary-to-secondary leak will not stop completely until the loops are drained.

Prior to and during the cooldown and stabilization period, the operator should check the status of several things so that he can decide on the best method for cooldown.

1. Estimate leak size by maintaining constant pressurizer level. Normal makeup will be adequate for a small tube leak. A large tube leak will require initiation of HPI.
2. Verify the operability of all TBV's. Any leaking or stuck open valves should be manually blocked closed (refer to SGTR with other failures section).
3. Verify the availability of both OTSG's. Assure both can be fed and are able to maintain pressure. If not, refer to SGTR with other failures section.
4. Verify operation of RC pumps for use of pressurizer sprays. If tripped and the RC conditions permit, restart one pump per OTSG (one in spray loop) as soon as possible.

7.0 ISOLATION OF AFFECTED SG

During the rapid cooldown and depressurization of the plant, isolation of the affected steam generator is recommended once the reactor coolant temperature approaches 500 F. Below this temperature, steam pressure will be below the secondary safety valve lift setpoints and reactor coolant subcooling requirements can be met. Isolation is defined as:

1. Stopping MFW and EFW to the affected generator; only enough water should be added to maintain level at the low limit. If in natural circulation enough water should be added to induce and maintain natural circulation.
2. If the condenser is available, the TBV's should be used if steaming is needed.
3. If the condenser is not available, the ADV's will have to be manually opened or the RCS must be cooled down by HPI cooling (very slow).
4. Switching both the main and emergency feed pump and other steam supplies to the unaffected SG.

The reactor coolant which leaks into the steam generator eventually must be steamed to limit the water accumulation in the generator. Steaming must be done when 95% on the operator range is reached. Steaming may also be needed to maintain natural circulation so that a reasonable rate of RCS cooldown to cold shutdown can be achieved. The decision to steam should be based on the actual situation; RCS

cooling, steam generator level, tube-to-shell ΔT, and the need to add feedwater for natural circulation will help make this decision.

Feedwater should be isolated until the leak size is known. Small leaks will not provide enough water to the generator to maintain a minimum level and therefore periodic feeding will be required to keep the shell cool and rate of cooldown constant. Large leaks will provide enough water without feedwater addition. Feedwater may be initially required to establish and maintain natural circulation in the loop with the tube leak. A decision to add feedwater must be based on the actual plant situation; RCS cooling, steam generator level, tube-to-shell ΔT, and the natural circulation condition will help make this decision.

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If the plant is in natural circulation, feedwater addition and steaming will be required from both loops to prevent the reactor coolant in either loop from going saturated in the "candy cane" of the hot leg and forming a steam bubble which will interrupt natural circulation. When natural circulation is stopped, further plant cooldown and depressurization will be significantly reduced.

Although the intent of isolation is to "bottle up" the affected SG and to use it as a storage tank for the additional reactor coolant that escapes during the remainder of the plant cooldown and depressurization, total isolation is not always possible. For large tube

leaks steaming may be required; feedwater may have to be added for some other situations. Level control in the steam generator and a high rate of RCS cooldown will provide a lower radioactive release for large leaks than a slow cooldown. A fast rate of cooldown will limit the total water that leaks into the steam generator, limiting the offsite releases. A timely cooldown will also allow the plant to be brought to a cold, depressurized condition before the BWST is drained. Therefore, a rapid rate of cooldown (100F/hr) should take priority over isolation of the OTSG.

Specific isolation actions for various plant conditions are addressed below (additional failures that may occur are covered in Section 9.0):

A. <u>Small Leaks (Approximately 20 gpm) with RC Pumps Running</u>

For this condition, the reactor coolant pumps will force circulation through the isolated loop so it will not stagnate and flash; a continuous uninterrupted cooldown with the other generator is mandatory. The cooldown rate should be close to normal, but the tube-to-shell temperature difference should be monitored and maintained within <u>normal</u> limits (60F) to avoid excessive tube stress. Feed addition should be maintained at the low level limits to keep the lower shell covered. Depressurization of the RCS is required continuously to maintain a

minimum subcooling margin while proceeding with RCS cooldown. If heat removal does become interrupted, then a path to the condenser from the affected OTSG should be opened to prevent radioactive release through the safety valves or ADV's of the isolated generator. Intermittent feeding to the isolated generator may be required to maintain a minimum water level if minor steam leaks occur. A leak of this size may not be large enough to fill the generator before the cooldown and depressurization are completed, however, the leak rate and the steam generator level should be monitored at all times. If the level increases or the leak rate changes, the generator may have to be steamed (preferably to the condenser). Steaming of the affected OTSG may be required at lower RC temperatures to maintain a reasonable rate of cooling. (COOLDOWN MODE 1 should be used for the entire transient if the condenser and RCP's are available - See Figure C-3, Sheet 2.)

B. <u>Small Leaks (Approximately 20 gpm) with RCP's Off</u>

Intermittent or minimal continuous feeding and steaming may be required to maintain natural circulation in the affected loop and to prevent stagnation and flashing in the RCS. The cooldown rate should be close to 100F/hr but may have to be lowered so that makeup can keep up with the leak and the RCS contraction. Pressurizer level should be maintained. The tube-to-shell temperature difference should be maintained within the emergency

limits (150F). Continuous feeding and steaming of the affected OTSG, at low flow rates; may be required to maintain 100 F/hr cooling and the tube-to-shell limits. (Cooldown Mode 2 should be used for duration of transient - See Figure C-3, Sheet 2.)

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C. Large Leaks/Ruptures with RCP's On or Off

If the tube leak is large enough, the affected OTSG will contribute significantly to RCS cooling. The best approach is to steam the affected OTSG at a rate adequate to maintain a constant level. The rate of cooling (100 F/hr) can then be controlled by steaming the unaffected OTSG. If the affected OTSG begins to fill, steaming should be increased to limit level to 95% on the operate range. The rate of RCS cooling may be temporarily exceeded to achieve level control. Once the 95% level is reached, continuous steaming of the leaking OTSG, without feedwater addition, will be required to maintain a constant rate of RCS cooling and to prevent overfill. If isolation of feedwater and adequate steaming of the affected OTSG is achieved early in the transient, the difficulties associated with OTSG overfill can be avoided. Toward the latter stages of the cooldown, the affected OTSG may begin to fill because the reduced core heat rejection will be unable to boil off the leaking reactor coolant. If this occurs, most of the heat removal should be transferred to the affected OTSG. Intermittent feeding and

steaming of the unaffected OTSG may be required to maintain natural circulation (if RCP's are tripped), 100 F/hr cooling, and the emergency tube-to-shell limit. (Cooldown Mode 2 should be used - See Figure C-3, Sheet 2.)

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8.0 COOLDOWN AND DEPRESSURIZATION TO COLD SHUTDOWN

The final step for mitigating tube leaks is to bring the plant to a completely depressurized condition. The tube leak rate will be the lowest when the RCS pressure is approximately equal to the steam generator pressure. To achieve this condition, the RCS must be depressurized so that the decay heat removal system can be started. The RCS subcooling requirements will not allow equalization of RCS and secondary system pressures while the steam generator is removing heat. When the decay heat removal system is started for heat removal, the steam pressure can be allowed to remain slightly higher than the RCS pressure. Reverse leakage will occur, but the effect is minimal if this condition is not maintained for extended periods of time.

Although the leak flow will be lowest when the RCS is placed on the decay heat removal system and the steam pressure is then increased by OTSG steam isolation, further cooldown and depressurization is required. Since the elevation of the hot leg "candy cane" is always higher than the tube leak, elevation head will cause the leak to continue. Special actions for handling the leak after the decay heat system is placed in the decay heat removal mode of operation will be described in Section 11.0.

Control of plant depressurization is needed to bring the plant to cold conditions; methods of depressurization are described in Section 11.0.

The approach chosen for cooldown to the decay heat removal system will depend on the conditions of the plant at the start of cooldown. To determine the mode of cooldown the leak size, condenser availability, RC pump availability, and other factors must be known.

The following subsection describes the various factors to be considered for final cooldown:

8.1 Selection of Plant Cooldown Mode

Figure C-3 shows the three modes of plant cooldown to be selected. There are two basic cooldown rates shown in this figure; "normal" and "emergency". The "emergency" approach applies to an RCS cooldown when either one or two generators are available.

A "Normal" cooldown is defined by limits as:

- A close-to-normal cooldown rate is used
- Tube-to-shell temperature limits do not exceed 60F

Fuel decompression limits are not violated deliberately

An "Emergency" cooldown is defined by limits as:

- Tube-to-shell temperature limits do not exceed 150F
- Fuel decompression limits may be deliberately violated
- The RC temperature should be dropped to 500F as fast as possible
- After 500F is reached the cooldown rate should be 100F/hr., as long as that rate can be held. Steaming of both OTSG's will be required at lower RCS temperatures. The cooldown rate should be reduced when the tube-to-shell temperature limit approaches 150F (unless there is danger of running out of BWST inventory).

The mode of cooldown to be selected is primarily based on two conditions: leak size and availability of the condenser. These two conditions take into account the severity of the event. If the leak is large or the condenser is not available, offsite releases will be large and the emergency approach is chosen. Small leaks with the condenser and RCP's available will not allow a significant release so a normal approach is selected.

Mode 1 (near "normal" cooldown) is selected when:

a. The condenser is available, and

b. The tube leak is small, and

c. The RC Pumps are operating.

Note that all conditions must be satisfied to use the "normal" mode.

Mode 1 is a "normal" cooldown except that the affected SG is isolated when the reactor coolant temperature (T_{hot}) is reduced to less than 540F. It should be used for SGTR's where the leak rate is small (well within the capacity of the MU system) and no additional equipment failures have occurred. For this mode, all normal plant equipment or systems (condenser, RC pumps, pressurizer sprays, etc.) must be available. The plant should be cooled within normal plant cooldown limits (fuel compression limits, 60F tube-to-shell ΔT's, etc.). This cooldown mode will most likely be preceded by a plant runback since the leak rate is small. This cooldown mode is the most likely condition which will be faced, based on tube failure histories with the OTSG.

The entire cooldown must be by the single operating generator. To cooldown with one generator under these conditions the following apply:

1. RC Pumps will keep circulation in both loops; if all RC Pumps were off the reactor coolant in the isolated generator would flash and prevent cooldown and depressurization. The RC Pumps also allow spray depressurization which is more rapid and controllable than PORV depressurization.

2. The leak is small enough and the cooldown is fast enough with one generator so that the water accumulation in the isolated generator will not build up and require steaming to lower the water level. The generator can truly be used to store the leaking reactor coolant. The level should still be closely monitored and steaming should be started if the water level approaches 95% on the operate range.

3. Continuous heat removal by the single generator is needed. The single generator will not only remove heat from the reactor coolant and core but also from the secondary fluid in the isolated generator, which acts as a heat source. At lower RC temperatures, the additional heat removal requirements on the good OTSG may result in a significant reduction in the rate of RCS cooling. If the rate of cooling is significantly reduced or interrupted, the isolated generator TBV's should be opened to increase RCS heat removal, preventing overfill of the affected OTSG.
4. Depressurization should be controlled so that it closely follows the cooldown. A minimum reactor coolant subcooled margin should be maintained so that the leak rate is as small as possible.
5. Feedwater to the ruptured generator should be isolated after a normal level is established. The feedwater will be cooled as heat is removed from the generator and shell heat will reduce in

the region contacted by the water in the generator. Steam generator tube-to-shell ΔT should be monitored and normal limits maintained. The rate of cooldown should be controlled to maintain normal limits.

The "emergency" mode is selected when:

a. the condenser is not available, or

b. the leak is large , or

c. the RPC's are not available.

Note that if either one of the above conditions exists, the "emergency" mode must be used.

Figure C-3 shows emergency modes for conditions when one generator or both generators are available.

The cooldown mode 2 shown in Figure C-3 is to be used for a SGTR when HPI must be used to makeup for the leak rate and both steam generators are available and removing heat when the plant cooldown is initiated. This cooldown mode can be preceeded by a reactor runback or by an automatic reactor trip (if the SGTR is large or other equipment failures occur). The general actions used for this cooldown mode are as follows:

1. Once the plant has been rapidly cooled down and depressurized to about 500F and 1000 psi, pressurizer level has been stabilized, and the isolation step has been performed:

   - If the condenser is not available the ADV's must be controlled manually to continue the cooldown. If the condenser is available, then the TBV's should be used.
   - If the RC Pumps were tripped on loss of subcooling margin they should be restarted if conditions permit; restart of the pump in the spray loop taking precedence to make the pressurizer spray available (See Pump Restart Criteria in Part II "Best Methods of Equipment operation). If the RC Pumps cannot be restarted then the PORV must be used for depressurization.

2. From 500F and 1000 psig, continue the plant cooldown and depressurization at up to 100F/hr with the unaffected or both OTSGs, as required to maintain cooling, until the decay heat system can be started.

   - Maintain <u>minimum</u> subcooling margins to minimize the leak.
   - Monitor the water level in the isolated SG and initiate steaming if the water level rises above 95% on the operate range.
   - If the leak does not provide enough water to keep a <u>minimum</u> level (natural circulation or forced level as appropriate), then feed the ruptured generator as required.

- If the plant is in natural circulation, both generators must be fed and steamed as necessary to maintain natural circulation in both loops.
- Monitor tube-to-shell limits and slow the plant cooldown so as not exceed a ΔT of 150°F (See Part II, "Cooldown With One Steam Generator Out of Service".)

3. After the decay heat removal system has been started, continue to cooldown and depressurize to cold shutdown so the loops can be drained and repairs started.

The above cooldown procedure is an emergency measure. Its goal is to get the plant to a depressurized state as quickly as possible to stop the leak and minimize offsite releases. In this mode, fuel compression limits do not apply. Mode 2 can be applied when all plant systems are available or under more degraded conditions such as a loss of offsite power. Figure C-3 outlines this cooldown mode; it shows the ways the plant can be controlled and identifies monitoring and corrective actions which are unique to a SGTR.

Mode 3 cooldown is identical to Mode 2 except that only one steam generator is available for heat removal when the plant cooldown is started. As indicated in Figure C-3, this could result from equipment failures that result in a loss of feedwater to one SG or a

loss of steam pressure control to one SG. The failed steam generator may or may not contain the SGTR. The actions required for Mode 3 are the same as Mode 2 except that isolation of the affected SG at less than 540F is not required. For Mode 3, significant offsite releases can occur under some failure modes. Figure C-3 outlines this third cooldown mode and the circumstances when large offsite doses could occur.

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Since the Mode 3 "one generator cooldown" can only come about because of failures, it will be discussed in more detail in the following sections which address failures.

A summary of the SGTR conditions and appropriate cooldown modes follows:

SGTR Conditions	Cooldown	Comments
Normal		
Small Leak RCP's On (Condenser Available)	Mode 1	Isolate Affected OTSG
Emergency		
Small Leak RCP's Off Condenser Unavailable)	Mode 2	Isolate Affected OTSG (and/or
Large Leak RCP's On	Mode 2	Isolate Affected OTSG
Large Leak RCP's Off	Mode 2	Isolate Affected OTSG
One Generator Cooldown	Mode 3	Refer to Section 9.0

9.0 PLANT CONTROL FOR TUBE LEAKS WITH OTHER FAILURES

Some other plant failures, in addition to the tube leak, can make plant control very difficult. Some failures will make it difficult to establish a stable condition for cooldown; some will cause the cooldown rate to be much slower than desired. Some failures can cause the offsite releases to be large. Some failures can be repaired or bypassed and some cannot. All of the failures that can make plant control difficult will be of two kinds:

- Those that limit the heat removal of the steam generator(s) (these failures will limit the ability to establish hot shutdown and will limit the cooldown rate).
- Those that limit the ability to depressurize the RCS (these failures will limit the depressurization rate and may require the cooldown rate be limited in order to stay within NDT limits).

Table C-3 summarizes the major failures and the actions required to correct them. The general approach for correcting other equipment failures that occur in addition to the tube leak is as follows:

- Simultaneously:
  a. Rapidly depressurize and cool the reactor coolant system (if possible) and bring the system to the point where the RCS pressure is below the setpoint of the steam safety valves. Most failures will not prevent this step and may even cause the depressurization.

b. Diagnose the additional failures in accordance with the principles shown elsewhere in these guidelines.

- Correct the failure prior to or during the final cooldown to cold shutdown. Do not delay the cooldown even if the failure limits the rate of RCS cooling and depressurization. Details of the effects of and corrections for failures are shown in Tables C-2 and C-3.

- If the failures cannot be corrected to enable both generators to be used or the RCS cannot be depressurized with use of spray or the PORV, then the cooldown will be difficult. In general, these kinds of failures are limited to:

  a. Failure of a steam safety valve (open) will not allow a "normal" rate of feedwater addition (excessive RCS cooling rates will result). Turbine bypass failures may be corrected by isolating the failed valve(s) and do not apply; MSRV failures cannot be isolated. The plant must be cooled down with the remaining generator even if it has the leaky tube. Radiation dose release may be high if the open steam valve is in the same generator as the tube leak. During cooldown, the generator with the steam failure (low pressure) must be continuously fed at approximately 100 gpm, preferably through the main feedwater nozzles. Periodic feeding will maintain cooling of the lower shell and keep the tube-to-shell temperature difference within limits. If the generator is completely depressurized, extreme care

must be exercised when feeding. A flow path through the warmup valve can be used to achieve low feed flow rates (100 gpm). If the RCP's are tripped, it is important to provide some feeding of the depressurized generator to establish or maintain natural circulation. If the steam failure is in the affected (ruptured) generator and the leak flow is large, the tube leak will usually provide adequate cooling. Unlike any other transients which do not recommend feeding a depressurized generator, a SGTR requires RCS cooldown to terminate the transient; therefore, careful feeding of a depressurized generator is allowed.

Steam leaks will most likely occur outside containment and be caused by valve failures rather than piping failures (TBS, MSRV's, etc). If the steam leak occurs inside containment, it will likely be a piping failure in a steam, feed, or drain line. If the steam failure is on the steam generator with the tube leak and is inside the containment, the reactor coolant from the tube leak will be returned to the containment sump. Although adverse containment conditions may result, the offsite releases will be minimal after the RC temperature reaches 540 F. The time requirements for termination of the SGTR transient may then be relaxed because HPI sump recirculation can be utilized and offsite releases are minimized.

b. Partial loss of feedwater, where no feedwater is available to one OTSG, will normally require cooldown on the remaining generator. The rate of RCS cooling must be slow to prevent violation of the tube-to-shell temperature limits on the OTSG without feedwater. If the RCP's are tripped, the rate of cooling will be even slower; reactor coolant will not naturally circulate in the idle loop. The stagnant loop must be closely monitored for saturation in the hot leg. Releases offsite will be significant because of the slow rate of RCS cooling and depressurization. Every effort should be made to realign feedwater to both generators. If realignment cannot be made, plans should be arranged to replenish the BWST inventory to avoid loss of injection water. If feedwater is lost to the ruptured OTSG and large leak flows are present, the leak will provide enough feedwater to maintain cooling of the affected OTSG.

c. Pressurizer valve failures (open or leaking) will require constant HPI to maintain subcooling. Failures in the pressurizer steam space will cause HPI to fill the pressurizer. RCS depressurization from this solid water condition will require throttling of HPI without loss of subcooling. Offsite releases of reactor coolant may be significant due to the lengthy time that may be required to completely

cool and depressurize the RCS in this manner. Therefore, depressurization of the RCS should take precedence over a quench tank disc rupture. The BWST inventory should be carefully monitored when this mode of depressurization is utilized for extended periods of time.

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d. Total loss of feedwater (MFW and EFW) will require a solid water cooldown without the benefit of secondary heat removal. Attempts should be made to obtain feedwater from any available source. For cases of this type which require excessive cooldown times, BWST inventory should be carefully monitored. HPI inventory from the BWST will flow into the affected OTSG and ultimately be released offsite. It will not be returned to the reactor building sump and cannot be reclaimed. Therefore, backup water supplies for the BWST may be necessary and should be prepared. Offsite releases will be very significant.

10.0 LIMITS FOR RCS COOLDOWN

The following is a summary of the basic limits which the operator should be aware of while proceeding with the RCS cooldown for all SGTR's:

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1. Fuel decompression limits apply to the case of small leaks with RC Pumps running and the condenser available; they may be violated for other cases.

2. Normal tube-to-shell temperature limits (60F) apply to the case of small leaks with RC Pumps running and the condenser available. Emergency tube to shell temperature limits (150 F) apply to all other situations, except:

   The emergency limits may be violated at management discretion, if loss of core cooling injection water is eminent (BWST draining). The steam generators are likely to be unusable if the emergency limits are violated, however core cooling takes priority.

3. Cooldown rate is near "normal" for small leaks with the condenser available and RC pumps running. Cooldown rate is 100 F/hr for all other cases except where tube-to-shell temperature limits restrict cooldown rate. Overcooling should be avoided and the pressurizer should not be allowed to drain. If the plant is on natural circulation, the cooldown rate should be reduced to avoid RCS voiding (the recognition of void formation is discussed in Part II, Volume 1 of the ATOG). Cooling with both OTSG's may be required to maintain 100 F/hr cooling.

4. The minimum subcooling margin should be maintained. Excessive subcooling should be avoided to keep a minimum leak rate.
5. The plant should be completely cooled down and depressurized before the BWST is drained. Simply placing the decay heat removal system into operation is not adequate.
6. It is desirable to place the plant on the decay heat removal system before poor quality backup water must be injected into the steam generator by EFW.

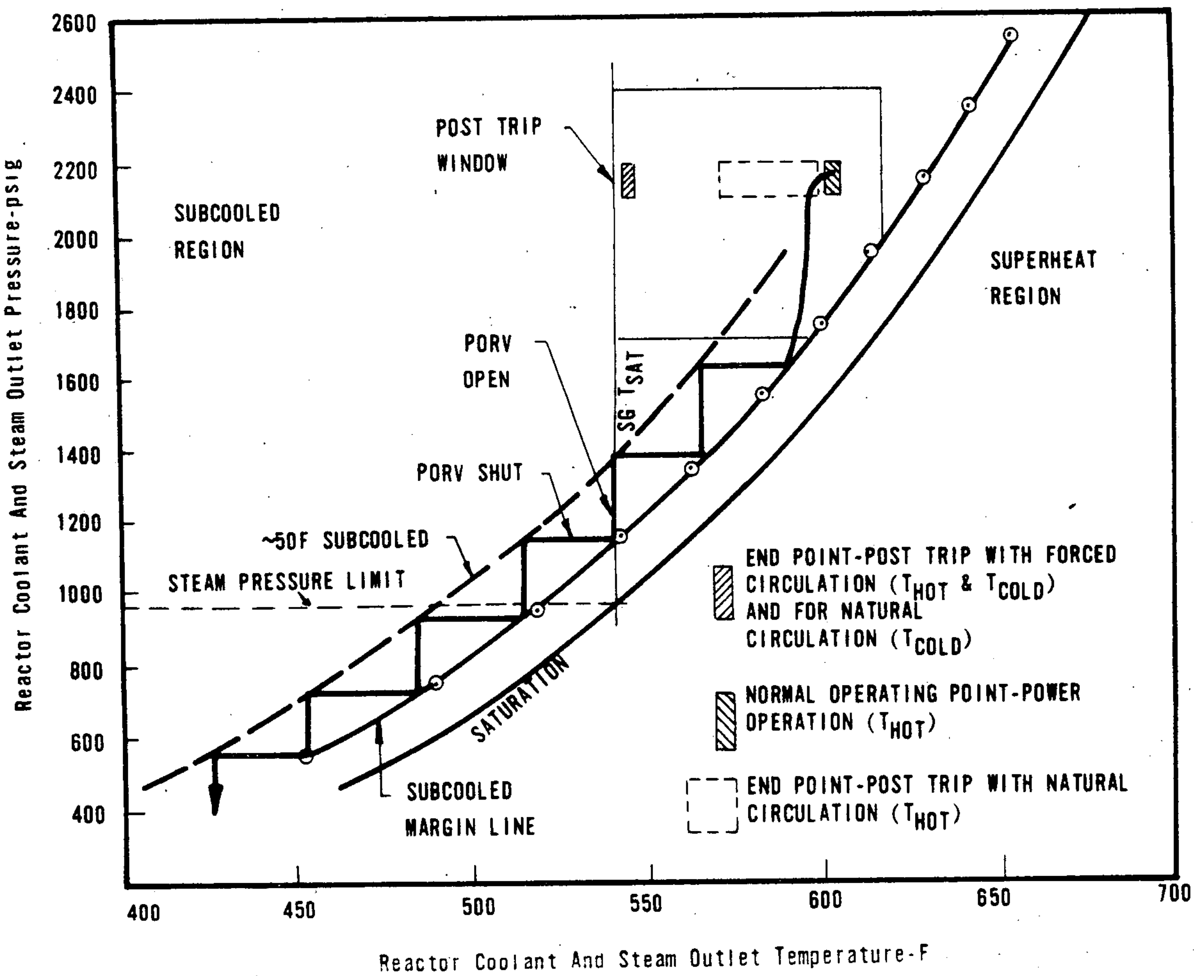
11.0 SPECIAL TOPICS RELATED TO COOLDOWN

11.1 Ways to Depressurizer the RCS

Two ways are available to depressurize the primary system. The first way is to use the pressurizer spray, and the second is to open the PORV. Use of the pressurizer spray is the preferred method since PORV operation can result in a quench tank rupture, abnormal reactor building pressure and temperature, and additional RB cleanup efforts. Pressurizer spray also provides better control and more rapid depressurization as opposed to cyclic PORV operation.

The pressurizer spray is dependent on RC pump operation. Consequently, if the RC pumps have been tripped, two RC pumps should be restarted as soon as the restart criteria are satisfied (See RC pump Restart Criteria in "Best Methods for Equipment Operation."). An early restart of the RC pumps allows the pressurizer spray to be used and also limits the need to add additional feedwater to the affected OTSG to sustain natural circulation. The performance of the pump motor and seal cavity cooling systems should also be periodically monitored to ensure that the pumps can be continuously run until the decay heat removal system is placed into a cooling mode of operation.

In the event the pressurizer spray is off or the RC pumps cannot be restarted, the pressurizer PORV must be used to depressurize the plant. In general, it will not be possible to open the PORV and still maintain pressurizer level and pressure control. Instead, cyclic operation (open-shut) will be necessary. The best way to maintain plant control is to continue the RCS cooldown and depressurize with the PORV in a sawtooth manner between the 50F subcooled line and the subcooling margin line, as shown on the P-T diagram below.



When the PORV is opened, pressurizer conditions should be closely monitored. A drop in pressurizer level should occur because, as pressure drops, some of the water within the pressurizer will flash to steam. To counteract this effect, adjustments in the MU/HPI flow may be necessary to maintain an adequate pressurizer water inventory and to prevent a loss of subcooling margin. Also, monitor the quench tank conditions and if possible, adjust the PORV "open" time to prevent failure of the rupture disc. However, RCS cooldown and depressurization to 500 F and 1000 psia should take precedence over quench tank disc rupture.

11.2 Placing The Decay Heat Removal System into Operation

Once the plant has been cooled down and depressurized, the decay heat removal system (DHRS) is used for final cooldown and depressurization. Special actions must be taken for tube leaks when the decay heat system is actuated to continue SGTR mitigation. Special actions must be taken to place the DHR system into operation during a natural circulation cooldown.

The most important items are (also refer to Appendix F):

1. As a general rule, if the RCS is saturated (which will only occur if several tubes are failed), it may not be possible to place the DHR into operation; (liquid level above the drop line is not known and a loss of decay heat suction would cause pump cavitation). However, if it is definitely known that no LOCA other than a tube leak exists, the decay heat removal system can be placed into operation. The reason is that the leak location is such that liquid will be trapped in the lower parts of the hot leg and the RCP internal "lip" will trap water between the pump and the vessel. When the RCS is saturated and the decay heat system is engaged, the decay heat pumps must be locally monitored for cavitation. The pumps must be shut down within 2-5 minutes after loss of suction to prevent cavitation damages.

2. One decay heat pump should be aligned in the decay heat removal mode; the other in the injection mode from the BWST <u>or</u> the HPI pump should be placed in "piggyback" on the decay heat pump for injection. Injection flow rate needs will determine the choice.

3. Due to the location of the decay heat removal drop line and return line, the system will not adequately cool the higher portions of the reactor coolant loop, especially if the plant has been cooled down on natural

circulation. Cooling of the lower portions of the loop and continued RCS depressurization can permit the reactor coolant flashing in the high parts of the hot leg which will inhibit further depressurization. This effect will be more pronounced if a natural circulation cooldown rather than a forced circulation was used. To avoid this possibility, the plant should be operated as follows:

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<u>Decay heat removal system startup after forced circulation cooldown</u>: (The following options may be available):

- Continue RC pump operation until the hot leg temperature is lower than 212F (One pump in each loop is adequate) or
- Place the decay heat removal system into operation at a higher temperature. After the DHRS is placed into operation, stop the RC pumps, monitor hot leg temperatures for saturation, and control cooldown rate to avoid saturation. If saturation occurs, "bump" a pump. A slight pressure rise may occur and pump "bump" should be avoided if the decay heat system can become overpressurized.

Operator Actions During DHRS Operation:

- Isolate the steam generator with the tube leak at the steam and feed lines. Steam pressure may rise for a short time, but thereafter will drop since the reactor coolant temperature will be lower than the steam generator liquid temperature; lack of heat transfer into the generator will not permit steam pressure to build up.

- Add nitrogen to the isolated steam generator to provide a slight overpressure. When the RCS is completely depressurized, the nitrogen pressure should be about 2-3 psi greater than RCS pressure.
- Do not isolate the remaining generator; it may have to be used for heat removal if the RCS repressurizes and reheats.
- Since the hot leg is at a higher elevation than the steam generator, a small leak rate can occur into the generator after the RCS is depressurized. Liquid can be allowed to accumulate in the isolated generator as long as continued heat removal occurs and reheat and repressurization is avoided. It may be desirable to manually drain liquid accumulation to avoid complete filling (600 inches on the wide range level) if a radioactive waste reservoir is available. Draining should not be performed above the design pressure of the waste systems.

Decay heat removal startup after natural circulation cooldown:

In general, the steps are similar to the forced circulation method, however, it is not likely that the hot leg reactor coolant can be cooled to 212F using the steam generators, consequently hot leg flashing is more likely. To avoid hot leg flashing:

- Just before placing the DHRS into operation "bump" a pump, if possible.

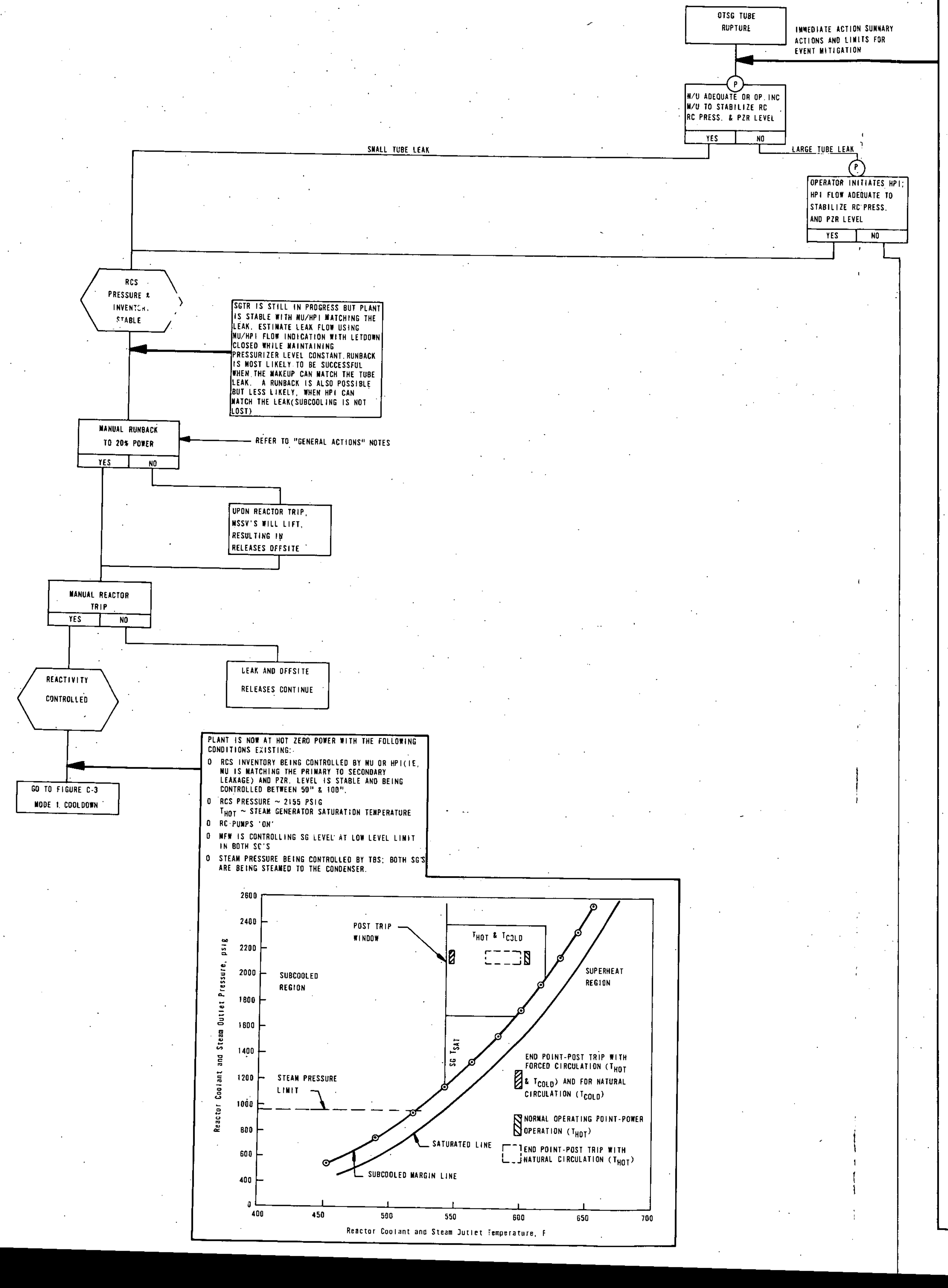
or

- When the decay heat system is placed into operation, control the rate of cooldown and further depressurization to avoid hot leg saturation.

FIGURE C-3

STEAM GENERATOR TUBE RUPTURE LOGIC DIAGRAM

Sheet 1 of 2



NOTES:

IMMEDIATE ACTION SUMMARY

- DIAGNOSE THE EVENT TO BE A TUBE LEAK OR RUPTURE.
- INCREASE MAKEUP OR START HPI AND ISOLATE LETDOWN TO STABILIZE PRESSURIZER LEVEL AND RCS PRESSURE TO AVOID A REACTOR TRIP.
- IDENTIFY THE AFFECTED STEAM GENERATOR USING RADIATION MONITORS.

REFER TO:

- ATOG PART I, SECTION III, D FOR PROCEDURE.
- ATOG PART II, SECTION II, APPENDIX C FOR DISCUSSION.
- ALSO REFER TO ATOG PART II, APPENDIX F, "LOCA".

EVENT RECOGNITION SYMPTOMS (BEFORE REACTOR TRIP)

- RADIATION ALARMS WILL OCCUR ON
  - CONDENSER VACUUM PUMP DISCHARGE MONITOR
  - STEAM LINE RADIATION MONITOR ON AFFECTED SG
- A SGTR IS A LOCA; DEPENDING ON THE LEAK SIZE; POSSIBLE LOCA SYMPTOMS ARE:
  - DECREASING RCS PRESSURE
  - DECREASING PRESSURIZER LEVEL
  - HIGH MAKEUP FLOW
  - LOW MAKEUP TANK LEVEL
- IF THE TUBE LEAK RATE IS VERY LARGE AND A REACTOR TRIP OCCURS UNAVOIDABLY, THE AFFECTED STEAM GENERATOR LIQUID LEVEL WILL BEGIN TO INCREASE.

CORRECTIVE ACTIONS/LIMITS

A. GENERAL ACTIONS FOR ALL TUBE LEAKS:

- BALANCE LEAK FLOW WITH MU/HPI TO MAINTAIN PRESSURIZER LEVEL AND RCS PRESSURE; AVOID REACTOR TRIP WHICH WILL LIFT THE MAIN STEAM SAFETY VALVES AND ALLOW RADIOACTIVE STEAM RELEASE.
- ATTEMPT TO RUN BACK THE REACTOR POWER, AND STEAM AND FEEDWATER SYSTEMS WITHOUT TRIPPING.
- AT ABOUT 20% POWER
  - PLACE TBS'S IN MANUAL AND OPEN
  - UNLOAD STEAM GENERATORS AND TRIP THE TURBINE
  - TRIP REACTOR
  - PLACE TBV'S BACK IN AUTOMATIC
- COOL THE REACTOR COOLANT SYSTEM DOWN AND DEPRESSURIZE TO ABOUT 540°F AND 1000 PSI (BELOW THE OPENING SETPOINT OF THE STEAM LINE SAFETY VALVES).
- ISOLATE FEEDWATER TO THE AFFECTED STEAM GENERATOR, BUT DO NOT ISOLATE THE STEAM LINES (NOTE: THERE ARE SOME EXCEPTIONS TO THIS GENERAL RULE WHICH ARE COVERED NEXT AND IN APPENDIX C "SPECIFIC ACTIONS".
- COMPLETELY COOL DOWN AND DEPRESSURIZE THE REACTOR COOLANT SYSTEM TO COLD SHUTDOWN. DURING THE COOLDOWN THE AMOUNT OF SUBCOOLED MARGIN SHOULD BE HELD TO A MINIMUM. EXCESSIVE SUBCOOLED MARGIN WILL INCREASE THE LEAK RATE.
- ENGAGE THE DECAY HEAT REMOVAL SYSTEM IF THE REACTOR COOLANT IS SUBCOOLED (DO NOT ENGAGE IF SATURATED BECAUSE THE LIQUID LEVEL IN THE RCS MAY NOT BE ABOVE THE DECAY HEAT LETDOWN LINE CONNECTION TO THE REACTOR COOLANT PIPING). INJECTION MUST BE MAINTAINED UNTIL THE PLANT IS COMPLETELY DEPRESSURIZED.

B. SPECIFIC ACTIONS (REFER TO APPENDIX C TEXT FOR DETAIL)

TUBE LEAK ONLY (NO ADDITIONAL SYSTEM OR COMPONENT FAILURES)

SMALL LEAKS (RCP'S ON)
ISOLATE STEAM AND FEEDWATER TO AFFECTED OTSG

SMALL LEAKS (RCP'S OFF)
ISOLATE FEEDWATER TO AFFECTED OTSG

LARGE LEAKS (RCP'S ON OR OFF)
ISOLATE FEEDWATER TO AFFECTED OTSG
(REFER TO APPENDIX C, SECTION 5)

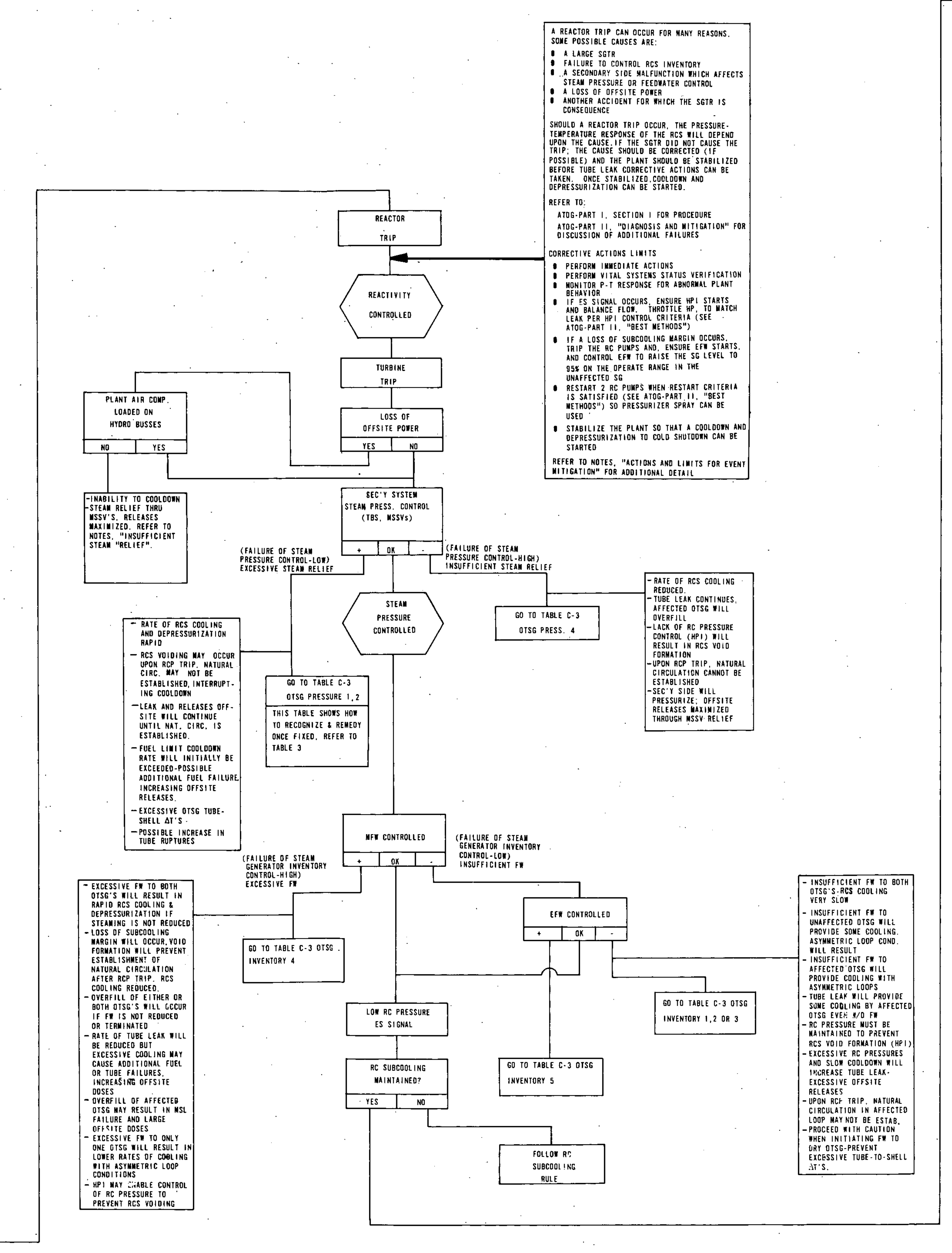
ADDITIONAL FAILURES (REFER TO TABLE C-3 FOR DETAIL)

- REPAIR OR BYPASS FAILURES PRIOR TO COOLDOWN SO THAT TWO GENERATORS ARE AVAILABLE FOR COOLDOWN AND DEPRESSURIZATION BY PORV OR SPRAYS IS AVAILABLE
- SOME FAILURES CANNOT BE REPAIRED:
  - TOTAL LOSS OF FEEDWATER; USE SOLID WATER DEPRESSURIZATION BY THROTTLING HPI
  - LOSS OF F/W TO ONE GENERATOR; COOLDOWN SLOWLY TO AVOID IDLE LOOP REACTOR COOLANT FLASHING
  - LOSS OF STEAM PRESSURE (ONE GENERATOR); FEED SLOWLY TO KEEP NATURAL CIRC AND/OR TUBE TO SHELL LIMITS ACCEPTABLE
  - PRESSURIZER LOCA; SOLID WATER COOLDOWN BY THROTTLING HPI

C. LIMITS

- FUEL DECOMPRESSION LIMITS APPLY TO SMALL LEAKS WITH RCP'S RUNNING
- NORMAL TUBE-TO-SHELL TEMPERATURE LIMITS (60°F) APPLY TO SMALL LEAKS WITH RCP'S RUNNING
- EMERGENCY TUBE-TO-SHELL TEMPERATURE LIMITS (150°F) APPLY TO ALL OTHER CASES EXCEPT:

  EMERGENCY LIMITS MAY BE VIOLATED IF BWST INVENTORY DEPLETION IS EMINENT
- COOLDOWN RATE IS 100°F/HR FOR ALL CASES EXCEPT WHERE TUBE-TO-SHELL TEMPERATURE LIMITS RESTRICT COOLING RATE
- MINIMUM RCS SUBCOOLING MARGIN SHOULD BE MAINTAINED, EXCESSIVE SUBCOOLING SHOULD BE AVOIDED.
- COMPLETELY DEPRESSURIZE RCS TO TERMINATE TRANSIENT. ACHIEVE BEFORE BWST INVENTORY DEPLETION OR POOR QUALITY BACKUP WATER MUST BE INJECTED INTO OTSG BY EFW.



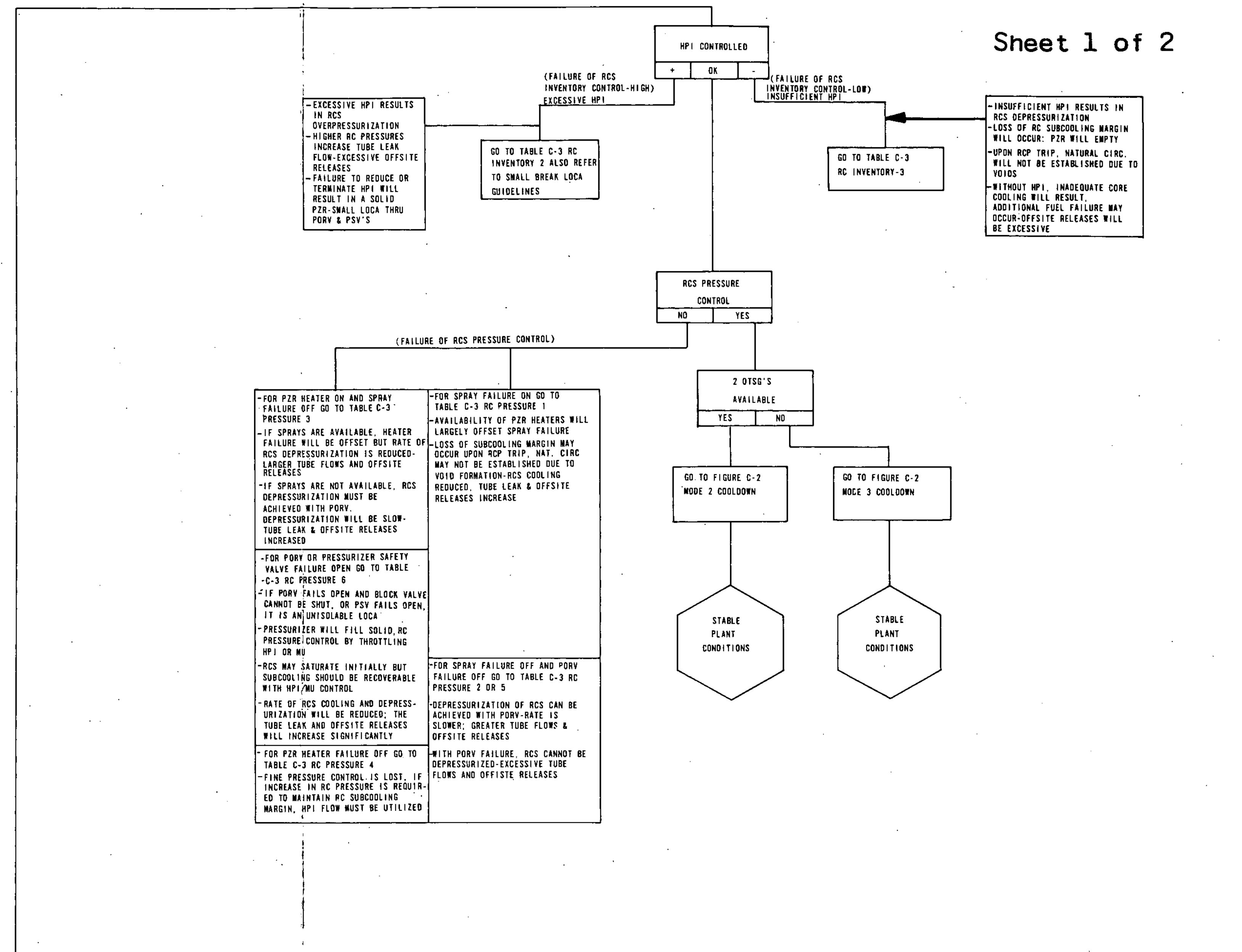


FIGURE C-3

STEAM GENERATOR TUBE RUPTURE LOGIC DIAGRAM

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APPENDIX D

LOSS OF OFFSITE POWER

1.0 GENERAL TRANSIENT DESCRIPTION

Loss of offsite power is a failure to provide power to the plant auxiliaries from an offsite source. It is not an "abnormal" transient or a failure to control the plant unless coincident failures occur.

This transient is initiated by the plant separating from the grid due to a grid upset. The unit auxiliaries are powered by the main generator through the Unit 1 auxiliary transformer during both normal operation and grid separation. Therefore, power is still supplied to all plant auxiliaries and the ICS (by design) begins to run the plant back to 15% power.

Since the turbine was supplying 100% power and "house loads" are only about 4% full power, the turbine protection system must partially close the turbine throttle valves to prevent overspeed. This results in a sudden increase in secondary pressure and temperature and consequently an increase in primary pressure and temperature. The steam produced during the runback will be dumped to the turbine through the turbine throttle valves, to the condenser through the turbine bypass system, and to the atmosphere through the code safety valves. The atmospheric dump valves are manually operated and therefore will not open.

With the present pressurizer electromatic relief valve setpoint (2450 psig) and high pressure trip setpoint (2300 psig), the plant will trip during the runback. The plant auxiliaries will switch automatically from the auxiliary transformer to the startup transformer which now has no power. Therefore, all power is lost to the plant auxiliaries except those components loaded on the 125V DC battery banks and 120 VAC vital bus.

The Hydro Generators start automatically if the external transmission system is lost (or if engineered safeguards actions is required). One hydro generator feeds through the 13.8 kv underground feeder. The other hydro generator feeds to the 230 kv switching station.

Approximately 23 seconds after starting, the Hydro generators will begin to accept loads. After the Hydro loading is complete the operator should start a makeup pump to re-establish makeup and RCP seal injection and start an instrument air compressor to maintain air pressure.

Because the reactor coolant pumps have lost power, natural circulation must be used for decay heat removal. To establish natural circulation the steam driven and both electric driven emergency feedwater pumps are automatically started. Water from both the steam and motor driven EFW pumps will be automatically directed to

the steam generators. This EFW is desirable when establishing natural circulation because it enters at the top of the OTSG and will provide a higher thermal center for heat removal.

However, the injection of EFW from the EFW pumps into the top of the OTSG's will slowly quench or condense the steam. As a result, the OTSG steam pressure will decrease at a rate dependent on the EFW flow rate and the decay heat rate.

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Therefore, EFW should be throttled by the operator to prevent this overcooling of the RCS. Following the EFW throttling guidelines the operator should throttle flow such that:

1. OTSG level is continuously increasing; and
2. Steam pressure does not decrease more than 100 psi below its target value.

The main condenser continues operation after a loss of offsite power (LOOP) because the circulating water flow will continue by automatic valve operations to allow gravity flow to the tail race.

Because of this, the turbine bypass system will dump to the condenser. The steam safeties will reseat in less than a minute after the LOOP and thereafter, the TBS will relieve steam to the condenser as necessary.

As the plant approaches steady state, the OTSG level will reach its setpoint and EFW flow will modulate to maintain this level. Steam pressure and temperature will return to the TBS setpoint. Primary temperature will increase slightly as steam temperature increases. Primary pressure will also increase as makeup flow raises pressurizer level back to its normal setpoint. When secondary pressure has increased to the TBS setpoint and pressurizer level is at its normal setpoint, primary to secondary heat transfer will again be equal and the plant will be at steady state.

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Actual Plant Loss of Offsite

On February 22, 1975, Arkansas Nuclear One, Unit 1 was operating at 100% power when a storm blew down a 500 kv transmission line near Little Rock, Arkansas. This resulted in a loss of offsite power.

The reactor and consequently, the turbine generator tripped due to under voltage to the control rod drive breakers. The reactor did not trip on high RC pressure as would be expected. The control rod drives are normally powered from startup transformer #1 and make a transfer to the auxiliary transformer upon LOOP. During this trip, there apparently was a momentary delay in this transfer and the control rods fell into the core. Buses H-1, H-2, and A-1 transferred to startup transfer #1 which had lost power due to the LOOP. Bus A-2 transferred to startup transformer #2.

Following the reactor trip, the reactor coolant and main feedwater pumps tripped placing the plant in a natural circulation mode. The diesel generator DG1 started automatically and emergency feedwater and makeup flow was initiated in about 30 seconds after the LOOP. After approximately 4-1/2 minutes, bus H1, H2, and A1 were switched to startup transformer #2 which was powered and one RC pump was started in each loop.

Figures D-1 to D-5 show the plant data. Figures D-4 and D-5 show the OTSG full range level and steam pressure versus time. Since OTSG level was slowly decreasing, the EFW flowrate was probably very small. Although the OTSG level is to be raised to 50% on the operate range after a LOOP, the failure to do so in this case did not adversely affect the transient for the first five minutes. However, failure to raise the level for an extended period of time would have led to possible loss of natural circulation.

The slow oscillations in steam pressure are probably due to the steam safety valves lifting and blowing down. Steam pressure leveled off at about 1010 psig which is the setpoint of their MADVs.

Figure D-1 and D-2 show the hot and cold leg temperatures versus time for loops A and B. The initial decrease in RC temperature after trip is followed by a slow rise in the hot leg temperature. The hot leg temperature should be approximately 20-40F higher than

the cold leg when natural circulation is developed. The cold leg temperature will approach the saturation temperature of the steam pressure.

Figure D-3 shows the loop A RC pressure versus time after trip. After an initial decrease due to T_{ave} decrease following a reactor trip, the pressure begins to increase. This is caused by the injection of makeup and the reactor coolant expansion due to the inreasing hot leg temperature.

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Figure D-6 is a P-T diagram of the ANO LOOP of February 27, 1975. This plot is similar to but not exactly the same as Figure D-10. Since the plant tripped on loss of power to the control rod drives, there was no initial spike in pressure.

Figure D-10 shows a temporary increase in both pressure and temperature before OTSG level reaches 50%. This increase is due to the increase in the ΔT required to establish natural circulation. The plant data for the February 22, 1975 LOOP does not show a similar trend. This difference is due to the long time interval used between plant data points.

Figure D-1 LOSS OF OFFSITE POWER - 2/22/75
LOOP A RC TEMPERATURES VS TIME

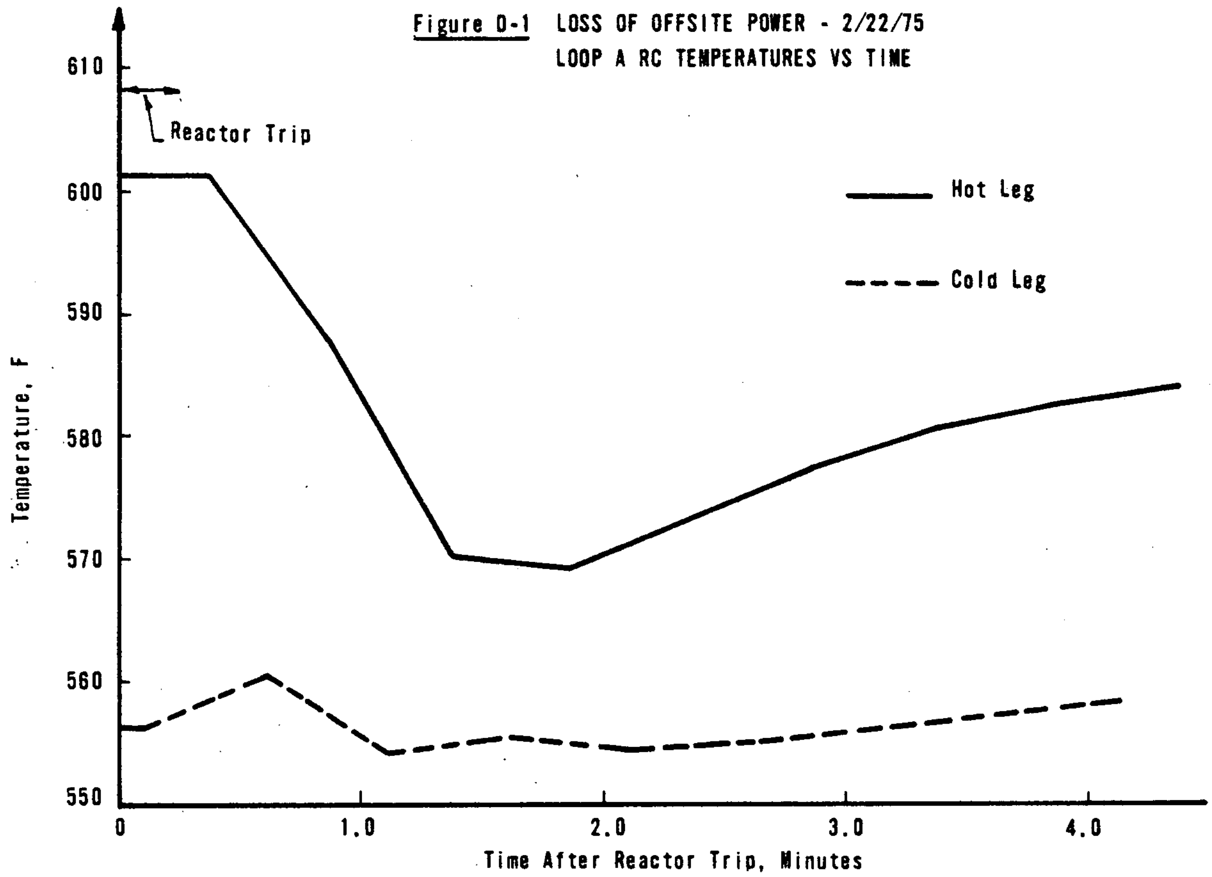


Figure D-2 LOSS OF OFFSITE POWER - 2/22/75
LOOP B RC TEMPERATURES VS TIME

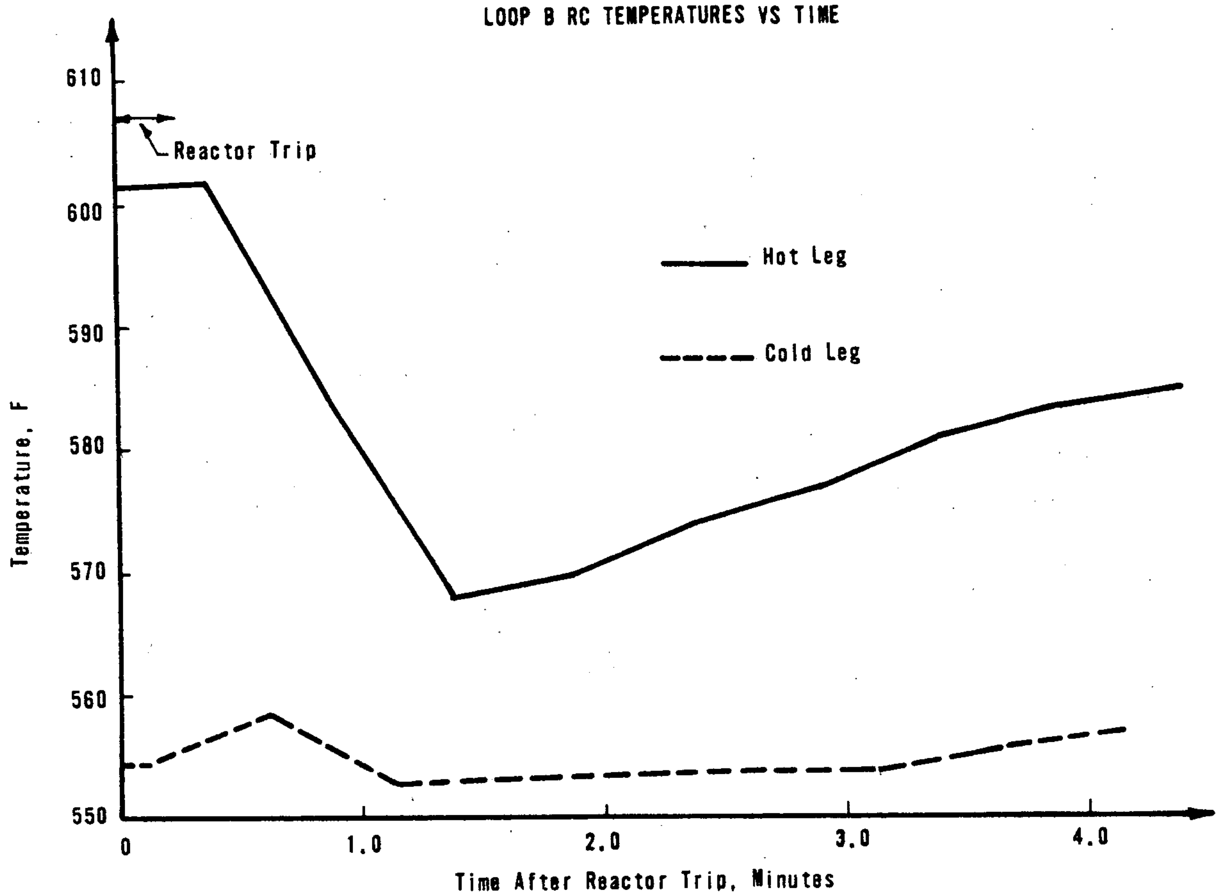


Figure D-3 LOSS OF OFFSITE POWER - 2/22/75
LOOP A PRESSURE VS TIME

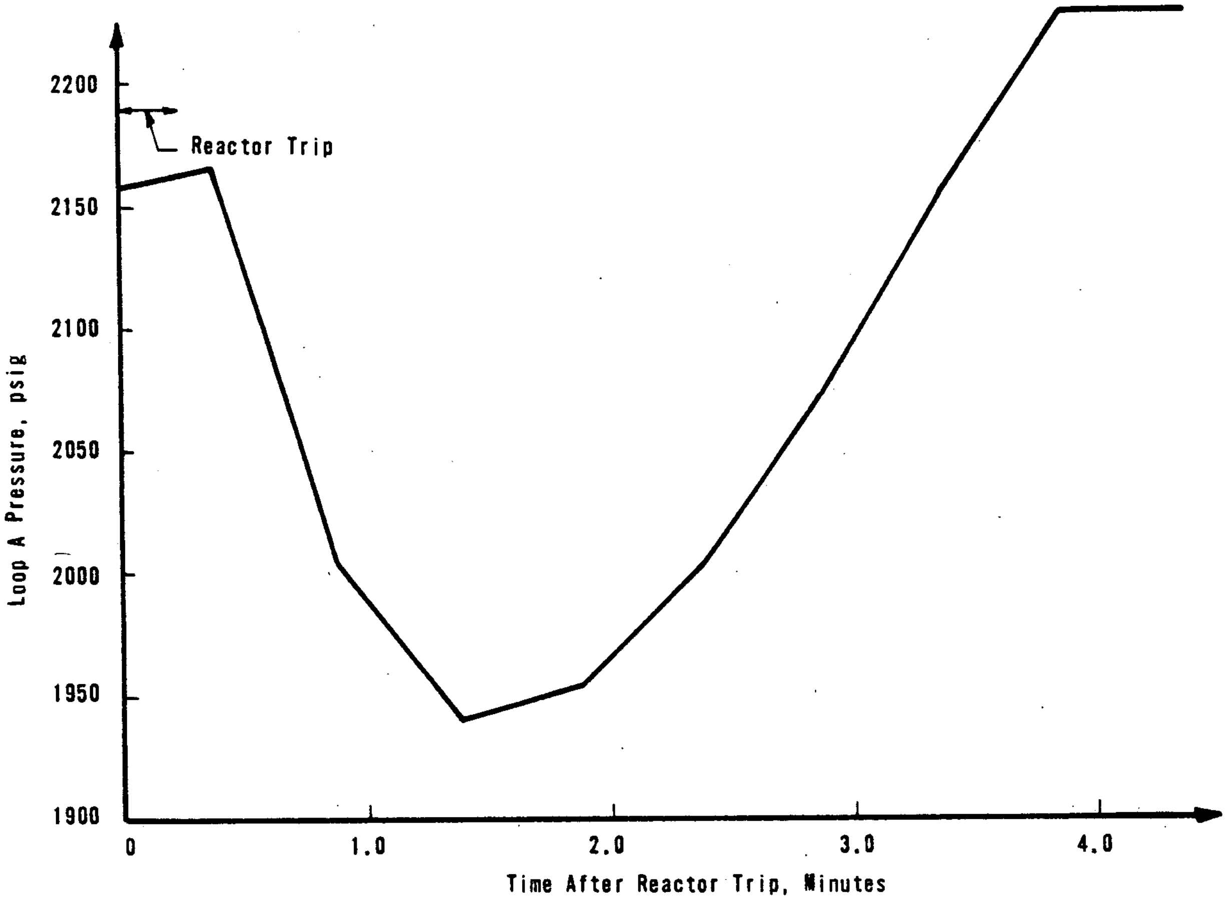


Figure D-4 LOSS OF OFFSITE POWER - 2/22/75
- OTSG FULL RANGE LEVEL VS TIME

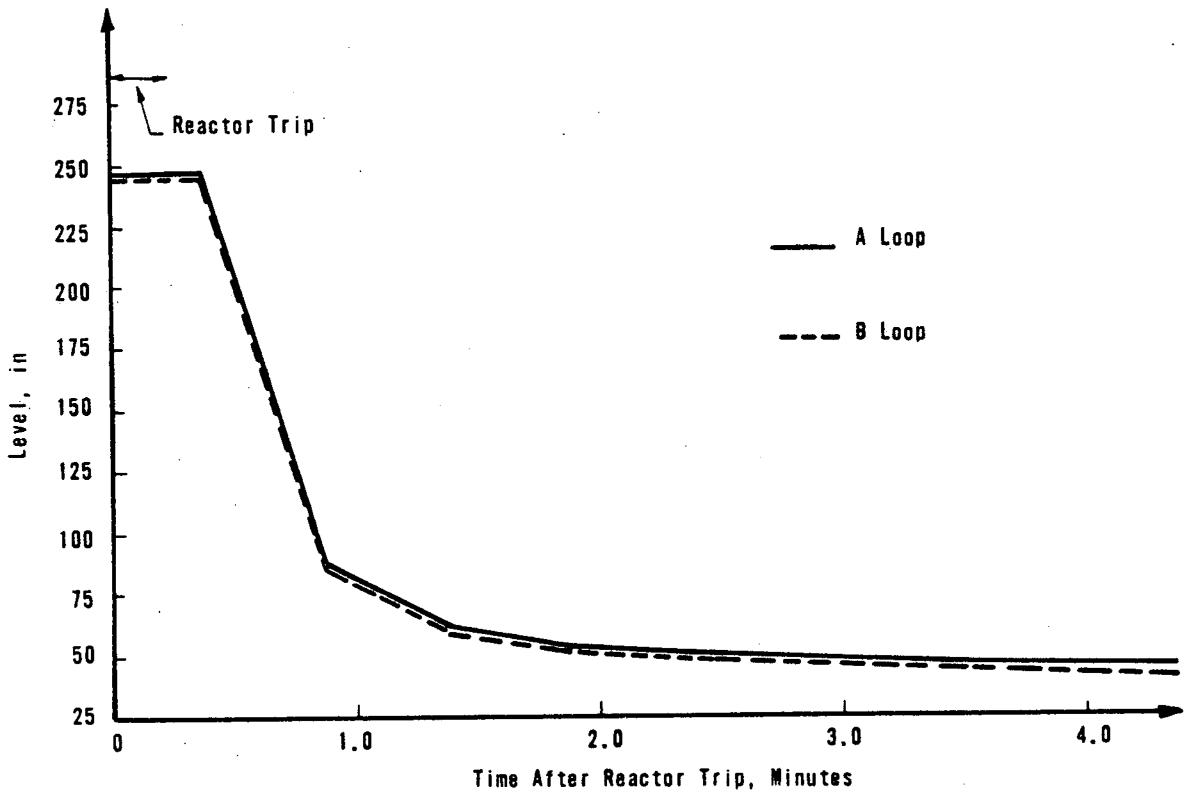


Figure D-5 LOSS OF OFFSITE POWER - 2/22/75
OTSG STEAM PRESSURE VS TIME

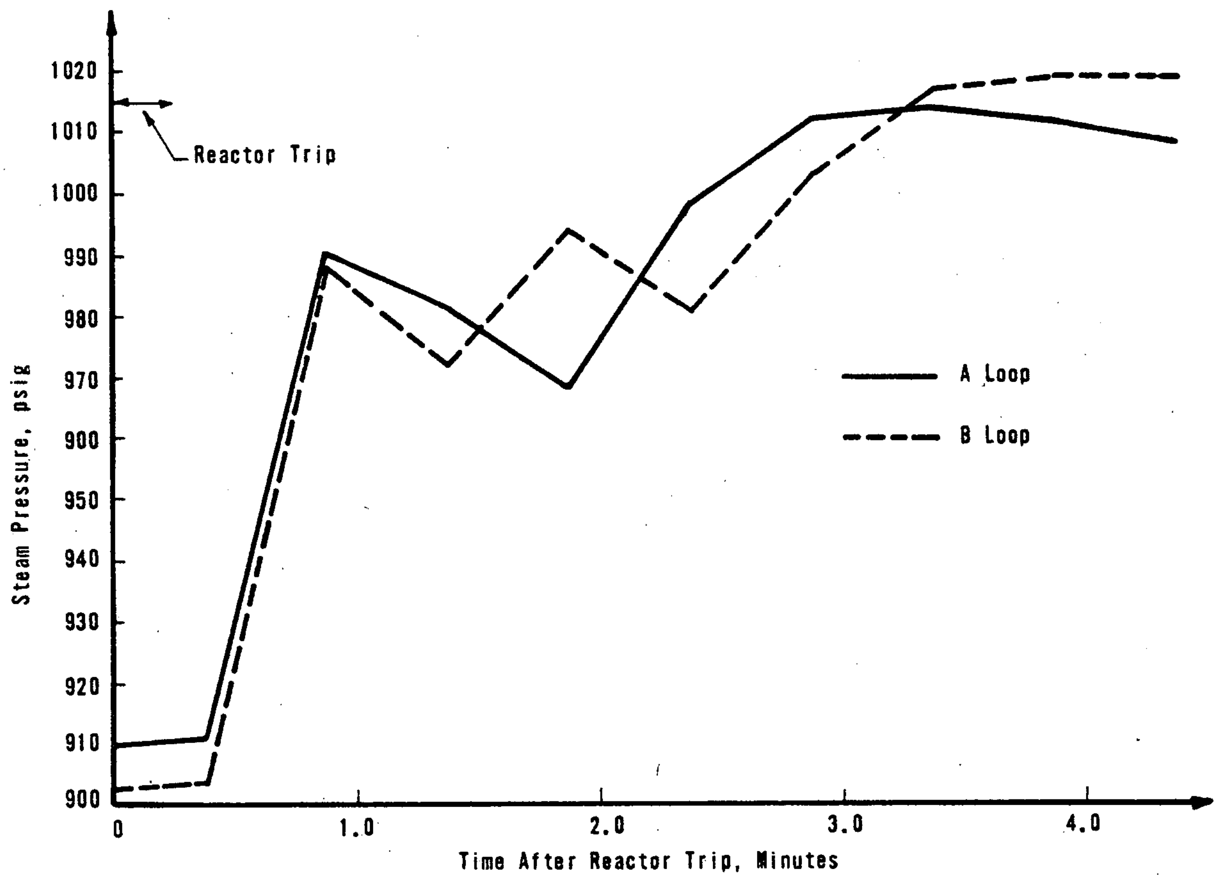
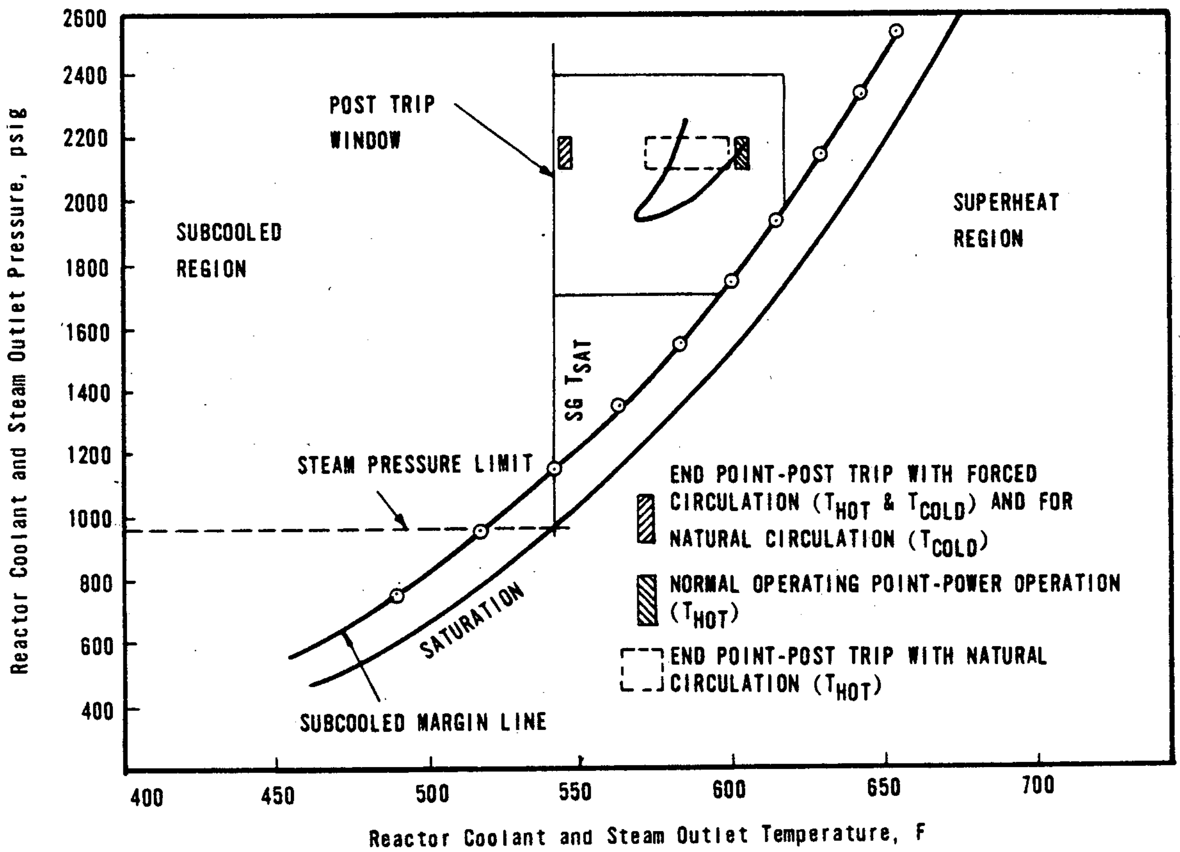


Figure D-6 LOSS OF OFFSITE POWER-2/22/75



2.0 OPERATOR ACTIONS SUMMARY WITH NO OTHER PLANT FAILURES

Major Operator Action

- Verify both Keowee hydro generators have started and are loading. If not start Lee station.
- Verify EFW has started to both OTSG's then throttle flow to both OTSG's such that:
  - a. OTSG level is increasing to 50% on operate range and
  - b. Steam pressure does not fall more than 100 psi below its target value.
- Start a MU pump, establish makeup and RCP seal injection
- Start instrument air compressor

Identifying Symptoms

- A loss of offsite power is a failure to provide power to the plant auxiliairies from the offsite source. It is <u>not</u> a failure to control the plant unless additional failures occur. A typical P-T plot for a loss of offsite power is shown in Figure D-10.
- Other identifying symptoms to help distinguish a LOOP from other transients are:
  - RC and MFW pump trip
  - Under frequency or under voltage on the 230 kv busses
  - Trip of condensate and circulating water pumps
  - Automatic start of hydro generators
  - Trip of instrument air compressors

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Discussion

This section will discuss how actions in accordance with Part I of ATOG will help the operator recognize a loss of offsite power and obtain a stable shutdown after it occurs.

As was pointed out, most of the plant response is automatic. The operator must take four actions: 1) assure restart of at least one makeup pump, 2) throttle EFW, 3) establish seal injection, and 4) start an instrument air compressor. The first two actions are intended to prevent draining of the pressurizer which would result in a possible loss of subcooling and a consequent loss of natural circulation cooling. Restoring seal injection is also required for reactor coolant pump protection. Starting the air compressor is needed to supply air to numerous control valves and instrumentation.

An operator using the guidelines for the event described would first perform the Immediate Actions of Part I, section I; then go to Part II, Vital System Status Verification. The first verification which might require action is to verify that letdown is flowing through the block orifice only. If the bypass valve is open it should be closed. Step 5.0 requires the operator to verify that feedwater has runback. However, both MFW pumps will trip after a LOOP and EFW will be automatically initiated. Therefore, the operator does not need to take any remedial action.

Step 8.0 is the crucial step for diagnosis of a LOOP. The 6900V and 4160V busses will auto transfer to the startup transformer after a turbine generator trip. If voltage is not regained on all buses after this transfer, a LOOP has occurred. This step requires verification of voltage on all busses; the voltage will not be on all busses, therefore, the operator will take the required remedial action.

The remedial actions to be taken for Step 8 are:

a. Start or verify the auto start and loading of the hydro generators (automatic); this should occur within 23 seconeds after turbine generator trip.
b. Verify that both the motor driven EFW pumps and the steam driven EFW pump started then raise OTSG operate level to 50% (automatic) while throttling EFW (manual) as needed to keep steam pressure within 100 psi of its target value. The LOOP will cause the condensate pumps to trip which will cause both MFW pumps to trip on low suction pressure. Tripping both MFW pumps will auto initiate the EFW system.
c. Restart at least one makeup pump and provide seal injection to the RCP's.

The instrument air compressor will not be operating since it is not powered from the standby bus. The operator must get power to the compressor from the standby bus. The air compressor supplies air to many major air operated controls and instrumentation.

3.0 LOSS OF OFFSITE AC POWER WITH OTHER PLANT FAILURES

Introduction

The previous section described loss of offsite power with no coincident failures. This section will show what symptoms indicate other equipment has failed and what steps the operator should take to correct the heat transfer from the core to the steam generators.

In addition to a possible failure of one of the major control functions discussed throughout these guidelines, both hydro generators can fail to start after a LOOP. While this is not one of these control functions, it can result in loss of RC Inventory and Pressure Control.

Branch Discussion

The loss of Offsite Power Logic Diagram (Figure D-9) has separate failure branches for loss of all power except batteries, loss of reactor inventory control (high and low), loss of secondary inventory control (high and low), loss of primary pressure control, and loss of secondary pressure control. This section will discuss each of these additional failure branches and illustrate how operator actions in accordance with procedures in Part I will restore proper control of the parameter in question.

Figure D-7 and D-8 show typical primary and secondary pressure and pressurizer and OTSG full range level trends for each of the failure branches.

Loss of Reactor Inventory Control (Low)

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A loss of reactor inventory (low) exists whenever makeup or HPI is insufficient to overcome a primary leak rate. The most probable primary leak is a failure to stop letdown, except through letdown orifice, after a reactor trip.

The makeup pump busses are automatically loaded onto the diesel generators. However, the operator will need to manually start the pumps. If he does not do so, the pressurizer will drain unless letdown is completely isolated. The instructions of part one only require isolating the block orifice bypass valve. Draining the pressurizer is especially likely if excessive EFW occurs and overcools the primary.

In either of the above cases (excessive letdown or inadequate makeup), the symptoms will be the same, decreasing pressurizer level and pressure with eventual draining of the pressurizer. The primary pressure will then quickly approach the saturation pressure of the primary temperature. As long as primary temperature has been maintained below 600F (which has a saturation pressure of 1500 psig), ES will actuate HPI.

The symptoms of uncontrolled reactor inventory (low) will be very similar to those of uncontrolled secondary inventory (high). The distinguishing characteristics of the two events is that primary temperature as well as pressure will decrease for the uncontrolled secondary inventory (high). As shown in Figure D-9, the P-T diagrams are considerably different. Figure D-8 shows typical parameter trends for this case.

The appropriate actions for an operator to regain control of primary inventory are as follows:

- Isolate letdown
- If pressurizer level continues to decrease, manually initiate HPI
- If the uncontrolled letdown occurs in conjunction with an excessive cooldown, eliminate cause of cooldown. (i.e., throttle feedwater, isolate the stuck open steam line valve, etc.)

Following the remedial action of step II.8 of Part I will correct the problem. If that is missed, subcooling will be lost and the operator can use III A to regain inventory control.

<u>Loss or Reactor Inventory Control (High)</u>

A loss of reactor inventory control (high) can result from excessive makeup or HPI. These can happen by:

1. Pressurizer level is controlled by varying makeup using the makeup control valve, therefore excessive makeup can occur by the failure of the makeup control valve (in an open position) or failure of inputs into the valve (i.e., erroneous pressurizer level measurements).
2. If HPI is initiated, the operator is allowed to throttle the flow when subcooling margin is regained. If he does not, excessive HPI can occur.

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An operator using ATOG, Part I, section II only, would not be able to directly diagnose excessive makeup. However, a slowly increasing pressurizer level will indicate the problem. It should be emphasized that excessive makeup is an extremely slow transient and there should be ample time for the operator to recognize and correct the problem. Several indications, such as high makeup flow and pressurizer level alarms, are available to assist him.

If HPI has been started manually or automatically due to loss of subcooling margin, Section III A of Part I will direct the operator to throttle the HPI once subcooling margin has been regained.

Figure D-7 shows typical parameter trends for loss of reactor inventory control (high).

The actions the operator can take to regain control of the reactor inventory are as follows:

1. Throttle or stop makeup using inline valves.
2. Increase letdown to maximum by opening both the letdown cooler isolation valve and bypass valve.
3. Trip the makeup pumps.

   Caution: The makeup pumps should only be tripped after all other actions have failed to stop the excessive makeup. Tripping the makeup pumps makes RCP seal injection unavailable.

Loss of Secondary Inventory Control (Low)

A loss of secondary inventory control (low) exists whenever there is too little EFW being injected into the steam generators to remove the decay heat output of the core. This can result from such things as:

1. The steam and motor driven EFW pumps have both failed to start.
2. EFW isolation valves have failed to open if closed.

The symptoms will be typical of a loss of primary to secondary heat transfer. OTSG level will decrease with eventual dry out of both OTSGs. RC temperature will increase along with a rise in pressurizer level and pressure. (See Figure D-8.) Some RC flow may continue even after the OTSGs are dry (i.e., not removing any heat). The increasing hot leg temperature maintains a density difference between cold and hot legs. However, this is not stable natural circulation nor does this flow adequately cool the core for long periods of time.

At Step 11.0 of Part I, Section II, the operator will see that primary to secondary heat transfer has been lost. Step 12.0 will direct him to follow Section III B for loss of heat transfer. EFW will be regained or the plant will be placed into HPI cooling and brought to a stable shutdown.

Loss of Secondary Inventory (High)

A loss of secondary inventory (high) exists whenever significantly more emergency feedwater is being injected into one or both steam generators than is required to remove the existing decay heat. The symptoms and corrective actions for excessive EFW are discussed in Appendix A and will not be repeated here. However, it should be emphasized that draining of the pressurizer due to overcooling may result in void formation in the primary. This may lead to an accumulation of steam at the top of each hot leg and a stopping of natural circulation cooling. Chapter D of Part II discusses methods of recovery from this condition.

Loss of Primary Pressure Control (Low)

During a LOOP, a loss of primary pressure control (low) can be the result of:

(1) Power being lost to all presurizer heaters.

(2) The pressurizer draining.

All pressurizer heaters are powered after a LOOP by the hydro generators. These heaters will compensate for ambient heat losses. If all heaters are lost, the steam bubble in the pressurizer will cool and condense. Eventually the bubble will collapse, the pressurizer will fill and primary pressure control will be lost.

An operator using Part I, Section II, will not diagnose the loss of heaters until subcooling margin is lost. However, this is a very slow process and the operator should have ample time to diagnose the event using the P-T display. The required operator actions are to regain the pressurizer heaters or increase makeup to compensate for the collapsing steam bubble.

The pressurizer may drain on an overcooling event (excessive EFW or stuck open steam line valve). If it does, control of primary pressure will be lost. RCS pressure will then approach the saturation pressure of the primary temperature. As long as the primary temperature is less than 600F (saturation pressure equals 1500 psig), ES will actuate and HPI cooling will begin.

An operator using Part I, Section II, will diagnose the overcooling at Step 12.0. Section III C will then instruct him on how to regain subcooling margin (if it was lost) and how to take steps to stop the overcooling and terminate the transient.

Loss of Steam Pressure Control (Low)

A loss of steam pressure control exists whenever one or both steam generators undergo a pressure reduction significantly below the safety valve or TBS reseating set point. This is an overcooling transient and will look similar (on the P-T curve) to the small steam line break transient discussed in Appendix E. There is nothing unique about loss of this control function coupled with a loss of offsite power, therefore, it should be identified and treated as discussed in Appendix E.

Loss of All Power Except Batteries

A loss of all power (or station blackout) exists when both hydro generators fail to start after a LOOP. Thereafter, until the Lee station is on line or one of the Keowee generators is started, the only power available to the plant auxiliaries are the 125V DC battery banks. This results is a loss of power to the following major components in addition to those lost due to the LOOP.

- All EFW isolation valves (however the valves are normally open)
- The motor driven EFW pump
- All makeup pumps
- Presurizer heaters
- Instrument air compressors

This event is one of the most serious that can happen to a pressurized water reactor. The main feedwater pumps have tripped placing the plant into a loss of feedwater transient. The plant relies on only the steam driven EFW pump for the injection of EFW into the steam generator for natural circulation. The HPI pumps are not powered and thus HPI cooling cannot be initiated. However, the operator must establish cooling water flow to the steam driven turbine within 12 minutes.

This situation cannot be maintained indefinitely because of the lack of primary inventory control and a finite supply of good quality secondary water. Therefore, the operator should get at least one hydro generator running or lineup power via the Lee station 100 kv feeder so that EFW and makeup can be remotely initiated and controlled.

Since the pressurizer heaters and makeup pumps have lost power, the operator has no control of primary pressure or inventory. Any voids that form in the primary will tend to accumulate in the hot leg and obstruct natural circulation. Therefore, the plant may windup in the reflux boiling mode. The operator should monitor natural circulation and, if necessary, raise the steam generator levels to 95% on the operate range.

An operator using the guidelines for a LOOP with no hydro generators would first perform the immediate actions of Part I, Section I and

verify vital system status in Section II. At Step 8.0 he will be directed to start the Hydro generators and initiate emergency feedwater. These are the two most important steps he can take to regain control of the plant. In the long run, offsite power must be regained or a hydro generator or the Lee station started before primary inventory control is lost.

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Figure D-9 LOSS OF OFF-SITE POWER DIAGRAM

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APPENDIX E

SMALL STEAM LEAK

1.0 GENERAL TRANSIENT DESCRIPTION

A small steam leak is a failure to control secondary pressure. It is an overcooling transient that results from too much primary to secondary heat transfer.

A small steam leak is defined as loss of steam up to about 25% of full flow. The largest leak assumed is that due to equipment failure (failed open turbine bypass valves). Larger steam leaks could exist only with piping failures; these are not as likely as component failures. However, the basic principles and sequence of events to be discussed also apply to larger breaks.

The steam leak causes a power mismatch between the heat source (the core) and the heat sink (the steam generators) and will cool the reactor coolant down. The ICS will pull rods to increase core power and maintain T_{av}. If initially the plant is operating at much less than full power, the plant will stabilize with reactor power greater than megawatt demand by an amount equal to the size of the steam leak. Depending on the leak location and size the operator may be able to locate and isolate the leak without tripping. For example,

if the leak results from a stuck open TBS valve, the mass flow through the leak goes to the condenser and the plant could operate indefinitely. The only indication will be the reactor-turbine power mismatch. An atmospheric leak will require makeup to prevent the hotwell from draining. Hotwell level and condensate level as well as visual inspection will indicate the leak. If, however, the leak is inside containment, and ES signal on high RB pressure will occur. ES actuation will happen in about ten minutes for a leak equivalent to 1% full flow and in about twenty seconds for a leak equivalent to 25% full flow.

If the plant is initially near full power, the steam leak may result in a reactor trip on high flux. Small leaks will not cause a reactor trip (up to about 3% flow) if the reactor is at high power. After trip the RCS will continue to cool down and depressurize due to the leak. The rate of cooldown will depend on prior power history (decay heat level) and the size of the leak.

Small steam leaks do not result in rapid overcooling and the reactor coolant temperature drop will not be severe. If no additional failures occur and the leak can be isolated, the plant may recover from a leak as large as 25% automatically with only a temporary loss of subcooled margin. HPI will be actuated by ES and will recover RCS pressure, although temporary drainage of the pressurizer may occur.

After the leak is isolated, the operator would have to take manual control of HPI and the TBV's to limit the RCS refill by limiting the reactor coolant reheat and swell.

The following transient example is based on a leak equivalent to 25% full flow and occurs with maximum decay heat. A larger leak or lower decay heat levels would result in a faster cooldown rate and could cause RCS saturation. As an example, refer to Figure 16 in Part II, "Diagnosis and Mitigation", showing the expected response to a larger steam leak.

If the leak is unisolable the opertor must isolate FW to the broken SG and allow it to boil dry. In addition, other failures may occur which will require operator action. These are discussed in Section 3.0 of this appendix. The operator should become familiar with the "Over-cooling Diagnnosis Chart", in Part II, "Diagnosis and Mitigation".

The P-T curve and sequence of events shown in Figure E-1 depict a typical small steam leak transient (stuck open TBS valves) that is terminated by the operator.

2.0 OPERATOR ACTIONS SUMMARY

Immediate Actions

- Start HPI if pressurizer level is less than 80" and RCS pressure is decreasing.
- If the leaking generator is obvious it should be isolated; but if it is not obvious the next step will help identify it.
- Stop all feedwater to both SG's; watch levels; the generator with the fastest level decrease is leaking; isolate it and restart feedwater to the generator(s) which repressurize.
- Close TBV's.

Follow remainder of Part I, Section III.C.

Identifying Symptoms

Small steam leak is an overcooling transient as shown below:

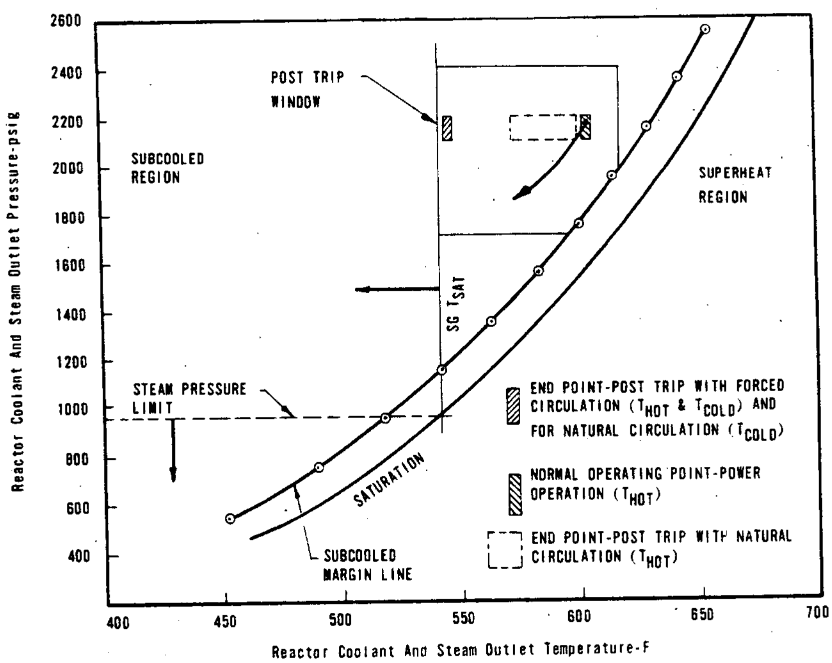

FIGURE E-2

Other key parameters to distinguish small steam leaks from other overcooling transients are:

- Decreasing level and pressure in one or both SG's without feedwater addition* (steam pressure will be lower than the 960 psig steam pressure limit)
- Near normal or low SG levels with feedwater addition

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*NOTE: Excess MFW can cause a slightly reduced steam pressure in the <u>unaffected</u> SG or slightly reduced pressure in <u>affected</u> SG if flow is through the upper nozzles; excess EFW can cause a large reduction of steam pressure in the <u>affected</u> SG; however, in both cases the SG with reduced pressure should have steady or increasing level indicating that inventory is not being lost.

This section will present an example of a small steam leak and discuss how operator actions in accordance with Part I will terminate the transient and provide recovery to stable shutdown conditions. The assumed transient will be the same shown in the previous section, i.e., failed open TBS valves from 100% FP.

A failure of all TBS valves in the open position results in approximately 25% excess steam flow. The reactor will trip on high flux within about eighteen seconds, the turbine will trip, and the ICS

will run back MFW. After the operator performs the immediate actions in Part I, Section I, he will verify vital system status in accordance with Section II. Steam pressure in both steam generators will decrease fairly quickly (∿ 70 seconds) below the normal post-trip pressure boundary of 960 psig. In step 12.0 of Section II the operator should note that primary to secondary heat transfer is excessive by observing lower than normal steam pressures and the overcooling trend indicated by T_h response on the P-T curve. (Refer to the P-T curve, Figure E-2.) The procedure directs the operator to Section III.C.

Step 1.0 in III.C requires the operator to manually start HPI if pressurizer level is below 80" and RCS pressure is decreasing. For this transient pressurizer level will go below 80" within about 40 seconds (see Figure E-3, "Time Relationship of Key Parameters") after trip, thus the operator will initiate HPI. Full HPI flow will not compensate for the reactor coolant contraction rate for this size leak; therefore pressurizer refill and RCS repressurization will not occur until the steam leak is isolated. Therefore, even with rapid isolation by the operator, a temporary loss of indicated pressurizer level may be expected. Note that the graphs shown in Figure E-3 represent automatic plant response, i.e., HPI is not started manually at 80". Reactor building pressure and temperature will be normal since the leak is through the TBS valves; therefore the operator will proceed to Step 5.0.

Steps 5.0, 5.2, 5.3, 5.4, 5.5, and 5.6 of III.C effectively isolate the steam generators from most failures that would cause overcooling. Step 5.2 requires the operator to place the TBV's in manual and close them. Successful completion of this step will terminate the overcooling transient in this case by isolating the leak. If the failure of the TBV's was such that it prevented manual closure (unlikely), the operator would recognize the failure and realize the cause for overcooling. Since the TBV's are failed open on both SG's he would have to close both TBS isolation valves to isolate the leak. If the operator can perform these actions within 90 seconds, subcooling margin will not be lost, therefore the RC pumps will remain on and MFW will be establishing the low level setpoint. However, if subcooled margin is lost (as shown in Figure E-1), it will be quickly regained once the steam leak is isolated.

Once the steam leak and overcooling has been terminated, the operator must perform a few actions to regain RCS inventory and pressure control. At this stage in the transient he still has HPI flow and the RCS is slowly reheating and will continue to reheat, without operator actions, until steam pressure increases to the MSRV setpoint. The actions he should take are given in the procedure and are as follows:

- Maintain RCS temperature at the present value by regaining control of the TBV's. If the TBV's had successfully closed under remote manual control then the operator can control

RCS temperature with manual TBV control. If, however, closure of the TBS isolation valves was required, he must correct the problem with individual TBV control before reopening the isolation valves. Since the HPI flow has increased the RCS inventory, allowing the RCS to reheat to the MSRV lift setpoints would result in a much higher pressurizer level and possibly a full pressurizer.

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- Throttle HPI when the subcooling margin is restored by using one HPI pump (preferably the normal makeup pump) and one injection line (preferably the normal makeup nozzle with the thermal sleeve). If the leak was isolated very rapidly and subcooled margin was not lost the operator could have throttled HPI as soon as RCS pressure began to increase.
- When pressurizer level returns on-scale low and is increasing (and the RCS is above the subcooled margin) the operator should terminate HPI and realign for normal makeup/letdown operation.

In the previous example, the operator followed the procedure in Part I and terminated the transient in a timely manner. For this particular example, the steam leak was isolable and steam pressure was quickly restored.

The transient of most concern for the operator is an unisolable leak in one SG upstream of extraction line isolation valves.

Turbine trip would ensure the leak is isolated from the good steam generator. However, continued feeding and blowdown of the broken SG would continue the overcooling (refer to Figure 16B in Part II). In this case the operator must take action to isolate feedwater to the broken SG and allow it to boil dry. He would then return to stable shutdown conditions by controlling DH removal with the good SG.

The procedure in Part I will result in isolation of the broken SG and restoration of stable shutdown conditions, regardless of whether the leak is inside or outside the RB. For example, if we assume the leak is inside the RB, then RB pressure and temperature would be increasing. Step 2.0 of III.C will direct the operator to Step 4.0. The operator will stop all feedwater to both SG's (4.0), close the TBV's (4.1), and determine which SG is broken by observing SG levels and pressures (4.2). He will then isolate that SG and allow it to boil dry and restart feedwater for DH removal with the good SG. FW should be stopped by closing the control valves; the pumps should not be stopped. Stopping feedwater is not dangerous as long as it is restarted before both steam generators dry out. When the broken SG is identified, then the motor-driven EFW pump for that SG, if in operation, may be stopped. If the leak was unisolable but outside of the RB, the operator would go to Step 5.0 of III.C (RB pressure and temperature normal) and would isolate the broken SG in much the same manner. The primary difference is that this section of the procedure also addresses the possibility of excessive feedwater as the cause of overcooling.

If the steam leak is inside containment, it is likely an ES trip on high RB pressure (4 psig) will occur. The trip will cause isolation of LPSW for the RC pumps. A loss of LPSW for more than ten minutes requires tripping of the RC pumps (refer to RCP control in Part II, Section I.E, "Best Methods for Equipment Operation"). It is desirable, for this transient, to maintain forced circulation of primary coolant as long as the RCS is above the subcooled margin. Therefore, following the ES trip on RB pressure (within ten minutes), the operator should take manual control and restore LPSW flow. If, for some reason, seal injection flow is lost along with LPSW, the operator must restore seal injection or LPSW within sixty seconds or trip the RC pumps.

3.0 SMALL STEAM LEAK WITH OTHER PLANT FAILURES

Introduction

The previous section described small steam leaks in general, but did not discuss other failures that might also happen at the same time. This section will show what symptoms to look for when other equipment fails and will show what steps the operator should take to restore the heat transfer from the core to the steam generators.

Figure E-4, "Small Steam Leak Logic Diagram", condenses the event tree into the main transient path and branches for the major failures that can occur in addition to the main transient. The P-T diagrams are shown in Figure E-4. The effects on other parameters is shown in Figure E-3, "Time Relationship of Key Parameters".

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The event that was chosen for simulation starts with the reactor at power. A small steam leak occurs that may or may not result in a reactor trip on high flux. If a trip does not occur, the plant will stabilize with a power mismatch with the possible conditions given in Note 1 on Figure E-4. The main path assumes a reactor trip does occur. All the data that is shown starts from the time of reactor trip. The failures are shown to occur when a component has to operate (for example, after a reactor trip the ICS shown automatically run back MFW; the failure shown is that MFW does not run back).

It must be remembered that all steam leaks will not look exactly like the examples used. The primary purpose of these examples is to promote operator familiarity with the symptoms and effects of steam leaks.

Branch Discussion

Figure E-4 has separate failure branches for loss of reactor inventory control (high and low), loss of secondary inventory control (high and low), and loss of secondary pressure control. (Minor failures, such as loss of pressurizer heaters, have relatively little impact on the overall transient. The operator has time to diagnose and correct such failures; therefore they are not shown.) Another valuable tool to facilitate operator recognition and identification of overcooling transients is Figure 22, "Overcooling Diagnosis Chart", in Part II, Section I.C, "Diagnosis and Mitigation". The operator should become familiar with this chart.

Figure E-3 is provided to show key distinguishing parameters for small steam leaks that have a time dependency important to the operator in identifying both the type and severity of transient. The parameter plots show typical responses to a steam leak equivalent to 25% full flow with automatic HPI actuation. Arrows, where used, show the effect of other failures and of operator actions on the time relationship.

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Loss of Reactor Inventory Control (High)

A loss of reactor inventory control (high) exists whenever makeup or HPI flows are excessive causing the pressurizer to fill and overpressurizing the RCS for the existing plant conditions. Steam leaks in the range of approximately 25% of full flow, even assuming maximum decay heat, will cause overcooling and RCS contraction rates greater than the full capacity of the HPI system. Therefore, pressurizer refill and RCS repressurization should not begin to occur until the steam leak is isolated (refer to Figure E-1 and sequence of events in Section 1.0 of this appendix). Larger steam leaks or lower decay heat levels would result in a faster cooldown rate and could cause RCS saturation even with full HPI flow (refer to Figure 16 in Part II, "Diagnosis and Mitigation", to see the expected response to a larger steam leak). However, for smaller steam leaks and when the overcooling has been terminated, full HPI flow will overwhelm the coolant shrinkage and result in a rapid increase in pressurizer level and RCS pressure. Rapid operator response may be required to prevent a solid pressurizer and RCS overpressurization.

The operator actions required to restore proper RCS inventory and pressure control are identical to those discussed in Appendix A under "Loss of Reactor Inventory Control (High)". To accomplish this objective, the operator must be able to control TBV's to limit reheating of the RCS.

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Loss of Reactor Inventory Control (Low)

A loss of reactor inventory control (low) exists whenever makeup or HPI flow is insufficent to overcome the primary leak rate or (as in this case) the coolant contraction rate, resulting in drainage of the pressurizer.

Again, as in Appendix A, too little makeup or HPI flow is not a major concern for this particular transient. If the overcooling is terminated before the pressurizer empties, the RCS will reheat and the resultant swell will restore pressurizer level. If the overcooling continues, ES will actuate and HPI will automatically initiate.

Following the actions specified in III.A of Part I will restore primary system inventory control and subcooled margin.

Loss of Secondary Inventory Control (High)

A loss of secondary inventory control (high) exists whenever significantly more feedwater (main or emergency) is being injected into one or both steam generators than is required by existing

plant conditions. It is an overcooling transient and response on the P-T curve is very similar to that of a small steam leak. Excessive main feedwater can be distinguished from small steam leaks by high MFW flowrates and high SG levels. Large feedwater mismatches can cause more severe effects on the RCS than small steam leaks and must be corrected much faster. A detailed discussion of excessive main feedwater is contained in Appendix A.

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Excessive main feedwater was defined in Appendix A as basically supplying more feedwater than could be boiled off to make superheated steam. The definition for excessive emergency feedwater (or excessive MFW after reactor trip) must differ slightly in that 1) the steam generator is at saturation conditions and 2) more importantly, whenever the natural circulation setpoint is in effect EFW (or MFW) <u>must</u> provide more flow than can be boiled off in order to raise SG levels to the appropriate setpoint. However, the <u>rate</u> at which FW builds SG levels can be excessive and overcool the primary system.

In addition, excessive emergency feedwater can cause depressurization of the affected SG to a larger extent than excessive main feedwater. This is due primarily to the increased condensing action introduced by spraying colder EFW in near the top of the tube bundle (into the steam space).

Should the EFW or MFW flow be excessive, the operator should recognize the overcooling as well as high FW flow. Following the actions in Part I, Section III.C for excessive primary to secondary heat transfer will terminate the excessive FW. Step 5.0 of III.C requires the operator to close the EFW regulating valves and steps 5.3 and 5.4 isolate MFW.

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Figure E-3 shows the impact of small steam leak overcooling compounded by overcooling due to excessive feedwater. The curves for RCS pressure and pressuizer level will shift to the left, i.e., pressure reduction and drainage of the pressurizer will occur faster. Excessive EFW (and excessive MFW through the upper nozzles) will also cause the steam pressure curve to shift to the left.

Loss of Secondary Inventory Control (Low)

A loss of secondary inventory control (low) exists whenever too little feedwater is being supplied to the steam generators resulting in too little primary to secondary heat transfer and overheating of the RCS. The operator should recognize this condition by low feedwater flow rates and low SG levels.

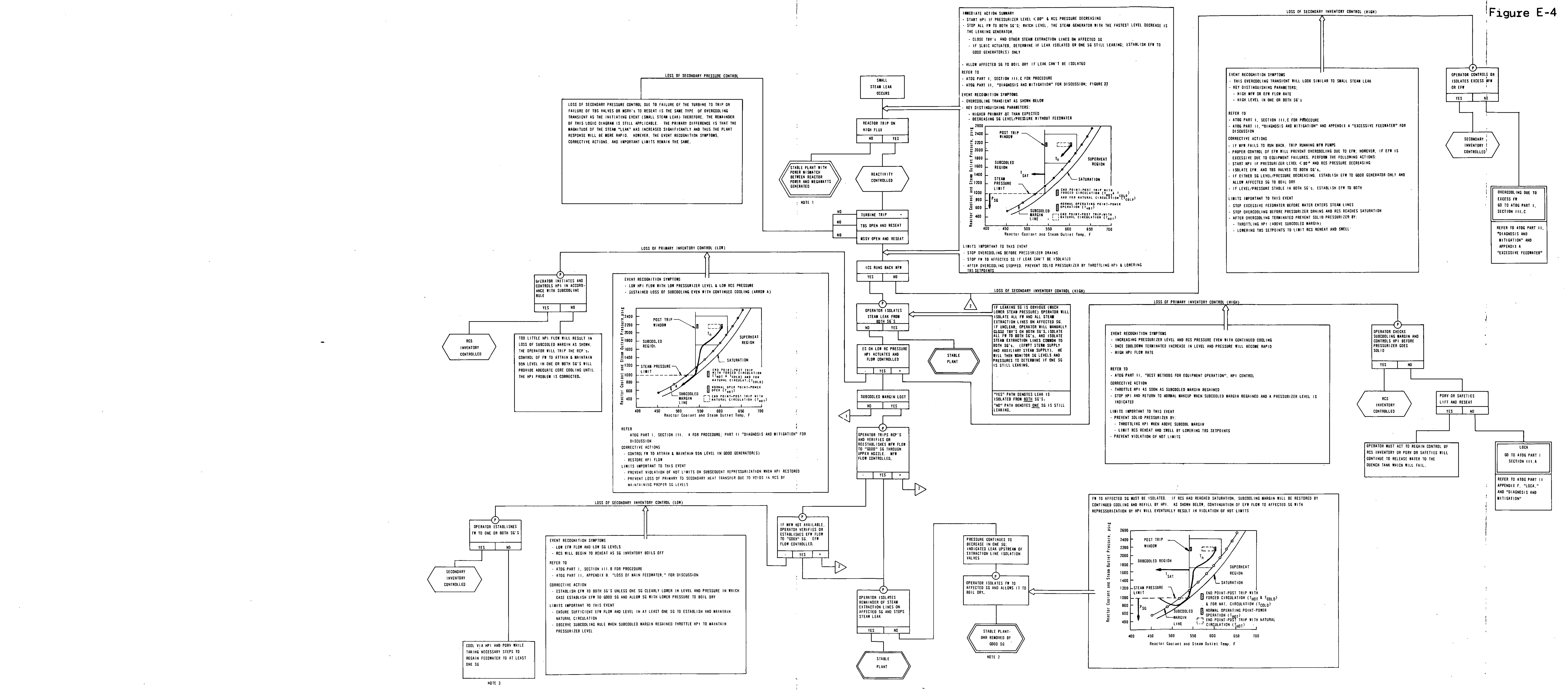
The operator may be required, by the procedures in Part I, to isolate all feedwater in order to determine which SG is leaking. However, he will also by procedure immediately restore feedwater to the good SG as soon as the broken SG is identified and isolated.

A detailed discussion concerning too little primary to secondary heat transfer is provided in Appendix B, "Loss of Main Feedwater".

Loss of Steam Pressure Control

A loss of steam pressure control exists whenever one or both steam generators undergo a pressure reduction significantly below the TBS reseat setpoint. It is an overcooling transient and, since a small steam leak is a loss of steam pressure control, it will look similar (on the P-T curve) to the main transient in Section 2.0 of this appendix. It can be caused by excessive emergency feedwater (discussed earlier) or by an unplanned steam flow through stuck open valves or a pipe break. EFW should be throttled to minimize pressure reduction while providing a gradual SG level increase as discussed in "Best Methods for Equipment Operation", in Part II. Section 2.0 of this appendix already covered loss of steam pressure control due to steam leaks alone.

Figure E-4 SMALL STEAM LEAK LOGIC DIAGRAM



NOTES:

1. IF SMALL STEAM LEAKAGE PATH IS THROUGH TBS VALVES PLANT MAY OPERATE WITH POWER MISMATCH INDEFINITELY WHILE SOURCE OF LEAKAGE IS BEING LOCATED AND CORRECTED. IF LEAKAGE PATH IS OTHER THAN TBS BUT OUTSIDE OF RB, A GRADUAL DECREASE IN SECONDARY WATER INVENTORY WILL OCCUR. IF LEAKAGE PATH IS INSIDE RB, CONTAINMENT TEMPERATURE AND PRESSURE WILL INCREASE, THE ESAS 4 PSIG TRIP SETPOINT CAN BE REACHED IN ~ 20 SECONDS FOR A 25% MISMATCH.

2. AT THESE POINTS "STABLE PLANT" DENOTES FACT THAT OVERCOOLING DUE TO SMALL STEAM LEAK HAS BEEN TERMINATED AND PROPER PRIMARY TO SECONDARY HEAT TRANSFER HAS BEEN ESTABLISHED USING GOOD SG. HOWEVER, THE OPERATOR STILL NEEDS TO RE-ESTABLISH NORMAL PRIMARY SYSTEM INVENTORY CONTROL BY:
   - THROTTLING HPI WHEN SUBCOOLED MARGIN RESTORED
   - REDUCING TBS SETPOINT TO LIMIT RCS REHEAT AND SWELL (ONLY ON GOOD SG IF ONE SG IS BROKEN)
   - TERMINATE HPI AND RE-ALIGN FOR NORMAL MAKEUP/LETDOWN OPERATION WHEN RCS IS ABOVE SUBCOOLED MARGIN AND PRESSURIZER LEVEL IS ON-SCALE LOW AND INCREASING.
   - THROTTLE FW FLOW TO OBTAIN A GRADUAL SG LEVEL INCREASE TO THE APPROPRIATE LEVEL SETPOINT
   - IF IN NATURAL CIRCULATION, THE OPERATOR SHOULD RESTART AN RCP WHEN SUBCOOLED MARGIN IS RESTORED

3. HPI COOLING IS NOT A STABLE END POINT. THE OPERATOR WILL PERFORM THE FOLLOWING ACTIONS TO STABILIZE THE PLANT:
   - WHILE ON HPI COOLING MONITOR INCORE THERMOCOUPLES TO PREVENT THERMAL SHOCK
   - RE-ESTABLISH FW TO AT LEAST ONE SG.
   - THROTTLE HPI WHEN SUBCOOLED MARGIN RESTORED
   - USE PRESSURIZER HEATERS TO DRAW A BUBBLE AND RESTORE RCS PRESSURE CONTROL
   - ISOLATE PORV AND CONTINUE NORMAL PLANT COOLDOWN

4. "STABLE PLANT" DENOTES FACT THAT OVERCOOLING DUE TO SMALL STEAM LEAK HAS BEEN TERMINATED. HOWEVER, FW AND TBV'S ARE ISOLATED ON ONE SG AND MAYBE BOTH. THE OPERATOR STILL NEEDS TO RE-ESTABLISH NORMAL PRIMARY TO SECONDARY HEAT TRANSFER BY:
   - RE-ESTABLISH NORMAL MFW FLOW THROUGH MFW NOZZLES ON BOTH SG'S (EFW IF MFW NOT AVAILABLE)
   - USE MANUAL TBV CONTROL TO LIMIT HEATUP RATE OF RCS.

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APPENDIX F

LOSS OF COOLANT ACCIDENTS

1.0 Introduction

A loss of coolant accident (LOCA) is defined as any event during which reactor coolant escapes due to loss of integrity of the primary coolant system. A LOCA can occur at any time (power operation, cool-down, heatup, cold shutdown) and can be either an initiating event or a result of another accident. Usually LOCA's release reactor coolant to the contaminant environment; however, there are some LOCA's which can release reactor coolant to the secondary plant (steam generator tube leaks) or to the auxiliary building (leaks in the MU, letdown, or decay heat removal systems).

LOCA's directly affect one of the five fundamental principles of reactor operation: "RCS Inventory Control". The loss of reactor coolant reduces the plant's ability to transport the core's heat to the steam generators. If the lost reactor coolant is not replaced, core cooling will be lost and fuel damage may occur. The Emergency Core Cooling Systems (ECCS) consisting of the High Pressure Injection (HPI), Low Pressure Injection (LPI), and Core Flooding Tanks are used to supply emergency core cooling water to prevent fuel damage. The main feedwater or emergency feedwater system is also used for core heat removal and RCS depressurization for some sizes of small break LOCA's.

A LOCA is a unique accident. Some of the major factors which "single out" the LOCA from other abnormal transients are as follows:

A. <u>A wide range of leak sizes is possible:</u>

The rate at which reactor coolant is lost from the primary system depends on the <u>break size</u> through which the reactor coolant can pass. LOCA's can range from small leak rates (leaking pressurizer relief or safety valves, leaking RC pump seals, cracks or breaks in instrumentation lines connected to the primary system) to large losses of reactor coolant due to cracks in or a severance of the RCS piping or its attachments (letdown line, HPI lines, decay heat drop line).

B. <u>Abnormal system conditions are a natural consequence of the event:</u>

When reactor coolant is lost at a rate greater than it is replaced, pressurizer level will be lost, the reactor coolant will become saturated, a loss of normal natural circulation (assuming the RCP's are tripped) is inevitable and reflux boiling may be the only means by which the steam generators can remove energy from the reactor coolant. For many LOCA's these conditions are temporary and can be prevented by operator isolation. For a LOCA which cannot be isolated, the evolution of the transient to abnormal conditions is unavoidable.

C. Steam generator heat removal may be degraded:

The transport of core heat to the steam generators may be severly reduced depending upon the size of the LOCA. Steam generator heat removal is needed for some small breaks, even though the break itself also removes energy from the primary system and uses the reactor building as a heat sink. If the break is large enough the steam generator is not needed as a heat sink. In some large breaks the RCS depressurizes below steam pressure, and the steam generator can even be a heat source if the temperature of the secondary inventory is higher than that of the reactor coolant.

D. Maintenance of hot shutdown is not a safe end condition in most cases:

Some LOCA's can be isolated and the loss of reactor coolant stopped. Hot shutdown is a safe condition. However, when isolation is not possible the loss of reactor coolant cannot be stopped. Then the RCS must be depressurized to reduce the leak, increase the ECCS flowrate, engage the LPI system and establish long term cooling. It is also desirable to depressurize before the BWST is empty to avoid HPI piggyback operation. Depressurization is also desirable to engage the LPI system so the steam generators do not have to be used with service water when the condensate storage tank empties (unless supply available from the other units). Depressurization is required for the tube leaks to limit the offsite doses and to avoid emptying the BWST (tube leakage is not returned to the sump and recirculation is not possible).

E. The containment environment will degrade:

If the break is inside the containment (most breaks will be) reactor coolant water and steam will collect in the reactor building. The pressure, temperature and radiation levels in the reactor building will increase, and use of emergency building cooling systems may be necessary. In addition to maintenance of core cooling, the operator must also control the reactor building environment to prevent or limit offsite radiation releases. Control of the environment is also desired to prevent equipment damage. Containment isolation is required to limit the offsite doses.

LOCA's can be complicated events because of the wide range of system conditions that can happen. The plant's safety systems, especially the ECCS, are capable of safely limiting the LOCA effects. Core cooling can be maintained solely through use of the ECCS, but operator actions and other equipment can be used to enhance the performance of the ECCS and to achieve better results.

The remaining portions of this appendix will describe the response of the RCS for the full spectrum of LOCA's, the general approach to accident mitigation and establishment of long term cooling, how the ECCS and other equipment are best used, and also outline control limits which apply to plant recovery.

2.0 LOCA CONCEPTS

Loss of coolant accidents are classified in three categories:

1. Large Break
2. Small Break
3. Small Leak

These categories are used because the response of the RCS is different. Each category will be discussed separately.

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Large Breaks

A large break is a major failure of the primary system pressure boundary which depressurizes the reactor coolant system rapidly and almost completely. This is a designer's event that establishes the size of the core flood tanks, the size of the LPI flow rate and the size of the containment cooling systems (sprays and coolers). A large break LOCA has never occurred at a commercial nuclear power plant.

As a rule of thumb, a LOCA with a break area greater than that of a 10-inch diameter hole is a large break. For breaks of this size or larger, the plant transient can be broken down into three distinct phases: blowdown, refill, and reflood.

Blowdown: Blowdown is a rapid depressurization of the primary system as reactor coolant escapes in large quantities. Large pressure gradients are created in the primary system and

low pressure zones are created at or near the break. These pressure gradients disrupt the normal flow patterns within the primary system, and the reactor coolant will be drawn to the break (low pressure zone) from all regions of the primary system. The reactor coolant will saturate and change to steam due to system depressurization and core heat removal.

Blowdown ends when the primary system and the containment pressures are the same; the loss of reactor coolant stops. From start to finish, the blowdown period can be from 20 to 100 seconds depending upon the break size and location. At the end of blowdown, the reactor coolant system will contain very little water and will be mostly steam.

During blowdown, RC pressure drops rapidly and falls below the secondary side pressure within a few seconds. The steam generators become a heat source as opposed to a heat sink because the temperature of the reactor coolant will be less than the secondary feedwater temperature. The steam generators remove heat only as long as the primary temperature is above the secondary temperature. Steam pressure will hang up initially and then slowly decrease as energy from the secondary is slowly transferred to the primary system.

Blowdown causes a reactor trip, and the fission process is shutdown to the decay heat level. But, because large quantities of steam are produced, the core will heat up; the fuel and cladding temperature will rise until cooling systems recover the core with water.

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During blowdown the HPI and LPI systems are actuated, and the core flood tanks discharge when the primary system pressure drops below 600 psig. These emergency core cooling systems have little or no impact on the blowdown phase of a large break. The blowdown transient is too fast, and the cooling water that does enter the primary system is "swept out" the break along with the reactor coolant. The ECCS systems restore core cooling during the second and third phase of a large break.

The blowdown also has a severe impact on the reactor building. Rapid increase in the containment pressure, temperature, and radiation levels occur. The emergency coolers and sprays are started by the ES by high containment pressure (4 and 10 psig) and water will accumulate in the reactor building emergency sump. The reactor building becomes the heat sink for the energy originally contained in the primary system.

Figure F-1 shows the primary system response during the blowdown transient fo a large break.

Refill: At the end of blowdown the primary system is at a low pressure, and little or no water is left in it (see end condition in Figure F-1). However, the CFT's, the LPI and the HPI will be adding water to the system. The emergency injection water entering the system accumulates and begins to refill the reactor vessel. Refill is defined as the time required to refill the reactor vessel lower head up to bottom of the fuel (see end of refill in Figure F-1). This is a short time (10-12 seconds) and for smaller breaks a refill period may not occur if enough water remains in the reactor vessel.

The refill phase of a large break cannot be detected by any plant instrumentation. As shown in Figure F-1, no water exists in the core during refill and the core will heat up until ECCS water enters the core. When the core is uncovered the hot zircaloy clad chemically reacts with the oxygen of the steam to give zirc oxide and release hydrogen. Hydrogen may accumulate in the loop or it may be vented through the break to the containment.

Reflood: The last phase of a large break is called reflood. During reflood, water from the ECCS refloods the RV up to the elevation of

the inlet nozzles and refills the system up to the elevation of the break. This is a slow process because a lot of the emergency cooling water is boiled off as it enters the core since decay heat levels are still high.

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Within 5 to 10 minutes the core will be recovered with water and the fuel clad temperatures reduced to within a few degrees of the saturation temperature (the core exit thermocouples should show approximately saturated conditions). Core cooling thereafter is maintained by continuously adding injection water from the low pressure injection system so that the core remains covered by water. The CFT will empty, and the HPI system can be secured (see Guidelines for HPI termination in "Best Methods for Equipment Operation").

Figure F-1 shows the reflood phase. Two examples are given: one for a large break at a high location in the system and one for a large break low in the system.

During the reflood phase, additional steam is created by:

a) boiling of emergency injection water in the core,
b) boiling of emergency injection water due to heat from the hot metal components, and

c) boiling of emergency injection due to heat transferred from the secondary side of the steam generators (this is a source of additional steam only if primary side water re-enters the tube region of the steam generators).

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This steam is vented out the break to the reactor building. The reactor building pressure and temperature conditions may rise during the early stages of reflood. When decay heat and the heat from the hot metal and steam generators drop, the steam release rate will decrease and the reactor building pressure and temperature will decrease as the sprays and coolers can remove more heat than is being released.

Figures F-2 and F-3 give a summary of the expected response (P-T diagram) of the RCS during large hot and cold leg breaks. The diagrams also outline the sequence of events. At the end of blowdown, the reactor coolant is in a saturated condition at or near the containment pressure. Thereafter, the core outlet thermocouples should be used to check the core outlet temperature for saturation. At the end of the blowdown the core outlet thermocouples may indicate superheated conditions since the core may be completely void of water. As the ECCS refloods the reactor vessel, the fuel rods will be quenched and the core outlet temperature should return to saturation.

Small Breaks

The second category of LOCA's are small breaks. Small breaks are not as severe as large breaks. The depressurization is much slower, the mass is lost at a lower flowrate, and the core cooling can be maintained throughout the accident as long as the ECCS is operating. For small breaks the operator can play a vital role in minimizing the consequences of the accident.

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As a rule of thumb, small breaks are between small leaks and large breaks:

Small Leak				Large Break
Breaks for which the leak rate is within the capacity of the MU system (area approximately equal to that of a 3/8" hole)	<	SMALL BREAK	<	(Area approximately equal to or greater than that of a 10 in. diameter hole).

Small breaks include cracks in the primary system piping, breaks of lines attached to the RCS (HPI lines, letdown line, spray line), and failed open pressurizer relief or safety valves. The reference hole sizes given above are not exact boundaries for small breaks. They are given only as physical reference points to promote a feel for

the spread in break size for this category of LOCA's. The response of the RCS during a LOCA is actually a better way to gauge whether a specified LOCA is a small leak, small break, or a large break.

A small break transient is characterized by a slow depressurization relative to large breaks. Flow conditions in the RCS change gradually and smoothly, and temperature and pressure gradients between different places in the primary system tend to be small. Rather than the distinct blowdown, refill, and reflood phases associated with large breaks, small breaks have a period of relatively high loop flow followed by a period of relatively low or no loop flow. The no loop flow conditions are a semi-stable period where circulation stops and the steam separates from the water. When the steam and water separate the reactor coolant system is in a "boiling pot" mode.

Figure F-4 shows a small break, including P-T responses and sequence of events. Early in the transient reactor coolant will circulate from the core to the steam generator, and the steam generator will remove heat. This part of the transient is called the "flow circulation" phase. Flow can circulate by natural or forced circulation. For very small breaks natural circulation (normal or two phase) will not be lost, and the primary system will return to a subcooled

state. For "larger" small breaks, the circulation flow phase will end almost immediately after the RC pumps are tripped. For all small breaks, core cooling is maintained during the "low circulation" phase.

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After the "flow circulation" portion of a small break, the reactor coolant "settles out". The water falls by gravity and collects in the lower regions, and the steam bubbles up through the water and collects in the high points of the system. A "boiling pot" water level will exist that will vary depending on:

- Break Size
- Break Location
- Primary to Secondary Heat Transfer (steam generator cooling)
- Number HPI/LPI Pumps Operating
- Decay Heat Levels

These variables will cause the response characteristics of the RCS to change in different ways after the reactor coolant "settles" into the "boiling pot". The effects of these variables will be discussed next.

To illustrate the effects of break size, steam generator cooling and the other small break variables, the following five specific examples will be considered:

1. Small breaks large enough to depressurize the RCS
2. Small breaks which stabilize at approximately secondary side pressure
3. Small breaks which may repressurize in a saturated condition
4. Small breaks without primary to secondary heat transfer
5. Small breaks within the pressurizer steam space

These discussions will also highlight the effects of break size on the "flow circulation" phase of a small break.

<u>Small Breaks Large Enough to Depressurize the RCS</u>

At the "large" end of the small break spectrum, the breaks release enough mass and energy from the system such that the primary system will continually depressurize to a stable low pressure condition. For these breaks the "flow circulation" period is over very rapidly. RCS pressure will also drop very rapidly below the secondary side pressure. Therefore, for these breaks primary to secondary heat transfer does not play a vital role in the accident. During the "boiling pot" period, the water levels in the system will quickly drop to the elevation of the break. Thereafter the water levels within the system will decrease until emergency cooling water flow-rate exceeds the rate of water boiloff in the core. The net amount

of water in the system will then increase slowly, and the water levels in the system will gradually build back up to the break elevation. For small breaks of this size the primary system will never refill water above the break.

Figures F-5a and F-5b show two examples of the system P-T response up to the time where the lowest system water levels occur. These figures show that these breaks depend upon the HPI and CFT's to maintain adequate water in the system to keep the core covered. When the system finally depressurizes, LPI increses the water level up to the break elevation. The steam generators provide very little cooling for this size of break and are not very important for accident mitigation.

Small Breaks Which Stabilize At Approximately Secondary Side Pressure

When the break size is smaller than that of the previous example, the leak rate and consequently the energy removal rate of the break decreases (the break removes energy when reactor coolant escapes). These smaller breaks cannot remove energy from the primary system fast enough to allow a continuous RCS depressuriaztion. Following the "flow circulation" portion of the transient, the system will stabilize at or near the secondary side pressure. For these breaks, the steam generators can move a significant amount of the core's decay heat by condensing primary system steam within the tube region. Steam generator operation is important for these break sizes. To promote reactor coolant steam condensation (commonly called reflux boiling) and to reduce RCS pressure, the SG water levels should be increased to 95% on the operate range. This must be done manually and is one of the more important operator actions.

The system response for small breaks within this size range is illustrated in Figure F-6. As shown in Figure F-6, the "flow circulation period" is short-lived, and the system stabilizes slightly above steam generator pressure. Because the RCS pressure remains high, HPI is the only means by which emergency cooling water can be supplied to ensure core cooling until the system depressurizes further. Since the generators are removing heat from the RCS by condensing steam, another important opertor action will be to begin a plant cooldown and depressurization by decreasing steam pressure. The primary system will depressurize because more steam will be condensed. As system pressure is decreased the ECCS system will flow at a

higher rate, and a refill of the system becomes more likely. During the refill, RCS pressure may stabilize or increase slightly (while still saturated) as the RCS water level coveres the condensing surface in the steam generators. For this break size range, a refill and repressurization is not likely until the RCS pressure has been lowered to the LPI pressure. Repressurization of the RCS when the RC is in a saturated state will be more likely for the next category of breaks in the small break range.

Small Breaks Which May Repressurize in a Saturated Condition

For the sizes smaller than the two previous examples the loss of reactor coolant mass and energy is sufficient to initially depressurize the RCS and results in enough steam formation to interrupt loop circulation. When the flow stops and the reactor coolant "settles out", the RCS may gradually repressurize. The repressurization occurs because the break alone is not large enough to depressurize the RCS and the rector coolant steam in the hot leg cannot be condensed because the reactor coolant water level is higher than the tubes. The primary system can repressurize as high as the pressurizer safey valve setpoint once enough RCS inventory has been lost and the reactor coolant water level drops below the upper feedwater injection level, reactor coolant steam will condense on the tube and the reactor coolant pressure will drop to the steam generator pressure. Figure F-6 illustrates this repressurization behavior as it would be observed on a P-T diagram. During the repressurization phase, pressurizer level may return on scale, as shown in Figure F-7, because the reactor coolant is heating up and expanding. For more severe cases, the pressurizer level may go off scale high. Because the reactor coolant is saturated, HPI flow should not be throttled. Two HPI pumps should be kept on full, and the PORV should be opened if RCS pressure increases above the PORV pressure setpoint.

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Summary

The previous three examples highlighted the importance of small break size and steam generator cooling and showed how these two variables interplayed to change the reactor coolant system pressure. The reactor coolant system pressure is one of the most important factors in small break control. When the reactor coolant system pressure is high the leak flow will be high and the HPI flowrate will be low. The steam generator is valuable because it condenses reactor coolant steam and lowers the RCS pressure. To get the best effect steam generator level should be raised and FW flow continuously run through the upper nozzles until the steam generator level is at 95% on the operate range.

Although decay heat level and HPI flow rate were not specifically addressed for their effects, their influence is important. As the decay heat level lowers with time (or if it was low at the start) the heat input to the reactor coolant is reduced and the HPI flow can more easily "match" decay heat. Pressure will fall quicker with lower decay heat levels than with high. HPI flowrate is important because it will keep the core covered and cooled when two HPI pumps are running. One pump will take much longer to match decay heat and if the system pressure is high the flowrate will be very low; steam generator cooling is desirable to reduce the RCS pressure to enable more HPI flow.

The next three examples will repeat the first three, except that steam generator cooling is not available. The final example will illustrate the importance of break location.

Small Break Without Primary to Secondary Heat Transfer

Steam generator heat transfer becomes more and more important as the break size gets smaller. This is due to the inability of the break to remove decay heat by release of reactor coolant. Without primary to secondary heat transfer, the RCS is more likely to repressurize or to depressurize more slowly. Figures F-8 to F-10 show system responses for three different break sizes. These break sizes are approximately the same as previously illustrated; those had steam generator cooling and these do not.

Figure F-8 shows the size where the break is large enough to continually decrease RCS pressure. For these breaks, which can depressurize the RCS below the normal secondary side pressure, the loss of secondary inventory has little or no effect on the transient. Core cooling is maintained by the ECCS. Because these breaks are fairly large, pressurizer level would rapidly fall and remain off-scale low.

Figure F-9 shows the next size smaller break. The RCS quickly depressurizes; the HPI's are actuated; and the RC pumps are tripped when the reactor coolant subcooled margin is lost. Pressurizer level also goes off-scale low. When flow circulation stops and the steam and water separate, the RCS pressure hangs up. RC pressure would normally drop to the secondary side pressure if primary to secondary heat transfer were available. If feedwater is restored and primary to secondary heat transfer established, the RCS pressure would drop and reflux boiling would start.

Figure F-10 shows the smallest break size. ES may not be automatically actuated and the RCS repressurizes when the secondary inventory is boiled off. It is possible for the RCS to repressurize to the pressure setpoints of the PORV or the pressurizer safety valves and for pressurizer level to go off-scale high. For any break size without secondary inventory, the operator should actuate the HPI (if not already on) and open the PORV when primary to secondary heat transfer is lost. The RC pumps should also be tripped if the RC subcooled margin is lost. The RCS is being cooled by "HPI LOCA Cooling". Figure F-10 shows a transient where no operator actions are taken for the first 20 minutes. After 20 minutes, Figure F-10 shows the effects of restoring feedwater and starting HPI. The RCS depressurization causes an outsurge from the pressurizer. A complete loss of pressurizer level may result. Thereafter, the system response is similar to that of a small break with primary to secondary heat transfer.

Small Breaks Within the Pressurizer Steam Space

For small breaks within the pressurizer, the system pressure-temperature response will be similar to that discussed previously. The initial depressurization will be faster because steam, rather than water, is released (steam has a higher energy content per pound than water does). The big change for pressurizer breaks is the response of pressurizer level. Figure F-11 shows a P-T diagram and transient pressure and pressurizer level histories for a stuck open PORV. As shown in Figure F-11, pressurizer level initially rises; this is due to the pressure reduction in the pressurizer and an insurge into the pressurizer from the RCS prior to reactor trip. Once the reactor trips, the reactor coolant contraction (normal post trip response) causes a decrease in pressurizer level. Because RC pressure is dropping, flashing occurs in the hot leg (saturation), and the steam bubbles expand and force water into the pressurizer. The pressurizer subsequently fills and will remain full for the remainder of the transient. In the long term, the pressurizer may contain a two-phase mixture, and the indicated level may show that the pressurizer is only partly full. This peculiar behavior is the main reason that pressurizer level cannot be relied on to measure reactor inventory until the reactor coolant is subcooled. If the break in the pressurizer is fairly small and can't be isolated, the HPI can refill the system and the pressurizer can be water solid and subcooled. This would not be the expected response for a stuck open PORV since the PORV can be isolated. For these cases, attempts to draw a pressurizer steam bubble will fail, and a solid water cooldown will be necessary.

Small Leaks

Small leaks are events where the loss of reactor coolant is within the capacity of the normal makeup system. Several possible occurrences which could fall in this category are leaking pressurizer safety or relief valves, leaking RC pump seals or tiny cracks in the primary system piping or attachments (instrument lines, RTD hotwells, or auxiliary system lines). This category would correspond to a 3/8 inch or less diameter hole in the primary system.

When the leakage rate is very small, there is no impact on the primary or secondary system. These very small leaks are discovered due to the necessity to add water to the letdown storage tank more frequently than normal. The action required is in the Technical Specifications.

At the upper end of the leak rate for small leaks, more direct symptoms may be observed, such as

1. Excessive MU Flow
2. Decreasing pressurizer level and pressure
3. Reactor trip

These symptoms would likely exist when the leak, combined with letdown flow, is greater than the capacity of the MU system. This transient is very slow, but pressurizer low level and pressure and letdown storage tank low level alarms show the imbalance in reactor inventory control. There is time to increase the makeup rate to the RCS and stop letdown. Subcooling should not be lost; but if a reactor trip occurs, the pressurizer

might drain and HPI will be started. The plant can be cooled down in a near normal manner unless a small leak in the pressurizer steam space occurs. If the steam release rate is greater than the steam production capacity of the pressurizer heaters, the pressurizer will fill and a solid water cooldown will be necessary if the leak is unisolable.

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3.0 POST LOCA - PLANT CONTROL

The primary objectives during a LOCA are to maintain core cooling, to cooldown and depressurize, and to establish a stable long-term cooling mode. Maintenance of core cooling is the first priority and is achieved through operation of the ECCS. Maintenance of core cooling will limit fuel damage and thereby limit radiation release. With essentially no actions other than verifying the rods fully inserted following a reactor trip and tripping the reactor coolant pumps on loss of subcooling, the core can be cooled by ensuring that the ECCS works when required and can be run continuously until their safety function is satisfied. In addition to ensuring core cooling, the operator must make sure that containment cooling is working and that long-term cooling must be established. The general things to be done after a LOCA are:

Core Cooling

1. Actuate HPI on loss of subcooled margin (if not automatically actuated).

2. Trip RC pumps on loss of subcooled margin.
3. Verify automatic start and injection flow rate of LPI and HPI after appropriate ES setpoints are reached.
4. Balance LPI and HPI as necessary.
5. Verify that Core Flood Tanks release when RC pressure drops below 600 psig. Isolate CFT's after discharge to prevent N_2 from entering RCS.

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6. Locate and if possible isolate the break (check for steam generaor tube leaks and leaks inside or outside containment).
7. Control steam pressure and raise steam generator level to 95% on the operate range if the reactor coolant subcooled margin is lost (except for the "bad" generator for tube ruptures) for small breaks. If a large LOCA has occurred, isolate and let both generators dry out.

Containment Control

8. Verify that containment cooling systems start (at 4 psig and 10 psig).
9. Verify that containment isolation has occurred and that cooling water services to important components (RC Pumps) are restored if they are to be used later.

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Long Term Cooling

10. Switch the suction of the LPI to the reactor building emergency sump when the BWST low level is reached. If the RCS is not depressurized enough to use LPI, place the HPI in piggyback on the LPI and use recirculation from the sump for injection.
11. Turn off containment sprays when containment pressure reaches 4 psig and is decreasing. Keep containment coolers running.
12. For large breaks in the sump recirculation mode, stop one LPI train and leave one running so long as the core exit thermocouples indicate the core is cooled. This will preserve one pump in case it is needed later. Continue to monitor LPI flow.
13. Start hydrogen purge system when containment hydrogen samples show a 3.5% concentration.
14. Take actions to prevent boron precipitation.

For large breaks, maintenance of core cooling leads directly to a stable long-term cooling mode because the break is large enough to depressurize the RCS and secondary pressure and inventory control have little or no effect on the course of the accident.

For small breaks, core cooling can be maintained using two HPI's. The core will be cooled, but reactor coolant will continue to be lost in large quantities as long as the RCS is at high pressure. Depressurization and cooldown using the steam generator is necessary to decrease the leak and to

allow for a potential refill of the primary system. Depressurization to cold conditions is essential for tube leaks because all the BWST water is lost out of the leak and cannot be recirculated from the sump; the plant must be placed on the decay heat removal system before the BWST drains.

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In the following sections, the general approach to accident mitigation (core cooling) and establishment of long-term cooling using the Abnormal Transient Operating Guidelines will be outlined for loss-of-coolant accidents. The full spectrum of LOCA's will be addressed with the exception of the very small breaks or leaks which would typically never result in loss of reactor coolant subcooled margin. For these accidents, the loss of reactor coolant can be matched by the normal makeup system or the HPI system; and plant control is essentially no different than any other abnormal transient which results in a reactor trip followed by a plant cooldown and depressurization.

General Overview

Table F-1 outlines the general approach to post-LOCA plant control from a core cooling standpoint. This sequential breakdown of operator actions has many of the same features described in the accident mitigation chapter for non-LOCA's except that actions to cool down the plant and to establish long-term cooling that are unique to LOCA have been included. If a LOCA can be located and isolated, however, the plant can be stabilized at or near a hot shutdown condition since the loss of reactor coolant has been

stopped. Although not shown in Table F-1, the containment environment must also be controlled; both short and long term actions from a containment integrity standpoint will be discussed.

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Immediate Actions and Vital System Status Checks

The operator actions required during the first 2-3 minutes of a LOCA are identical to those for any abnormal event. These actions include the immediate actions and verification that systems are working properly as outlined in the Accident Mitigation Chapter. For LOCA's, some of the required actions are of special significance:

1. Verification that ES starts HPI and LPI and that containment cooling and isolation have occurred.
2. Trip RC pumps if the reactor coolant subcooled margin is lost.

The ES initiates and aligns the HPI and LPI systems for emergency injection, starts emergency containment cooling systems (sprays and fan coolers) for high pressure containment conditions, and isolates non-vital containment penetrations to limit offsite dose releases. Tripping the RC Pumps, as explained in the chapter on Best Methods of Equipment Operation, is a preventive action specifically for a small break. If the RC pumps are tripped on loss of reactor coolant subcooling margin, enough reactor coolant will be retained within the primary system (not lost out the break) and HPI will maintain the core covered and cooled. Both loss of subcooling margin and ES actuation are alarmed (visually and audibly) in the control room to alert the operator of the plant's status. By verifying that the ES-actuated systems automatically start (or by applying corrective actions in the event of failure in ES-actuated systems) and tripping the RC pumps, the operator ensures:

1. Adequate core cooling for the vast majority of possible LOCA's.
2. Containment integrity so that the offsite doses will be within acceptable limits.

<u>Monitoring</u>

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During and immediately following the immediate actions, the P-T and other parameters should be monitored to determine if the abnormal transient is a LOCA and not some other accident. Many overcooling accidents will look like small breaks, and all small break LOCA's will not look the same. In some cases, a LOCA can only be determined by showing that some other accident is not underway. Usually the accidents that can be eliminated readily are overcooling accidents; these can often be eliminated by reviewing steam pressure, secondary water level, and feed flows.

The operator should get a "feel" for LOCA's by comparing the examples of this section with other transients. LOCA's do have some unique characteristics; these are shown in Table F-2. (This is Table 4a from the "Abnormal Transient Diagnosis and Mitigation" chapter, repeated here for convenient reference.)

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Table F-2. HOW TO DISTINGUISH LOCA'S FROM OTHER TRANSIENTS

Unique Characteristics of LOCA's

- Rapid system depressurization to saturated conditions with little or no change of reactor coolant temperature (characteristic of all but the very smallest breaks)

- Sustained saturation (HPI does not return the reactor to a subcooled state within ~10 minutes after actuation)

- Containment radiation (only for breaks in containment)

  NOTE: A steam or feed line leak inside containment will cause high pressure, temperature and humidity but will not cause high radiation.

- Steam pressure, feed flow and steam generator level do not indicate overcooling (this helps to differentiate LOCA's from overcooling transients).

- High steam line radiation alarms (tube leaks only)

- Low letdown storage tank level (in the absence of all of the above, this indicates a leak outside the containment)

  NOTE: LOCA'S CAN BE DIFFICULT TO DETECT, ESPECIALLY IF THE BREAKS ARE SMALL. THEY CAN OCCUR INSIDE THE CONTAINMENT OR OUTSIDE. STEAM GENERATOR TUBE LEAKS ARE LOCA'S. IF THERE IS ANY DOUBT THAT AN ACCIDENT IS A LOCA, ASSUME THAT IT IS AND TAKE APPROPRIATE LOCA ACTIONS UNTIL CLEARLY PROVEN OTHERWISE. THE GENERAL ACTIONS INCLUDE HPI COOLING, RC PUMP TRIP, AND COOLDOWN TO COLD CONDITIONS.

For most LOCA's the general symptoms shown in Table F-2 will be apparent, and other indications will supply additional evidence that a LOCA is in progress. Table F-3 scopes the characteristics of a wide range of LOCA break sizes and summarizes the other evidence to be used to diagnose that a LOCA is occurring. Table F-3 includes both the pressure-temperature response characteristics and other event or plant symptoms for the complete spectrum of LOCA's. Some LOCA's can be isolated. Table F-4 gives symptoms for LOCA's that can be located and isolated. (This is Table 4b from the "Abnormal Transient Diagnosis and Mitigation" chapter repeated here for convenient reference.)

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Some LOCA's (such as large breaks) have such distinctive characteristics that a quick diagnosis is assured. For small LOCA's the event may not be properly diagnosed for some time into the event, but core cooling is assured so long as the ECCS systems are flowing fully and the RC pumps are tripped on loss of subcooled margin. The symptoms in Tables F-2, F-3 and F-4 should be studied because some LOCA's can be isolated and the loss of reactor coolant stopped.

Corrective Actions for LOCA's - Introduction

As outlined in the Accident Mitigation chapter, the actions to be taken in response to abnormal plant symptoms are aimed at restoring and controlling primary-to-secondary heat transfer or starting backup cooling methods. For LOCA's, in addition to restoring primary-to-secondary heat transfer, an RCS cooldown and depressurization must be started. The plant is not in a safe condition until it is depressurized and the ECCS systems are aligned for long-term cooling (unless the break is isolated). For large breaks, the plant will depressurize quickly and only actions to establish and maintain long-term cooling are required.

The three symptoms for which operator actions are based during any abnormal event are:

1. Lack of Adequate Subcooled Margin
2. Lack of Primary to Secondary Heat Transfer (overheating)
3. Too much Primary to Secondary Heat Transfer (overcooling)

During a LOCA all of the above symptoms can exist at some time during the transient. Lack of adequate subcooling and/or primary to secondary heat transfer are the expected symptoms, because the loss of reactor coolant will result in saturated conditions and impede the transport of core heat to the steam generators. The following sections will show how to control the RCS for LOCA, control containment cooling and ensure isolation, and establish long-term cooling.

Corrective Actions for LOCA's - Loss of Subcooling Margin

Figure F-12 outlines the symptoms of and general actions for a lack of subcooling margin for a LOCA. Tripping the RC Pumps is a preventive action for small breaks as discussed previously, and manual HPI actuation serves as a backup to the normal start feature provided by the ES. Verification of the HPI flows is a general action to be taken following HPI actuation at any time, and is required to ensure that the greatest amount of pumped flow enters the core. The chapter on Best Methods outlines specific ways to correct abnormal HPI flows. The third action in Figure F-12 is raising the steam generator water level to 95% on the operate range. Controlling the water level high in the steam generator is required for small LOCA when the pumps are tripped because it aids reflux boiling (boiling in the vessel and condensing in the steam generator) for decay heat removal and can help to establish natural circulation. When raising the water level in the steam generators, full feedwater flow is not required. FW should be throttled and the water level raised in a continuous and controlled manner as described in the Best Methods chapter. The last action, which is to attempt to locate and isolate the break, should be taken whenever a LOCA is suspected. Typically, the PORV and block valve should be routinely closed under these conditions as a precautionary measure. In addition, Table F-4 also outlines symptoms of other specific LOCA's which can be located and shows how they can be isolated. By performing the four actions given in Figure F-12, core cooling is assured for the full spectrum of small breaks. If the reactor coolant returns to a subcooled state due to break isolation or system refill by HPI, the plant can be cooled

down in a near normal manner if no other abnormal plant symptoms exist. If saturated conditions persist with heat transfer to the steam generators (see example P-T diagram on Figure F-12), additional evidence to support a LOCA diagnosis should be gathered (see Tables F-2 and F-3). With evidence to confirm or suspect a LOCA, an immediate cooldown of the system to establish a stable long-term cooling mode should be started. If lack of primary to secondary heat transfer (overheating) exists, these conditions must be treated to restore use of the steam generators for plant cooldown.

Corrective Actions for LOCA's with Lack of Primary-to-Secondary Heat Transfer

Figure F-13 outlines the symptoms of and general corrective actions for a lack of primary to secondary heat transfer (overheating) during a LOCA. Actions are shown for two types of overheating transients. The first type (right-hand portion of Figure F-13) is overheating due to a loss of secondary inventory (no feedwater), and the corrective action is to use HPI cooling (open PORV with two HPI pumps on) until feedwater is restored. The second type is overheating due to reverse heat transfer (RC pressure drops below secondary pressure) or due to loss of steam condensing surface in the steam generators as the primary system refills. For reverse heat transfer symptoms to exist, a fairly "large" small break must be in progress because the break is large enough to depressurize the plant. For these LOCA's the primary and secondary systems can be recoupled by lowering steam pressure. If RC pressure follows steam pressure, small break cooldown procedures should be followed to establish long-term cooling. For plant symptoms which imply a lack of primary to secondary heat transfer due to the inability to condense steam in the generators, methods are available to stimulate a return to "normal" natural circulation. The method used to start natural circulation is to lower steam saturation temperature about 50F below hot leg temperature and bump an RC pump (see pump restart criteria in the "Best Methods" chapter). When the pump is bumped, RCS pressure will drop to secondary pressure as primary steam is swept into and condenses in the steam generators. As a result of the reduction in primary system pressure, the HPI can add more water to the

primary system and the system may refill and establish natural circulation. If the primary and secondary systems do not recouple after four successive pump bumps, steam temperature should be dropped until it is 100F below the hot leg temperature and a RC pump should be started and run continuously. After these actions, a cooldown can be started to establish a stable long-term cooling mode. If the RC pumps are not available, the PORV can be opened (HPI cooling) to minimize the RCS pressure increase and to increase HPI flow when lack of primary to secondary heat transfer symptoms exist.

Corrective Actions for LOCA's with Overcooling

LOCA's are not overcooling events. Nevertheless, the P-T response of the RCS during a LOCA can resemble a severe overcooling event (too much primary to secondary heat transfer) initially because the reactor coolant pressure and temperature will drop to saturation and steam pressure can also be low. Low steam pressure in both steam generators is possible (without equipment failures) because LOCA's can block the primary to secondary heat transfer process. This similarity between non-LOCA and LOCA events will be short-lived because the non-LOCA overcooling event will return to a subcooled condition fairly rapidly after HPI is actuated.

Figure 22 in the Abnormal Transient Diagnosis and Mitigation chapter outlines diagnostic techniques and corrective actions for too much primary to secondary heat transfer. Corrective actions are identified for high steam generator level (too much feedwater) and low steam pressure conditions. These actions are also appropriate for LOCA's because equipment failures can occur in conjunction with a LOCA to cause excessive heat transfer, even when the reactor coolant is saturated. If steam pressure or feedwater control in one steam generator is lost, that steam generator can be isolated and allowed to boil dry with no loss of core cooling capability.

Cooldown for Small Breaks

The actions identified in Figures F-12 and F-13 and Figure 22 of the Abnormal Transient Diagnosis and Mitigation chapter allow the operator to maintain the plant in a safe condition; but since the plant is still

at high pressure, the leak will continue (unless it has been isolated). Cooldown and depressurization is required. At this point into the accident, the conditions can be one of the following:

1. Isolated LOCA with the plant at hot shutdown (forced or natural circulation).

2. Small break with HPI "on" and
   a. Saturated coolant conditions with primary to secondary heat transfer (RC pumps "on", restarted when treating lack of primary to secondary heat transfer, or "off"); or
   b. Subcooled coolant conditions with primary to secondary heat transfer (RC pumps "on" or "off"); or
   c. Saturated coolant conditions (RC pumps "off") with no primary to secondary heat transfer (HPI cooling).
3. Large break with the primary system completely depressurized.

Cooldown and depressurization from any of these conditions is possible with or without steam generator heat transfer. Steam generator heat transfer will permit a better cooldown and depressurization, but the plant can be depressurized without it. Large breaks do not require depressurization, but long term cooling must be established. The following section will discuss cooldown from each of these conditions.

Small Break Cooldown (With Primary to Secondary Heat Transfer)

a. Subcooled Cooldown

If the reactor coolant has a subcooled margin, the plant can be cooled in a near normal manner to get on the decay heat removal system for long-term cooling. The RC Pumps should be started (see RC Pump restart criteria) if the plant is in a subcooled natural circulation condition on at least one generator. With RC Pumps running and the reactor coolant subcooled, the steam generator water levels can also be controlled at the normal low level limits. If the LOCA has been isolated, a pressurizer bubble can be drawn and cooldown can proceed normally. If the LOCA is not isolated and not in the pressurizer, a pressurizer bubble can be drawn using the pressurizer heaters. The HPI injection system must be throttled to match the leak and to maintain the reactor coolant subcooled as the plant is cooled and depressurized.

If the LOCA is in the pressurizer, a solid water cooldown is necessary because RC pressure is controlled by the discharge pressure of the HPI pumps. To maintain plant control, HPI flow must be reduced slowly to reduce RCS pressure and coordinated with RCS temperature control (by the secondary system) to maintain desirable cooldown limits. Tight control of HPI is necessary at low RC temperatures to avoid a sudden flow increase and subsequent increase in system pressure.

For plant cooldown (with or without a pressurizer steam bubble) the depletion rate of BWST should be monitored. If low BWST levels occur, the LPI pumps must be aligned to take suction from the reactor building sump and the HPI-LPI systems aligned in the piggyback mode (HPI pumps take suction from discharge of the LPI pumps) so as to maintain continuous RCS inventory control. The CFT's may also be isolated if the plant is cooled down in a subcooled state.

b. Saturated Cooldown

For some small breaks, a plant cooldown with saturated reactor coolant conditions will be necessary. In this mode, the steam pressure controls the reactor coolant temperature and RCS pressure. Pressurizer bubble control is not possible when the RCS is saturated. When steam pressure is reduced, the RCS pressure will drop as long as primary steam can be condensed in the steam generator tube region. For saturated cooldown, a high steam generator level is required and the HPI cannot be throttled as long as the subcooling margin is lost. The RCS may refill and reestablish subcooling during saturated plant cooldown because (1) the leak flow decreases and HPI flow increases as RCS pressure is reduced, and (2) decay heat is slowly dropping. If the RC Pumps are off, the need to apply corrective action for loss of primary to secondary heat transfer (see left side of Figure F-13) may again be necessary. BWST level should be monitored and, if it gets low, the ECCS should be realigned to draw from the sump to maintain continuous HPI injection. If the RCS remains in a saturated condition, the CFT

isolation valves should not be closed unless the tank's water volume has been depleted. The plant cooldown and depressurization should be continued to acquire approximately 150 psig to establish long-term cooling.

When 150 psig is reached the normal decay heat removal system can be engaged if the reactor coolant is subcooled. The normal decay heat removal system cannot be engaged if the reactor coolant is saturated, because water level may not exist at the hot leg suction nozzle. When subcooled with the normal decay heat removal system started up, continued makeup either from the HPI or from the LPI (aligned to the sump or BWST) will probably be required. If the LPI is used for makeup, then one string will have to be aligned to the sump or BWST and the other aligned in the "normal" decay heat removal mode. Cooling in this configuration is required until the reactor coolant temperature is below 212F. At this time the additional makeup can be stopped if the break location is above the hot leg suction of the decay heat drop line. If it is below this level or it is not known where the break is, makeup should continue.

Small Break Cooldown (Without Primary to Secondary Heat Transfer, HPI Cooling)

If a LOCA occurs and primary to secondary heat transfer is not possible due to a total loss of feedwater, HPI cooling is used to maintain the core cooled. Generally, as discussed in the Backup Cooling Methods chapter, this is not a preferred method of plant control and alternative actions to

restore primary to secondary heat transfer should be started at once. Should long-term reliance on HPI cooling be necessary, the operator's primary responsibilities while the system is saturated are:

1. To open the PORV and leave it open.
2. To monitor the performance of the HPI system and realign the HPI in piggyback with LPI to draw from the sump on recirculation if the BWST runs out.

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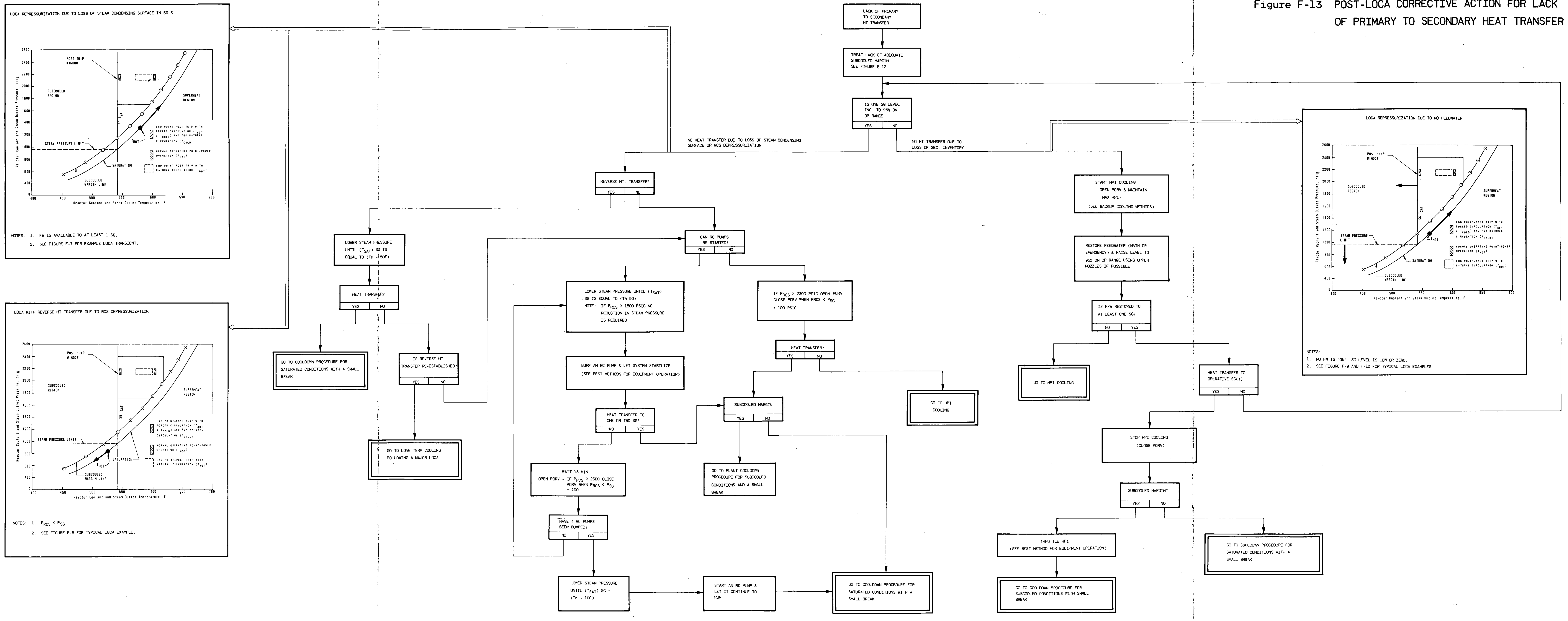
While saturated conditions and no primary to secondary heat transfer exists, it is not possible to make the plant depressurize. Depending on the break size and location, the number of HPI pumps operating, and the plant decay heat level, some different system trends that can occur are:

1. A slow but continuous depressurization as decay heat drops and the break is able to remove more energy.
2. Repressurization to the PORV or pressurizer code safety valve setpoint followed by a slow depressurization as decay heat drops.
3. Either of the above followed by reestablishment of subcooled conditions (some small breaks will never allow a system refill and re-establishment of subcooled conditions)

Should the RCS return to a subcooled condition, HPI can then be throttled to slowly depressurize the plant as the reactor coolant temperature drops. Because there is little or no circulation within the RCS, the incore thermocouples must be used to gauge the reactor coolant subcooling margin. The

Figure F-13 POST-LOCA CORRECTIVE ACTION FOR LACK OF PRIMARY TO SECONDARY HEAT TRANSFER



HPI must be carefully controlled to maintain the primary pressure and temperature conditions to the left of the subcooling margin curve but to the right of the RV brittle fracture limits (see Figure 25) until the DHR system can be engaged.

HPI cooling will eventually allow establishment of a long-term cooling mode where the DHR System (normal or LPI mode) can be used to keep the core cooled. This mode of core cooling (HPI cooling) should be maintained only until alternate means of cooling are possible.

Containment Control for LOCA's

Depending on the size of the break, containment coolers (at 4 psig) or containment sprays (at 10 psig) may be initiated. The only operator actions needed are to monitor their performance to make sure they are working properly. Monitoring performance includes:

1. Ensure that containment pressure is kept below design.
2. Ensure that the cooling systems are operating
   - Sprays - Check spray flow rate
   - Coolers - Check flow rate and check cooler ΔP's for leaks. If leaks occur, the leaking cooler should be isolated to prevent dilution of the sump.
3. Once containment pressure turns around and decreases to 4 psig, the sprays can be shut off by bypassing the ES channel, stopping the pumps, and closing the injection valves. Coolers should be kept running.

Containment isolation may also be actuated (non-essential at low RC pressure of 1500 psig, remainder at high RB pressure of 4 psig). Two general actions are desired:

1. <u>All</u> penetrations not necessary for core or containment cooling that connect to the containment atmosphere or to the RCS should be checked to make sure they are closed.

2. Containment isolation (at RB pressure of 4 psig) will close off cooling water to the RC Pumps. They should be tripped to prevent damage (per the RC Pump trip criteria in the Best Methods chapter) if cooling water is not rapidly restored.

<u>NOTE</u>: LOCA blowdown forces may have broken cooling water lines near the RC Pumps; usually only a big LOCA will have enough force to do this and the RC Pumps will not be needed later. But caution should be given to reopening the lines; the component cooling water surge tank level should be watched when re-establishing flow to the RCP cooling jacket and seal coolers. If the surge tank level drops abruptly, the lines are leaking and must be reisolated to prevent loss of cooling pump NPSH. No surge tank or flow indications are available for LPSW to the RCP air coolers and bearing coolers. However, a rupture of these lines <u>may</u> cause a large reduction in flow through the RB coolers which can be monitored.

Large break LOCA's will cool the RC system down fast and draw heat from the steam generators. The rapid heat removal may cause enough steam pressure loss to result in a trip of the MFW pumps and subsequent initiation of EFW. For a large break the steam generators are not needed, but EFW should be allowed to go to 50% automatically.

Long Term Cooling

Long-term cooling is defined as the time after a LOCA where the Decay Heat Removal System, either in a normal or emergency mode (LPI), is operating and can be used for core heat removal. The duration of long-term cooling is the period between the onset of long-term and the end of core cooling requirements by the ECC system. The end of core cooling requirements is the time when the core is removed from the reactor vessel or other permanent means are used for core heat removal. The exact duration of long-term cooling will vary depending on several factors, including the size of the break and the radiation release. For the worst case LOCA (i.e., a large break), the duration of the long-term cooling period may vary from one month to a maximum of one year depending upon the resulting accident consequences. For large breaks, long-term cooling could begin as soon as 30 minutes after the break occurs. For small breaks, long-term cooling conditions may take days to achieve depending on how fast the plant can be cooled and depressurized.

Table F-5 presents a summary of actions required to establish and maintain long-term cooling following a loss-of-coolant accident. Table F-5 includes both core cooling and containment related actions.

To Establish Long-Term Cooling

To establish long-term cooling after a small break, the decay heat removal system can be aligned with one train in the decay heat removal mode (sub-cooled only) and the other train in the LPI mode (possibly with HPI piggy-back from the sump or from the BWST). If the system is saturated, recirculation from the sump in the LPI mode (possibly with HPI piggyback) establishes long-term cooling. To establish long-term cooling for large breaks, the LPI system is placed in the recirculation mode from the sump.

Boron Precipitation

Within twenty-four hours after a LOCA, actions should be taken to preclude the possibility of boron precipitation. If the RCS is subcooled and one DHR train is successfully operating in the normal decay heat removal mode, then no further actions are required to prevent boron precipitation. However, if the RCS is not subcooled, other actions will be required. With a large hole in certain areas of the RCS, the reactor can, acting like a distiller, boil off almost pure steam and leave impurities (mostly boron) to concentrate in the vessel. If enough boron accumulates, core flow blockage might occur. To limit the boron concentration, the following steps should be taken:

1. Open the series dump-to-sump valves located upstream of the DHR drop-line isolation valves. Verify flow through this dump line.
2. If Step 1 is not successful, open the DH drop line isolation valves and the alternate sump flow path (LP-105 on Unit 1; backflow through 'B' train sump suction line for Units 2 & 3).

3. If Step 2 is not successful, open the auxiliary spray to the pressurizer. This will route relatively diluted injection water to the area above the core. The flow path is through the auxiliary spray line into the pressurizer, out of the pressurizer through the surge line into the hot leg and then into the vessel.

Preserving the LPI System for Long Term

If a large break LOCA has occurred, or if a small break LOCA has occurred that is in a location that does not permit use of the decay heat system in the normal mode (for core cooling or prevention of boron precipitation), it may be desirable to take one decay heat pump (in addition to the standby pump) out of service to preserve it for the future (so it can be placed back in use if the operating train develops problems). One decay heat pump can be removed from service safely if sump recirculation is in progress and the LPI flow in each train (from the remaining pump through the cross-connect line) is equal to or greater than 1000 gpm.

Hydrogen Purge

Following a large break, hydrogen can accumulate within the containment. The potential sources of hydrogen inside the containment are:

1. Radiolytic decomposition of water within the sump and reactor core
2. Metal-water reaction of the fuel pin cladding
3. Dissolved hydrogen within the reactor coolant
4. Corrosion of aluminum

Because hydrogen can build up, samples of the containment atmosphere should be drawn and analyzed once the ECCS has been aligned for long-term cooling. If the hydrogen concentration exceeds 3.5%, the hydrogen purge system should be manually started. Under worst case conditions, operation of the hydrogen purge system should not be required for more than 10 days after the LOCA occurs. This action is important, when required, because it prevents the accumulation of a large concentration of explosive hydrogen.

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Table F-5 SUMMARY OF LONG-TERM COOLING ACTIONS

ACTIONS TO MAINTAIN CORE COOLING

1. ECCS Alignments

   a. For saturated reactor coolant conditions (RCS pressure ≤ 150 psig):

      - When BWST reaches low level limit, transfer suction of LPI pumps to containment sump.
      - If LPI flow in each train is ≥ 1000 gpm, HPI can be stopped. If LPI flow in each train can be maintained ≥ 1000 gpm with one LPI pump, the other LPI pump can be stopped to preserve it for future use.
      - If neither or only one LPI train flow is > 1000 gpm, run two HPI pumps in the piggyback mode.

   b. For subcooled reactor coolant conditions (RCS temperature ≤ 280F):

      - If both LPI pumps are operative, place one LPI train in normal decay heat removal mode. Cool down to 100F with decay heat coolers.
      - If only one LPI pump is operative, maintain SG cooling. Control RCS inventory using HPI with suction from BWST or in piggyback mode. Start normal decay heat removal when second LPI pump becomes available.

2. Start Action within 24 Hours to prevent boron precipitation following large breaks.

   a. Try to establish normal decay heat removal if two LPI pumps are available, or
   b. Establish gravity draining from the hot leg to the reactor emergency sump, or
   c. Start the auxiliary spray to the pressurizer.

CONTAINMENT ENVIRONMENT ACTION

1. Containment Sprays

   a. On low BWST level, switch suction of spray pumps to RB sump (switched when LPI pump suctions are switched).
   b. Once containment pressure turns around and decreases below 4 psig, the sprays can be shut off.

   NOTE: Emergency coolers should be kept running.

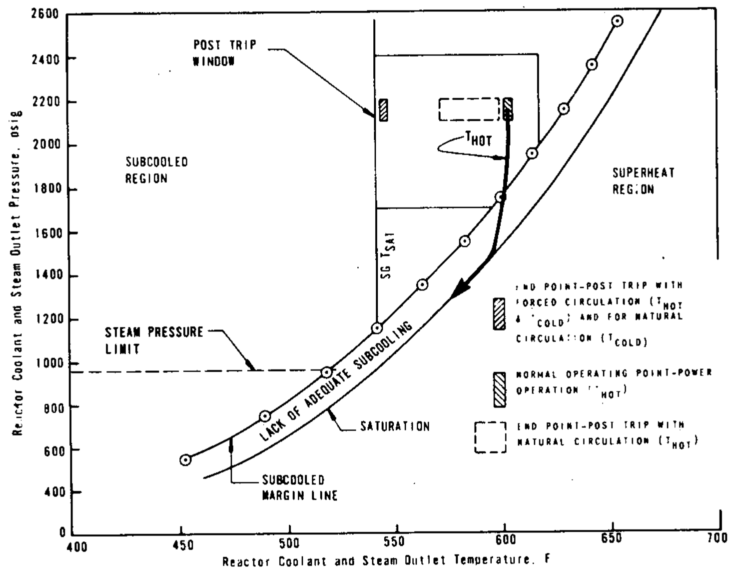
2. Hydrogen Purge

   a. Draw samples of containment atmosphere and monitor hydrogen concentration.
   b. Start hydrogen purge system when hydrogen concentration exceeds 3.5%.

Figure F-12 POST-LOCA CORRECTIVE ACTION FOR LACK OF ADEQUATE SUBCOOLED MARGIN

LACK OF ADEQUATE SUBCOOLED MARGIN

EXPECTED POST TRIP RESPONSE FOR LOCA WITH LOSS OF SUBCOOLING



*1. TRIP RC PUMPS

*2. CONFIRM 2-HPI ON AT MAX. CAPACITY & BALANCE FLOWS

*3 CONFIRM FW FLOWING THROUGH UPPER NOZZLES AND RAISE LEVEL TO 95%

4. ATTEMPT TO LOCATE AND ISOLATE THE LOCA (SEE TABLE F-4)

SUBCOOLED MARGIN?

YES	NO

YES: THROTTLE HPI (SEE BEST METHODS FOR EQUIPMENT OPERATION)

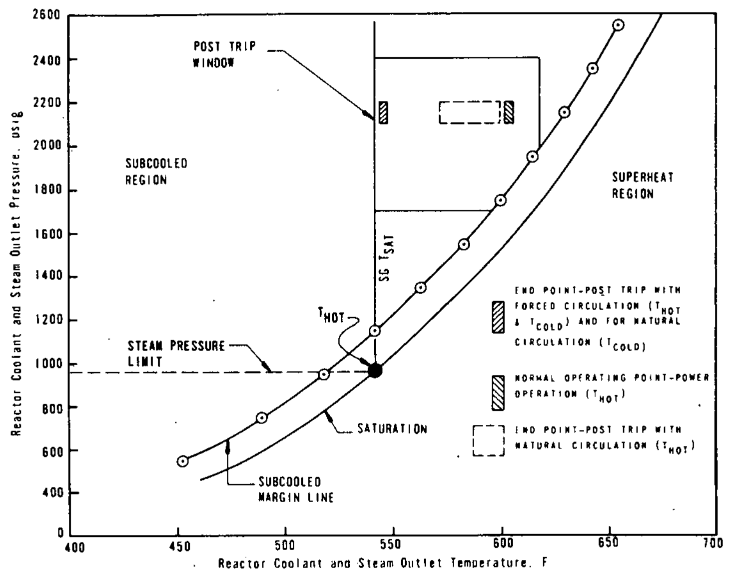
ABNORMAL SYMPTOMS? (OVERHEATING OR OVERCOOLING)

NO	YES

NO: GO TO PLANT COOLDOWN PROCEDURE FOR SUBCOOLED CONDITIONS WITH A SMALL BREAK

YES: TREAT OVERCOOLING OR OVERHEATING PER PROCEDURE (SEE FIGURES 22 OR F-13)

EXPECTED POST LOCA P-T CONDITION WITH SUSTAINED SATURATION & PRIMARY TO SECONDARY HEAT TRANSFER



NOTE: 1. T_{COLD} COULD BE EQUAL TO T_{HOT} OR SLIGHTLY SUBCOOLED.

2. SEE FIGURE F-6 FOR A TYPICAL LOCA EXAMPLE.

NO (SUBCOOLED MARGIN): HEAT TRANSFER IN BOTH SG?

YES	NO

YES: GO TO PLANT COOLDOWN PROCEDURE FOR SATURATED CONDITION WITH A SMALL BREAK

NO: GO TO LACK OF PRIMARY TO SECONDARY HT. TRANSFER (GO TO FIGURE F-13)

*FOR MORE DETAILS, GO TO ATOG PART II BEST METHODS OF EQUIPMENT OPERATION.

Table F-4 SYMPTOMS FOR LOCA'S THAT CAN BE LOCATED OR ISOLATED

THIS CHART WILL AID IN LOCATING SOME BREAKS; ALL BREAKS CANNOT BE LOCATED. SOME BREAKS WHICH CAN BE LOCATED CAN ALSO BE ISOLATED AND THE LOCA CAN BE STOPPED. IT MAY BE DIFFICULT TO DISTINGUISH SMALL STEAM LINE LEAKS INSIDE CONTAINMENT FROM LOCA'S; BUILDING ENVIRONMENT WILL CHANGE FOR BOTH AND THE STEAM PRESSURE WILL NOT ALWAYS BE LOW. HOWEVER, A LOCA WILL CHANGE BUILDING RADIATION LEVELS.

SYMPTOMS FOR LOCA'S THAT CAN BE ISOLATED

(Symptoms or alarms most likely to show location are underlined)

FAILURE	LOCATING SYMPTOMS	ISOLATING HARDWARE
Makeup and purification system outside containment and letdown coolers	- <u>Low letdown storage tank level</u> - <u>High component cooling water surge tank level</u> (for breaks in letdown cooler) - Local sump levels, radiation alarms - High CC discharge temperature from letdown coolers	Letdown valve **1) upstream of coolers
Seal return line and seal return cooler outside containment	- <u>Low letdown storage tank level</u> - <u>High RCW radiation</u> - <u>High RCW surge tank level</u> (for breaks in seal return cooler) - Local sump levels, radiation alarms - High seal return flow - High RCW seal return cooler discharge temperature (local)	Seal return**1) isolation valve
Pressurizer electromatic relief valve	- <u>Flow Monitor Alarm</u> - <u>High quench tank level</u> - <u>High quench tank temperature</u> (These will only be good when the quench tank rupture disk is good)	PORV isolation valve
Makeup-letdown imbalance (this is not a break, but is a loss of coolant)	- <u>High letdown storage tank level</u> - Bleed holdup tank level - Makeup flow rate (+) seal injection flow (-) letdown flow	Letdown control**1) valve
Decay heat removal line break outside containment (decay heat removal system in operation-plant is cooled down)	- <u>High or low decay heat removal flow</u> - <u>Low pump suction press.</u> - <u>Local sump and local</u> radiation alarms	Decay heat letdown** drop line valve
Decay heat cooler tube leak (decay heat removal sys. in operation- plant is cooled down)	- <u>High LPSW temperature at DH cooler outlet.</u>	Cooler isolation valves

SYMPTOMS FOR LOCA'S THAT CANNOT BE ISOLATED

(Symptoms or alarms most likely to show location are underlined)

FAILURE	LOCATING SYMPTOMS
Steam Generator Tube(s)	- <u>High steam line radiation</u> - <u>High steam generator level</u> - High condenser radiation
Pressurizer Safety Valves	- <u>Flow Monitor Alarm</u> - <u>High quench tank level</u> - <u>High quench tank temperature</u> (These will only be good while the quench tank rupture disk is good)
HPI Injection Line Break	- <u>Flow imbalance between injection**3) lines</u> (High flow will be through broken line)
RC Pump Seal Failure	- <u>High seal return temperature ($\sim 350^\circ F$)</u> combined with: <u>Low stage and upper stage pressures are equal and high</u>
RCS Instrumentation Lines - Pressurizer Level - Pressures - RC Flow	 - <u>False low level reading</u> - <u>False low pressure</u> - <u>False high or low flow compared with known pump operation</u>

**Footnotes:

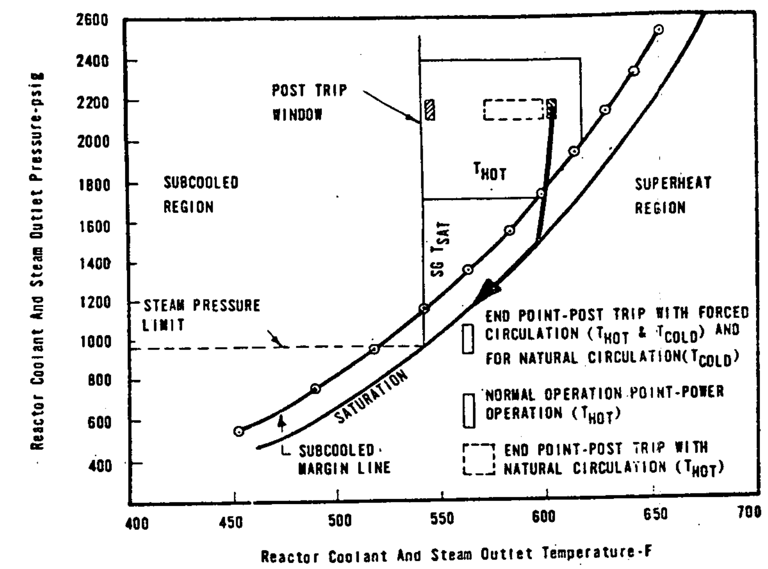
1) Do not allow letdown storage tank to drain or operating makeup pump will lose suction and fail.

2) Inadequate Core Cooling Guidelines for loss of decay heat removal should be implemented.

3) Break cannot be isolated to prevent either loss of reactor coolant or loss of injection water, but the orifice will limit the HPI flow out the break. Balancing the two main injection lines for maximum flow, which is done after any HPI actuation, will ensure adequate pumped flow enters the core. It should be noted that with three HPI pumps started automatically, Train A flow will be 40-50% higher than Train B flow regardless of RCS pressure.

Table F-3 SUMMARY OF GENERAL LOCA SYMPTOMS

A. INITIAL P-T DIAGRAM CHARACTERISTICS



a. Rapid system depressurization to saturated reactor coolant conditions.

b. Sustained saturation (i.e., HPI does not return the reactor coolant to a subcooled state within 5-10 minutes).

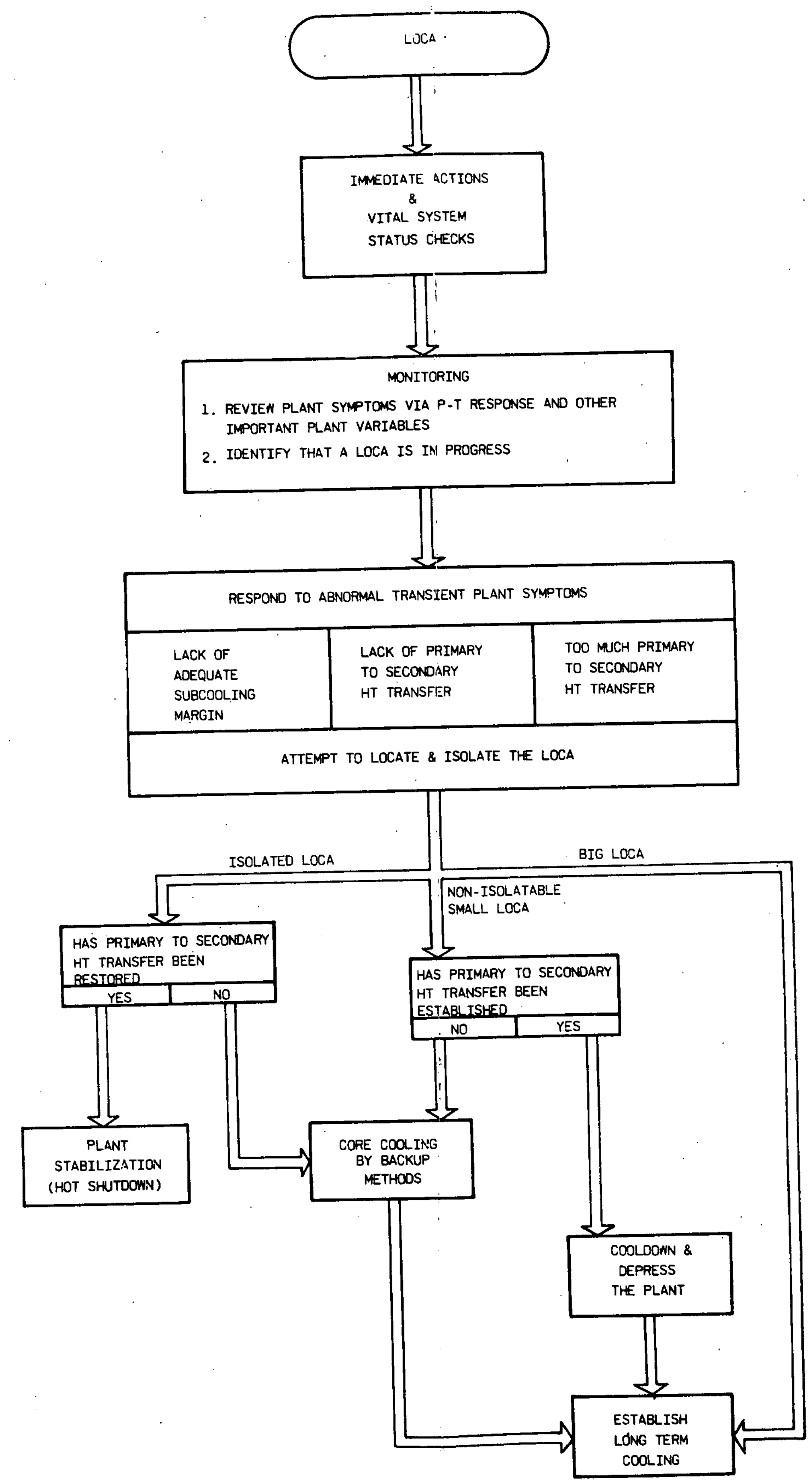
B. POSSIBLE EVENTS OR OTHER PLANT LEVEL SYMPTOMS (EXTRACTED FROM LOCA CONCEPTS SECTION)

	SMALL LEAK	SMALL BREAK Small	SMALL BREAK Large	LARGE BREAK
1. Excessive Makeup	X	X	-	-
2. Decreasing Pressurizer Level and Pressure	X	X	X	X
3. Reactor Trip	X	X	X	X
4. ES Actuation (Low RC Pressure)		X	X	X
5. Loss of Subcooled Margin		X	X	X
6. Lack of Primary to Secondary Heat Transfer (System Repressurization along Saturation Curve)		X		
7. Reverse Primary to Secondary Heat Transfer			X	X
8. Rapid Depletion of CFTs				X
9. Rapid Drop in RCS Pressure to where LPI System becomes Operative				X
10. Rapid Increase in RB Pressure and Temperature*			X	X
11. Increasing RB Radiation Levels*	X	X	X	X
12. Inadequate Core Cooling Symptoms**				X

*Degraded containments conditions can occur for other events such as steam or feedwater line breaks inside containment. These non-LOCA events would not have high containment radiation levels. High containment radiation levels are thus a good indicator that a LOCA is in progress.

**For large breaks, the core exit thermocouples can indicate superheated coolant conditions from approximately the end of blowdown to up to 10 minutes into the reflood portion of the event. This is an expected condition. Because the RC pressure is reduced to where the LPI system is fully operational, actions specified for ICC are not required. The core's temperature excursion will be terminated when CFT and LPI water reflood the reactor vessel.

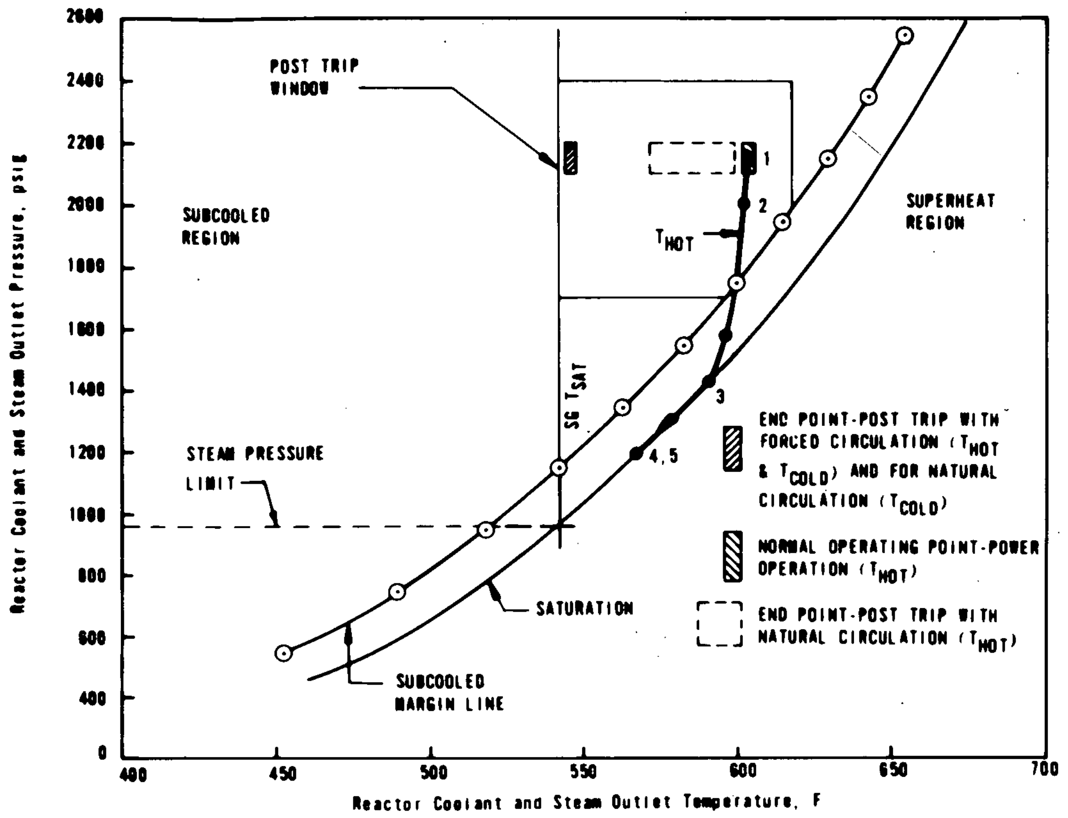
Table F-1 GENERAL POST-LOCA CORRECTIVE ACTION TO MAINTAIN CORE COOLING

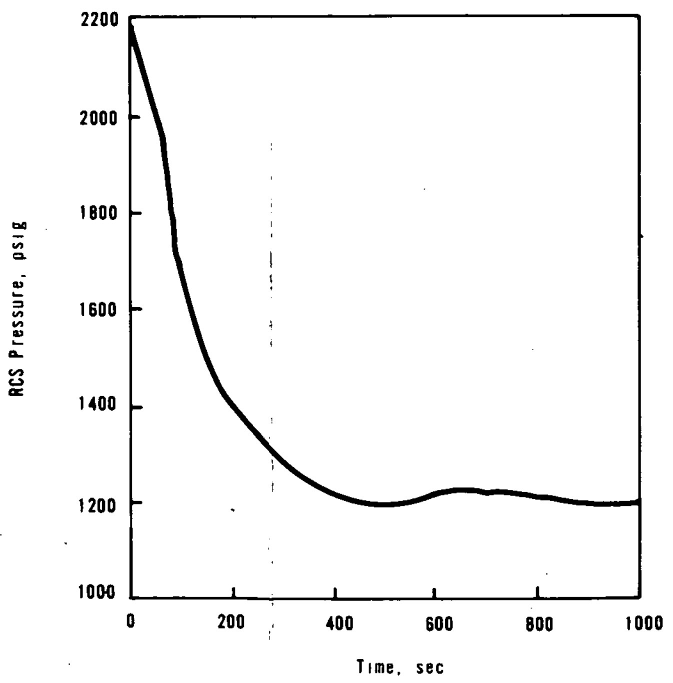


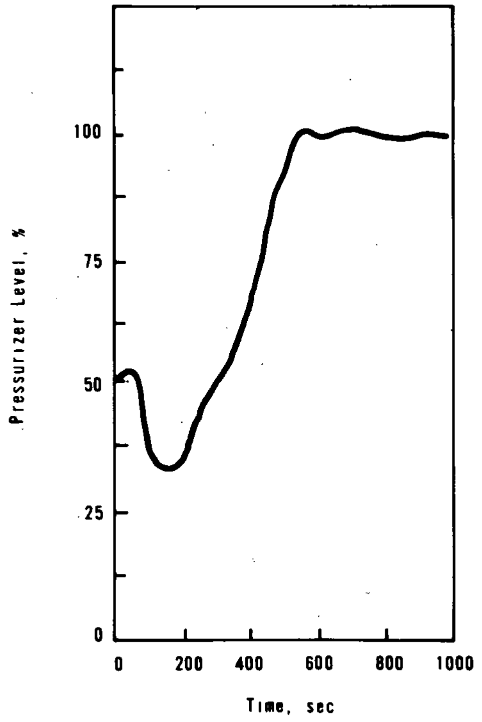
STUCK OPEN PORV

REFERENCE POINTS	TIME (SECONDS)	REMARKS
1	0	LOCA OCCURS WITH REACTOR INITIALLY AT POWER; STUCK OPEN PORV IS EQUIVALENT TO ABOUT A ONE-INCH DIAMETER HOLE.
1-2	0-60	RCS PRESSURE DROPS AS THE PRESSURIZER STEAM SPACE IS VENTED OUT THE BREAK. PRESSURIZER LEVEL INCREASES.
2	60	PRESSURE DROPS TO THE RPS TRIP SETPOINT; THE REACTOR TRIPS.
2-3	60-185	AS RCS PRESSURE DROPS, THE REACTOR COOLANT SUBCOOLING MARGIN IS LOST (RC PUMP TRIP IS REQUIRED), AND ES IS ACTUATED ON LOW RCS PRESSURE.
3	185	THE HOT LEG SATURATES.
3-4	185-490	RCS IS IN TWO-PHASE NATURAL CIRCULATION. PRESSURE AND TEMPERATURE CONDITIONS DECREASE ALONG THE SATURATION CURVE AND STABILIZE AT ABOUT 1200 PSI. PRESSURIZER LEVELS ARE INCREASING AS THE STEAM SPACE IS DEPLETED.
4	490	THE PRESSURIZER FILLS.
4-5	>490	SYSTEM STABILIZES AT ABOUT 1200 PSI, AND TWO-PHASE NATURAL CIRCULATION IS NEVER LOST. HPI WILL EVENTUALLY RETURN THE SYSTEM TO A SUBCOOLED STATE. WITHOUT ISOLATION OF THE PORV, A SOLID WATER COOLDOWN WOULD BE REQUIRED.

Figure F-11 SYSTEM RESPONSE FOR SMALL BREAK WITHIN PRESSURIZER STEAM SPACE



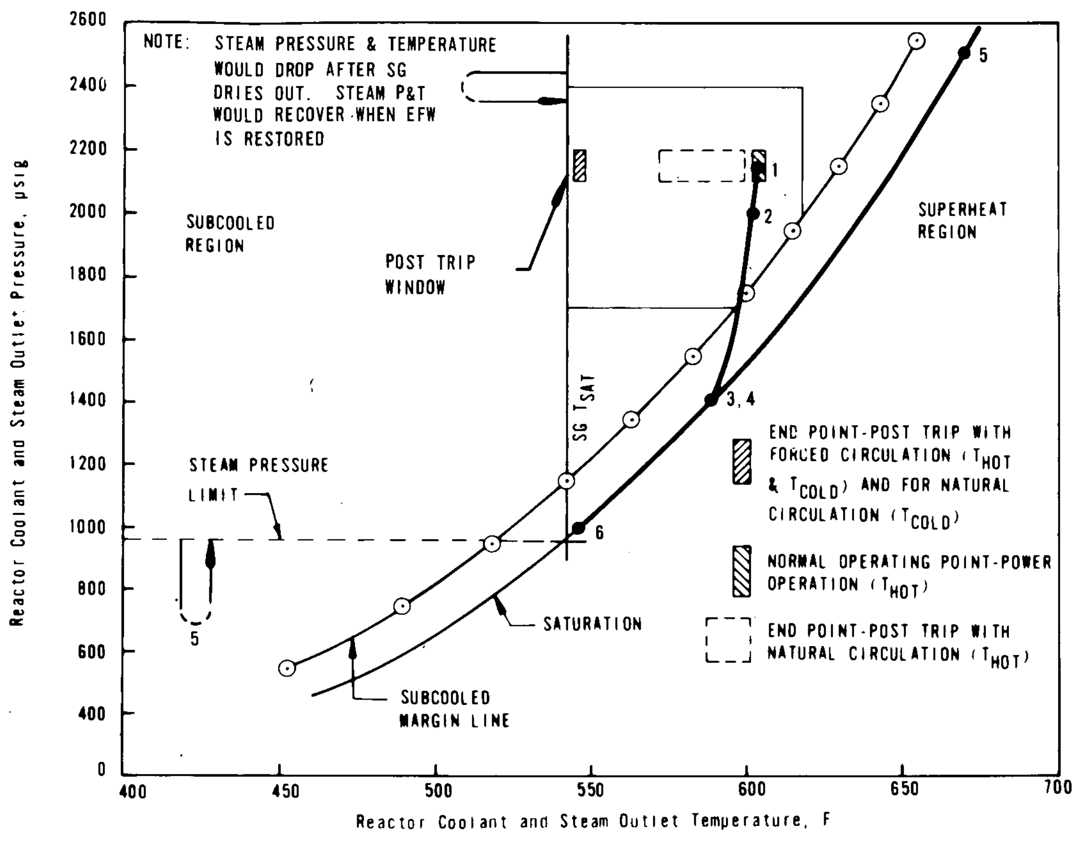


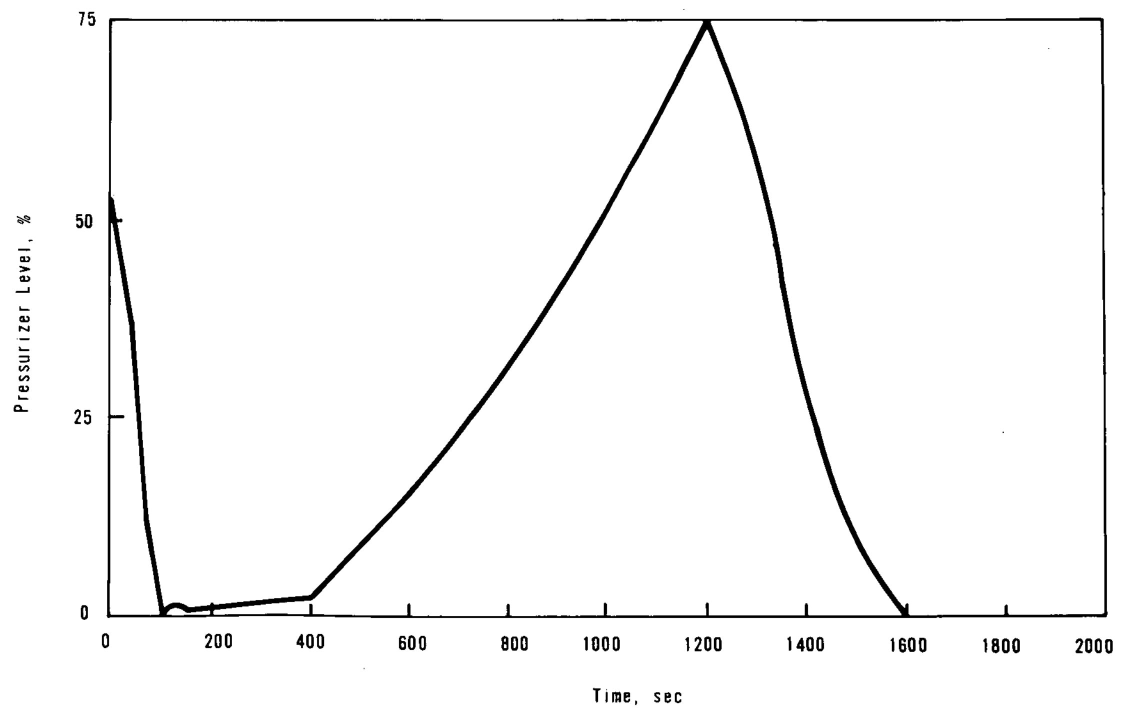


0.01 FT^2 BREAK AT PUMP DISCHARGE
(BOTH HPI AND EFW ARE ASSUMED TO
BE DELAYED FOR 20 MINUTES)

REFERENCE POINT	TIME (SECONDS)	REMARKS
1	0.0	LOCA OCCURS WITH REACTOR INITIALLY AT POWER; BREAK IS EQUIVALENT TO 1.35 INCH DIAMETER HOLE.
1-2	0-50	RCS PRESSURE AND PRESSURIZER LEVEL DROP DUE TO THE LOSS OF REACTOR COOLANT.
2	50	RCS PRESSURE DROPS TO RPS TRIP SETPOINT; THE REACTOR TRIPS. LOSS OF OFFSITE POWER OCCURS, AND THE RC PUMPS AND MAIN FEED ARE LOST. EFW FAILS TO START.
2-3	50-120	AS RCS PRESSURE DROPS, THE REACTOR COOLANT SUBCOOLING MARGIN IS LOST AND ES IS ACTUATED ON LOW RCS PRESSURE. HPI IS ASSUMED TO FAIL.
3	120	HOT LEG SATURATES.
3-4	120-280	SG'S SLOWLY BOIL DRY.
4	280	SG COOLING IS ESSENTIALLY LOST.
4-5	280-1200	RCS REPRESSURIZES DUE TO LACK OF PRIMARY TO SECONDARY HEAT TRANSFER AND THE BREAK'S INABILITY TO REMOVE ENOUGH ENERGY (I.E., BREAK IS TOO SMALL TO KEEP THE SYSTEM PRESSURE DOWN).
5	1200	OPERATOR ACTION TO START EFW AND HPI IS ASSUMED.
5-6	>1200	REFLUX BOILING IS ESTABLISHED; AND THE RCS SLOWLY DEPRESSURIZES TO THE SECONDARY SIDE SATURATION CONDITIONS.

Figure F-10 SYSTEM RESPONSE FOR SMALL BREAK THAT REPRESSURIZES IF FEEDWATER IS LOST

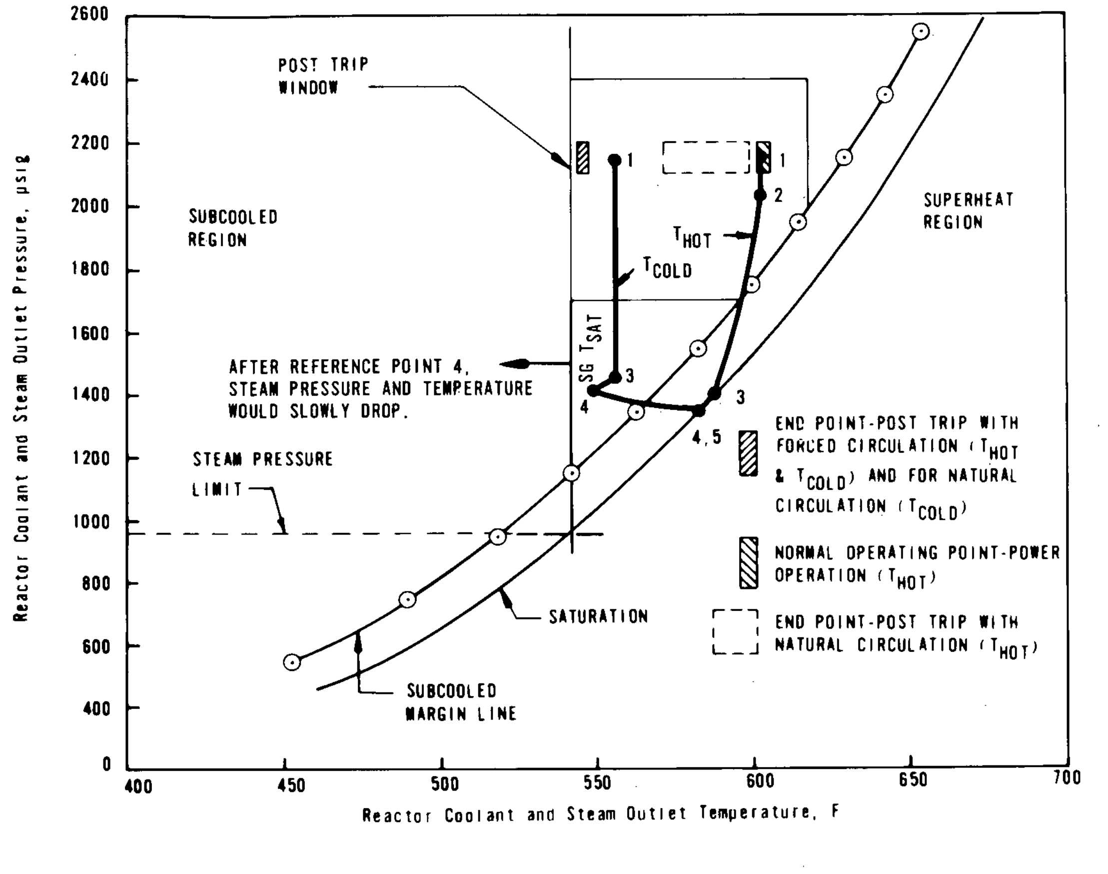


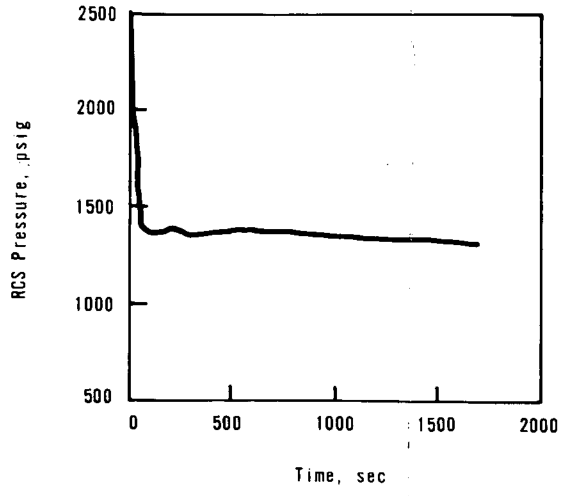


0.02 FT^2 BREAK AT PUMP DISCHARGE (NO FEEDWATER)

REFERENCE POINTS	TIME (SECONDS)	REMARKS
1	0.0	LOCA OCCURS WITH REACTOR INITIALLY AT POWER; BREAK IS EQUIVALENT TO 1.9 INCH DIAMETER HOLE.
1-2	0-29	RCS PRESSURE AND PRESSURIZER LEVEL DROP DUE TO LOSS OF REACTOR COOLANT.
2	29	RCS PRESSURE DROPS TO RPS TRIP SETPOINT; THE REACTOR TRIPS.
2-3	29-60	AS RCS PRESSURE DROPS, THE REACTOR COOLANT SUBCOOLED MARGIN IS LOST; THE RC PUMPS ARE TRIPPED; AND ES IS ACTUATED ON LOW RCS PRESSURE.
3	60	HOT LEG SATURATES.
3-4	60-250	THE RCS STABILIZES IN PRESSURE AND THE STEAM GENERATORS SLOWLY BOIL DRY.
4	250	SG COOLING IS ESSENTIALLY LOST.
5	>250	THE RCS HANGS UP IN PRESSURE AS THE BREAK IS NOT LARGE ENOUGH TO CONTINUALLY DEPRESSURIZE THE SYSTEM. THE CORE IS BEING COOLED BY HPI COOLING. THE PLANT WILL SLOWLY COOLDOWN AND DEPRESSURIZE AS DECAY HEAT DROPS.

Figure F-9 SYSTEM RESPONSE FOR SMALL BREAK THAT STABILIZES AT HIGH RCS PRESSURE WITHOUT FEEDWATER





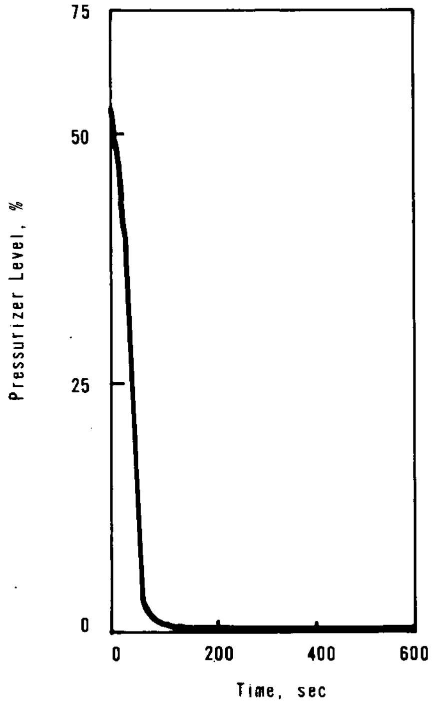
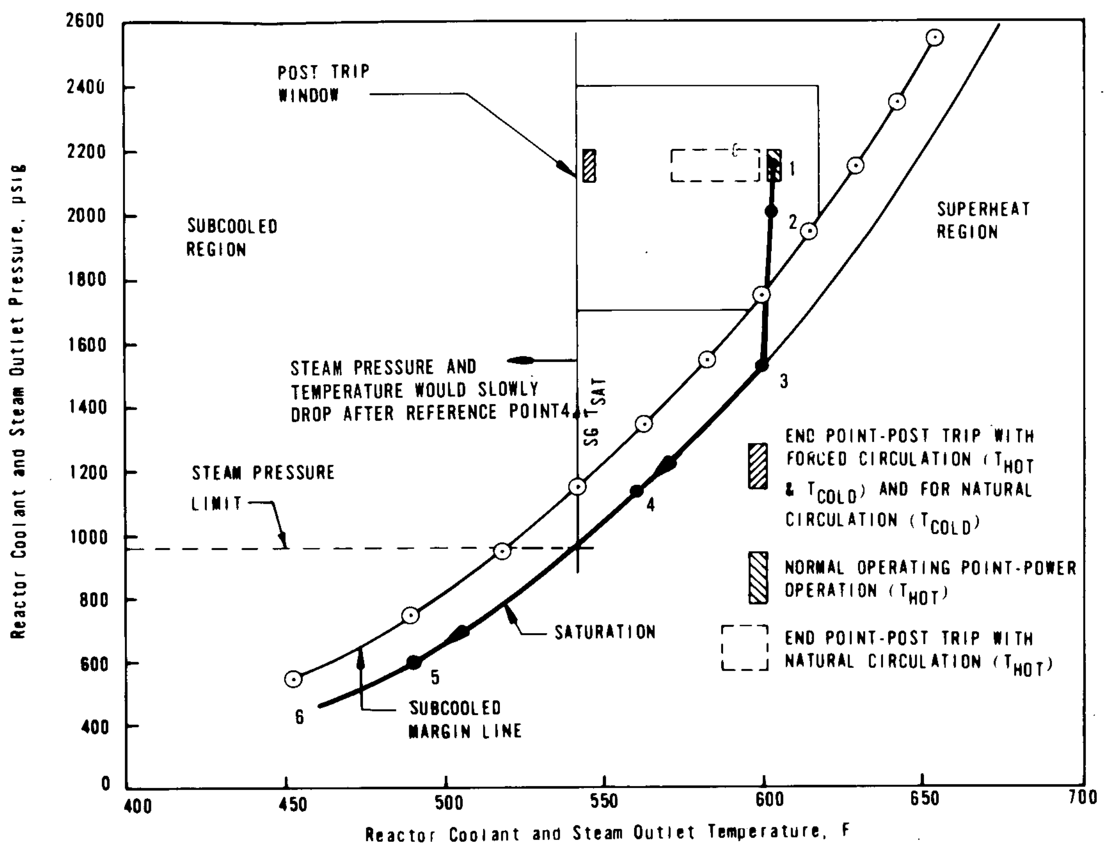
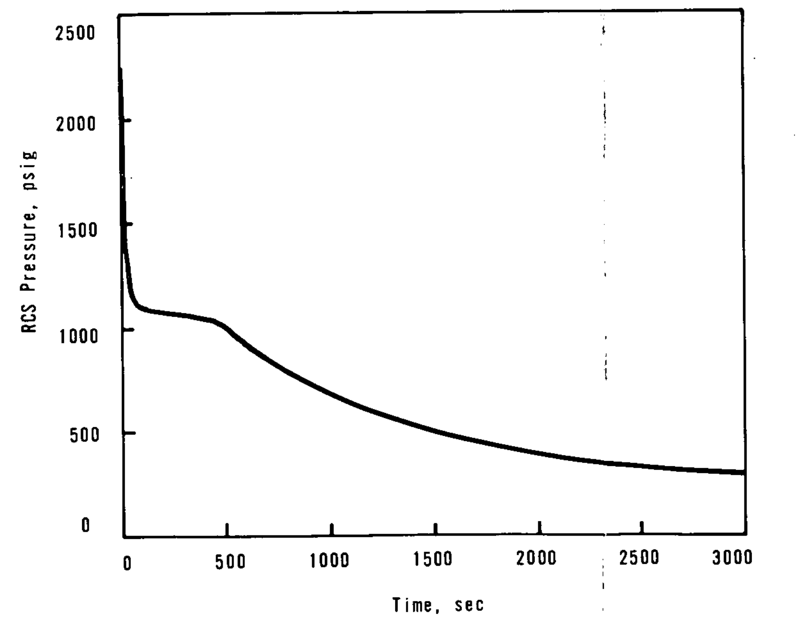


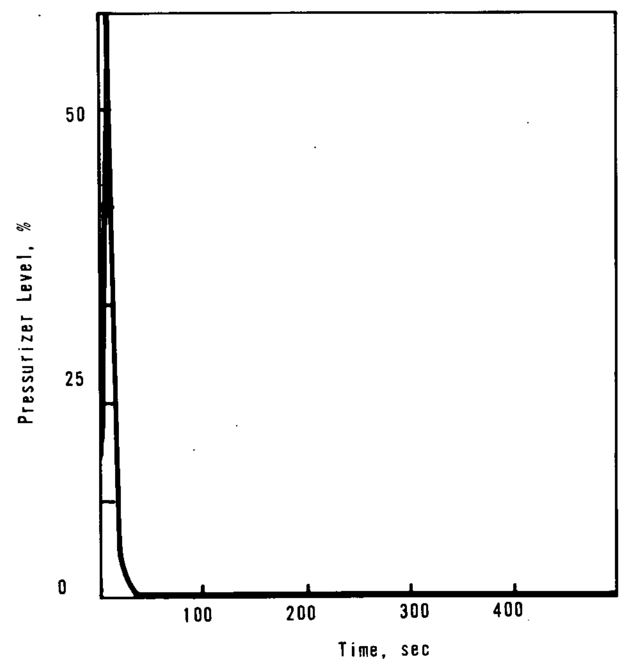
Figure F-8. SYSTEM RESPONSE FOR SMALL BREAK WHICH DEPRESSURIZES THE RCS WITHOUT FEEDWATER

0.07 FT^2 BREAK WITHOUT FW

REFERENCE POINTS	TIME (SECONDS)	REMARKS
1	0	LOCA OCCURS WITH REACTOR INITIALLY AT POWER; BREAK IS EQUIVALENT TO 3.6 INCH DIAMETER HOLE.
1-2	0-10	RCS PRESSURE AND PRESSURIZER LEVEL DROP DUE TO LOSS OF REACTOR COOLANT.
2	10	RCS PRESSURE DROPS TO RPS TRIP SETPOINT; THE REACTOR TRIPS.
2-3	10-24	AS RCS PRESSURE DROPS, THE REACTOR COOLANT SUBCOOLED MARGIN IS LOST (RC PUMP TRIP IS REQUIRED) AND ES IS ACTUATED ON LOW RCS PRESSURE.
3	24	THE HOT LEG SATURATES.
3-4	24-150	THE RCS SLOWLY DEPRESSURIZES; THE RC PUMPS ARE TRIPPED; RESIDUAL FEEDWATER IS SLOWLY BOILED OFF.
4	150	SG COOLING IS ESSENTIALLY LOST.
4-5	150-1200	THE RCS CONTINUES TO SLOWLY DEPRESSURIZE BECAUSE THE BREAK CAN REMOVE DECAY HEAT BY RELEASING REACTOR COOLANT TO THE RB.
5	1200	RCS PRESSURE HAS DROPPED TO 600 PSIG; THE CFT BEGINS TO ADD WATER TO THE REACTOR VESSEL.
5-6	>1200	THE RCS CONTINUES TO DEPRESSURIZE WITH HPI COOLING IN PROGRESS.



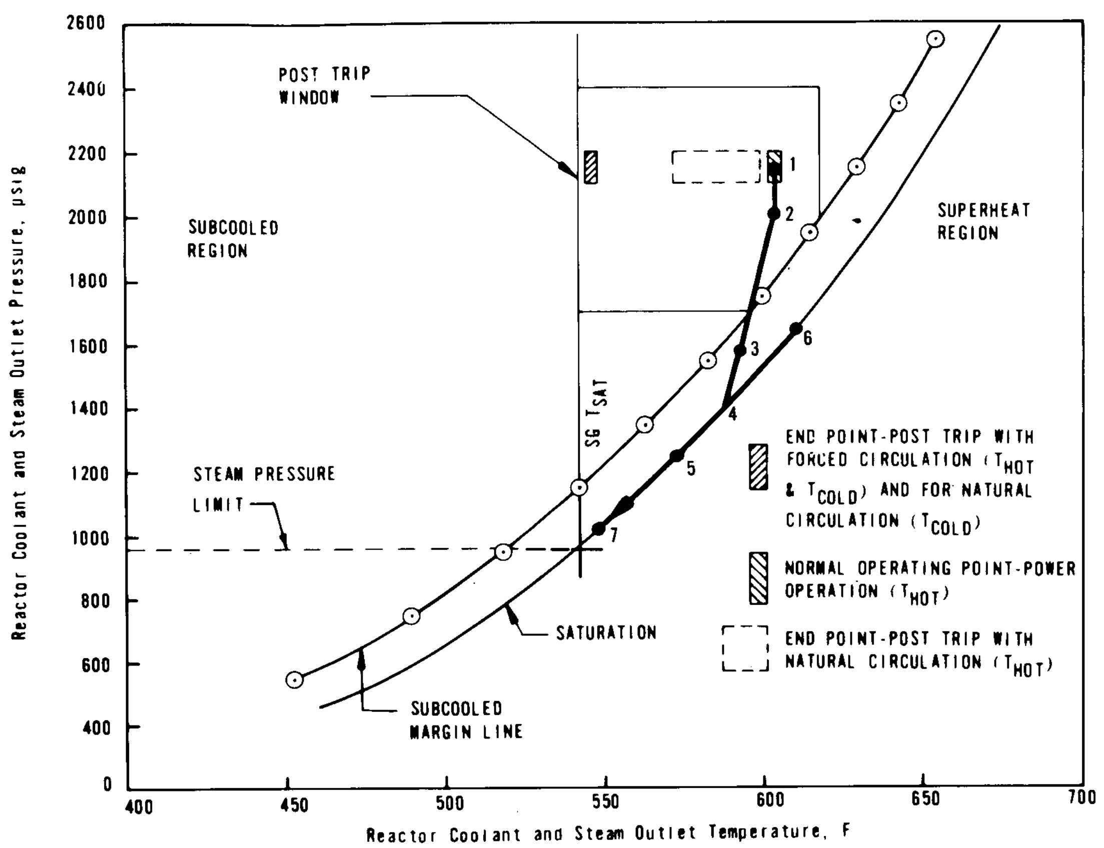


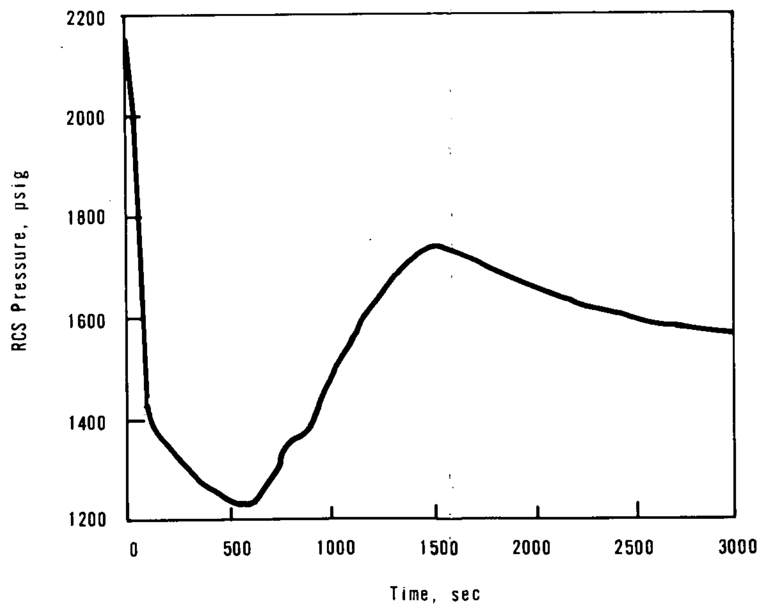


$0.01\ FT^2$ BREAK AT PUMP DISCHARGE

REFERENCE POINTS	TIME (SECONDS)	REMARKS
1	0	LOCA OCCURS WITH REACTOR INITIALLY AT POWER; BREAK IS EQUIVALENT TO 1.35 INCH DIAMETER HOLE.
1-2	0-50	RCS PRESSURE AND PRESSURIZER LEVEL DROP DUE TO THE LOSS OF REACTOR COOLANT.
2	50	PRESSURE DROPS TO RPS TRIP SETPOINT; THE REACTOR TRIPS.
2-3	50-80	RCS PRESSURE CONTINUES TO DROP; THE REACTOR COOLANT SUBCOOLING MARGIN IS LOST (PUMP TRIP REQUIRED).
3	80	ES ACTUATION ON LOW RCS PRESSURE.
3-4	80-100	HOT LEG APPROACHES SATURATED CONDITIONS AS THE PRESSURIZER EMPTIES. THE RC PUMPS ARE TRIPPED AND MFW FLOW IS DIVERTED TO THE UPPER NOZZLES.
4	100	THE HOT LEGS ARE SATURATED.
4-5	100-600	RCS IS IN TWO-PHASE NATURAL CIRCULATION.
5	600	THE "FLOW CIRCULATION" PHASE STOPS; A LOSS OF PRIMARY TO SECONDARY HEAT TRANSFER OCCURS BECAUSE STEAM CONDENSATION IN SG TUBES IS NOT POSSIBLE.
5-6	600-1250	RCS REPRESSURIZES BECAUSE NO CORE HEAT CAN BE REMOVED BY STEAM GENERATORS. STEAM BUBBLE IN HOT LEG PIPING IS SLOWLY GROWING IN SIZE. REACTOR COOLANT EXPANDS INTO THE PRESSURIZER AS THE SYSTEM REPRESSURIZES.
6	1500	STEAM CONDENSATION STARTS IN SG TUBES.
6-7	>1500	REFLUX BOILING IS ESTABLISHED; AND THE PRIMARY SYSTEM PRESSURE AND TEMPERATURE SLOWLY DECREASE TO THE SECONDARY SIDE SATURATION CONDITIONS.

Figure F-7 SYSTEM RESPONSE FOR SMALL BREAK WHICH REPRESSURIZES IN A SATURATED STATE





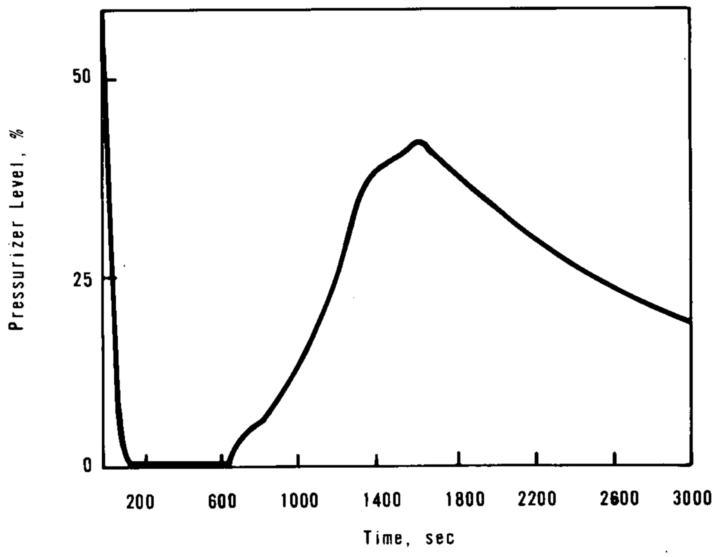
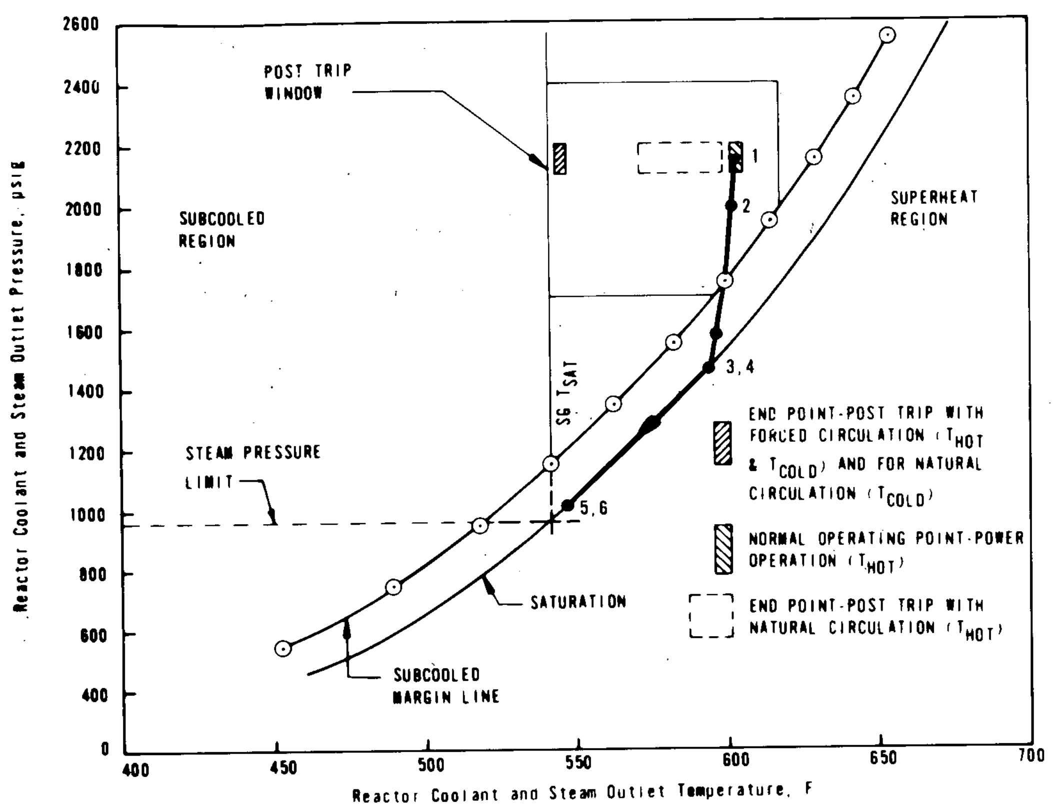


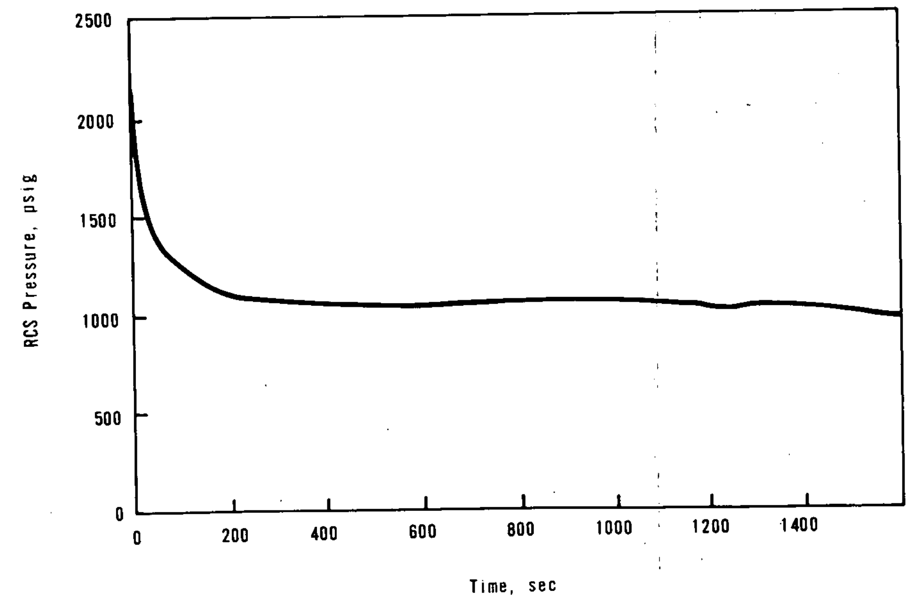
Figure F-6 SYSTEM RESPONSE FOR SMALL BREAK WHICH STABILIZES AT SECONDARY SIDE PRESSURE

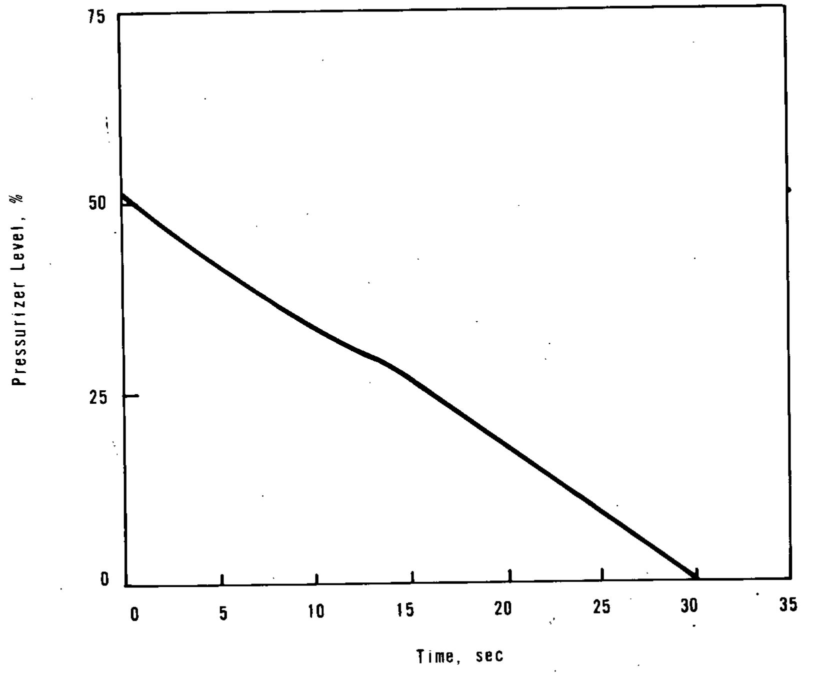
0.04 FT² COLD LEG BREAK

REFERENCE POINTS	TIME (SECONDS)	REMARKS
1	0	LOCA OCCURS WITH REACTOR INITIALLY AT POWER; BREAK IS EQUIVALENT TO 2.7 INCH DIAMETER HOLE.
1-2	0-15	RCS PRESSURE AND PRESSURIZER LEVEL DROP DUE TO THE LOSS OF REACTOR COOLANT.
2	15	PRESSURE DROPS TO RPS SETPOINT; THE REACTOR TRIPS.
2-3	15-23	RCS PRESSURE CONTINUES TO DROP AND SUBCOOLING MARGIN IS LOST. AN RC PUMP TRIP IS REQUIRED.
3	23	ES ACTUATED ON LOW RC PRESSURE.
3-4	23-30	THE PRESSURIZER EMPTIES AND THE HOT LEG CONDITIONS APPROACH SATURATION.
4	30	HOT LEG SATURATES.
4-5	30-340	THE RC PUMPS ARE TRIPPED; MFW FLOW IS DIVERTED TO THE UPPER NOZZLES; AND HEAT CONTINUES TO BE TRANSPORTED FROM THE CORE TO THE SG BY TWO-PHASE NATURAL CIRCULATION. RCS PRESSURE AND TEMPERATURE ARE SLOWLY APPROACHING THE SECONDARY SIDE SATURATION CONDITIONS.
5	340	THE FLOW CIRCULATION PHASE ENDS; THE SYSTEM IS IN A REFLUX BOILING MODE.
6	>340	THE RCS STABILIZES AT OR NEAR SECONDARY SIDE CONDITIONS. OTSG LEVELS SHOULD BE SLOWLY RAISED TO 95% ON OPERATE RANGE. THE RCS WILL REMAIN IN THIS CONDITION FOR A LONG TIME PERIOD AND THEN BEGIN A SLOW COOLDOWN AND DEPRESSURIZATION AS DECAY HEAT DROPS.

NOTE: OPERATOR ACTION TO COOLDOWN THE PLANT SHOULD BE INITIATED TO ACHIEVE A LONG-TERM COOLING MODE.



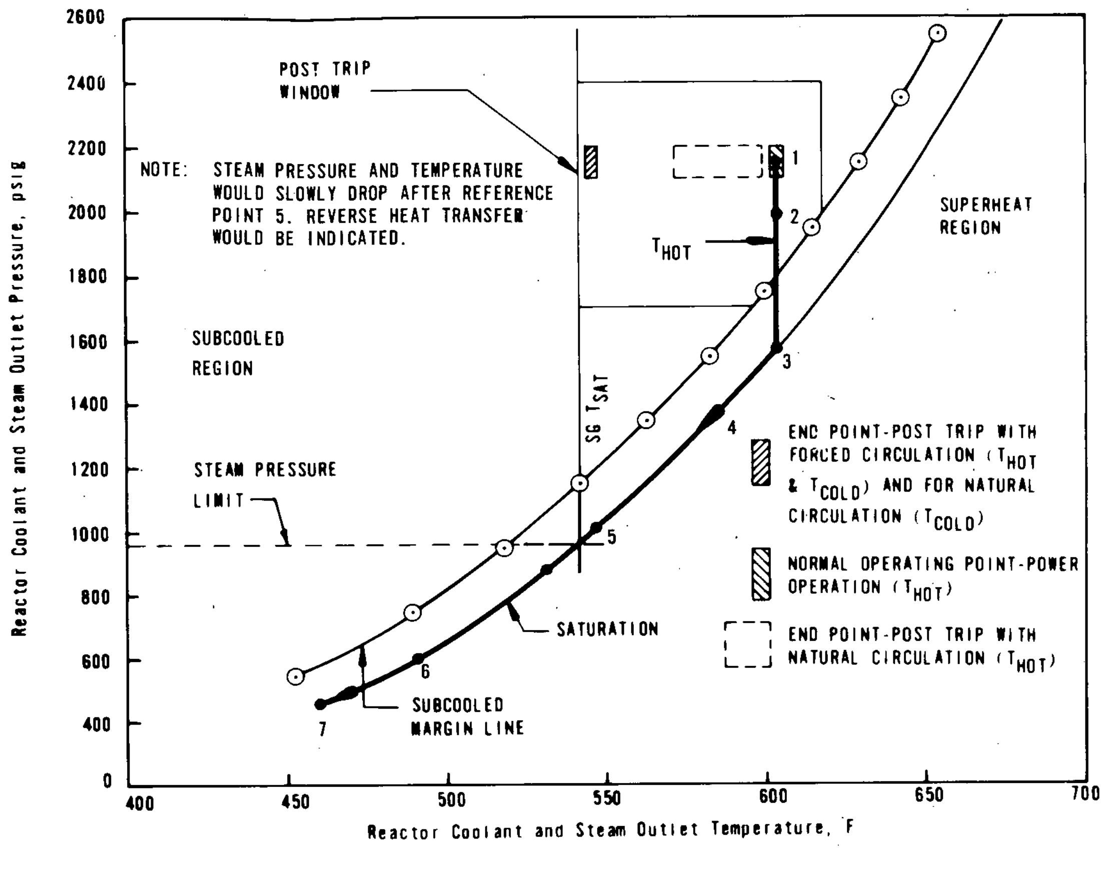


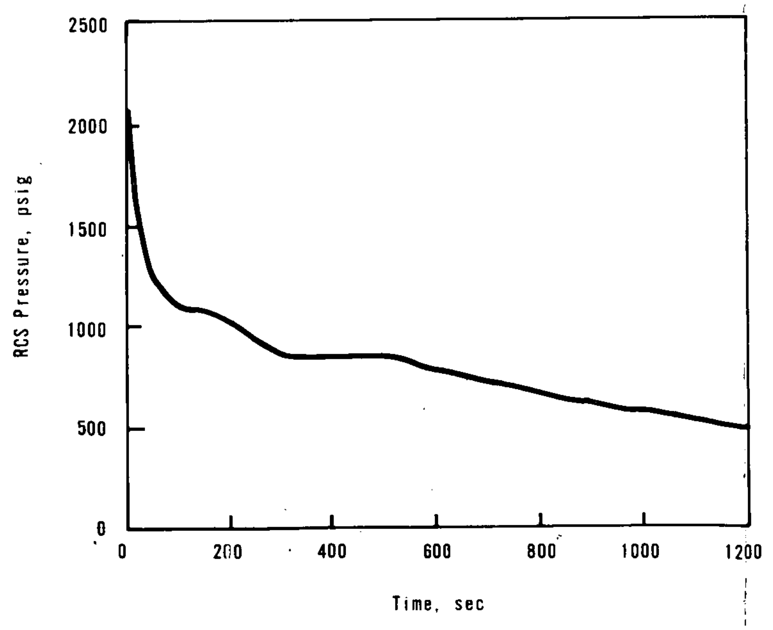


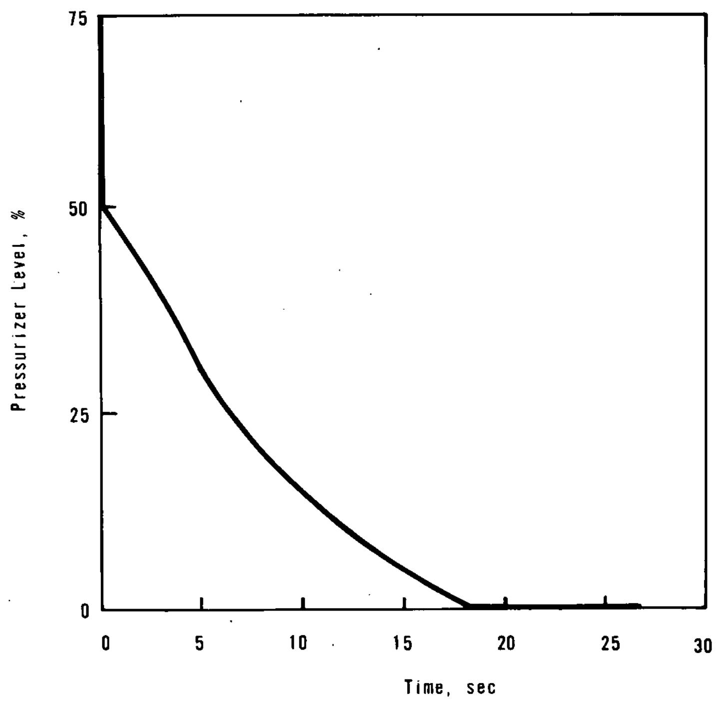
0.1 FT^2 COLD LEG BREAK

REFERENCE POINTS	TIME (SECONDS)	REMARKS
1	0	LOCA OCCURS WITH REACTOR INITIALLY AT POWER. BREAK IS EQUIVALENT TO 4.3 INCH DIAMETER HOLE.
1-2	0-5.0	RCS PRESSURE AND PRESSURIZER LEVEL DROP DUE TO LOSS OF REACTOR COOLANT OUT THE BREAK.
2	5.0	RPS TRIP SETPOINT IS REACHED; REACTOR TRIPS.
2-3	5-18	RCS DEPRESSURIZES TO SATURATED CONDITIONS; ES IS ACTUATED ON LOW RC PRESSURE; AND THE PRESSURIZER EMPTIES.
3	18	PRESSURIZER IS COMPLETELY EMPTY AND THE HOT LEG IS SATURATED.
3-4	18-140	THE RCS CONTINUES TO DEPRESSURIZE. THE RC PUMPS ARE TRIPPED; MFW FLOW IS DIVERTED TO THE UPPER NOZZLES; AND TWO-PHASE NATURAL CIRCULATION EVOLVES.
4	140	THE "FLOW CIRCULATION" PHASE ENDS; THE RCS IS IN REFLUX BOILING.
4-5	140-270	THE RCS CONTINUES TO DEPRESSURIZE. STEAM PRESSURE DROPS DUE TO REDUCED PRIMARY TO SECONDARY HEAT TRANSFER.
5	270	PRIMARY PRESSURE DROPS BELOW THE SECONDARY PRESSURE. THE SG'S ARE NOW A HEAT SOURCE.
5-6	270-910	RCS PRESSURE CONTINUES TO DROP.
6	910	RCS PRESSURE HAS DROPPED TO 600 PSIG AND THE CFT'S BEGIN ADDING WATER TO THE REACTOR VESSEL.
7	>910	THE RCS WILL CONTINUE TO DEPRESSURIZE SLOWLY UNTIL THE LPI SYSTEM BECOMES OPERATIVE.

Figure F-5b SYSTEM RESPONSE FOR SMALL BREAK WHICH CONTINUALLY DEPRESSURIZES THE RCS (0.1 FT^2) COLD LEG BREAK)



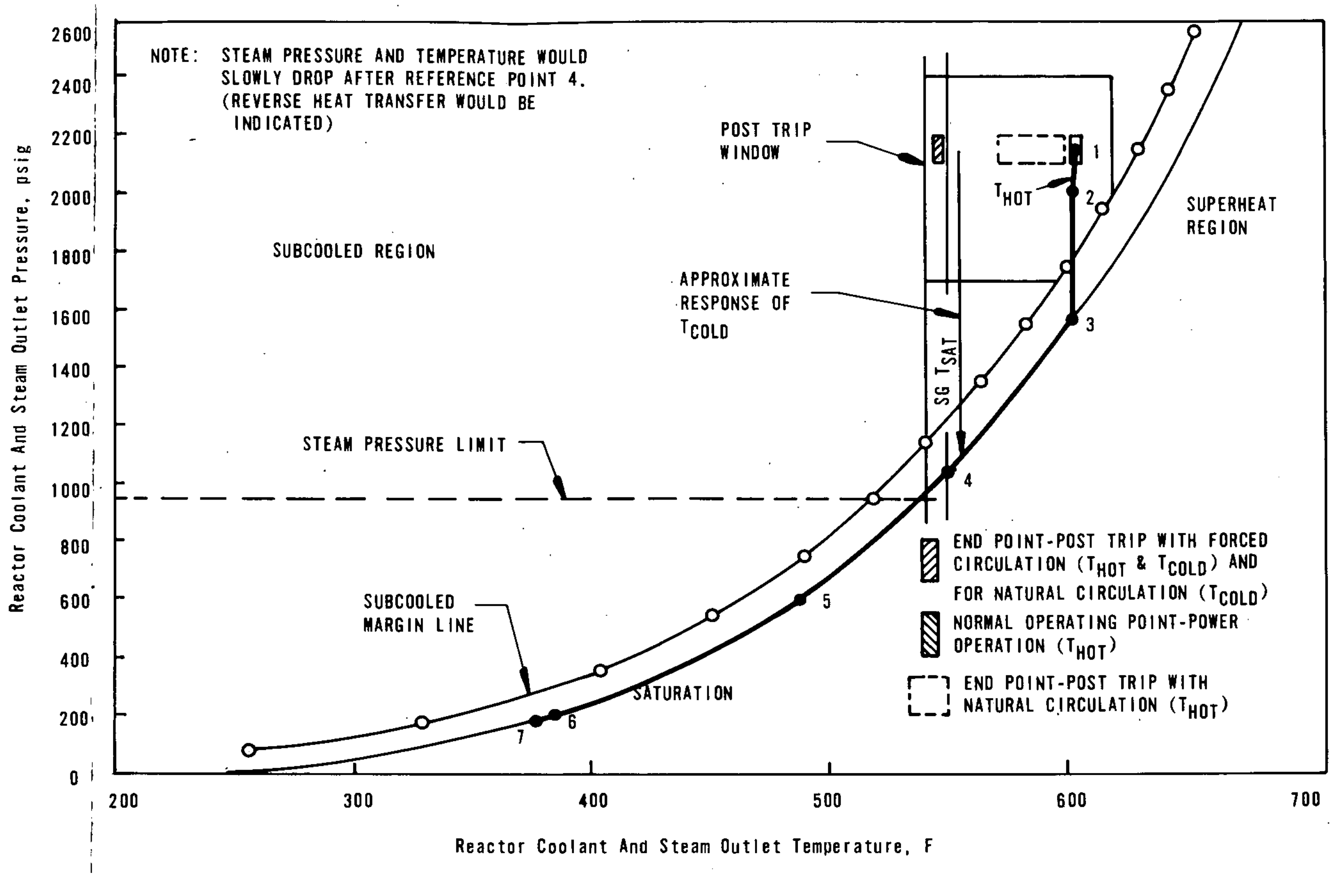


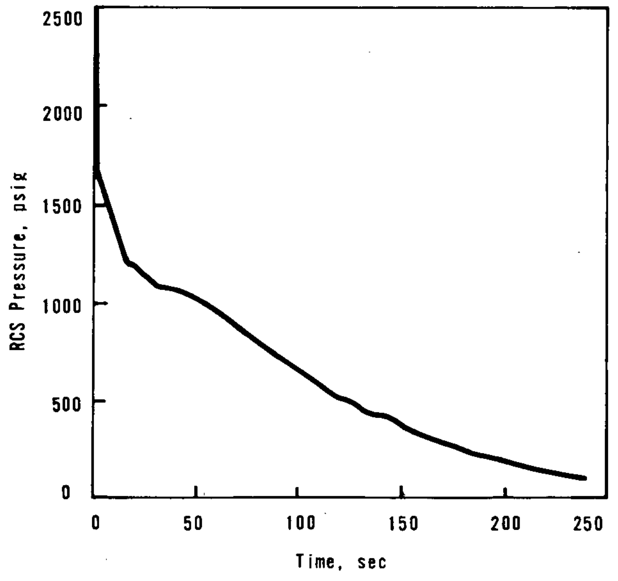


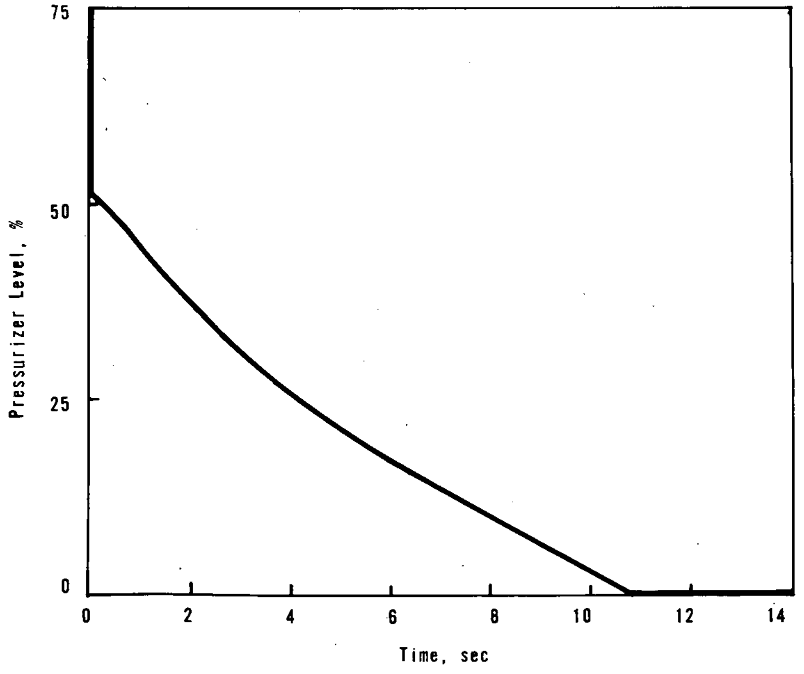
0.5 FT^2 BREAK IN COLD LEG PIPE

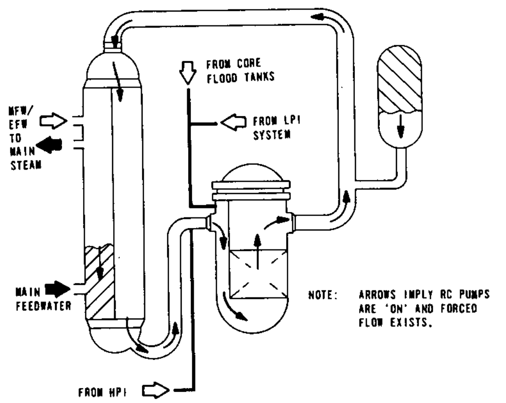
REFERENCE POINTS	TIME (SECONDS)	REMARKS
1	0	LOCA OCCURS WITH REACTOR INITIALLY AT POWER; BREAK IS EQUIVALENT TO 9.5 INCH DIAMETER HOLE.
1-2	0-.2	PRESSURE DROPS DUE TO RELEASE OF REACTOR COOLANT OUT THE BREAK.
2	0.2	RPS TRIP SETPOINT IS REACHED; A REACTOR TRIP IS INITIATED.
2-3	0.2-0.5	BREAK IS LARGE ENOUGH TO DEPRESSURIZE THE RCS TO SATURATED CONDITIONS BEFORE PRESSURIZER EMPTIES. WITHIN 0.5 SEC. ES IS ACTUATED AND THE RC PUMPS SHOULD BE TRIPPED BECAUSE SUBCOOLING IS LOST.
3	0.5	HOT LEG SATURATES.
3-4	0.5-40	RC PUMPS ARE TRIPPED AND MFW FLOW IS DIVERTED TO THE UPPER NOZZLES. THE RCS CONTINUES TO DEPRESSURIZE IN A SATURATED STATE AND THE HPI BEGINS TO DELIVER FLOW TO THE RCS.
4	40	THE "FLOW CIRCULATION" PERIOD ENDS, AND THE PRIMARY SIDE PRESSURE AND TEMPERATURE DROP BELOW THE PRESSURE AND TEMPERATURE OF THE SG's. THE SG's BECOME A HEAT SOURCE.
4-5	40-110	BOTH THE HOT AND COLD LEG ARE SATURATED, AND RC PRESSURE CONTINUES TO DROP.
5	110	RC PRESSURE HAS DROPPED TO 600 PSIG AND THE CFT's BEGIN ADDING WATER TO REACTOR VESSEL.
5-6	110-200	RCS PRESSURE CONTINUES TO DROP. ES ACTUATION AT 500 PSIG STARTS LPI.
6	200	RCS PRESSURE IS BELOW 200 PSIG AND THE LPI PUMPS BEGIN TO DELIVER FLOW TO THE REACTOR VESSEL.
7	>200	THE RCS BEGINS TO REFILL BACK TO THE ELEVATION OF THE BREAK. CORE COOLING WILL BE MAINTAINED SO LONG AS LPI IS CONTINUED.

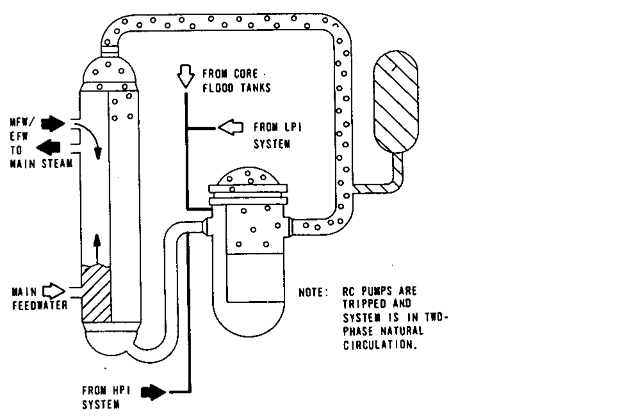
Figure F-5a. SYSTEM RESPONSE FOR SMALL BREAK WHICH CONTINUALLY DEPRESSURIZES THE RCS (0.5 FT^2 BREAK IN COLD LEG PIPE)











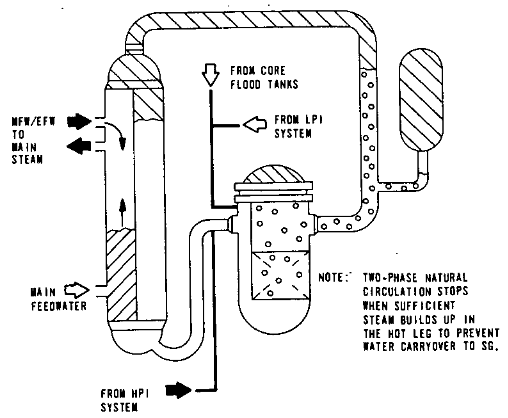


Figure F-4. TYPICAL SYSTEM RESPONSE DURING FLOW CIRCULATION PHASE OF A SMALL BREAK

TYPICAL SYSTEM RESPONSE UP TO REACTOR TRIP (POINTS 1 TO 2 ON P-T DIAGRAM)

1. THE SMALL BREAK OCCURS AND SUBCOOLED REACTOR COOLANT IS LOST AT A FASTER RATE THAN CAN BE SUPPLIED BY THE MU SYSTEM.
2. PRESSURIZER LEVEL DROPS, AND RC PRESSURE DECREASES AS THE PRESSURIZER STEAM SPACE EXPANDS. LITTLE OR NO CHANGE OCCURS IN THE REACTOR COOLANT TEMPERATURE; THE REACTOR COOLANT REMAINS SUBCOOLED.
3. THE REACTOR TRIPS ON VARIABLE TEMPERATURE AND PRESSURE (DIAGONAL PORTION OF THE RPS TRIP ENVELOPE), AND THE RODS ENTER THE CORE AND SHUT DOWN THE FISSION PROCESS. CORE POWER DROPS TO DECAY HEAT LEVELS.
4. THE TURBINE TRIPS, AND A MAIN FEEDWATER RUNBACK TO CONTROL SG LEVEL AT THE LOW LEVEL LIMIT BY THE ICS IS STARTED.
5. THE REACTOR COOLANT TEMPERATURE DROPS (SIMILAR TO THAT WHICH OCCURS AFTER ANY REACTOR TRIP); AN INCREASE IN THE OUTSURGE FROM THE PRESSURIZER OCCURS TO COMPENSATE FOR TEMPERATURE REDUCTION (SHRINKAGE) OF THE REACTOR COOLANT; AND PRESSURIZER LEVEL AND RC PRESSURE CONTINUE TO DROP.

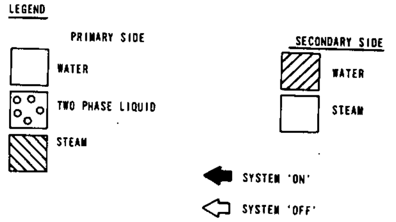
POST-TRIP EVOLUTION TO TWO-PHASE NATURAL CIRCULATION (POINTS 2 TO 3 ON P-T DIAGRAM)

1. RCS PRESSURE AND PRESSURIZER LEVEL CONTINUE TO DROP DUE TO THE LOSS OF REACTOR COOLANT AND RCS CONTRACTION FOLLOWING THE REACTOR TRIP.
2. BECAUSE COOLANT IS BEING LOST, RCS PRESSURE DROPS FASTER THAN NORMAL. AS A RESULT, THE REACTOR COOLANT SUBCOOLING MARGIN IS LOST, ES IS ACTUATED ON LOW RCS PRESSURE, THE PRESSURIZER EMPTIES, AND THE HOT LEG SATURATES.
3. HPI IS STARTED; THE RC PUMPS ARE TRIPPED MANUALLY; MFW IS AUTOMATICALLY DIVERTED TO THE UPPER NOZZLES WHEN THE RC PUMPS ARE TRIPPED; AND THE PRESSURIZER LEVEL GOES OFF SCALE LOW.
4. RCS FLOW DECREASES TO NATURAL CIRCULATION LEVELS AS THE PUMPS COAST DOWN.
5. STEAM BEGINS TO COLLECT IN THE PRIMARY SYSTEM LOOPS
   - STEAM ENTERS THE HOT LEG FROM THE PRESSURIZER
   - SOME OF THE REACTOR COOLANT FLASHES TO STEAM BECAUSE THE SYSTEM PRESSURE IS DROPPING
   - STEAM CAN POSSIBLY BE FORMED IN THE CORE IF THE REACTOR COOLANT IN THE CORE SATURATES AND BOILING OCCURS.
6. TWO-PHASE NATURAL CIRCULATION IS IN PROGRESS.

   NOTE: AT THIS POINT INTO A SMALL BREAK, THE SYSTEM RESPONSE CAN GO IN TWO DIRECTIONS. IF THE BREAK IS VERY SMALL, THE STEAM WITHIN THE PRIMARY SYSTEM CAN BE CONDENSED WITHIN THE STEAM GENERATOR OR BY COLD HPI WATER, AND THE REACTOR COOLANT CAN BE RETURNED TO A SUBCOOLED STATE. ONCE THE REACTOR COOLANT SUBCOOLING MARGIN IS ESTABLISHED, THE HPI CAN BE THROTTLED AND A HOT SHUT DOWN CONDITION CAN BE ACHIEVED. IF THE BREAK IS SOMEWHAT LARGER, ADDITIONAL STEAM WILL BE FORMED (SEE NEXT SEQUENCE).

SYSTEM RESPONSE TO THE END OF THE "FLOW CIRCULATION" PHASE (POINTS 3 TO 4 ON P-T DIAGRAM)

1. THE RCS P&T IS SLOWLY DROPPING AND APPROACHING THE SATURATION PRESSURE AND TEMPERATURE OF THE STEAM GENERATORS.
2. ADDITIONAL STEAM IS FORMED, AND THE STEAM AND WATER START TO SEPARATE (I.E., STEAM CAN RISE FASTER BECAUSE IT IS LIGHTER).
3. STEAM WILL START TO ACCUMULATE IN LARGE AMOUNTS IN THE HIGH POINTS IN THE SYSTEM (UPPER HEAD OF REACTOR VESSEL AND UPPER HOT LEG PIPING).
4. AS STEAM COLLECTS IN THE UPPER HOT LEG PIPING, THE TWO-PHASE NATURAL CIRCULATION FLOW DROPS; THIS ENCOURAGES BOILING IN THE CORE AND MORE STEAM PRODUCTION.
5. THE VOLUME OF THE HOT LEG OCCUPIED BY STEAM GROWS; AND THE UPPER HOT LEG 180° ELBOWS BECOME COMPLETELY OCCUPIED BY STEAM.
6. TWO-PHASE NATURAL CIRCULATION STOPS AS WATER IS NO LONGER CAPABLE OF FLOWING UP THE HOT LEG PIPING AND INTO THE STEAM GENERATORS.
7. THE "FLOW CIRCULATION" PHASE IS OVER.



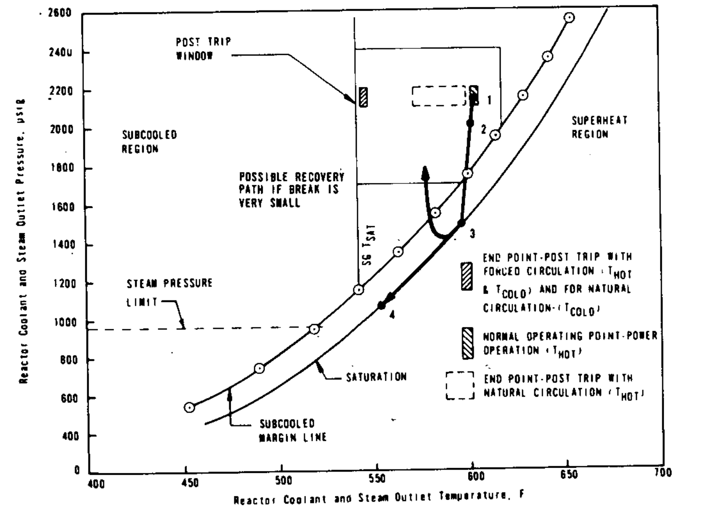
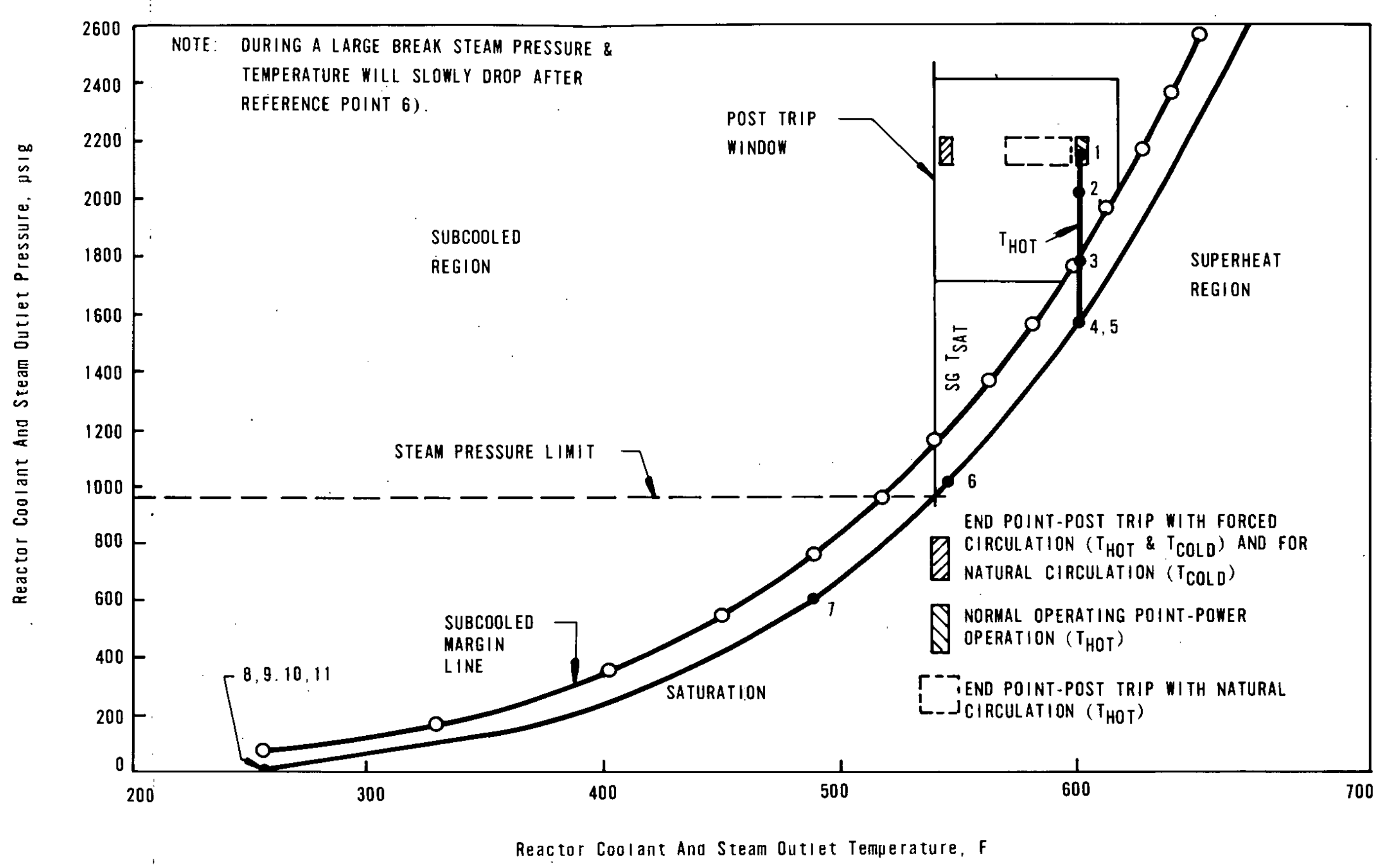
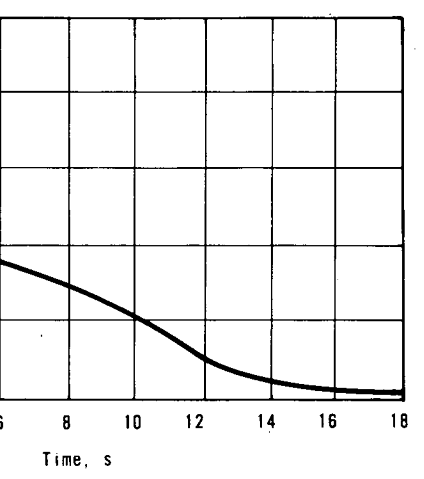


Figure F-3. SYSTEM RESPONSE FOR LARGE BREAK IN HOT LEG PIPING

DOUBLE-ENDED HOT LEG BREAK

REFERENCE POINTS	TIME (SECONDS)	REMARKS
1	0.0	LOCA OCCURS WITH THE PLANT INITIALLY AT 100% POWER.
2	<1.0	RCS PRESSURE DROPS TO RPS TRIP SETPOINT AND A REACTOR TRIP IS INITIATED.
3	<1.0	SUBCOOLING MARGIN IS LOST.
4	<1.0	ES ACTUATED ON LOW RC PRESSURE AND HIGH RB PRESSURE. HPI AND LPI WILL BECOME OPERATIVE WITHIN 35 SECONDS.
5	<1.0	THE REACTOR COOLANT SATURATES IN THE HOT LEGS.
6	4.0	RCS PRESSURE DROPS BELOW SECONDARY SIDE PRESSURE; THE SG'S BECOME A HEAT SOURCE INSTEAD OF A HEAT SINK.
7	9.0	RCS PRESSURE DROPS TO 600 PSIG AND BORATED WATER FROM THE CFTs BEGINS TO ENTER THE REACTOR VESSEL.
8	18	THE END OF BLOWDOWN IS REACHED AS THE RCS AND CONTAINMENT EQUALIZE IN PRESSURE (~ 20-40 PSIG). THE RCS IS ESSENTIALLY VOID (LITTLE OR NO WATER EXISTS) AND THE CORE OUTLET THERMOCOUPLES WOULD INDICATE SUPERHEATED CONDITIONS.
9	19	END OF REFILL. THE RV HAS BEEN REFILLED UP TO THE BOTTOM OF THE ACTIVE CORE. AT THIS TIME, THE HPI AND LPI SYSTEMS ARE FULLY OPERATIVE.
10	30	THE CFT WATER VOLUME IS DEPLETED.
11	350	THE CORE IS RECOVERED BY WATER AND THE FUEL'S TEMPERATURE EXCURSION IS TERMINATED. THE CORE EXIT THERMOCOUPLES RETURN TO SATURATED CONDITIONS.





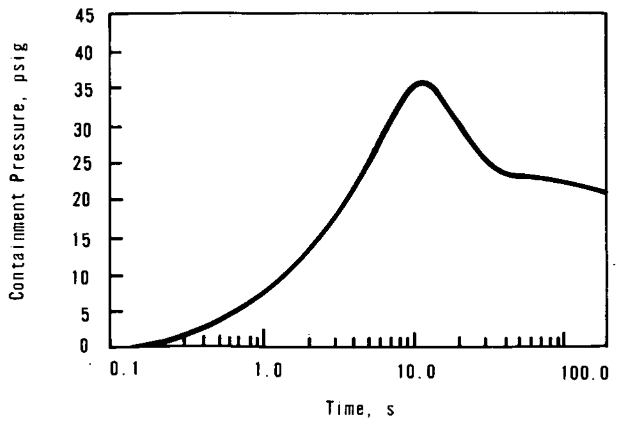
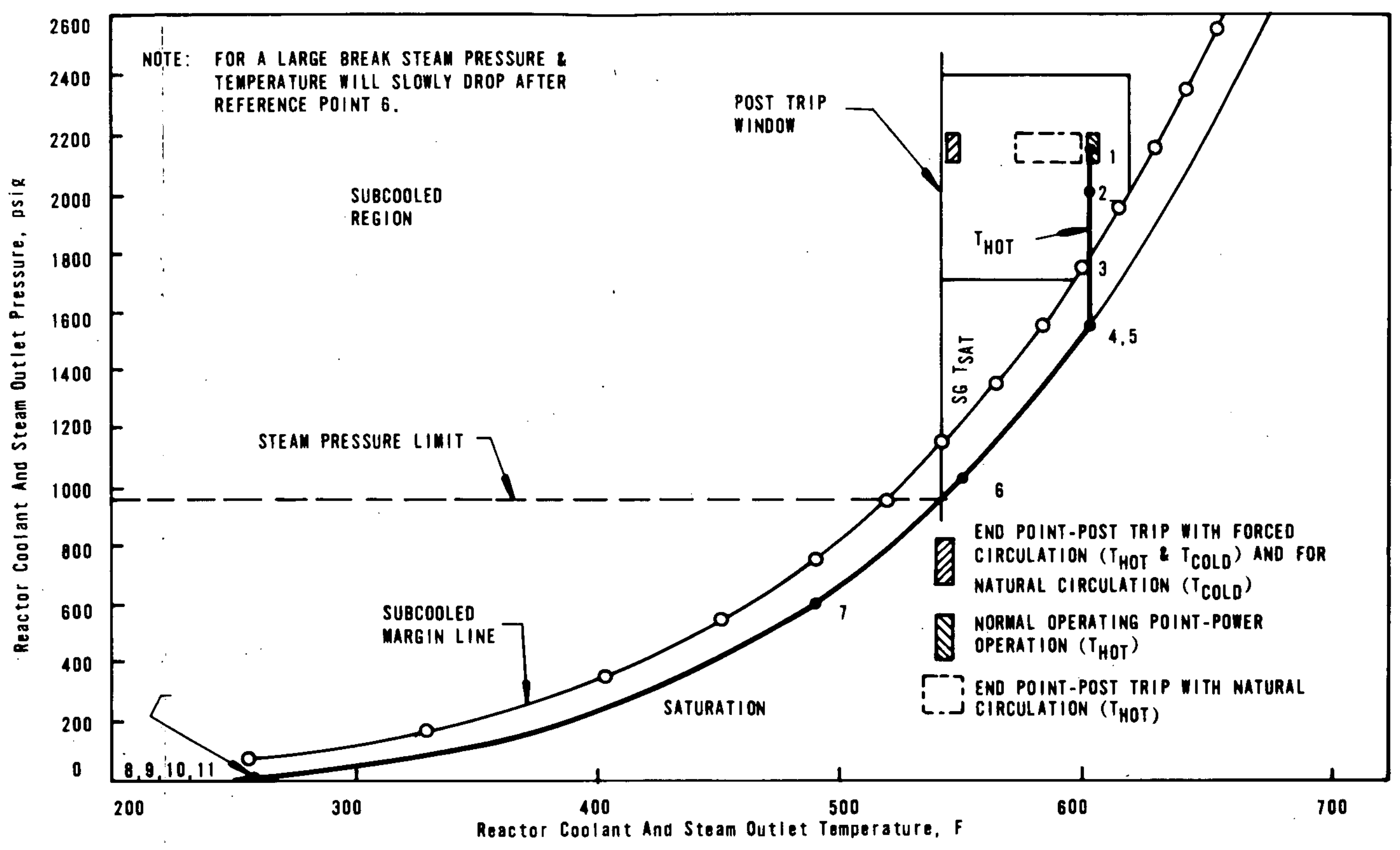
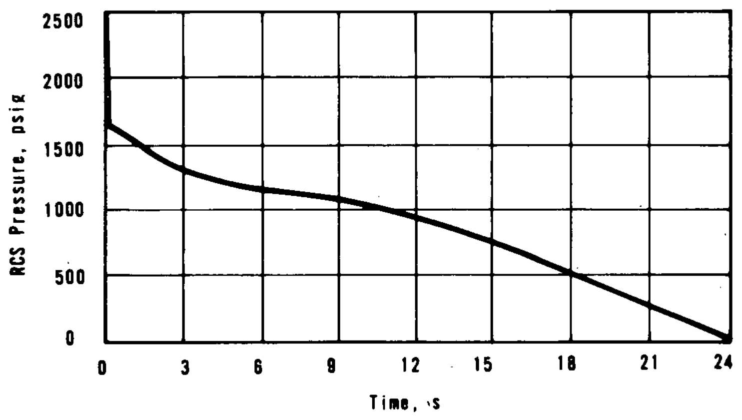


Figure F-2. SYSTEM RESPONSE FOR LARGE BREAK IN COLD LEG PIPING

DOUBLE-ENDED COLD LEG BREAK AT DISCHARGE OF RC PUMPS

REFERENCE POINTS	TIME (SECONDS)	REMARKS
1	0.0	LOCA OCCURS WITH THE PLANT INITIALLY AT 100% POWER.
2	<1.0	RCS PRESSURE DROPS TO RPS TRIP SETPOINT AND A REACTOR TRIP IS INITIATED.
3	<1.0	SUBCOOLING MARGIN IS LOST.
4	<1.0	ES ACTUATED ON LOW RC PRESSURE AND HIGH RB PRESSURE. HPI AND LPI WILL BECOME OPERATIVE WITHIN 35 SECONDS.
5	<1.0	THE REACTOR COOLANT SATURATES IN THE HOT LEGS.
6	12.0	RCS PRESSURE DROPS BELOW SECONDARY SIDE PRESSURE; THE SG's BECOME A HEAT SOURCE INSTEAD OF A HEAT SINK.
7	17.0	RCS PRESSURE DROPS TO 600 PSIG AND BORATED WATER FROM THE CFTs BEGINS TO ENTER THE REACTOR VESSEL.
8	24.0	THE END OF BLOWDOWN IS REACHED AS THE RCS AND CONTAINMENT EQUALIZE IN PRESSURE (~ 20-40 PSIG). THE RCS IS ESSENTIALLY VOID (LITTLE OR NO WATER EXISTS) AND THE CORE OUTLET THERMOCOUPLES WOULD INDICATE SUPERHEATED CONDITIONS.
9	35.0	END OF REFILL. THE RV HAS BEEN REFILLED UP TO THE BOTTOM OF THE ACTIVE CORE. AT THIS TIME, THE HPI AND LPI SYSTEMS ARE FULLY OPERATIVE.
10	50.0	THE CFT WATER VOLUME IS DEPLETED.
11	380.0	THE CORE IS RECOVERED BY WATER AND THE FUEL'S TEMPERATURE EXCURSION IS TERMINATED. THE CORE EXIT THERMOCOUPLES RETURN TO SATURATED CONDITIONS.





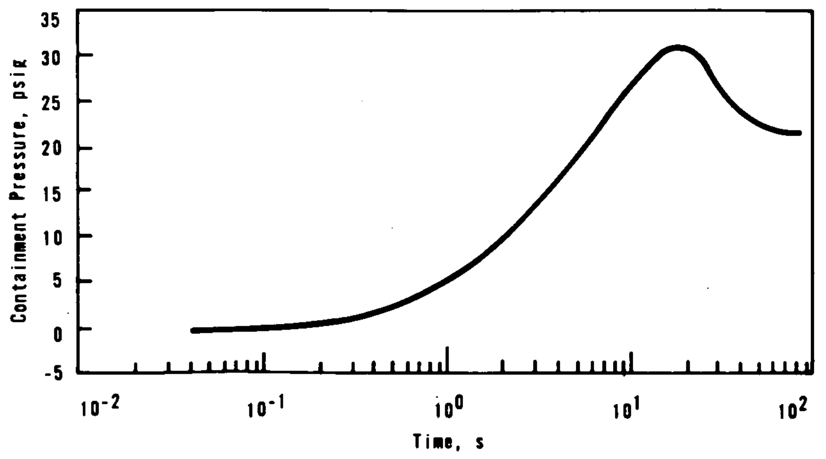
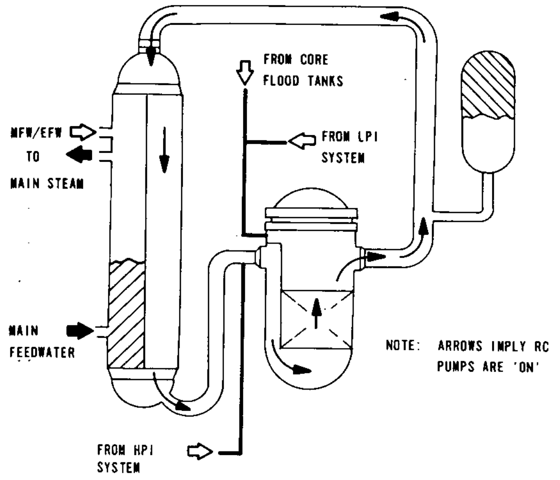
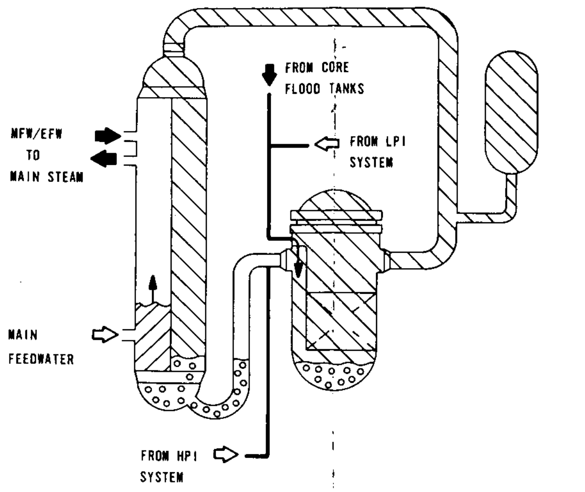


Figure F-1. FLUID HISTORY DURING A LARGE BREAK



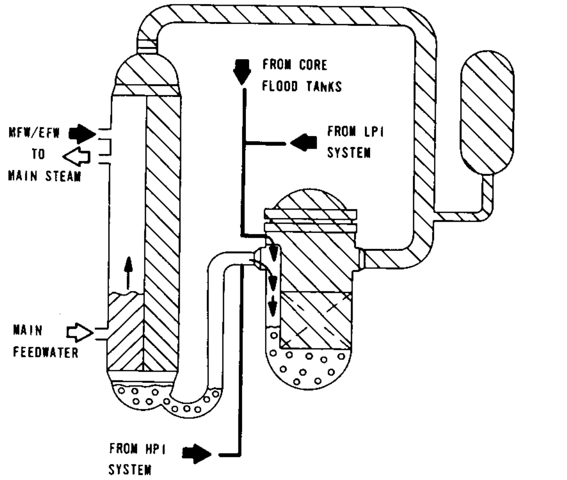
PRE-LOCA INITIAL CONDITION

RCS INVENTORY IS NORMAL WITH THE REACTOR AT POWER AND THE RC PUMPS 'ON'.



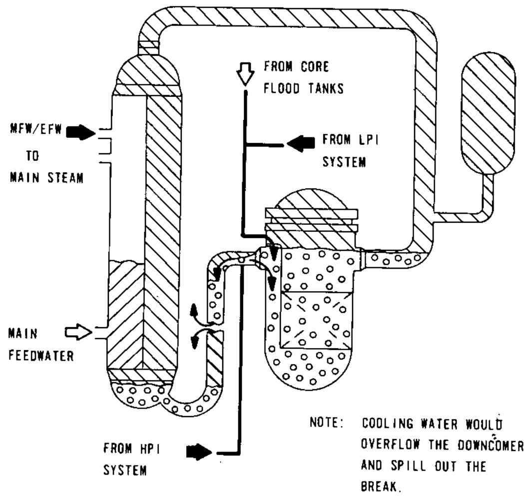
END OF BLOWDOWN

A LARGE BREAK HAS OCCURRED AND NEARLY ALL OF THE INITIAL REACTOR COOLANT INVENTORY HAS ESCAPED TO THE RB. BLOWDOWN LASTS ANYWHERE FROM 20-200 SECONDS DEPENDING ON THE BREAK SIZE AND ENDS WHEN THE RCS AND RB PRESSURES EQUALIZE. AT THE END OF BLOWDOWN THE CFT'S WILL BE OPERATIVE AND THE HPI/LPI SYSTEMS WILL BE FUNCTIONAL WITHIN 35 SECONDS FOLLOWING ESAS ACTUATION.

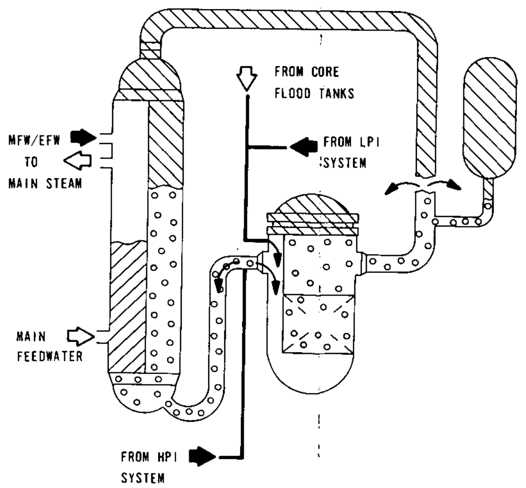


END OF REFILL

THE ECCS HAS REFILLED THE REACTOR VESSEL UP TO THE BOTTOM OF THE CORE REGION.



REFLOOD - COLD LEG BREAK



REFLOOD - HOT LEG BREAK

DURING REFLOOD THE ECCS FLOODS THE REACTOR VESSEL WITH WATER, RESTORES CORE COOLING, AND REFILLS THE SYSTEM UP TO THE ELEVATION OF THE BREAK. THE CFT'S EMPTY DURING REFLOOD, AND THE LPI SYSTEM PROVIDES THE COOLING WATER TO REFILL THE SYSTEM. CORE COOLING WILL BE MAINTAINED AS LONG AS LPI FLOW IS CONTINUED.

LEGEND

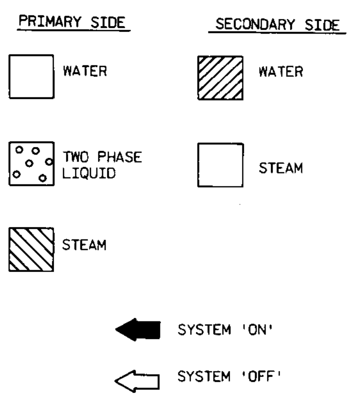
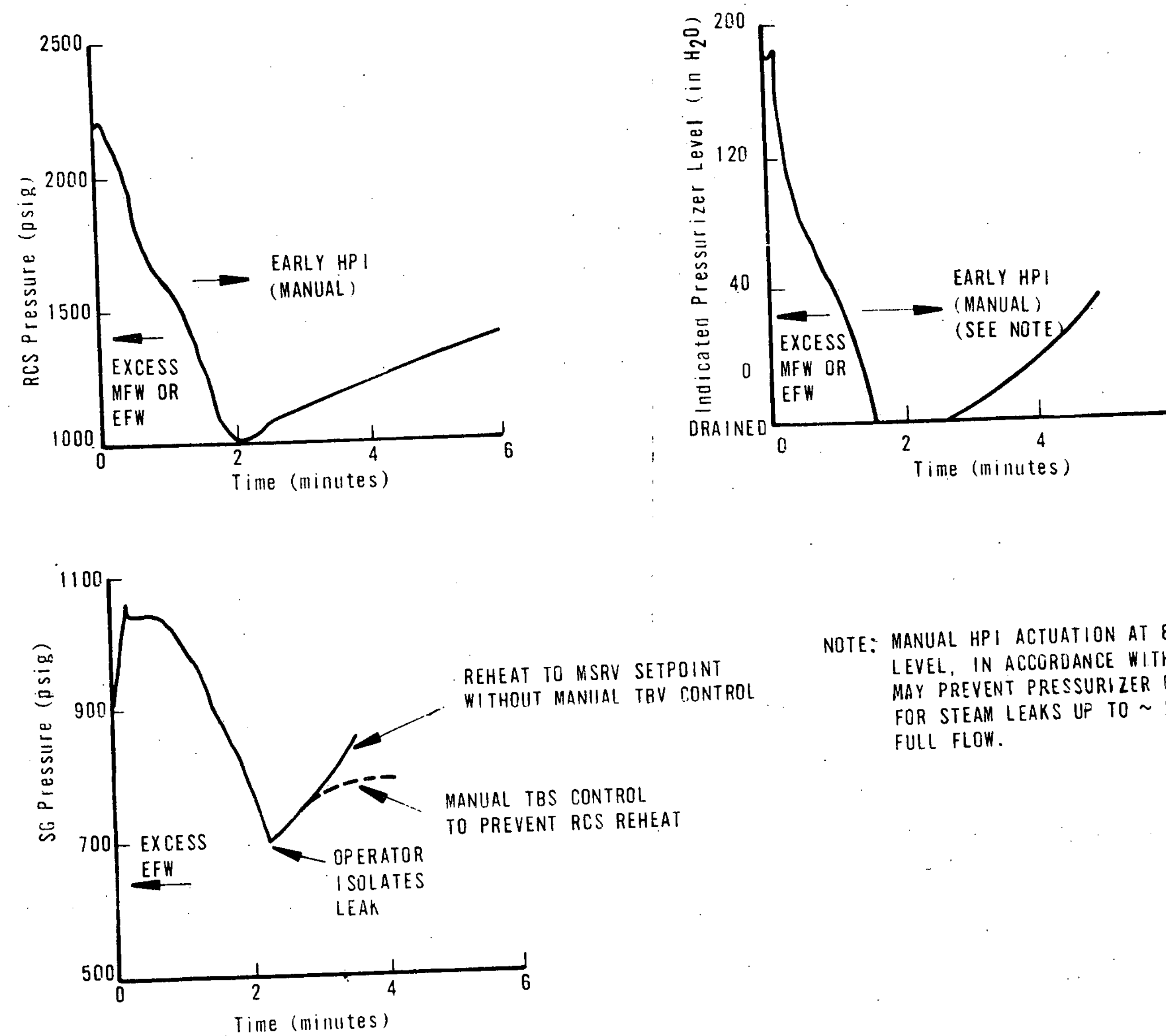
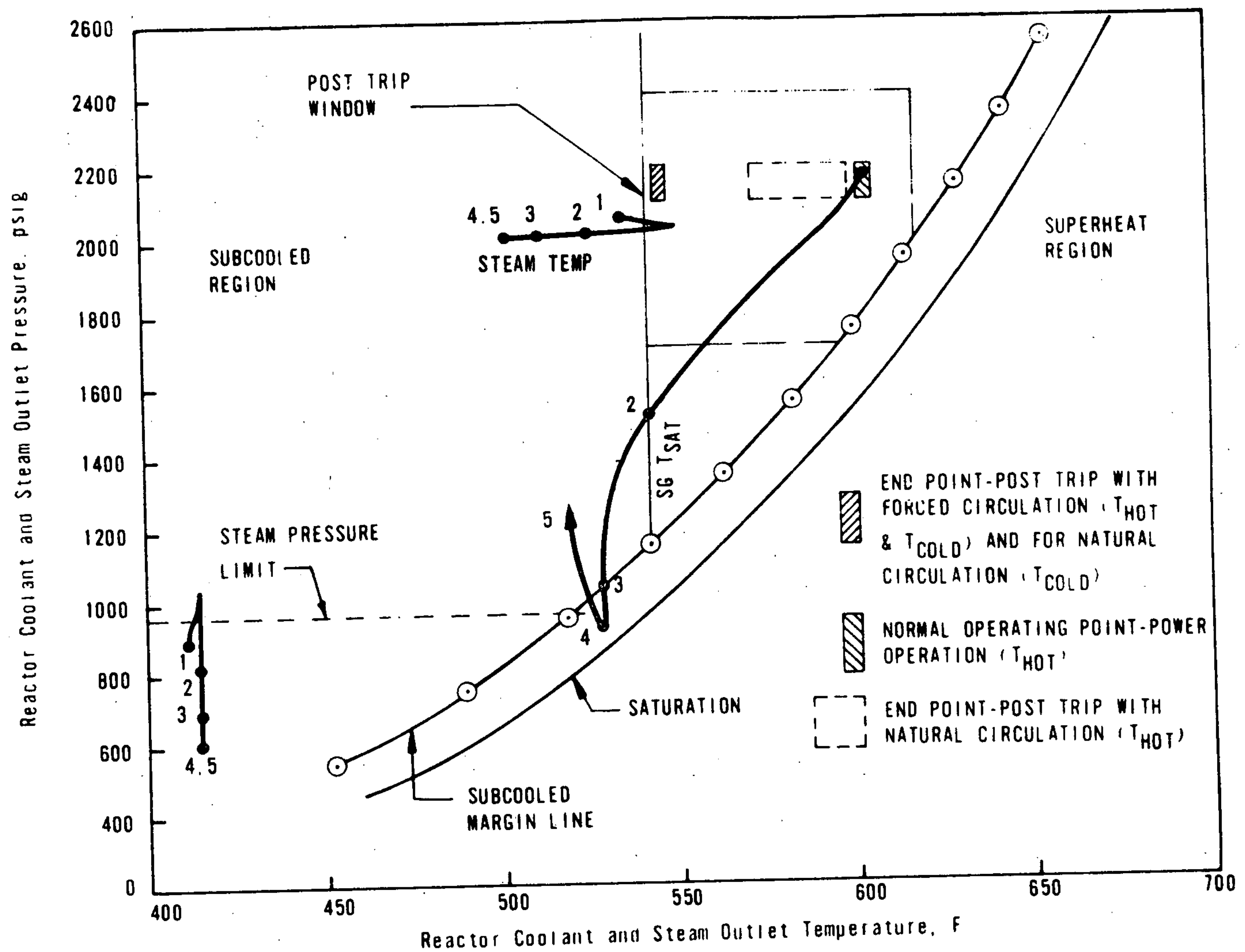


Figure E-3 TIME RELATIONSHIP OF KEY PARAMETERS



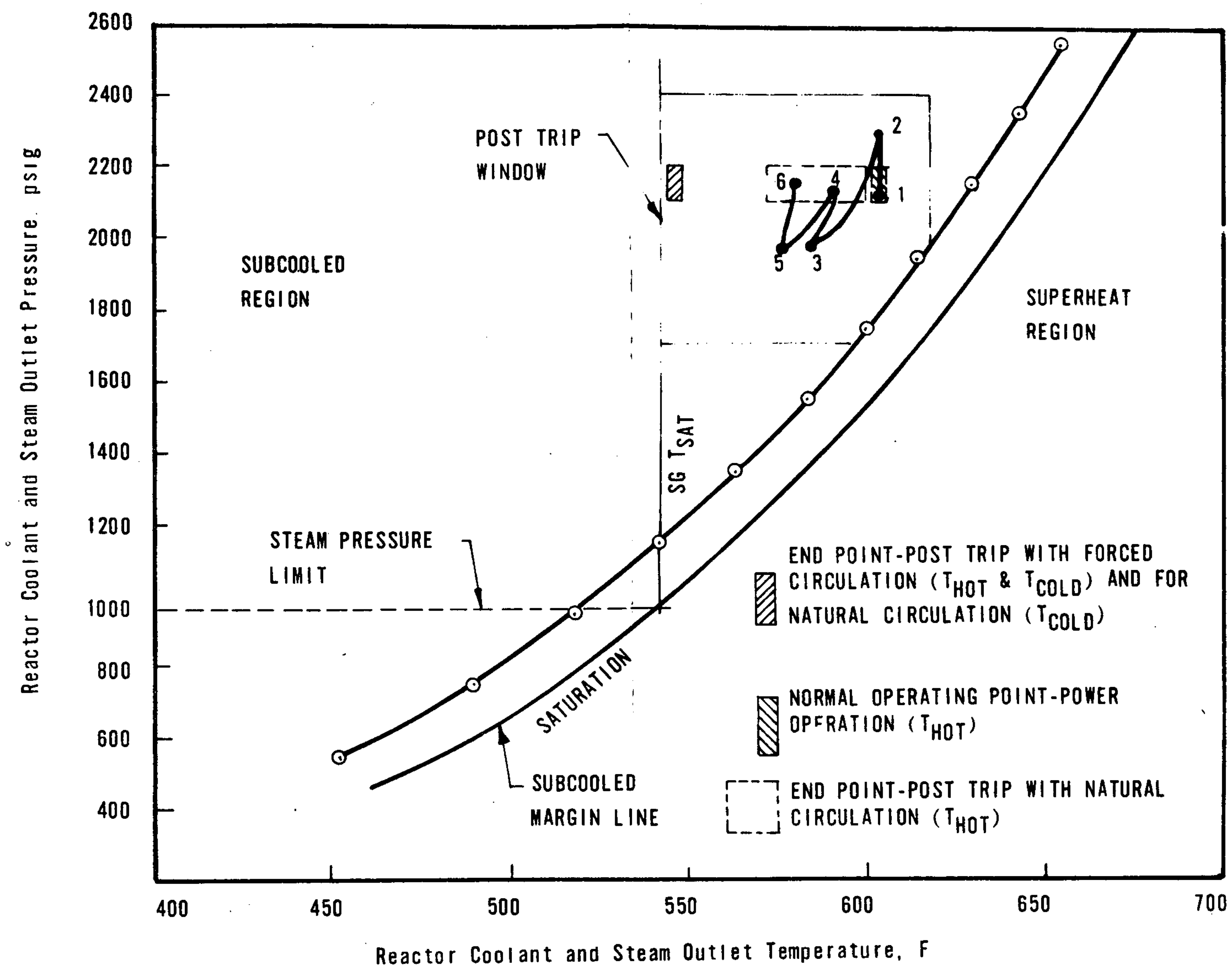
NOTE: MANUAL HPI ACTUATION AT 80" LEVEL, IN ACCORDANCE WITH PART I, MAY PREVENT PRESSURIZER DRAINAGE FOR STEAM LEAKS UP TO ~ 25% OF FULL FLOW.

Figure E-1 SMALL STEAM LEAK



Reference Points	Time (Seconds)	Remarks
1	0	TBS valves fail open from 100% full power allowing 25% of full power steam flow.
1-2	0-95	Reactor trip on high flux (~ 18 seconds), turbine trip, ICS runs back MFW, makeup valve goes wide open.
2	95	Pressurizer drained; ES trip on low RC pressure, starts HPI.
3	100	Subcooled margin lost. Operator trips RC pumps; MFW diverted to upper nozzles.
4	120	Operator isolates TBS to stop steam leak. RCS pressure begins to increase.
5	300	Pressurizer level > 80" and increasing; operator stops HPI and realigns for normal makeup/letdown. To prevent excessive swell due to reheating, operator should regain manual control of TBV's; otherwise RCS could reheat until SG pressure reaches the MSRV setpoint.

<u>Figure D-10</u> TYPICAL LOSS OF OFFSITE POWER P-T RESPONSE



Reference Points	Time (Seconds)	Remarks
1	0	Plant separates from grid.
1-2	0-15	Reactor trips on high pressure. Hydro generators begin loading within 23 seconds after reactor trip.
2-3	15-75	Plant cools down as steam is relieved to atmosphere and steam safety valves and to condenser by TBS.
3-4	75-240	T_{hot} increases to develop required temperature difference for natural circulation.
4-5	240-480	RCS is cooled by natural circulation. Steam pressure is dropping due to quenching effects of EFDW.
5	480	OTSG levels reach 50% on the operate range.
5-6	>480	Steam pressure, T_{av} and RCS pressure return to normal post-trip values as decay heat adds energy to the system.

Figure D-8 TYPICAL PARAMETER TRENDS FOR LOOP/LOSS OF SECONDARY INVENTORY CONTROL (LOW)

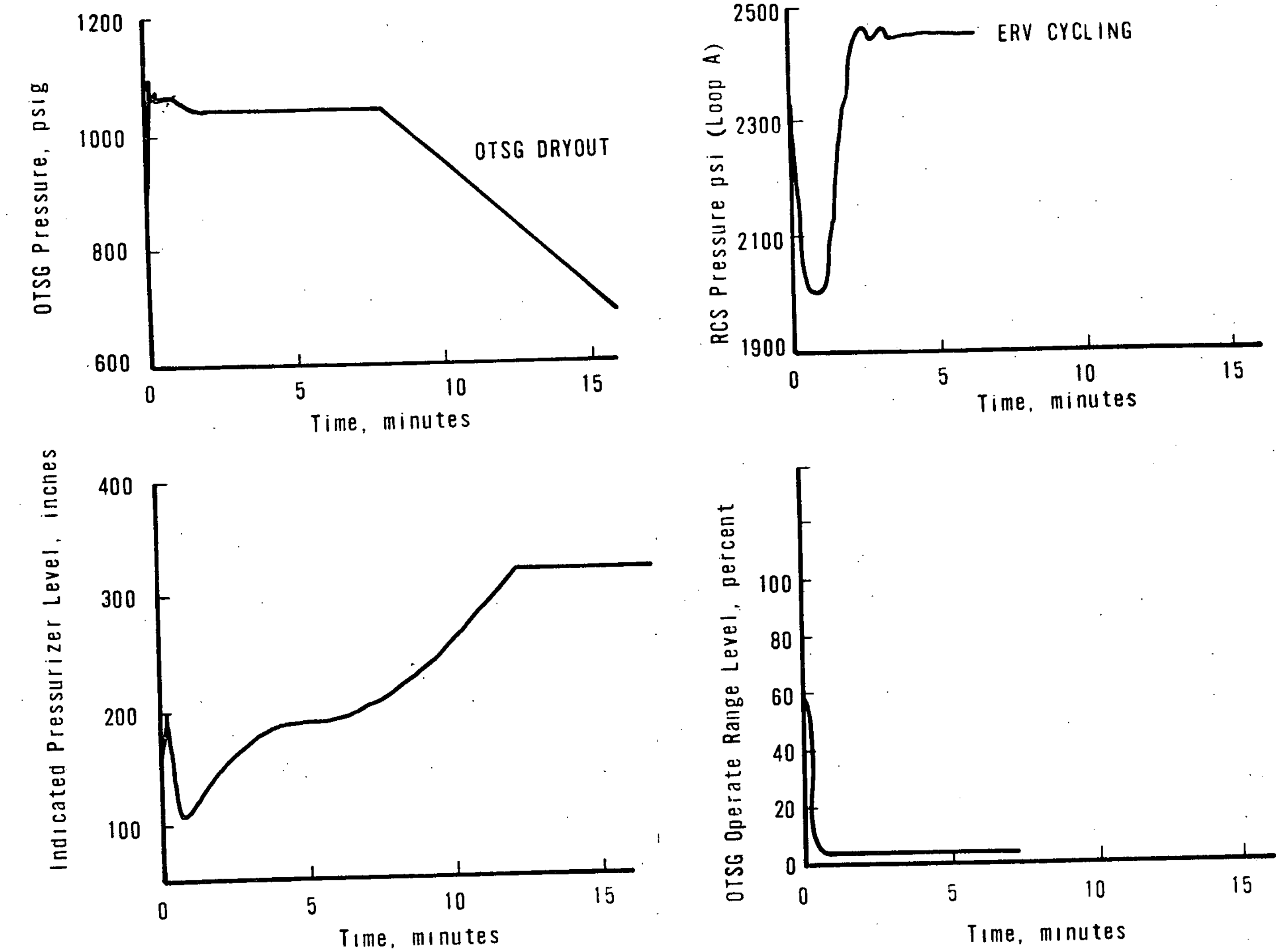


Figure D-7 TYPICAL PARAMETER TRENDS FOR LOOP

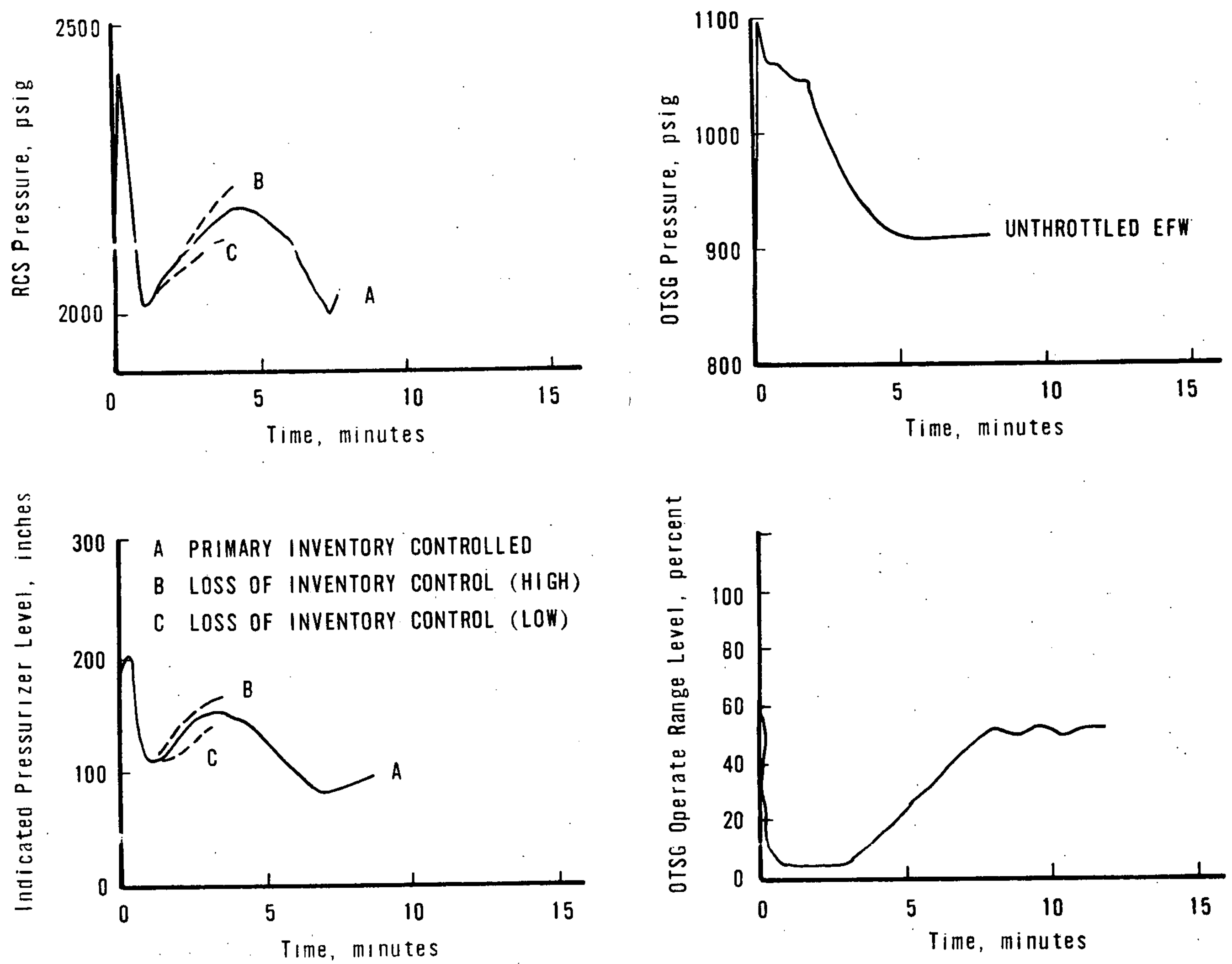


TABLE C-3

EFFECTS OF FAILURES ON STEAM GENERATOR TUBE LEAK CONTROL

FAILURES WHICH AFFECT RC INVENTORY CONTROL

(NOTE: OVERCOOLING AND OVERHEATING TRANSIENTS CAN CAUSE REACTOR COOLANT EXPANSION OR CONTRACTION AND POSSIBLY RESULT IN LOOP VOIDING; THESE ARE COVERED IN THE STEAM GENERATOR INVENTORY AND PRESSURE CONTROL CHARTS)

FAILURE	EFFECT ON TUBE LEAK CONTROL AND PLANT COOLDOWN	CORRECTIVE ACTIONS TO ALLOW A CONTROLLED COOLDOWN
RC INVENTORY 1. LOSS OF NORMAL LETDOWN	THE LOSS OF LETDOWN PREVENTS THE USE OF FEED AND BLEED TO INCREASE THE RCS BORON CONCENTRATION DURING COOLDOWN. IT ALSO PREVENTS THE USE OF NORMAL BORON SAMPLING. IN SOME CASES THE COOLDOWN CONTRACTION MAY NOT PROVIDE ENOUGH ADDITIONAL VOLUME TO ADD BORON WITHOUT BLEED AND FEED (FOR EXAMPLE, IF COOLDOWN STARTS WITH THE PRESSURIZER AT A HIGH LEVEL).	- ATTEMPT TO RESTORE LETDOWN; CAUSES OF FAILURE COULD INCLUDE CLOSED VALVES OR LOSS OF CCW TO THE LETDOWN COOLERS. - ADDITION OF BORON WITHOUT LETDOWN CAN BE DONE BY METHODS DEPENDING ON THE RCS INVENTORY AND WHETHER IT HAS OR DOES NOT HAVE ADEQUATE SUBCOOLING. IN EACH CASE BORON SAMPLING IS NOT POSSIBLE SO THE BORON CALCULATIONS SHOULD ASSUME THAT THE CONCENTRATION IN THE RCS IS THE SAME AS IT WAS PRIOR TO THE ACCIDENT EVEN THOUGH HPI MAY HAVE INCREASED IT. 1. IF SUBCOOLING IS AMPLE, REDUCE THE MAKEUP/HPI FLOW AND COOL THE REACTOR COOLANT TO OBTAIN A VOLUME CONTRACTION; ADD CONCENTRATED BORIC ACID TO THE MAKEUP TANK AND INJECT TO OBTAIN A HIGH PRESSURIZER LEVEL (NOTE: IF HPI IS OPERATING, THEN ONE MAKEUP PUMP MUST BE REALIGNED TO THE MAKEUP TANK, THE OTHER HPI PUMP SHOULD STAY ALIGNED FOR ECCS INJECTION FOR THE TUBE LEAK). THIS VOLUME CONTRACTION APPROACH CAN BE REPEATED AS REQUIRED. STARTUP NEUTRON MONITORS SHOULD BE CHECKED; IF THE COUNT RATE INCREASES THE COOLDOWN SHOULD BE HALTED. 2. IF THE SYSTEM IS SATURATED OR NEAR SATURATION OR IF HPI COOLING HAS BEEN USED AND THE SYSTEM IS WATER SOLID AND A PRESSURIZER BUBBLE CANNOT BE DRAWN, THEN THE CONTRACTION METHOD CANNOT BE USED. CONTINUOUS HPI INJECTION FROM THE BWST AND FLOW OUT THROUGH THE PORV OR BREAK WILL ALLOW A FORM OF BLEED AND FEED THAT WILL INCREASE THE CONCENTRATION SUFFICIENTLY FOR SHUTDOWN (IF A LARGE VOLUME OF THE BWST IS PUMPED INTO THE RCS).
RC INVENTORY 2. TOO MUCH HPI (OPERATOR ERROR-FAILURE TO THROTTLE HPI AFTER SUBCOOLED, MARGIN REQUIREMENTS ARE SATISFIED).	- TOO MUCH HPI WILL RESULT IN WATER SOLID CONDITIONS, WATER RELIEF OUT OF THE PRESSURIZER, AND HIGH RC PRESSURES. THIS TENDS TO INCREASE THE PRIMARY-TO-SECONDARY TUBE LEAK RATE. IF TOO MUCH MU IS ADDED AT REDUCED RCS TEMPERATURES, NDT LIMITS CAN ALSO BE VIOLATED. - A RESULTANT PORV/SAFETY VALVE FAILURE TO RESEAT CAUSED BY WATER RELIEF WILL REQUIRE CONTINUED HPI. THE BWST INVENTORY WILL BE DRAINED FASTER SINCE A PATH, IN ADDITION TO THE TUBE FAILURES, EXISTS THRU WHICH REACTOR COOLANT CAN BE LOST. ALTHOUGH WATER WILL BE RETURNED TO THE SUMP THROUGH THE PRESSURIZER LEAK (I.E., CAN BE RECIRCULATED), COOLDOWN WILL BECOME MORE DIFFICULT BECAUSE SYSTEM SATURATION WILL MOST LIKELY OCCUR EARLY IN THE TRANSIENT AND WHEN SUBCOOLING IS RESTORED THE PRESSURIZER WILL BE WATER SOLID. RECIRCULATION FROM THE SUMP MAY OR MAY NOT BE POSSIBLE DEPENDING ON THE TRADE OFF BETWEEN TUBE LEAK FLOW AND PRESSURIZER VALVE LOSS.	- HPI MUST BE CONTROLLED, TO ALLOW AN ORDERLY PLANT COOLDOWN AND DEPRESSURIZATION, PER THE GUIDANCE PROVIDED IN THE "BEST METHODS" SECTION. SHOULD A SOLID WATER SYSTEM EVOLVE, A PRESSURIZER STEAM BUBBLE CAN BE RE-ESTABLISHED (TURN ON HEATERS WHILE COOLING DOWN SLIGHTLY). IF A NON-ISOLABLE VALVE FAILURE OCCURS, A SOLID WATER COOLDOWN AND DEPRESSURIZATION WILL BE NECESSARY. WHEN SOLID, RCS PRESSURE WILL BE CONTROLLED BY THE DISCHARGE HEAD OF THE HPI PUMPS. - IF DEPLETION OF THE BWST APPEARS POSSIBLE, RCS COOLDOWN SHOULD BE ACCELERATED TO ENGAGE THE CORE FLOOD TANKS AND THE DHR SYSTEM. - BACKUP SOURCES OF BORATED WATER SHOULD BE PREPARED FOR INJECTION.

FAILURE	EFFECTS ON TUBE LEAK CONTROL AND PLANT COOLDOWN	CORRECTIVE ACTIONS TO ALLOW A CONTROLLED COOLDOWN
RC INVENTORY 3. NOT ENOUGH HPI/MAKEUP (AUTOMATIC INITIATION OF ONLY ONE PUMP; MANUAL INITIATION OF ONLY ONE PUMP; OPERATOR THROTTLES PREMATURELY; FLOW BLOCKAGE IN INJECTION LINE)	- FAILURE TO SUPPLY ENOUGH HPI TO MAKEUP FOR THE PRIMARY-TO-SECONDARY LEAKAGE AND THE RCS CONTRACTION VOLUME DURING COOLDOWN WILL RESULT IN A LOSS OF PRESSURIZER LEVEL AND REACTOR COOLANT SUBCOOLED MARGIN. - WHEN THE REACTOR COOLANT SATURATES, THE TUBE LEAK RATE WILL DECREASE AND NATURAL CIRCULATION MAY STOP. - DEPENDING ON THE TUBE LEAK RATE, ONE HPI PUMP WILL RESTORE THE SUBCOOLING MARGIN, BUT THE TIME WILL BE CONSIDERABLY LONGER THAN IF TWO PUMPS WERE RUNNING. - COOLDOWN MAY BE SLOWER IF THE REACTOR COOLANT IS SATURATED, OR IF THE RETURN TO SUBCOOLING IS DELAYED.	- IF ONLY ONE HPI PUMP IS STARTED, AND THE SUBCOOLED MARGIN IS LOST, START THE SECOND HPI PUMP (IF POSSIBLE); OR IF BOTH PUMPS ARE RUNNING ENSURE THAT THEY ARE RUNNING AT THE HIGHEST POSSIBLE FLOW (~ 250 GPM PER INJECTION LINE). - IF IT IS NOT POSSIBLE TO START A SECOND HPI PUMP, OR IF THE MAXIMUM POSSIBLE FLOW CANNOT BE ATTAINED THROUGH THE INJECTION LINES (~ 250 GPM PER LINE), THEN THE RCP'S MUST BE TRIPPED WHEN THE SUBCOOLED MARGIN IS LOST. THE PLANT COOLDOWN METHOD WILL DEPEND ON WHETHER SUBCOOLING IS RESTORED AND WHETHER NATURAL CIRCULATION EXISTS (SEE "RC PUMP RESTART CRITERIA" IN THE "BEST METHODS" CHAPTER). IF THE RC PUMPS CANNOT BE RESTARTED THE REACTOR COOLANT SYSTEM MUST BE DEPRESSURIZED USING THE PORV. THE MOST LIKELY CONDITION TO OCCUR DURING COOLDOWN IS RESTORATION OF THE SUBCOOLING MARGIN; NATURAL CIRCULATION WILL PROBABLY EXIST AT ALL TIMES DURING THE COOLDOWN (SATURATED OR SUBCOOLED). IF THE SUBCOOLING MARGIN IS RESTORED, THE RC PUMPS CAN BE RESTARTED AND COOLDOWN CAN PROCEED IN A STRAIGHT FORWARD FASHION.

TABLE C-3

EFFECTS OF FAILURES ON STEAM GENERATOR TUBE LEAK CONTROL

Sheet 3 of 4

FAILURES WHICH AFFECT RC PRESSURE CONTROL

FAILURE: RC PRESSURE 1
SPRAY VALVE FAILS "ON"

EFFECT ON TUBE LEAK CONTROL AND PLANT COOLDOWN:

THE UNCONTROLLED ADDITION OF PRESSURIZER SPRAY WILL CAUSE A SLOW DEPRESSURIZATION OF THE RCS BECAUSE THE SPRAY CAPACITY IS HIGHER THAN THAT OF THE HEATERS. THE TRANSIENT IS SLOW; BUT, IF IT LASTS LONG ENOUGH THE SUBCOOLING MARGIN CAN BE LOST IF CORRECTIVE ACTION IS NOT TAKEN. THE EFFECTS OF THE SPRAYS MAY BE COUNTERED BY STEAM GENERATOR COOLING; IF THE PLANT IS CONTINUALLY COOLED AND RC TEMPERATURE IS LOWERED THE SUBCOOLING MARGIN MAY BE MAINTAINED. THE TIME TO LOSS OF SUBCOOLING MARGIN WILL DEPEND ON THE DECAY HEAT RATE, HPI (MAKEUP) FLOW AND THE SIZE OF THE TUBE LEAK. THE RATE OF DEPRESSURIZATION WILL ALSO DEPEND ON THE RATE OF SPRAY FLOW, THE SPRAY TEMPERATURE, THE STEAM SPACE IN THE PRESSURIZER (A SMALL STEAM VOLUME WILL ALLOW A GREATER RATE OF DEPRESSURIZATION), AND THE NUMBER OF HEATERS OPERATING. AS SUBCOOLING IS REDUCED THE LEAK RATE WILL LOWER; BUT WHEN THE MARGIN IS LOST THE RC PUMPS MUST BE TRIPPED (SPRAY WILL STOP) AND HPI TURNED ON FULL. SUBCOOLING WILL RETURN, BUT UNLESS THE SPRAY CAN BE REPAIRED AND CONTROLLED, THE PORV MUST BE USED TO DEPRESSURIZE THE PLANT.

CORRECTIVE ACTIONS TO ALLOW A CONTROLLED COOLDOWN:

SPRAY FLOW CONTROL WITH RC PUMPS RUNNING IS PREFERRED FOR RCS DEPRESSURIZATION. TO REGAIN CONTROL FOR DEPRESSURIZATION THE FOLLOWING STEPS MAY BE DONE (IN ORDER OF PRIORITY):

1. ATTEMPT MANUAL CONTROL OF SPRAY VALVE
2. CLOSE THE SPRAY BLOCK VALVE. DEPRESSURIZATION FOR COOLDOWN CAN BE BY OPENING AND CLOSING THE BLOCK VALVE TO DEPRESSURIZE IN "STEPS" AS THE SYSTEM IS COOLED BY REDUCING STEAM PRESSURE. A HIGHER THAN "NORMAL" PRESSURIZER LEVEL WILL BE HELPFUL SINCE THE PRESSURIZER LEVEL SHOULD STAY ON SCALE.
3. TRIPPING SELECTED RC PUMPS TO REDUCE THE SPRAY FLOW (BUT KEEPING AT LEAST ONE RC PUMP ON TO MAINTAIN FORCED CIRCULATION WHEN THE SUBCOOLING MARGIN EXISTS). THE FOLLOWING TABLE WILL GIVE GENERAL GUIDANCE FOR THE EFFECTS OF RUNNING VARIOUS PUMPS (SPECIFIC GUIDANCE CANNOT BE GIVEN BECAUSE THE TUBE LEAK RATE, PRESSURIZER STEAM VOLUME, AND SPRAY TEMPERATURE WILL HAVE DIFFERENT EFFECTS). THIS TABLE WAS CALCULATED FOR NORMAL OPERATING CONDITIONS.

NUMBER OF RC PUMPS RUNNING		SPRAY FLOW (% FULL FLOW)
SPRAY LOOP	OPPOSITE LOOP	
2	2	100
2	1	92
2	0	84
1(SPRAY LINE NEXT TO RUNNING PUMP)	2	60
1(SPRAY LINE NEXT TO RUNNING PUMP)	1	53
1(SPRAY LINE NEXT TO IDLE PUMP)	2	50
1(SPRAY LINE NEXT TO RUNNING PUMP)	0	41
1(SPRAY LINE NEXT TO IDLE PUMP)	1	38
1(SPRAY LINE NEXT TO IDLE PUMP)	0	26
0	2	20
0	1	~0

AS A RULE OF THUMB TRIPPING ONE PUMP IN EACH LOOP WILL PROVIDE A GOOD BALANCE BETWEEN THE SPRAY FLOW RATE AND THE HEATER CAPACITY. IT WILL ALSO PROVIDE GOOD FORCED CIRCULATION FOR COOLDOWN.

FAILURE: RC PRESSURE 2.
PRESSURIZER SPRAYS FAIL "OFF" (RC PUMP OFF AND CAN'T RESTART OR SPRAY VALVE FAILED SHUT OR PRESSURIZER IS WATER SOLID)

EFFECT ON TUBE LEAK CONTROL AND PLANT COOLDOWN:

NORMAL RCS PRESSURE CONTROL IS LOST. TUBE LEAK WILL CONTINUE AT A HIGH RATE AS LONG AS RCS PRESSURE IS HIGH. TO DEPRESSURIZE THE PLANT THE PORV MUST BE UTILIZED.

CORRECTIVE ACTIONS TO ALLOW A CONTROLLED COOLDOWN:

TO DEPRESSURIZE THE PLANT USING THE PORV, AN OPEN-SHUT OPERATION MUST BE UTILIZED. THE PLANT SHOULD BE CONTINUALLY COOLED AND THEN DEPRESSURIZED INTERMITTENTLY BETWEEN THE SUBCOOLED MARGIN LINE AND 50F SUBCOOLING. WHEN THE PORV IS OPENED, BOTH PRESSURIZER LEVEL AND REACTOR COOLANT SUBCOOLING SHOULD BE MONITORED. IF PRESSURIZER LEVEL DROPS BELOW ~ 40" OR THE REACTOR COOLANT SUBCOOLED MARGIN IS APPROACHED, THE PORV SHOULD BE CLOSED. AN EXAMPLE OF THE PREFERRED P-T CONTROL METHOD IS SHOWN IN FIGURE C-2 (SEE MODE 3 COOLDOWN).

IF THE PRESSURIZER IS WATER SOLID THEN A BUBBLE MUST BE DRAWN BEFORE DEPRESSURIZING WITH THE PORV. IF A BUBBLE CANNOT BE DRAWN (HEATERS OFF) THEN THE RATE OF DEPRESSURIZATION WITH WATER RELEASE THROUGH THE PORV WILL BE SLOW.

FAILURE	EFFECT ON TUBE LEAK CONTROL AND PLANT COOLDOWN	CORRECTIVE ACTIONS TO ALLOW A CONTROLLED COOLDOWN
RC PRESSURE 3. PRESSURIZER HEATERS FAILED "ON"	IT IS UNLIKELY THAT THIS WILL HAPPEN. AT HIGH PRESSURES THE SPRAYS WILL OVERCOME THE HEATERS SO NO EFFECT WILL BE FELT, BUT THE SPRAY FLOW WILL BALANCE THE HEATERS AT ~ 1300 PSIG RC PRESSURE AND FURTHER DEPRESSURIZATION CANNOT OCCUR. IF THE SPRAYS ARE OFF RC PRESSURE WILL SLOWLY INCREASE AS LONG AS THE HEATERS ARE COVERED BY WATER; AUTOMATIC INTERLOCKS WILL CUTOFF THE HEATERS IF PRESSURIZER LEVEL IS LOW (UNLESS THE INTERLOCKS HAVE FAILED). DEPENDING ON THE TUBE LEAK RATE THE HEATERS MIGHT HAVE TO BE USED TO CONTROL SUBCOOLING, ESPECIALLY TOWARD THE LATER STAGES OF COOLDOWN WHEN THE COOLDOWN RATE IS REDUCED. HOWEVER, FOR MOST COOLDOWNS THE HEATERS ARE NOT REQUIRED.	MANUAL CUTOFF OF PRESSURIZER HEATERS SHOULD BE TRIED. IF THAT FAILS THE BREAKERS CAN BE THROWN TO CUTOFF POWER. IF SUBCOOLING MARGIN IS APPROACHED THEN THE BREAKERS CAN BE RESTORED TO CONTROL SUBCOOLING.
RC PRESSURE 4. PRESSURIZER HEATERS FAILED "OFF"	THIS IS MORE LIKELY THAN THE ABOVE. FOR MOST CASES LOSS OF THE HEATERS WILL NOT AFFECT COOLDOWN, BUT IF THE TUBE LEAK IS LARGE AND SUBCOOLING MARGIN IS LOST THEN HPI WILL HAVE TO BE USED TO REGAIN SUBCOOLING MARGIN DURING THE COOLDOWN. PRESSURE CONTROL WILL BE DIFFICULT IF THE PRESSURIZER FILLS SOLID; THE TUBE LEAK RATE WILL INCREASE AS RC PRESSURE RISES OR SUBCOOLED MARGIN IS INCREASED.	HEATERS ARE AUTOMATICALLY LOADED ON THE BUS CONNECTED TO THE HYDRO UPON LOSS OF OFFSITE POWER. IF THE HEATERS CANNOT BE POWERED, COOLDOWN AND DEPRESSURIZE THE PLANT NORMALLY. IF SUBCOOLING MARGIN IS LOST THEN START HPI AND ATTEMPT TO KEEP SUBCOOLING WITH A PRESSURIZER BUBBLE BY REDUCING RCS TEMPERATURE WITH THE STEAM GENERATOR. IF THE BUBBLE IS LOST SPRAYS BECOME INEFFECTIVE AND THE PORV MUST BE USED FOR DEPRESSURIZATION (AND TO RELIEVE HPI WATER). A SOLID WATER COOLDOWN IS REQUIRED. IN MOST CASES CONTINUED COOLING WITH THE STEAM GENERATORS WILL KEEP THE SUBCOOLED MARGIN.
RC PRESSURE 5. PORV (OR BLOCK VALVE) CLOSED AND SPRAYS OFF	DEPRESSURIZATION IS NOT POSSIBLE; TUBE LEAK WILL CONTINUE.	PREPARE ADDITIONAL SOURCES OF BWST AND CONDENSATE.
RC PRESSURE 6. PRESSURIZER SAFETY VALVE FAIL OPEN; OR PORV FAILS OPEN AND BLOCK VALVE CANNOT BE CLOSED.	THIS IS AN UN-ISOLABLE LOCA; HPI AND REACTOR COOLANT EXPANSION FILL THE PRESSURIZER SOLID WITH WATER. EARLY IN THE TRANSIENT THE REACTOR COOLANT MAY BE SATURATED (DEPENDING ON NUMBER OF HPI PUMPS OPERATING, THE DECAY HEAT LOAD, AND STEAM GENERATOR TEMPERATURE). THE TUBE LEAK WILL CONTINUE AND WILL DEPEND ON THE AMOUNT OF SUBCOOLING (WHICH DEPENDS ON HPI INJECTION RATE AND STEAM GENERATOR SATURATED TEMPERATURE.)	A SOLID WATER COOLDOWN WILL BE REQUIRED WITH THE RCS DEPRESSURIZATION RATE CONTROLLED BY THE DISCHARGE PRESSURE OF THE HPI PUMPS.

EFFECTS OF FAILURES ON STEAM GENERATOR TUBE LEAK CONTROL

FAILURES WHICH AFFECT STEAM GENERATOR INVENTORY CONTROL

FAILURE	EFFECTS ON TUBE LEAK CONTROL AND PLANT COOLDOWN	CORRECTIVE ACTIONS TO ALLOW A CONTROLLED COOLDOWN
OTSG INVENTORY 1. LOSS OF MAIN FEEDWATER	LOSS OF MAIN FEEDWATER HAS NO EFFECT ON PLANT COOLDOWN, IF EMERGENCY FEEDWATER IS STARTED.	WHEN MAIN FEEDWATER IS LOST THE OPERATOR SHOULD VERIFY THAT EMERGENCY FEEDWATER STARTS AND DELIVERS FLOW TO THE CORRECT STEAM GENERATOR. EMERGENCY FEEDWATER MUST BE CONTROLLED IF THE RC PUMPS ARE TRIPPED AND A HIGH OTSG LEVEL IS REQUIRED SO THAT OVERCOOLING EFFECTS WHICH MIGHT CAUSE PRESSURIZER DRAINING WILL BE AVOIDED. A HIGH LEVEL SHOULD BE AVOIDED IN THE OTSG WITH THE TUBE LEAK TO LIMIT THE WATER ACCUMULATION; SUFFICIENT LEVEL SHOULD BE BUILT TO PROVIDE SUBCOOLED NATURAL CIRCULATION. THE TUBE LEAK WILL CONTINUE TO ADD WATER TO THE OTSG AND IF THE EFW LEVEL IS HIGH, STEAMING MAY BE REQUIRED TO PREVENT WATER ACCUMULATION IN THE STEAM LINES. WHEN EFW IS STARTED, THE STEAM SUPPLY TO THE EFW PUMP TURBINE DRIVER SHOULD BE SWITCHED FROM THE GENERATOR WITH THE TUBE LEAK TO THE "GOOD" GENERATOR TO AVOID RADIOACTIVE RELEASES FROM THE TURBINE EXHAUST TO THE ATMOSPHERE.
OTSG INVENTORY 2. LOSS OF EFW TO OTSG (MFW IS OFF)	THE RATE OF COOLDOWN WILL BE SLOWER IF ONLY ONE GENERATOR IS USED FOR COOLDOWN. THE EFFECTS WILL BE DIFFERENT DEPENDING ON AVAILABILITY OF OTHER EQUIPMENT AND WHETHER THE GENERATOR HAS THE TUBE LEAK. IF THE RC PUMPS ARE OFF, AND EFW IS AVAILABLE TO ONLY ONE GENERATOR THEN NO COOLING WILL OCCUR IN THE OTHER GENERATOR UNLESS THE GENERATOR WITHOUT EFW HAS A LARGE TUBE LEAK (LEAK WILL PROVIDE COOLING). THE REACTOR COOLANT IN THE IDLE GENERATOR LOOP WILL NOT CIRCULATE AND THE STAGNANT REACTOR COOLANT MAY FLASH INTO STEAM AS THE RCS DEPRESSURIZED. IF A SATURATED STEAM BUBBLE GROWS FURTHER, DEPRESSURIZATION WILL BE SLOW, BECAUSE THE IDLE HOT LEG WILL CONTROL THE PRESSURE. IF EFW IS ONLY AVAILABLE TO THE GENERATOR WITH THE TUBE LEAK, RELEASES WILL BE HIGH. IF REACTOR COOLANT PUMPS ARE RUNNING, TUBE TEMPERATURES WILL FOLLOW THE COOLDOWN BUT STEAM GENERATOR SHELL COOLING WILL NOT OCCUR IN THE DRY GENERATOR. THE TUBE-TO-SHELL TEMPERATURE LIMITS IN THE DRY GENERATOR MAY BE VIOLATED DURING THE COOLDOWN UNLESS THE COOLDOWN RATE IS SLOWED. IF THE COOLDOWN RATE IS SLOWED THEN THE POSSIBILITY OF RUNNING OUT OF BWST INVENTORY INCREASES.	- THE EFW CROSSOVER VALVES MUST BE REALIGNED SO THAT BOTH GENERATORS CAN BE FED. CONTROL OF EFW SHOULD BE AS LISTED IN "OTSG INVENTORY 1" ABOVE. - IF EFW CANNOT BE RESTORED TO BOTH GENERATORS THEN THE PLANT MUST BE COOLED DOWN USING THE ONE GENERATOR METHOD OUTLINED IN THE "BEST METHODS" SECTION.
OTSG INVENTORY 3. LOSS OF EFW/MFW TO BOTH OTSG'S.	PLANT COOLDOWN CANNOT BE MANUALLY STARTED UNTIL FEEDWATER IS RESTORED. CORE COOLING MUST BE MAINTAINED BY CONTINUED INJECTION OF HPI. GUIDANCE FOR HPI COOLING IS PROVIDED IN THE "BACKUP COOLING" SECTION OF THE ATOG GUIDELINES.	HPI COOLING (SEE "BACKUP COOLING") MUST BE INITIATED TO MAINTAIN CORE COOLING. WITH A TOTAL LOSS OF FEEDWATER, THE SG'S WILL BOIL DRY AND DEPRESSURIZE. SOME COOLING OR RETENTION OF A SG WATER LEVEL MAY OCCUR IN THE AFFECTED SG DUE TO THE PRIMARY-TO-SECONDARY LEAKAGE. FEEDWATER (MAIN OR EFW) SHOULD BE RESTORED TO AT LEAST ONE SG AS SOON AS POSSIBLE. A TIMELY RESTORATION OF PRIMARY TO SECONDARY HEAT TRANSFER IS NECESSARY BECAUSE HPI COOLING PLACES A BIG DEMAND ON BWST INVENTORY. SINCE THE BWST INVENTORY IS NOT RECOVERABLE (I.E., LOST TO THE SECONDARY PLANT THROUGH THE LEAK), THE POTENTIAL FOR A LOSS OF CORE COOLING CAPABILITY INCREASES PROPORTIONATELY WITH THE TIME PERIOD THAT HPI COOLING IS MAINTAINED. IF LONG TERM RELIANCE ON HPI COOLING REQUIRED, THE FOLLOWING ACTIONS/PRECAUTIONS SHOULD BE REMEMBERED. 1. BE PREPARED TO ENGAGE A SUPPLY OF BORATED WATER TO BACKUP THE BWST. 2. THE CORE EXIT THERMOCOUPLES SHOULD BE MONITORED IF THE REACTOR COOLANT RETURNS SUBCOOLED. APPROPRIATE ACTION (HPI THROTTLING) TO PREVENT RV BRITTLE FRACTURE SHOULD BE TAKEN (SEE "BEST METHODS FOR EQUIPMENT OPERATION"). 3. ALIGN EFW SUPPLY FROM ONE OF THE OTHER TWO OCONEE UNITS.
OTSG INVENTORY 4. MFW OVERFEED	A FAILURE WHICH CAUSES EXCESSIVE MFW TO ONE OR BOTH GENERATORS CAN RESULT IN WATER IN THE STEAM LINES AND CAUSE RCS OVERCOOLING. PRESSURIZER DRAINING MAY OCCUR. THE EXCESSIVE FEEDWATER ADDITION MUST BE STOPPED PRIOR TO START OF COOLDOWN. IF EXCESSIVE FEEDWATER IS ADDED TO THE GENERATOR WITH THE TUBE LEAK, THE EXCESSIVE LEVEL MUST BE STEAMED TO LOWER THE LEVEL TO NO MORE THAN 95% ON THE OPERATE RANGE. DURING THIS STEAMING PERIOD RADIOACTIVE RELEASES WILL CONTINUE. WHEN STEAM GENERATOR LEVEL REACHED 90% ON THE OPERATE RANGE, THE HIGH LEVEL LIMITER WILL TRIP THE MAIN FEEDWATER PUMPS AND THE EFW PUMPS WILL AUTOMATICALLY START.	- CORRECTIVE ACTIONS TO STOP EXCESSIVE FEEDWATER ARE SHOWN IN ATOG APPENDIX A AND OTHER INFORMATION IS SHOWN IN THE "ACCIDENT DIAGNOSIS AND MITIGATION" CHAPTER. IN SUMMARY THOSE ACTIONS ARE: 1) ATTEMPT TO CLOSE THE FEEDWATER CONTROL VALVES IF OVERFILL IS SLOW AND THE AFFECTED SG IS OBVIOUS. 2) IF THE OVERFILL IS FAST OR THE CONTROL VALVES CANNOT BE CLOSED STOP THE MFW PUMPS - START EFW AND VERIFY OPERATION - CONTROL EFW TO LIMIT OVERCOOLING - IF THE PRESSURIZER HAS DRAINED, HPI MUST BE USED TO RESTORE LEVEL. - TO LIMIT RELEASES AND TO LIMIT WATER ENTERING INTO THE STEAM LINES THE GENERATOR WITH THE TUBE LEAK SHOULD BE STEAMED TO LOWER THE WATER LEVEL TO AT MOST 95% ON THE OPERATING RANGE. INTERMITTENT OR CONTINUOUS STEAMING MAY BE REQUIRED TO MAINTAIN THE WATER LEVEL AS THE LEAK CONTINUES TO ADD WATER AS THE COOLDOWN CONTINUES. IF THE EXCESSIVE FEEDWATER WAS IN THE GENERATOR WITHOUT THE TUBE LEAK IT MAY BE STEAMED TO THE APPROPRIATE WATER LEVEL FOR OPERATION (RC PUMPS "ON" OR "OFF"). PLANT COOLDOWN SHOULD BE CONTROLLED BY THE "GOOD" STEAM GENERATOR. - TO PERMIT COOLDOWN AT A REASONABLE RATE WATER MUST BE ADDED TO THE STEAM GENERATOR WITH THE LEAK TO MAINTAIN AN ACCEPTABLE TUBE-TO-SHELL ΔT. IF THE OVERFEED WAS IN THAT GENERATOR, THERE SHOULD BE NO PROBLEM SINCE A LARGE INVENTORY HAS ACCUMULATED. HOWEVER, IF ALL FEEDWATER WAS STOPPED TO ONE GENERATOR, AND IT BOILED DRY, THEN A WATER LEVEL MUST BE ADDED TO ALLOW A REASONABLE RATE OF COOLDOWN. IF MFW CANNOT BE USED EFW CAN BE STARTED.
OTSG INVENTORY 5. EFW OVERFEED (MFW STOPPED)	- THE EFFECTS OF EFW OVERFEED ARE MUCH THE SAME AS MFW OVERFEED (OTSG INVENTORY 4). - EFW OVERFEED WILL HAVE A GREATER OVERCOOLING AND CONTRACTION EFFECT THAN MFW, CONSEQUENTLY PRESSURIZER DRAINING IS MORE LIKELY. - IF THE OVERFEEDING CONTINUES, STEAM PRODUCTION MAY STOP; WHEN THIS OCCURS THE EFW FEED PUMP TURBINE DRIVER MAY SLOW DOWN, AND PUMPED FLOW RATE MAY DECREASE. (THIS EFFECT DEPENDS ON THE VOLUME OF STEAM TRAPPED IN THE STEAM LINES). HOWEVER, WHEN THE EXCESS FEEDWATER IS STOPPED, THE EFW PUMP WILL RESUME NORMAL SPEED AS STEAM PRODUCTION PICKS UP.	- CORRECTIVE ACTIONS FOR EFW OVERFEED ARE MUCH THE SAME AS FOR MFW. THEY ARE SUMMARIZED HERE: 1) MANUALLY CLOSE THE EFW CONTROL VALVE TO THE OVERFILLING STEAM GENERATOR. 2) IF THE CONTROL VALVE CANNOT BE CLOSED, THEN MANUALLY CLOSE THE ISOLATION VALVE UPSTREAM OF THE CONTROL VALVE. - ONCE THE OVERFEED HAS BEEN STOPPED, THE GENERATOR AND FEED SYSTEM MUST BE RETURNED TO SERVICE FOR COOLDOWN. FEEDWATER CONTROL MUST BE RE-ESTABLISHED. WHEN FEEDWATER WAS STOPPED, THE GENERATOR WILL PROBABLY DRY OUT, UNLESS QUICK MANUAL CONTROL STOPPED THE FEEDWATER AND ALSO CONTROLLED THE CONTROL VALVE. TO PREPARE FOR COOLDOWN A WATER LEVEL MUST BE RE-ESTABLISHED IN BOTH GENERATORS. A WATER LEVEL MUST BE ESTABLISHED IN THE STEAM GENERATOR WITH THE TUBE LEAK TO PERMIT SHELL COOLING (MAINTAIN CORRECT TUBE-TO-SHELL ΔT) AND CONTINUOUS CONTROLLED FLOW MUST BE AVAILABLE TO THE "GOOD" GENERATOR FOR COOLDOWN. IN ORDER TO RESTORE THE FEEDWATER CONTROL THE FOLLOWING MAY BE DONE: 1) MANUALLY CONTROL THE CONTROL VALVE 2) OPEN THE CONTROL VALVE WIDE, AND MANUALLY CONTROL THE ISOLATION VALVE

TABLE C-3

EFFECTS OF FAILURES ON STEAM GENERATOR TUBE LEAK CONTROL

FAILURES WHICH AFFECT STEAM PRESSURE CONTROL

FAILURE	EFFECT ON TUBE LEAK CONTROL AND PLANT COOLDOWN	CORRECTIVE ACTION TO ALLOW A CONTROLLED COOLDOWN
OTSG PRESSURE 1. TBS FAILS OPEN	- THE STEAM PRESSURE DECREASE WILL CAUSE REACTOR COOLANT CONTRACTION AND IN COMBINATION WITH THE TUBE LEAK WILL NEARLY ALWAYS CAUSE PRESSURIZER DRAINING REGARDLESS OF THE NUMBER OF VALVES OPEN. VERY SMALL LEAKS (VALVE DOES NOT RESEAT TIGHTLY) WILL HAVE LITTLE EFFECT. - THE TIME TO DEPRESSURIZE THE GENERATOR TO ABOUT 600 PSIG FOR EXAMPLE, WILL DEPEND ON THE NUMBER OF VALVES OPEN. SEVERAL COMBINATIONS OF FAILURES ARE POSSIBLE: A. A SINGLE VALVE CAN FAIL TO CLOSE AFTER OPENING; STEAM PRESSURE (IN ONE GENERATOR) WILL DROP TO AROUND 600 PSIG 15 TO 20 MINUTES AFTER TRIP. B. A CONTROL SIGNAL ERROR CAN CAUSE THE VALVES IN ONE STEAM LINE TO FAIL OPEN; STEAM PRESSURE (IN ONE GENERATOR) WILL DROP TO 600 PSIG 8 TO 12 MINUTES AFTER TRIP. C. A CONTROL SIGNAL ERROR CAN CAUSE ALL VALVES IN BOTH STEAM LINES TO FAIL OPEN; STEAM PRESSURE (IN BOTH GENERATORS) WILL DROP TO 600 PSIG 3 TO 5 MINUTES. - THE EFFECT ON THE AMOUNT OF RCS CONTRACTION AND PRESSURIZER DRAINING WILL DEPEND ON THE NUMBER OF VALVES OPEN, WHETHER FEEDWATER IS STOPPED OR CONTINUED, AND THE AMOUNT OF HPI FLOW. - SUBCOOLING MARGIN WILL NOT BE LOST FOR SINGLE VALVE OPENINGS IF HPI IS STARTED, BUT IT MAY BE LOST WHEN ALL THE TURBINE BYPASS VALVES FAIL OPEN. IF THE SUBCOOLING MARGIN IS LOST RECOVERY WILL OCCUR WHEN HPI IS ABLE TO REFILL THE SYSTEM AND RESTORE PRESSURIZER LEVEL. - IF HPI IS ON, THE SUBCOOLING MARGIN WILL BE RECOVERED (IF IT WAS LOST) AND THE AMOUNT OF SUBCOOLING WILL INCREASE. THE TUBE LEAK FLOW WILL GET LARGER WHEN REACTOR COOLANT TEMPERATURE DROPS. THE TUBE LEAK FLOW WILL GET LARGER IF BOTH REACTOR COOLANT TEMPERATURE DROPS AND THE REACTOR COOLANT PRESSURE INCREASES. - IF HPI AND OVERCOOLING CONTINUE AN NDT VIOLATION WILL OCCUR. - LOOP VOIDS CAN FORM BECAUSE OF THE STEAM PRESSURE FAILURE; THE TUBE LEAK WILL ALSO TEND TO REDUCE SUBCOOLING. THE SIZE OF THE VOIDS FORMED WILL DEPEND ON THE NUMBER OF VALVES OPEN, THE SIZE OF THE TUBE LEAK AND HPI FLOW. NATURAL CIRCULATION CAN BE LOST IF VOIDS IN THE LOOP ARE LARGE ENOUGH, AND VOIDS CAN FORM IN THE REACTOR VESSEL HEAD FOR THE LARGEST STEAM PRESSURE LOSSES. IF VOIDS ARE FORMED IN THE HEAD AND REACTOR COOLANT PUMPS ARE NOT OPERATING, THE STEAM BUBBLE WILL LIMIT THE COOLDOWN BECAUSE IT WILL NOT PERMIT DEPRESSURIZATION. HPI ALONE CANNOT ELIMINATE THE VOIDS; HEAT MUST BE REMOVED BY THE STEAM GENERATOR TO COMPLETELY ELIMINATE VOIDS, BUT HPI WILL REDUCE THE VOLUME OF THE VOIDS. - LARGE STEAM LEAKS WILL CAUSE AN UNCONTROLLED COOLDOWN. COOLDOWN WITH OPEN TBS VALVES WILL REQUIRE THAT THE FAILED TBV(S) BE BLOCKED CLOSED, THE VALVES BE REPAIRED SO THAT THEY CAN BE USED FOR A CONTROLLED COOLDOWN, OR THAT FEEDWATER BE VERY CAREFULLY CONTROLLED TO A DEPRESSURIZED GENERATOR. REACTOR COOLANT OVERCOOLING CAN VIOLATE THE TUBE-TO-SHELL ΔT LIMITS AND PLACE THE TUBES IN TENSION. THIS MAY INCREASE THE TUBE LEAK RATE IF THE TUBE CRACK IS PERPENDICULAR TO THE CENTERLINE OF THE TUBE. - IF THE CONDENSER IS WORKING, THE FAILED TURBINE BYPASS VALVES WILL DISCHARGE TO THE CONDENSER; RADIATION RELEASES WILL BE REDUCED BY PARTITIONING. IF THE CONDENSER IS NOT AVAILABLE, THE TURBINE BLOCK VALVES WILL ISOLATE THE LEAK AND RESTORE STEAM PRESSURE, BUT RELEASES WILL GO TO THE ATMOSPHERE THROUGH THE STEAM SAFETY VALVES AND THE MADV'S (IF OPERABLE). COOLDOWN WILL BE VERY SLOW. HOWEVER, LOSS OF THE CONDENSER IS VERY UNLIKELY SINCE THE GRAVITY FEED CIRCULATING WATER SYSTEM WILL SUPPLY BACKUP COOLING WATER. LOSS OF GRID POWER WILL AUTOMATICALLY SWITCH TO THE BACKUP SUPPLY FROM THE NORMAL CIRCULATING WATER PUMPS. HOWEVER, LOSS OF GRID POWER WILL TRIP THE AIR COMPRESSOR SUPPLYING AIR TO THE TBS; WHEN THIS OCCURS THE TBS WILL SHUT. EITHER THE NORMAL AIR SUPPLY MUST BE LOADED ON THE HYDRO BUS MANUALLY OR THE BACKUP AIR SUPPLY MUST BE STARTED SO THAT THE TBS CAN BE CONTROLLED.	- THE STEAM LEAK WILL COOL THE REACTOR COOLANT TO 500F (OR LOWER DEPENDING ON THE LEAK RATE). IF THE COOLDOWN RATE IS RELATIVELY SLOW, THE OPERATOR CAN DEPRESSURIZE WITH THE PRESSURIZER SPRAYS, BUT CAREFUL CONTROL OF HPI TO MAINTAIN PRESSURIZER LEVEL IS NEEDED. IF THE COOLDOWN RATE IS FAST IT IS BEST TO RESTORE STEAM PRESSURE CONTROL BEFORE DEPRESSURIZING THE RCS. - TO DETERMINE WHICH GENERATOR LOOP HAS THE STEAM LEAK: - CHECK STEAM PRESSURE, DETERMINE WHICH GENERATOR HAS THE LOWEST PRESSURE, OR - STOP FEEDWATER TO BOTH GENERATORS (WHEN LEVEL EXISTS IN BOTH); THE GENERATOR WITH THE FASTEST LEVEL DECREASE IS LEAKING. (NOTE: IF A LARGE TUBE LEAK EXISTS IN THE DEPRESSURIZED GENERATOR IT WILL COOL FASTER AND DEPRESSURIZE MORE THAN THE "GOOD" GENERATOR, BUT LEVEL MAY NOT LOWER SIGNIFICANTLY). - ONCE THE GENERATOR WITH THE STEAM LEAK HAS BEEN DETERMINED, STOP FEEDWATER TO THAT GENERATOR (OR IF ALL FEEDWATER HAS BEEN STOPPED, RESTORE FEEDWATER TO THE GOOD GENERATOR) AND STABILIZE THE REACTOR COOLANT WITH HPI; CONTROL REHEAT REPRESSURIZATION BY CONTROLLING HPI AND BY CONTROLLING STEAM PRESSURE WITH THE TBS ON THE GENERATOR THAT DOES NOT HAVE THE STEAM LEAK. - TO DETERMINE WHETHER THE LEAK IS THROUGH THE TBS OR THE STEAM SAFETY VALVES. - ATTEMPT TO MANUALLY CLOSE THE TBS VALVES IN THE GENERATOR LOOP WITH THE STEAM LEAK (NOTE: IF THE TBV'S ARE VERIFIED AS BEING SHUT AND THE OTSG REPRESSURIZES, THEN THE SAFETY VALVES CAN BE ELIMINATED. - VISUALLY EXAMINE THE TBS VALVES LOCALLY. - ONCE THE PLANT HAS BEEN STABILIZED, CORRECTIONS CAN BE MADE TO PERMIT COOLDOWN. THE METHOD FOR COOLDOWN WILL DEPEND ON WHETHER THE STEAM LEAK CAN BE CONTROLLED AND WILL DEPEND ON WHICH GENERATOR HAS THE TUBE LEAK. (THE "MODE" FOR COOLDOWN IS GIVEN IN FIGURE C-3, SHEET 2). BEFORE COOLDOWN IS BEGUN THE TBS MUST EITHER BE RESTORED TO SERVICE OR MANUALLY CONTROLLED. - TO RESTORE THE TBS TO SERVICE ATTEMPT MANUAL CONTROL WITH THE HAND/AUTO STATION; IF THIS DOES NOT WORK THE BLOCK VALVES SHOULD BE MANUALLY CONTROLLED AS A SUBSTITUTE FOR THE TBS VALVES. THE GENERATOR WITH THE TUBE LEAK SHOULD ONLY BE STEAMED TO LIMIT THE WATER LEVEL AND THE VALVES SHOULD ONLY BE OPENED AS NECESSARY TO KEEP THE WATER LEVEL LOW. COOLDOWN SHOULD BE WITH THE GENERATOR WITHOUT THE TUBE LEAK UNLESS THE PROPER RATE OF COOLDOWN CANNOT BE MAINTAINED. - IF AT ANY TIME DURING THE COOLDOWN THE WATER LEVEL OF THE AFFECTED GENERATOR APPROACHES 95% ON THE OPERATE RANGE BECAUSE OF THE TUBE LEAK, THAT GENERATOR MUST BE STEAMED TO REDUCE THE LEVEL. FEEDWATER SHOULD BE LIMITED TO THE GENERATOR WITH THE TUBE LEAK TO AVOID FILLING THE GENERATOR. ENOUGH WATER LEVEL SHOULD BE MAINTAINED TO KEEP THE TUBE-TO-SHELL ΔT IN LIMITS AND TO MAINTAIN NATURAL CIRCULATION. - IF THE OVERCOOLING HAS CAUSED VOIDS TO FORM IN LOOPS THEN SATURATION HAS EXISTED AND THE RC PUMPS HAVE BEEN TRIPPED. THE HPI SYSTEM SHOULD BE STARTED TO REFILL THE SYSTEM AND TO RESTORE SUBCOOLING. ALTHOUGH SUBCOOLING IS RESTORED, THE STEAM BUBBLES MAY STILL EXIST IN THE UPPER LOOPS OR IN THE HEAD. VOIDS CAN BE REMOVED BY COOLING WITH THE STEAM GENERATORS. IF THE SYSTEM IS SUBCOOLED, THE RC PUMPS CAN BE RESTARTED AND RUN TO "WASH" THE STEAM BUBBLE INTO THE STEAM GENERATOR WHERE IT CAN BE CONDENSED BY STEAM GENERATOR COOLING. IN NEARLY ALL CASES FOR TUBE LEAKS THE HPI SYSTEM WILL RESTORE SUBCOOLING; SATURATION WILL CONTINUE ONLY IF VERY LARGE TUBE LEAKS EXIST (MORE THAN ONE TUBE) OR IF ANOTHER LOCA HAS ALSO OCCURRED. IN THESE REMOTE CASES THE "PUMP RESTART CRITERIA" OF THE BEST METHODS" CHAPTER OF ATOG PART II SHOULD BE USED. - IF THE RC PUMPS CANNOT BE RESTARTED, AND THE OVERCOOLING HAS CAUSED VOIDS, THEN DEPRESSURIZATION CANNOT BE ACCOMPLISHED. BACKUP SUPPLIES OF BORATED WATER FOR INJECTION SHOULD BE PREPARED.
OTSG PRESSURE 2. MAIN STEAM RELIEF VALVE FAILS OPEN	- THE EFFECTS OF ONE STEAM SAFETY VALVE FAILING OPEN ARE THE SAME AS IF ONE TBS VALVE FAILED OPEN (SEE OTSG PRESSURE 1) EXCEPT THAT A MAIN STEAM SAFETY VALVE CANNOT BE CLOSED OR BLOCKED. - AS LONG AS FEEDWATER CONTINUES TO THE DEPRESSURIZED GENERATOR, OVERCOOLING WILL CONTINUE. - SINCE THE STEAM SAFETY VALVE CANNOT BE REPAIRED, A ONE GENERATOR COOLDOWN WILL BE REQUIRED. THE GENERATOR TO BE USED FOR COOLDOWN MAY OR MAY NOT BE THE ONE WITH THE TUBE LEAK. - IF THE TUBE LEAK IS IN THE SAME GENERATOR AS THE FAILED SAFETY VALVE THEN RADIOACTIVE RELEASES WILL BE VERY HIGH AS LONG AS THE TUBE LEAK CONTINUES AND THE GENERATOR STEAMS.	THE GENERAL APPROACH FOR COOLDOWN IS: - DETECT THE STEAM LEAK AS OUTLINED IN OTSG PRESSURE 1 - STOP THE OVERCOOLING TRANSIENT BY STOPPING ALL FEEDWATER TO THE GENERATOR WITH THE STEAM LEAK. - PROCEED WITH A ONE GENERATOR COOLDOWN AS OUTLINED IN THE "BEST METHODS" CHAPTER ATOG PART II. - CORRECTIVE ACTIONS FOR VOIDS SHOULD BE PERFORMED, IF THEY OCCUR AS OUTLINED IN OTSG PRESSURE 1.
OTSG PRESSURE 3. FAILED CLOSED TBV(s)	FAILURES RESULTING IN CLOSURE OF THE TBV'S CAN PRODUCE THE FOLLOWING CONDITIONS: 1. ALL TBV'S ON BOTH SG'S FAIL CLOSED. 2. ALL TBV'S ON ONE SG FAIL CLOSED. 3. ONE TBV FAILS CLOSED. CLOSURE OF THE TBV(S) PREVENTS STEAM DUMP TO THE CONDENSER AND PLACES MORE RELIANCE OF USE OF SECONDARY SAFETY VALVES TO CONTROL/LIMIT STEAM PRESSURE. CLOSURE OF ALL THE TBV'S ON BOTH STEAM GENERATORS REDUCES AUTOMATIC STEAM PRESSURE CONTROL SOLELY TO THE SSVs. IMMEDIATELY FOLLOWING A REACTOR TRIP, THIS FAILURE DOES NOT PRODUCE ABNORMAL SYSTEM BEHAVIOR (AN OVERCOOLING OR OVERHEATING TRANSIENT). PLANT COOLDOWN IS NOT POSSIBLE, HOWEVER, UNTIL VARIABLE STEAM PRESSURE CONTROL IS RESTORED. IF PLANT COOLDOWN IS DELAYED FOR AN EXTENDED TIME PERIOD, THE FOLLOWING CONSEQUENCES CAN EVOLVE: 1. THE LEAK RATE WILL CONTINUE, AND BWST INVENTORY WILL SLOWLY DECREASE. THE EVENTUAL LOSS OF MU CAPABILITY CAN LEAD TO AN ICC CONDITION. 2. OVERFILL OF THE AFFECTED SG AND WATER ADDITION TO THE STEAM LINE, DUE TO PRIMARY TO SECONDARY LEAKAGE, CANNOT BE PREVENTED. 3. FEEDWATER SOURCES WILL BE SLOWLY DEPLETED; A LOSS OF ALL FEEDWATER WILL REQUIRE DEPENDENCE ON HPI TO KEEP THE CORE COOLED. FAILURE OF ALL THE TBV'S ON ONE GENERATOR WILL REQUIRE A ONE SG COOLDOWN. IF THE AFFECTED SG MUST BE USED FOR PLANT COOLDOWN, HIGHER STEAM PLANT CONTAMINATION AND OFFSITE RELEASES WILL OCCUR. FAILURE OF ONE TBV DOES NOT IMPAIR PLANT COOLDOWN CAPABILITIES IN THAT EACH LOOP HAS AMPLE ADDITIONAL CAPACITY.	AT LEAST ONE TBV MUST BE OPERATIONAL TO COOLDOWN THE PLANT SINCE OPERATION OF THE ADV'S IS PRESENTLY QUESTIONABLE. IT IS ALSO DESIRABLE TO RESTORE CONTROL TO AS MANY TBV'S AS POSSIBLE TO ACHIEVE MAXIMUM COOLDOWN RATES AND USE OF NEAR NORMAL COOLDOWN ACTIONS. IF TBV FAILURES ARE INDICATED, THE FOLLOWING CORRECTIVE ACTION SHOULD BE ATTEMPTED: 1. ATTEMPT REMOTE MANUAL CONTROL FROM THE CONTROL ROOM. 2. CHECK THE STATUS OF THE AIR SUPPLY SYSTEM AND START THE BACKUP AIR COMPRESSOR IF NECESSARY. 3. IF ALL TBV'S ARE FAILED CLOSED AND ITEM 1&2 ABOVE FAIL, ACTIONS TO RETURN THE ADV TO SERVICE SHOULD BE STARTED. 4. IF ONE OR MORE TBV ARE OPERATIVE ON ONE SG, A PLANT COOLDOWN PER THE ONE GENERATOR METHOD OUTLINED IN THE "BEST METHOD" SECTION SHOULD BE CONDUCTED (EVEN IF THE AFFECTED SG MUST BE CONTINUOUSLY STEAMED TO COOLDOWN THE PLANT). IF TBS OPERATION IS TEMPORARILY LOST, PREPARATIONS TO ENGAGE A BACKUP SUPPLY OF BORATED WATER TO THE BEST AND ALTERNATE FEEDWATER SOURCES SHOULD BE MADE. ONCE CONTROL OF THE TBV'S OR ADV'S IS RESTORED, AN IMMEDIATE PLANT COOLDOWN AND DEPRESSURIZATION SHOULD BE STARTED.

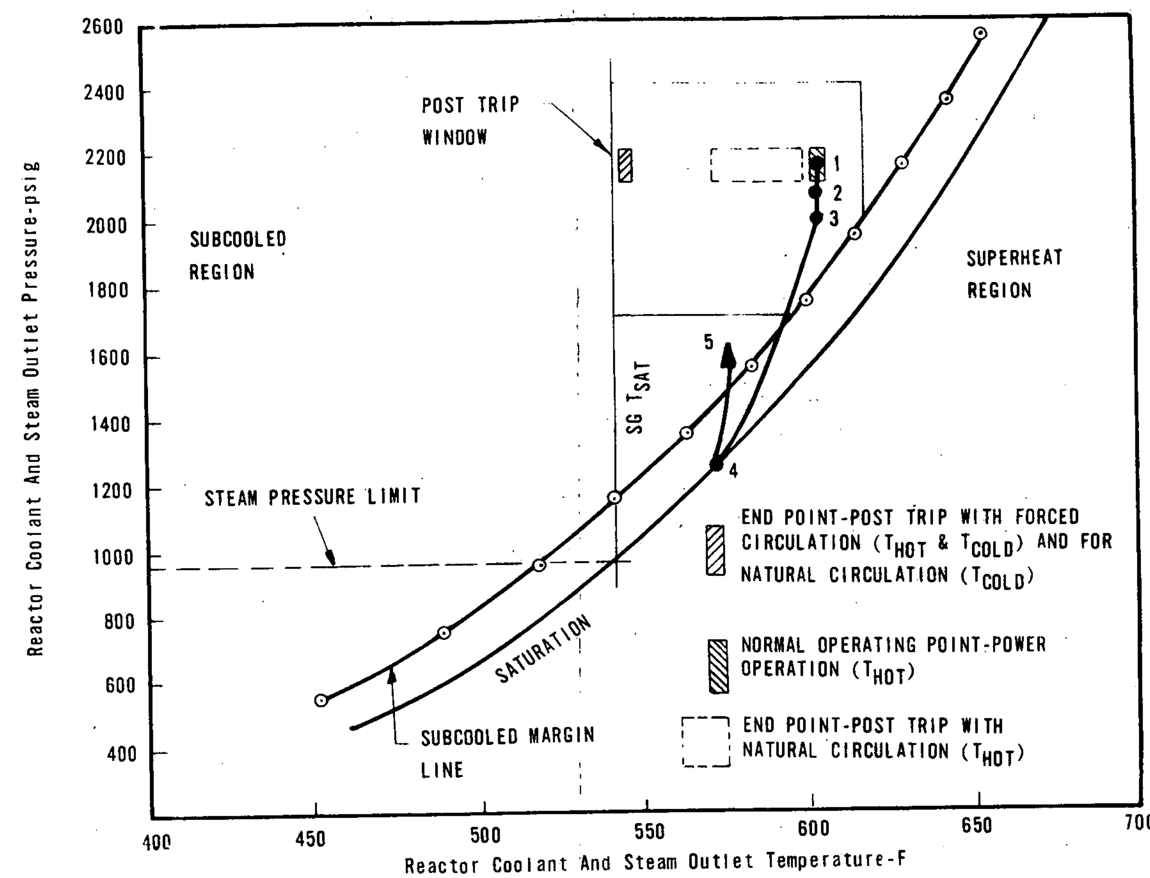
TABLE C-2 ACTIONS FOR A SGTR IN COMBINATION WITH A NON-ISOLABLE STEAM LEAK

	LARGE SGTR[1] WITH				SMALL SGTR[2] WITH			
	LARGE STEAM LEAK[3] ON		SMALL STEAM LEAK[4] ON		LARGE STEAM LEAK[3] ON		SMALL STEAM LEAK[4] ON	
	AFFECTED[5] SG	UNAFFECTED[6] SG	AFFECTED[5] SG	UNAFFECTED[6] SG	AFFECTED[5] SG	UNAFFECTED[6] SG	AFFECTED[5] SG	UNAFFECTED[6] SG
Steam Leak Mitigation Actions	Stop FW to affected SG (tube leak will provide cooling)	Maintain minimal FW to unaffected SG[9]	N/A	N/A	Maintain minimal FW to affected SG[9]	Maintain minimal FW to unaffected SG[9]	N/A	N/A
SG(s) removing heat when cooldown and depressurization started	Both SG's	Predominantly affected SG	Both SG's	Both SG's	Predominantly unaffected SG	Predominantly affected SG	Both SG's	Both SG's
SGTR Isolation Actions	N/A	N/A	• Stop or reduce feedwater to affected SG when $T_{hot} \sim$ 500F (tube leak will provide cooling)[9]	Stop or reduce feedwater to affected SG when $T_{hot} \sim$ 500F (tube leak will provide cooling)[9]	N/A	N/A	• Stop steam from and FW to affected SG when $T_{hot} <$ 540F.[10]	Stop steam from and FW to affected SG when $T_{hot} <$ 540F.[10]
Cooldown Rates	Emergency[7]	Emergency	Emergency	Emergency	Emergency	• Emergency[7] if condenser is not available. • Normal[8] if condenser is working.	Emergency	• Emergency[7] if condenser is not available. • Normal[8] if condenser is working.

NOTES:

1. Large SGTR corresponds to a primary to secondary leak rate which requires HPI to control RCS inventory.
2. Small SGTR corresponds to a primary to secondary leak rate which is within the capacity of the normal MU system.
3. A large steam leak reduces steam pressure more than 200 psi.
4. A small steam leak reduces steam pressure less than 200 psi and allows the reactor coolant to stabilize at a slightly reduced temperature condition.
5. Affected SG is the OTSG with the SGTR.
6. Unaffected SG is the OTSG without the SGTR.
7. Emergency cooldown means:
   - Tube-to-shell temperature limits do not exceed 150F
   - Fuel decompression limits may be violated
   - The RC temperature should be dropped to $\sim$ 500F as fast as possible
   - After 500F is reached, the cooldown should be at 100F/hr as long as that rate can be held. The cooldown rate should be reduced when the tube-to-shell temperature limit approaches 150F (unless there is danger of draining the BWST).
8. Normal cooldown means:
   - Tube-to-shell temperature limits do not exceed 60F
   - Fuel decompression limits should not be violated
   - The cooldown is at a "normal" rate
9. Provide minimal or periodic feeding to maintain tube-to-shell ΔT and/or natural circulation if RCP's are tripped.
10. If RCP's are tripped, provide minimal or periodic feeding and steaming to maintain natural circulation in affected loop.

Figure C-2 TYPICAL SYSTEM RESPONSE FOR LARGE SGTR WHICH RESULTS IN A REACTOR TRIP



Reference Points	Time (Seconds)	Remarks
1	0	A SGTR occurs from 100% FP. (Initial leak rate ~ 400 gpm)
1-2	0-440	RC pressure and pressurizer level slowly drop. • Pressurizer heaters come on; • High MU flow alarm occurs. Steam line and condenser radiation alarms sound.
2	440	Pressurizer level falls below 40 inches; pressurizer heaters turn off automatically.
2-3	440-660	RC pressure and pressurizer level continue to drop because the leak rate exceeds MU.
3	660	Reactor trips on variable low pressure (P_{RCS} ~ 2000 psig).
3-4	660-680	The turbine trips; MFW runs back; the reactor coolant subcooled margin is lost (RC pump trip required); ESAS actuated on low RC pressure; HPI starts.
4	680	The pressurizer completely drains and the reactor coolant is saturated.
4-5	680-900	The RC pumps are tripped and EFW starts. The operator throttles EFW to prevent overcooling. The reactor coolant subcooled margin is reestablished, and the operator throttles HPI to stablize pressurizer level and system repressurization.
5	900	The plant is stable with decay heat being removed by natural circulation. The RC pumps can be restarted and a plant cooldown and depressurization can be initiated. A minimum subcooled margin is maintained to keep RC pressure and leak rate low.

Figure A-6 TIME RELATIONSHIP OF KEY PARAMETERS

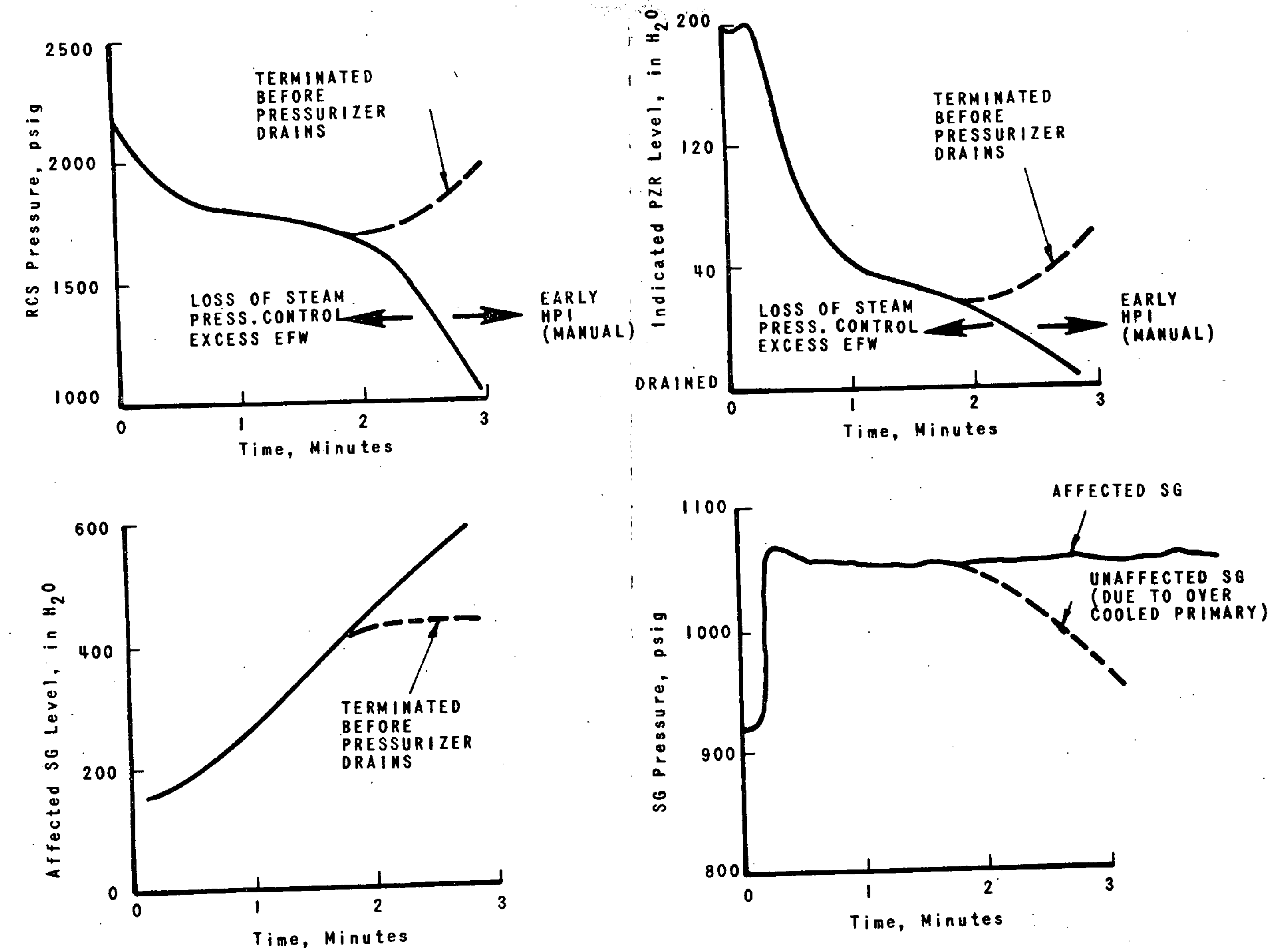
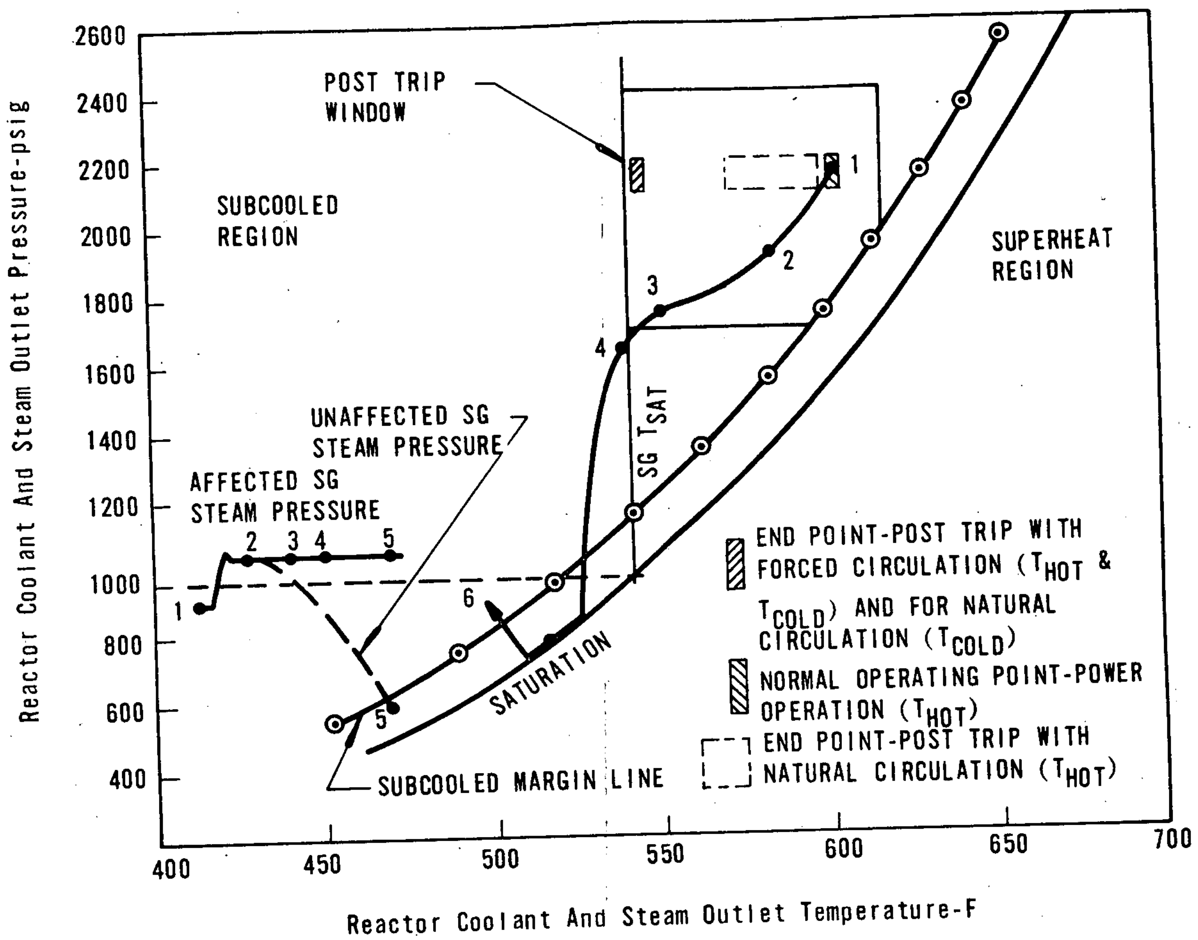
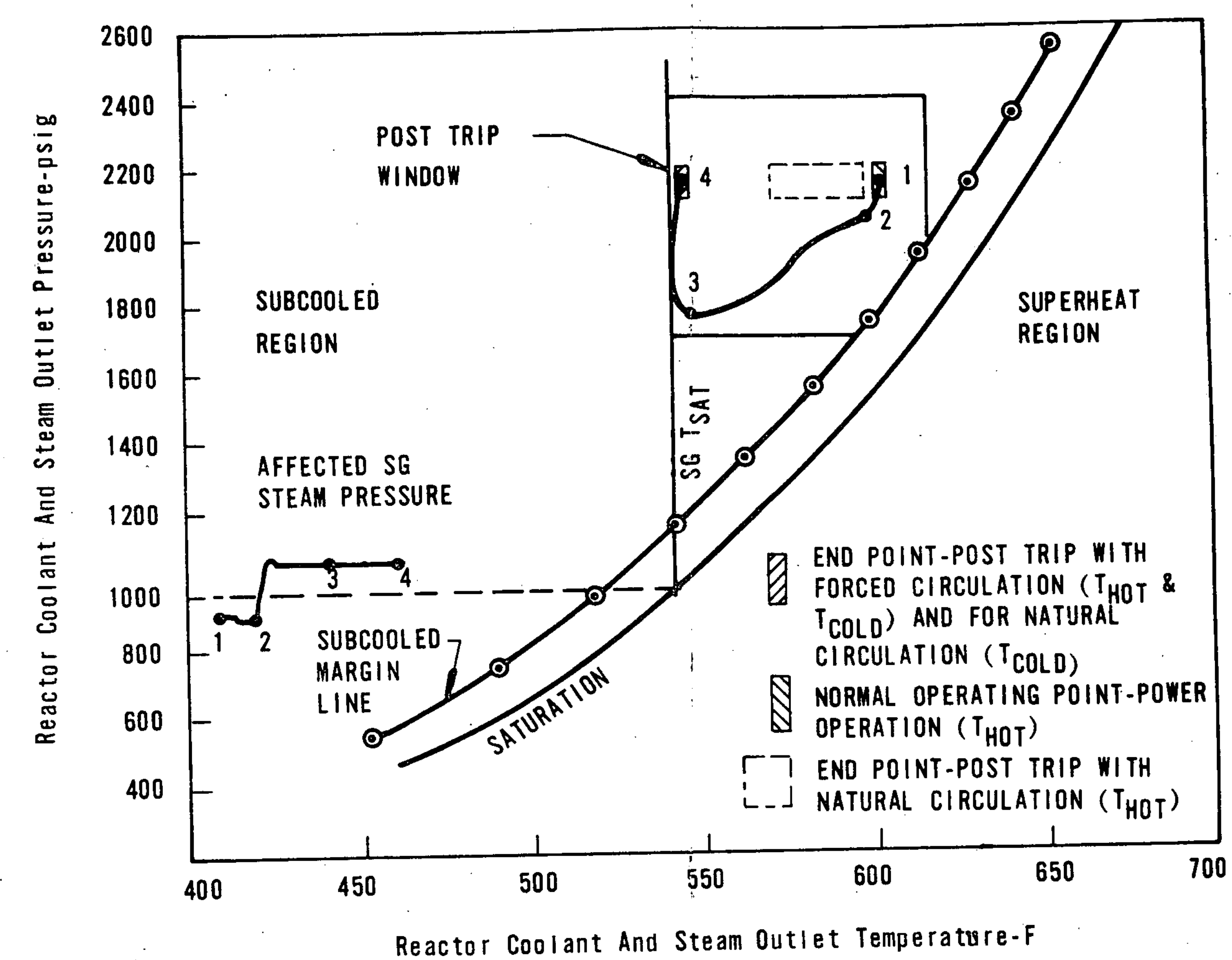


Figure A-2 EXCESSIVE FEEDWATER NOT TERMINATED BY ICS



Reference Points	Time (Seconds)	Remarks
1	0	Reactor trip from 100%. ICS fails to run back MFW to OTSG A and begins to overfeed OTSG A.
2	60	Operator senses rapidly decreasing pressurizer level and RCS pressure and starts HPI. OTSG A level 70% on operate range and increasing.
3	180	OTSG A full. High level trip of the MFW pumps fails.
4	250	Pressurizer empty. RCS rapidly approaches saturation. On loss of subcooling margin, the operator trips RC pumps and MFW is diverted to upper nozzels but also continues to feed through main block valve on OTSG A.
5	330	MFW flow to OTSG B and RCS cooling cause steam pressure to decrease. The operator notices the high OTSG A level and continuing MFW flow and acts to stop FW addition to OTSG A.
6	410	The excessive MFW to OTSG A has terminated and the operator has terminated FW to OTSG B at 95% on the operate range. Refill of the pressurizer by HPI has begun to repressurize the RCS and subcooling is regained.

Figure A-1 EXCESSIVE FEEDWATER TERMINATED BY ICS



Reference Points	Time (Seconds)	Remarks
1	0	Excessive feedwater addition begins.
1-2	0-60	Slight overcooling of RCS occurs due to excessive feedwater addition. ICS pulls rods to compensate for reduction of T_{ave}.
2	60	Reactor trip on high flux or low pressure. (Note: Depending on severity and power history, a reactor trip may or may not occur.)
2-3	60-200	RC P&T decrease due to loss of fission power and higher than normal secondary inventory. The ICS initiates a feedwater runback and the MFW addition stops. Pressurizer level decreases because of reactor coolant contraction.
3	200	Minimum pressurizer level reached.
3-4	>200	Normal system pressure restored by RCS reheating, operation of MU system, and pressurizer heaters. Primary system is left in a stable, hot shutdown condition.