BABCOCK & WILCOX
NUCLEAR POWER GENERATION DIVISION

TECHNICAL DOCUMENT

NUMBER

76-1123298-00

ATOG

ABNORMAL TRANSIENT

OPERATING GUIDELINES

PART II

VOLUME 1

FUNDAMENTALS OF REACTOR CONTROL

FOR

ABNORMAL TRANSIENTS

DATE: 3-23-82

PAGE

BABCOCK & WILCOX
NUCLEAR POWER GENERATION DIVISION

NUMBER
76-1123298-00

RECORD OF REVISION

REV. NO.	CHANGE SECT/PARA.	DESCRIPTION/CHANGE AUTHORIZATION
00		Original Issue - J. J. Kelly Plant Engineering

Prepared by Joseph J Kelly, Project Engineer Date 3/22/82
Reviewed by [signature], Operations Support Serv. Date 3/22/82
Approved by [signature] Manager, OPS Date 3/25/82
Approved by E. H. Dowling, Mgr Ops & Cont. Eng Date 3-26-82

ATOG GUIDELINES
PART II

VOLUME 1 FUNDAMENTALS OF REACTOR CONTROL FOR ABNORMAL TRANSIENTS

List of Figures

VOLUME 1 FUNDAMENTALS OF REACTOR CONTROL FOR ABNORMAL TRANSIENTS

List of Tables

ATOG GUIDELINES PART II

INTRODUCTION

The accident at Three Mile Island has caused the Nuclear Industry's perspective of emergency operation to change. That accident was difficult for the plant operators to handle because several things were happening at once. Loss of Main Feedwater, Loss of Emergency Feedwater, and Small Break LOCA occurred at the same time. An incorrect interpretation of pressurizer level misled the operator to think the core was covered when it was not. The operator acted on that misleading information and core cooling was stopped when he shut down Emergency High Pressure Injection and the Reactor Coolant Pumps. The combination of multiple failures and incorrect interpretation of information are the two factors which have caused a new perspective of emergency operation to be developed.

In the past emergency procedures and operator training concentrated on single event accidents. But accidents do not usually happen with only single failures; several things often go wrong at the same time. These guidelines have been developed so that operator can understand what has gone wrong so he can circumvent failures and still keep the core cool with the available equipment.

When failures of equipment occur they frequently cause a change in the heat transfer from the core to the steam generators. When the reactor is

operating normally all the heat produced by the core is being removed by the steam generators; primary and secondary system pressures, temperatures, and levels are stable. Heat transfer is balanced. Any transient will cause an upset in the heat transfer from the core to the steam generators and the main objective of emergency procedures is to restore and maintain adequate core cooling. Heat transfer will be affected in different ways depending on what equipment has operated incorrectly. When the heat transfer changes the effects will show up in primary and secondary system pressures, temperatures, and levels. Pressures and temperatures are parameters from which three basic symptoms of improper heat transfer can be derived and used to discover what has gone wrong. These guidelines will use those heat transfer symptoms as the source of information for the operator action. Recognition of just three basic heat transfer symptoms will given the knowledge needed so the operator can restore and maintain adequate core cooling.

Correlation Between Part I and Part II

Use of symptom-oriented procedures involves a different approach to plant control and requires a shift in emphasis in operator training. The operator is no longer limited by, nor is he required to solely rely on, the designer's foresight in providing the key alarms and indications for every conceivable event that could occur. The symptom approach of Part I will work, regardless of the event. By training on the ATOG approach, the operator will have a thorough understanding of heat transfer, plant control, and the various options available for controlled core cooling when

systems and equipment fail. Part II of ATOG was written to provide the basis for this understanding with the intent that it be used as part of the operator training program.

Part II, Volume 1

Volume 1, "Fundamentals of Reactor Control for Abnormal Transients," provides the basic background necessary for understanding heat transfer and builds on this information to enable the operator to recognize abnormal conditions when they develop and take the appropriate actions to correct them. Volume 1 covers primarily information regarding the heat transfer process and mechanisms, including subcooling and natural circulation. Volume 1 also shows how to use the P-T diagram and this knowledge of heat transfer to diagnose abnormal transients and mitigate them.

The preferred method of core cooling is with controlled primary to secondary heat transfer and many abnormal transients involve restoring a balance to this heat removal path. However, this is not always possible; therefore, Volume 1 discusses core cooling methods when the steam generators are not available.

Volume 1 also covers operational methods for key systems (feedwater, HPI, etc.) and equipment for various conditions, provides guidance on verification of plant stability, discusses the fundamentals of reactor building control, and provides instructions on the use of the guidelines.

Volume 1 contains a considerable amount of information and should be studied periodically for optimum comprehension and retention. Three major points should be kept in mind when reading Volume 1:

1. Understanding heat transfer is essential.
2. Relationship of symptoms and control functions.
3. Differentiating rules and guidelines.

Heat Transfer

One aspect of plant control and the use of ATOG cannot be overstressed: the importance of understanding heat transfer and primary pressure-temperature relationships. A thorough grasp of the heat transfer process and P-T relationships will enable the operator to:

- recognize abnormal conditions (symptoms)
- evaluate plant response to corrective actions
- implement backup cooling methods when needed

Although virtually any event or combination of events could conceivably occur, they all present the common threat of disrupting core cooling. Thus, the major thrust of ATOG is to maintain some form of controlled core cooling, whether it be by the steam generators or ECC systems. Simply put, understanding heat transfer allows recognition of symptoms of abnormal transients. Recognition of symptoms allows implementation of the appropriate sections of Part I. Implementation of the appropriate sections of Part I and verification of plant response allows stabilization and restoration of controlled core cooling.

Symptoms and Control Functions

Section III of Part I of the guidelines is divided into four main sections to address the basic heat transfer symptoms (lack of subcooled margin, overheating, and overcooling) and the special case of SG tube rupture. Part II discusses the importance of "control functions". These control functions are:

1. RC inventory
2. RC pressure
3. SG inventory
4. SG pressure

A fifth control function, reactivity, is also important in heat transfer considerations. However, reactivity is quickly controlled by reducing core power to the decay heat level by automatic reactor trip, manual reactor trip, and/or emergency boration.

If control of one of these four functions is lost, it will impact primary to secondary heat transfer and become evident as one of the three symptoms. For example, a loss of SG inventory control low (loss of feedwater) will result in a loss of primary to secondary heat transfer (overheating). Conversely, a loss of SG inventory high (too much feedwater) will result in excessive primary to secondary heat transfer (overcooling).

When a symptom appears, one or more of these functions are not being controlled properly. Regaining control of these four functions will restore controlled core cooling.

Rules and Guidelines

Volume 1 provides guidance on the operation of systems and equipment for many conditions and various events. When an action must always be taken for the conditions specified it is called a rule and is enclosed in a box for emphasis. For example, the RC pumps must always be tripped whenever the subcooled margin is lost; therefore, it becomes the RC pump trip rule.

Whenever specified actions are recommended, but not always mandatory, they are considered guidelines. For example, Table 6 in Volume 1 provides guidelines for RC pump operation for different plant conditions.

ATOG was designed for maximum flexibility in order to address the spectrum of conceivable transients. Therefore, rules have been kept to the minimum necessary. The user should remember, however, that the guidelines are also important and should be followed whenever they are applicable and feasible.

Part II, Volume 2

Volume 2, "Discussion of Selected Transients," provides detailed coverage of six specific initiating events. Although the ATOG concept is a break from the traditional event-oriented approach, Volume 2 was structured in this manner to meet the following objectives:

1. Validate the ATOG Concept

   Most operators involved in the initial implementation of the ATOG concept will be experienced with use of event-oriented procedures

and may understandably be resistant to a different approach. Volume 1 discusses ATOG in a general overview manner and, standing alone, may not fully promote user confidence in the concept.

Therefore, Volume 2 is provided to give examples of representative events and how the use of ATOG will lead to successful mitigation. Although the event is given, the diagnosis and mitigation is written with the assumption that the operator (in the example) is unaware of the specific cause.

In addition, the discussion on each transient demonstrates successful mitigation of events compounded by other failures using the same basic ATOG procedure. This highlights the relative simplicity of using a single, comprehensive procedure as opposed to several discrete procedures.

The transients depicted in Volume 2 are derived from more realistic analyses than previously used for design bases accident analyses. Thus these transient discussions should give the operator a better feel for how the plant would actually respond should similar conditions occur. Where available, actual plant data from representative transients is used.

2. Amplify Volume 1

The structure of Volume 2 provides a ready vehicle for conveying more detailed information about transient types (e.g., overcooling) and peculiarities and complexities of specific events. This is especially true for the appendices covering SG tube rupture and small break LOCA. These two events are unique in that they cannot be quickly terminated and stabilized. They impact many facets of plant operation and their mitigation is highly dependent on specific conditions at the time of occurrence (including the size of the leak). Consequently, considerably more event specific information is provided in these two appendices.

The entire purpose of these guidelines is to give an overview on reactor transients, their diagnosis and control, so transients as severe as the Three Mile Island accident will be prevented. Because transients will not follow a planned course, anything can happen. Consequently, a symptom-oriented approach is necessary to ensure transient control.

These guidelines should provide enough background and understanding so that no matter what happens, the operator will have sufficient understanding to correctly respond to the transient using the principles of heat transfer control.

CHAPTER A

BASIC HEAT TRANSFER

Introduction and Summary

This section of the guidelines gives the basic principles of heat transfer that are important for removing heat from the core so that it can be properly cooled. The chapter is divided into three parts: 1) "Basic Heat Transfer", 2) Addendum A - "Subcooled, Saturated, Superheated Water", and 3) Addendum B - "Natural Circulation". Addendum A and Addendum B give information on two general subjects. The part on "Basic Heat Transfer" covers two related topics: 1) the general process for heat removal through the steam generators, and 2) the ways the operator can control that heat transfer.

The preferred way to protect the core and prevent fuel failure is to control the rate of heat removal by transferring core heat to the steam generators. Other ways to protect the core do exist; they are covered in a later section call "Backup Cooling Methods".

To control core heat removal with the steam generator the operator should balance the heat generated by the core with the heat removal through the steam generators. This section will show the fundamentals of heat transfer control and how the operator applies these fundamentals to get balanced heat removal.

Heat Transfer Equations

The path for heat flow from the core to the steam generator is:

Core Heat $\longrightarrow$ reactor coolant

Reactor coolant heat $\longrightarrow$ Steam generator water and steam

The steam generator then releases the heat either to the atmosphere or to the condenser.

The concepts of heat sinks and heat sources are useful. For the first heat transfer path the core is the heat source for the reactor coolant and the reactor coolant is the heat sink. When the plant is tripped the reactor coolant pump heat becomes a significant heat source. For the second heat transfer path the reactor coolant is the heat source for the steam generator water and steam and the steam generator water and steam is the heat sink. The atmosphere and the condenser are heat sinks for the steam from the steam generator. In some unusual cases the reactor coolant can be colder than the steam generator fluid; then the steam generator is a heat source which passes heat to the reactor coolant sink.

Two "kinds" of heat can be transferred to the steam generators:

1. Generated heat which includes RC pump work and nuclear heat which is the heat made within the core by the fission process; it includes decay heat

2. Stored Heat - which is the heat of the metal parts of the reactor coolant system and of the reactor coolant

When the reactor is operating at steady state and heat removal is balanced the steam generators will remove the nuclear heat and RC pump heat as it is generated and reactor coolant temperatures will not change. In other words the stored heat will stay the same.

If the steam generators remove more heat than the core and RC pumps are creating then they will remove both generated heat and stored heat; reactor coolant temperatures will drop. Normal cooldown is a condition when both generated heat and stored heat are being removed within a specified rate; this is a controlled condition. If the condition is abnormal or not controlled then it would be called <u>overcooling</u> and corrective actions would have to be taken to bring it under control.

On the other hand, if the steam generators remove less heat than the core is creating then the nuclear heat will increase the amount of reactor coolant stored heat; reactor coolant temperatures will increase. Heatup from 0% to 15% power illustrates a controlled example where the stored heat of the reactor coolant is increased by

heat addition from the core nuclear heat and the reactor coolant pumps. If a condition exists where the reactor coolant temperatures increase abnormally it is called overheating; corrective actions would have to be taken to bring overheating under control.

Equations can be used to describe the heat transfer path from the core to the steam generators. When the heat transfer is balanced:

Equation 1) $\dot{Q}_{core} = \dot{Q}_{reactor\ coolant}$

for the heat transfer path from the core to the reactor coolant

and

Equation 2) $\dot{Q}_{reactor\ coolant} = \dot{Q}_{steam\ generator\ fluid}$

for the heat transfer path from the reactor coolant to the steam and water in the secondary side of the steam generators

$\dot{Q}$ is heat rate - units are BTU/hr.

When heat transfer is balanced all the way from the core to the steam generator Equation 1 equals Equation 2. But when heat transfer becomes unbalanced they will not be equal. Interruptions of the heat transfer path can happen when the reactor coolant is not a good heat sink for the core ($\dot{Q}_{core} \neq \dot{Q}_{reactor\ coolant}$); or when the steam generator fluid is not a good heat sink for the reactor coolant ($\dot{Q}_{core} \neq \dot{Q}_{steam\ generator\ fluid}$).

The unbalanced condition of concern for core heat transfer to the reactor coolant is when there is not enough heat transfer from the core to the reactor coolant. This can happen when the core is partly covered by water and partly by steam or covered completely by steam; then $\dot{Q}_{core} \neq \dot{Q}_{reactor\ coolant}$. When this happens not enough nuclear heat can be transferred from the core to the reactor coolant and the core will heat up. The stored heat of the fuel clad will increase which will result in increased fuel pin temperatures.

When the steam generator heat flow path becomes unbalanced then the steam generator fluid will remove too much or too little heat from the reactor coolant and it will be an <u>overcooling</u> or <u>overheating</u> condition. When this happens during a transient $\dot{Q}_{reactor\ coolant}$ will increase or decrease depending on the heat removal by the secondary side. The reactor coolant temperatures will change in order that temperature (thermal) equilibrium can be re-established between the primary and secondary side fluids. To show the effects Equations 1 and 2 can be written to add temperature terms:

Equation 1 ($\dot{Q}_{core} = \dot{Q}_{rc}$) can be written as:

<u>Equation 1a</u> $\dot{Q}_{core} = M_{rc} Cp_{rc} (T_h - T_c)$

where: M_{rc} = reactor coolant system mass flow rate (lbm/hr)

Cp_{rc} = specific heat capacity of the reactor coolant (BTU/lbm-F)

T_h = core outlet temperature (F)

T_c = core inlet temperature (F)

Equation 2 ($\dot{Q}_{rc} = \dot{Q}_{sg}$) can be expanded as follows:

Equation 2a: $\dot{Q}_{sg} = UA\Delta T$

where: U = overall heat transfer coefficient

A = total area of heat transfer surface

ΔT = temperature differential across the heat transfer boundary

Overall heat transfer coefficient is dependent on many factors including the fluid conditions (primarily density and flowrate) on both sides of the boundary and the properties of the boundary (primarily the thickness and thermal conductivity of the barrier and oxide layers). For this discussion we can assume that the properties of the boundary (steam generator tube walls) remain constant and therefore can be ignored.

The secondary side of the steam generator has three different regions along the tube bundle during power operation: nucleate boiling, film boiling, and superheat. Each region has a different coefficient (U), surface area (A), and temperature differential across the tube wall (ΔT). The nucleate boiling region has the highest U of the three and accounts for approximately 70-85% of the total heat transfer into the steam generator over the power range. The heat transfer coefficient decreases by a factor of 3-10 in the film boiling region and again by another factor of 3-10 in the superheat region.

The heat transfer surface areas and ΔT's involved for each of the three regions vary over the power range with the two boiling regions accounting for an increasingly higher percentage of the total heat transfer with increasing power levels. Thus, to determine the effects of transients on secondary heat removal during power operation, the effects in each of the three regions along the tube bundle must be studied.

However, for the purposes of these guidelines, we are primarily concerned with control of heat removal by the steam generators after a reactor trip. After trip the steam generators are at saturation conditions with two basic regions, saturated water and saturated steam. Almost all of the heat transfer occurs in the water region and most of the heat transfer in the water region occurs in the nucleate boiling portion below the steam/water interface. Saturated water is absorbing the latent heat of vaporization and the nucleate boiling provides a much higher heat transfer coefficient (U). Below this level the water is saturated with a considerably lower heat transfer coefficient, although this heat transfer coefficient is still much higher than exists in the steam space.

Very little heat transfer occurs in the steam space (primary side temperature can be considered equal to T_{hot} throughout the steam

space). Even though the area is large, the heat transfer coefficient is small due to low steam flow rates and low density with respect to the water region. During forced circulation the ΔT across the tube walls in the steam space is also very small as T_{hot} is close to T_{sat} of the steam. The ΔT is larger between T_{hot} and T_{sat} during natural circulation but the heat transfer coefficient is even smaller due to the lower primary flowrates.

The primary factors affecting heat transfer in the water region are surface area and the ΔT between the primary and secondary sides. Surface area is increased by increasing feedwater flow to raise level. The primary increase in area takes place in the saturated water region. Even though most of the heat transfer occurs in the nucleate boiling region, overall heat transfer is increased because the area of the steam space (with a very small heat transfer coefficient) is decreased and replaced by area in the saturated water region (with a relatively much larger heat transfer coefficient).

The major method of affecting primary to secondary ΔT is on the secondary side by varying steam pressure. When steam pressure is decreased (e.g., by opening turbine bypass valves) saturation temperature also decreases which increases the ΔT across the tube wall. The higher ΔT causes heat transfer ($\dot{Q}_{sg}$) to increase thus cooling the primary side. Of these two factors (surface area and ΔT), ΔT is

the major factor. Affects due to surface area are only significant while the surface area is changing. Once a constant water level is maintained, changes in heat transfer are primarily due to changes in ΔT.

Heat transfer can be increased significantly by injecting feedwater (main or emergency) through the upper nozzles. The increase in heat transfer is due to two factors. First, and most significant, the spray of feedwater into the steam space reduces steam pressure similar to the action of pressurizer spray. This reduces the saturation temperature which increases heat transfer as described previously. Second, where water contacts the tube surfaces in the steam space the heat transfer coefficient is increased, essentially replacing steam area with water area as in the case of raising steam generator level. Emergency feedwater will have a greater cooling effect than main feedwater through the upper nozzles (for the same flowrate) due to its colder temperature.

Assuming a minimum adequate level is maintained in the steam generators, variations in steam pressure will have a greater effect on heat transfer than variations in level. The best method to decrease heat transfer is to close the turbine bypass valves and allow the steam generator pressure to increase. Allowing steam generator level to decrease will not have an appreciable effect on heat transfer until the level becomes inadequate (too low for maintaining natural circulation or virtually dry with forced circulation).

In summary, the operator can control primary to secondary heat transfer after reactor trip by controlling two major parameters on the secondary side (assuming the capability of the reactor coolant to transport core heat to the steam generators remains intact). The operator can increase heat transfer by reducing steam pressure or by raising steam generator level. He can decrease heat transfer by allowing the steam generator pressure to increase.

FOOTNOTE

Equations 1a and 2a have been simplified to show the general heat transfer process. To be complete additional heat transfer terms would have to be included. All of the water that flows through the reactor coolant system loops does not flow through the core and get all the way to the steam generators. Some flow is let down to the makeup system, some goes to the pressurizer spray and there is some "leakage" through spaces in the internals. This amount of flow is small and it has been ignored for these equations. Also, all the heat of the core does not go to the steam generators; some of it is lost through the "skin" of the piping to the reactor building or through the letdown water. But this amount of heat is small compared to the total amount and it has been neglected. Heat is also added by reactor coolant pumps (as in plant heatup to power operation), but it is small compared to core heat when the reactor is at power (but the reactor coolant pumps are a large heat source after trip or at low power).

Control of heat transfer requires control of all the parameters in these two equations. Some are fixed by design or properties of fluids; the remainder can be influenced by the operator. The general methods of heat transfer control are to be discussed next.

Control of Heat Transfer

The preferred way of removing heat from the core is to transfer the heat to the reactor coolant and then transfer the reactor coolant heat to the secondary fluid in the steam generators. Steam generator heat removal is controlled by adjusting steam pressure and feedwater. To keep the core-to-steam generator heat transfer in balance the heat removal rate from the steam generators must be equal to the heat generation rate of the core. In order to balance the heat removal two very basic conditions must be satisfied: 1) There must be enough liquid reactor coolant in the vessel and piping to transfer the heat to the steam generators, and 2) the steam generator pressure and level (feedwater flow rate) must be balanced at the correct heat removal rate. Figure 1 illustrates these fundamental methods of heat transfer control.

Figure 1 shows the controls that the operator can use to change heat transfer.

The five fundamental functions of heat transfer control are:

- Reactivity control (core heat output control)
- Reactor coolant pressure control
- Reactor coolant inventory control
- Steam generator pressure control
- Steam generator inventory control

When an abnormal transient occurs, one or more of these five functions will be out of control. It is the operator's job to determine which are, and to make corrections to restore the right heat transfer balance so the core heat can be removed by the steam generators.

1. Reactivity control - Reactivity control is usually taken care of automatically by ICS rod control or by reactor trip. Reactor trip lowers the core heat output to the decay heat level.

2. Reactor Coolant Inventory Control - The link between the core and the steam generator is the reactor coolant. It is the fluid which transports the heat. To do its job best the coolant should be in a liquid state, that is, subcooled. (Discussion of subcooling is given in Addendum A.)

3. Reactor Coolant Pressure Control - The reactor coolant system is pressurized to keep the reactor coolant in a liquid state.

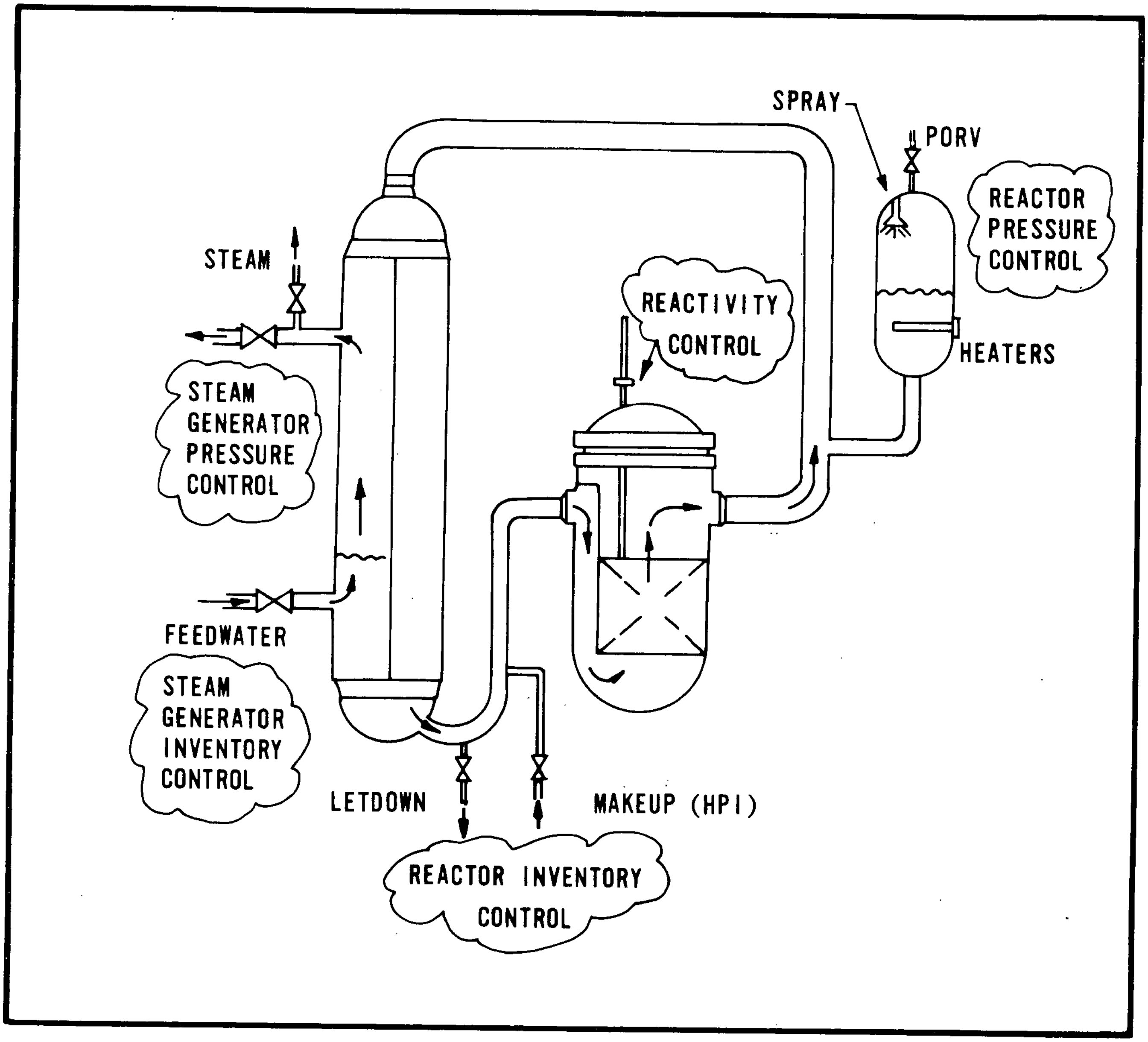
4. Steam Generator Inventory Control - The reactor coolant transfers its heat to the water and the steam in the secondary side of the steam generator. The water-steam inventory is the heat transfer fluid which removes the heat from the reactor coolant. In order for it to remove heat at the correct rate the amount of fluid and its flow rate through the steam generator must be controlled.
5. Steam Generator Pressure Control - The water temperature of the reactor coolant is best controlled by controlling the pressure of the steam generator. In combination with reactor pressure control, steam generator pressure control will maintain the reactor coolant in a subcooled liquid state.

Each one of these control functions will be discussed individually as they relate to heat transfer.

Steam Generator Pressure Control

Heat transfer from the reactor coolant to the steam generators goes to both the steam and water in the generator. After reactor trip the steam and feedwater in the generator are saturated, and changes of steam pressure will cause a direct change in the saturation temperature of the steam and of the feedwater. A review of the

Figure 1 FUNDAMENTAL METHODS OF HEAT TRANSFER CONTROL



saturated water and steam sections of the ASME Steam Tables will show how much the steam and water temperature will be changed by increasing and decreasing steam pressure. There are situations where the operator controls the steam pressure by manually increasing or decreasing steam pressure using the turbine bypass valves or the atmospheric dump valves. When the steam pressure is lowered the heat transfer from the reactor coolant to the steam generator increases because the steam and water in the steam generator become a colder heat sink causing more heat to flow away from the reactor coolant. Two reasons combine to create the colder heat sink: first, the saturation temperature of the steam and water is reduced by lowering the steam pressure which causes the rate of boiloff to increase. The increased boiloff takes away more heat. Second, the increased boiloff requires more feedwater flow to be added to maintain level. The feedwater inlet temperature is colder than the water already in the steam generator and so its addition contributes to the colder heat sink. Because a colder secondary heat sink exists the primary side temperature will drop as heat is transferred.

Steam pressure can be lowered in two ways:

- By opening the steam line and releasing steam (turbine bypass, steam line break, atmospheric dump valves, steam to EFW pump turbine).
- By spraying Main Feedwater or cold Emergency Feedwater into the steam space and condensing it. This is similar to the way pressurizer pressure is reduced by the pressurizer spray.

Steam pressure can also increase; but normally it will only increase from the operating condition to the reactor trip condition where it will be limited by the steam safeties or by the turbine bypass valves, so the affect on reactor coolant temperature is small. But if steam pressure is low because of a failure, for example a steam line break, the change in reactor coolant temperature could be much larger. When the steam break is isolated the reactor coolant adds heat to the generator and causes the steam pressure to increase. The operator can limit the increase in reactor coolant temperature under these conditions by lowering the turbine bypass valve setpoint and keeping steam pressure low.

Steam Generator Inventory Control

Heat transfer from the reactor coolant goes to both the steam and the feedwater in the secondary side of the steam generators. When changes of feedwater flow or steam pressure occur the volumes occupied by the steam or water will change and the heat transfer will change. For example, when the volume of water increases, it occupies space formerly occupied by steam, so the mass of steam has to decrease. This changes the relative amounts of OTSG tube surface area covered by water and steam. Because water has a greater heat capacity than steam does it is a better heat sink for heat transfer from the reactor coolant than steam is. Simply stated there are

more pounds of water in a cubic foot to absorb heat than there are of steam. If the water inventory increases then the generator will become a better heat sink for the reactor coolant, but if the water inventory decreases or is lost the generator will lose some or all of its ability to absorb heat from the reactor coolant.

For example, after trip when the core heat is nearly constant, if the water level in the steam generator is raised rapidly without changing steam pressure, the reactor coolant temperature will drop and stay low until the feedwater addition reaches a new level and that level is held. Once the new level is fixed the reactor coolant will reheat and temperatures will return to their former values.

This cooling effect of feedwater is caused by the inlet feedwater temperature which is colder than the general temperature of the bulk of the fluid in the steam generator. The inlet feedwater temperature allows a colder heat sink to be established in the steam generator.

The steam generator level can, however, be increased slowly after trip without a large drop of reactor coolant temperature by controlling the rate of addition of feedwater.

Too much inventory can also be the result of overfeeding with the Emergency Feedwater System. Even though its flow rate is lower, Emergency Feedwater will have a proportionally larger cooling effect on reactor coolant than main feedwater because:

a) it comes on when the reactor is tripped and core heat is lowest,

b) it is colder ($T_{\text{inlet Feedwater}}$ is less), and

c) it has a steam pressure reduction effect that main feedwater does not normally have (main feedwater will cause some pressure reduction when it enters through the upper nozzles following a loss of reactor coolant pumps).

On the other hand, if steam generator inventory is too low (insufficient feedwater or loss of feedwater can lower the water level), the reduced heat sink will not allow the reactor coolant to transfer all of its heat to the steam generator. When the steam generator's heat sink is reduced, the reactor coolant must retain more of the core heat and it will heat up.

For example, if <u>all</u> feedwater is lost, the water in the generator will boil away and only steam will remain to remove heat. But because the steam does not have enough heat capacity, the reactor coolant must retain the core heat and the reactor coolant

temperatures will increase. When all feedwater is lost the reactor coolant pressure will increase to the PORV setpoint and the reactor coolant will eventually become saturated as the core continues to add heat. The steam remaining in the generator will flow out through the steam lines and steam pressure will drop; loss of the steam eliminates the heat sink of the steam generators altogether.

Finally, another part of steam generator inventory control is feedwater temperature. The heat sink of the generators will be affected by an abnormally low feedwater temperature. A reduction of feedwater heating steam or loss of a feedwater heater will cause reactor coolant temperature to decrease. Usually ICS operation will stabilize the plant, but the decreased feed temperature will cause a change in the heat sink and an increase of heat transfer from the reactor coolant.

The operator should ensure the rate of feedwater addition is controlled properly to maintain the steam generator inventory. Level measurements in the steam generator downcomer give a good indication of the steam generator inventory for control.

Reactor Coolant Inventory Control

Reactor coolant heat transfer can be affected by changes in the amount of mass of fluid in the reactor coolant system or by changes in the density of the reactor coolant.

Several ways exist to vary the mass of reactor coolant: LOCA or small break, and changes in HPI or makeup, RC pump seal injection, seal return, and letdown. Several ways also exist to vary the density of the reactor coolant. As shown by the previous discussions of steam generator pressure and inventory control, changes of the rate of heat transfer from the reactor coolant to the steam generator can cause the reactor coolant to cool down when the steam generators remove too much heat (low steam pressure, too much feedwater); or the reactor coolant can heat up when the steam generators don't remove enough heat (not enough feedwater). These effects cause density changes in the reactor coolant; the coolant contracts or expands accordingly.

Regardless of the cause, changes in inventory in the reactor coolant system have two effects:

1) A loss of mass can affect the ability of the reactor coolant to transport heat from the core to the steam generators. If the RC pumps are not running steam can collect in the hot legs and block natural circulation. When circulation stops and heat transport stops then the steam generator temperature will not "set" the temperature of the reactor coolant; T_{cold} will not change when T_{sat-SG} changes.

If the mass of the reactor coolant system continues to decrease and the core is mostly covered by steam it will not provide a sufficient heat sink and the core will retain the heat and heat-up. Fuel failures can result if this situation is not corrected.

2) A change of mass or density can affect the ability of the pressurizer to provide pressure control of the reactor coolant system (this will be discussed next under Reactor Coolant Pressure Control).

Operator control of reactor coolant inventory requires the ability to balance mass increases or decreases by adding water with makeup or ECCS systems or removing mass with the letdown. Control of reactor coolant density changes requires control of the steam generator pressure and inventory.

The inventory of the reactor coolant system cannot be measured directly. But the operator has two indications to determine if the inventory is sufficient for core cooling. Pressurizer level is an accurate measure of the inventory when the reactor coolant is subcooled (except for a rare possibility when free hydrogen gas may exist in the loops; this condition will only likely exist after fuel failures caused by uncovering of the core). The other measure is

the incore thermocouples; if these read subcooled or saturation temperature then enough mass exists in the reactor vessel to cover and cool the core. But the incore thermocouples will not show if the loops are full.

Reactor Coolant Pressure Control

Reactor coolant pressure control is required to keep the reactor coolant subcooled so the coolant is in the best state to transfer the heat from the core to the steam generators. For all cases of reactor operation except LOCA's, RCS pressure control is provided by the pressurizer. (Reactor coolant pressure control is different for LOCA's and small breaks than for other plant conditions. It is discussed in detail in Appendix F.) Use of pressurizer heaters and spray is the usual way of increasing and decreasing RCS pressure when a steam and water interface exists in the pressurizer. The purpose of the heaters is to maintain the reactor coolant in a subcooled condition; the spray retards pressure increases to limit operation of the pressurizer relief and safety valves. Neither the heaters nor spray have enough capacity to prevent large abrupt pressure changes, but they can moderate small changes. As a backup the PORV can be used to reduce pressure but it is not as desirable to use as the spray.

RCS pressure control by the pressurizer can be lost in two ways:

1) The steam-water interface in the presurizer can be lost either by draining the pressurizer or if the pressurizer fills solid with water
2) The heaters and spray can fail.

Each of these is discussed below:

Draining the Pressurizer: If the pressurizer level drops sufficiently to uncover the heaters, the heaters cannot provide pressure control because no water is available to be boiled by the heaters to create steam. If the pressurizer drains completely, RCS pressure will then be controlled by the highest fluid temperature in the system.

When the pressurizer drains the reactor coolant system pressure will decrease to the saturation pressure of the reactor coolant in the hot leg. Pressure will then be controlled indirectly by the steam generator (the steam generator sets the reactor coolant loop hot and cold leg temperatures). Since the hot leg is at the highest temperature the reactor coolant in the hot leg will flash to steam. In effect the hot leg will become a pressurizer.

Filling the Pressurizer:

Spray depressurizes the reactor coolant system by condensing the steam in the pressurizer. If the pressurizer fills with water, the spray cannot be effective for depressurizing because the steam space is lost.

When the pressurizer fills, the reactor coolant system may or may not lose subcooling and become saturated depending on what caused it to fill. If the filling was caused by HPI or makeup and the steam generator is still removing heat, then the RCS will stay subcooled because the makeup (HPI) pumps will cause the pressure to stay at the PORV setpoint and the steam generator will keep the temperature controlled. If the filling was caused by heatup and swell because the steam generators were not removing enough heat, then the system may become saturated because the heat from the core will only go into the reactor coolant and not out the steam generators.

When the pressurizer fills, either because of heating the reactor coolant or because of too much HPI, the water will be lost through the pressurizer valves. This loss is considered to be a LOCA, even if the action was deliberately done.

Failure of Heaters and Spray:

A failure of the spray and heaters in the pressurizer control system can also cause a loss of pressure control. If the spray fails and cannot be turned off the system will depressurize. Depressurization may also occur if the heaters fail in the "off" mode. The reverse is not true; failure of the spray in the "off" mode will only limit the ability to depressurize. Unless something else happens to the plant, pressure increases and decreases will not occur. If the heaters fail "on" pressure increases will not occur because the spray will operate to provide a balance. However, if the spray is not working then the heaters can cause the system to pressurize and cause coolant (steam) to be lost from the pressurizer relief valves; subcooling will not be lost as long as water covers the heaters. When only steam covers the heaters they will no longer raise pressure and subcooling can gradually drop. If the heaters fail "on" when they are uncovered, no water exists to cool them and they will burn out.

Reactivity Control

Reactivity control is usually taken care of automatically by ICS rod control or by reactor trip. Reactor trip lowers the core heat output to the decay heat level. The operator must verify rod insertion and decreasing reactor power to ensure the reactivity control systems

function properly. After the trip no more heat transfer control can be achieved by use of the rods, unless the rods did not fully insert. If one or more rods are stuck out after trip the operator should manually trip them. If one or more rods remain stuck out the operator should begin emergency boration and a reactivity balance calculation should be performed to ensure a shutdown margin in excess of 1% $\Delta k/k$ is achieved.

Summary

The preceding discussion introduced the concept of reactor-steam generator heat transfer and the balance that heat transfer must have. When an imbalance of heat transfer occurs, its effects will often be transmitted throughout the steam and reactor coolant systems. The purpose of understanding heat transfer is to understand its effects so the operator can step in and diagnose what has gone wrong and correct it. An understanding of the major influences on reactor-steam generator heat transfer control (reactor inventory control, reactor pressure control, steam generator pressure control, steam generator inventory control, and reactivity control) will allow the operator to focus on achieving controlled heat transfer and stable plant conditions without necessitating the identification of specific failures. Thus, an understanding of the principles of heat transfer and the control methods permits a more direct and efficient approach to abnormal transient diagnosis and correction.

The effects of changing one of the controls will nearly always cause changes in other parts of the system and therefore will require other controls to be changed to balance heat transfer. The controls are interdependent because they affect total heat transfer from core to steam outlet.

The core cooling with the steam generator can occur as long as two things exist:

- The reactor coolant can transport the heat. The best way to do this is with subcooled liquid. Reactor Coolant Inventory and Pressure Control contribute towards this.
- The heat removal is controlled by the steam generator. Steam Generator Inventory Control and Pressure Control aid this.

Usually an abnormal transient will be caused by a failure of one or more of the heat transfer controls. The understanding of the control influences allows the operator the freedom of two approaches to abnormal transient correction:

1. He can treat the symptoms by manipulating equipment to regain heat transfer control without knowing exactly which equipment has failed. Consequently, proper heat transfer can be restored quicker and more accurately than if the operator had to hunt for the equipment failure. In some instances, treating the symptoms will also uncover the failed equipment.

2. He can use these control failures as symptoms of poor heat transfer to discover the equipment that has failed and by doing so, isolate it, remove it from service, or repair the equipment.

Understanding the influence each of these controls have on overall heat transfer will also give an understanding of what the outcome of an action is. All operator actions will have some consequence to heat transfer and a knowledge of the heat transfer will allow judgements to be made about the general effects.

Table 1 is a summary of the previous discussion. Like all summaries, material has been condensed. When that happens, information has been left out. The table should only be used to provide an overview.

The next section builds on the information about heat transfer and extends those principles into a disciplined approach to accident diagnosis and recovery.

Table 1 HOW FAILURES AFFECTING HEAT TRANSFER CAN AFFECT REACTOR OPERATION

CONTROL PRINCIPLE	FAILURE	EQUIPMENT WHICH MIGHT HAVE FAILED	EFFECTS ON REACTOR-STEAM GENERATOR
Steam Generator Pressure	Low	Steam Release - - Turbine Valves Open - Turbine Bypass Open - Steam Safeties Open - Other Steam Extraction Open - Steam Piping Break Steam Condensation - - Emergency Feedwater On	T_{av} Drops; Reactor Subcooling Increases; Reactor Coolant Shrinks; Pressurizer May Drain; If Pressurizer Drains, Then Reactor Coolant Will Saturate.
Steam Generator Inventory	High Level (Too Much Feedwater) Overcooling	- Feedwater Control Valves Open or Don't Close After Trip - Feedwater Pump Overspeed - ICS Controls; Power to ICS - Operator Error in Manual - Emergency Feedwater Not Controlled	- Steam Generator Level Increases; Superheat Lost; T_{av} Drops; Reactor Subcooling Increases; Reactor Coolant Shrinks; Water Can Enter Steam Lines; MFW Will Probably Not Cause Pressurizer Draining if not severe or terminated early. - If Emergency Feedwater Is Uncontrolled After Trip, The Overcooling may Be Enough To Drain The Pressurizer; Reactor Coolant Will Saturate.
	Low or No Level (Not Enough Feedwater) Overheating	- Loss of Feed Because of Many Possible Failures In Feedwater And Condensate System - Feedline Break	- Steam Generator Level Lost; T_{av} Increases; Pressurizer Fills; Reactor Subcooling Lost; Pressurizer Relieves Through Safeties (LOCA).
Reactor Coolant Inventory	Low	- Loss of Coolant - Failure of Letdown, Makeup, Seal Injection, HPI - Overcooling (Too Much Feedwater, Low Steam Pressure)	- Pressurizer May Drain; Reactor Coolant Will Saturate; If In Natural Circulation Flow to Steam Generators May Be Blocked By Steam In Hot Legs.
	High	- Failure of Letdown, Makeup, Seal Injection, HPI - Overheating (Not Enough Feedwater)	- If HPI or Makeup Failure Fills the Pressurizer, the RCS Will Go to 2500 PSI but Will Remain Subcooled. - If Overheating Failure Fills the Pressurizer, the RCS Will Go to 2500 PSI and the RCS Temperature Will Increase; Subcooling Will Be Lost. - In Either Case, a LOCA Through the Pressurizer Safeties Will Occur.
Reactor Coolant Pressure	Low	RCS Pressure Control Equipment - Spray Fails On, and - Heaters Fail Off Loss of Reactor Coolant Inventory Control (Low)	- If RCS Pressure Drops Too Low, Subcooling Will Be Lost.
	High	RCS Pressure Control Equipment - Heaters Fail On, and - Spray Fails Off Loss of Reactor Inventory Control (High)	- RCS Pressure will Increase to 2500 PSI; Coolant Will Be Lost Until the Heaters uncover.

ADDENDUM A

(SUBCOOLED, SATURATED, SUPERHEATED WATER)

The state (solid, liquid or gas phase) of the water in the reactor coolant system or the steam system is determined by the pressure and temperature conditions which exist. The terms subcooled, saturated, and superheated are normally used within operating procedures. These terms mean the following:

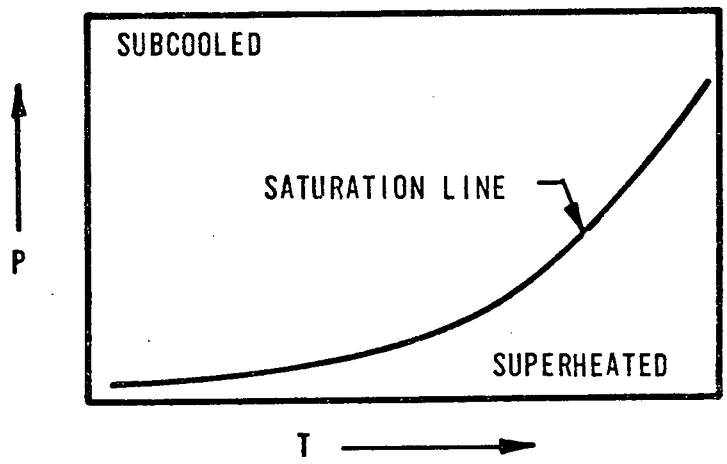
Subcooled: Water can exist only in the liquid phase.

Saturated: If heat is added to subcooled water a temperature, for the existing pressure, will be reached where the water can exist either as a liquid or as a gas (steam). At this point, the liquid is called satuated water and the gas is called saturated steam. The liquid and steam phases both can exist at this temperature and pressure. Heat must be added to saturated water to change it to saturated steam. Heat can also be removed from saturated steam to change it to saturated water. The heat required to make the change is called the latent heat of vaporization for heat added and the latent heat of condensation for heat removed.

Superheated: Water can exist only in a gaseous or steam phase. This phase can be distinguished from saturated conditions because the temperature will be higher than the saturation temperature for the existing pressure.

The normal state of the steam coming out of the steam generator is superheated during power operation and saturated after trip.

The state of the reactor coolant can be determined by watching the RCS pressure and temperature on a pressure-temperature diagram (see below):



P-T conditions which are to the left and above the saturation line are in the subcooled region, and P-T conditions to the right and below the saturation line are in the superheated region.

Subcooling

Subcooled conditions are maintained in the reactor coolant (except pressurizer) during normal operation. During a reactor transient it is desirable to maintain the reactor coolant subcooled. When subcooled:

1. The primary loops are solid water and a water level is present within the pressurizer.
2. The pressurizer water level is a true measurement of RCS inventory. (NOTE: A very special case can exist when the reactor coolant is subcooled and a water level is in the pressurizer but the loops are not full. In that case pressurizer level is not a true measurement of inventory. That condition is when there is a large amount of free gases in the loop. The gases will be mostly H_2, that have been created after a large amount of fuel failure. Since this would be an uncommon event, reliance on pressurizer level is usually acceptable when the reactor coolant is subcooled.)
3. The reactor coolant is liquid and is ideal for heat removal from the core and heat transport to the steam generator by either forced or natural circulation.
4. RC pressure can be maintained by the pressurizer and can be regulated by using normal procedures and equipment (spray, heaters, and regulation of pressurizer level by the MU and/or HPI system and letdown).
5. RC temperature can be controlled by the secondary system (with feedwater available) and can be regulated by adjusting feedwater flow and steam pressure.

Subcooling should be checked in all parts of the loop especially when natural circulation is removing heat. The operator should check T_{hot} and T_{cold} in both loops and the core exit thermocouples. Anytime subcooling is lost the HPI system should be turned on full.

HPI Subcooling Rule: Two HPI pumps should be run at full capacity when:
The ES is actuated and the HPI is automatically started.
The reactor coolant subcooled margin is lost and the HPI is manually started.

NOTE: All three HPI pumps start on automatic initiation but only two are required. Therefore, if all three are operating properly, the operator should secure one of the HPI pumps supplying train 'A'.

Saturation

A loss of subcooling can happen when the pressurizer drains or is filled solid (if the pressurizer is solid because of HPI and cooling is by the steam generators then the Reactor Coolant can stay subcooled). A loss of subcooling can be caused by an overheating or overcooling transient or a loss of reactor coolant. Saturated conditions can exist in isolated pockets of the loop (i.e., within one or both hot leg pipes or the reactor vessel head but not in the cold leg pipes) or within the system as a whole, as would be the case during a major LOCA. Therefore, temperatures should be checked in the hot legs of both loops. When the RCS is saturated:

1. The reactor coolant temperature and pressure will not show whether the saturated fluid is liquid or gas (steam).

ADDENDUM B

NATURAL CIRCULATION

When the reactor coolant pumps are tripped forced circulation is lost and an alternate method of removing core decay heat must be found. The preferred method is to transport this heat to the steam generators by natural circulation of the reactor coolant. Natural circulation is possible as long as the following requirements are met: 1) a heat source is available to produce warm (low density) water; 2) a heat sink is available to produce cold (high density) water; 3) a flow path (loop) is available connecting the warm and cold water; and 4) the cold water is at a higher elevation than the warm water. Requirements 1, 2 and 3 are met by the following: decay heat in the core is the heat source, water on the secondary side of the steam generators provides a heat sink, and the hot and cold legs connect the two. Requirement 4, "the cold water is above the warm water," involves a concept called thermal center. In reality heat is transferred continuously as the water moves up through the core and again as it moves down through the steam generator. The thermal center is the point in the core or the steam generator where the primary water is at average temperature. It can be used to represent the entire column of water in its "average" conditions.

Thermal Center Definition

1. Core thermal center: That elevation in the core which the coolant may be considered to go from T_{cold} to T_{hot}.

2. Steam generator thermal center: That elevation in the steam generator at which the coolant may be considered to go from T_{hot} to T_{cold}.

Requirement 4 for natural circulation can be met if the thermal center of the steam generator is at a higher elevation than the thermal center of the core. This will put the "average" cold water above the "average" hot water, the cold water (more dense) will sink, the hot water (less dense) will rise and there will be circulation.

The rate of natural circulation (gpm) depends on the following things:

- The friction (resistance to flow) of the piping and components around the primary loops: this is determined when the plant is designed and built; the operator has no control over it.
- The strength of the heat source: this depends on the available decay heat, which is a function of past power history and time since the reactor trip. It will, of course, decrease with time after trip. The operator has no control of this after trip except to make sure the reactor is shut down so that the only heat input is decay heat.
- The strength of the heat sink: the colder the heat sink is the more it will be able to cool the primary coolant passing through the steam generator. This will make the water more dense and the natural circulation flowrate will increase. The operator can make the heat sink colder by 1) lowering secondary steam pressure

(opening the turbine bypass or valves more), this will lower secondary saturation temperature which will increase heat transfer across the tubes; or 2) lower feedwater temperature (for example, shift from main feedwater to emergency feedwater), this will increase the heat transfer across the tubes by providing a larger primary to secondary ΔT.

- Difference in height between the core thermal center and the steam generator thermal center: The larger this difference is the more imbalance will exist between the high cold water and the lower hot water and more natural circulation flow will result. The core thermal center is fixed, but the operator can control the steam generator thermal center by two methods: 1) most of the heat transfer occurs in the violent boiling area just below the established secondary side water level. Therefore, the operator can raise the thermal center by raising the steam generator water level; 2) the operator can add FW through the EFW nozzles at the top of the generator. This will put feedwater high in the generator and thereby raise the average height (thermal center) of heat removal. This only works <u>while</u> FW is being added. If FW is stopped, the thermal center will move back down to just below the water level.

In summary, the natural circulation flowrate can be changed by changing the difference in temperature (density) between the hot water and the cold water or, changing the difference in height between the core thermal center and steam generator thermal center. This can be expressed in equation form as:

$$\Delta P_{driving\ head} = h_{eff} (\rho_c - \rho_H)$$

where: $\Delta P_{driving\ head}$ = available driving head for natural circulation

h_{eff} = distance between core thermal center and steam generator thermal center (effective height)

ρ_c = density of cold water at steam generator thermal center

ρ_H = density of hot water at core thermal center

This is shown graphically in Figure 2.

Natural Circulation (All Other Conditions Normal)

When the reactor coolant pumps are tripped the operator should check two things to make sure natural circulation is being initiated properly. First he should make sure the reactor coolant remains subcooled. If it does not he should make every effort to restore subcooling (the methods for doing this are discussed in the accident mitigation chapter of these guidelines). Second, he should make sure the thermal center is being raised in both steam generators. Normally automatic equipment will transfer MFW injection to the upper nozzles and increase level to 50% on the operating range of each steam generator when the RC pumps trip. The operator should monitor this process while keeping the following in mind:

- As long as MFW is flowing at sufficiently high rates into the top of the generator it is not necessary to get a level in the generator to have natural circulation. If the heat source (decay heat) is high enough, the MFW may come in and boil right off and go out as steam. This is acceptable; the thermal center is high and natural circulation will develop.
- If MFW is not available natural circulation can be initiated using emergency feedwater. Again, the level will be raised automatically to 50% on the operating range when the RC pumps are tripped.
- With two EFW pumps running or with low decay heat levels it is likely that the reactor coolant will be overcooled and could drain the pressurizer even with proper automatic control. This is due to the fact that the automatic control will provide essentially full EFW flow until the steam generator levels approach the level setpoint. If the pressurizer drains, subcooling will be lost. As was pointed out in the heat transfer chapter, this will not happen if the rate of EFW flow is limited. The operator can do this by throttling EFW flow. After initiation of EFW the operator should watch steam pressure, pressurizer level, and cold leg temperatures. If necessary, EFW should be throttled to prevent overcooling. Guidelines for throttling EFW are discussed in the Best Methods chapter.

Ideally, main feedwater is best used for natural circulation if MFW and the associated controls in the ICS are available. MFW can be

injected through the upper nozzles to establish a higher thermal center (while flow exists) with less likelihood of overcooling than is possible with colder EFW.

Figure 3 shows how RCS temperature and pressure, and steam generator temperature and pressure will vary during the transition to natural circulation using EFW. Approximate times for the transient are also included. The times are approximate because the rate of recovery of the steam pressure depends on the amount of decay heat available. When steady state is reached, the cold leg temperatures (T_{cold}) will be just about equal to the saturation temperature in the steam generators. The hot leg and incore thermocouple temperatures will increase as necessary to develop the driving head required for flow (by developing a density change between T_h and T_c). The best measures to use to see if natural circulation has started are the coupling between T_c and the steam generator temperature, the relationship of T_h and the incore thermocouples, and the temperature difference between T_h and T_c. When the incore thermocouples, T_h, and T_c are subcooled, they should follow steam generator T_{sat} when it changes; the temperature difference between T_h and T_c should be $\sim$ 50-60F (maximum decay heat with both SG levels at 20 ft.) and an average of the five highest incore thermocouples should track T_h within approximately 10F. If T_c only is subcooled and T_h and the incores are saturated, natural circulation characteristics should be the same as if they are both subcooled; however, the incores will not provide verification of flow since both T_h

and the incores are saturated. Once natural circulation is established and the higher steam generator levels are reached, the operator must ensure feedwater is available to replace the steam generator water being boiled off removing decay heat and to maintain the RCS subcooled. Transition to natural circulation using MFW will look very similar. The major difference would be slightly less primary system cooldown (with the same feedwater flow rates) while the SG levels are being established due to higher temperature feedwater.

Natural circulation flow will regulate itself. That is, as the heat source (decay heat) dies down the ΔT (T_h - T_c) will go down and there will be less driving head available; therefore, flow will go down.

Natural Circulation - Abnormal Operation

The discussion so far concerned expected or normal natural circulation conditions. That is, the RCS is subcooled, the level in both steam generators is 50% on the operate range and both steam generators are being steamed. This section will discuss off normal conditions:
1) natural circulation with one OTSG, 2) natural circulation with saturated RCS, and 3) recognition of loss of natural circulation.

One OTSG

There may be times when an operator does not want to steam a generator (OTSG tube leak) or cannot steam a generator (steam line break and isolated generator is dry). If he is also in natural circulation he can expect the following:

T_{hot} on both loops will be about equal; T_{cold} on the operating generator will be equal to T_{sat} in the operating steam generator; T_{cold} in the isolated generator will not be equal to T_{sat} in the isolated generator, it will probably be much colder being influenced by seal injection water temperature coming into the idle pumps; (T_h-T_c) on the operating steam generator should again be 50-60F but the level in the operating steam generator may have to be raised above 50% to maintain adequate natural circulation flow. Steady state operation under these conditions is stable and safe. Plant cooldown, however, is complicated because the cooldown of the loop with the isolated steam generator will lag behind the steaming steam generator. If there is water in the isolated steam generator it will become a heat source instead of a heat sink. In fact, the isolated generator may add enough heat to cause the reactor coolant in its hot leg to flash to steam. If this happens, that hot leg will act as a pressurizer and slow down the depressurization during cooldown. This will also slow down the cooldown rate. The operator must carefully watch subcooling in both loops under these conditions and make sure adequate margin is maintained by regulating the rate of cooldown with steam pressure control of the operating steam generator.

Natural Circulation with a Saturated RCS

A subcooled reactor coolant system is the desired state, however, natural circulation can remove core heat when the RCS is saturated. As long

as the four requirements for natural circulation are met heat will be removed from the core and transferred to the steam generator. The problem with saturated natural circulation is that the operator doesn't know how much of the reactor coolant is steam and how much is water (see discussion of saturation in Addendum A). If the RCS is losing inventory steam will form in the hot legs and eventually stop natural circulation flow (this is a violation for the requirement that a flow path exists connecting the hot water and the cold water*).

Another form of natural circulation could still exist under these conditions called boiler-condenser cooling (boiling in the core and condensing in the steam generator) but it requires a higher steam generator level (95% on operator range). This method is discussed in detail in the Backup Cooling Methods chapter of these guidelines.

The point to remember is that primary inventory (mass) is unknown under saturated conditions and therefore, every effort should be made to keep the RCS subcooled.

*This could also be violated by a large collection of non-condensible gases in top of the hot legs, however, such a collection could only exist following a core uncovery. At that point the operator would be using inadequate core cooling procedures.

Recognition of Loss of Natural Circulation

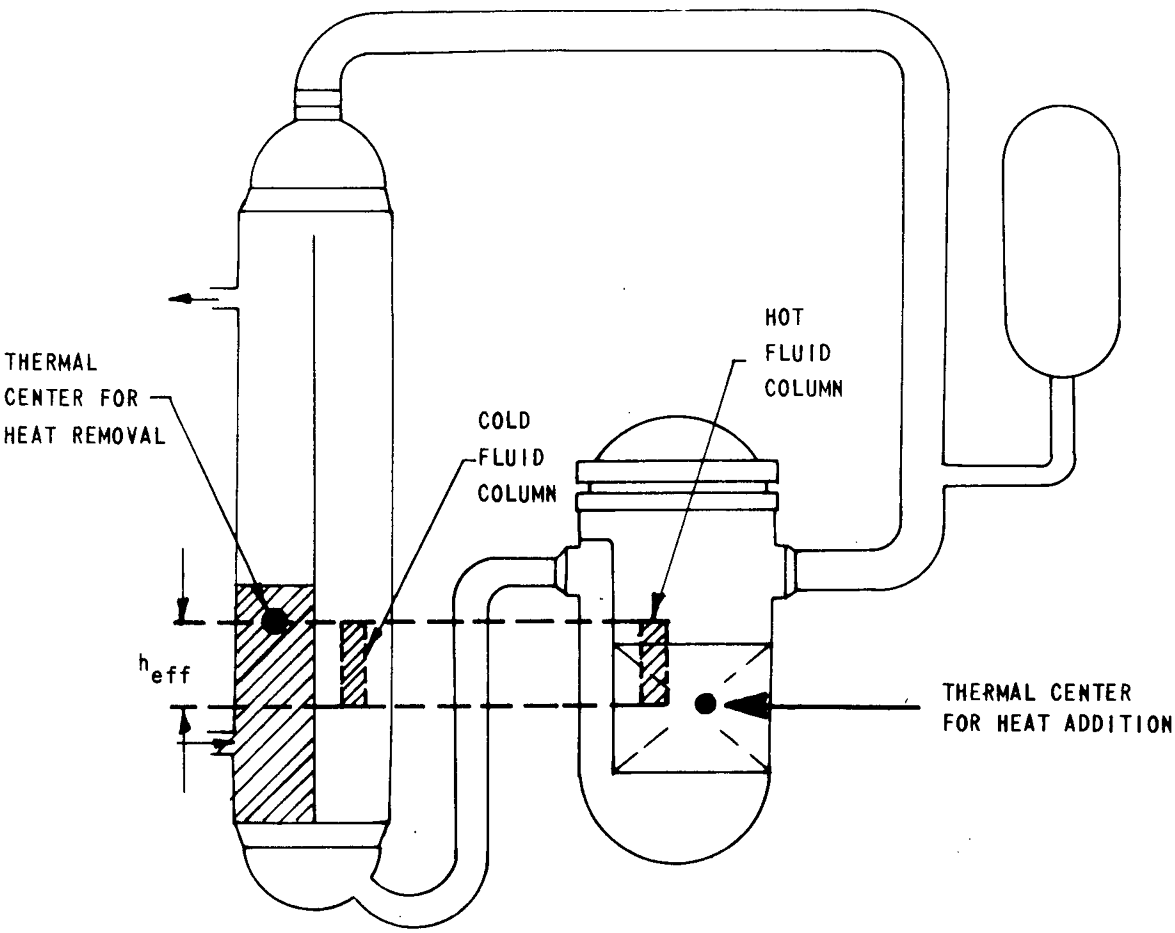
A loss of natural circulation can occur for various reasons and several indications may be available. If the RCS is subcooled, a loss of natural circulation flow is more than likely a result of inadequate heat removal by the steam generators. The thermal center in the steam generators may be too low. At low decay heat levels or during single loop cooldown, the SG levels may have to be raised above 50% on the operate range to induce or maintain natural circulation flow. When natural circulation flow exists, T_{hot} and the incore thermocouples will track together within $\sim$10F (although there will be some time lag due to long loop transport times). In addition, T_{cold} and T_{sat} of the SG should track together.

The best single indication of a loss of natural circulation flow when the RCS is subcooled is a divergence developing between the incore thermocouples and T_{hot}. When the flow is lost, the incore thermocouples will begin a continual increase toward saturation. The rate will depend on the decay heat level. T_{hot} indications may also increase but can actually decrease and begin to converge with T_{cold}. In any case, T_{hot} will not increase as rapidly as the incore thermocouples and the two indications will diverge. Another indication of loss of natural circulation is a "decoupling" between T_{sat} in the SG and T_{cold}. If T_{cold} ceases to follow T_{sat} natural circulation flow is lost. However, this is not as positive an indication as the divergence between the incore thermocouples and T_{hot}.

When the RCS is saturated and natural circulation flow is lost, this divergence may not develop significantly. The best indication of a loss of natural circulation flow when the RCS is saturated is a trend of incore thermocouple temperature vs. RCS pressure increasing along the saturation curve. Flow can be lost due to low thermal centers in the SG's or blockage due to voids in the RCS. When saturated, SG levels should be maintained at 95% on the operate range and full HPI flow should exist. If voids exist in the RCS, it is possible that boiler-condenser cooling was in progress. As the RCS refills, cooling in this manner is expected to be lost when the RCS liquid level increases above the SG tubes. However, in this case cooling should be restored by continued refill and by following the actions specified in Section III.B of Part I (RCP bumps, reducing SG pressure, etc.).

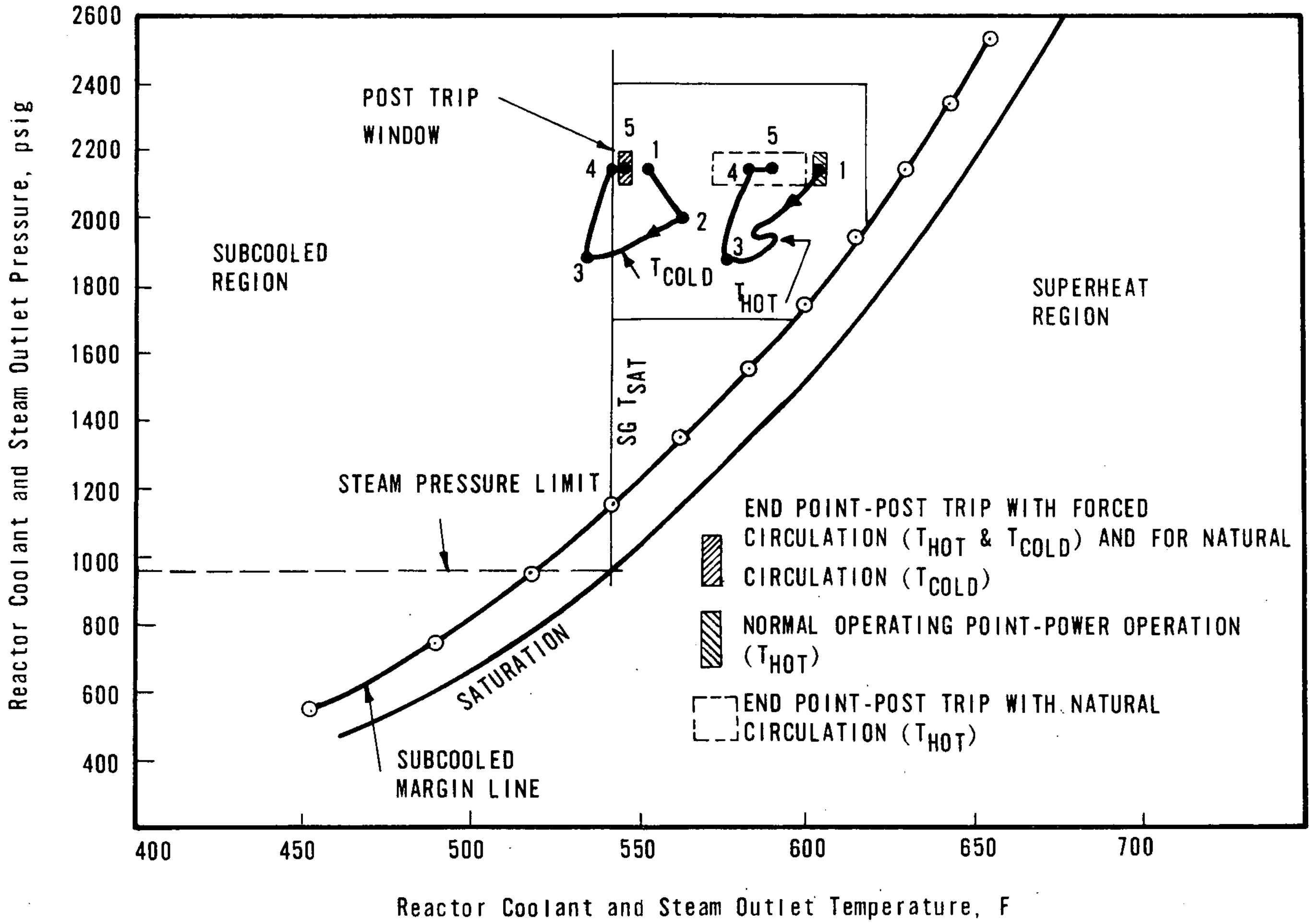
FIGURE 2

ILLUSTRATION OF PARAMETERS CONTRIBUTING TO NATURAL CIRCULATION DRIVING HEAD



$$\Delta P_{\text{driving head}} = h_{eff} (\rho_C - \rho_H)$$

Figure 3 TRANSITION TO NATURAL CIRCULATION USING EFW



Reference Points	Time (Seconds)	Remarks
1	0	RCP's trip; reactor trip.
2	60-90	T_{cold} reaches maximum value.
3	400	OTSG's at required level; RCS pressure at minimum value; recovery of RCS pressure begins.
3-4	>400	Steam pressure being restored by decay heat; TBV's shut.
4-5	>400	RCS pressure normal. OTSG pressure still low due to initial injection of EFW.
5	Depends on available decay heat	Steady state; TBV's begin relieving steam. Primary $\Delta T \sim 40F$.

CHAPTER B

USE OF THE P-T DIAGRAM

Introduction

The previous chapter provided the fundamentals of reactor heat transfer control and also presented information about natural circulation, subcooling, saturation and superheating. These basics are the background information needed to diagnose transients and follow through with the correct operator actions. This chapter builds on that information.

The foundation for abnormal transient diagnosis and operator action is the reactor coolant pressure-temperature diagram (P-T) which is used to show how changes of heat transfer affect plant operation. Examples of reactor coolant system pressure and temperature response for normal trips are shown; the response is also shown for a few selected abnormal events. These examples will show the difference between transients in which all systems and equipment function properly and those which have several failures.

The P-T diagram is used to identify a transient "type". There are two general "types" of transients which cause the core to steam generator heat transfer to be abnormal: overheating (inadequate heat transfer), and overcooling (excessive heat transfer). Changes of the amount of subcooling can also occur for a number of reasons. The P-T diagram can be used to find out in general what may be wrong and can be used to

narrow down the number of possible failures. Observing the P-T diagram is the first step for abnormal transient diagnosis; the second step is to observe a few pertinent parameters associated with the "type" of transient to narrow down the possible failures.

The P-T diagram will be used to monitor actions taken by the operator to see if they are producing the right effects. When equipment failures cannot be found or cannot be fixed the P-T diagram can be used to follow the effects of operator corrections as the plant is steered toward the best possible condition.

The diagram may also be used to ensure the plant has stabilized after a transient has been terminated.

Description of the P-T Diagram

Figure 8 shows the P-T diagram with information pertinent to normal power operation. The features of plant power operation that this diagram shows include the saturation line which applies to both primary and secondary water and steam conditions. Above the saturation line is the subcooled water region; below it is the superheated steam region.

The reactor coolant information displayed also shows the RPS trip envelope. A small window shows the expected normal reactor power operation point. This point is based on T_{hot} leg; if $T_{cold\ leg}$ were

shown on the figure it would be to the left. The size of the window is based on an expected approximate instrument error and also an allowance from the desired setting due to ICS control of minor plant variations. Actual "normal" power operation could be anywhere within this window and be acceptable. This window is a "moving" window because T_{hot} will change as plant power goes up and down.

Steam generator outlet pressure is shown as a line crossing the saturation line, and steam generator outlet temperature is also shown. The point where these two lines cross in the superheat region is the "normal" steam outlet operating point at power. The amount of superheat is shown as the difference between the saturation temperature (where the steam pressure line meets the saturation curve) and the steam operating temperature. The amount of superheat will change when the power level changes. (Note: In an actual P-T display, superheat will only be shown if steam temperature is measured. If steam temperature is calculated for steam pressure it will always show saturation temperature even at power.)

Figure 9 shows a P-T diagram for post-trip conditions. Most of the feature of Figure 8 are also shown on Figure 9. The important difference between Figures 8 and 9 is a line that shows the subcooling margin from the saturation curve. This subcooling margin line is only to be used to gauge the condition of the reactor coolant and not

the steam generator fluid. Because the reactor coolant conditions around the loop can be different and because the conditions can be different from one loop to the other this line must be compared to reactor coolant pressures and temperatures in the hot and cold legs of both loops. The amount of subcooling margin was chosen based on the ability to accurately measure the reactor coolant temperatures and pressures (instrument errors) during degraded reactor building environmental conditions (LOCA or SLB). It also includes an extra 5F to allow for temperature variations from the point of measurement in the system. This will give assurance that the reactor coolant is truly subcooled when above the margin line and has the ability to move the heat from the core to the generator.

If the subcooling margin is lost, the assumption should be made that subcooling has been lost (ie., the RCS is at saturation). The subcooling rule that was given in Addendum A should be invoked (it is repeated here):

> HPI Subcooling Rule
>
> Two HPI Pumps should be run at full capacity when:
>
> The ES is actuated and the HPI is automatically started.
>
> The reactor coolant subcooled margin is lost and the HPI is manually started.

NOTE: All three HPI pumps start on automatic initiation but only two are required. Therefore, if all three are operating properly, the operator should secure one of the HPI pumps supplying Train 'A'.

The P-T diagram can also be used to monitor and control HPI and RC pump operation. When HPI is initiated it can only be throttled when the subcooling margin is regained. In general, if the RC pumps have been tripped they can be restarted anytime the subcooling margin is regained and OTSG level exists (i.e., heat sink available). Exact details of HPI and RC pump control are given in the chapter called "Best Methods for Equipment Operation".

Figure 9 also shows a "post trip" operating window. This window has been drawn to show where the reactor coolant pressure and temperature should end up after reactor and turbine trip. The size of the window has been compiled from a review of several actual reactor trips (plus computer simulations) with and without equipment failures; its size is not exact and it is possible for a trip (with minor failures) to end slightly outside the window and still have a stable plant. Some judgement will have to be applied. However, this window gives a good "first" basis for determining if the plant is operating correctly after a trip. If the reactor coolant system pressure and temperature move outside the window after trip <u>and</u> do not return in a fairly short time (about 2 to 3 minutes) then an abnormal transient is in progress and

operator corrective actions are needed. A review of other plant readouts may be required to find out the exact cause. After the corrective actions have been taken the plant will be stabilized and the stable point can be inside or outside of the window (Criteria for plant stability are given in the chapter entitled, "Post Accident Stability Determination".)

An abnormal transient is also indicated by the steam pressure and steam saturation temperature lines. Generally if steam pressure falls below 960 psig after trip, some failure has occurred and the operator should begin a diagnosis of the plant. A steam temperature of 542F corresponds to 960 psig, therefore, if steam temperature is lower than 542F after a trip an abnormal condition is indicated. A loss of reactor coolant to steam generator heat transfer may also be noted when T_c does not follow T_{sat} in the steam generator.

The "post trip window" shows two end points: One is for natural circulation. When the RC pumps are off T_{cold} will be nearly the same as steam temperature but T_{hot} will be greater. The value of T_{hot} will depend on the decay heat level. The other end point shows forced circulation. When the reactor coolant pumps are running T_{hot} and T_{cold} will be almost the same after trip and both will be almost the same temperature as steam temperature. Nearly every trip will end at either the forced or the natural circulation point if <u>all</u> equipment operates

correctly and no equipment failures have happened. If some minor equipment failures have occurred (a leaky steam safety valve for example) the end point will be somewhere else inside the window.

The post trip window is a good gauge for determining if systems are operating correctly after a trip. If the reactor coolant temperature and pressure path stay inside this window or if the transient path goes outside this window slightly but returns, then the transient is going as expected and the core cooling with steam generator heat transfer is correct. However, severe excessive feedwater transients must be discovered <u>before</u> the transient path goes outside this window. This will be discussed in more detail later. If the reactor pressure and temperature are moving away from this window and do not return then an abnormal transient is in progress and corrective actions for abnormal transients should be implemented. These corrective actions are directed toward restoring control of reactor-steam generator heat transfer which is the preferred method for core cooling.

Successful transient mitigation can end with reactor temperature and pressure inside the window, but the plant can be stabilized outside the window. In some cases it is desirable to achieve stability outside this window.

Figure 9 also shows steam pressure; as illustrated its value is at the 960 psig "lower" steam pressure limit. After trip steam pressure will normally be approximately 1010 psig. Steam temperature is also shown. After trip the steam temperature should decrease to the steam generator saturation temperature (approximately 546F) which is set by the steam generator pressure of 1010 psig. (Note: In an actual P-T display, steam temperatures will always be shown at saturation temperature if it is calculated from steam pressure rather than measured.)

Steam pressure and temperature are very important parameters to review to determine if the plant is working correctly after trip. These two parameters in combination with reactor coolant pressure and temperature, will show if the secondary side is: 1) removing the right amount of heat from the reactor coolant, and 2) indicate if the reactor coolant is transporting the core heat to the steam generator so the steam generator can remove the heat. It is important to note that other parameters that are not displayed on the P-T diagram must also be checked to ensure proper primary to secondary heat transfer. For example, excessive main feedwater will not initially cause noticeable steam generator pressure or temperature reduction. By the time excessive feedwater causes the transient path to leave the post-trip window, the overcooling of the reactor coolant will cause the pressurizer to be in a nearly-drained condition. Therefore, main feedwater flowrates and SG levels must be checked very early following a reactor trip.

Heat Transfer Characteristics Shown by the P-T Diagram

This section will show examples of various transients on the P-T diagram. Both normal and abnormal transients are shown for comparison. The transients to be illustrated include:

- A normal reactor-turbine trip with no failures
- Transients that show the effects of equipment failures before trip
- Transients that show the effects of single and multiple equipment failures after trip.

These examples are used to show how reactor coolant system pressure and temperature and steam pressure change when different failures cause changes in heat transfer.

P-T Transient - Normal Trip

Figure 10 shows the typical response of both primary and secondary plant parameters following a reactor trip. Individual important parameters are shown as well as the P-T diagram. The shape of the reactor coolant P-T characteristic path from power operation (above 15%) to hot zero power is always like this unless an abnormal transient is in progress. The "dip" of the curve is due to cooldown of the RCS

to near T_{sat} of the steam generators for the turbine bypass system (TBS) setpoint. The cooldown results in coolant shrinkage which causes a pressurizer outsurge and pressure reduction. After the RCS reaches a temperature slightly above T_{sat} of the SG's, the reactor coolant will repressurize and stabilize. Depending on prior power operating history the low point of the "dip" will have different values, but the characteristic shape will always exist. When the plant trips the steam pressure will initially rise to the safety valve setpoint and then quickly settle out at the post trip turbine bypass valve setpoint and steam temperature will fall from the superheated condition to saturation temperature (if steam temperature is measured; if derived from steam pressure, saturation temperature will always be shown).

A similar P-T characteristic shape can also be seen for some abnormal transients, especially those that are caused by secondary side overcooling. Small LOCA's which depressurize the RCS "slowly" will not show the characteristic repressurization upturn (unless they are very small leaks or they are isolated). Individual parameters are shown versus time to show the approximate time for stabilization. Since stabilization takes a certain amount of time the overcooling characteristic can mask failures that would not show up while the "overcooling" trend exists. Since overcooling can be caused by too much feedwater or low steam pressure, one of the immediate post trip operator actions includes a review of the steam pressure, MFW flow, and steam generator level to assure that the trip is "normal" and not combined with an over-cooling transient.

Indications of a normal trip as shown by the P-T diagram include:

1. Hot and cold leg temperatures will stabilize in 2-3 minutes.
2. Reactor coolant pressure will stabilize in 5 to 6 minutes.
3. T_{cold} will be nearly equal to saturated steam temperature indicating that reactor coolant is transferring heat to the steam generators.
4. Steam pressure will stabilize in 2 to 3 minutes.
5. Reactor coolant subcooled margin will increase.

P-T Characteristics - Abnormal Transients - Before Trip

Although many transients will go so fast that operator action before trip is unlikely, the changes in displayed parameters prior to trip can provide clues as to the type of transient (overheating, overcooling, etc.). When the reactor trips the trend of the accident can be covered up by the P-T change caused by the cooling effects of the trip so the characteristics that occur in the short time before trip can help identify the trend.

Operator action in response to a change from the normal position in the P-T window may be possible, and trip may be avoided, but usually transients will happen too fast for the operator actions to be successful. Nevertheless, some of the indications before trip will help to determine what may be occurring.

Figures 11, 12, 13,and 14 show pre-trip movements on the P-T diagram. Steam pressure and RC temperature and pressure will respond differently depending on the cause. The events represented by these curves are:

Figure 11 - Overheating Transient
Figure 12 - Overcooling Transient
Figure 13 - Overpressure Transient
Figure 14 - Depressurization Transient

P-T Characteristics - Abnormal Transients - After Trip

Figures 15, 16, 17, 18, and 19 show examples of transients which may occur because of failures either on the primary or secondary side. These examples show transients which end as expected and also go past the expected point because of additional failures. Those transients which are corrected properly will follow the expected course and will end up in the "post trip window" near the normal post trip end point. When the path goes outside the window, the transient is defined as abnormal and the direction reactor coolant pressure and temperature move toward can be classified as overheating or overcooling. In combination with overheating or overcooling the reactor coolant temperature and pressure path can also move toward more or less subcooling.

These trends, overheating, overcooling, and loss of subcooling, are the first indications to check in transient diagnosis and correction. In the case of overcooling, which can be masked by the normal post-trip response, other parameters such as MFW flow and SG levels, which are not shown on the P-T diagram, must be checked very early in the transient.

An abnormal transient will show different characteristics depending on the failures that may have occurred. Some characteristics of RC pressure and temperature and of steam pressure that show undesired heat transfer on the P-T diagram are:

1. Reactor coolant subcooled margin is lost

   - The trend may be caused by overheating or overcooling.
   - The trend may be caused by loss of reactor coolant. Subcooling will be lost for all except the very smallest breaks.

2. Steam pressure is much lower than normal

   A value of 960 psig has been established as a limit similar to the "post trip" window for the Reactor Coolant P-T. If steam pressure drops below this limit after trip, then an abnormal condition may exist. A corresponding value of 542F has also been chosen for saturated steam temperature.

   - Steam pressure may be low because of a failure in the steam lines. Overcooling will result. Subcooling may or may not be lost.
   - Steam pressure may be low because of a loss of all feedwater. Overheating will result. Subcooling will be lost.

- Steam pressure may be low because a large amount of reactor coolant has been lost and cannot pass core heat to the feedwater in the generator to create steam. Large LOCA's can cause this or an Inadequate Core Cooling (ICC) situation can cause this. Both LOCA and ICC are discussed in detail as separate topics later.
- Steam pressure may be low due to excessive EFW (or MFW through the upper nozzles). Overcooling will result and subcooling may be lost.

3. Steam generator saturation temperature and Tcold do not correspond (not coupled) (Lack of primary to secondary heat transfer)

   - When T_{cold} does not change when T_{sat}-SG changes, then heat transfer from the reactor coolant to the steam generator is interrupted. Natural circulation has probably stopped when this occurs and the reactor coolant may heat up. The reactor coolant condition can be subcooled or saturated. If the reactor coolant is superheated, natural circulation has been lost.

The transients used as examples are:

Figure 15 - Loss of Main Feedwater

- 15a) Shows Loss of Main Feedwater with EFW actuated. The important feature of this transient is that the main feedwater heat sink is quickly replaced with an EFW heat sink; the trend looks similar to a normal reactor trip.

- 15b) Shows Loss of Main Feedwater with EFW delayed. Important features of 15b) are: 1) loss of steam pressure, and 2) the reactor coolant heats up and would eventually saturate at 2500 psi. This is an indication of lack of primary to secondary heat transfer.

Figure 16 - Small Steam Line Break

- 16a) Shows an unisolable break that is terminated by stopping main feedwater and EFW and allowing the generator to boil dry. The important feature is that the reactor coolant was overcooled before isolation; after isolation when the "bad" generator boiled dry it was no longer able to remove heat from the reactor coolant. The "good" generator, which is pressurized, is the heat sink; it allows the reactor coolant to return to stable subcooled conditions.

- 16b) Shows an unisolable break that is not terminated. Continued feeding of FW and boil off causes extreme reactor coolant overcooling. Since HPI is running the RCS might be overpressurized at low temperature violating NDT limits.

Figure 17 - Excessive Feedwater

- This transient is shown to be corrected by ICS operation and looks similar to a normal trip. Were the transient to continue, water could enter the steam lines and cause damage but the amount of damage and its effects are not known. The RCS would overcool to saturated conditions (i.e., drain the pressurizer) by the time water entered the steam lines. An example of a severe excessive feedwater transient (disregarding possible steam line damage) is given in Appendix A of Volume 2.

Figure 18 - Small Break LOCA in the Pressurizer Steam Space

- The important feature of this transient is that water will flow into the pressurizer from the reactor coolant loops. Although the pressurizer will show a level it is not a good indication of reactor coolant inventory when the reactor coolant is saturated.

- 18a) Shows a LOCA with the break isolated after the accident starts. Refill and repressurization of the reactor coolant system allow a normal cooldown with a pressurizer bubble.
- 18b) Shows a LOCA that is not isolated. Subcooling does not return as quickly although the entire reactor coolant system fills with water. Cooldown after this accident will be with a pressurizer full of water.

Figure 19 - Small Break LOCA in the RCS Loop Water Space

- This transient is different from Figure 18 because the pressurizer does not fill with water from the loops as a result of the break.
- 19a) Shows a small break with MFW used to remove heat.
- 19b) Shows the same break with no MFW and EFW delayed. The effect of the heat transfer to the steam generators can be seen by comparing the RC pressures with and without FW. With no FW the RC system pressurizes to 2500 psi. At this pressure the leak rate is highest and HPI flow is lowest; use of the steam generator helps to reduce the leak flow and increase the HPI flow to cover the core. 19b also shows that steam pressure is lost because no steam generator inventory exists to create steam.

Figure 8 POWER OPERATION P-T DIAGRAM

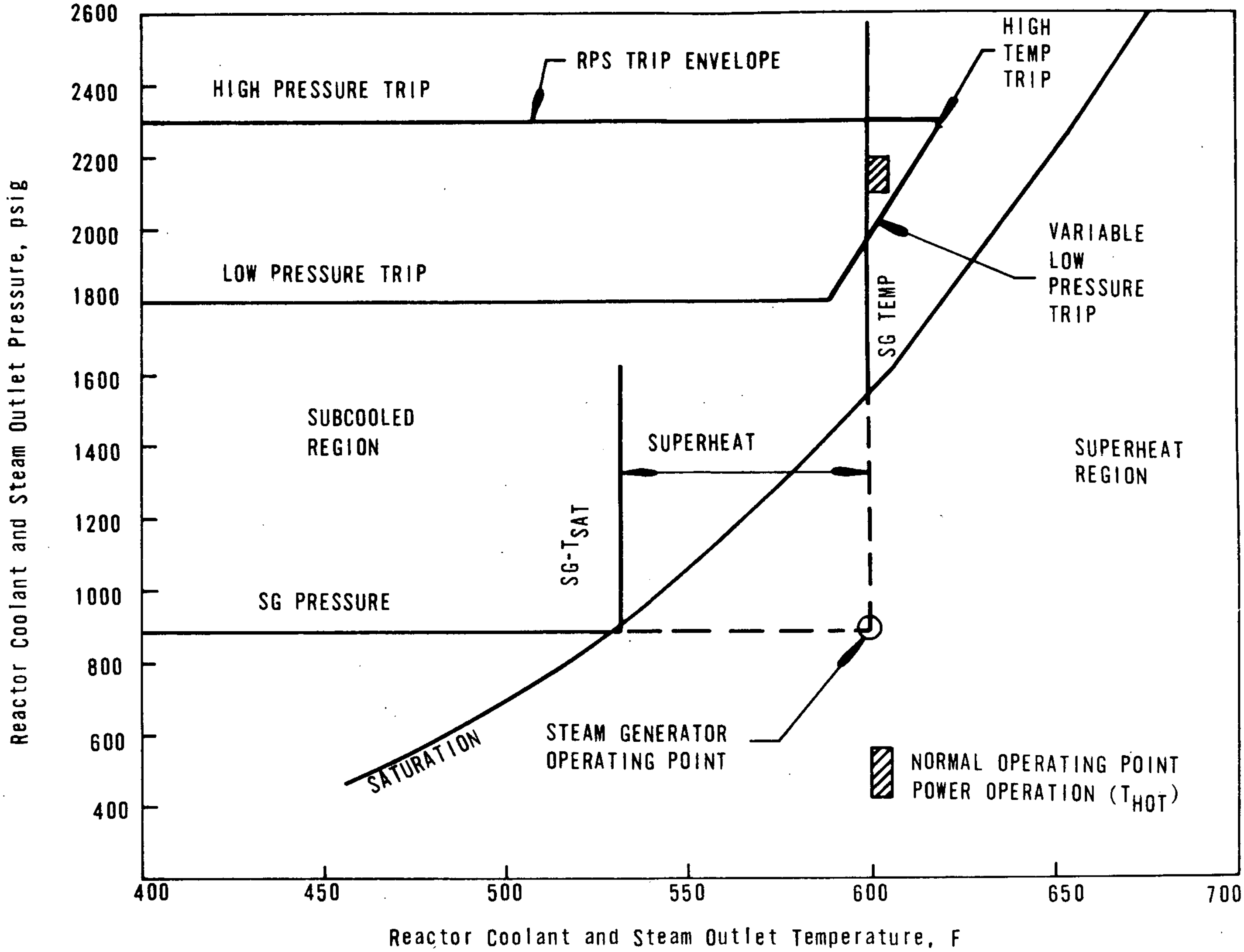


Figure 9 POST TRIP P-T DIAGRAM

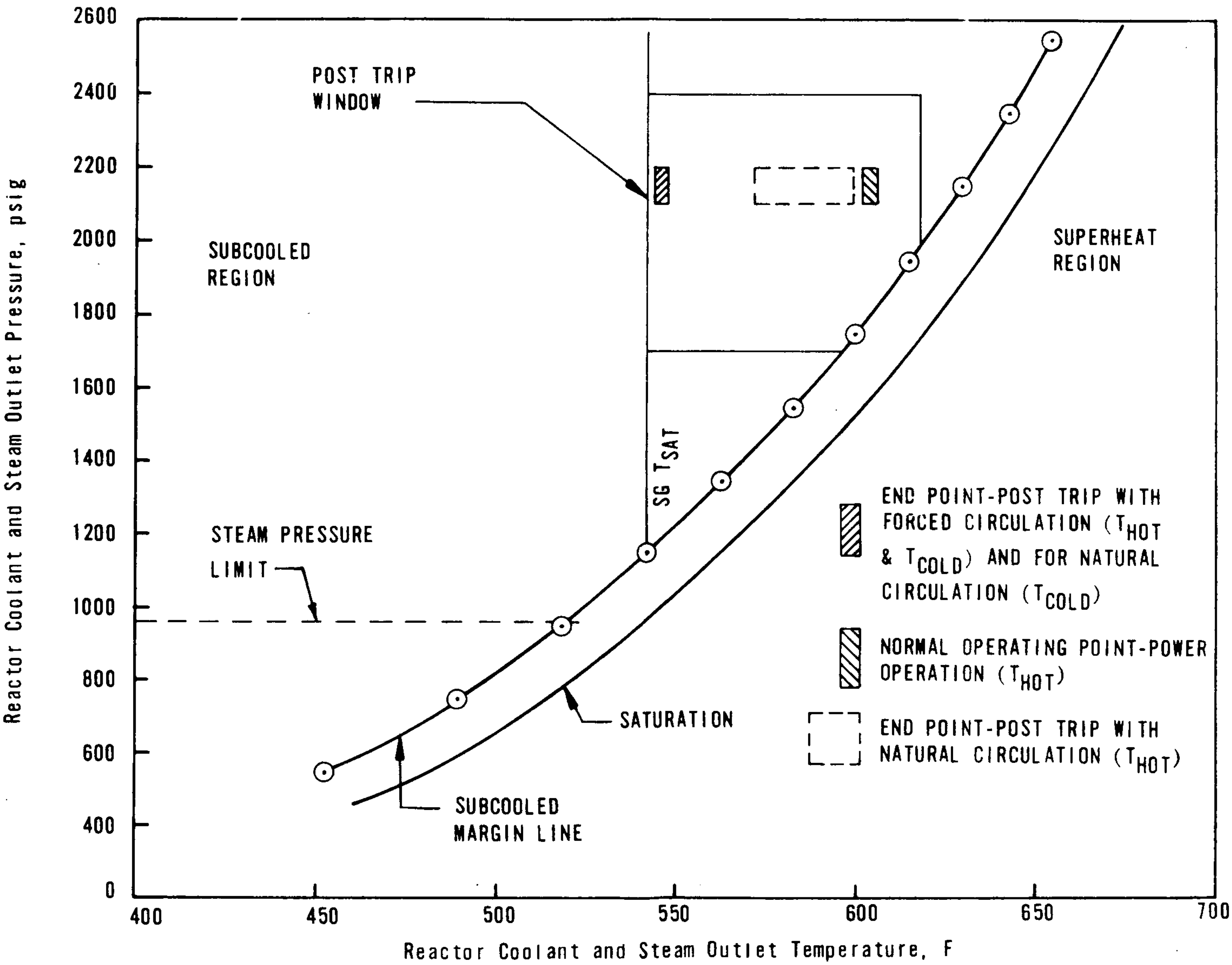


Figure 10 TYPICAL RESPONSE OF MAJOR PLANT PARAMETERS FOLLOWING A REACTOR TRIP

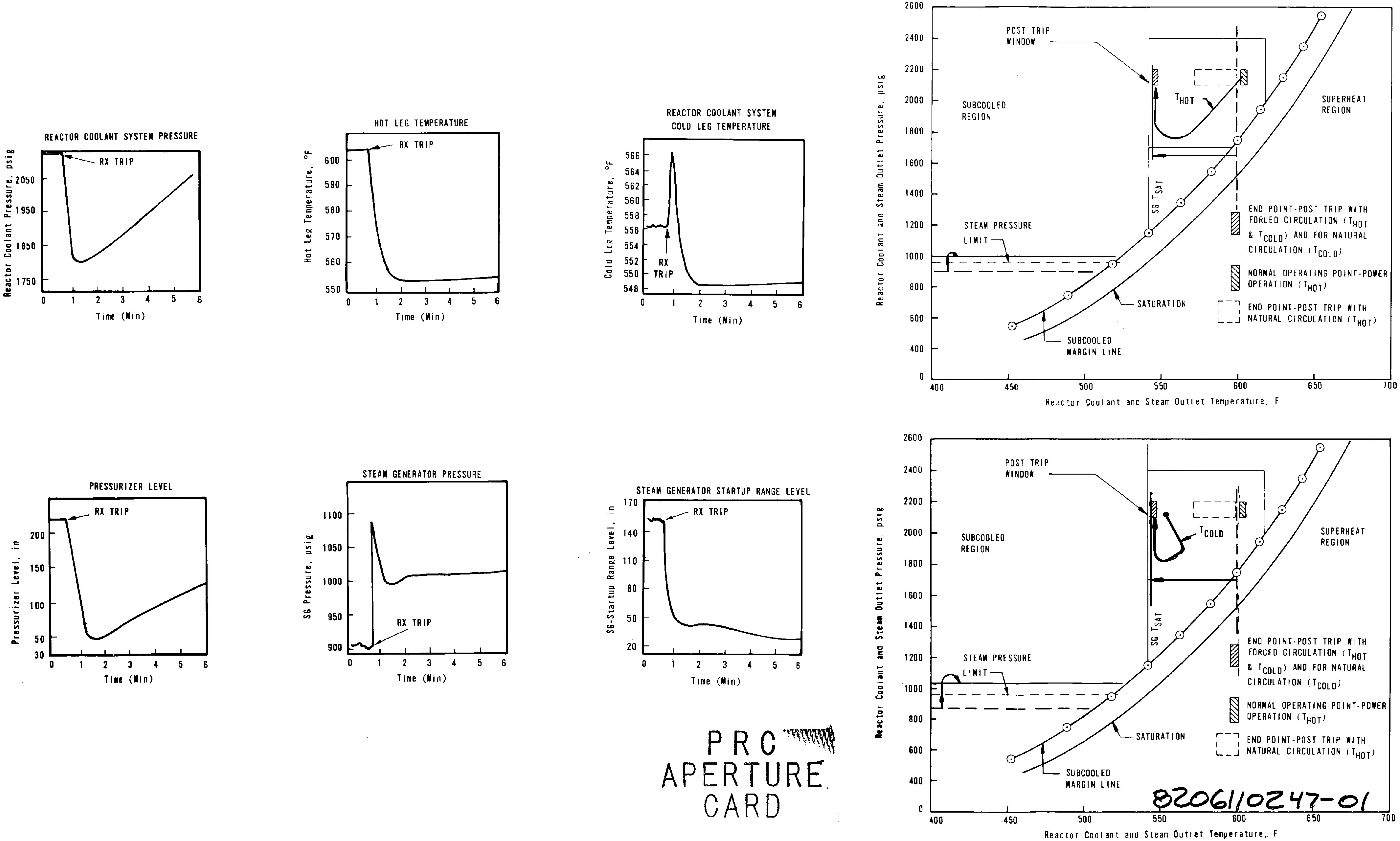
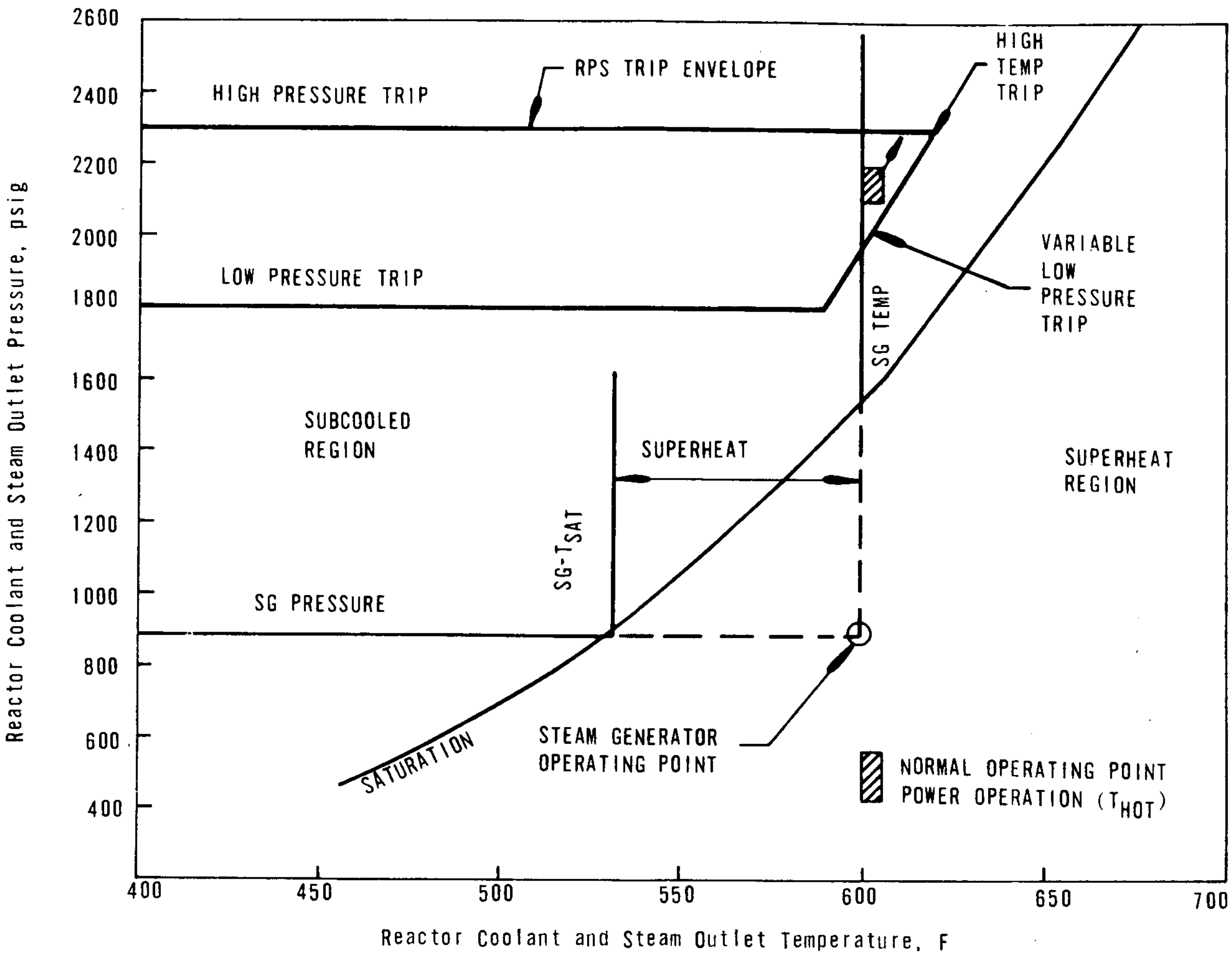


Figure 11 OVERHEATING TRANSIENT (PRE-TRIP)



Plot shows an increase in both Pressure and T_{hot}. A slight increase in superheat and steam pressure is also possible.

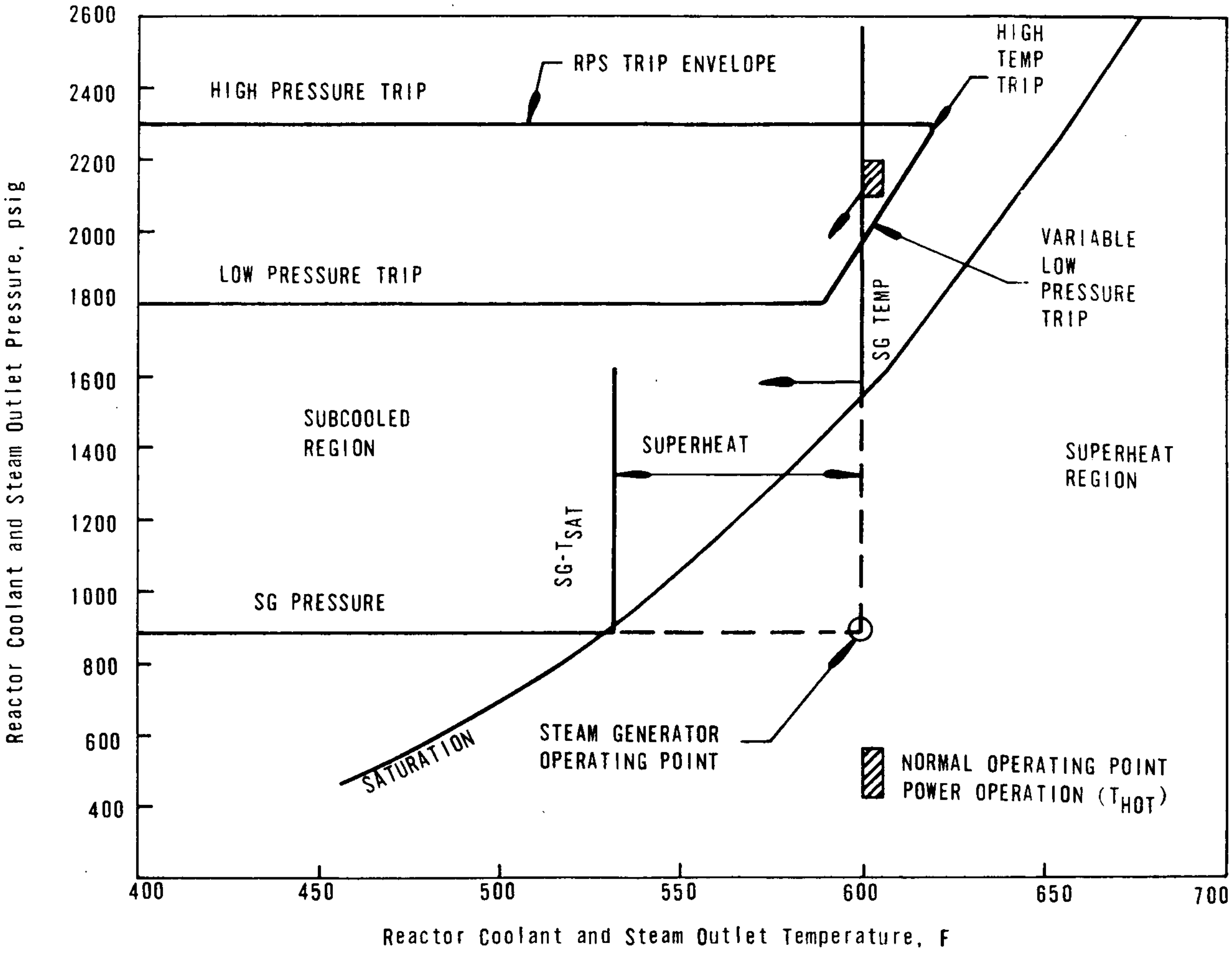
Possible Causes

- Decrease in or loss of main feed-water
- ICS malfunction causing steam pressure increase (Turbine valves closing)

Possible Alarms

- High - RC Pressure
- High - Pressurizer level
- Low - MFW Pump Flow
- Low - MFW Pump Suction Pressure
- High Main Steam Temperature

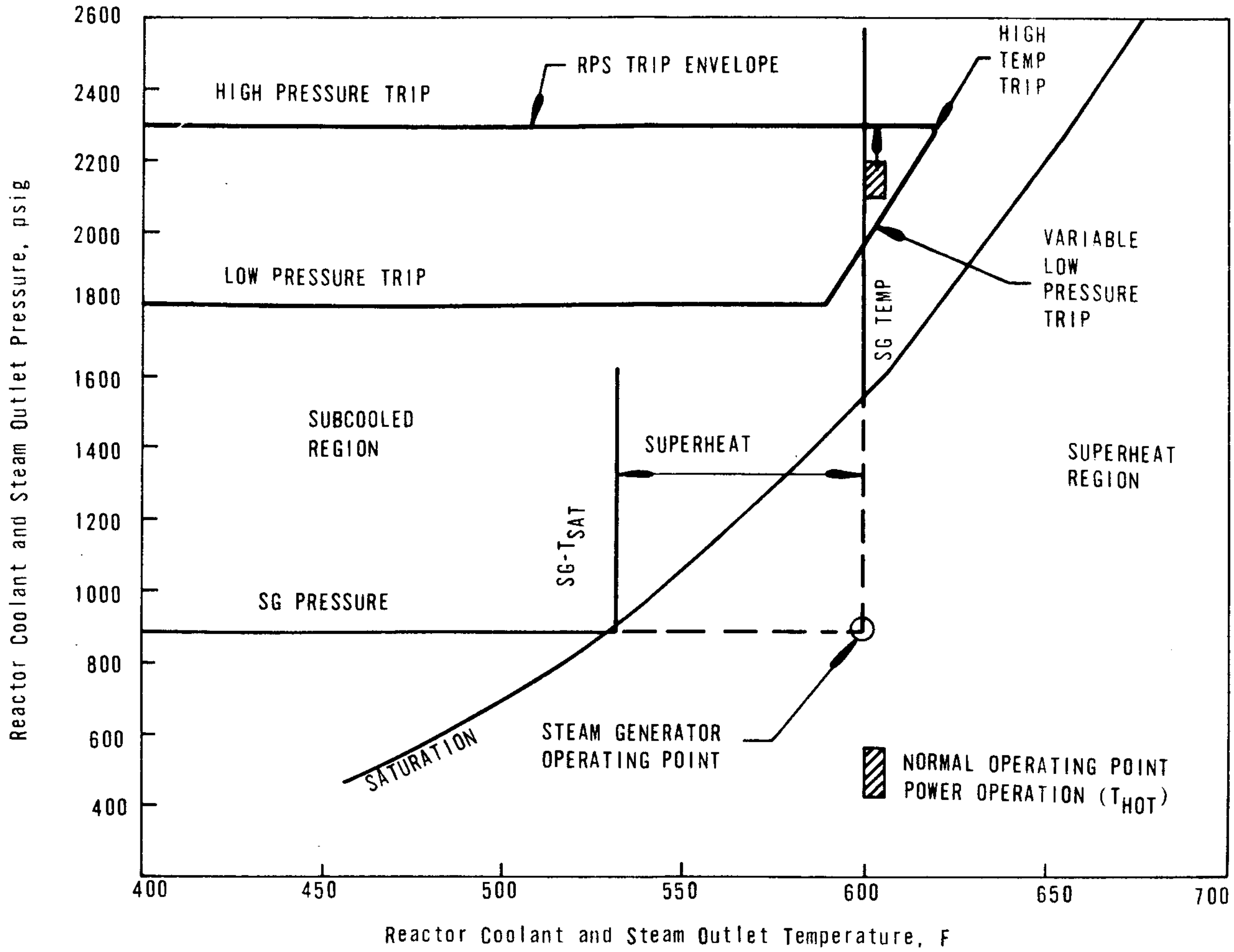
Figure 12 OVERCOOLING TRANSIENT (PRE-TRIP)



Plot shows a decrease in both RC Pressure and T_{hot}. A drop in superheat and SG pressure is also possible depending on the event.

Possible Causes	Possible Alarms
• Excessive Feedwater	• Low RC Pressure
• Decrease in Feedwater Temperature	• Low Pressurizer level
• Decrease in Steam Generator Pressure (steam leaks)	• High Makeup Flow
	• High Steam Generator level

Figure 13 OVERPRESSURE TRANSIENT (PRE-TRIP)



Plot shows an increase in RC Pressure with little or no change in T_{hot}.

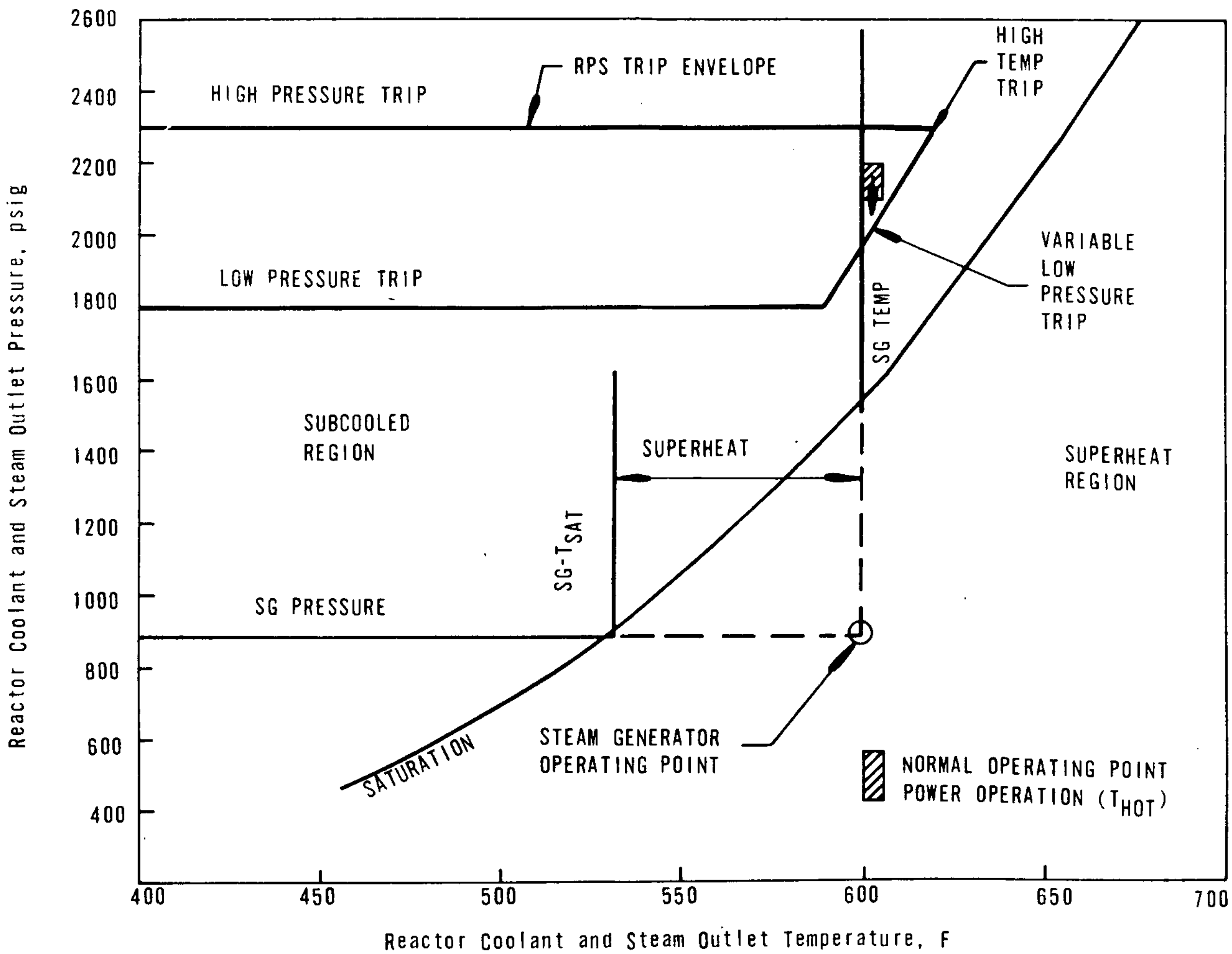
<u>Possible Causes</u>

- Too much Makeup
- Insufficient Letdown
- Pressurizer Heater Misoperation

<u>Possible Alarms</u>

- High Reactor Coolant Pressure
- High Pressurizer Level
- High Makeup Flow

Figure 14 DEPRESSURIZATION TRANSIENT (PRE-TRIP)



Plot shows a decrease in RC Pressure with little or no change in T_{hot}.

Possible Causes	Possible Alarms
• Loss of Makeup; too much letdown	• Low RC Pressure
• Pressurizer Spray Misoperation	• Low Pressurizer level
• Small LOCA	• Makeup Pump Trip
	• Low Seal Injection Flow
	• High or low Makeup Tank level

Figure 15 LOSS OF MAIN FEEDWATER

Reference Points	Time (Seconds)	Remarks
1	0	Accident starts. Main feedwater pumps trip.
1-2	0-40	RCS P&T decrease due to loss of fission power. Typical post trip response (overcooling trend) with a corresponding decrease in pressurizer level.
2	40	EFW delivered to both steam generators, with level controlled at low level (RC pumps on).
2-3	40-225	System stabilizes with decay heat removal through the steam generators. RC temperature approaches the saturation temperature for the secondary pressure, and pressurizer level increases because of makeup.
3-4	225-600	Pressurizer level restored and steady with normal RC pressure restored by the pressurizer heaters.
4	600	STABLE PLANT CONDITIONS

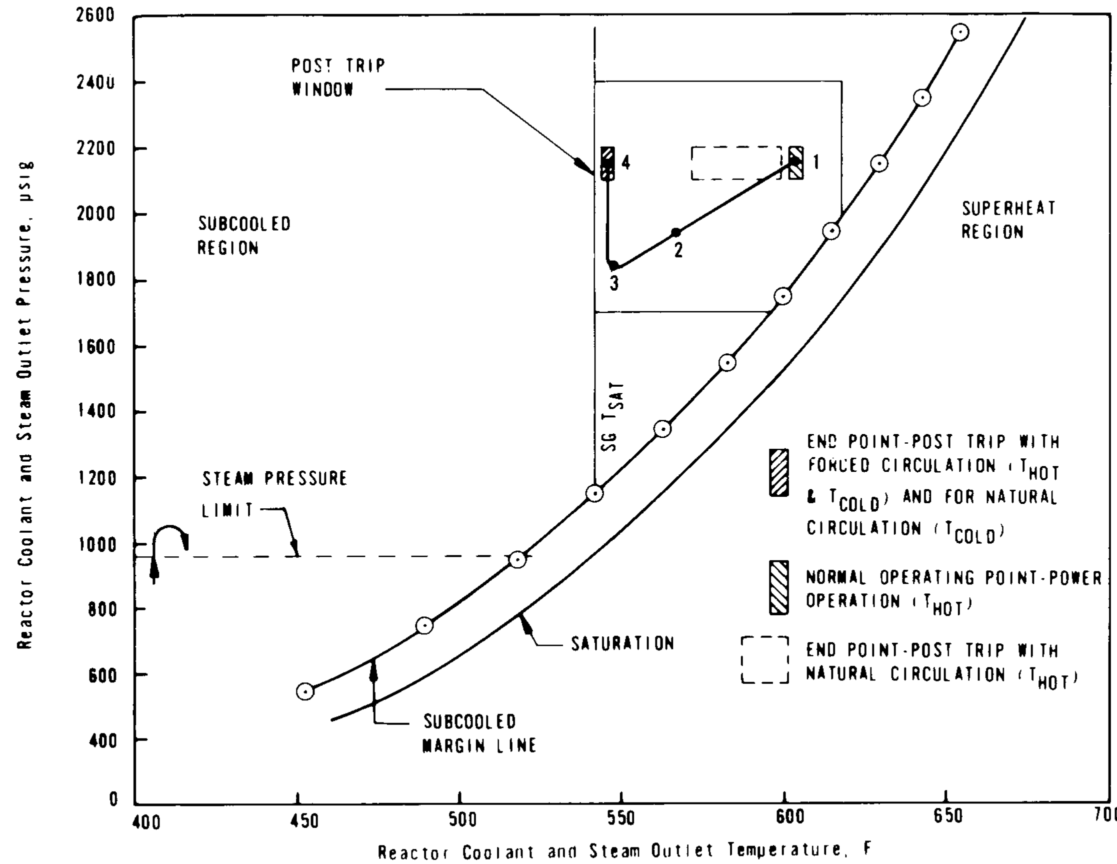


Figure 15b. TOTAL LOSS OF MAIN AND EMERGENCY FEEDWATER

Reference Points	Time (Seconds)	Remarks
1-2	0-40	Same as Figure 15a.
2	40	EFW fails; no feedwater is delivered to either steam generator.
2-3	,0-50	System continues to exhibit post trip overcooling trend; feedwater inventory is being boiled off.
3	50	Steam generators are almost dry.
3-4	50-175	RCS reheats and repressurizes due to loss of primary to secondary heat transfer. Pressurizer level also increases.
4	175	PORV lifts and cycles to control RCS pressure.
4-5	175-620	PORV controls RCS pressure. RC temperature continues to heatup; the primary system is approaching saturated conditions.
5	620	Pressurizer fills with water. RC pressure would stay at the PORV or safety valve setpoint and saturated primary conditions would develop. Plant is left in an abnormal condition; EFW and/or HPI must be established to maintain core cooling before water boils out of the vessel.

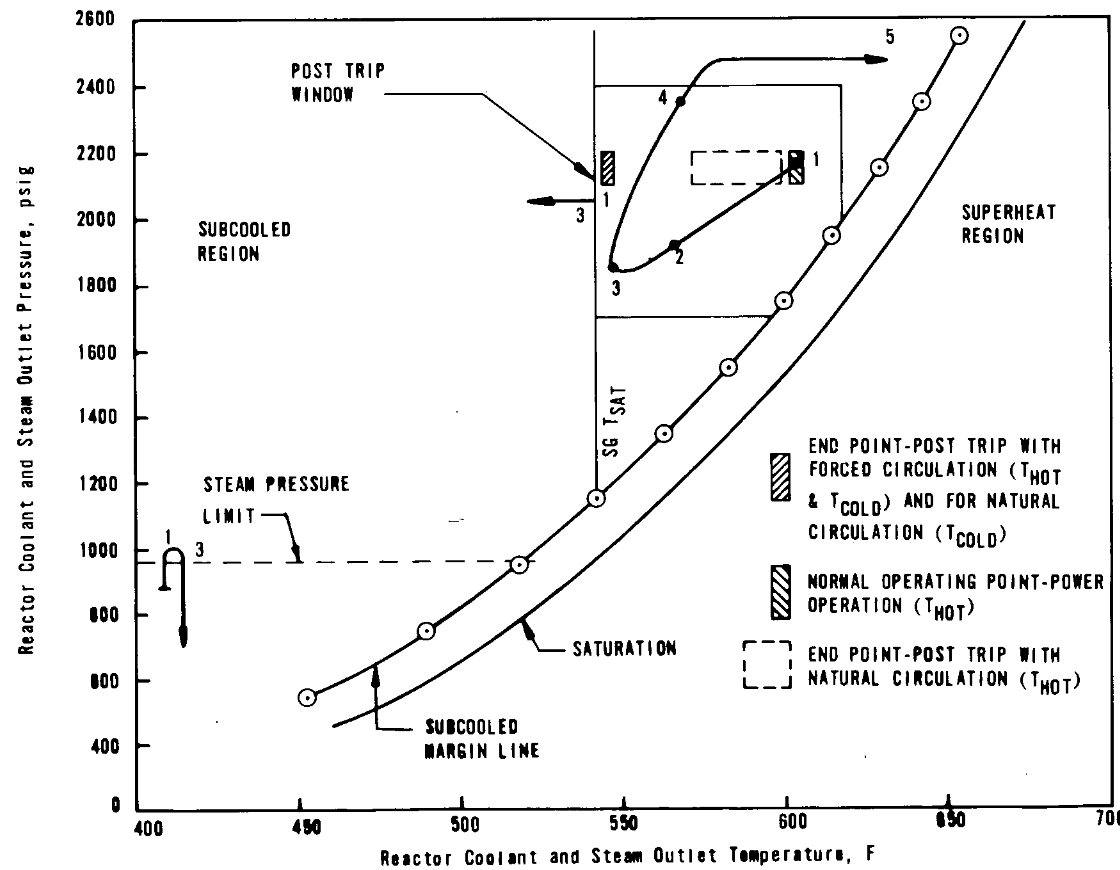


Figure 16 SMALL STEAM LINE BREAK (0.5 FT^2)

Reference Points	Time (Seconds)	Remarks
1	0	SLB occurs (0.5 ft^2 leak).
1-2	0-5	Increase of steam flow causes slight reduction in T_{av}. ICS attempts to keep T_{av} up by pulling rods.
2	5	Reactor trip on high flux turbine trip.
2-3	5-20	RCS P&T drop due to loss of fission power and excessive primary to secondary heat transfer.
3	20	ES actuation on low RC pressure; pressurizer level indication off-scale low.
3-4	20-30	RCS P&T continues to decrease. During this time, the operator trips the RC pumps.
4	30	Hot leg saturates; steam voids exist in top of hot legs.
4-5	30-120	RCS P&T conditions decrease along saturation curve. After turbine trip, one steam generator repressurizes while affected steam generator pressure is very low (<300 psi). Operator isolates MFW to depressurized steam generator.
5	120	Affected steam generator boils dry. Good generator is removing decay heat by use of MFW.
5-6	120-500	HPI collapses steam voids. RCS returns to subcooled state and pressure increases as pressurizer level is restored. RCS is slowly reheating.
6	500	Subcooled margin has been restored with increasing pressurizer level.
6-7	7500	Operator throttles HPI and controls steam pressure in good steam generator to prevent water solid conditions as the RC repressurizes. Plant is left in a stable, subcooled condition.

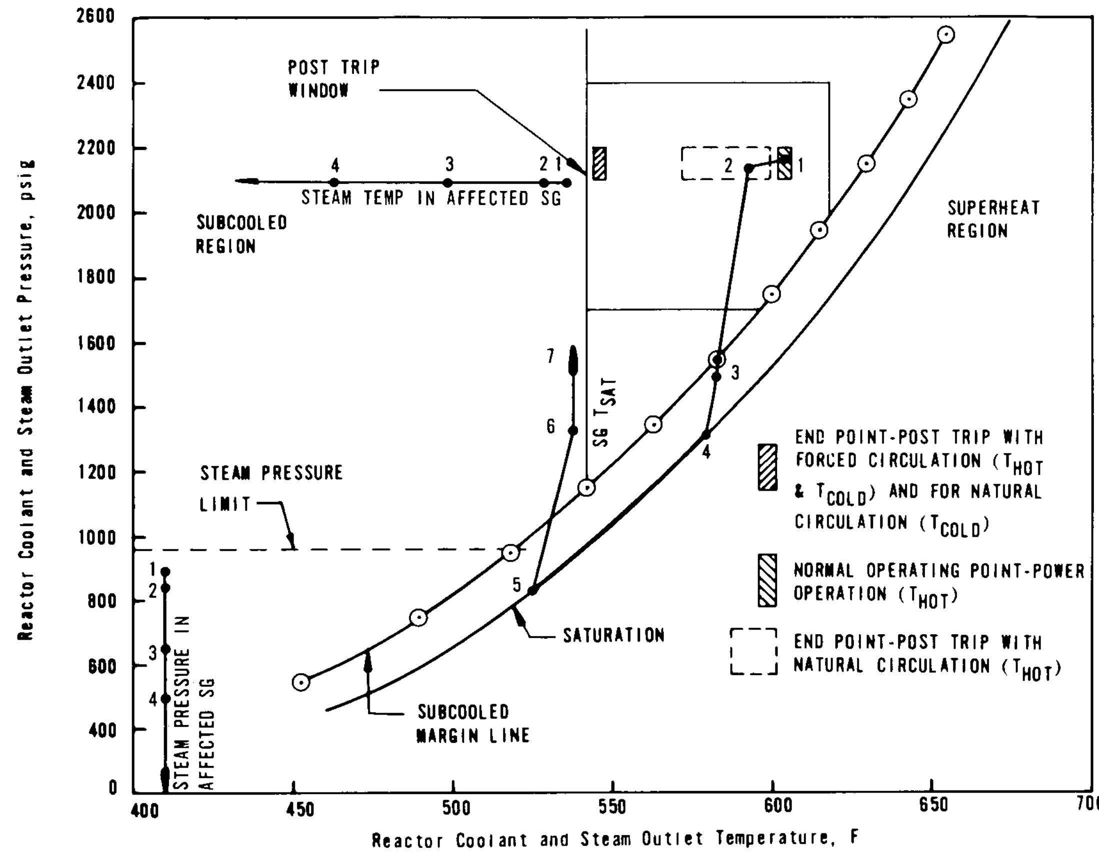


Figure 16b. STEAM LINE BREAK (UNISOLABLE) WITH NO OPERATOR ACTION TO ISOLATE FW TO AFFECTED STEAM GENERATOR

Reference Points	Time (Seconds)	Remarks
1-4	0-30	Same as Figure 16a.
4-5	30-250	MFW delivered to both generators (most of the flow would go to the depressurized generator). RCS P-T continues to drop along saturation line until HPI and primary to secondary heat transfer collapses voids.
5	250	RC returns to subcooled state.
5-6	250-500	RCS continues to overcool due to continued addition of FW to the affected steam generator. Pressurizer will refill with cold water.
6	$\geq$500	RCS left in abnormal condition. FW to affected generator must be isolated. Following FW isolation, HPI must be throttled and steam pressure controlled in good generator to limit potential for solid water condition and violation of NDT limits.

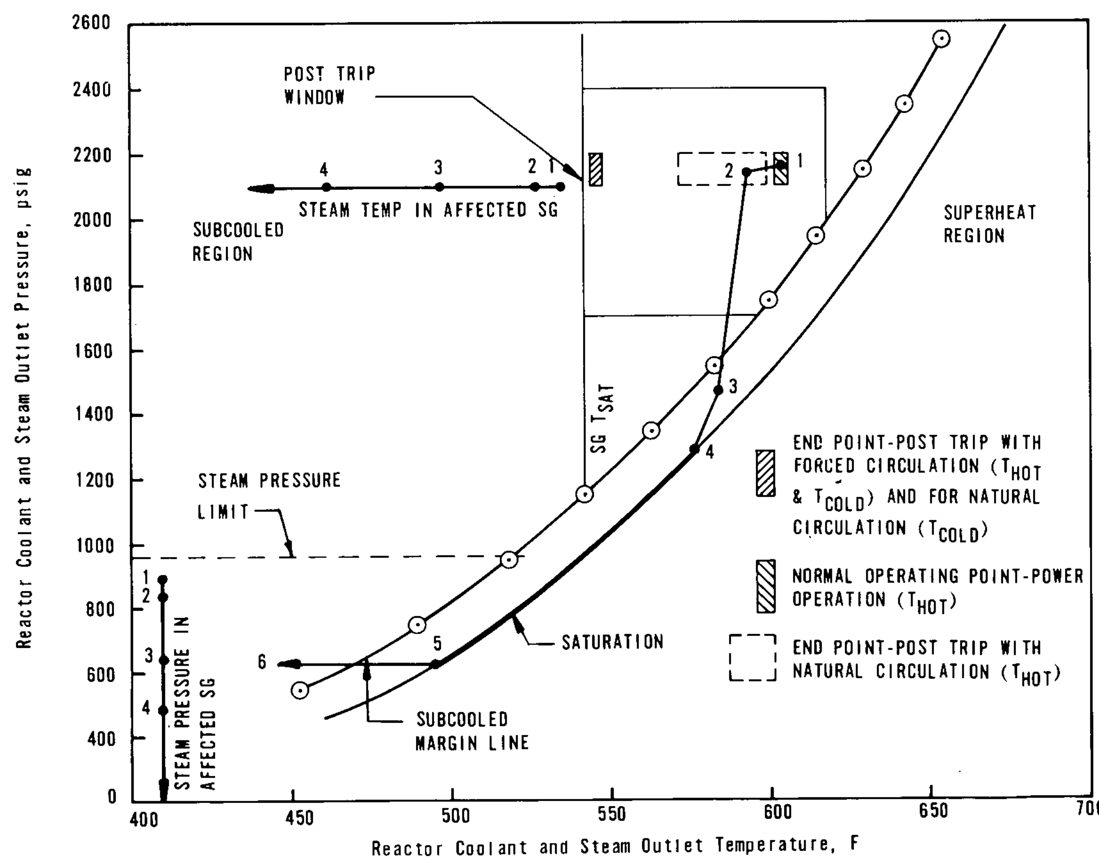
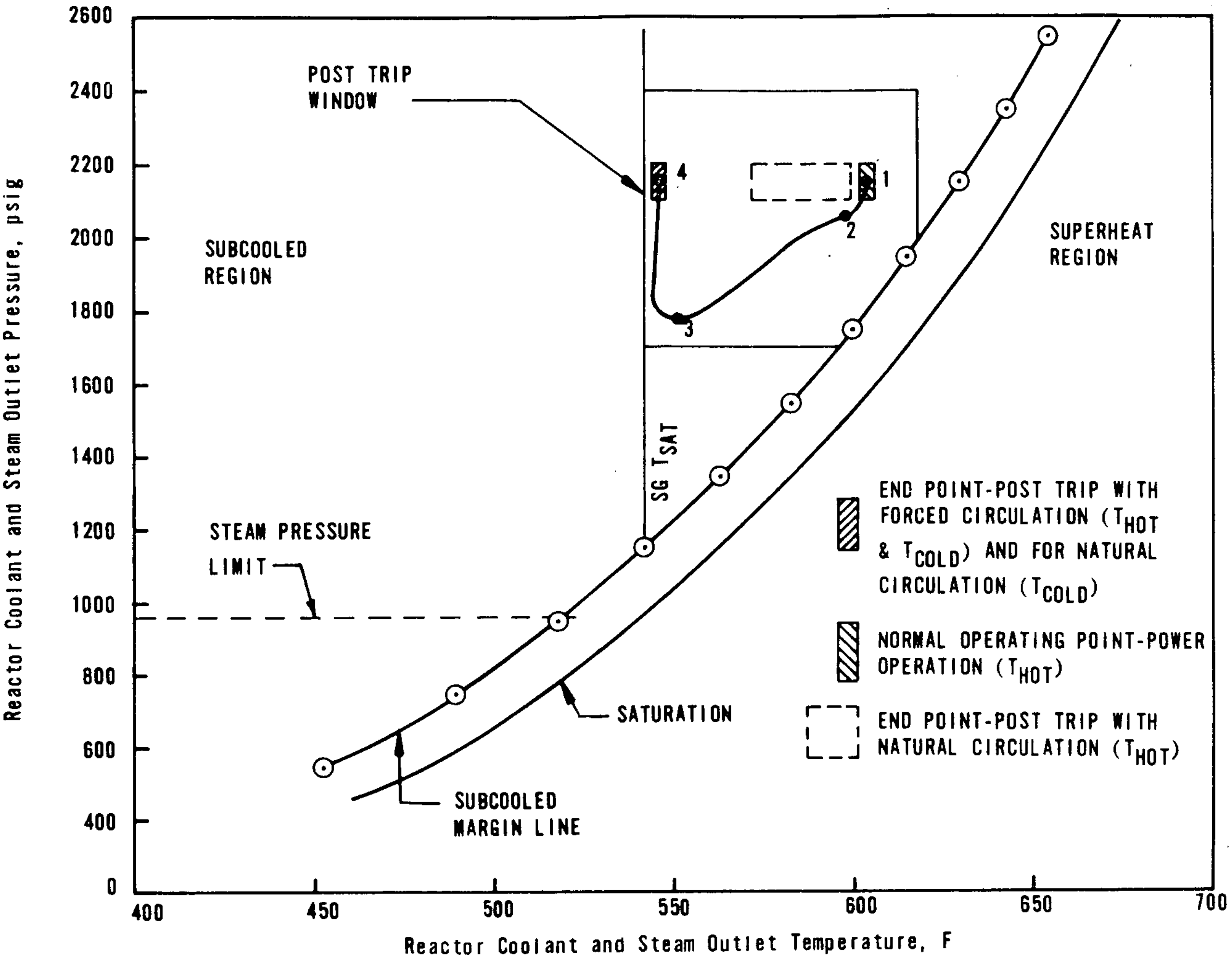


Figure 17 EXCESSIVE FEEDWATER



EXCESSIVE MAIN FEEDWATER ADDITION TO ONE STEAM GENERATOR (DURING POWER OPERATION)

Reference Points	Time (Seconds)	Remarks
1	0	With the plant operating at 100% power, a failure of the MFW pump controller allows pump overspeed. 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}, but rod withdrawal is limited by the high flux limiter.
2	60	Manual reactor trip.
2-3	60-200	RC P&T decreases due to loss of fission power and higher than normal secondary inventory. The ICS initiates a feedwater run-back 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 operation of MU system and pressurizer heaters. Primary system is left in a stable, hot shutdown condition.

Figure 18 SMALL LOCA IN PRESSURIZER STEAM SPACE

Reference Points	Time (Seconds)	Remarks
1	0	PORV assumed to open.
1-2	0-60	Pressure drops due to discharge of pressurizer steam out of PORV. Little or no change of RC temperature occurs. An insurge of reactor coolant into pressurizer would occur and an increase in pressurizer level would be observed.
2	60	Reactor trip on low RC pressure.
2-3	60-125	RCS P&T decrease due to loss of fission power and primary to secondary heat transfer; general post trip-overcooling trend results. Pressurizer level drops. MU can't keep up with the leak, and RC pressure drops.
3	125	Subcooling margin lost; operator trips RC pumps and starts HPI. Level automatically controlled to 50% on operate range with MFW flow through the upper nozzles.
4	185	Hot leg saturates and an insurge into the pressurizer occurs. Operator action is assumed to isolate the PORV block valve. LOCA is isolated.
4-5	185-600	HPI in combination with condensation of the RC steam on the generator tubes leads to collapse of steam voids within primary system. System returns to a subcooled state and repressurizes as pressurizer level is restored to an indicated level.
5	600	Subcooling margin established; operator throttles HPI and restarts RCP's.
5-6	600-700	Operator stops HPI and restarts normal MU and letdown.
6	700	STABLE PLANT CONDITIONS.

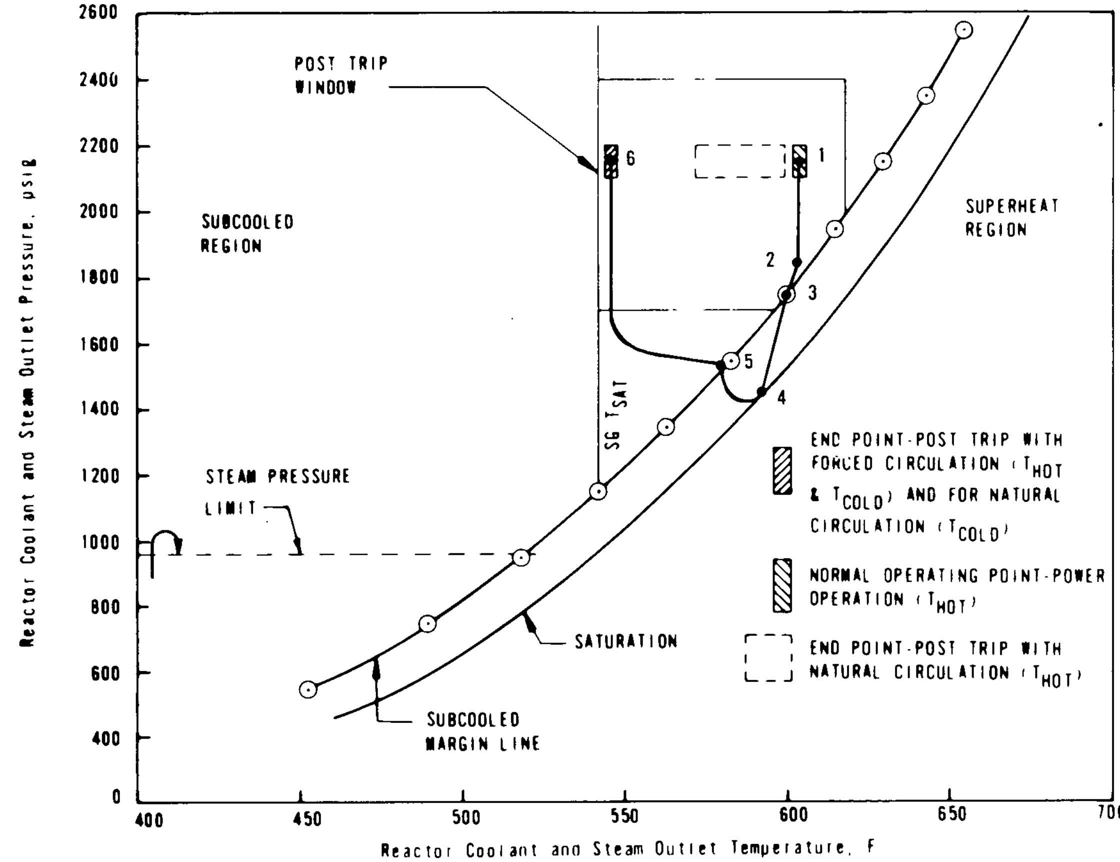


Figure 18b. STUCK OPEN PORV (NO ISOLATION)

Reference Points	Time (Seconds)	Remarks
1-4	0-185	Same as Figure 18a except PORV is not isolated, leak continues and operator raises OTSG levels to 95%.
4-5	185-400	RCS is in two-phase natural circulation condition. P&T conditions decrease along saturation line and stabilize at about 1200 psia. Pressurizer level is increasing as pressurizer steam space depletes.
5	400	Pressurizer level indicates full scale; leak flow changes from steam to a steam-water mixture.
5-6	400-1000	Quench tank ruptures. Primary system remains stable at approximately 1200 psia with the hot leg saturated. HPI exceeds core boil-off and steam voids are slowly being collapsed.
6	>1000	Operator initiates plant cooldown and depressurization to place plant in a safe condition. For this size break a return to a subcooled state would be expected during cooldown. Solid-water cooldown would be required thereafter unless PORV is isolated.

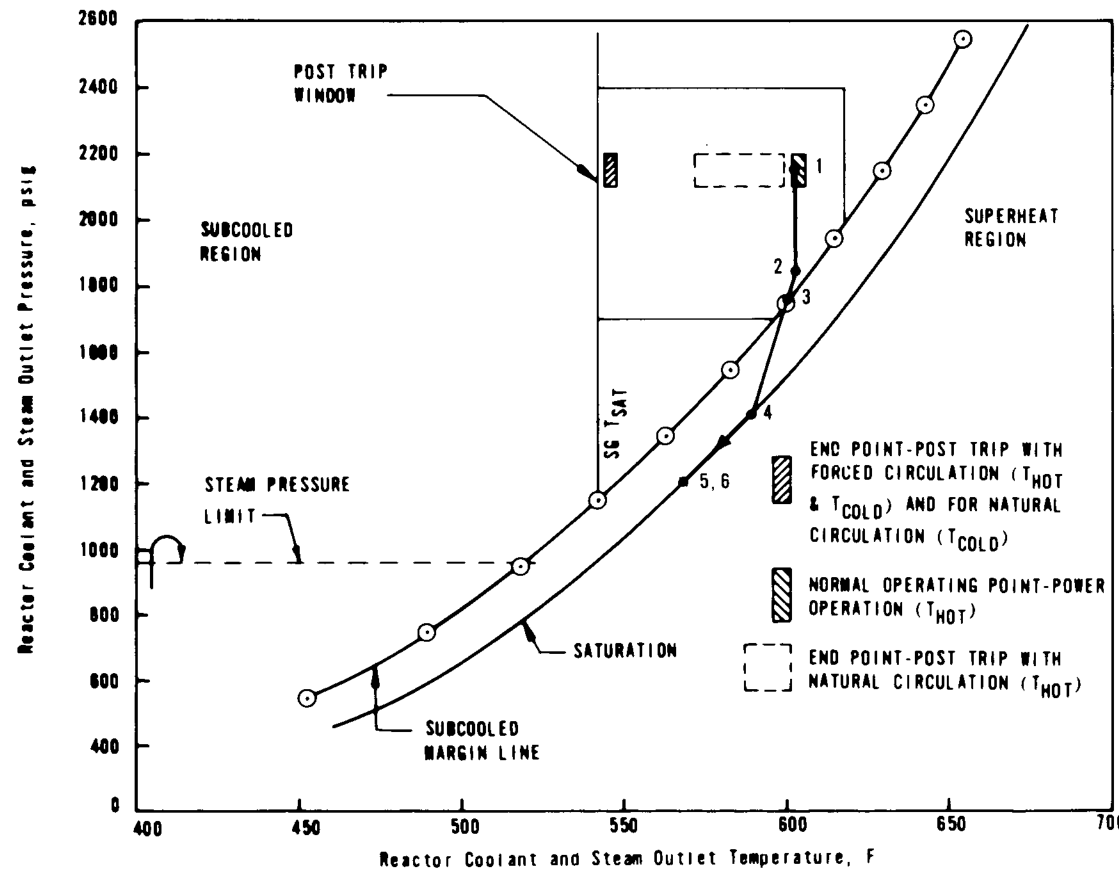


Figure 19 SMALL LOCA IN RCS WATER SPACE

Reference Points	Time (Seconds)	Remarks
1	0	LOCA occurs; break is equivalent to 1.35 in OD hole at discharge of RC pump.
1-2	0-50	Pressure drops due to release of reactor coolant out break; pressurizer level decreases.
2	50	Reactor trip on low RC pressure.
2-3	50-90	RCS P&T drops due to loss of fission power and primary to secondary heat transfer; general post trip overcooling trend results. Pressurizer level indication goes off scale low. Because MU can't keep up with leak, pressure drops.
3	90	Subcooling margin lost; operator trips RCP's and starts HPI.
3-4	90-120	SG levels are automatically raised to 50% (when it reaches 50%, operator will manually increase to 95%).
4	120	Pressurizer drains and hot leg saturates.
4-5	120-600	RCS in two-phase natural circulation mode. P&T decrease along saturation curve and stabilize at approximately 1225 psi.
5	600	Two-phase natural circulation stops; steam bubble in top of hot leg prevents liquid carry over to steam generator and steam generator cannot remove heat.
5-6	600-1500	RCS repressurizes because all heat from core is going to reheat the reactor coolant. RCS stays on the saturation curve. Steam bubble in hot leg is slowly increasing in size; condensation of RCS steam on tubes is not yet possible. Pressurizer level is increasing.
6	1500	Boiler-condenser cooling established; RVS hot leg water level is low enough to allow RCS steam to condense on steam generator tubes.
6-7	>1500	Pressurizer level decreases. RCS P&T decrease along saturation curve and will stabilize at approximately 1200 psi. Operator should initiate a plant cooldown and depressurization to recover plant.

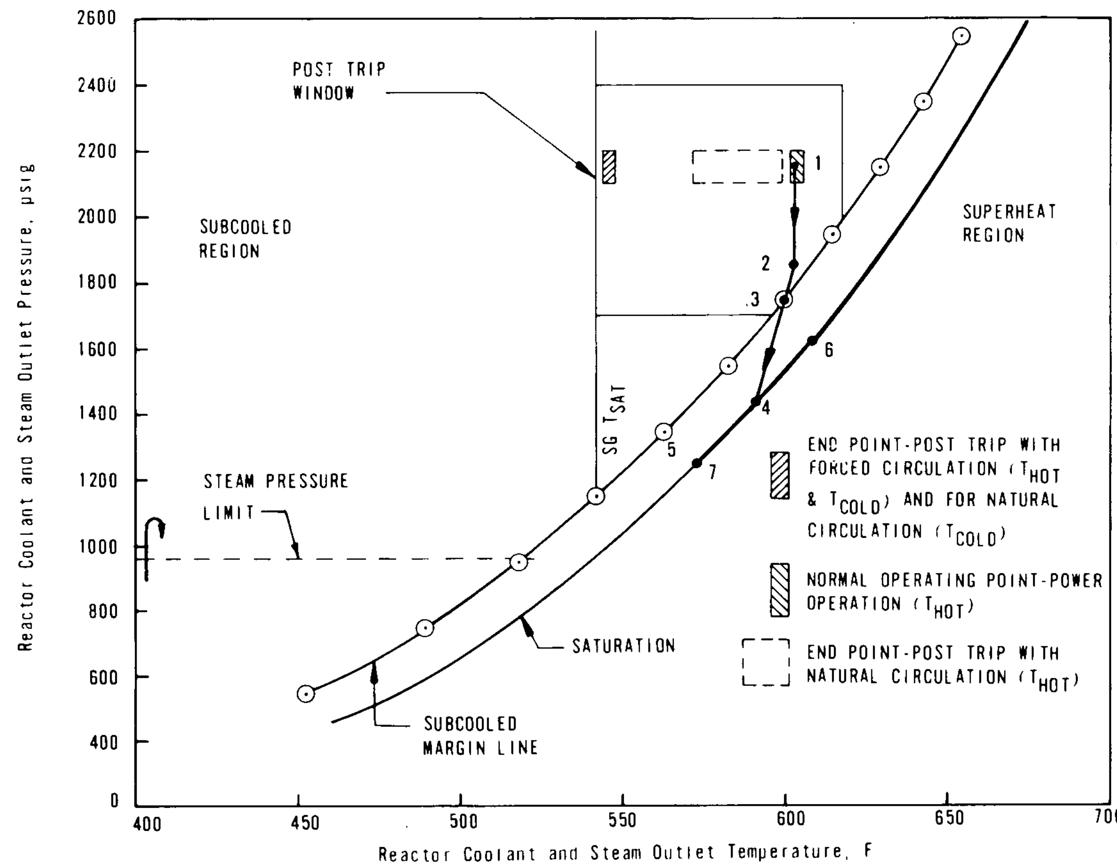
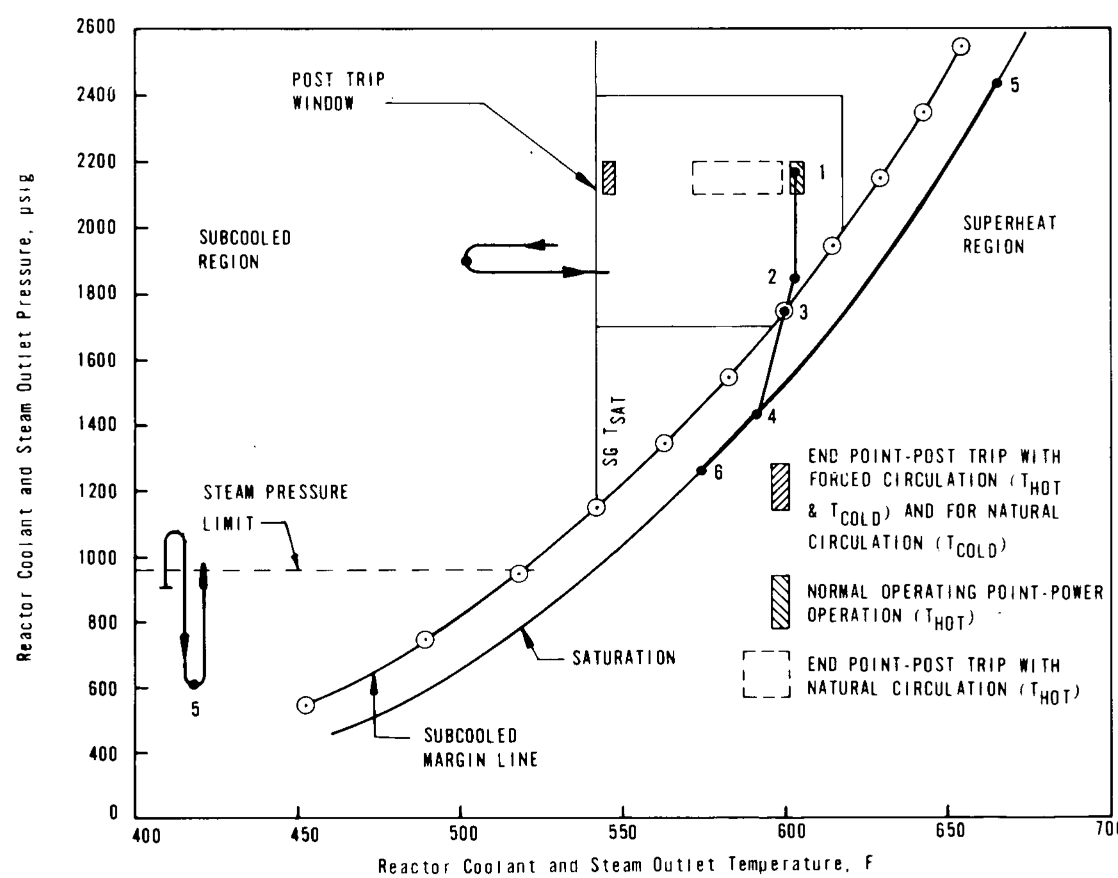


Figure 19b. 0.0 FT^2 BREAK AT PUMP DISCHARGE WITH LOSS OF MFW AND EFW DELAYED FOR 20 MINUTES (LOCA IN RCS WATER SPACE)

Reference Points	Time (Seconds)	Remarks
1-4	0-120	Same as Figure 19a. except that a total loss of feedwater has occurred.
4-5	120-1200	Steam generators boil dry; P&T increase along saturation line due to lack of primary to secondary cooling. Core cooling is being maintained by the HPI. Pressurizer level is increasing. Steam pressure will slowly drop once inventory is boiled off.
5	1200	Operator restores EFW system operation. EFW flow is started to both steam generators and steam pressure is restored.
5-6	>1200	With EFW on boiler-condenser cooling is started. The RCS P&T drops along saturation curve and will stabilize at approximately 1200 psi. Pressurizer level drops to zero indication because of coolant contraction. Operator should initiate a plant cooldown and depressurization to recover plant.



CHAPTER C

ABNORMAL TRANSIENT DIAGNOSIS AND MITIGATION

Introduction

The chapter shows how an abnormal transient can be diagnosed and mitigated using the information provided by the P-T diagram and the concepts on heat transfer discussed in the previous chapters. A simplified flow chart of the approach to be used to diagnose and mitigate an abnormal transient is provided in Figure 20A "General Plant Accident Mitigation". It is broken down into a few separate steps although these steps will "blend" together into one continuous process in actual practice. The Abnormal Transient Guideline Procedures are implemented whenever an automatic reactor trip occurs (although the operator may have manually tripped the reactor after recognizing something was wrong) or a forced shutdown is necessary.

The guidelines are provided in Part I. They list the appropriate operator actions necessary to mitigate an abnormal transient. They follow the approach outlined in Figure 20A. The guidelines incorporate the following features.

1. Use of the P-T diagram which provides a constant feedback to the operator on his success or failure after taking each step in Part I. This diagram should be checked frequently to make sure things are progressing as expected. It will thus give the operator early indications of subsequent failures that are delayed after the initial event, or multiple failures that were covered by the predominant event and didn't appear until that one was corrected.

2. The guidelines are constructed such that the operator makes an attempt to correct the problem with a given piece of equipment or system (e.g., EFW to correct loss of main feedwater). If that fails he is instructed to go on to the next available system (i.e., HPI cooling). The failure of the EFW system is not given priority attention in the body of Part I, protection of the core is.

3. The operator is given frequent "present plant status" (PPS) aids throughout the procedure to help him maintain proper orientation.

4. If new symptoms appear he is instructed to recycle (go to the section that treats that symptom) to the appropriate part of the procedure.

Immediate Actions

The first block in Figure 20A is the "Immediate Actions" block. The immediate actions should be completed in the first 2-3 minutes. The first action to be made is to determine if a reactor trip has occurred or plant conditions requiring a forced shutdown exist. If a reactor trip has occurred the operator should manually trip the reactor and turbine, then proceed to the next post trip step of the ATOG procedure which is "Vital Systems Status Verifications". However, if plant conditions warrant a forced shutdown, the operator should initiate the appropriate shutdown

should initiate the appropriate shutdown procedure. If a reactor trip should occur during the forced shutdown operations, the posttrip ATOG procedures should be implemented.

Vital Systems Status Verification

The next major block on Figure 20A is the Vital Systems Status Verifications. This section requires reviewing specific plant status items including the P-T diagram to determine if they are behaving as they should for a normal reactor trip. If the specific plant status items cannot be verified as performing as expected, the operator should perform the specified remedial actions. The procedures provide specific remedial actions for each plant status item which cannot be verified. The plant status items which are checked first are the normal automatic post trip functions which control core reactivity, primary and secondary inventory, and primary and secondary pressure.

During the performance of the immediate actions and status checks, certain conditions require specified "standard" actions. These conditions are:

- Reactor Trip
- ES Actuation
- Loss of grid power

These conditions can occur separately or in various combinations. The actions to be taken are given in Table 2.

In addition, two transients require a very fast check of indications and fast corrective actions. Excessive main feedwater requires the operator to quickly terminate feed flow* to prevent water spilling into the steam line, and steam generator tube rupture requires fast action to depressurize and begin cooldown to limit the offsite doses. Table 3 shows the actions required for these transients.

Finally, the operator needs to check the P-T diagram for "overcooling", "overheating", or "loss of subcooling" conditions. The P-T diagram is the foundation for transient diagnosis and for the actions to correct abnormal transients.

Abnormal Transient Diagnosis and Treatment

Although the type of transient may have become evident during the

*The operator should act to terminate excessive feedwater by tripping the MFW pumps rather than rely on the automatic MFW pump trip on high SG level.

first 2 or 3 minutes after trip, plant monitoring is required to make sure that the transient is going as expected. Generally, after 2 or 3 minutes the plant will begin to stabilize within the "Post Trip Window" (examples of this were given in the P-T Diagram Chapter). Actions have already been taken to identify and handle the "fast" transients and the systems which should be operating have been checked to make sure that they are working correctly. Further plant monitoring should begin. At this stage the effort should now be to make sure that the plant stabilizes as it should. To do this the P-T diagram is kept under surveillance. If reactor coolant pressure and temperature stabilize within the P-T post trip window, and steam pressure is above the low steam pressure limit, the transient is probably not abnormal and a quick check of the following should be made to ensure system and equipment parameters are within expected values:

Heat Transfer Balance Indicators

- P-T diagram (for RC pressure and temperature and subcooling)
- Pressurizer Level
- Steam generator level and pressure

Equipment Status and Operation (depending on what was started);

- Makeup/HPI flow rates and pump status
- Main or emergency feedwater flow rates and pump status
- RC pump operation including cooling water and seal injection service
- Position of important valves (letdown, PORV, feedwater isolation and control valves, pressurizer spray valve)
- Containment isolation and cooling systems
- Power supplies (AC and DC)

Once these reviews are completed a more thorough check can be conducted and a decision made to determine if the plant is stable. (See the Chapter on "Post Transient Stability Determination".)

But if the first review of the P-T indicates that the reactor coolant pressure and temperature are not going to remain within the post-trip window (or return to it), or that steam pressure is below the steam pressure limit, then something is wrong with heat transfer and corrective actions are required to bring the heat transfer into balance.

RC pressure will decrease toward the saturation pressure causing a "loss of subcooling margin".

The overcooling is caused by a temperature decrease on the secondary side of the steam generators. Since the secondary steam is saturated after reactor trip (no superheat) the steam pressure must also be decreasing. Figure 20B shows this decrease in secondary pressure and temperature. The steam pressure must go below 960 psig after RC trip for the RC temperature to go out of the post trip window on the low temperature side. The steam pressure before RC trip would have been around 900 psig but superheated. Following turbine trip the steam pressure will probably go up to the safety valve setpoint and then decrease down to the turbine bypass valve setpoint.

The characteristics of an "overheating" type transient are shown in Figure 20C. The figure shows the T_{hot} - RC pressure trace. Initially the RC pressure and temperature decrease after a reactor trip. However, as the primary to secondary heat transfer is lost the RC average temperature increases. As the temperature increases the RC expands causing the pressurizer level to increase and RC pressure to go up. This pressure increase will continue until stopped by the pressurizer PORV or safety valves opening. However, the RC temperature will continue to increase. This will lead to a "loss of subcooling margin".

Diagnosis

There are two general abnormal transient types:

- Excessive Primary to Secondary Heat Transfer - "overcooling".
- Not enough Primary to Secondary Heat Transfer - "overheating".

A loss of reactor coolant subcooling margin can also occur and can be combined with or caused by either "overheating" or "overcooling".

The characteristics of an "overcooling" type transient are shown on a P-T diagram in Figure 20B. The figure shows T_{hot} and T_{cold} coming together as the reactor coolant reaches an isothermal condition following reactor trip because the reactor cooloant pumps are running. If no RC pumps are running, T_{hot} and T_{cold} will not come together because a temperature difference will develop across the core which creates the thermal driving force for natural circulation of the reactor coolant.

The excessive heat removal by the steam generators will cause the average reactor coolant temperature to go down. As the temperature goes down the reactor coolant will contract causing the pressurizer level to go down. If the effect cannot be offset by increased makeup flow to the RCS the pressurizer level will go down which will cause the RCS pressure to also go down. If the pressurizer empties, the

Without heat being transferred to the secondary side, the secondary side steam will gradually cool which will also cause the SG pressure to decrease.

A direct "loss of RC Subcooling" without a simultaneous "overcooling" or "overheating" caused by a secondary system malfunction is shown in Figure 20D. This type loss of subcooling margin would be caused by a loss of RC inventory. This loss in inventory will cause the RC pressure to fall but without the large decrease in RC temperature associated with the "overcooling" type transient.

Mitigation

The path for correction is charted and shown on Figure 21, "Accident Mitigation Approach". The chart keys on the three general characteristics displayed on the P-T diagram: overcooling (too much steam generator heat transfer), overheating (insufficient steam generator heat transfer), and loss of subcooling. The chart is a reference table that ties together a wide variety of information for corrective actions. With the exception of LOCA, corrective actions for all abnormal transients are provided in this section. LOCA is discussed separately in considerable detail in Appendix F of these guidelines. Specific information for mitigation of overcooling and overheating transients are provided in Figures 22 and 23 respectively.

Corrective Actions for Overcooling (too much steam generator heat transfer)

Figure 22 shows the corrective actions to be taken for overcooling. The chart is largely self-explanatory so only a brief discussion will be given. Information provided by the chart will not be repeated.

Overcooling is always caused by failures on the secondary side. The usual sources of failure are low steam pressure or excessive main or emergency feedwater or by combinations of high feedwater and low pressure. The P-T diagram shown is typical for a more severe case of overcooling; usually excessive feedwater alone (unless severe and not terminated within 2-4 minutes) or small reductions in steam pressure will not cause loss of subcooling. But the general trend shown by the P-T diagram is characteristic of overcooling. Some LOCA's can also cause a loss of steam pressure because the RCS will depressurize, cool and draw heat away from the steam generators; this will be temporary for small breaks.

Once this trend is exhibited, checks should be made on steam pressure, steam generator level, loop T_{cold} temperatures, and main or emergency feedwater flow. If the cause is obvious, then actions to isolate the cause should be taken.

If the subcooling margin is lost during an overcooling transient, the subcooling rule should be followed; two HPI pumps should be turned

on and run until the subcooling margin is restored. When the subcooling margin is lost the RC pumps should be tripped and not restarted until subcooling is restored; the reason for this is that a LOCA may have occurred, and its P-T characteristics and overcooling characteristics are similar.

Overcooling transients induced by steam pressure and/or feedwater control failures on only one SG may be obvious when the T_{cold} temperatures are compared. If the magnitude of the overcooling is significant, T_{cold} in the affected loop will always lead T_{cold} in the loop with the good SG (i.e., T_{cold} in the affected loop will be lower).

Detection of overcooling by low steam pressure and determination of which generator has failed can be done by two methods. The first way is to stop all feedwater <u>when a level exists in both generators</u> (if level exists only in one generator the one without level is likely to be failed). When feedwater is isolated to both generators, the level should fall at a faster rate in the failed generator. Detection <u>is</u> possible before both generators boil dry and feedwater should be restored to the "good" generator before it dries out. A unique feature of steam leaks is that the steam generator with the low pressure will transfer the heat from the RCS and lower its temperature; the

"good" generator will not. Consequently the RCS temperature can fall below the temperature of the "good" generator. When this occurs the pressure in the good generator will drop below the TBS setpoint, they will close, and steam generator level will drop slowly or not at all. Consequently, one steam generator will retain level, so stopping all feedwater to both generators is not dangerous. However, when one steam generator boils dry, then the remaining generator will begin to transfer heat and level will drop. Feedwater must be restored before it is dry.

The second (but less reliable) way to identify a leaking generator is to compare the rate of drop of steam pressure in both generators. The failed generator will permit steam pressure to fall faster than the good generator will. A differential pressure between the two steam generators of about 100 psi, with the failed generator lower, will show the correct one to isolate. The 100 psi differential will show up rapidly for large leaks and slower for small leaks. A steam leak of about 5% total flow (about equal to one main steam safety valve stuck open) will show this trend within 3 to 5 minutes. However, this magnitude of pressure differential may exist only for a short duration. Therefore, comparison of level changes in <u>isolated</u> SG's is preferred, if the affected SG is not obvious. In very

rare instances it is possible, but not likely, that both steam generators will leak. The plant should be temporarily placed in HPI cooling with all feedwater stopped to see if repairs can be made. If a rapid "running repair" cannot be made the feedwater flow should be throttled considerably to both generators and the plant should be cooled down. This will be a very difficult operation. Thermal stress limits should be maintained. It is better to use main feedwater for this than EFW. A discussion of cooldown on one generator is given in the "Best Methods Section". Those principles will also apply here except two steam generators are to be used.

Corrective actions for excessive feedwater are shown on the overcooling diagnosis chart (Figure 22) and discussed in Appendix A.

Corrective actions for low feedwater temperature are also shown on Figure 22.

Corrective Actions for Overheating (not enough steam generator heat transfer)

Figure 23 shows the corrective actions to take when the reactor coolant cannot transfer heat to the steam generators. Reactor-to-steam generator heat transfer is not coupled; T_c and SG T_{sat} do not track together. In general there are three causes of insufficient heat transfer:

- There is no inventory in the steam generator to receive the heat (loss of all feedwater)
- The reactor coolant cannot transport the heat to the generator because there is insufficient inventory (LOCA)
- Circulation has stopped (forced and natural)

Natural circulation can be temporarily interrupted because of reactor coolant contraction after a severe overcooling transient. A long interruption would not be expected since HPI will refill the system and natural circulation will normally restart. Loss of natural circulation would be expected for most LOCA's or for an extended loss of feedwater. Therefore, to restore natural circulation for either of these the failure must be corrected (feedwater restored) and subcooling should be restored (to ensure the best natural circulation). Since loss of natural circulation heat transfer is the most probable of the overheating conditions, the "overheating" corrective actions include restoration of heat transport as an integral part of the action.

Loss of all Feedwater

Overheating when all feedwater is lost can take different paths depending on the decay heat level, when feedwater was lost and whether HPI was operating before it was lost. The P-T diagram illustrated shows a total loss of all feedwater immediately after reactor trip from full power, with HPI cooling started when the operator recognized the loss of primary to secondary heat transfer and loss of all feedwater. Regardless of the path, the loss of all feedwater from power operation will exhibit two clear characteristics on the P-T diagram:

- After the normal post-trip coöldown the RCS will begin to reheat and repressurize beyond the normal post-trip "window" (point 2 on Figure 23) as the SG's boil dry.
- Steam pressure and steam temperature will drop because there is no feedwater.

The corrective actions for this transient are to attempt to restore feedwater; failing to do so, HPI cooling should be started when primary to secondary heat transfer is lost. Two HPI pumps should be started and run at full capacity and the PORV manually opened. Although the subcooling rule requires that HPI be started when the subcooling margin is lost, the expected path for this transient is that high pressure is reached before the subcooling margin is lost. This will result in losing primary inventory out of the PORV for

about 30 or 40 minutes before subcooling margin is lost. This could lead to conditions of core uncovery. By starting HPI when primary to secondary heat transfer is lost core protection is assured.

When HPI cooling (with the PORV open) is started under these circumstances, all but one RC pump should be tripped. This will reduce the heat load. One RC pump should continue to run as long as possible to maintain forced core cooling. When the subcooling margin is lost all the RC pumps must be tripped.

Continued operation without feedwater and with HPI cooling will allow subcooling to be restored when the heat removed by the HPI flow matches the decay heat. When the subcooling margin is restored the HPI flow may be throttled. A reactor coolant pump should also be restarted at this time. Control of HPI flow to keep the minimum subcooling margin is important to minimize reactor vessel thermal shock. Thermal shock occurs when the reactor coolant is subcooled and no circulation exists because the cold HPI water will slowly flow into the downcomer and flow next to the hot reactor vessel wall. Restart of a reactor coolant pump will thus help prevent brittle fracture because it mixes the HPI water and the reactor coolant. (See Figure 25 and the "Best Methods" chapter for a discussion of HPI throttling and RC pump restart.)

LOCA

The other condition in which heat transfer to the steam generators can be interrupted is during a LOCA. LOCA is discussed in great detail in Appendix F and will not be covered here in detail. However, this section will show how LOCA's are to be identified and will show how to locate those that can be isolated.

Although some small breaks will allow the reactor coolant to transfer heat to the steam generators, some will not. The most significant characteristic that shows poor heat transfer is an increase of T_{hot} along the saturation line when a steam generator level exists. T_{hot} is increasing because the reactor coolant is absorbing the core heat and not passing it to the generators. T_{cold} will usually, but not always, follow the same path that T_{hot} does.

Steam pressure and steam generator saturation temperature will gradually drop because little or no heat is being absorbed. Figure 13 of Appendix F (LOCA) shows these P-T characteristics.

Figure 23 indicates that LOCA's can cause poor heat transfer and includes three references for supporting actions:

(1) Table 4a shows how to distinguish LOCA's from other transients.

(2) Table 4b shows symptoms for LOCA's that can be located and shows which equipment to use for isolation for those that can be isolated).

(3) Appendix F shows the corrective actions for LOCA's.

Table 4a 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 5-10 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 AND 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.

Pressurizer Control System Failures

Two failures of the pressurizer controls can occur that can change RC pressure. These are not serious events because they are slow, but if they are left without correction plant control can become more difficult.

Failure of pressurizer heaters (on) with no spray operation will cause the RC pressure to increase to the PORV setpoint at a constant reactor coolant temperature. (If the spray is operating it will stop the heater pressure increase.) Steam will be released to the quench tank until the heaters are turned off. The normal makeup control will continue to add reactor coolant until the letdown storage tank level is lost at which time the pressurizer level will begin to drop. Although this is a very slow transient and should be easy to correct (manual cutoff or power disconnect) if it is left unattended, the following equipment damage can result:

Quench Tank Failure

Makeup Pump Failure on Loss of Suction

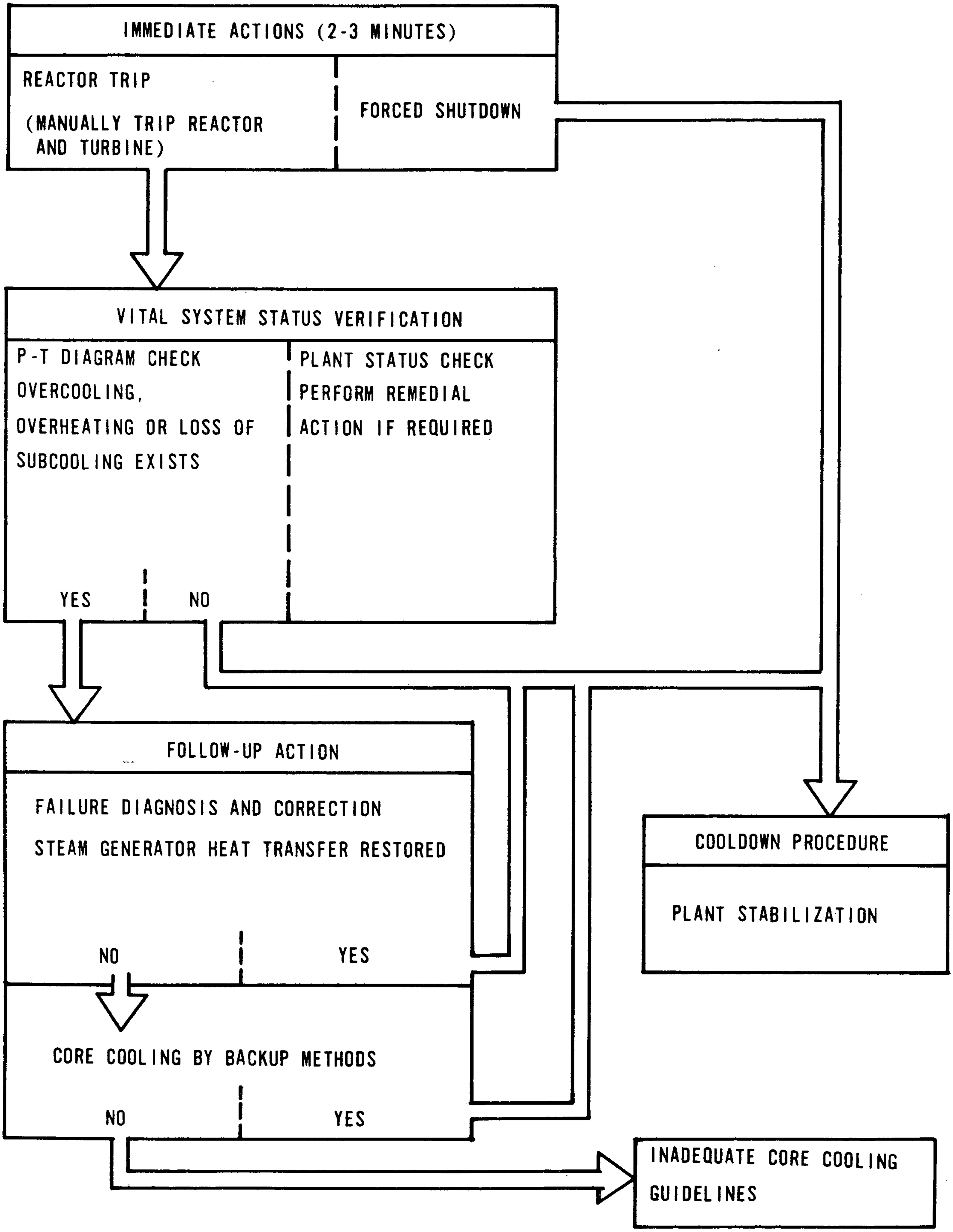
Heater Burnout when they Uncover

A spray failure (on) will cause a pressure decrease at a constant RC temperature until the reactor coolant becomes saturated. This may be corrected by blocking spray flow.

Neither failure is considered serious because there is ample time for correction. Pressure and pressurizer level alarms will sound far in advance of the time when action is required.

Figure 20A

GENERAL PLANT ACCIDENT MITIGATION



PLANT STATUS INDICATOR	OPERATOR ACTION REQUIRED	BASIS FOR ACTION
1. Turbine Trip and/or Reactor Trip	• Verify that all control rods (except APSR's) are on bottom; verify power decreasing. • Manually trip both the reactor and turbine. • If one or more control rods are not fully inserted begin boration (at a later time a stuck rod may be driven in). • Isolate letdown bypass of block orifice (if on high flow bleed cycle at time of trip).	After reactor or turbine trip, the operator should ensure that the fission process is shutdown. The simplest method is to ensure the rods are fully inserted; if not, the operator should manually trip both the turbine and reactor and ensure that a rapid decrease in neutron flux has occurred. Compensation for a stuck rod will have to be by boration to maintain a subcritical margin when the plant is stabilized or plant cooldown is required. When a reactor trip occurs, Tave will decrease due to the loss of core fission power, and an outsurge from the pressurizer will be caused by the contraction of the reactor coolant. The MU control valve will open to increase MU in response to a decrease in pressurizer level. To minimize the potential for a loss of pressurizer level and/or indication, the operator should manually isolate the letdown bypass of block orifice. It is not necessary to isolate "normal" letdown.
2. ES Actuation	• Confirm that the HPI and LPI (<500 psig) are started. • Verify that at least one train in each ECCS is on (pump on). If not, try to start ECCS manually. • Verify, by review of ECCS flow indication, that flow exists in the injection lines. • Confirm containment isolation (for high containment pressure) (non-essential on low RCS pressure). • Confirm containment cooling systems start (on high containment pressure). • Trip RC Pumps on loss of subcooling margin. • If containment isolation has stopped cooling water to the RC Pumps, either reinstate or trip pumps.	When an ES actuation occurs, the operator should assure that at least one train is operative (one pump on) and that flow is present. At this point, HPI/LPI flow balancing is not required, but can be done later.
3. Loss of Offsite Power (LOOP)	• Verify that at least one Keowee generator starts and automatic loading is completed. If the Keowee unit connected to the 13.8 KV buss fails to start, manually transfer the buss to the running Keowee unit.	Upon loss of normal and standby power sources, the two 4160 volt Engineered Safeguard buses are energized, powered by at least one Keowee generator. Bus load shedding, bus transfer to the Keowee generators, and pickup of critical loads is automatic. When a loss of power occurs, the operator should ensure that at least one Keowee generator starts and that loading is completed, and he should try to start the other. (See ATOG Guidelines, Part II, Section 2, "Loss of AC Power," for details about equipment which is automatically loaded and for equipment which must be manually started.)

PLANT STATUS INDICATORS	OPERATOR ACTION REQUIRED	SYMPTOMS	BASIS FOR ACTION
1. Reactor Trip	Immediately following reactor trip examine SG levels for excessive feedwater. ● If SG level is "high" and MFW flow is still on, stop further MFW addition. ● Allow SG level to decrease to appropriate setpoint; then resume FW addition by - manual control of MFW or - EFW addition if MFW has been isolated.	High SG Level High FW Flow	Excessive MFW is the addition of water to the SG at a rate faster than it can be boiled off. It can result in overcooling of the RCS and water spillage into the steam lines must be avoided. Overfill of the SG can occur very rapidly because of the large flow capacity of the MFW system; this is especially true following a reactor trip. Following a reactor trip the operator should assure that MFW runs back. If a FW flow remains high and SG level is increasing, MFW should be controlled (trip pumps). Do not rely on automatic MFW pump trip at high SG level. NOTE: Ensure EFW starts if MFWP's are tripped and throttle EFW to prevent overcooling.
2. Steam Line or Condenser Air Ejector Radiation Alarm Note: Radiation monitors may indicate a tube rupture prior to reactor trip.	Confirm radiation monitor reading supports alarm; start an immediate cooldown and depressurization of the RCS. The cooldown should continue to cold shutdown.		Steam generator tube leaks or ruptures are LOCA's which result in contamination of the main steam system and offsite dose release. To minimize the offsite dose release, a complete cooldown and depressurization of the RCS is required to reduce the primary to secondary leakage and to prevent unnecessary discharge to the atmosphere through the steam generators. Since this is a LOCA, HPI must be kept on if subcooling is lost but this will keep reactor coolant pressure high and continue the leak. Cooldown is required to lower RC pressure to stop the leak.

Figure 20b EXCESSIVE PRIMARY TO SECONDARY HEAT TRANSFER

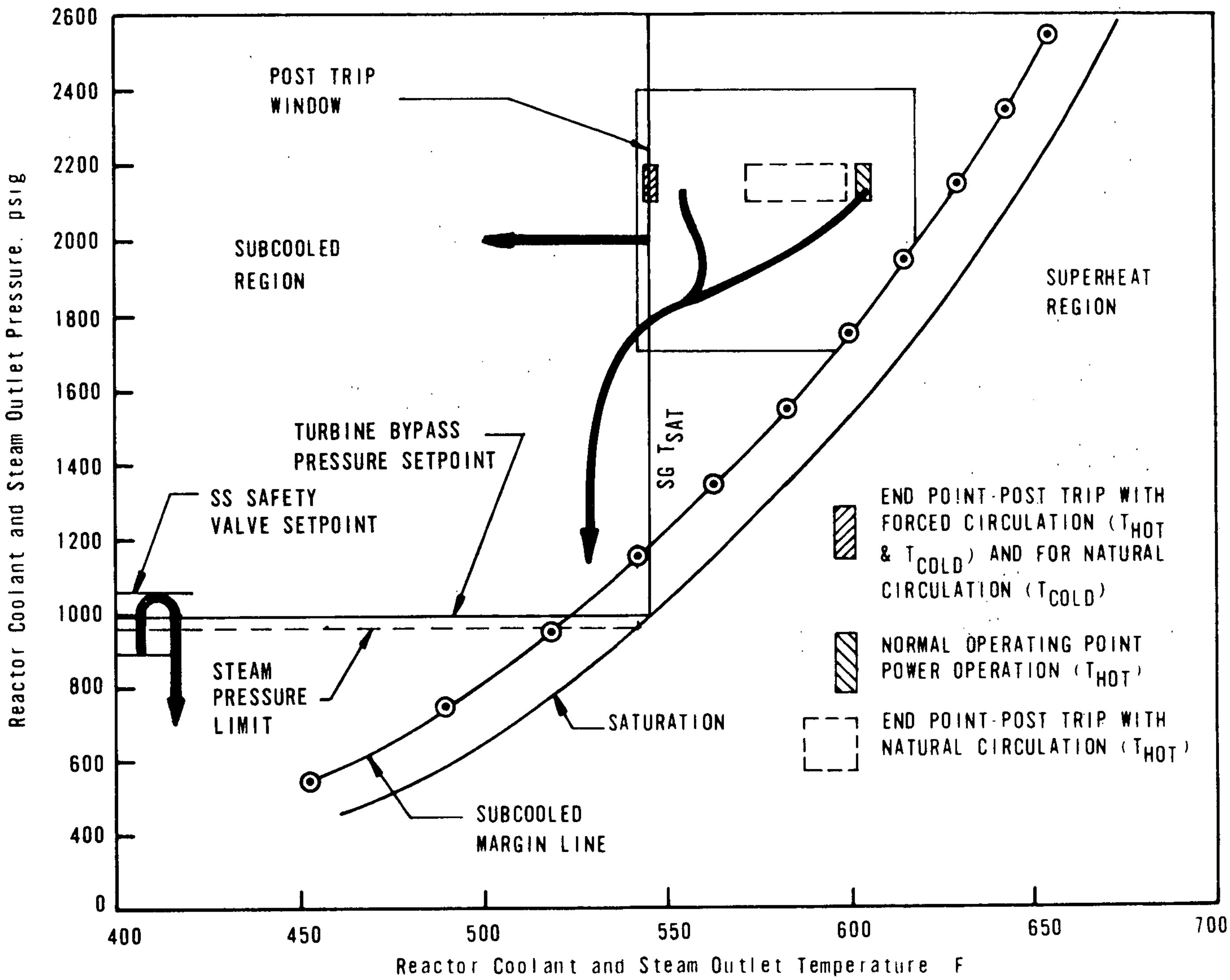


Figure 20c LOSS OF PRIMARY TO SECONDARY HEAT TRANSFER

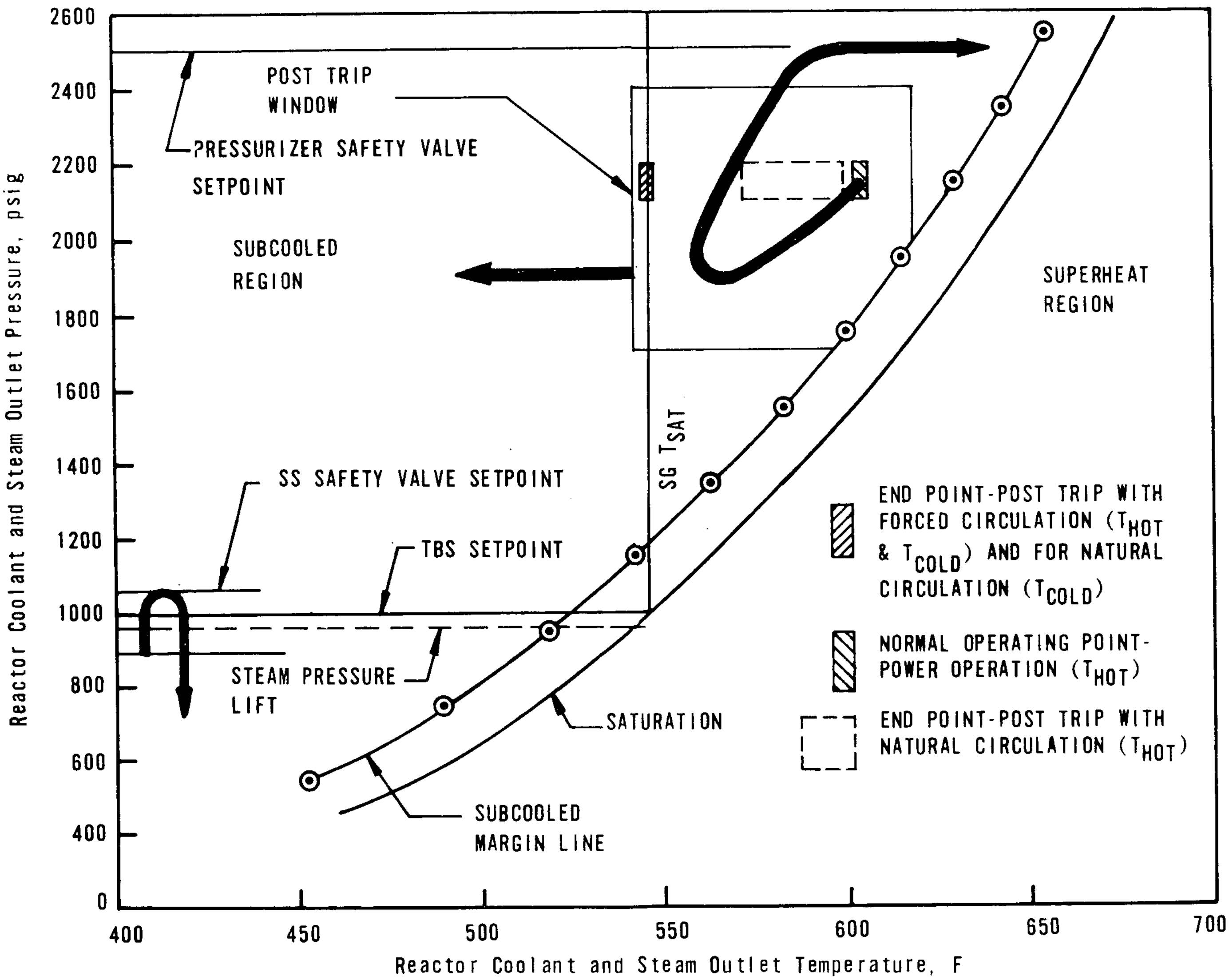


Figure 20d INADEQUATE SUBCOOLING MARGIN

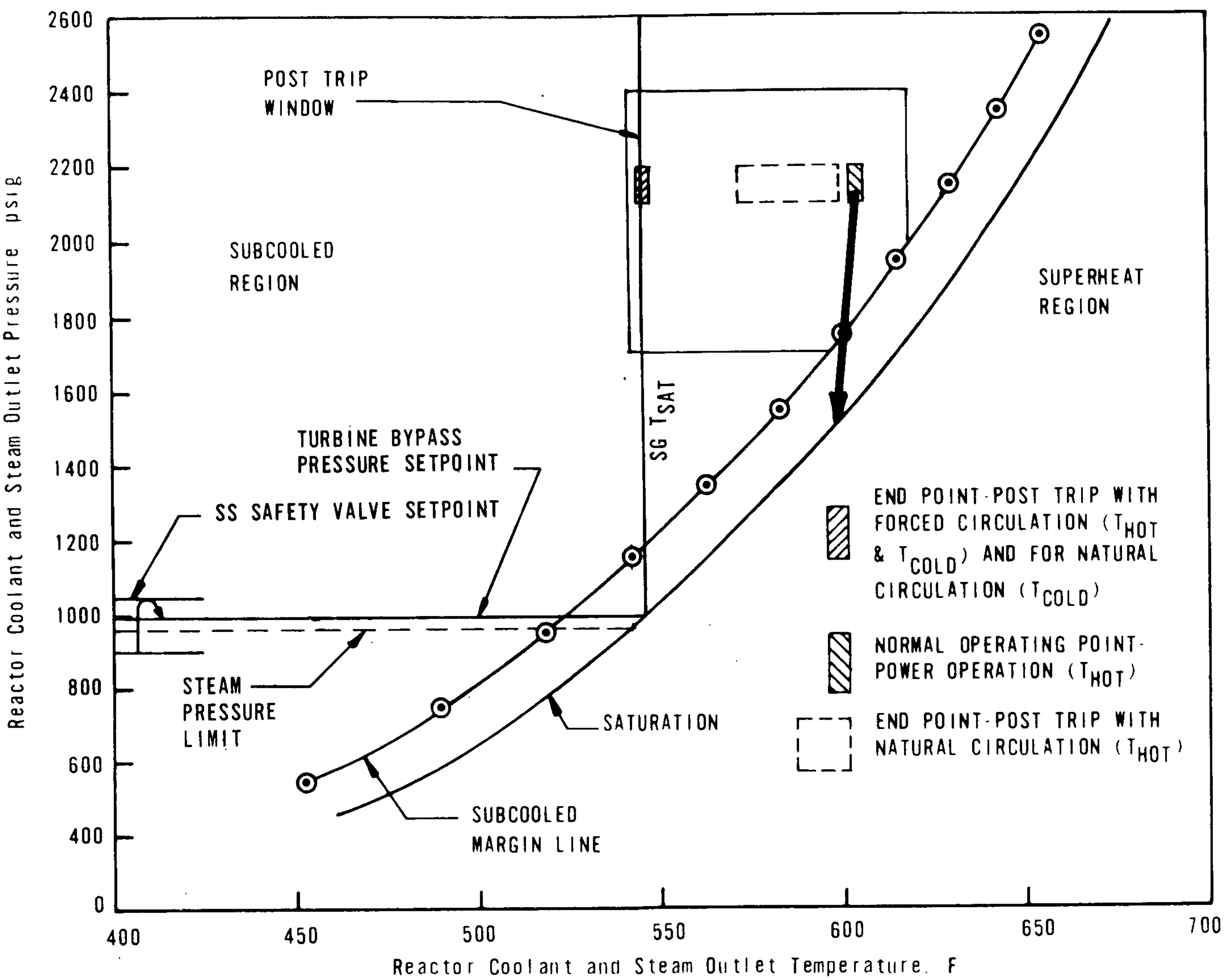


Figure 21 ACCIDENT MITIGATION APPROACH

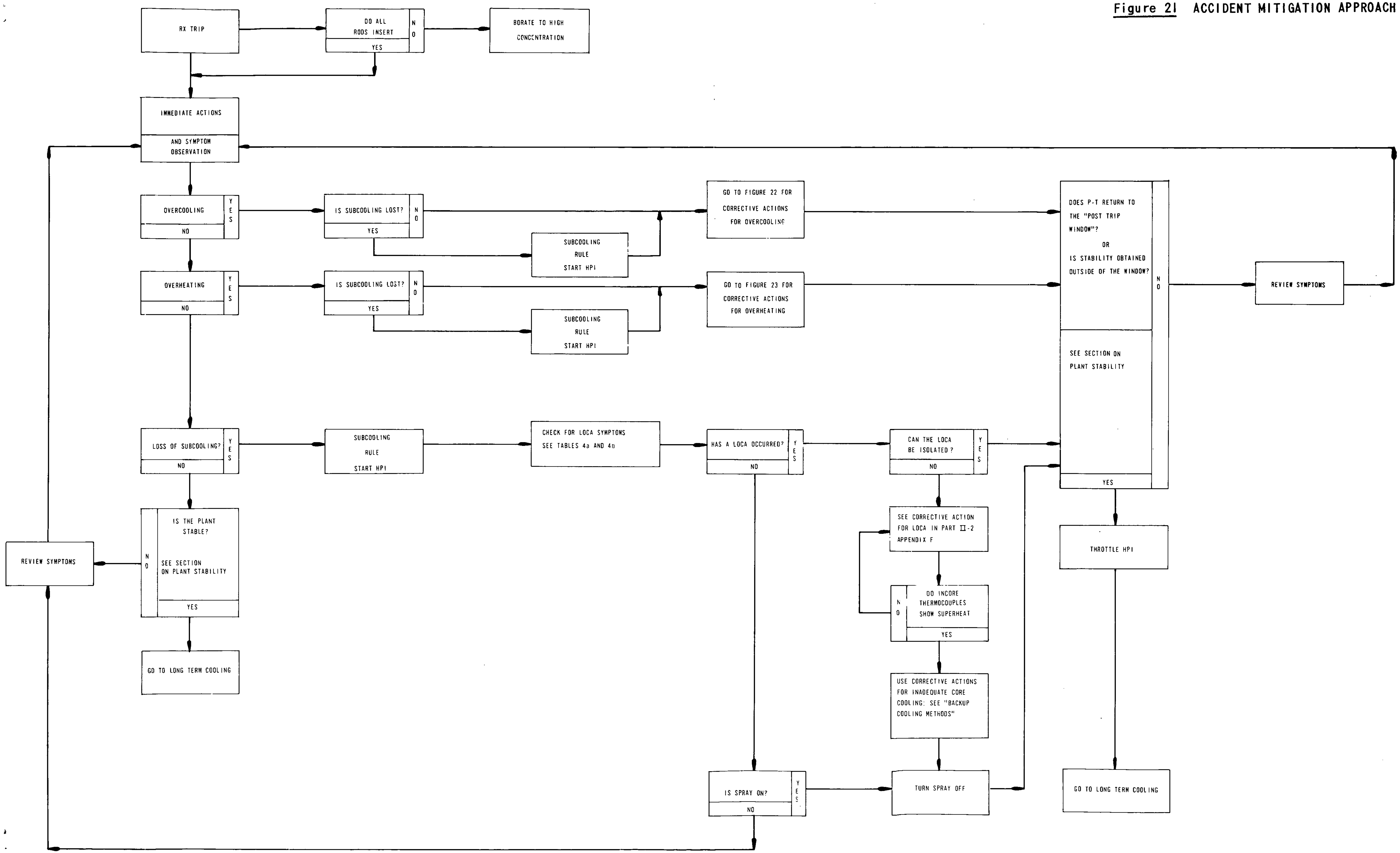


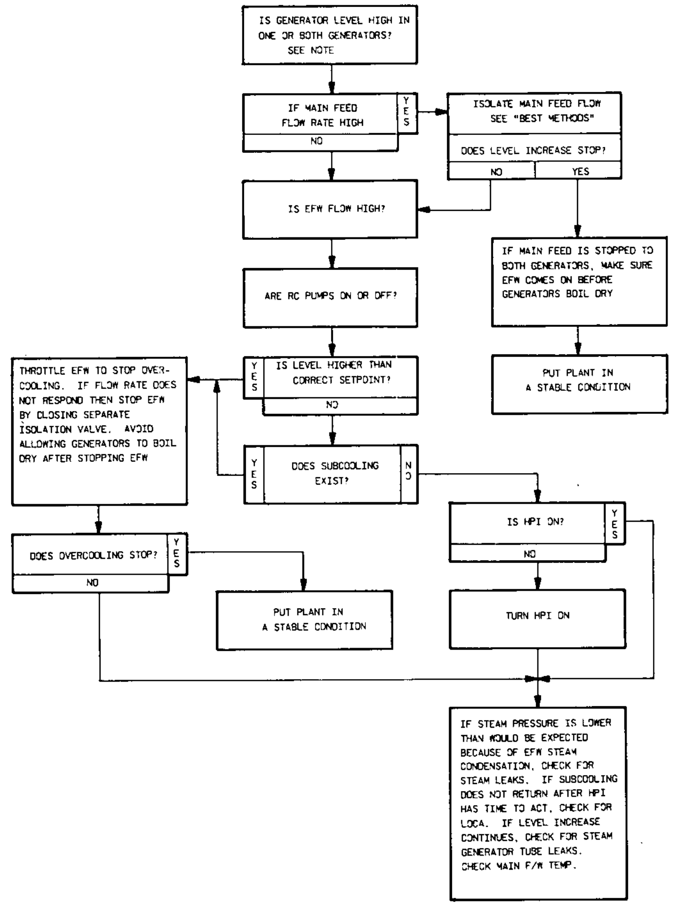
Figure 22 OVERCOOLING DIAGNOSIS CHART

CORRECTIVE ACTIONS FOR LOW MAIN FEEDWATER TEMPERATURE

DURING NORMAL OPERATION LOW MAIN FEEDWATER TEMPERATURE, WHICH OCCURS BECAUSE OF PARTIAL LOSS OF STEAM HEATING TO THE FEEDWATER HEATERS WILL BE HANDLED BY THE ICS. NO SAFETY PROBLEMS ARE EXPECTED. HOWEVER, IF EXCESSIVE FEEDWATER HAPPENS (ESPECIALLY TO BOTH STEAM GENERATORS) THE HEATING SURFACE OF THE GENERATOR WHICH CONVERTS WATER TO STEAM WILL BE COVERED WITH WATER. STEAM PRODUCTION WILL BE LOST. IF THE FEEDWATER HEATING IS LOST AND LOW TEMPERATURE FEEDWATER ENTERS THE GENERATOR, EXTREME OVERCOOLING WILL OCCUR. THE EFFECTS OF THIS MAY BE TO OVER STRESS PORTIONS OF THE STEAM GENERATOR AND REACTOR COMPONENTS.

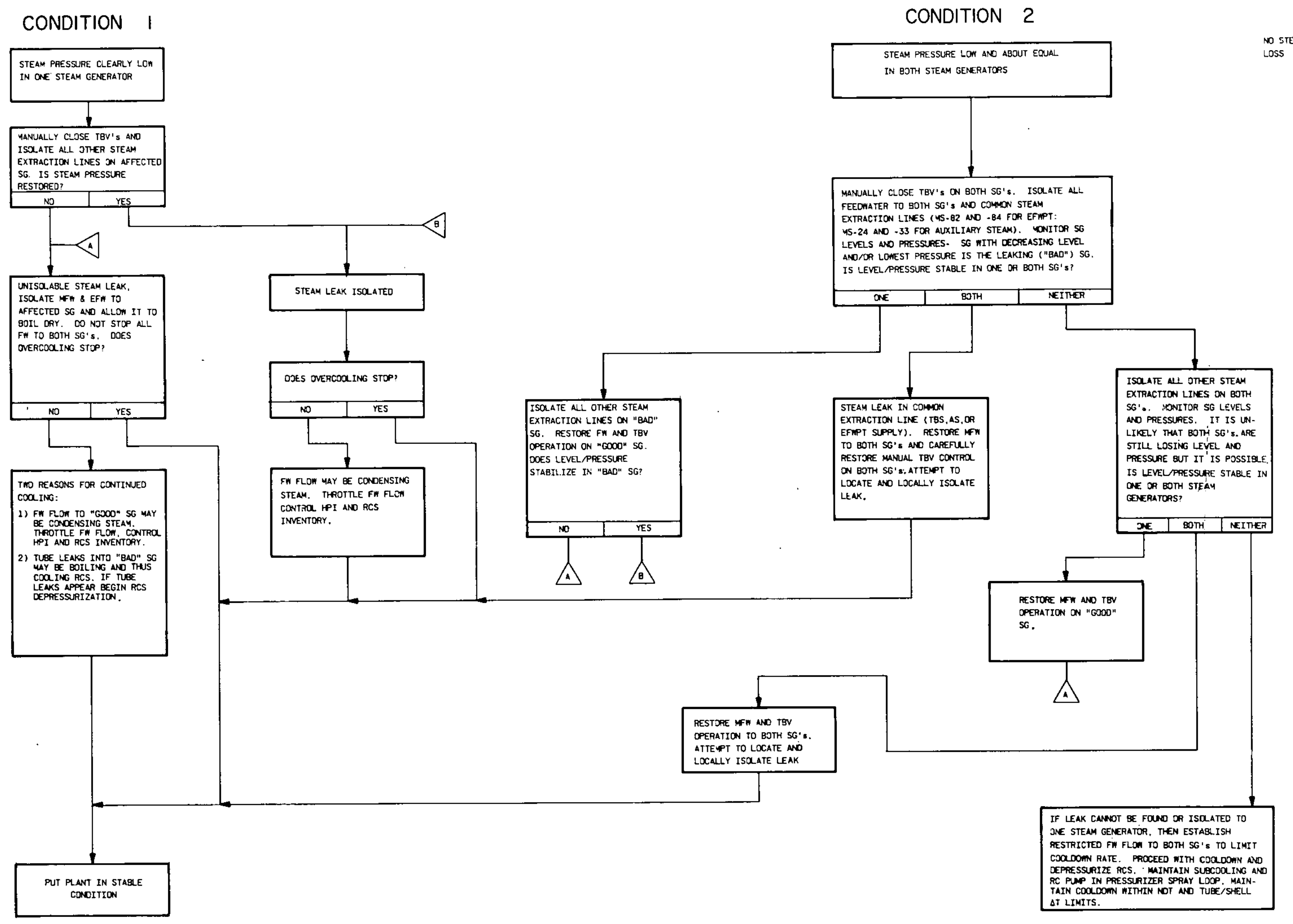
THE ACTIONS TO PREVENT COLD FEEDWATER FROM ENTERING THE GENERATOR ARE THE SAME AS THOSE FOR PREVENTING A HIGH STEAM GENERATOR LEVEL.

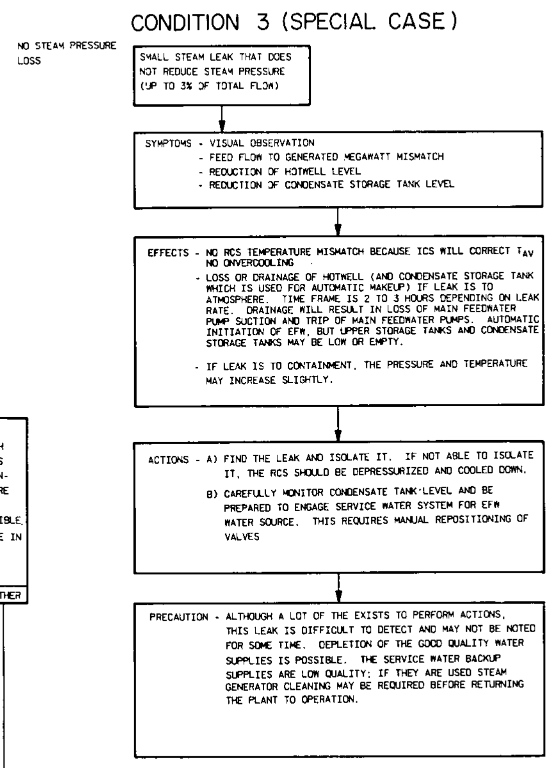
CORRECTIVE ACTIONS FOR HIGH STEAM GENERATOR LEVEL



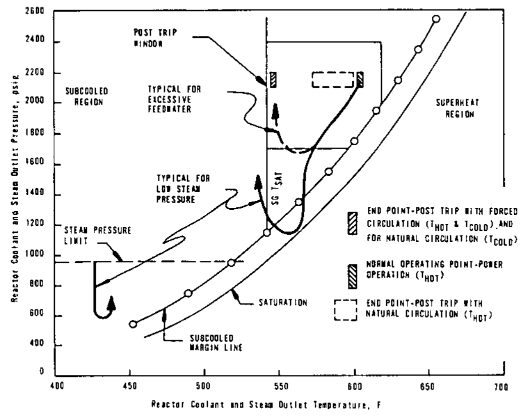
NOTE: BOTH MFW PUMPS SHOULD TRIP AUTOMATICALLY ON HIGH LEVEL IN ONE OR BOTH STEAM GENERATORS. HOWEVER, RUNAWAY MFW CAN BE A VERY RAPID TRANSIENT WITH POTENTIALLY SEVERE CONSEQUENCES. THEREFORE THE OPERATOR SHOULD ACT TO TERMINATE MFW RATHER THAN RELY ON THIS TRIP FEATURE.

CORRECTIVE ACTIONS FOR LOW STEAM PRESSURE





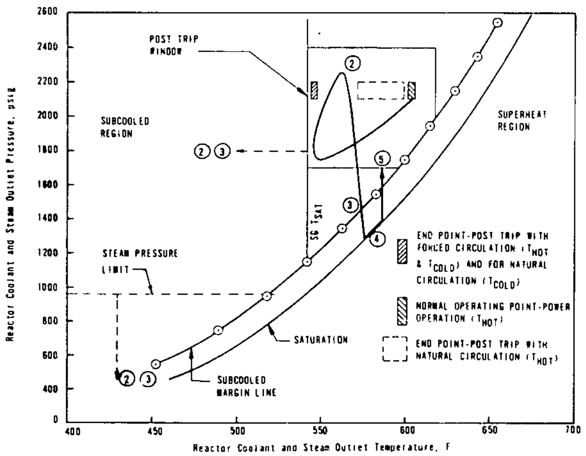
TYPICAL P-T DIAGRAM



THREE GENERAL CAUSES	LOW STEAM GENERATOR PRESSURE	HIGH STEAM GENERATOR LEVEL	LOW MAIN FEEDWATER TEMPERATURE
OBVIOUS SYMPTOMS:	- STEAM OUTLET PRESSURE	- S.G. LEVEL - MAIN F/W FLOW - EFW FLOW	- MAIN F/W TEMP

GENERAL APPROACH:
- TREAT MOST OBVIOUS SYMPTOMS (STEAM PRESSURE, LEVEL, FIRST)
- IF STEAM PRESSURE IS LOW, ISOLATE FEED AND STEAM LINES, IF THAT DOES NOT WORK STOP FEEDWATER
- IF LEVEL IS HIGH, CHECK MAIN AND EFW FEED FLOWS - STOP FEEDING HIGH LEVEL GENERATOR

PRECAUTIONS:
- EXCESSIVE MAIN FEEDWATER MUST BE STOPPED BEFORE WATER ENTERS THE STEAM LINES
- OVERCOOLING BECAUSE OF LOW STEAM PRESSURE CAN CAUSE THE REACTOR COOLANT TEMPERATURE TO BE VERY LOW WHEN THE STEAM GENERATOR SHELL IS STILL HOT; TUBE TENSION STRESSES CAN BE EXCESSIVE; IF STEAM PRESSURE IS BELOW 350-400 PSI IN ONE OR BOTH GENERATORS, THE TUBES CAN STRETCH AND MAYBE FAIL.
- OVERCOOLING CAN CAUSE TEMPORARY LOSS OF SUBCOOLING; HPI SHOULD BE STARTED; WHEN SUBCOOLING RETURNS HPI SHOULD BE THROTTLED TO PREVENT FILLING PRESSURIZER SOLID AND FORCING WATER THROUGH SAFETIES
- STEAM LEAKS TO CONTAINMENT CAN (WITH TIME) CAUSE CONTAINMENT OVERPRESSURE; MAKE SURE CONTAINMENT COOLERS ARE WORKING; IF POSSIBLE AVOID CONTAINMENT SPRAYS; IF NO RADIATION EXISTS VENT THE CONTAINMENT.



OVERHEATING OCCURS WHEN THE REACTOR COOLANT CANNOT TRANSPORT THE CORE HEAT TO THE STEAM GENERATORS FOR HEAT REMOVAL. NATURAL CIRCULATION WILL NORMALLY BE LOST FOR AN EXTENDED TIME (VS BRIEFLY INTERRUPTED). T_{HOT} WILL BE SATURATED (T_{COLD} WILL USUALLY ALSO BE SATURATED). SINCE THE STEAM GENERATOR CANNOT REMOVE HEAT, STEAM PRESSURE AND T_{SAT} SG WILL DECREASE. GENERALLY ONLY TWO CONDITIONS WILL PERMIT OVERHEATING; LOCA's AND LOSS OF ALL FEEDWATER.

① ATTEMPT TO RESTORE MFW OR EFW

② START HPI COOLING WHEN PRIMARY TO SECONDARY HEAT TRANSFER IS LOST

STOP ALL BUT ONE RC PUMP TO REDUCE THE REACTOR COOLANT HEAT LOAD

③ TRIP ALL RC PUMPS ON LOSS OF SUBCOOLING MARGIN

OTHER REFERENCES:
SEE "BEST METHODS FOR EQUIPMENT OPERATION" FOR HPI COOLING AND RC PUMP TRIP

④ SATURATED CONDITIONS WILL EXIST UNTIL FEEDWATER IS RESTORED OR HPI FLOW MATCHES DECAY HEAT (HOURS)

⑤ SUBCOOLING MARGIN RESTORED BY HPI COOLING, HPI MUST BE THROTTLED TO PREVENT NDT PROBLEMS

ONE RC PUMP SHOULD BE RESTARTED WHEN SUBCOOLING MARGIN IS REGAINED. A MOMENTARY LOSS OF SUBCOOLING MARGIN MAY OCCUR ON RESTART

OTHER FEFERENCES:
SEE "BEST METHODS FOR EQUIPMENT OPERATION" FOR HPI THROTTLING AND RC PUMP RESTART

SEE FIGURE F-13, APPENDIX F, "LOCA" IN PART II-2 "DISCUSSION OF SELECTED TRANSIENTS"

FOR:
A - P-T DIAGRAM CHARACTERISTICS
B - CORRECTIVE ACTIONS

SEE TABLE 4A "HOW TO DIFFERENTIATE A LOCA FROM OTHER TRANSIENTS"

SEE TABLE 4B "SYMPTOMS FOR LOCA's THAT CAN BE LOCATED OR ISOLATED"

Table 4B SYMPTOMS FOR LOCAs 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	- Low letdown storage tank level - High component cooling water surge tank level (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	- Low letdown storage tank level - High RCW radiation - High RCW surge tank level (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	- Flow Monitor Alarm - High quench tank level - High quench tank temperature (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)	- High letdown storage tank level - 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)	- High or low decay heat removal flow - Low pump suction press. - Local sump and local radiation alarms	Decay heat letdown** drop line valve
Decay heat cooler tube leak (decay heat removal sys. in operation- plant is cooled down)	- High LPSW temperature at DH cooler outlet.	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)	- High steam line radiation - High steam generator level - High condenser radiation
Pressurizer Safety Valves	- Flow Monitor Alarm - High quench tank level - High quench tank temperature (These will only be good while the quench tank rupture disk is good)
HPI Injection Line Break	- Flow imbalance between injection**3) lines (High flow will be through broken line)
RC Pump Seal Failure	- High seal return temperature (~350°F) combined with: Low stage and upper stage pressures are equal and high
RCS Instrumentation Lines - Pressurizer Level - Pressures - RC Flow	- False low level reading - False low pressure - False high or low flow compared with known pump operation

**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.

CHAPTER D

BACKUP COOLING METHODS

The normal method of core cooling is by transferring heat from the core to the steam generators using the reactor coolant. This is the preferred method and is the one the operator is most familiar with. However, if heat transfer via the steam generators is lost other means of core cooling are available. This section will discuss these backup cooling methods and explain their uses and limitations. Three instances will be covered:

1. Loss of All Feedwater
2. Restoration of Natural Circulation
3. Inadequate Core Cooling

These methods will not be necessary if the normal method of cooling using the steam generators is working.

Backup Cooling by HPI for Loss of All Feedwater

A complete loss of feedwater is not a likely event, but it can occur because of multiple equipment failures or operator error. If primary to secondary heat removal is lost because of the loss of feedwater, the core energy is removed by the reactor coolant (HPI) and released to the reactor building, which serves as the heat sink instead of the steam generator. The core is kept covered and cooled by HPI.

A total loss of all feedwater is illustrated in Figure 24a and is discussed below:

1. With the plant at power, a loss of main feedwater would result in a reactor trip (anticipatory trip on loss of MFW or RPS actuation on high RCS pressure). A loss of all FW could also occur during hot shutdown or plant heatup/cooldown.

2. If emergency feedwater also fails, the secondary side of the steam generators will boil dry and the RCS will then heat up due to decay heat.

3. Subcooling will decrease and the reactor coolant will expand into the pressurizer.

   NOTE: The rate at which the above occurs will depend upon the initial inventory in the steam generators and the core decay heat level. For example, the RCS heatup rate may be as high as 4F/min with high decay heat or as low as 1F/min with low decay heat following boil off of the SG inventory.

4. RC pressure will increase as the steam space in the pressurizer is compressed due to the insurge of reactor coolant and steam/water relief out of the PORV or the pressurizer safety valves will occur.

5. The reactor coolant will eventually saturate and boiling will occur throughout the core.

6. Without corrective action, the reactor coolant will slowly be vaporized to steam and relieved to the containment and core damage will result.

To avoid these consequences the operator should make every attempt to regain feedwater to at least one steam generator. This includes main feedwater, emergency feedwater, condensate and booster pumps (if steam generator pressure is low enough) or service water. If feedwater cannot be regained after primary to secondary heat transfer is lost, he should manually start two HPI pumps and open and leave open the PORV. HPI flow should be balanced to give the maximum flow possible.

The event can be recognized by observation of generator level and equipment status checks or through use of the P-T diagram. Figure 24b illustrates the P-T response of the RCs after a loss of main feedwater from 100% power with no EFW and appropriate operator action.

When HPI is started the RCS will eventually go to a water solid condition (subcooled) with RC pressure controlled by a combination of the HPI pump head rise and the relief capability of the PORV. The number of running reactor coolant pumps should be reduced to one to reduce the heat load. The time to become water solid depends on the core decay heat level and the number of HPI pumps. If core decay heat is high, the reactor coolant will saturate and its pressure will rise along the saturation curve (points 5-6 and Figure 24b). The running RCP should be tripped when subcooling margin is lost. The water inventory in the RCS will drop because of core boiling and relief out the PORV, until the amount of water added by HPI can take away all the decay heat of the core. Until that time water from the RCS and HPI water together are needed to take out core heat. When the heat removal capacity of the HPI water can take out all the decay heat, HPI is said to "match" decay heat. When HPI matches decay heat the liquid inventory in the RCS will stabilize. As HPI heat removal capacity begins to exceed decay heat generation, RCS inventory will start to increase, steam will be condensed and the system will slowly go to a water-solid (subcooled) state (points 6-7 on Figure 24b). If the core decay heat level is low at the time feedwater is lost, HPI may be enough to take out all the heat and the RCS may remain subcooled (water solid) until feeedwater is restored.

When all feedwater is lost it is very important that two HPI pumps be run until subcooling exists. One HPI pump is adequate for core heat removal, but the time required for one HPI pump to match decay heat is much longer than two pumps. The time to match 100% decay heat level with one HPI pump is approximately 70 minutes and approximately 8 minutes with two HPI pumps.

When subcooled conditions (based on both incore thermocouples and hot leg RTD's) are reached, the operator should start one reactor coolant pump and throttle HPI flow to maintain the reactor coolant subcooled but within equipment design limits such as the brittle fracture limit of Figure 25, "RCS Pressure/Temperature Limits". If a reactor coolant pump cannot be restarted the operator should still throttle HPI to maintain an adequate subcooling margin and avoid NDT problems with the reactor vessel. The intent is to run an RC pump to improve thermal mixing and minimize thermal shock to the reactor vessel. However, if the pump is not ready in all respects to be run it should not be run. Core protection is assured by the HPI cooling.

In summary, HPI cooling will maintain core cooling if secondary heat removal is lost. It is not a normal operating mode for many reasons; three examples are:

1. High RC pressure may occur causing the pressurizer relief and safety valves to be cycled with water and two phase flow discharge. This increases the potential for a LOCA that cannot be isolated. (Failure of code relief valve).

2. Long-term operation (subcooled solid water operation) must be closely monitored to prevent exceeding equipment design limits (NDT and RV Brittle Fracture).

3. The degraded containment environment may cause failure or bad readings of instrumentation.

Consequently, secondary cooling should be restored as quickly as possible so that normal primary to secondary heat transfer can be resumed.

Restoration of Natural Circulation

When RC pumps are off, heat is removed from the reactor core by natural circulation as discussed in Addendum B. Some abnormal transients can lead to a loss of natural circulation but methods exist to restore it if it is lost. The intent of this section is to highlight the recovery measures and to give an understanding of why certain actions are recommended and when they are to be taken. Brief discussions of other issues on natural circulation are also provided.

A loss of natural circulation can occur for two reasons, which are:

Reason 1. Insufficient secondary inventory control (i.e., not enough feedwater)

Reason 2. Formation of steam voids within the hot leg which are of sufficient volume to block water flow to the steam generator (i.e., not enough reactor coolant).

The previous section addressed system operation when natural circulation is lost due to insufficient feedwater (Reason 1). The only way to recover natural circulation under that condition is to restore feedwater. Void formation (Reason 2) is more complicated because the reactor coolant system can operate differently depending on what

has happened. The two principle events which lead to void formation are overcooling transients and loss of coolant accidents. For these transients, voids are formed in the following manner:

Overcooling: Too much primary to secondary heat transfer causes a drop of RCS temperature which causes a contraction of fluid inventory, a decrease in reactor coolant pressure, and a loss of pressurizer liquid. Some of the steam in the pressurizer flows into the RC piping and collects in the hot legs. Because the RC pressure drops, some of the reactor coolant may flash and cause void formation in the hot legs.

LOCA: A LOCA results in a loss of RC inventory and a reduced RC pressure. Voids are formed directly as a result of loss of RC inventory and also because of flashing of the reactor coolant as RC pressure drops. The RC temperature does not drop as much as it would for an overcooling event.

Figure 26 illustrates the buildup of steam voids and the formation of a steam bubble in the upper hot leg piping.

The size of the steam bubble will depend on the rate of system overcooling or loss of inventory versus the rate at which HPI adds water

to the RCS to refill it. If HPI flow is large compared to the other things no steam bubble will form at all and natural circulation will not be lost.

If a steam bubble does form its size has a direct effect on primary to secondary heat transfer. If the bubble is big enough such that the hot leg level is at or below the secondary side feedwater level then steam can be condensed within the steam generator tubes and the steam generators can still remove a large amount of decay heat. This is a boiling mode of natural circulation (boiler-condenser cooling) and is illustrated in Figure 27. Boiler-condenser cooling is an expected small break LOCA condition. If the steam bubble is smaller and steam cannot be condensed in the steam generator tubes (see Figure 28) then primary to secondary heat transfer will be much lower. In this condition the RCS may heat up and might repressurize. Several examples of transient conditions that could get into this mode of operation are:

1. Small LOCA's where HPI can match the leak rate (at reduced RCS pressures) and refill the RCS.
2. A severe overcooling event (e.g., major steam line break) in combination with delayed actuation of HPI.
3. A total loss of feedwater, where EFW is restarted after the RCS is in a highly voided condition and the HPI is refilling the RCS.

Figure 27 shows boiler-condenser cooling (boiling in the reactor vessel and condensing in the steam generator tubes) with a saturated hot leg and RCS pressure near the steam generator pressure. For this condition, it is important to ensure that 1) SG level is at 95% on the operating range (to allow the condensed reactor coolant to flow over the cold leg pump elevation and into the core) and, 2) HPI is on at a high capacity (two pumps). A check of containment pressure and temperature conditions should also be made to see if the cause is a LOCA. If LOCA conditions are indicated, an immediate plant cooldown at design rates (100F/hr) should be inititated. The P-T diagram should be monitored to see if subcooled natural circulation returns or boiler-condenser cooling is lost.

If natural circulation has been lost and steam cannot be condensed in the steam generators (that is, the steam bubble is in the top of the candy cane but not low enough to be in the steam generator tube region) the RCS will repressurize. This mode of operation will be indicated (see Figure 28) by saturated hot leg conditions with Reactor Coolant pressure above the steam generator pressure (SG pressure may be dropping due to lack of primary to secondary heat transfer). As indicated in Figure 28 the same actions identified for boiler-condenser cooling apply to this operating mode. In this mode, the HPI is refilling the RCS. During refill, steam in the upper region of the hot leg piping will be compressed and/or condensed as the water level in the loops and steam generator rises. In some

cases (i.e., low decay heat with all HPI pumps on) subcooling and natural circulation will occur with minor increases in RC pressure. Under other circumstances, it may be difficult to fully condense the steam in the hot leg and restore natural circulation. Figure 28 shows the actions to take to restart natural circulation if the steam generator can be used as a heat sink and the RC pumps are available for restart (see the RC pump restart guidelines in the "Best Methods for Equipment Operation" section). If the RC pumps are available, pump bumps (short run times of 10 seconds duration) are allowed. This momentary use of forced circulation tries to force reactor coolant steam condensation by mixing it with liquid reactor coolant and by moving the steam into the generator tubes where it can condense. Use of the PORV to limit RCS pressure rise and to increase HPI flow is also allowed (separate or in conjunction with RCP operation). To be effective the steam generator must be a heat sink; the steam generator saturation temperature selected is somewhat arbitrary; it was chosen to ensure a strong temperature gradient for condensation. When the pumps are bumped and steam is condensed the RCS pressure will drop a much as several hundred psi. HPI flow will increase to help refill of the voids. If natural circulation starts the RC pressure will stay low; if natural circulation does not start the RCS will repressurize and another bump can be used about 15 minutes later (see the pump restart guidelines).

Finally, a LOCA of a certain size could depressurize the RCS below steam generator pressure before it settles to an equilibrium with HPI (HPI will automatically start because this size LOCA will drop pressure below the ES setpoint). If this happens the operator should lower the steam generator pressure (using the TBS) until the saturation temperature on the secondary side is 50F below the primary saturation temperature. This will ensure the steam generators are heat sinks. Other actions are the same as discussed above for boiler-condenser cooling (Figure 27).

Inadequate Core Cooling (ICC)

The first objective of the operator during any abnormal transient is to keep the core cooled. As discussed in Addendum A, core cooling is taking place whenever the reactor coolant is in a subcooled or saturated state and the core is covered. If the reactor coolant becomes superheated, the core has been uncovered and is not being adequately cooled; that is, decay heat is not being removed fast enough and the temperature of the fuel and cladding are increasing. This, in turn, causes the reactor coolant to heat up, flash to steam, and become superheated.

Inadequate core cooling is not expected. However, any transient can become an inadequate core cooling event if enough equipment failures happen. These events have a low probability of occurrence. Some examples where ICC conditions could develop are:

1. Small LOCA with a total failure of the HPI system.

2. Total loss of feedwater (both main and EFW) with a total failure of the HPI system.

3. A total loss of power (including a failure of both Keowee generators to start) with a failure of the steam turbine-driven EFW pump to run.

4. During a small break, tripping the RC pumps at a time period when the RC void fraction is about 70% or greater.

The intent of the Inadequate Core Cooling (ICC) guidelines is:

1. To allow the operator to identify when core cooling is inadequate.

2. To provide the operator with a way to estimate the severity of the accident.

3. To identify those systems which are vital so that the operator's attention will be focused on these items in his attempts to re-establish core cooling.

4. To identify some known alternative actions to try to correct or minimize the consequences of the accident until normal cooling can be re-established. These actions are based on the severity of the accident.

ICC is indicated when the reactor coolant pressure and temperature (T_{hot} RTD or incore thermocouples) enter the superheat region. This condition can occur with and without forced circulation. If the RC pumps are operating superheated conditions imply that the reactor coolant is nearly all steam (see Figure 24a-Time IV). That is, the

liquid in the RCS has been lost, due to a leak in the primary system or boiled off out the safety valve (assuming the PORV is isolated) by decay heat, and the steam mass left within the system is not enough to remove core heat even though it is circulated by the RC pumps. When the RC pumps are off, core cooling is accomplished by keeping the core covered with a steam-water mixture. If not enough cooling water (HPI) is supplied to make up for losses, the core will become uncovered and the core exit fluid temperature will become superheated (see Figure 24a - Time IV.)

Superheated temperatures, as indicated by the core exit thermocouples, are ICC symptoms. (NOTE: Incore Thermocouples are the only valid temperature measurement when the RC is not circulating). These indicators can also be used to estimate how bad the situation is. Analyses have been performed which show the relationship between core exit steam temperature and fuel cladding temperature for various RC pressures (see Figure 29). This figure gives the following information:

1. When the RCS P&T conditions are superheated but to the left of curve 1 on Figure 29, and ICC condition exists but it is not bad enough to cause core damage.

2. If the RCS P&T conditions reach or exceed Curve 1 on Figure 29, the cladding temperature in the high power regions of the core may be 1400F or higher. Above this temperature, there is a chance for rupture of the fuel rod cladding material. A chemical reaction between the cladding and the water at high temperatures also occurs and will add heat to the fuel rods increasing the chances of fuel failure. The clad-water reaction also causes free hydrogen formation which collects in the reactor loops and may escape to the building. The accumulation of hydrogen in the RCS can also block natural circulation when water is added to the RCS.

3. If RCS P&T conditions reach or exceed Curve 2 of Figure 29, the cladding temperatures in the high power regions of the core may be 1800F or higher. This is a very serious condition. At this level of ICC, significant amounts of hydrogen are being formed, and core damage may be unavoidable. Extreme measures are warranted to prevent major core damage.

If an ICC condition develops, the operators should try to get equipment working to supply water to the reactor and/or steam generator. The general strategy during ICC as follows:

1. To check vital equipment

   - All available HPI on with flow into RCS. For a total loss of feedwater, the HPI must be manually started. ES will not be actuated automatically since RCS pressure does not decrease.
   - FW to at least one steam generator with level at 95% on operating range level instrumentation.

2. Start any backup equipment to correct for problems found in vital equipment check.

   - Start standby MU/HPI pumps.
   - Take suction from any available borated water source.
   - Start MFW if EFW is not operating.
   - Start any backup pumps which can supply water to the steam generator if MFW and EFW are not operating.

3. Minimize the consequences of the event if conditions degrade.

   - Start an RC pump to circulate primary system fluid (water or steam) through core. This action will make available water trapped in the lower region of the reactor vessel and the loops for core cooling (see Figure 24a - TIME IV) and provide improved heat transfer due to forced convection which will provide additional time to restore emergency injection.

- Attempt to decrease RC pressure by opening the PORV and high point vents in order to increase the rate of available high pressure injection.
- If secondary cooling is available, decrease primary pressure by decreasing SG pressure. This action is directed at making the core flood tanks and LPI system available to restore core cooling.

In general, the ICC strategy depends on operator action to locate and correct the cause of low RCS inventory or to take alternate actions to make backup sources of cooling water available. Some of the actions identified above can be detrimental to major components, and others carry a certain amount of risk, but keeping the core cooled is the first priority. For example, an emergency cooldown/depressurization of the system may impose high thermal stresses on the SG internals; this action can be shown to be acceptable but it challenges the design to its limit. A second example would be the restart of one of more RC pumps. This action carries some risk because later on a pump trip may leave the RCS with less water than before. These risks are small compared to those which could happen with extensive core damage. Because the severity of the ICC condition can be estimated (by using Figure 29), the appropriate actions have been picked so that the risk of the action is small compared to the core consequences if the action is not taken. These actions are

outlined below and are based on where the RC pressure-incore thermocouple temperature (P-T/C) combination corresponds to the curves of Figure 29.

If the P-T/C combination is between the saturation curve and Curve 1 superheated conditions exist and the operator should:

1. Verify emergency cooling water is being injected through all HPI nozzles into the RCS,
2. Initiate any additional sources of cooling water available such as the standby makeup pump,
3. Verify the steam generator level is being maintained at the emergency level,
4. If steam generator level is not at 95% of operating range, raise level to the 95% level,
5. If the desired steam generator level cannot be achieved, actuate any additional available sources of feedwater.
6. Establish 100F/hr. cooldown of RCS via steam generator pressure control until secondary steam saturation temperature is 100F below the incore thermocouple temperature.
7 Open core flooding line isolation valves if previously isolated.
8. If RC pressure increases to 2300 psig, open the pressurizer PORV to reduce RC pressure and reclose PORV when RC pressure falls to 100 psi above the secondary pressure.

These actions are directed toward depressurization of the RCS to a pressure at which the ECCS water input exceeds core steam generation. The alignment of other sources of cooling water is the recognition that the injection of the HPI system alone is not sufficient to exceed core boil off.

If the P-T/C combination is between Curve 1 and Curve 2 of Figure 29, the operator should do the following:

1. Start one RC pump in each loop; do not defeat RC pump interlocks.
2. Depressurize the steam generator as rapidly as possible to 400 psig or as far as necessary to achieve a 100F decrease in secondary saturation temperature, but not below the steam pressure necessary for EFW pump turbine to deliver EFW.
3. Immediately continue the plant cooldown by maintaining a 100/hr cooldown rate until the secondary saturation temperature is low enough to achieve a 150 psig RC pressure.
4. Open the power operated relief valve (PORV), as necessary, to relieve RCS pressure and vent non-condensible gases.
5. Open all high point vents to aid RCS depressurization and refill until the RCS is 50F subcooled or RCS pressure decreases to 150 psig.

The operator action in starting the RC pumps will provide forced flow core cooling and will reduce the fuel cladding temperatures.

The rapid depressurization of the steam pressures will help to depressurize the primary system to the point where the core flood tanks will actuate. Stopping the depressurization at 400 psig (or as far as necsessary to achieve a reduction in secondary saturation temperature of 100F) will maintain the OTSG tube to shell temperature difference within the design limit. The continued cooldown to 150 psig will reduce the primary system pressure to the point where the Low Pressure Injection System can supply cooling. The opening of the PORV and high point vents will also help to depressurize the primary system. The PORV should be closed when the primary pressure is within 50 psi of the secondary pressure and then should only be used as necessary to maintain the primary system pressure at no greater than 50 psi above the secondary system pressure. This method of operation will minimize the loss of water from the primary system through the PORV. The high point vents should not be closed until the RCS is 50F subcooled or RCS pressure decreases to 150 psig.

If the P-T/C combination is to the right of Curve 2 of Figure 29, the operator should:

1. Depressurize the steam generators as rapidly as possible down to low pressure while ensuring sufficient steam pressure remains in the steam generators to operate the turbine drive EFW pump. (If the turbine driven EFW pump is being supplied with steam from another unit or the auxiliary boiler, then depressurize the steam generators to as low a pressure as possible.)

2. Start the remaining RC pumps. Defeat starting interlocks; do not defeat the overload trip circuit.

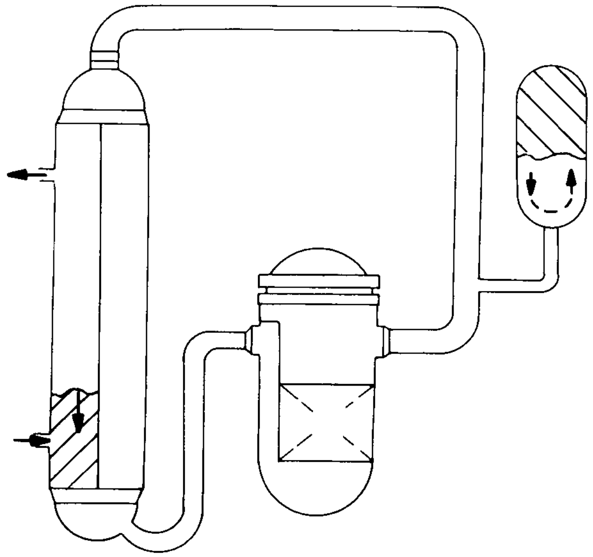
3. Open the PORV and leave it open.

The goal of these actions is to depressurize the RCS to a level where the core flooding tanks will fully discharge and the LPI system can be actuated, thus providing prompt core recovery.

After reaching Curve 2, significant core damage may have occurred which will add significant radioactive contaminants to the reactor coolant.

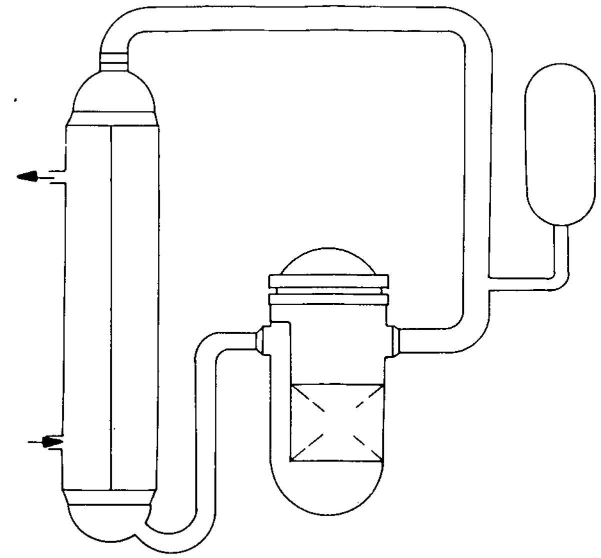
Special cooldown precautions need to be followed to contain these contaminants. These include isolating the DH rooms.

Figure 24a BACKUP COOLING BY HPI FOR LOSS OF ALL FEEDWATER (NO OPERATOR ACTION)



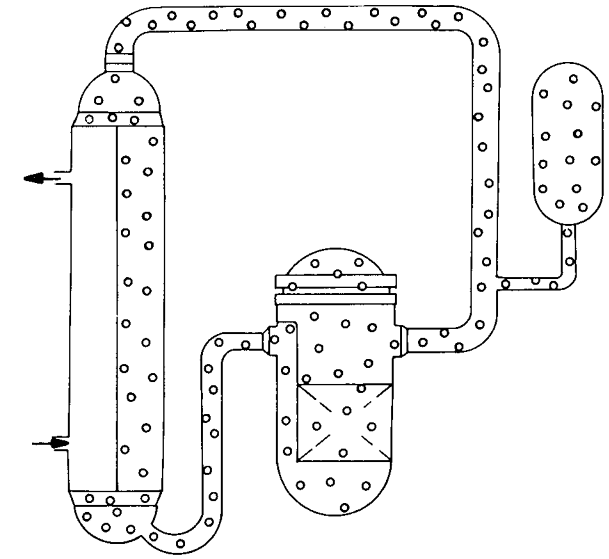
TIME I

1. LOSS OF FEEDWATER
2. REACTOR TRIP ON HIGH RCS PRESSURE (OR ANTICIPATORY TRIP).
3. PRESSURIZER LEVEL DECREASES (NORMAL RESPONSE FOLLOWING A REACTOR TRIP) THEN INCREASES BECAUSE OF MU ADDITION AND REHEAT OF REACTOR COOLANT.
4. SECONDARY SIDE BOILS DRY.
5. EFW DOES NOT START.
6. RC PRESSURE SHOWS NORMAL POST TRIP RESPONSE THEN INCREASES AS PRESSURIZER LEVEL IS RESTORED.



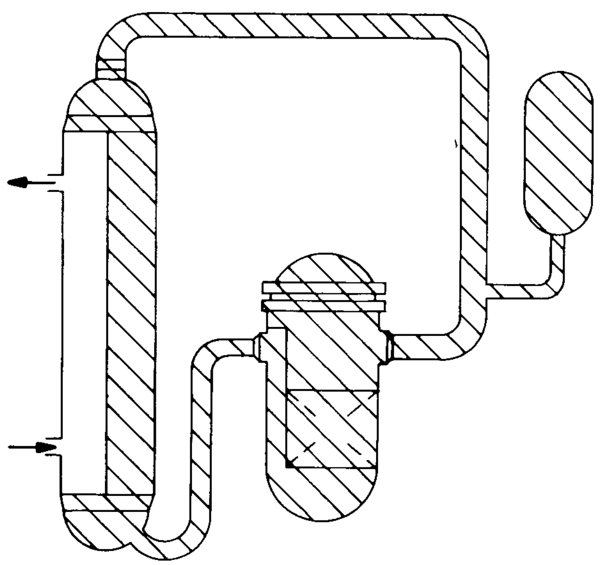
TIME II

1. RC HEATS UP DUE TO LOSS OF SECONDARY HEAT SINK AND EXPANDS INTO PRESSURIZER.
2. RC PRESSURE INCREASES TO PORV SETPOINT. PRESSURIZER STEAM IS EJECTED OUT OF PORV.
3. PRIMARY SYSTEM GOES WATER SOLID (STILL SUBCOOLED) AT 2500 PSIG. CONTINUED HEATUP OF REACTOR COOLANT CAUSES WATER RELIEF OUT OF PORV OR SAFETY VALVES.
4. RC TEMPERATURE IS SLOWLY APPROACHING SATURATED CONDITIONS @ 2500 PSIG.



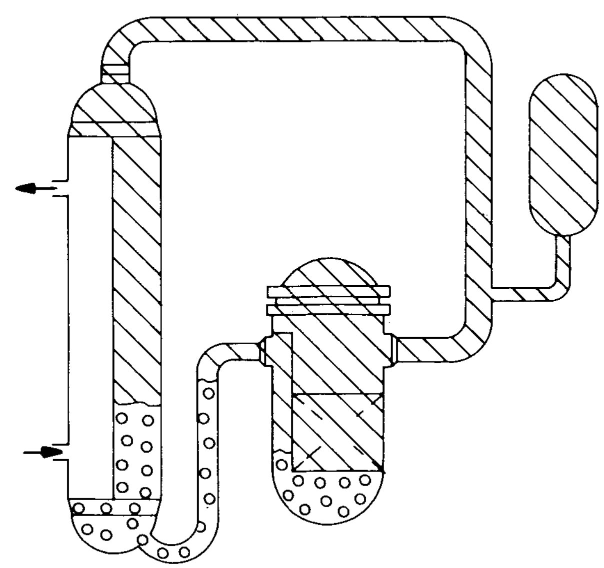
TIME III

1. RCS GOES SATURATED AT 2450 PSIG.
2. STEAM IS CREATED IN CORE BECAUSE OF BOILING THROUGHOUT THE CORE.
   - IF RC PUMPS ARE RUNNING, STEAM WILL BE DISTRIBUTED AROUND THE LOOP (SEE ABOVE).
   - IF RC PUMPS ARE OFF, THE STEAM WILL SEPARATE FROM THE REACTOR COOLANT AND COLLECT IN THE UPPER REGION AT RV AND HOT LEG.



TIME IV (RC PUMPS ON)

END CONDITION FOR TOTAL LOSS OF FW WITHOUT HPI ACTUATION. SYSTEM WOULD COMPLETELY VOID (STEAM ONLY IN RCS). CORE TEMPERATURE WILL INCREASE AND CAUSE SUPERHEATED STEAM TO FORM. INADEQUATE CORE COOLING CONDITIONS EXIST.



TIME IV (RC PUMPS OFF)

END CONDITION FOR TOTAL LOSS OF FW WITHOUT HPI ACTUATION. WITH RC PUMPS OFF EARLY IN THE EVENT, REACTOR COOLANT CAN BE TRAPPED IN LOWER REGIONS OF LOOP AND REACTOR VESSEL. CORE WILL HEATUP WHEN MIXTURE LEVEL DROPS BELOW TOP OF FUEL. INADEQUATE CORE COOLING WILL EXIST.

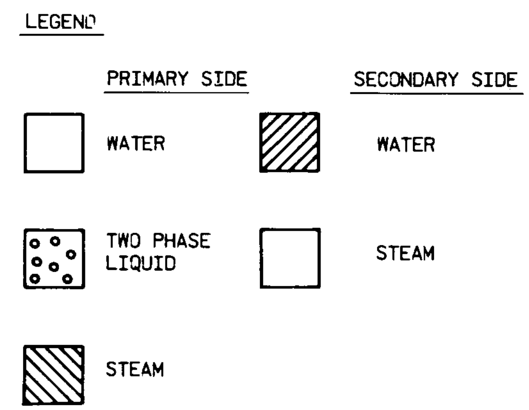
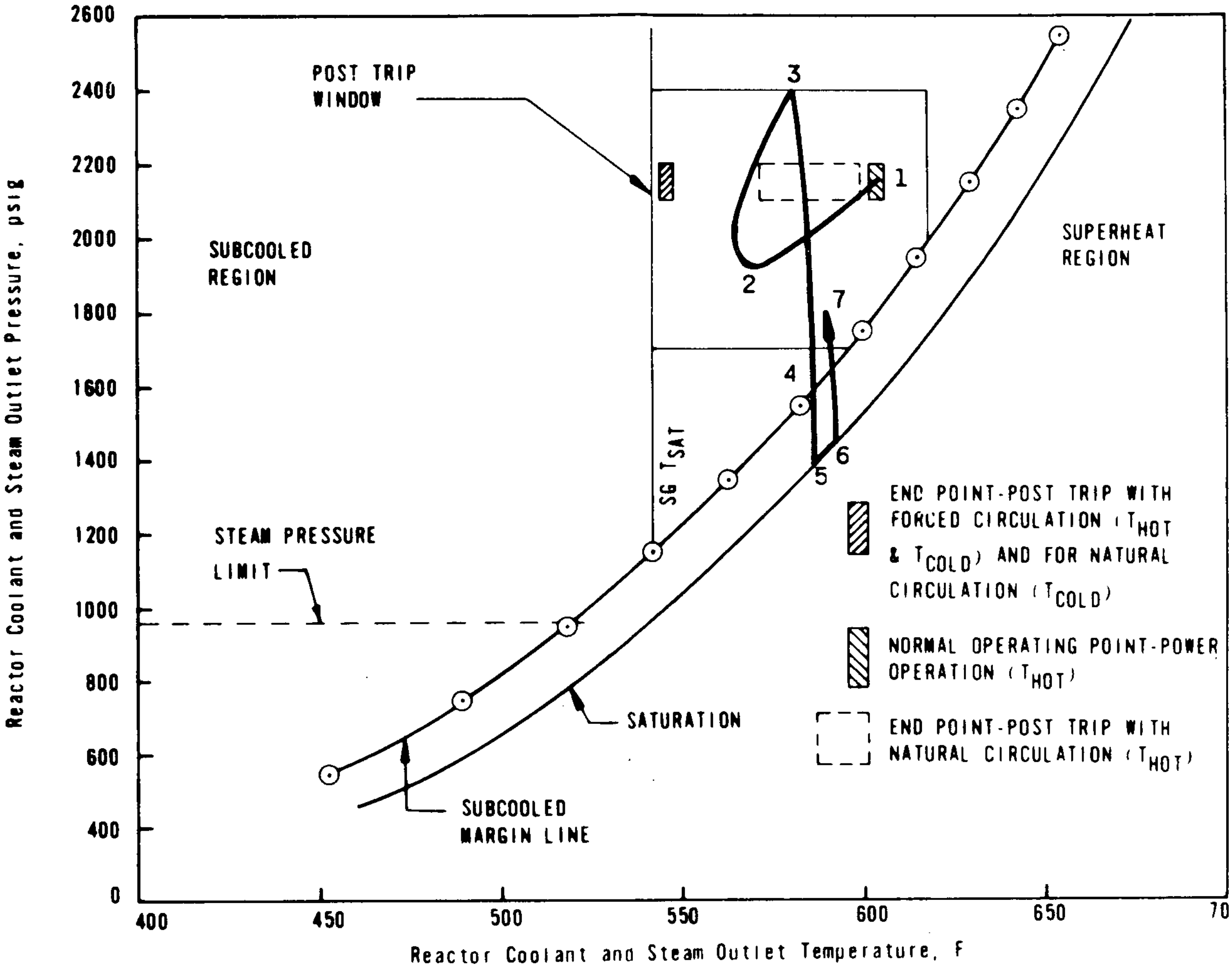


Figure 24b BACKUP COOLING BY HPI FOR LOSS OF ALL FEEDWATER (WITH OPERATOR ACTION)



Reference Points	Time (Minutes)	Remarks
1-2	0-1	Reactor tripped on anticipatory loss of feedwater. Normal post-trip cooldown and depressurization in progress. EFW does not initiate.
2	1-2	Steam generators dry. RCS begins to reheat and repressurize due to loss of secondary cooling.
3	3-4	Operator diagnoses loss of heat transfer, opens PORV, starts two HPI pumps and balances HPI flow. PORV release rate exceeds HPI capacity initially and RCS begins to depressurize. Operator trips all but one RC pump to reduce heat input.
4	5-6	Subcooled margin is lost. Operator trips remaining RC pump.
5	6-7	RCS reaches saturation.
6	7-8	Pressurizer in solid or near solid condition. HPI flow "matches" decay heat and begins to repressurize RCS to subcooled conditions.
7	8-10	RCS subcooled margin restored and RCS is beginning to cool due to HPI flow and PORV release. Operator throttles HPI flow to maintain subcooled conditions at a pressure lower than the safety valve setpoint and restarts an RC pump to promote thermal mixing of HPI to minimize thermal shock.

Figure 25 RC PRESSURE/TEMPERATURE LIMITS

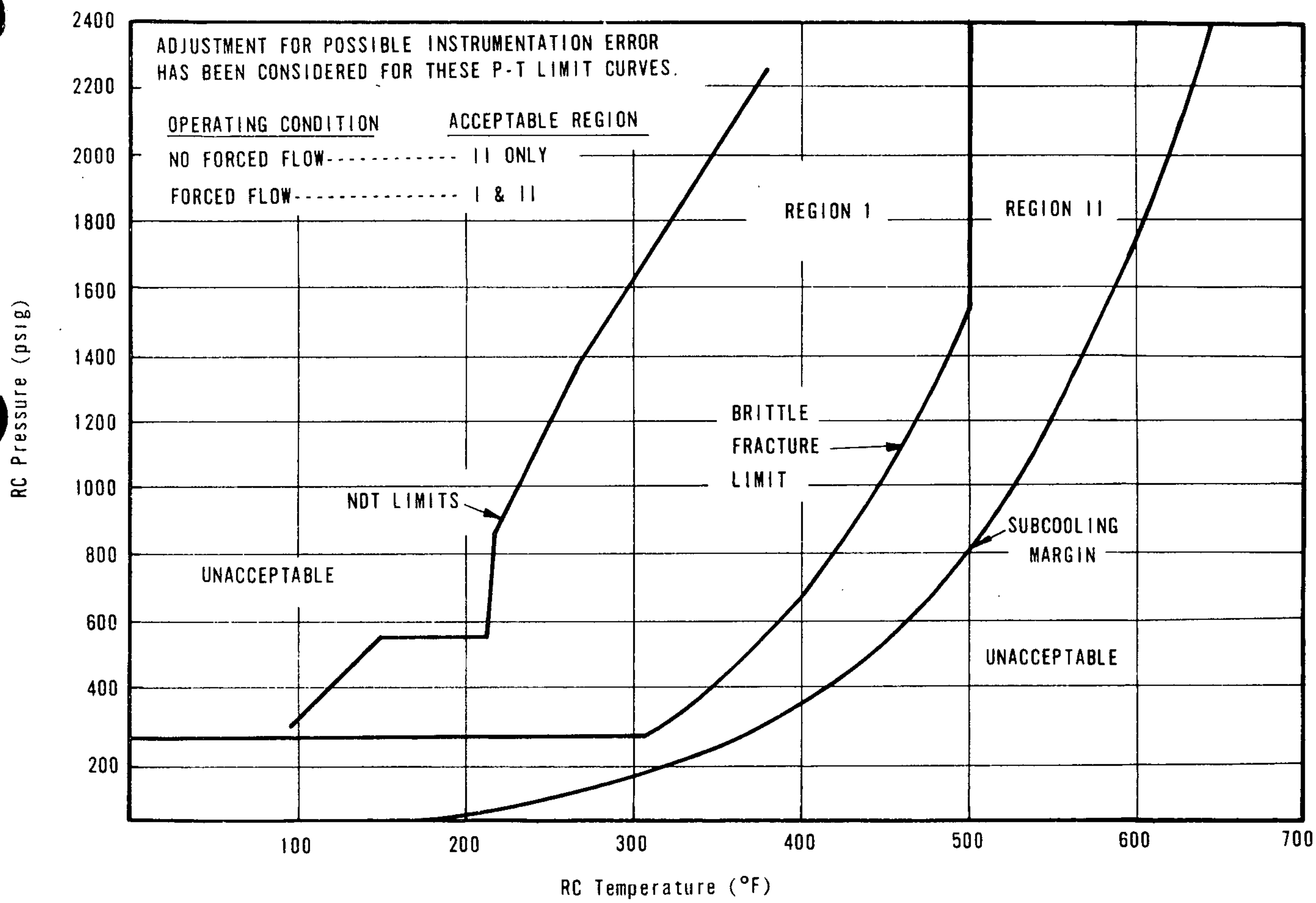
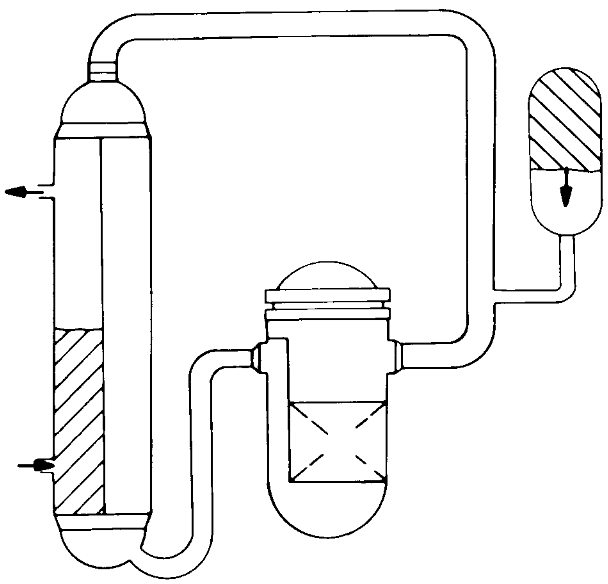
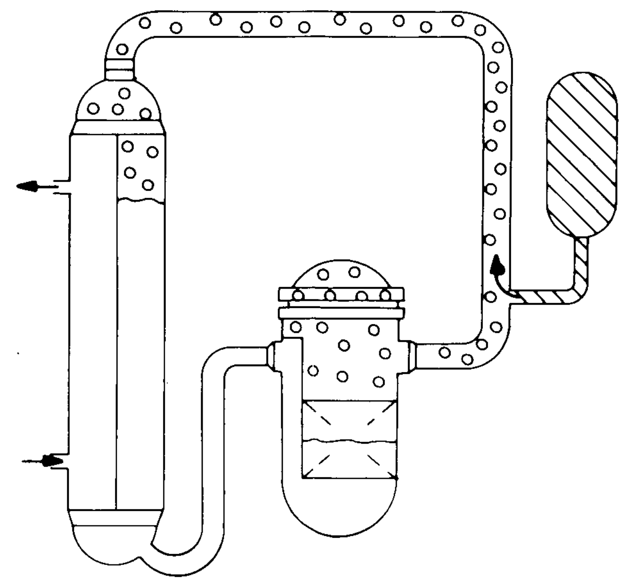


Figure 26 ILLUSTRATION OF LOSS OF NATURAL CIRCULATION DUE TO BUILDUP OF STEAM IN THE REACTOR COOLANT SYSTEM



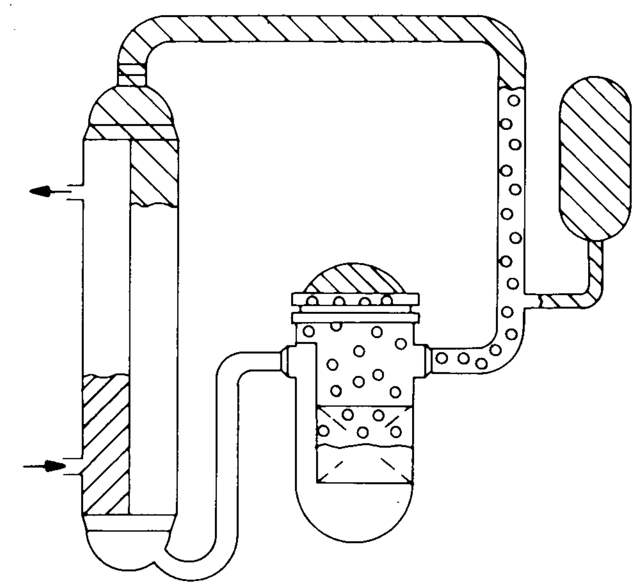
TIME I

1. DECREASING PRESSURIZER LEVEL BECAUSE OF LOSS OF REACTOR COOLANT (LOCA) OR CONTRACTION OF REACTOR COOLANT (OVERCOOLING).
2. REACTOR TRIP ON LOW REACTOR COOLANT PRESSURE
3. ES ACTUATION ON LOW REACTOR COOLANT PRESSURE
   - HPI ACTUATION



TIME II

1. PRESSURIZER LIQUID VOLUME IS LOST; STEAM FROM PRESSURIZER CAN ENTER RCS LOOPS WHEN THE RCS DEPRESSURIZES.
2. REACTOR COOLANT PRESSURE DROPS TO A VALUE ABOUT EQUAL TO STEAM GENERATOR PRESSURE. OPERATOR TRIPS RC PUMPS WHEN SUBCOOLED MARGIN IS LOST.
3. STEAM FORMS IN HOT LEG BECAUSE OF ACCUMULATION OF PRESSURIZER STEAM AND BECAUSE OF FLASHING OF REACTOR COOLANT. STEAM IS IN THE FORM OF BUBBLES WITHIN RCS.
4. 2-PHASE NATURAL CIRCULATION WILL OCCUR - BOILING MAY OCCUR IN CORE.
5. IF SUFFICIENT STEAM IS CREATED, IT WILL START TO COLLECT IN UPPER REGION OF LOOP BECAUSE STEAM CAN RISE AT A FASTER VELOCITY THAN WATER.



TIME III

1. STEAM SEPARATES IN UPPER REGIONS OF LOOP.
2. 2-PHASE NATURAL CIRCULATION STOPS.
3. SIZE OF THE STEAM BUBBLE DEPENDS ON SEVERITY OF ACCIDENT. HPI FLOWRATE, AND HPI STARTUP TIME.

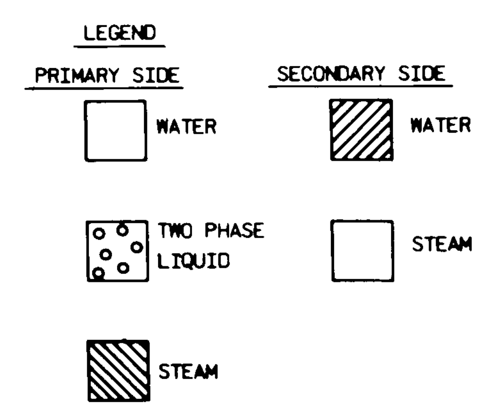
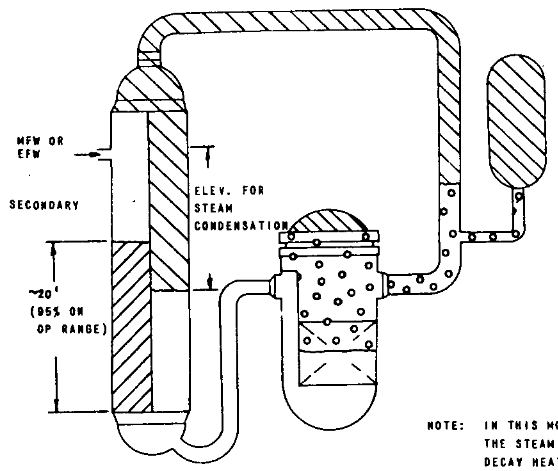


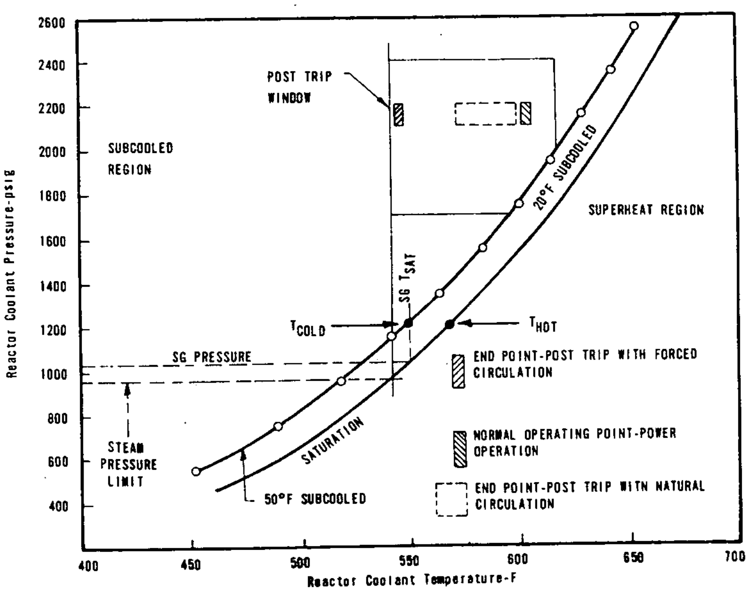
Figure 27 BOILER-CONDENSER COOLING



NOTE: IN THIS MODE OF OPERATION THE STEAM GENERATORS CAN REMOVE DECAY HEAT BY CONDENSATION OF PRIMARY COOLANT STEAM. THE STEAM THAT IS CONDENSED KEEPS A WATER LEVEL IN THE STEAM GENERATOR TUBES TO FORCE FLOW UP THE COLD LEG AND INTO THE REACTOR VESSEL.

LEGEND

PRIMARY SIDE	SECONDARY SIDE
WATER	WATER
TWO PHASE LIQUID	STEAM
STEAM	



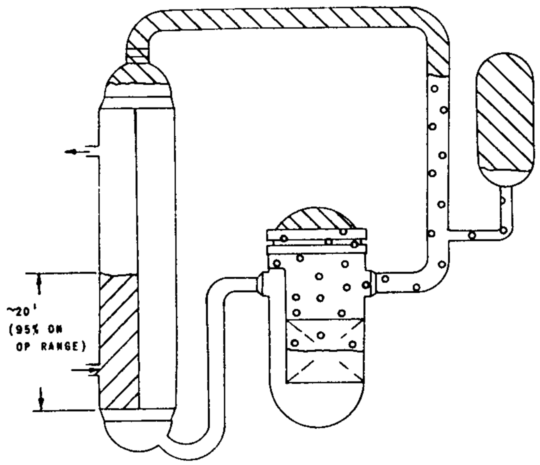
NOTES ON P-T DIAGRAM

1. RC pressure is slightly higher than steam generator pressure.
2. T_{hot} is equal to T_{sat} for existing RC pressure.
3. T_{cold} is equal to T_{sat} for existing steam pressure.

OPERATOR ACTION REQUIRED

1. Turn HPI on to highest flow rate.
2. Verify AFW flowing through upper nozzles (or start EFW) and raise steam generator level to 95% on operate range.
3. Start plant cooldown at 100F/hr.
4. Monitor plant conditions for a loss of core boiling/SG condensing or a return to normal natural circulation (subcooling).

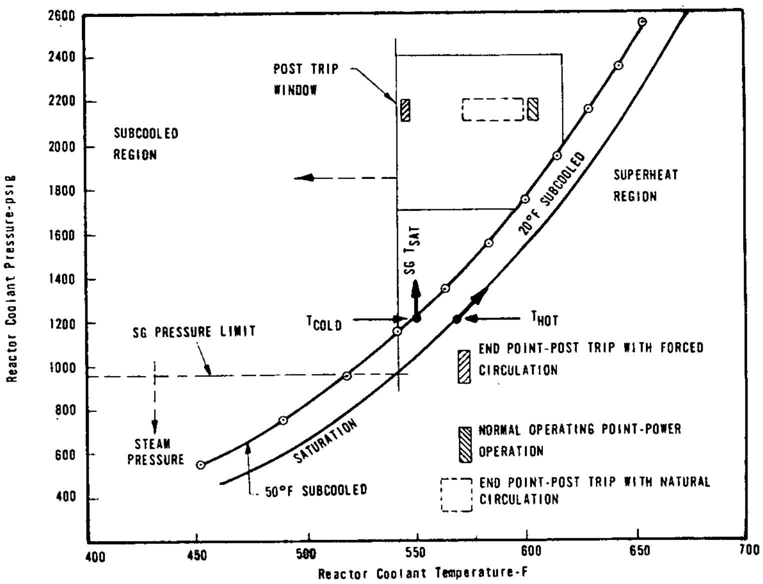
Figure 28 LOSS OF NATURAL CIRCULATION - SYSTEM REFILL BY HPI



LEGEND

PRIMARY SIDE	SECONDARY SIDE
WATER	WATER
TWO PHASE LIQUID	STEAM
STEAM	

NOTE: HPI IS REFILLING THE PRIMARY SYSTEM. THE STEAM GENERATOR CANNOT REMOVE DECAY HEAT BECAUSE THE HOT LEG IS FULL OF STEAM AND FLOW IS BLOCKED.



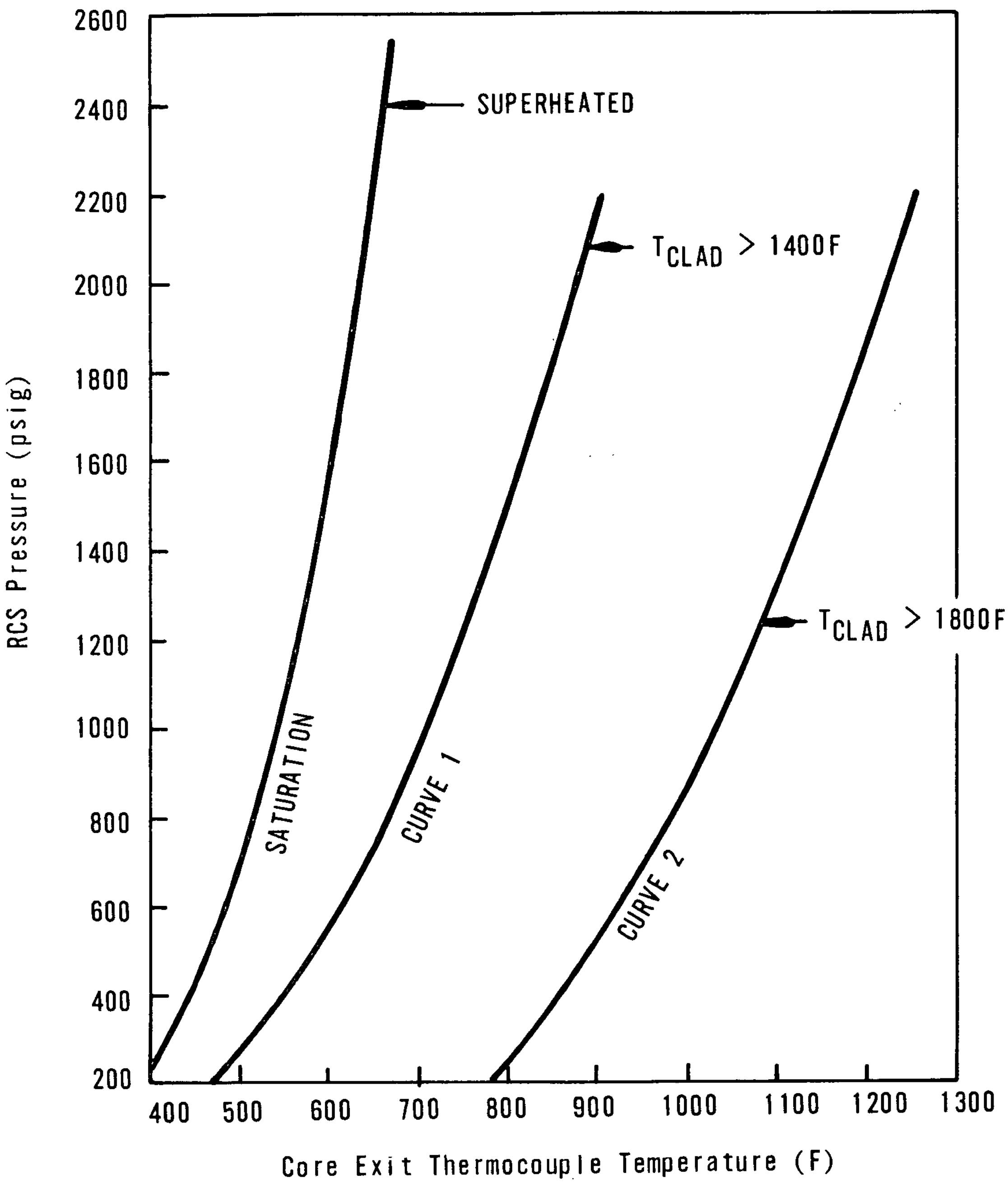
NOTES ON P-T DIAGRAM

1. T_{hot} is equal to T_{sat} for existing RC pressure.
2. RC pressure will increase above steam generator pressure and can go as high as the pressurizer safety valve setpoint (2500 psig).
3. Steam pressure may drop because heat transfer from reactor coolant is low.

OPERATOR ACTION REQUIRED

1. Same as boiler-condenser cooling.
2. Establish the steam generators as a heat sink (SG T_{sat} should be about 50F less than the incore thermocouple temperature).
   - Open PORV
   - Bump one RC pump

Figure 29 CORE EXIT FLUID TEMPERATURE FOR INADEQUATE CORE COOLING



CHAPTER E

BEST METHODS FOR EQUIPMENT OPERATION

During an abnormal transient the operator has to perform several actions to control different systems. This section will show the best ways to do the following things:

- Start and Stop RC Pumps
- Throttle or Stop HPI
- Throttle or Stop Emergency Feedwater
- Stop Main Feedwater
- Use the Incore Thermocouples
- Cooldown with One Generator out of Service
- Use the High Point Vents

RC Pumps

During the course of an abnormal transient the RC pumps may be stopped and at a later time may be restarted depending on the kind of transient and the conditions of the reactor coolant system.

In general the reasons for stopping the pumps are:

- To prevent pump damage
- To prevent possible core damage if a small break LOCA occurs

In general the reasons a pump may be restarted are:

- To start natural circulation if it has stopped
- To allow a faster rate of cooldown and RCS depressurization

- To provide core cooling if the core has become uncovered (inadequate core cooling)
- Prevent brittle fracture of the reactor vessel when the reactor coolant is subcooled and no circulation exists.

This section will show the rules for stopping and restarting RC pumps.

RC Pump Trip

RC pumps must be tripped during a small break LOCA if the subcooling margin is lost to prevent core damage. If the pumps were kept running they would force steam and water by the break; because the water is forced by the break more reactor coolant mass is lost than if they were not running. As long as the pumps continue to run the core will be cooled by the steam and water mixture circulating through the core. But if the pumps are tripped at a later time, when very little liquid remains in the RCS, the steam and water remaining in the vessel and loops will separate. Steam will collect in the high points and water will collect in the low points. If enough water does not collect in the vessel the core will be uncovered and will not be adequately cooled. Core damage can result. Analyses show that a later pump trip can be dangerous, but an early pump trip is safe.

The RC pumps must be tripped upon loss of subcooling margin. A loss of coolant accident will nearly always cause a loss of subcooling

margin. Other transients, such as severe overcooling or loss of all feedwater can also cause loss of subcooling margin. Because the effects of failure to immediately trip RC pumps during a LOCA can be very serious, the operator should trip the pumps on the loss of subcooling margin without trying to find out the cause.

To avoid failures which may cause the pumps to trip late, the following rule is given:

RC PUMP TRIP RULE
The RC pumps shall be tripped immediately whenever the subcooling margin is lost.

NOTE: IT IS ABSOLUTELY MANDATORY TO TRIP THE RC PUMPS IMMEDIATELY BUT IF THE PUMP ARE NOT TRIPPED IMMEDIATELY (I.E., WITHIN TWO MINUTES) WHEN THE SUBCOOLING MARGIN IS LOST IT IS MANDATORY THAT THEY SHOULD NOT BE TRIPPED AT A LATER TIME. THE OPERATOR MUST MAKE SURE THAT COOLING WATER AND SEAL INJECTION ARE WORKING TO PREVENT PUMP DAMAGE. THESE SERVICES MUST BE MAINTAINED FOR SEVERAL HOURS. IF MECHANICAL DAMAGE TO THE PUMPS CAN OCCUR THEN TWO PUMPS (ONE IN EACH LOOP) CAN BE STOPPED. THE TWO REMAINING PUMPS MUST BE KEPT RUNNING. IF THEY FAIL THE TWO PUMPS WHICH WERE STOPPED SHOULD BE STARTED EVEN IF MECHANICAL DAMAGE CAN OCCUR. THE OPERATOR MUST ALSO TRY TO GET AS MUCH HPI FLOW INTO THE RCS AS POSSIBLE.

The RC pumps can be tripped to prevent mechanical damage in all cases except the one noted above. Mechanical damage is not expected to cause safety problems unless total seal failure occurs.

It is desirable to trip the pumps to prevent mechanical damage in case they must be restarted at a later time. Preserving the pumps for long-term cooling or cooldown is desirable, and it is recommended that they be shut down if high vibration or loss of cooling services occurs. Limits on continued pump operation are given in the "Plant Limits and Precautions". These limits apply to normal and emergency services.

Table 5, "Rules for RC Pump Trips" summarizes these requirements. Included in this table are the limits on pump operation because of failures of cooling water and seal injection. These limits are shown because containment isolation can affect cooling water.

When the RC pumps are tripped to prevent mechanical damage, main feedwater will be automatically diverted through the upper nozzles and the steam generator level setpoint will be changed to 50% on the operating range. The setpoint will also change to 50% automatically if EFW is in operation. The operator should make sure that natural circulation starts. If component cooling water only is lost, time exists to raise the water level before the pumps are tripped. If LPSW cooling water only is lost, ten minutes are available for

raising SG water level before the RC pumps must be tripped. If the pumps are tripped on loss of subcooling margin, natural circulation may or may not start depending on the amount of steam in the RCS. Nevertheless a check on natural circulation is desired. Actions to establish natural circulation when the pumps are tripped because of subcooling margin are given in the pump restart criteria which follows.

TABLE 5 - RULES FOR RC PUMP TRIPS

RULE	REASON
1. The RC pumps shall be tripped immediately whenever subcooling margin is lost.	Precludes the potential for uncovering the core (ICC) during a small loss of coolant accident due to a pump trip when the amount of water in the RCS is low.
2. *If LPSW cooling water to the RC pump motor is lost and the pumps are running, cooling water must be restored within 10 minutes or the RC pumps must be tripped.	Pump trip precludes motor failure and minimizes the chance of a fire inside containment due to lack of cooling water to the RC pump motors.
3. *If seal injection <u>and</u> component cooling water are lost to a RC pump for a period longer than 60 seconds, the pump(s) must be tripped within the next 30 seconds and the seal return line closed 90 seconds later.	Pump trip precludes damage to the pump seals and the chance of a LOCA. Injection and/or CCW can be reinitiated by following pump manufacturer's instructions. If possible, an engineering assessment should be performed before restarting since seal failure could occur due to a high temperature in the seal cavity.

* These rules do not apply if the pumps were not tripped immediately after the subcooling margin was lost. The operator should try to restore cooling water.

RC Pump Restart

Core cooling and plant control are best if the RC pumps are running. Pumps can be restarted after trip if the reactor coolant conditions are right. Therefore, to complement the RC Pump Trip Rule given previously, conditions when the pumps can be restarted are given. These conditions cover both LOCA and non-LOCA events and have been carefully chosen so that a pump restart followed shortly afterwards by an inadvertent trip will prevent fuel damage for small breaks.

Restart of the RC pumps is desirable for several reasons:

- Prevent brittle fracture of the RV when the reactor coolant is subcooled with HPI flow but no forced circulation exists.
- If natural circulation was lost, the pumps can be used to restart it.
- If the plant must be cooled down and depressurized, the RC pumps will permit use of the pressurizer spray.
- Cooldown will be faster with forced circulation and the decay heat removal system can be placed in operation before the BWST is depleted.
- If severe Inadequate Core Cooling (ICC) conditions exist the RC pumps must be restarted.

The major effect of restarting the RC pumps is to increase the rate of heat transfer from the core to the steam generators; or if natural circulation has stopped and there is no heat transfer from the core to the steam generators then a pump bump will help to restart natural circulation. Because the purpose of restarting the pumps is to increase core-to-steam generator heat transfer, it is necessary that the steam generator

be available for heat removal. The steam generator will remove heat if: 1) the steam generator saturation temperature is lower than the RCS incore thermocouple temperature - a 50F temperature difference is a good rule of thumb to use, and 2) the steam generator is fed with main or emergency feedwater. It is necessary to start the pumps in the loop with the operating steam generator if only one is in service, and it is best to start the pumps in the loop with the pressurizer spray if possible. Since it is preferable to keep the pumps operable, a pump restart is not desired if mechanical damage can result.

One RC pump may be restarted and run to prevent brittle fracture when the reactor coolant is subcooled and the core is being cooled by HPI cooling and no circulation exists. For this unusual situation, which can be caused by a prolonged loss of all feedwater, one RC pump may be run even though there is no steam generator cooling. However, it should not be run if mechanical damage can occur. If subcooling is lost the RC pump should be stopped. See the "Backup Cooling" Chapter for details about brittle fracture.

Inadequate Core Cooling is a condition where the reactor coolant is superheated. This is a condition when core damage could occur. For this condition, exceptions are taken: 1) RC pumps can be restarted if the steam generators are <u>not</u> available, and 2) If severe ICC conditions exist, RC pumps must be restarted even if mechanical damage can occur. For all other cases of pump restart, mechanical damage should be avoided.

When the RC pumps are restarted the operator should expect to see pressure changes in the RCS.

- If the reactor coolant is subcooled and the pressurizer is filled solid, an abrupt rise in pressure will occur.
- If the reactor coolant is subcooled with a near normal pressurizer level, almost no change will occur.
- If the reactor coolant is two-phase and saturated, a pressure drop will occur when the heat removal rate of the steam generator increases.

Table 6, "RC Pump Restart Guidelines", shows the conditions when the pumps can be restarted. The table is divided into three parts: subcooled, saturated, and superheated. Guidelines for restart of the pumps in the subcooled and saturated conditions are dependent on the existence of liquid or two-phase natural circulation. Generally, if natural circulation does not exist the RC pumps are "bumped" to try to start natural circulation; if natural circulation does start then that is a good indication that a large amount of water is in the RCS. "Bump" means to start a pump and run it for 10 seconds, then turn it off.

When it is "bumped" it will cause hot reactor coolant in the vessel and hot leg to move into the steam generator; and a pump "bump" will cause cold water in the steam generator to move into the reactor vessel. This

will establish communication between the thermal centers and initiate natural circulation (if enough water is in the RCS). When the RCS is saturated the "bump" may or may not start circulation, but it will help to depressurize the RCS by condensing reactor coolant steam in the generators and allow more HPI to flow into the system.

The "bumps" are used only every 15 minutes because: 1) that will limit the liquid flow out of the break and 2) it will take some time for natural circulation to develop and stabilize. Between "bumps" the development of natural circulation can be checked.

Table 6 shows two columns when the RCS is saturated - one with natural circulation and one without. Both show that HPI is on.

When natural circulation exists the steam generator T_{sat} will control the incore thermocouple temperature; if T_{sat} is changed the incore thermocouple temperature will follow. If the incore thermocouple temperature does not change when T_{sat} changes, the steam generators are not coupled to the reactor coolant system. Extended saturation <u>with</u> the steam generators available as a heat sink can only exist because of a LOCA.

For the condition with no natural circulation the operator is directed to perform several "bumps"; if after four bumps natural circulation does not start, and one hour has elapsed since the reactor trip, then one RC pump

should be run for cooldown as long as at least one OTSG is available as a heat sink. Natural ciruclation will not start when there is not enough water in the RCS. The reasons for this exception, which goes against other requirements that do not permit RC pump operation when the subcooled margin is lost, are that the RCS should be depressurized and placed on the decay heat system before the BWST runs dry (to avoid HPI recirculation from the sump) and that the several "bumps" have consumed time. The one hour time limit has allowed the decay heat load to drop sufficiently so that the HPI system is now capable of adding enough water to make up the flow out of the break and remove all of the heat. There is no chance for the core to become uncovered when the RC pumps are run at this time when the HPI system is working.

HPI Control

The HPI system is used for emergency injection of borated water to make up for lost inventory from a small break. It may also be actuated for other reasons. The operator will have to control the flow rate in different ways depending on the cause of its actuation. The general control actions are:

1. Maximize the flow for ECCS small break*
2. If flow is abnormally low in one train, open the associated header cross-connect valve and verify proper flow.
3. Throttle the flow to prevent runout and cavitation of the HPI pumps at low pressure
4. Throttle or stop the flow to prevent filling the pressurizer solid when the RCS is subcooled (except during HPI cooling as described in "Backup Cooling Methods").
5. Stop the HPI system when the LPI system is operating
6. Throttle the HPI to prevent brittle fracture when the RCS is subcooled and no circulation exists.

*With 3 HPI pumps running, Train A flow should be 40-50% higher than Train B flow. However, only two HPI pumps (one supplying each train) are required.

Each one of these topics will be addressed and the best ways for handling HPI will be shown. The discussion will be divided into two sections: Maximizing HPI Cooling, and Throttling HPI.

NOTE: Manual actuation of HPI should be accomplished on a component level as opposed to a system level actuation by the ES. System level actuation may also result in other actions (e.g., Keowee start, non-essential RB isolation) that may not be desired. Therefore, the operator should manually actuate HPI by opening the BWST suction valves, starting two HPI pumps (one pump supplying each train) and opening the injection line isolation valves.

Maximizing HPI Cooling

> HPI SUBCOOLING RULE: Two HPI pumps should be run at full capacity when:
>
> - The ES is actuated and the HPI is automatically started.
> - The reactor coolant subcooled margin is lost and the HPI is manually started.

NOTE: All three HPI pumps start on automatic initiation but only two are required. Therefore, if all three are operating properly, the operator should secure one of the HPI pumps supplying Train 'A'.

When the HPI system is started for either of these two conditions, its purpose is to remove decay heat either by "once through cooling" or by allowing the reactor coolant to transfer heat to the steam generator. "Once through cooling" or HPI cooling occurs when the injection water passes through the core, picks up heat, and exits through a break or the PORV. In order to be most effective the HPI flow to the core must be the greatest amount possible. One HPI pump will satisfy core cooling requirements, but two HPI pumps are preferred. Balancing the HPI flow is required to ensure that the greatest amount of pumped flow enters the core. When the system is automatically actuated the operator should check the flow indicators on both injection lines. If one injection line reads low flow, the associated cross-tie discharge valve should be opened and cross-tie flow should be checked. This ensures adequate HPI flow enters the core even if the 'C' pump or one of the main injection valves fails. These actions, if required, must be completed within ten minutes of ES actuation.

HPI Throttling

After it is started the HPI must be run at full capacity until the reactor coolant system conditions allow it to be terminated or throttled. Guidelines for throttling or termination and the reasons are given below:

Guideline 1 for HPI termination or throttling:

HPI operation may be terminated if the LPI system has been started and has been flowing at a rate in excess of 1000 gpm in each injection line for 20 minutes.

This condition is applicable to a large LOCA when the RCS depressurizes enough to allow the LPI to flow into the reactor vessel. Since LPI will provide emergency injection at a much greater flow rate than the HPI, HPI can be stopped. The 20-minute delay is used to make sure that the primary system will not repressurize and result in a loss of LPI flow. The minimum flow requirement of 1000 gpm is used to make sure that the injection flow can remove decay heat with no loss of reactor vessel water inventory after HPI is stopped. 1000 gpm flow to each injection line is required to make sure at least 1000 gpm gets into the RCS. A possible break in one of the two LPI/CFT lines would allow LPI water to be lost out the break and not reach the reactor vessel.

Guideline 2 for HPI termination or throttling:

- HPI may be throttled any time the reactor coolant subcooled margin is restored.
- HPI may be stopped any time the reactor coolant subcooled margin is restored and pressurizer level is on scale "low" ($\geq$100") and increasing. Normal makeup should be restarted. The one exception to this guideline is the case where core cooling is provided solely by HPI. In this case HPI can be throttled when the subcooled margin is restored but not stopped until secondary heat removal is established, even though the pressurizer will be solid.

These guidelines apply to both LOCA and non-LOCA transients and are intended to limit the amount of water going into the RCS so that the pressurizer will not fill solid and let water discharge through the pressurizer valves. The pressurizer can fill for two reasons:

- Continued HPI injection
- Reheat and swell of the reactor coolant after an overcooling transient has been stopped.

Although the core will be covered and safe if HPI is not throttled, it is desirable to do so. If there is any doubt about throttling HPI then don't do it. If water were allowed to flow through the pressurizer valves, the plant conditions could get worse. Continued flow through the valves could fill the quench tank and cause the ruptured disc to fail releasing water to the containment, or the pressurizer valves could fail to reclose and a LOCA would result. The one exception is HPI cooling when the PORV is intentionally opened to provide a once-through cooling path for HPI water. If the reactor coolant was very cold the repressurization could cause a violation of the NDT limits.

An overcooling transient causes the reactor coolant to shrink. If HPI is started additional water is added to the RCS. When the overcooling is stopped the core heat will cause the reactor coolant and the added water to swell. It can expand enough to fill the pressurizer. In order to limit the amount of filling the HPI can be throttled when the reactor

coolant subcooled margin is restored and the HPI can be stopped when the subcooled margin is restored and a low pressurizer level indication is shown. The 100 inch pressurizer level indication was chosen because it is just above the pressurizer heater cut-off level at 80".

If the overcooling was severe throttling HPI alone may not be enough to prevent the pressurizer filling; therefore, the reheat of reactor coolant must also be limited. This can be done by lowering steam pressure. In most cases a steam pressure reduction of 100 to 200 psi will work; however, the operator can monitor the effects of steam pressure by monitoring T_{cold} and pressurizer level and control steam pressure as necessary. The operator should be careful not to lower steam pressure too much or the pressurizer will drain. In many cases throttling or termination of HPI and lowering steam pressure will keep the P-T from returning to the "post trip window". This is an acceptable end point if the system is stable.

For many reactor events that use the HPI system, subcooled reactor coolant conditions will be returned in the first several minutes. When the reactor coolant subcooled margin is established the following general procedure to control RCS inventory should be followed:

<u>HPI Control After RC Subcooling is Regained</u>

1. Avoid too much subcooling (high RCS pressure). There is a tendency to think that if "adequate" subcooling margin is good, then 200F subcooling must be better. The easiest way to get this subcooling

is to allow HPI to run unthrottled and raise RCS pressure. This may lead to unnecessary lifting of the primary safety valves (the time between 2200 psig and 2500 psig with two HPI pumps running unthrottled is about 60 seconds) or in some cases, violation of NDT and brittle fracture limits. Also, there are transients such as as a steam generator tube rupture where the higher RCS pressure makes the outcome worse (large leak rates). Therefore, the operator should throttle HPI and begin to <u>stabilize</u> RCS pressure as soon as subcooling margin is regained.

2. Check pressurizer level.
3. If pressurizer level is less than 100 inches, maintain HPI but reduce the amount of flow that is being added to the RCS.
   - If two or more HPI pumps are running, stop all but one pump.

     NOTE: Run the HPI pump which normally supplies seal injection.
   - If pressurizer level is on scale, throttle HPI using HPI injection valves and attempt to stabilize pressurizer level.

     NOTE: Do not decrease HPI flow below low flow limits (> 35 gpm).
   - Maintain HPI at the reduced flow rate if pressurizer level and subcooled margin stabilizes (i.e., HPI is matching a leak).
   - If pressurizer level continues to increase above 100 inches, control HPI per Item 4 below (except during HPI cooling as described in "Backup Cooling Methods").

4. If pressurizer level is increasing and greater than 100 inches (indicated), realign the HPI system into the normal makeup and letdown mode.
   - Monitor reactor coolant subcooling; restart HPI if subcooling margin or pressurizer level is lost.
   - Reset ES after HPI has been stopped (if RCS pressure is high enough).
5. If the RCS is subcooled and the pressurizer level is increasing rapidly, it may also be necessary to open the turbine bypass valves to decrease steam pressure to prevent the RCS from going water solid.

Guideline 3 for HPI termination or throttling

- The HPI must be throttled to prevent pump runout and cavitation damage. The maximum allowable flow per pump is 550 gpm.

This guideline is implemented for pump protection so that core cooling will continue. Calculations show that, even when the reactor coolant system is at atmospheric pressure, the injection line orifices will limit HPI flow to 540 gpm per pump. However, the additional flow through the recirculation lines can result in total pump flow slightly greater than the 550 gpm limit.

Guideline 4 for HPI termination or throttling

- The HPI low flow limit is about 35 gpm. Total pump flow should not be throttled below this limit.

Pump overheating and damage can occur at very low flows. The total flow is a combination of the recirculation flow and the injected flow.

Guideline 5 for HPI termination or throttling:

- HPI must be throttled to prevent brittle fracture of the reactor vessel when the reactor coolant is subcooled but not circulating.

The RCS pressure/temperature combination must be kept within certain limits to assure reactor vessel integrity. These limits are dependent on whether there is forced flow, or NO forced flow:

Forced Flow

As long as at least one reactor coolant pump (RCP) is running, the RCS pressure and temperature must be kept within the normal technical specification NDT limits (Region I & II of Figure 25).

With at least one RCP running, any cold leg RTD can be used to determine the temperature for comparison to the NDT limit. However, due to back flow in the cold leg pipes without an operating pump the cold leg RTD in

these loops will indicate temperatures slightly lower (∿ 2F) than in the cold leg pipes with running RC pumps due to the relatively cold HPI and seal injection water added to the back flow.

No Forced Flow

If the RC pumps are NOT running, the RC pressure/temperature combination must be kept within the no forced flow region of Figure 25 (Region II). The reactor vessel downcomer temperature will be colder during no flow conditions than during Forced Flow conditions. The "Interim Brittle Fracture Limit" of Figure 25 is designed to account for these colder temperatures. These colder temperatures occur because the HPI flow entering the RCS does not completely mix with the reactor coolant as would happen if the RC pump were operating. The HPI water, which can be as low as 40F per Technical Specifications, will enter the cold leg pipe then flow into the RV downcomer to cool the reactor vessel walls. These colder RV temperatures will cause the allowable RV pressure to be lower.

The "Interim Brittle Fracture Limit" requires throttling the HPI flow. This will reduce the brittle fracture probability by first reducing the RV pressure and second reducing the HPI cooling of the RV.

The "Interim Brittle Fracture Limit" of Figure 25 is based on several conservative assumptions, consequently, small violations of this limit are more tolerable than similar violations of the subcooling margin. However,

if the "Interim Brittle Fracture Limit" is exceeded, the RCS pressure should be reduced to regain the no forced flow operating region as quickly as possible.

a) Monitoring Thermocouple Temperatures

With no RCPs running, the average of the five highest thermocouple (TC's) temperature readings should be used to determine the RC temperature for Figure 25. This will assure that the subcooling margin is maintained and that the brittle fracture limit is not exceeded.

The use of the 5 highest TC's is preferred for the following reasons:

1) The operator monitors the five highest TC's for other reasons (ICC), i.e., the data is available and the operator need not perform additional data reduction.
2) The conservatisms in the brittle fracture analysis are more than adequate to support the slightly higher system pressures and slightly lower downcomer temperatures by using the five highest rather than five lowest thermocouple readings.
3) The allowable temperature span between the subcooling margin limit and the NDT limit for RCS operation is relatively narrow. Consequently, the operator should use the same instrument for avoiding both limits. If the subcooling margin is determined by averaging the five highest thermocouple readings, the margin to the brittle fracture limit should be determined with the same readings to avoid overlapping limits.

Without a RC pump on and HPI, with or without natural circulation the cold leg temperature detectors cannot measure the RV downcomer temperature because the RC loop flow and HPI flow mix downstream of the temperature detector. The ratio of the HPI flow to RC loop flow is substantial. Consequently, the resulting mixed temperature of the two fluids will be substantially lower than the cold leg temperature indication. The ratio of the two flows will vary with the size of break in the reactor coolant pressure boundary. The larger the break the more the HPI flow and the less the reactor coolant flow.

During natural circulation the hot leg temperature detector with loop flow can be used to indicate core outlet temperature. However, to simplify the operating instruction the same thermocouples are to be read, whether or not natural circulation exists. Therefore, the operator does not have to determine if natural circulation exists or to switch from one measuring device to another.

If HPI flow does not exist, the operator should use the normal P-T limit curve during natural circulation and the cold leg temperature detector in the loop(s) with natural circulation flow.

b) RCS Pressure Control

With no RCPs running, throttling the HPI flow is the only method for gradually reducing RCS pressure. Also, without primary to secondary heat transfer, the rate of cooldown is dependent on HPI cooling through the break (opening the PORV is necessary only if the break is so small that RCS pressure begins increasing). Therefore, careful and consistent throttling of HPI flow is the only available means to ensure that the RC pressure/temperature combination remains within the no forced flow operating region of Figure 25.

c) Restoring Natural Circulation

If a RCP cannot be started, natural circulation should be obtained to provide some HPI mixing as well as providing good heat transfer from the primary to secondary coolant. With natural circulation, the cold HPI water will mix with cold leg flow and reduce the thermal shock to the reactor vessel. However, the RC pressure/temperature should still be maintained within the no forced flow operating region of Figure 25. The brittle fracture concern is eliminated entirely when RCPs are running and RCS P/T is maintained within Tech Spec NDT limits. Therefore, as soon as subcooling is obtained in the RCS, a RCP should be restarted. Then the reactor vessel downcomer pressure/temperature should be kept within the normal NDT limit.

HPI VALVE CONTROL

The HPI valves will be used to reduce HPI flow until the RC system pressure becomes low enough to initiate the LPI or DHR system. If a small break exists, the RC pressure will have to be reduced to the LPI operating pressure. If only a loss of feedwater exists then the RC pressure need only be reduced to the DHRS operating pressure (unless secondary cooling is re-established) and then the pressurizer power operated relief valve should be closed.

Without feedwater the time to depressurize will depend on the size of the break. The HPI flow will be performing two functions. It will be maintaining system pressure which will be a function of the HPI pump head and the choked flow out the break. It will also be removing decay heat from the core. The amount of decay heat will determine the amount of HPI flow needed and the HPI flow will establish the RC pressure. Consequently, as the decay heat level decreases the HPI flow can be throttled back which will cause the RC pressure to reduce.

With feedwater and natural circulation or a pump operating, the HPI flow to the core is needed only to control pressure. The steam generator will remove heat. Consequently, the RCS can be depressurized much quicker. The steam generator can cool the core as quickly as possible up to the 100F/hr. limit. Simultaneously, the HPI flow will be throttled to maintain the RC pressure within the acceptable P-T limits.

The HPI flow rate should be balanced among all injection nozzles to distribute the HPI flow around the reactor vessel downcomer as much as possible. This will limit localized cooling of the RV.

During plant cooldown a situation may occur where the RC pressure cannot be reduced by throttling HPI flow. This can be caused by hot water flashing to steam in either of the RC hot leg 180 degree elbows (due to no flow in one or both loops) or in the pressurizer (PORV closed).

RC pump operation and opening the pressurizer PORV is required to eliminate the problem.

The operator should attempt to remove the RC loop steam and hot water by bumping a RC pump in the loop with natural circulation flow. If no natural circulation flow exists any RC pump can be bumped.

If a break does not exist in the pressurizer the PORV should also be opened to reduce RC pressure.

These guidelines are summarized in Figure 30.

Feedwater Control

Abnormal transient operation with main or emergency feedwater requires special attention to feedwater control. Failures can cause too much water to be added. Excessive main feedwater addition can fill the steam lines with water (steam lines may fail) and cause undesirable overcooling, especially if feedwater heating is lost and cold water is added to the steam generator. Excessive emergency feedwater can have the same general effects, but it will cause a more severe cooldown (for the same flowrate) because of the greater steam pressure reduction effect due to the colder water. Both excessive main and emergency feedwater may require that quick actions be taken to stop it. Emergency feedwater may also cause overcooling when the steam generator level is being raised even though no failures have occurred. In order to limit overcooling, emergency feedwater should be throttled. This section will recommend the best methods for manual control.

Main Feedwater Overfill

The procedural guidelines in Part I assume worst case (very rapid) MFW overfill conditions and therefore direct the operator to immediately trip both MFW pumps. This section, however, presents less severe actions that can be taken in the event of slower overfill transients.

Both MFW pumps should trip automatically when one SG reaches a level of 95% on the operate range. However, the operator should act to terminate excessive feedwater as soon as it is discovered rather than rely on the

pump trip. Therefore, to regain control and stop main feedwater overfilling when a failure of the controls occurs, the following gives a series of increasingly more severe actions:

- Attempt to manually control the feedwater pumps with the hand/auto station; this may not work if the controls to the pumps have failed, so be prepared to quickly take the next step.
- Close the feedwater isolation valve to the high flow generator. This action is preferred to closing the control valves or running back feedwater with the ICS feedwater demand hand station because those controls may have failed and could have been the reason for the excessive feedwater. Closing the isolation valve cuts off feedwater to only one generator and does not cause a total loss of main feedwater.
- Trip the main feedwater pumps. This is the quickest and surest method of stopping the overfill. It is also the preferred method if the OTSG is overfilling rapidly. This will stop all feedwater to both generators (it will be a loss of feedwater, but since the generators have a large inventory the heatup effects will be delayed). This action can be taken if both generators have failed feedwater or the other actions do not work.
- Flow should be monitored in all cases; it will show the effects of corrective action faster than level. The corrective actions must be taken within 2-3 minutes to prevent steam generator overfill (water level at the top of the shroud) with large excessive MFW flow rates.

If all main feedwater has been stopped, the operator should make sure emergency feedwater starts so it can start to inject when the generator water level boils down to the automatic setpoint.

Emergency Feedwater Overfill

To stop emergency feedwater from filling one steam generator:

- Attempt to close the control valve to high level/high flow steam generator. This may not work if the valve controls have failed.
- Trip one EFW pump (select the motor-driven pump supplying flow to the high level/high flow generator) and close the cross connect valve to prevent the turbine-driven pump from supplying flow to the bad generator. To restore emergency feedwater operation the failed control valve may be closed manually and the bypass valve around the control valve may be opened. The pump can be restarted and the cross connect reopened.

To stop emergency feedwater from filling both steam generators (this condition may happen if power supplies or station air are lost to the control valves):

- Ensure control valves are operating properly on nitrogen. If not, check N_2 valve lineup. If control valves cannot be used, then:
- Isolate normal EFW feedlines and open alternate flow paths using ICS-controlled valves. If these valves have also failed, then:

- When steam generator level is high, stop pumps, allow the steam generator level to drop and restart one pump. Use that one pump to "batch" feed the generator by starting and stopping the pump. Infrequent starts and stops of the pump are expected because the steam generator level will take about 5 to 10 minutes to boil the inventory to a low level.
- If no control of EFW can be obtained the pumps can be stopped and HPI cooling can be used. This method is not desirable, but it will keep the core cool. Every attempt should be made to maintain EFW to at least one generator, even if the operation is not steady.

FW Throttling (MFW and EFW)

Anytime MFW flow is diverted through the upper nozzles or EFW is actuated and automatically increases steam generator level to 50% on the operating range, it can cause significant overcooling. The 50% level is required to establish natural circulation any time all RC pumps are tripped.

Steam generator level must also be manually increased from 50% to 95% on the operating range when the subcooled margin is lost. The 95% level will permit primary coolant steam condensation during boiler-condenser cooling in case a small break LOCA has occurred. Anytime the subcooled margin is lost the level should be raised to 95%; if the subcooled margin is regained while the level is increasing then it does not need to be continued to the 95% level, but must be raised to 50% if the RC pumps are not running.

Steam Generator Level Rule

Anytime subcooled margin is lost, levels in the operable steam generators must be raised to 95% on the operating range using FW flow (MFW or EFW) through the upper nozzles in accordance with the FW throttling guidelines.

Exception: If the loss of subcooled margin was due to a loss of secondary steam pressure control, do not attempt to raise level in the affected steam generator(s) until steam pressure control is regained.

Overcooling can result because FW flow through the upper nozzles injects water into the steam space of the generator. As the flow sprays into the steam space it causes steam condensation and a reduction of steam pressure; when the level increases the inventory accumulation is a colder heat sink than is needed to balance decay heat. The combination of the steam pressure reduction and the colder heat sink causes the overcooling.

Automatic level control will provide essentially full FW flow until level approaches the setpoint. When the natural circulation setpoint is in effect this maximum flowrate can cause significant overcooling. Addition of FW at the maximum rate is not needed to achieve stable natural circulation; it is also not necessary to raise the level from 50% to 95% at the highest possible rate. FW can be throttled to control the level and limit the overcooling. Full flow of FW is not

needed, but continuous flow is. A continuous addition of FW into the steam space will cause the thermal center for natural circulation to be high in the generator, and continuous addition will cause primary steam condensation. However, if natural circulation is lost (for example, due to a delay in EFW actuation or an interruption in MFW flow), then FW flow should not be throttled until natural circulation starts.

It is not mandatory to limit the rate of FW addition to prevent overcooling, but throttling is preferred to control the plant better. For example, if a severe overcooling transient caused loss of the subcooling margin and the RC pumps were tripped, the addition of FW at full flow through the upper nozzles would cause the overcooling transient to be much worse. Throttling is desirable to control the severity of this type of transient.

EFW can result in considerably more overcooling for the same flowrates because it is much colder than MFW. Therefore, when MFW is not available, the following additional guidelines for EFW flow control should be used.

Guidelines for EFW Throttling:

- EFW may be throttled any time it is started immediately after loss of main FW when it is injecting into both steam generators and RC pumps are off (EFW should not be throttled when in automatic control with the low level setpoint in effect).
- EFW may be throttled any time after it is started and natural circulation exists in one or both steam generators.

The limits on EFW Throttling Are:

- Steam generator level must be gradually raised to the setpoint; the steam generator level must never be allowed to decrease if level is still below the applicable setpoint (see setpoints below).
- Flow into the steam generators must be continuous at all times until the setpoint is reached.

Restrictions on EFW Throttling:

- EFW must be turned on full if natural circulation stops and the steam generator level is below the setpoint.
- EFW must be turned on full if its actuation was delayed. It can be throttled when natural circulation starts.
- EFW must be turned on full if it is only injecting into one generator. It can be throttled when natural circulation starts.

Steam Generator Level Setpoints:

- 30" on the startup range when one or more RC pumps are operating.
- 50% on the operating range with two steam generators (it may be necessary to raise the level higher than 50% if only one steam generator is working) when no RC pumps are operating.
- 95% on the operate range when the subcooling margin is lost.

The amount that EFW can be throttled depends on the decay heat load which can vary depending on the prior operating power history. To increase level the flow must be greater than that required to remove the decay heat.

Because the decay heat can be different the amount of flow needed to remove decay heat and increase level is different; therefore, no fixed flow rate can be established. However, the maximum flow rate can be gauged by its effects. Generally, the flow rate should not drop steam pressure by more than about 100 psi below the pressure setpoint. For example, after a trip the turbine bypass set pressure is 1010 psi so the EFW flow should not cause steam pressure to drop below around 900 psi. If the operator has adjusted steam pressure to a different setting the steam pressure drop should stay within 100 psi of that setting. The 100 psi change in steam pressure is a rule of thumb for limiting the cooling of the RC system.

The effects of EFW throttling can also be seen in pressurizer level (if the reactor coolant is subcooled and is circulating). Pressurizer level indication should be visible and not drop out of range because of EFW.

The most important effect is to maintain natural circulation. If natural circulation has been previously established and the EFW flow rate is enough to maintain natural circulation, then it is the right flow (if pressurizer level and steam pressure are about right). Generally, natural circulation is established and maintained when T_{hot} and the incore thermocouples track together.

EFW should not be throttled if it does not automatically start after loss of main feedwater when RC pumps are off. It also should not be throttled if only one generator is available for heat removal. For either of these conditions natural circulation could be stopped or not started. Small break LOCA primary steam condensation might also be restricted for either situation.

EFW throttling is not mandatory, but is desirable to limit overcooling and possible pressurizer draining. If there is doubt about throttling EFW, then don't do it.

<u>EFW Throttling Rule</u>

EFW should <u>not</u> be throttled whenever:

- The low level setpoint (one or more RC pumps on) is in effect <u>or</u>
- The natural circulation setpoint (all RC pumps off) is in effect <u>and</u>
  - EFW actuation was delayed <u>or</u>
  - Natural circulation stops and levels are below the setpoint <u>or</u>
  - EFW is injecting into only one steam generator.

(EFW <u>may</u> be throttled whenever natural circulation starts)

Use of the Incore Thermocouples

The incore thermocouples can be used for a variety of reasons. Information about the incores is given in different chapters. The following summarizes that information:

1. They are used to detect core uncovering. They are the most valid indication of core cooling. If the incore thermocouples clearly indicate superheated conditions, then the actions to counter Inadequate Core Cooling should be taken.
2. They provide indication of natural circulation. T_{hot} should read within 10F of the incore thermocouples when the plant is subcooled and solid water natural circulaiton is occurring. When the reactor coolant is saturated the incore thermocouples do not provide as good an indication of natural circulation.
3. They provide an indication of NDT margin when no forced circulation exists. The reading of the five highest thermocouples displayed should be averaged, and compared to the Region II limits on Figure 25, "RC Pressure/Temperature Limits." If that number is beyond the NDT limit the HPI pumps should be throttled or an RC pump started.
4. They are the only valid indication of core outlet conditions when no circulation exists.

Cooldown with One Steam Generator Out of Service

Attempting to cool the plant down using one "good" steam generator can cause excessive thermal stresses in the other "bad" steam generator if it is dry and the cooldown rate is large. Although one steam generator can remove the decay heat and the stored heat needed to cooldown, the dry steam generator is not properly cooled because the shell stays hot.

During normal cooldown the shell of each generator is cooled by liquid in the lower part and by steam in the upper part. When the shell is not cooled and the tubes are cooled by reactor coolant, the tubes can get much colder than the shell causing them to contract relative to the shell. But because the tubesheets hold the tubes in a fixed position and the shell does not shrink the tubes go into tension. If they get cold enough the tension stresses will be greater than the yield stress and they will permanently stretch. If the tubes are cracked, flawed, or thinned they may fail. Consequently, limits are placed on the tube-to-shell ΔT. For normal cooldown this limit has been conservatively set at 100F. However, in an emergency situation when cooldown is absolutely required the limit has been relaxed to 150F ΔT, with the understanding that any transient which results in exceeding the design ΔT limit of 100F requires specific stress evaluation to determine SG tube integrity.

Cooldown with one generator at the highest rate of cooldown should not be done unless it is absolutely necessary. The choices to be made prior to cooldown are:

- Stay at stable hot conditions until the generator is repaired and returned to service.
- Cooldown at a slow rate so that the tube-to-shell temperature limit does not exceed the "normal" ΔT of 100F.
- Cooldown at a more rapid rate, but do not allow the tube-toshell temperature limit to exceed the "emergency" ΔT of 150F.

The need to cooldown can only be established after a review of the plant status. There are a limited number of reasons why cooldown may be required; these include:

- <u>LOCA</u> - small or intermediate break LOCA's will require cooldown so that the primary system can be depressurized. Depressurization will slow down or stop the leak rate. Tube leaks or ruptures especially required depressurization to stop the leakage into the steam generator.

- <u>BWST Draining</u> - in conjunction with LOCA, it is desirable to have the plant completely cooled down to avoid recirculation from the sump using the HPI system. For tube leaks which do not return water to the sump it is absolutely required to have the plant depressurized before the BWST drains.

- Surge tank draining - to avoid using backup service water with poor water chemistry in the steam generator, it is desirable to have the plant on the decay heat removal system before the surge tank is drained (for the unlikely case that the condenser becomes unavailable and condensate cannot be transferred from the other units).

- Accidents other than LOCA - most accidents will not require cooldown for mitigation, so the plant can be placed in hot shutdown while the "bad" steam generator is repaired.

However, some situations, such as fires, may have left the plant so badly damaged that a decision to cool down is necessary to avoid unknown side effects.

In order to cool the plant down rapidly with one generator out of service it will be necessary to add water to the "bad" generator so the shell can be cooled. If the generator is completely dry, the shell will only cool by heat loss through the insulation to the reactor building; the average shell cooldown rate will be low (around 3-5F per hour). Water addition to the generator will allow the shell to cool faster and the rate will depend on whether a water level can be maintained. If water can accumulate and cover the lower part of the shell the average rate of shell cooldown will be about 20F/hr. But if a water level cannot be built and the shell is mostly cooled by steam then the average rate of shell

cooldown will be around 10F/hr. Since the rate of shell cooldown is greatest when water is in contact with it the preferred way to add water is with the main feedwater system. However, the main feedwater flow rate must be carefully controlled so the tubes do not "overcool". The cooldown rate of the plant will be limited by the cooldown rate of the shell and the cooldown limit is based on the tube-to-shell ΔT limit. The tube-to-shell limit can be calculated by averaging the five shell thermocouples and subtracting the reactor coolant average temperature (However, in some rare cases T_{av} might not represent the average tube temperature; these cases can occur if the hot leg is steam bound and no circulation is occurring. If T_{hot} is increasing but T_{cold} is fixed, then T_{cold} should be used rather than T_{hot}.)

To illustrate a plant cooldown two examples are given. Both of the examples assume that a tube leak has occurred. Therefore, the plant must be cooled down; it cannot stay at hot conditions. The first example shows a tube leak with a generator that can hold pressure but cannot be steamed. In this case a water level can be built to cool the lower part of the shell. The second example shows a tube leak with a generator that cannot hold pressure (a failed steam safety valve could do this). In this case a water level cannot be built because of constant steaming (in fact, a water level could be built if

the main feedwater system were allowed to operate at high capacity, but the tubes would cool down extremely fast and the tube-to-shell temperature limit would be violated). The examples are illustrated in Figure 31a and 31b.

Both examples follow the recommended procedure for tube leaks. That is, that plant is runback, depressurized and cooled rapidly from 550F (T_{av}) to $\sim$500F (T_{av}) at a rate 240F/hr. After that the RCS is cooled down and depressurized at $\sim$ 100F/hr until the "emergency" tube-to-shell limit of 150F is reached. At that time the cooldown is slowed and follows the cooldown rate of the shell and the tube-to-shell ΔT is the controlling limit. When the plant first reaches the "emergency" tube-to-shell temperature limit of 150F the RCS pressure will be around 400 to 450 psig and the tube leak rate will be lowered.

The procedure shown in Figure 31a with a generator that can hold pressure, is to add water to the generator when the first stage of cooldown is completed (i.e., at 500F RCS T_{av}). The water level should be gradually increased to a high level (around 50% on the operate range) so the lower portion of the shell is cooled by water. Steam will be created in the generator and will help cool the upper shell. Main feedwater through the lower nozzles is preferred, but MFW or EFW through the upper nozzles will also be adequate; both must be controlled to prevent overcooling. The cooldown after 150F emergency tube-to-shell limit is reached will be about 20F/hr.

The procedure to be used when steam pressure cannot be maintained is shown in Figure 31b. The RCS should be cooled down at 100F/hr while slowly adding main feedwater (if possible) or emergency feedwater until the "emergency" tube-to-shell temperature limit is reached. When that limit is reached the cooldown rate should be slowed so as not to exceed the emergency tube-to-shell temperature limit. The rate of feedwater flow should be around 100 gpm, but actual flow rate will be dictated by the circumstances. A continuous low flow rate is desired rather than an interrupted "batch" feeding rate. Main feedwater will be difficult to control at this low flow rate and it may be necessary to use the "bypass" flow valve around the control valves.

If the reactor coolant pumps are not operating or have been shut off sometime during the cooldown sequence, natural circulation will not occur in the loop with the generator out of service. If the reactor coolant in that loop is at a high temperature when the RCS depressurization begins, it may flash to steam. The steam will collect in the candy cane and that loop will "become a pressurizer". The system pressure will "hang up" at that pressure, preventing further depressurization. In some cases not much can be done to prevent this except slowing the rate of cooldown. If possible one or more RC pumps should be run, at least periodically, to preclude steam voids collecting in the idle loop. If steam does form, and any reactor coolant pump

can be "bumped", it will help to mix the fluid so cooldown can continue. If a reactor coolant pump cannot be started, then an alternate method to stimulate circulation and cool the stagnant water can be obtained by spraying EFW on the tubes. If neither EFW nor RC pumps can be used the cooldown rate will have to be slowed. These actions should be performed only for situations, such as tube ruptures, where it is necessary to continue the cooldown. If continued cooldown isn't required, the plant should be held stable with continued natural circulation in the one loop to remove decay heat until the voids in the idle loop are collapsed due to losses to ambient and/or HPI or until the idle steam generator is returned to service (i.e., the problem requiring single steam generator cooldown is corrected).

Use of High Point Vents During Inadequate Core Cooling

Following a loss of coolant accident (LOCA), it is necessary to remove the decay heat from the core to prevent cladding damage. Core heat removal is accomplished by supplying cooling water to the core. If the supply of cooling water to the core is decreased or interrupted, a lower mixture level in the core will result. If the upper region of the core uncovers, cooling for these regions will be by forced convection to superheat steam, which is a low heat transfer regime. Continued operation in this mode may result in elevated fuel cladding temperatures with subsequent core damage and possible hydrogen generation due to Zr - H_2O reaction.

During inadequate core cooling, significant hydrogen generation due to metal-water reaction begins when a cladding temperature of 1800F is attained. Therefore, if during a small LOCA, the operator has indications that the fuel cladding temperature is at or above 1400F (Figure 29), he should open the high point vents at the hot leg and at the vessel head. This is a precautionary action to prevent noncondensible gases, which are being formed in the core, from accumulating within the steam generator tubes.

The steam generators are expected to be utilized within the inadequate core cooling procedure in order to depressurize the primary system and lead to subsequent actuation of the core flooding and/or low pressure injection systems. Therefore, concentration of noncondensible gases within the steam generator tubes should be minimized in order not to degrade the steam generator heat removal process.

Opening the high point vents is also somewhat beneficial in relieving the RCS pressure and allowing a greater safety injection flowrate, but this is a limited effect due to the vent's small size.

When the subcooled margin is regained or the RCS depressurizes to the point where the LPI system can be operated, then noncondensible production due to core damage has ceased. The operator can then close the high point vents and continue a "normal" shall break cooldown.

Loss of LPI Recirculation Ability

The ability to recirculate water from the reactor building emergency sump can be lost during recirculation or when trying to start recirculation. If this ability is lost, the cause should be determined and flow started as soon as possible. There are two general causes, 1) loss of sump water, and 2) loss of both flow paths from the sump. Loss of sump water can occur because of:

A. A steam generator tube rupture - A steam generator tube rupture is a LOCA; however, the water which leaks from the RCS does not accumulate in the reactor building sump. Instead, it leaks out of the reactor building via the secondary system. When the BWST is depleted and time has come to transfer to the RB sump, there will be no sump water. This transient must be mitigated as explained elsewhere in these guidelines to prevent the loss of the reactor building sump water.

B. Diluted sump water - Diluted sump water caused by a leak from a non-borated water source such as the LPSW system or the feed-water system will make the sump water unusable if the boron concentration becomes too low. The diluted water should not be added to the reactor core because it could allow the reactor to go critical. This would make the core cooling problem worse. This problem must be corrected by adding borated water to the recirculation flow and by stopping the dilution process, e.g., if a feedwater line is adding water to the reactor building, the flow to the bad generator should be stopped.

Both sump flow paths can be lost if both the RB emergency sump inlets become clogged or if both the sump valves fail to open. If both of the sump inlets becomes clogged, it may be possible to back flush the line to clear away any debris that might be covering at least one of the sump line inlets. If both the sump motor operated valves fail to operate remotely, then local manual operation of the valves should be

attempted to open at least one of the valves. However, the problem may not be electrical, e.g., the valve stem or disk may be binding, and manual operation of the hand wheel may not work, but some kind of mechanical leverage may be able to open the valves. Local attempts to open these valves may not be possible because of high radiation levels.

If the cause for a loss of sump water or flow path cannot be corrected, the operator should attempt to cool the reactor core with the DHRS. This method of cooling will be successful if the cooling water can flow through the RCS without leaking to the reactor building. To accomplish this, the RCS water must be subcooled to prevent steaming out the break and the RCS water level must be below the break elevation to stop RCS water from continually leaking to the reactor building and high enough to prevent a vortex formation as water is drawn into the DHRS suction pipe (this would be the same elevation as required for normal DHRS operation when draining the RCS). The RCS water is brought subcooled by increasing the LPI cooling as much as possible. When the RCS becomes subcooled, DHRS operation can be initiated and the RCS water level will inherently drop to the break location. If the the break location is high enough vortex formation in the DHRS drop line will not occur.

However, if the break elevation is too low then a vortex will form in the DHRS letdown line and LPI cooling will need to be continued. Vortex formation can be detected by noting LPI pump cavitation.

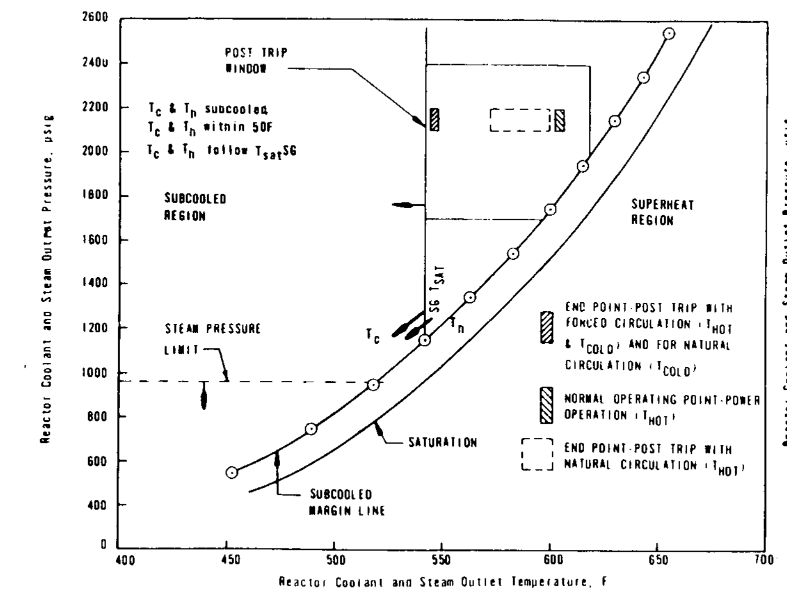
Because of the possibility of vortex formation the transition from LPI to DHRS cooling should be made cautiously by switching one LPI train at a time from LPI to DHRS mode of operation.

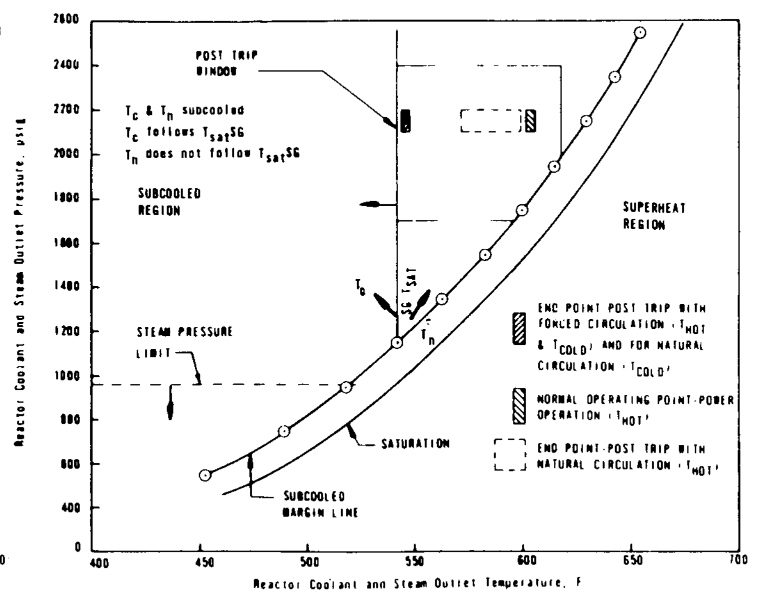
During LPI cooling water will continually be lost out the break until the water level in the reactor building increases to the elevation equal to the water level required in the RCS for DHRS operation. This will require more water than that contained in the BWST. Therefore, sources of borated water in addition to the BWST water will need to be used for core cooling. These additional sources will need to be pumped to the reactor vessel for core cooling until the sump recirculation can be established or the RB is flooded to the RCS level needed for DHRS operation so that the DHRS can be put into operation. If the RB needs to be flooded, the operator should prepare for equipment and instrument failures due to water submergence. For example, the DHRS suction line valves should be opened before they are submerged because after they are submerged they may not operate.

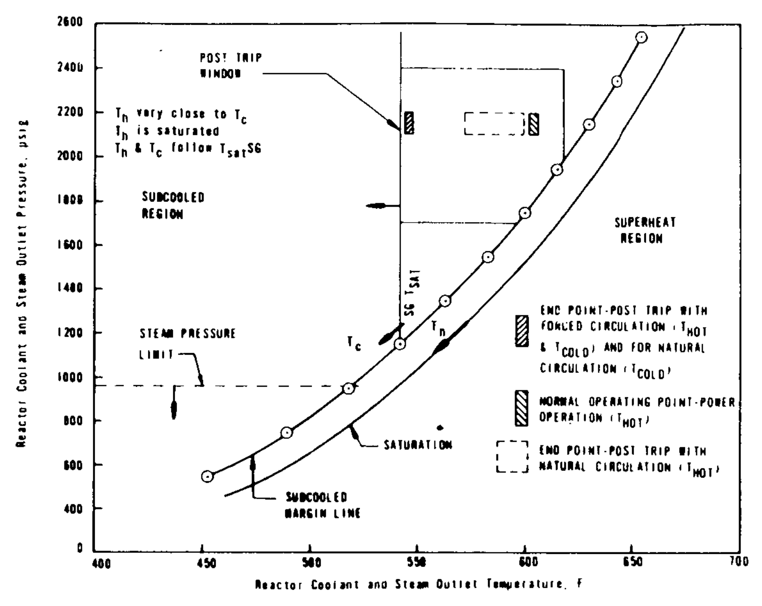
If available, the pressurizer level indication should be used as a visual aid to assure an adequate water level is over the DHRS suction nozzle. Water may need to be added periodically to makeup for leakage from the RCS and DHRS.

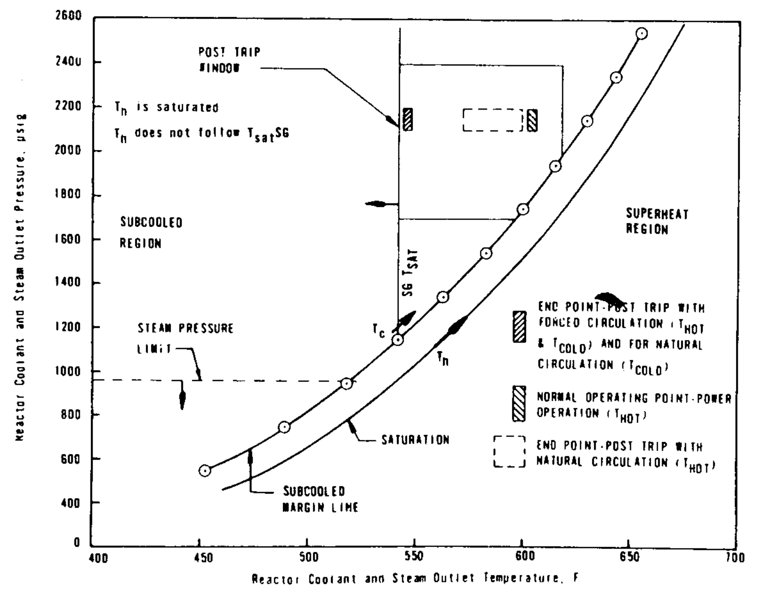
During DHRS operation the RCS water temperature should be held constant to prevent volumetric water changes which cause the RCS water level to fluctuate and the RCS water temperature should be held below the boiling point to prevent water loss by steam flowing out the break.

Table 6 RC PUMP RESTART GUIDELINES





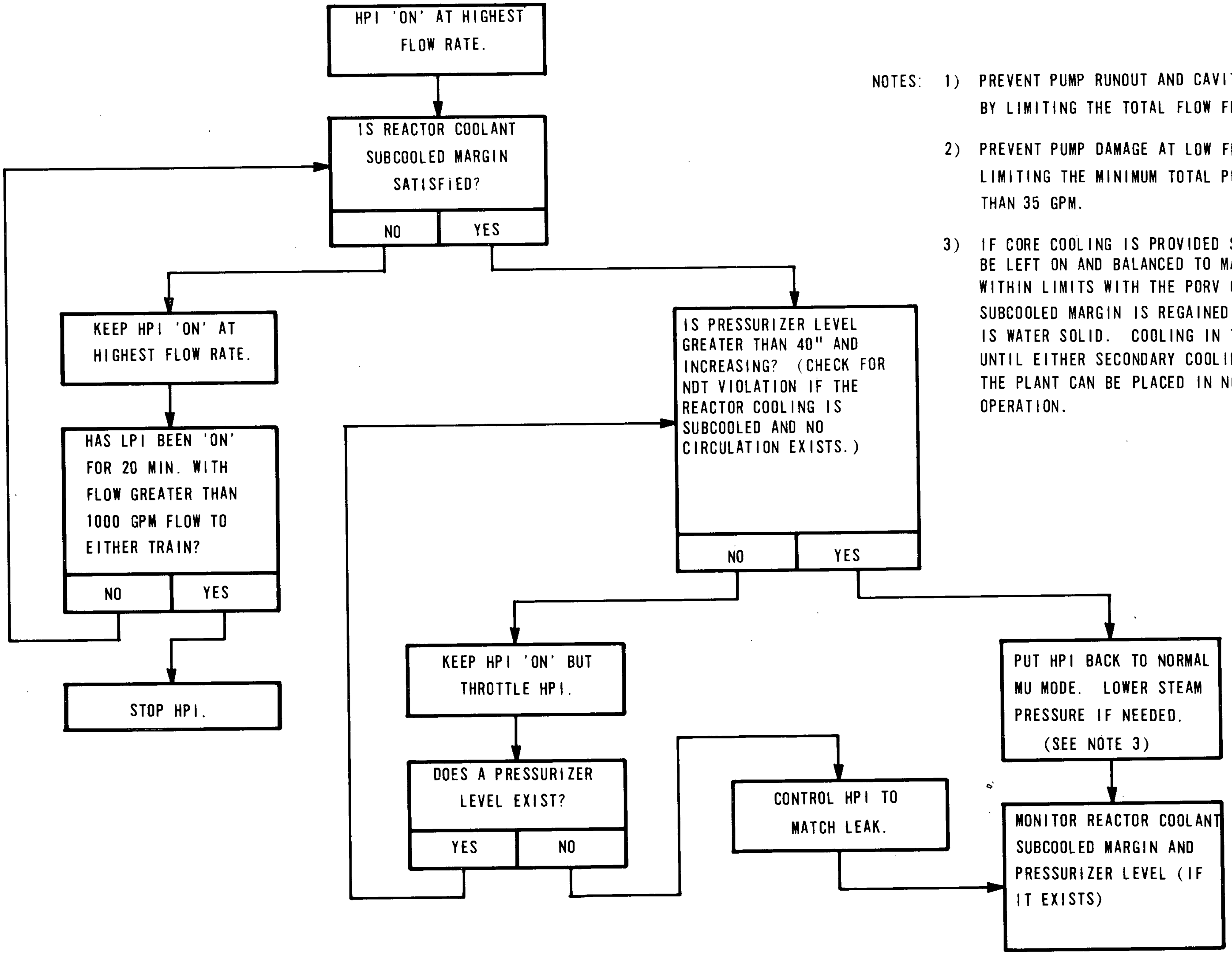




(INADEQUATE CORE COOLING)

REACTOR COOLANT CONDITION	SUBCOOLED WITH NATURAL CIRCULATION EXISTING	SUBCOOLED WITH NO NATURAL CIRCULATION	SATURATED WITH NATURAL CIRCULATION EXISTING (HPI ON)	SATURATED WITH NO NATURAL CIRCULATION (HPI ON)	SUPERHEATED WITH CLAD TEMPERATURE GREATER THAN 1400F (CURVE 1 OF FIGURE 29)	SUPERHEATED WITH CLAD TEMPERATURE GREATER THAN 1800F (CURVE 2 OF FIGURE 29)
RESTART ACTIONS	• CONFIRM THAT NO RC PUMP DAMAGE WILL OCCUR. • CONFIRM THAT AT LEAST ONE STEAM GENERATOR IS A HEAT SINK. • START AND RUN THE RC PUMPS IN THE LOOP WITH THE GOOD STEAM GENERATOR.	• CONFIRM THAT NO RC PUMP DAMAGE WILL OCCUR • ESTABLISH AT LEAST ONE STEAM GENERATOR AS A HEAT SINK (T_{SAT} OF STEAM GENERATOR IS AT LEAST 50F COLDER THAN INCORE THERMOCOUPLES. • BUMP PUMP ONCE TO SEE IF NATURAL CIRCULATION STARTS. IF IT DOES, RESTART AND RUN. IF NOT, WAIT 15 MINUTES THEN BUMP AGAIN. SPECIAL CASE: WHEN THE PLANT IS SUBCOOLED WITH NO NATURAL CIRCULATION AND THE HPI SYSTEM IS ON, THE COLD HPI WATER MAY CAUSE BRITTLE FRACTURE OF THE REACTOR VESSEL. ONE RC PUMP MAY BE STARTED TO CAUSE FLUID MIXING. STEAM GENERATOR COOLING IS NOT REQUIRED.	• DO NOT RESTART. • PROCEED WITH RCS COOLDOWN BY GRADUALLY LOWERING STEAM PRESSURE AND LOWERING RCS PRESSURE WITH THE PORV.	• CONFIRM THAT NO RC PUMP DAMAGE WILL OCCUR. • ESTABLISH AT LEAST ONE STEAM GENERATOR AS A HEAT SINK (T_{SAT} OF STEAM GENERATOR IS AT LEAST 50F COLDER THAN INCORE THERMOCOUPLES). • BUMP PUMP ONCE TO SEE IF NATURAL CIRCULATION STARTS. IF IT DOES, PROCEED WITH RCS COOL-DOWN BY GRADUALLY LOWERING STEAM PRESSURE AND LOWERING RCS PRESSURE WITH THE PORV. IF IT DOES NOT START, BUMP ALTERNATE PUMPS (ONE EVERY 15 MINUTES) UNTIL ALL FOUR HAVE BEEN BUMPED. IF NATURAL CIRCULATION HAS STARTED, PROCEED WITH COOLDOWN. IF NOT, LOWER STEAM PRESSURE UNTIL THE STEAM TEM-PERATURE IS 100F COLDER THAN THE INCORE THERMOCOUPLE TEMPERATURE. RUN THE RC PUMPS AND CONTINUE COOLDOWN AND DEPRESSURIZATION.	IF RC PUMP PROTECTIVE INTERLOCKS PERMIT, START AND RUN ONE PUMP IN EACH LOOP.	START AND RUN ALL RC PUMPS EVEN IF PUMP DAMAGE CAN OCCUR.
	SPECIAL PRECAUTION: IF SUBCOOLING MARGIN IS LOST IMMEDIATELY AFTER PUMP RESTART AND DOES NOT RETURN IN ABOUT 2 OR 3 MINUTES, THE RC PUMPS MUST BE TRIPPED AND NOT RESTARTED UNTIL THE SUBCOOLING MARGIN IS REGAINED.					

Figure 30 HPI CONTROL LOGIC



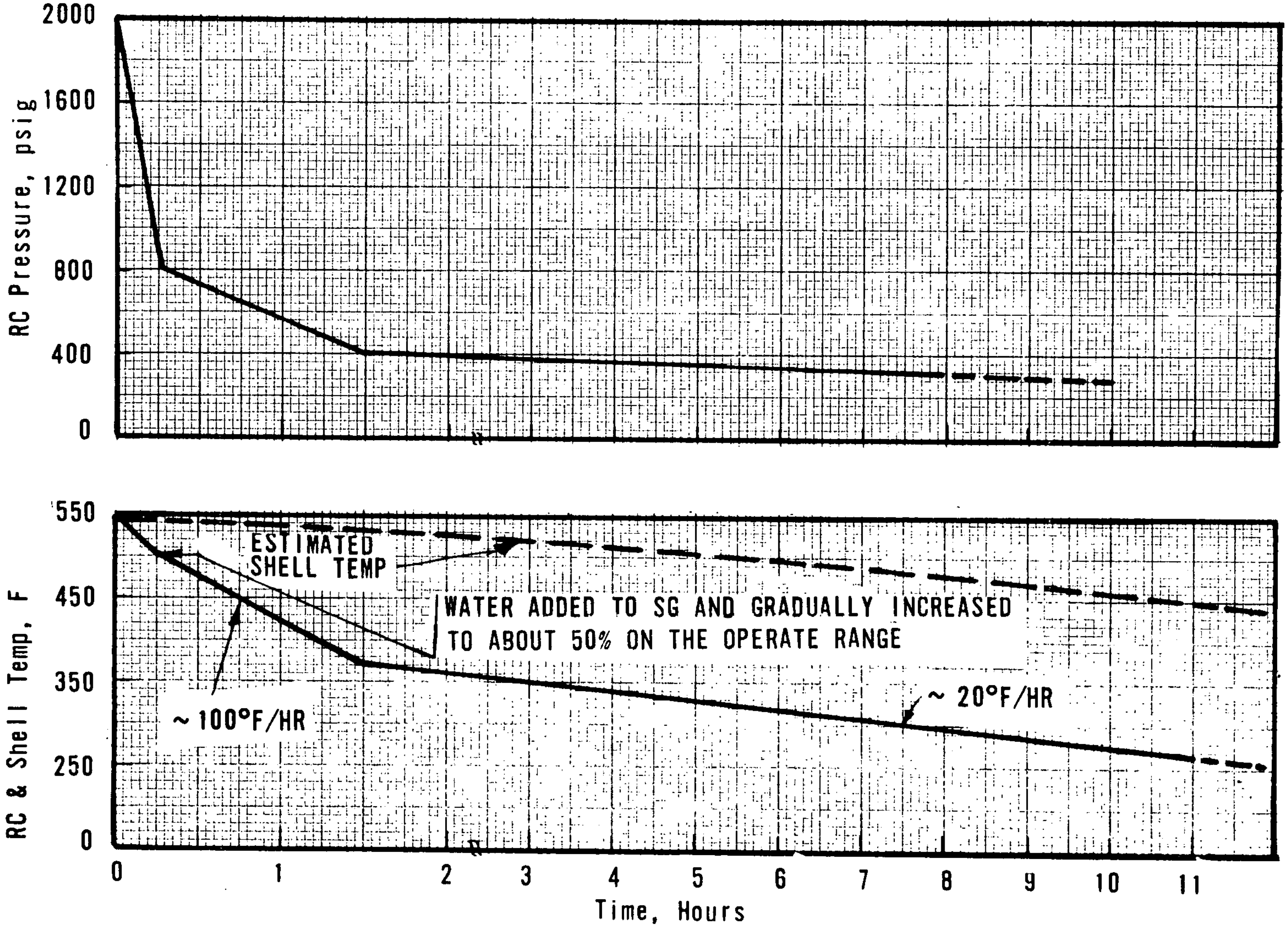
NOTES:

1) PREVENT PUMP RUNOUT AND CAVITATION DAMAGE BY LIMITING THE TOTAL FLOW FROM ONE PUMP TO 550 GPM.

2) PREVENT PUMP DAMAGE AT LOW FLOW CONDITIONS BY LIMITING THE MINIMUM TOTAL PUMP FLOW TO MORE THAN 35 GPM.

3) IF CORE COOLING IS PROVIDED SOLEY BY HPI, HPI MUST BE LEFT ON AND BALANCED TO MAINTAIN RCS PRESSURE WITHIN LIMITS WITH THE PORV OPEN EVEN AFTER THE SUBCOOLED MARGIN IS REGAINED AND THE PRESSURIZER IS WATER SOLID. COOLING IN THIS MODE MUST CONTINUE UNTIL EITHER SECONDARY COOLING IS REESTABLISHED OR THE PLANT CAN BE PLACED IN NORMAL DECAY HEAT REMOVAL OPERATION.

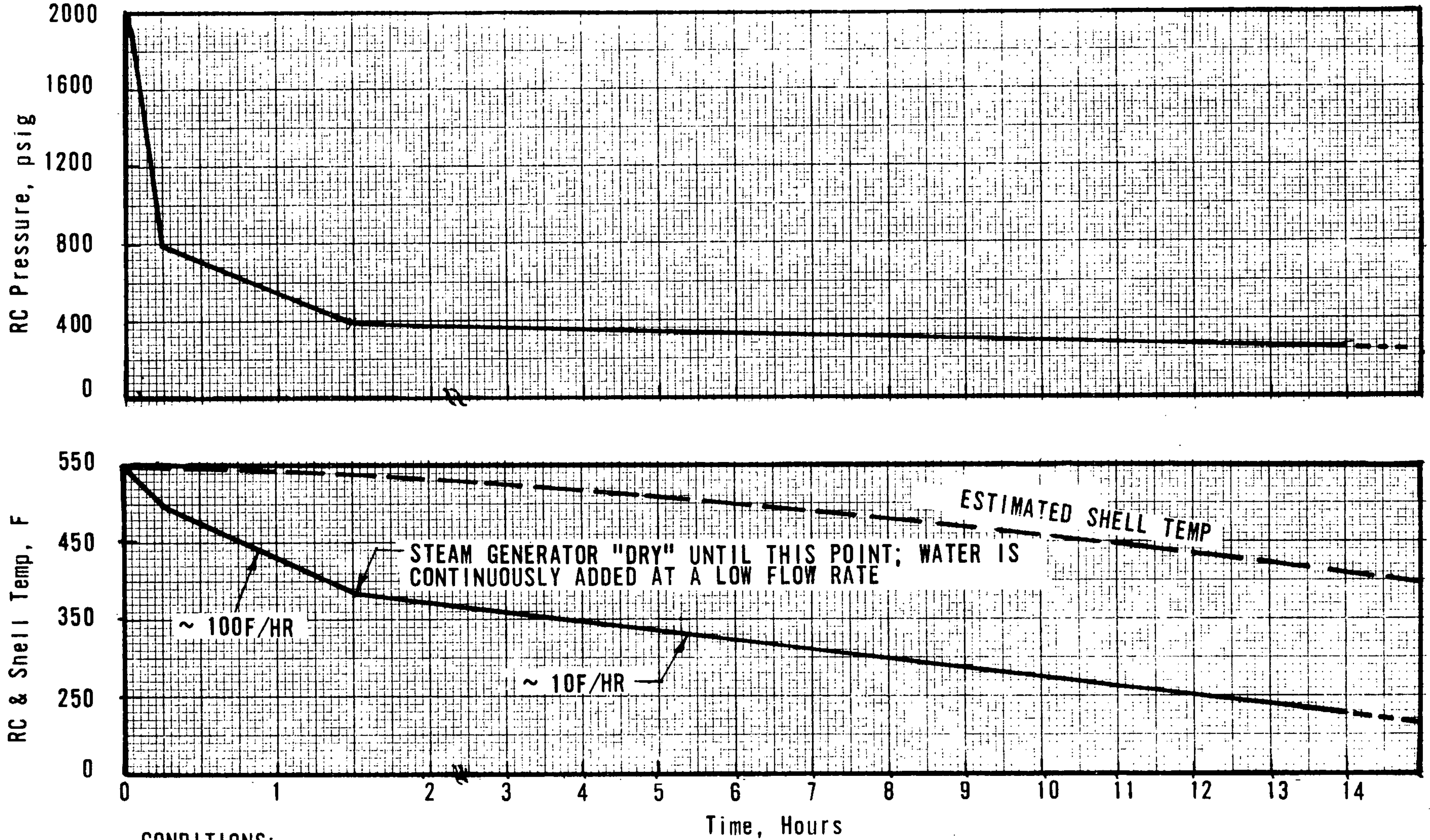
Figure 31a COOLDOWN ON ONE STEAM GENERATOR (STEAM PRESSURE CONTROLLED)



CONDITIONS:

- RC PUMPS ON OR OFF (PROBABLY ON)
- STEAM GENERATOR "ISOLATED" BUT HAS INVENTORY OF WATER ADDED AFTER THE INITIAL DEPRESSURIZATION TO 500°F
- STEAM PRESSURE MAY SLOWLY DECREASE DUE TO LEAKS AS COOLDOWN PROCEEDS IF IT CANNOT BE VENTED

Figure 31b COOLDOWN ON ONE STEAM GENERATOR (STEAM PRESSURE NOT CONTROLLED)



CONDITIONS:

- RC PUMPS ON OR OFF (PROBABLY ON)
- STEAM LEAK IN GENERATOR; STEAM PRESSURE IS AMBIENT
- SLOW FEEDING OF GENERATOR BEGINS AT ~ 1 1/2 HRS - IT IS DOUBTFUL THAT LEVEL WILL BUILD FOR SEVERAL HOURS: LOWER SHELL WILL NOT BE COVERED FOR SOME TIME; SHELL COOLING IS BY STEAM CONDENSATION

CHAPTER F

POST TRANSIENT STABILITY DETERMINATION

To determine if the transient has been brought under control four general areas must be checked.

1. Reactivity Control - The reactor must have a subcritical margin of at least 1% $\Delta k/k$.
2. Core Heat Removal Control - The core must be covered and cooled; the heat removal rate is equal to or slightly greater than the core heat generation rate.
3. Radiation Release Control - Release to offsite is terminated.
4. Plant Equipment is Operating Correctly - Equipment to maintain the plant safe and stable is operating and within design duty; equipment failures have been bypassed, isolated or repaired.

Several things around the plant must be checked to make sure these four general rules are being met. The following basic check list defines the more important items. The list is divided into two cases. Case I applies to LOCA's which can be stopped by complete isolation of the leak and to all other transients. Case II applies to LOCA's which cannot be isolated. The difference between the two parts is simple: a reactor leak that cannot be stopped is a transient that cannot be positively terminated. However, a leak can be reduced to the smallest amount possible and become stable for "long-term cooling". Steam generator heat removal can be used for some small leaks but HPI must be kept running to keep the reactor coolant subcooled. Subcooling can be regained for some very small break

sizes at a time when the decay heat decreases and HPI is able to refill the RCS loops and add water to the pressurizer.

Case I - All transients (including LOCA's which can be isolated)

1. Reactor coolant pressure and temperature are preferably within the "post-trip window" of the P-T diagram; however, pressure and temperature may be anywhere on the P-T diagram within a region bounded by: a) NDT limits, b) the subcooling margin, c) an RC pressure upper limit equal to the PORV setpoint minus 100 psi, d) fuel pin compression limits and e) RCP NPSH requirements, if applicable. Subcooling will exist in the hot and cold legs of both loops.
2. The "long term" trend of reactor coolant pressure and temperature is constant or slowly decreasing with time. "Short-term" fluctuations of temperature and pressure are small and can be attributed to periodic operations of other equipment (pressurizer heaters, spray, or feedwater).
3. Pressurizer level is within the indicated range.
4. If forced circulation exists (RC pumps on) then reactor coolant T_{av} is about equal to the saturation temperature of the water in the steam generator (or generators) that is removing the heat.

5. If natural circulation exists, $T_{cold\ leg}$ will be about equal to the saturation temperature of the water in the steam generator (or generators) that is removing the heat. The difference between incore thermocouples and T_{hot} in the operating loop (or loops) will track within 10F. If only one generator is removing heat the other reactor loop will be subcooled.
6. Steam generator (or generators) level will be at the correct setpoint (either natural or forced circulation setpoint) and will be steady.
7. Steam generator pressure is steady and is below the safety valve opening setpoint.
8. The core is at least 1% Δ k/k subcritical on rods and boron. If more than one rod did not fully insert the core is at 1% Δ k/k subcritical on boron alone (assuming all rods out).
9. If the accident caused water to enter the containment (reactor or steam generator) and the containment environment was increased, it will now be reduced to near normal levels. Pressure will be close to barometric pressure; average containment temperature will be near prior operating temperature; relative humidity will be about 100%.
10. If radioactive water leaks occurred in auxiliary buildings those areas will be sealed and the spillage either trapped or drained to storage tanks.

11. The component failure (or failures) which caused the transient is known. It has been bypassed, isolated, repaired, or otherwise handled so that it no longer compromises plant safety.

12. Components which support plant safety are operating near their design point (examples: pumps are operating away from the minimum shutoff flow and have adequate NPSH, throttle valves are near the proper opening, electric motors are in the normal service range, electronic equipment is environmentally protected). If a component is operating off design and future failure is possible, then redundant or alternate equipment is on standby and ready to replace the equipment which might fail.

13. Stored water (condensate tank, BWST) is adequate for long term use or alternates are readily available.

14. Instrumentation to monitor plant performance is operating correctly. Potential failures of critical instrumentation have been identified and alternate instrumentation is available.

Case II - LOCA's which cannot be isolated

NOTE: With the exception of steam generator tube leaks, all reactor coolant leaks outside the containment can be isolated. Although a tube leak is "inside" the containment a direct path outside the containment exists through the steam lines.

Many of the criteria of Case I apply to this part except that the reactor coolant will not always regain the subcooled margin and operating conditions that depend on subcooling will not apply. The very smallest reactor coolant leaks may allow the reactor coolant system to repressurize (because of continued High Pressure Injection) and some amount of subcooling may be regained, but it is not likely that enough margin will occur. Consequently, the criteria for LOCA stability does not include the subcooling margin. Also because subcooling may not exist the hot legs may have steam binding and natural circulation may not exist; therefore, the criteria do not include natural circulation requirements (however, it can exist for very small breaks and could be checked). A reactor-steam generator heat transfer balance cannot usually be accomplished because of saturated (or near saturated) conditions which may not permit the reactor coolant to move the heat from the core to the steam generator, but some heat transfer to the steam generator is possible for small breaks. The steam generator operating level should be at the 95% level for small breaks to permit condensation of primary side steam. Pressurizer level cannot be relied upon if saturation exists.

The most important criterion for LOCA is to keep the core covered. This condition is confirmed by readings of the incore thermocouples and the hot leg RTD's; both should show that the reactor coolant is saturated (or even subcooled) but not superheated.

The continued loss of coolant from a LOCA will not permit the accident to be truly terminated, but the leak rate can be minimized. Lowering RCS pressure is the best way to lower the leak rate. This can be done by loss through the leak, by opening the PORV, or by lowering secondary side pressure. Long term loss of coolant when the RCS is depressurized occurs in two ways: 1) steaming out of the leak because of continued boiling, and 2) water loss because the head of water is above the break and water will "run" out of it. The rate of leak will depend on the system pressure, the decay heat level (which causes boiling), and the elevation of the leak (a leak high in the system will have a lower flow rate than a leak low in the system). The leak rate will also depend on the hole size.

The criteria for stability is that the leak rate is as low as possible and that the flow into the core keeps it covered. It may take a very long time to recover from some LOCA's and during that time there will be two general stages when the leak rate diminishes. The first stage is when the reactor coolant system is depressurized to atmospheric pressure (big breaks will depressurize rapidly, smaller breaks will take longer); the second stage is when the core heat drops so that it cannot boil the water in the reactor vessel. Steaming will stop at that time (which may be as long as several months after the accident). Until the water in the vessel becomes subcooled (incore thermocouples will read less than 212F), the plant must be operated by injecting containment sump water in the recirculation mode or by

continuing to inject fresh borated water from other sources. When the vessel water becomes subcooled the operator has the option to transfer one train of LPI to the decay heat removal mode and keeping the other train on sump recirculation. The reason one train is left on recirculation is that it will keep water above the hot leg suction for decay heat removal. Decay heat removal has the advantage of rapid RCS cooldown, but it must be carefully monitored to make sure the decay heat pump does not lose suction (or it will fail), and to make sure the decay heat pump does not run at shut-off head.

Because the leak may continue a long time until the decay heat system is engaged, an arbitrary definition of stability is given. The following criteria define post-LOCA long term stability:

1. The core is covered. Incore thermocouple readings show saturated or subcooled reactor coolant.
2. ECCS injection is in the "long term cooling" mode. Long term cooling exists when the ECCS is operating with recirculation from the containment emergency sump. (NOTE: A decision may have been made not to transfer but to bring in backup water to refill the BWST. Nevertheless, if recirculation could have been started, "long term cooling" is considered to have started).
3. The reactor coolant system is depressurized to near atmospheric pressure so that the leak rate is as low as possible. The LPI system is used to cool the core. (NOTE:

If the break size did not permit depressurization before the BWST was empty, and HPI "piggyback" recirculation had to be used while further depressurization took place the plant is not considered to be stable until the pressure and leak rate are as low as possible).

4. Steam generator level is at 95% on the operate range and is steady.

5. Reactor coolant pumps are off (operation of RC pumps could move water past the break and increase the leak rate).

6. The following criteria from the previous part also apply: Numbers 7, 8, 9, 10, 11, 12, 13, 14.

7. For the special case of steam generator tube leaks (LOCA's):

   a) Feedwater (main and emergency) has been stopped to the bad generator.

   b) Steam created by boiling the RCS leakage is directed to the condenser (if it is operating).

   c) The plant is on decay heat removal or standby backup borated water sources are available to replenish BWST inventory.

CHAPTER G

FUNDAMENTALS OF REACTOR BUILDING CONTROL

1.0 Introduction

The Three Mile Island Accident provided some very significant insight to reactor building performance. Two distinctly separate conditions existed at TMI-2 that lead to a conclusion that a more comprehensive approach for reactor building control to prevent radiation release is very important:

- The reactor building pressure stayed very low for a significantly long time before reaching 4 psig when automatic reactor building isolation of penetrations could occur. During this time period the core was uncovered and very high concentrations of radioactive material were released to the reactor building. Fortunately, the reactor building penetrations that were open did not permit a direct release of reactor building gases to the site; unfortunately the normal sump line was open and radioactive liquid was pumped to the auxiliary building and indirect releases did occur.

This TMI situation illustrates the need for operator action to control radiation releases by controlling the reactor building boundary

at all times. Operator action prior to automatic system actuation may be appropriate; operator action may also be needed for conditions when systems used for automatic reactor building isolation control fail and the automatic isolation requirement exists.

- Another condition that existed at TMI was an accumulation hydrogen in the reactor coolant system. Hydrogen burning can be very dangerous and can cause damage to the reactor building or to the equipment in the reactor building.

This situation illustrates the need to control the reactor building environment to preserve the integrity of the reactor building and its operating equipment. Operator action is required to control the reactor building environment (radiation, pressure and temperature, H_2, and other physical phenomena) so that:

- Radiation release from the reactor building to the site surroundings is limited
- Radiation release to the auxiliary building atmosphere or fluid systems is limited or controlled acceptably
- Systems or components inside the reactor building that are used for reactor building control or for core cooling are protected.

The main purpose of the reactor building is to prevent unacceptable releases to the environment of radioactive material leaking from the reactor coolant system. Although the reactor building serves this purpose during normal operation and abnormal transients, this discussion will be limited to abnormal transients. Proper reactor building control during abnormal transients, especially during a Loss of Coolant Accident, will limit the amount of radioactive material leaking to the environment. Some reactor building controls are automatic and some are manual. However, the operator must be prepared to take manual action if the automatic controls fail.

2.0 DIAGNOSIS AND CONTROL

The two basic objectives of reactor building control are: control reactor building isolation and to control the reactor building environment. This section discusses how these objectives are reached by diagnosing and treating reactor building problems.

This chapter will first discuss the general approach to diagnosis and control of the reactor building, including backup actions if things are not responding as they should. (Section 2.1. and 2.2)

Then the following sections (2.3 and 2.4), will be more specific in describing what parameters to monitor, when to take action, and what the action should be.

2.1 General Approach For Diagnosis and Control

Figure 40 illustrates the general approach to be used for reactor building control following an abnormal transient.

The following is a brief discussion of Figure 40.

Abnormal Transients Which Change the Reactor Building Internal Environment.

The first block on Figure 40 refers to the abnormal transients which change the reactor building environment.

Reactor building control is required at all times, especially for reactor building conditions following a LOCA or steam/feedline break. Of these events the LOCA imposes the most difficult problems; a LOCA can result in releases of radioactive material, H_2, and acidic water (boric acid) but a steam or feed line break can only release steam and water (if no SG tube rupture). However, small amounts of radiation existing in the reactor building prior to the secondary system breaks can be released; the effects are expected to be negligible (unless a steamline break causes a SG tube leak). Either event can cause equipment damage and require repair prior to placing the plant back into service.

Following the ATOG instructions, the operator maintains adequate core cooling and then ensures RB control. Adequately cooling the core will limit the H_2 and radiation levels in the reactor building. Loss of core cooling (ICC) will aggravate the ability to control the

reactor building and can require more drastic actions than would otherwise be required. In fact, if ICC occurs the emphasis on reactor building control will become greater because of the greater potential of radiation release and the necessity for using the reactor building for subsequent core cooling.

<u>First Line of Action - Limit Radiation Release by Assuring the Reactor Building Leak Paths are Sealed</u>

The second block shows the first line of defense against radiation release. Attention should be devoted to isolation of leak paths that:

- should be normally closed but may not be (for example; the reactor building purge line
- could be isolated early by the operator before the setpoints are reached which initiate automatic actions, e.g., reactor building normal sump
- paths that can transfer liquid reactor coolant to the auxiliary building.

Conversely, paths that should remain open are those needed for core cooling or reactor building control; for example, cooling water should be available for operating reactor coolant pumps.

The need to take action for reactor building isolation is based on the following symptoms:

- High reactor building pressure
- Reactor coolant system pressure (indicates a LOCA)
- High reactor building radiation

The principal reason for use of radiation as a symptom is that it is possible to have high radiation without high building pressure or low reactor coolant pressure.

Confirmation that isolation is effective is provided by valve operator lights.

In addition to isolating the reactor building penetrations, the penetration ventilation system will need to be started on high reactor building pressure. This function is for controlled removal of any leakage into the penetration room via the penetrations. If a high reactor building pressure is not reached the penetration room ventilation system should not be started because it causes a vacuum in the penetration room which will draw the radioactive gases out of the reactor building.

Second Line of Action - Control Reactor Building Internal Environment to Protect the Reactor Building

The next line of defense is preservation of the reactor building integrity. This requires preventing the pressure from exceeding the reactor building design pressure. Pressure can be high within the containment for two reasons:

- the mass and energy from the break increases the pressure.
- hydrogen burning will increase the pressure.

To reduce pressure caused by mass and energy, building coolers and sprays should be used. Purging to remove hydrogen is acceptable provided the reactor building radiation level is low; as an alternate, hydrogen flammability can be surpressed by containment sprays.

Third Line of Action - Eliminate or Reduce the Cause for the Change in the Reactor Building Internal Environment

The next line of defense is to eliminate or reduce the cause of the change in the reactor building environment to:

- minimize the continued radiation release to the reactor building
- minimize the continued mass and energy release
- minimize hydrogen release to the reactor building

To minimize the amount of radiation and hydrogen released to the reactor building continued core cooling is required; and, if possible, the leak to the reactor building should be isolated. The symptoms for core cooling are discussed elsewhere in these guidelines.

Increased core cooling via the secondary system or LPI will reduce the amount of energy released to the reactor building.

If a steam or feedline break has occurred, isolating feedwater to the bad SG will also stop mass and energy release to the reactor building. This is performed in Part I followup actions for excessive heat transfer.

Fourth Line of Action - Control Reactor Building Internal Environment to Protect RCS Heat Removal Equipment and Instrumentation and Reactor Building Environment Control Equipment and Instrumentation

The fourth line of defense is to limit radiation release by protecting the reactor building control equipment and protecting the RCS heat removal equipment. This is done by a variety of activities, including continuing the actions to control radiation, building pressure, temperature, hydrogen, and controlling reactor building isolation valves on lines thst supply services to equipment in the containment (i.e., RC pumps and letdown cooler cooling water). Some additional actions are needed:

- Control sump water level to prevent flooding of useful equipment (such as decay heat drop line valves, instrumentation connections, RC pumps, RC ventilation ducts)
- Control sump water level to ensure that LOCA recirculation can occur
- Control sump water chemistry (acid) to limit corrosive effects, especially stress corrosion
- Control building spray into the reactor building to avoid unnecessary contact with operating equipment.

Control of the sump water level is important, but must be viewed in perspective. There must be a sufficient height to permit ECCS recirculation, but the upper limit is somewhat dependent on the post accident situation. An upper limit is used to prevent flooding of equipment that could be used to mitigate the accident; for example the decay heat drop line valves can be used to establish decay heat removal. But if the post accident sump water is very radioactive it may be better to avoid recirculating fluid. In this case the "upper limit" can be higher than the elevation of the decay heat valves. No single "fixed" limit exists.

To limit water level accumulation the following actions should be considered (as appropriate):

- Limit or isolate unwanted leakage into the reactor building from all sources if possible; possible sources are:
- Reactor building cooler cooling water
- Component cooling water to RC pumps, letdown coolers
- Secondary side leaks
- Primary side leaks (LOCA)

Control of sump water pH is performed by adding NAOH. The symptom to monitor is pH. The acceptable pH range is between 7.0 and 8.0.

Reduction of the amount of radioactive material in the reactor building atmosphere limits the amount of radioactive material that can be released offsite. Cooling and condensing steam with the building coolers will help this, and if very high radiation levels are reached sprays can be used. Both actions will deposit much of the particulate matter and iodine in the sump; noble gases will not be affected significantly.

2.2 General Back-Up Actions to Limit Radiation Release

Two conditions can exist for which special back-up actions are required:

- Loss of reactor building integrity
- Loss of equipment needed for reactor building control

Reactor building integrity can be lost for several general reasons:

- Failure to isolate penetrations that permit a direct link between the reactor building atmosphere and the site (air to air penetrations).
- Failure to isolate penetrations that permit a transfer of radioactive liquid from the reactor building to the auxiliary building and then to the site.
- Failure of the reactor building structure, although highly improbable, could be caused by high pressure due to mass and energy release, H_2 burning, external or internal missiles, or even extensive core melting (Note: This failure is no covered in this discussion; national studies in this area have not been completed and little information is known).

Generally, the single best symptom to determine reactor building integrity is offsite radiation levels. General actions to take for radiation release control are:

- Isolate penetration if possible
- Reduce reactor building pressure as fast as possible to reduce the driving force for leakage; use of reactor building sprays to remove radioactivity and pressure is required.

2.3.1 Diagnosis

Specific parameters are monitored that will indicate the potential for excessive radiation release from the reactor building. These parameters are:

- Reactor building radiation
- Reactor building pressure
- RC pressure

Each of the three conditions listed above relate to a potential release of radioactive material as follows:

1. High radiation - if the radiation level in the reactor building is too high then any gas or liquid leaking from the reactor building may contain excessive radiation per volume of leakage. Therefore, a small amount of leakage can give a large release of radioactive material.

2. High reactor building pressure - the reactor building pressure is the driving force which pushes the radioactive gases and liquids out of the reactor building. Consequently, as the reactor building pressure increases, the leak rate of radioactive materials will tend to increase.

3. Low reactor coolant pressure - a low reactor coolant pressure indicates a possible LOCA. The LOCA is the one abnormal transient which, as it releases a large amount of energy and radiation to the reactor building, can cause both a high reactor building radiation condition and a high reactor building pressure condition. This combination of high radiation and high pressure can result in an excessive radiation release.

If these parameters exceed the values specified in Table 7, the potential for too much radiation release exists and control actions should be made.

2.3.2 Isolate Penetrations of Non-Safety Fluid Systems Used in Power Operation

Following an abnormal transient which releases radioactive material into the reactor building, all reactor building penetrations (except those needed for safety functions) should be isolated or sealed to attenuate leakage of radioactive materials from the reactor building. These penetrations include fluid system piping, electrical wires and access openings. Most of these penetrations are isolated or sealed before going to power operation. The only penetrations not isolated or sealed are those for non-safety fluid systems used in power operation

(e.g. reactor coolant letdown) or for safety fluid systems (e.g. cooling water to the reactor building coolers). Except for the penetrations of safety fluid systems, these penetrations need to be isolated whenever the potential exists for leaking radioactive material from the reactor building.

The penetrations of non-safety fluid systems used in power operations (listed in Table 9) should be isolated if any of the three control parameters exceed their limit (with the exception of two penetrations to be discussed later).

The reactor building isolation system monitors the reactor building pressure and reactor coolant system pressure. It does not monitor reactor building radiation. The isolation system will automatically isolate the penetrations of non-safety fluid systems if the high reactor building pressure or low reactor coolant system pressure limits are reached. If the high reactor building radiation limit is reached (without high building pressure or low RCS pressure) or the isolation system fails to operate automatically, the operator must manually initiate penetration isolation.

As stated earlier two penetrations are treated differently. These are the penetrations for the component cooling water system and the low pressure service water to the reactor coolant

pump motors. These are isolated automatically only on high reactor building pressure. These two systems provide cooling water to the RC pumps motors and the letdown cooler. This equipment is important in the mitigation of abnormal transients. Therefore, this equipment should remain operational as long as possible cooling water should be re-established unless it is known that a large LOCA has occurred, in which case RCP operation will not be required.

These two systems do not pose a path for radiation release from the reactor building space; they do not carry radioactive fluid and they operate at a pressure which is higher than the pressure the reactor building would reach during an abnormal transient. Consequently, radiation would not be expected to flow from the reactor building to the the fluid system and out the penetration to the environment.

However, these systems could leak unborated water into the reactor building. This leakage would be undesirable for a large LOCA which depends on reactor building emergency sump water for recirculation cooling of the core, because the leaking water would go to the reactor building emergency sump and dilute the boron concentration of the water leaking from the RCS. The large LOCA releases a large amount of energy and it is possible that this energy could cause a

break in a LPSW or CC water line. Therefore, these lines should not be unisolated if a large LOCA has occurred. For smaller LOCAs, caution should be exercised in reopening the lines. The CCW surge tank level should be monitored and, if it drops abruptly, CCW should be reisolated. No surge tank or flow indications are available for the LPSW lines to the RCP air and bearing coolers. However, a rupture of these lines may cause a reduction in flow through the RB coolers which can be monitored.

The control actions discussed above are summarized in Table 7. Most of these mitigating actions are accomplished automatically by the ES system. Since it is possible that high RB radiation could occur before ES actuation on high RB pressure, operator action may be needed.

Control With Equipment Failure

If the ES system fails to auto isolate the penetrations, the operator should manually isolate the penetrations.

If an automtic isolation valve does not close, the operator should check its redundant isolation valve to see it if has automatically closed. If at least one valve has closed for each reactor building penetration, then no other immediate action is required. If both

valves on a penetration fail to close then the operator should try individual remote operation of one or both or the redundant isolation valves. If neither valve will close remote attempts should be made to locally close the isolation valve <u>outside</u> the reactor building. This may not be possible if radiation levels are too high. If the failed isolation valve can be locally reached and still cannot be closed, a determination should be made of what caused the valve <u>inside</u> the reactor building to fail open and then attempt to correct the problem and close the valve.

If neither of the redundant isolation valves closed, then follow the system out from the penetration to the next set of valves and close them.

2.3.3 Selectively Operate Penetration Isolation Valves

If all the operator wanted to do was to prevent leakage from the reactor building he would simply close all penetration isolation valves (both non-safety and safety penetrations). However, during and after abnormal transients, the operator will need to use some of the fluid systems which pass through these penetrations to maintain core cooling and to control the reactor building environment. Consequently, the operator will have to selectively operate the reactor building isolation valves as he chooses

between the need for reactor building isolation and fluid system operations. The decision to open the valves should be accompanied by a judgement of consequences (e.g. the penetration path cannot be reclosed).

The penetrations the operator needs to manipulate can be divided into two groups as follows:

a. The penetrations of non-safety fluid systems used during power operation. These are the penetrations which close automatically on high reactor building pressure. They are for the fluid systems which are not considered to perform safety functions. These are listed in Table 9.

b. The penetration of safety fluid systems.

   - DHRS suction line
   - Reactor building emergency sump
   - High pressure injection
   - Seal injection
   - Low pressure injection
   - Auxiliary feedwater
   - Main steam line
   - Main feedwater line
   - Reactor building spray
   - LPSW to reactor building coolers

Operating Penetrations of Non Safety Fluid Systems

When the penetrations for non-safety fluid systems are isolated, the operator needs to be aware of what is affected so that he can make appropriate compensation. For example: following large steam line break inside the reactor building the RC pumps will be manually tripped on loss of subcooling margin. The LPSW and Component Cooling to the pumps and motors will be lost by reactor building isolation on high reactor building pressure. When the subcooling margin is returned in the RCS, RC pumps may be restarted. However, they cannot be restarted until LPSW, Component Cooling and seal return are reestablished to the pumps and motors. In the same scenario, the component cooling water has to be reestablished to the letdown coolers before letdown flow is reestablished for RC inventory control and boron measurements.

Operating Penetrations of Safety Fluid Systems

The penetrations of safety systems may either be open or closed depending on the safety function. If the safety function is required immediately following an abnormal transient the isolation valves will be automatically opened.

Some of these isolation valves (e.g. HPI) remain open during power operation, but they still have the automatic valve opening feature just in case the valve happens to be closed.

If the safety function is not required immediately following an abnormal transient, the isolation valve may need to be manually opened. (i.e. emergency reactor building sump valves).

There are many reasons for operating these safety system penetration isolation valves. Some of these are as follows:

The reactor building emergency sump valves need to be opened following a large LOCA to provide core recirculation flow with the LPI system.

The DHRS suction valves are opened to provide long term cooling.

The LPSW penetration valves supplying cooling water to the reactor building fans remain open following a transient requiring reactor building isolation. Reactor building cooler isolation valves exist but require operator action to close. Separate isolation valves are provided for each cooler. These valves are used to isolate cooling water to a cooler should the cooler develop a leak. The LPSW is not borated, thus any leakage from the cooler would dilute any borated water collected

in the sump after a LOCA. Therefore, to prevent boron dilution of the reactor building sump water, the LPSW to the leaking cooler must be isolated. The LPSW flow to and from each cooler is measured. A leak is indicated if the LPSW flow to a cooler is more than the LPSW flow from the cooler.

If a steam leak occurs inside the reactor building, the operator needs to stop feedwater flow to the leaking SG to prevent excessive mass and energy release to the reactor building. Although the steam leak can cause reactor building isolation on high pressure, the reactor building does not become filled with radioactive fluid as with a LOCA. Consequently, reactor building isolation does not carry the same significance that it does for a LOCA.

The main feedwater, auxiliary feedwater, and steam pipes penetrate the reactor building and therefore need to be considered in preventing radiation release from the reactor building. However, they will not be discussed in this chapter on Reactor Building Control. These systems provide the major method of reactor coolant heat removal; therefore, control of these penetrations is discussed in other sections of these guidleines along with core heat removal.

The HPI and LPI line isolation valves remain open for post-accident high pressure injection and low pressure injection.

The cooling water to the quench tank should be reopened if PORV or safety valve release causes the quench tank temperatures to increase.

2.3.4 Penetration Room Ventilation System Control

Most of the reactor building penetrations exit the reactor building through the penetration room. The most likely passage for reactor building leakage is through these penetrations even when they are isolated or sealed. The design assumption is that the penetration leakage into the penetration room will be 50% of the total reactor building leakage. The penetration room has a ventilation system to control this leakage. The ventilation system takes this leakage, dilutes it with outside air, then passes it through filters to remove radioactive materials and then sends it up the vent stack for maximum atmospheric dispersion.

This penetration room ventilation system starts automatically on high reactor building pressure because the high pressure is the driving force causing leakage.

The penetration room ventilation system should be started only on high reactor building pressure as shown in Table 8. The penetration room ventilation system should not be started on high reactor building radiation or low RC pressure because the

penetration room ventilation system may needlessly pull radiation out of the reactor building because of the vacuum it produces in the penetration room. For a high reactor building pressure condition the contents of the reactor building can leak through the penetrations into the penetration room. The amount of radiation leaking to the penetration room may be significant and should be controlled by filtering the leakage and releasing it via the vent stack.

Control With Equipment Failure

For a failure of the ES system to auto start the penetration room ventilation system, the operator should manually initiate the system.

If the penetration room ventilation system fails to operate, the operator should attempt to start the small reactor building purge fan taking suction from the atmosphere and blowing to the vent stack and assure that the penetration room ventilation system valves are open. The purge fan flow up the stack will tend to create an exhaust flow from the penetration room to the vent stack while diluting any gas flow from the penetration room.

Follow-up action should include determining why the penetration room system failed to operate and make appropriate repair.

2.4 Details of Reactor Building Internal Environment Control

As mentioned in Section 1.0 controlling the reactor building internal environment is one of the two general requirements for reactor building control.

After an abnormal transient, the environment inside the reactor building can become harsh enough to cause failures of equipment and structures. The reactor building environment needs to be controlled to reduce and hopefully eliminate failures of equipment and structures (including failure of the reactor building) required to control the harsh environment itself and to control core heat removal. Reactor building control requires controlling the following to bring them within acceptable limits.

a. Reactor building pressure and temperature control

b. Reactor building hydrogen control

c. Reactor building emergency sump chemistry control

d. Reactor building radiation control

e. Reactor building emergency sump level control.

Control of each of these will now be discussed in more detail.

2.4.1 Reactor Building Pressure and Temperature Control

These two parameters (temperature and pressure) are coupled together, e.g., if the temperature is reduced the pressure will also be reduced and vice versa. Therefore, control of these two parameters is discussed together as follows:

The reactor building pressure and temperature must be reduced to:

a. Prevent exceeding the design pressure and temperature of the reactor building.

b. Reduce the driving force for reactor building leakage.

c. Prevent equipment damage.

The reactor building design limits for pressure and temperature should not be exceeded to prevent damaging equipment inside the reactor building. Reducing the reactor building pressure also reduces the driving force for leakage of radioactive materials.

2.4.1.1 Diagnosis Method

The operator diagnoses the potential for exceeding the reactor building design limits by monitoring the following parameters:

- reactor building pressure
- reactor building temperature

If either of these parameters exceed the limits given in Table 8, then the potential for increased leakage and thermal effects on operating equipment exists. The possibility of exceeding the reactor building design limits also exists.

2.4.1.2 Control

If the high reactor building pressure (4 psig) or temperature (130 F) limit is reached, the three reactor building coolers should be started with maximum LPSW flow to the coolers. If the 4 psig high reactor building pressure limit is reached, the fan speed should be switched to low in anticipation of reaching 10 psig and the dense air caused by this higher pressure.

If the high reactor building pressure limit is reached, the ES system will automatically:

a. start the three reactor building fans (if not already on)

b. switch all fans to low speed

c. fully open the LPSW supply valve to each cooler.

No automatic actions are made if the high reactor building temperature limit is reached. If the reactor building pressure exceeds the 4 psig high reactor building pressure limit, but the pressure will not exceed the 10 psig high high reactor building pressure limit (e.g. the reactor building pressure is decreasing) then two fans can be put on high speed to reduce the reactor building pressure more quickly. Likewise, if the reactor building pressure did exceed 10 psig but has dropped back below 10 psig two fans can be put into high speed. The operator should also try to reduce the amount of energy being released to the reactor building by:

1. Assuring that no feedwater is being added to a steam generator which has a break inside the reactor building.
2. Increase the heat removal by the LPI cooler. (if in operation)
3. Increase the heat removal by the steam generators. (if possible)
4. Isolate any break in the RCS if possible

The objective is to reduce the reactor building pressure and temperature below 4 psig and (by Oconee) respectively (lower if possible) and maintain the parameters below these values.

Reactor Building (RB) Pressure Rule

Whenever RB pressure $\geq$ 4 psig:

a. Isolate all automatic RB isolation valves

b. Start all RB coolers

c. Start RB penetration room ventilation system

Whenever RB pressure $\geq$ 10 psig:

a. Start both RB sprays systems

b. Put all RB coolers fans in low speed

Reactor Building (RB) Temperature Rule

Whenever the average RB Temperature $\geq$ 130 F

a. Start all RB coolers

Whenever the average RB temperature $\geq$ (By Oconee)

a. Start all RB spray systems.

Control With Equipment Failures

The primary method of reducing the reactor building pressure and temperature is with the reactor building coolers. If the reactor building coolers fail to keep the reactor building pressure below 10 psig or the reactor building temperature is going to exceed the reactor building design temperature, a backup cooling method should be considered. The backup method is the reactor building spray system. This operation should

be automatic by the ES on high building pressure of 10 psig. However, manual initiation should be made if the ES does not auto actuate the reactor building spray systems or if the reactor building high high temperature limit is reached (i.e., without a high reactor building pressure actuation).

The reactor building spray can be secured after the reactor building pressure and temperature are reduced to 4 psig and (by Oconee).

If the reactor building coolers or reactor building spray fail to operate, determine the cause of failure so that repairs [illegible] be made and the systems put back into operation. The system auxiliary diagrams can help in this area.

The operator should concentrate on maintaining adequate core cooling to keep down the reactor building pressure. Ambient heat losses from the reactor building will inherently assist pressure reduction but the effect is small and cannot [illegible] expected to mitigate pressure increases caused by rapid energy releases. Following a LOCA the reactor building pressure will probably not reach the design pressure limit with failure of all reactor building coolers and sprays as long as the operator maintains adequate core cooling.

The operator should not attempt to reduce reactor building pressure by opening the reactor building purge valves. The valves may not operate at high pressure and if they do, they may fail open which would release large quantities of radioactive material after a LOCA. A steam line break does not release much radiation, however, a steam line break should not increase the pressure enough to reach the design pressure limit, so there would not be a need to open the purge valve after a steam line break.

After a steam or feed line break the reactor building pressure cannot reach the design pressure limit with failure of the coolers and spray if the operator promptly isolates feedwater flow to the bad generator. However, the design temperature limit might be exceeded for a steam line break because the steam is superheated. The superheated steam release will cause the temperature to go unproportionally high compared to the reactor building pressure.

2.4.2. Hydrogen Control

The hydrogen concentration is controlled to limit hydrogen burn in order to prevent:

a. Equipment damage within the reactor building

b. Structural damage to the reactor building or penetrations.

2.4.2.1 Diagnosis Method

The reactor building hydrogen concentration should be periodically analyzed following a LOCA to determine the rate of hydrogen build up in the reactor building. The sample frequency should be (later). The reactor building atmosphere should be as close as possible to homogeneously mixed prior to sampling. Following a LOCA, several hours are required to get a homogeneous mixture with the reactor building fans running. Therefore, immediately after a LOCA has been detected by high reactor building radiation the reactor building fans should all be started and run continously.

2.4.2.2 Control

Hydrogen is controlled to keep the concentration below the value at which hydrogen burns. Hydrogen control is accomplished in two ways. First, local pockets of hydrogen concentration are eliminated by using the reactor building cooling fans to mix the air and vapor contents of the reactor building. This will disperse any local concentration of hydrogen and create a homogeneous mixture for sampling and purging the hydrogen in the reactor building. A valid indication of the hydrogen concentration cannot be made unless the hydrogen is evenly dispersed throughout the reactor building. Several hours are required to evenly mix the hydrogen. Therefore, the reactor building fans should be started as soon as a

LOCA is suspected i.e. when high radiation appears in the reactor building.

If a high reactor building radiation limit is reached, fuel failure is possible which would cause high volumes of hydrogen from Zr-water reaction. To mix up the atmosphere all reactor building fans should be operated even if the reactor building pressure and temperature conditions do not warrant fan operation. Even if cooling water is not available to the RB coolers, the fans should be operated to mix the reactor building atmosphere.

When the average hydrogen concentration reaches 3.5%, the hydrogen purge system should be used to prevent the hydrogen concentration from accumulating above this value by removing hydrogen from the reactor building. However the hydrogen should not be purged if the reactor building radiation is above the high-high limit (see Table 8) because this will cause an excessive radiation release from the reactor building.

Reactor Building (RB) Hydrogen Rule

Whenever RB Hydrogen concentration $\geq$ 3.5 vol% start hydrogen purge if RB radiation $\leq$ (By Oconee)

Whenever RB Hydrogen concentration $\geq$ 4.0 vol% start all RB spray systems

Control with Equipment Failures

If the hydrogen purge system cannot keep the reactor building hydrogen concentration below 4.0 vol%, the reactor building spray should be turned on. The spray water droplets tend to suppress the burning of hydrogen; therefore, the probabil of ignition will be reduced. The spray droplets will also suppress burning if ignition occurs and help remove the heat from a hydrogen burn. The spray should be stopped when the concentration returns to $\leq$ 3.5% (if the reactor building pressure is also within its limits). Also, if the hydrogen concentration exceeds 4.0 vol% the reactor building pressure should be maintained as low as possible. This is beca hydrogen burning will cause a sudden pressure spike and this pressure spike will be added onto any pressure existing in the reactor building. The objective is to keep this overall pressure below the reactor building design pressure.

Zircaloy-water reaction can be a major contributor to t overall hydrogen concentration. This contributor can be ma negligible by assuring adequate core cooling. Therefore, it is <u>very important</u> to maintain core cooling to reduce this source of hydrogen.

If the reactor building cooler fans cannot be started, the reactor building recirculation vent and CRDM fans should be run to mix the reactor building gases.

2.4.3 Emergency Sump Chemistry Control

The reactor building emergency sump water boron concentration following a LOCA should not be diluted. This is to assure the recirculation water has sufficient boron to maintain the core subcritical. Boron dilution can be caused if a fluid system leaks non-borated water to the reactor building sump such as a main feedwater leak or a leak in a reactor building cooler. If this occurs the source of non-borated water should be isolated. Boron precipitation can also cause a loss of boron (i.e. core boiling distilling RC). The operator should take appropriate actions to prevent boron precipitation as discussed in Appendix F.

2.4.3.1 Diagnosis Method

The reactor building emergency sump water pH and boron concentration are measured after a LOCA by taking a sample from the LPI recirculation flow.

2.4.3.2 Control Method

pH

The reactor building sump water following a LOCA will be acidic due to the large amount of boron. The pH should be increased to at least a pH of 7.0 (between 7.0 and 8.0) to reduce stress corrosion cracking and other adverse chemical effects. The pH can be increased by adding NaOH to the emergency sump water via the LPI recirculation line 30 minutes after switching to recirculation flow.

Boron

If the reactor building emergency sump boron concentration is low the operator should attempt to locate the cause. Such possibilities are boron precipitation in the RCS or non-borated fluid systems leaking into the RCS, i.e.:

- reactor building cooler cooling water leak
- Main or auxiliary feedwater leak
- Steam line leak

If the low boron concentration limit is reached, borated water must be added to bring the sump boron concentration back above the low concentration limit.

2.4.4 Reactor Building Radiation Control

The primary objective of reactor building radiation control is to keep the radioactive materials which leak from the RCS out of the atmosphere and in the reactor building sump. When opening reactor building isolation valves or removing any building sump water, care should be taken to prevent any radiation releases to the environment.

The best way to limit the amount of radiation in the reactor building is to assure adequate core cooling.

2.4.4.1 Diagnosis Method

The reactor building radiation level is monitored. If it exceeds the limits in Table 8, the following control actions are taken.

2.4.4.2 Control Method

If the high reactor building radiation level is reached, start all reactor building cooler fans at the maximum speed allowed by reactor building pressure. This will lead to a homogeneous reactor building atmosphere which will lower the reactor building dome radiation to reduce radiation emitting from the reactor

building and condense the reactor building steam to get radioactive materials into the sump water.

If the high-high reactor building radiation level is reached, hydrogen purging must be stopped if in progress. The operator must also check for inadequate core cooling and take appropriate action for core cooling as outlined in the ICC section.

Reactor Building (RB) Radiation Rule
Whenever RB radition ≥(By Oconee):
a. Isolate all automatic RB isolation valves <u>except</u> LPSW to RC motors and CCW
b. Start all RB coolers
Whenever RB radiation ≥ (BY Oconee) stop hydrogen purge

2.4.5 Reactor Building Emergency Sump Level

The sump water level should be maintained high enough for LPI (HPI) or reactor building spray recirculation flow, but not so high that it submerges equipment important to core heat removal or reactor building control. If the water level is too low, more borated water should be added. If it becomes too high consideration of lowering sump level should be made depending on the core cooling method being used (sump recirculation or DHR). The amount of radiation in the sump must also

be considered before attempting to remove any of the sump water.

2.4.5.1 Diagnosis Method

The reactor building emergency sump level should be measured and kept between the values shown in Table 8. Exceeding the maximum level must be avoided at all times (if possible) to prevent submerging equipment and instrumentation.

The level must be kept above the low level limit whenever the reactor building sprays or LPI are in the recirculation mode.

2.4.5.2 Control Method

High reactor building emergency sump level - If the sump water radiation level of the reactor building sump is less than specified in Table 8, the excess sump water may be drained from the reactor building using either the normal sump drain or emergency sump drain. The operator should also check for other fluid systems leaking into the sump and attempt to isolate them. These could be:

- reactor building cooler cooling water
- Main or emergency feedwater

- Component cooling water
- Steam line.

Low reactor building emergency sump water level - borated water should be added to the reactor building sump through the HPI or LPI flow.

The operator should check for a steam generator tube rupture or some other possible source of water loss such as inadvertent pumping or draining from the sump or break in LPI or spray recirculation line.

3.0 USING REACTOR BUIDLING CONTROL EQUIPMENT

3.1 Reactor Building Coolers

To maximize the cooling ability of the reactor building coolers the cooling water supply valves should be fully opened and the fan speed should as high as possible.

Reactor Building Cooler Fan Speed Guideline

1. Whenever the RB pressure exceeds 4 psig and is increasing, the RB fan speed should be put in low in anticipation of the RB pressure exceeding 10 psig.
2. Whenever the RB pressure is below 10 psig, two RB fans can be put in high speed to increase cooling and mixing.

Restrictions exist on the operation of the reactor building cooler fans to prevent the motors from tripping on motor overload. The fans have two speeds. Whether or not a fan can be operated in high speed depends on the reactor building pressure and the number of other fans already operating in high speed. As the reactor building pressure increases, the atmosphere of the reactor building becomes more dense. The denser gas causes the fan motors to work harder and possibly trip on overload. Therefore, the fans should not be operated in high speed above a 10 psig reactor building pressure.

The fans blow into air ducts and these ducts restrict the air flow. The greater the air flow the larger the restriction and the more the fan motors have to work. With all three reactor building fans operating, the motor load is large enough to cause a fan motor to trip on motor overload. Therefore, no more than two fans should be operated in high speed.

Starting fan motors and changing the fan motor speed also cause the motors to heatup. Therefore, the number of starts and speed changes must be limited.

The restrictions on the fan operation are:

1. When the reactor building pressure is < 10 psig a maximum of two fans should be operated at high speed.
2. When the reactor building pressure is $\geq$ 10 psig each fan must be operated in low speed.
3 Each fan is allowed one fan start or two fan speed changes per hour.

3.2 Reactor Building Spray

The reactor building spray system takes suction from the same source as the LPI system. Consequently, when the LPI is switched from the BWST to the reactor building emergency sump, the reactor building spray will also be switched. Likewise, if the operator

does not switch the LPI to the reactor building emergency sump on low BWST level, both LPI flow and reactor building spray flow will be lost. Following a LOCA, the reactor building sump water temperature will approximate the saturation temperature of the reactor building. The reactor building spray pumps will take suction from this water source as do the LPI pumps. This water will provide reactor building cooling as it is sprayed into the reactor building. However, the reactor building spray cooling ability can be increased by taking partial suction from the LPI recirculation flow downstream of the LPI coolers. This will provide subcooled water to the reactor building spray header. Two valves, LP-15 and LP-16, can be opened downstream of the LPI coolers to divert part of the cooled recirculation water to the reactor building spray pumps while the spray pumps continue to take suction from the reactor building sump.

The reactor building spray can be detrimental to equipment located inside the reactor building. Therefore, its use should be limited to between 10 psig increasing reactor building pressure (start spray) and 4 psig decreasing reactor building pressure (stop spray). To avoid dead heading a reactor building spray pump, the minimum pump flow rate is 60 gpm. When drawing water from the BWST the maximum pump flow rate is 1800 gpm to avoid reactor building spray pump run out.

When drawing water from the reactor building sump the maximum pump flow rate is 1000 gpm to meet spray pump NPSH requirements.

Reactor building pressure switches are used to start the reactor building spray. Therefore, the reactor building spray system will still automatically start on high high reactor building pressure if the reactor building pressure indicators fail.

3.3 Reactor Building Purge System

Do not attempt to reduce reactor building pressure by opening the purge system valves. The purge system valves are not designed to operate under accident conditions. They may not open, and if they do open they could fail in the open position.

3.4 Reactor Building Hydrogen Sampling

A representative hydrogen sample probably can not be obtained until after 3 hours of continuous reactor building cooler fan operation. However, reactor building hydrogen samples should be taken as soon after a LOCA as possible.

3.5 Reactor Building Temperature Monitoring

The reactor building temperature should be determined by averaging the available temperature monitor indications.

3.6 Reactor Building Environmental Effects on Equipment and Instrumentation

The reactor building environment after a transient such as a LOCA can be harsh causing equipment failures. Some equipment will be more susceptible to failure than others. Even the equipment designed to withstand LOCA conditions can fail. Consequently, equipment failures can be expected and will tend to be more frequent than during normal plant operation and because of the harsh environment the equipment failures in the reactor building cannot be repaired. Some of the factors contributing to the harsh reactor building environment are described below:

1. High Pressure and Temperature

   The energy released by a LOCA, steam line break or hydrogen burn can significantly increase reactor building pressure and temperature. The hydrogen burn can be localized at the equipment.

2. Missiles and Steam/Water Jets

Breaks in the high pressure fluid systems can cause missiles and steam/water jets which can damage equipment and instrumentation.

3. Chemistry

Boric acid and NaOH are added to the reactor building. The NaOH will tend to neutralize the boric acid but will also react with aluminum to create hydrogen.

4. Radiation

High Radiation - primarily gamma - will be generated from LOC water accumulated in the reactor building floor and steam and gases in the atmosphere.

5. Reactor Building Flooding

The water level in the reactor building sump may increase high enough to submerge equipment.

6. Humidity/Reactor Building Spray

This excessive water can cause electrical equipment to short out.

Equipment failures include failure to operate on command and incorrect equipment operation such as instrumentation indicating the

wrong values. Therefore, where possible the operator should make allowances for possible equipment failures. Most importantly, instrument readings should be checked whenever possible by another indication because of the possibility of erratic instrument readings.

3.7 Penetration Room Ventilation System

The penetration room ventilation system uses charcoal filters to remove iodine isotopes. The charcoal will ignite when its temperature reaches about 660 F. This temperature could be reached if the charcoal filters have accumulated iodine isotopes and air flow through the filter is lost. The decay of the iodine isotopes will generate heat and gradually heat up the charcoal filter to ignition temperature. Should a fan stop running, the cross connection between the two penetration room vent ducts should be opened to allow cooling air through both filters.

The heat up time of the charcoal filter is slow, e.g. ∿5 hours to combustion under worst conditions. This should be sufficient time to get air flow through the idle filter.

4.0 DISCUSSION OF SOME TRANSIENTS WHICH CHANGE THE REACTOR BUILDING ENVIRONMENT

The two abnormal transients having the greatest effect on the

reactor building are the LOCA and the steam line break in the reactor building. These two transients have been selected for a discussion of how they impact the reactor building.

4.1 LOCA

The LOCA changes the reactor building condition more than the steam line break does because, in addition to adding mass (water) and energy, it can also add hydrogen and large amounts of radioactive material. The amount of hydrogen will not be significant unless core damage occurs. However, large amounts of radiation will be released to the reactor building even if adequate core cooling is maintained.

The following items will be discussed in relation to a LOCA:

- Reactor Building Pressure and Temperature
- Hydrogen production
- Equipment failures
- Radiation release

4.1.1 Reactor Building Pressure and Temperature Changes

A LOCA releases hot water to the reactor building. This hot water flashes to steam and heats up the reactor building air. The steam will expand, cool and depressurize. In contrast, the air will

heat up and pressurize. The decreasing steam temperature and increasing air temperature will come to equilibrium at a common temperature. This temperature will determine the reactor building pressure. The pressure will be the sum of the pressure due to air and the pressure due to the steam.

For example: If the final reactor building temperature is 200 F, the corresponding pressure would be determined as follows:

Assume the initial air temperature is 100F (or 570R) at 0% relative humidity (R. H.), and the reactor building pressure is 0 psig (or $\sim$15 psia). The pressure due to air will increase approximately as an ideal gas:

$$\frac{\text{initial pressure x initial volume}}{\text{initial temp.}} = \frac{\text{final pressure X final volume}}{\text{final temp.}}$$

The final reactor building volume will be slightly less because of the water added to the reactor building sump, but for this example no change will be assumed. The final air pressure will be:

$$\frac{\text{initial press.}}{\text{initial temp.}} \text{ X final temp. } =$$

$$\frac{\text{15 psia}}{\text{570R}} \text{ X 660R } =$$

17.4 psia

The final steam pressure will be P_{sat} at 200F or 11.6 psia. The total pressure will be 17.4 psia + 11.6 psia or 29.0 psia (14.0 psig).

For an initial reactor building average temperature of 110F the total building pressure as a function of final temperatures is shown in Fig. 32. This figure was developed by repeating the above example for various building final temperatures. One important assumption is that sufficient water is added to raise the reactor building relative humidity to 100%. This is true for a LOCA.

The key point is that with an initial reactor building temperature of 110F or above, the reactor building pressure for a given temperature will not exceed that shown in Figure 32 (except for a hydrogen burn which will be discussed later).

Large increases in reactor building temperature will not occur for small LOCAs such as a stuck open PORV. The rate of water and energy release will be so small that the pressure increase will not be large. It may not even be high enough to reach the 4 psig reactor building isolation setpoint. However, radiation will be released to the reactor building and the RCS pressure may reach the ES setpoint.

4.1.2 Hydrogen Generation

If the core is not adequately cooled, the major source of hydrogen can be from Zr-water reaction. The hydrogen generation rate is shown in Figure 33. The extent of the Zr-water reaction is controlled by the ECCS performance. The emergency core cooling equipment is sized to limit the hot spot fuel temperature to 2200F or less during a LOCA, thereby limiting the zirconium-water reaction to less than one-tenth of one percent. Consequently, if the operator keeps the core adequately cooled Zr-water reaction will be an insignificant source of hydrogen and the most significant source of hydrogen will be from the radiolytic decomposition of water (mostly in the reactor building sump). Some chemically produced hydrogen will also come from zinc-boric acid reaction (one pound of zinc yield 5.5 SCF of H_2) and the initial hydrogen content ($\sim$472 SCF) of the reactor coolant.

If the sources of radiolytic and chemical hydrogen are calculated, a time dependent increase in hydrogen can be determined as shown in Figure 34 (assuming a 1% Zr.-water reaction). For this curve the hydrogen concentration in the reactor building reaches the purge setpoint (3.5%) in 780 hours. The required purge flow rate is a function of the hydrogen generation rate which decreases with time. For Figure 34 the hydrogen purge rate starts at 12 SCFM to keep the hydrogen concentration at 3.5%.

If the water level drops to the point where significant amounts of fuel are uncovered, large quantities of hydrogen gas can be generated very quickly by the metal-water reaction between the hot Zircaloy and steam. The amount of hydrogen that was estimated to be in the TMI-2 reactor coolant system is shown in Figure 35. It can be seen that 173,000 SCF was generated in the period between 2 and 4 hours after the accident. At least 63,000 SCF was released into the containment and burned approximately 9 hrs. 50 minutes after the accident causing a 28 psig pressure pulse inside the reactor building. The remaining hydrogen was slowly removed from the RCS over the next several days either by a combination of venting the pressurizer into the containment and degassing the letdown flow in the makeup tank or by a small leak in the area of the control rod drives. The hydrogen concentration in the reactor building subsequently reached a maximum concentration of 2.3 volume percent on April 3rd (6 days after the accident). A 95 CFM recombiner operated frequently during the period of April 2nd to May 1st and slowly reduced the hydrogen concentration to about 0.8 percent.

It can be seen that even if all of the plant's waste gas storage tanks were empty, they could only accommodate a small fraction of the hydrogen generated. Thus, the capability to vent these tanks into the reactor building is highly desirable, especially if the

vent stream could be passed through a recombiner. Other important facts to be considered are:

1. The hydrogen in the reactor coolant system does not present an explosive hazard, and if the RCS pressure is kept high, the hydrogen occupies a relatively small volume (e.g., 173,000 SCF = 2125 cu. ft. @ 220 psig and 450F) and should not interfere with forced circulation cooling; however, it can interfere with natural circulation cooling if it blocks the top of the hot legs. Adequate core cooling will prevent accumulation of H_2 in the RCS.

2. If all the hydrogen is vented to the reactor building too rapidly, the potential for an explosion is quite high.

After a LOCA the reactor building fans should be on to mix the reactor building atmosphere. This will probably take considerable time. Figure 36 shows it took several hours to mix the radioactive particles. This is also a good indication of how long it will take to mix H_2.

A hydrogen explosion will cause a rapid increase in reactor building pressure which will be in addition to any pressure existing in the reactor building. Figure 38 shows the effect of a H_2 explosion on the reactor building pressure temperature curve following a LOCA.

The radiolytic decomposition of water also adds oxygen to the reactor building atmosphere but the amount is not large enough to have any significant impact on the flammability limit for the hydrogen.

4.1.3 Equipment Failures Inside the Reactor Building

The harsh environment inside the reactor building can cause equipment failures such as pressurizer heaters, radiation monitors etc.

Failure of Radiation Monitors

One of the TMI-2 area radiation monitors inside the reactor building, the personnel hatch monitor, was removed during one of the first post-accident reactor building entries for analysis. The results of the analysis showed that a rubber boot on the instrument's cable connector was not on securely; therefore, when the reactor building sprays were on, a drop of water caused short between two high voltage connections.

The TMI-2 dome monitor also failed. Failures that leave the indicator needle on scale are particularly confusing because they

often indicate believable values (e.g., TMI-2 dome monitor indicating a constant 40 R/h after about 2 months of decay, see Figure 37).

A similar problem arose during a different open PORV incident caused by the loss of power to the non-nuclear instrumentation. Since the power supply was a -10 V to +10 V system, the loss of power produced 0 V, which corresponded to a mid-scale reading. The mid-scale reading is usually very close to the normal reading for that process variable, so the operators have difficulty identifying which instruments are functioning properly and which have failed.

4.1.4 Behavior of Noble Gases and Iodine During A LOCA

Noble Gases Behavior

The noble gas nuclides - primarily Kr-85, Kr-88, Xe-131m, Xe-133m, and Xe-133 - will always follow the steam and/or coolant flow and will accumulate wherever there is a gas phase, i.e., in the pressurizer steam space, in the makeup tank cover gas, or in the reactor building atmosphere. A significant fraction of all noble gas nuclides will be soluble in the reactor coolant while the system is at a high pressure, but the gas will rapidly leave the liquid

phase upon depressurization. After 10 to 20 minutes at atmospheric pressure, less than a few percent will remain in solution. All noble gases are chemically inert, so there is no effective way to remove them following a reactor accident other than to attempt to contain them and wait for decay, which should take 60 to 760 days. After this period, the remaining noble gas activity is so small that it no longer presents a significant offsite radiological risk.

Iodine Behavior

The behavior of iodine nuclides - primarily I-132, I-133, and I-135 - is complicated by the fact that the iodine can exist in several different chemical forms, each of which has somewhat different behavior characteristics.

Elemental iodine in a dry environment is very volatile at all temperatures above room temperature. If water is present, the elemental iodine will quickly react with it in the following manner:

$$I_2 + H_2O \rightarrow H^+ + I^- + HOI$$

$$3HOI \rightarrow 3H^+ + 2I^- + IO_3^-$$

The first of these reactions is very rapid; the second is much slower. There are several things to note about these reactions:

1. They produce H, which tends to make the solution more acid, so the reaction will proceed more rapidly if the solution is alkaline (basic).
2. If the reactions went to completion, there would be no iodine activity in the gas phase; however, in most reactor environments some iodine (0.01 to 0.1%) is always detectable in the gas phase. There is speculation that HOI (which is called hypoidous acid) is somewhat volatile, and since the HOI concentrations are so low, the second reaction proceeds very, very slowly and does not go to completion.

Sodium hydroxide is added to the containment sump water. This will increase the iodine removal rates when the spray is put on recirculation flow because it raises the pH.

If elemental iodine (I_2) becomes airborne as a vapor, it can be removed very quickly by the reactor building sprays. The removal half-time is approximately 2 to 3 minutes.

Charcoal filters on the penetration room ventilation system will remove elemental iodine very effectively. The previous operating history of the charcoal filters does affect the removal effectiveness to some degree, but in general the removal effectiveness remains high. Of course if gross condensation occurred in the charcoal bed, the pore structure would become clogged with water

and the absorption effectiveness of the charcoal would be destroyed. (An important point to remember about charcoal is that it acts to lower the vapor pressure of materials passing through it; therefore, condensation could occur at relative humidities of about 90%.)

4.1.5 Reactor Building Radiation Monitoring During a LOCA

Figure 37 shows the dome monitor readings following the TMI-2 accident. Several interesting features are shown in this figure.

1. The large dose rate reduction produced by the reactor building sprays.
2. The large increase in dose rate between March 29 and April 2 when the RCS was degassed by venting steam to the reactor building.
3. The increase in dose rate on April 6, when the waste gas tanks were vented into the reactor building.
4. The series of linear decreases in dose rate indicating exponential removal rates, which surprisingly did not relate to radioactive decay but rather to other removal processes (e.g., the large decrease on or about May 1 seemed to relate to operation of the reactor building coolers).
5. The constant dose rate, about 40 R/h, after May 15, which may be anomalous and indicative of instrument failure.

Figure 36 shows the response of the reactor building dome monitor following a February 26, 1980, transient. The detector is unshielded and thus showed the correct dose rate response relative to the other detectors in the reactor building. There are several important observations that can be made based on Figure 36:

1. The fuel handling bridge monitor was reading about a factor of 500 lower than the reactor building dome monitor. Apparently, the hot steam and air carry the activity to the top of the reactor building; thus, even though the bridge monitor and dome monitor are both exposed to large gas volumes in the reactor building there is not enough mixing to prevent radiation dose rates being several orders of magnitude different.

2. The reactor building incore instrument area monitor responded to the decay of the N-16 activity immediately after reactor trip, then eventually climbed to correspond to the dome monitor reading. The decrease in both the dome monitor reading and the reactor building incore instrument area monitor reading corresponds to the mixing rate provided by the reactor building fan-coolers.

3. The core was properly cooled so that no core damage occurred. However, the radioactive material released was great, causing a significant change in the reactor building radiation monitor readings.

The reactor buildings at both TMI-2 and the event in Figure 36 had significant radiation concentration gradients for several hours and therefore the assumption of a well mixed building atmosphere cannot be justified on the basis of the fan-cooler operation until several hours after a LOCA.

4.2 Steam Line Break

A steam line break releases mass (water) and energy as does the LOCA. However, the amount released will be less as long as the feedwater flow to the broken SG is stopped early in the transient. If not stopped the feedwater system will continually add feedwater to the SG which will leak into the reactor building. Therefore operator action for this transient is important in reducing the affect to the reactor building.

A steam line break will release superheated steam to the reactor building. The energy content of the steam is very high compared to the water released. Consequently, when the reactor building

air is heated up and the steam is cooled, they will meet at a temperature high enough to maintain superheated water in the reactor building atmosphere.

In other words, the reactor building atmosphere will not reach 100% relative humidity and the pressure temperature relationship will tend to be like that shown in Figure 39. It will not follow the reactor building pressure temperature curve as does the LOCA, instead it will tend to be below and to the right of the pressure temperature curve.

If the reactor building spray is turned on, the pressure temperature conditions will rapidly return to the reactor building pressure temperature curve. The pressure will drop slightly but the temperature will drop significantly.

TABLE 7

SUMMARY OF REACTOR BUILDING ISOLATION CONTROL ACTIONS

PARAMETER	LIMIT	CONTROL ACTION
High RB Pressure	$\geq$ 4 psig	- Isolate all penetrations of non-safety fluid systems used in power operation.* - Start Penetration Room Ventilation System.
Low RCS Pressure	$\leq$ 1550 psig	- Isolate all penetrations of non-safety fluid systems used in power operations except those for LPSW to RCP motors and CC water.
High RB Radiation	by Oconee	- Isolate all penetrations of non-safety fluid systems used in power operations except those for LPSW to RCP motors and CC water.
		*If a large LOCA does not exist, then LPSW and CCW for the RCP's should not be isolated.

TABLE 8

SUMMARY OF REACTOR BUILDING INTERNAL ENVIRONMENT CONTROL ACTION

PARAMETER	LIMIT	CONTROL ACTION

High RB Pressure	$\geq$ 4 psig	- Start 3 RB coolers with
		  - fans in low speed in anticipation of reaching 10 psi
		  - LPSW control valves to coolers wide open
		- When RB pressure returns below 10 psig or if RB pressure does not reach 10 psig, switch 2 fans to high speed.

High high RB Pressure	$\geq$ 10 psig	- Start 2 RB spray systems - When RB pressure returns to 4 psig, stop spray. - Switch RB cooler fan speed to low if in high - When spray is on recirculation, take partial spray suction from LPI cooler discharge if needed for RB spray pump NPSH requirements.
High RB Temperature	$\geq$130F	- Start all 3 RB coolers and if RB pressure $\leq$ 10 psig run two fans in high speed, other in low speed; provide max LPSW to coolers.

TABLE 8 (Cont'd)

PARAMETER	LIMIT	CONTROL ACTION
High-high RB Temperature	by Oconee	- Start 2 RB spray systems - Stop spray when RB temp returns to (by Oconee) if RB spray not being used to reduce RB pressure - Check for steam line break and stop feeding bad generator.
High RB Radiation	by Oconee	- Start RB coolers and fans (on high speed if allowed by RB pressure) to mix possible hydrogen being released & condense radioactive steam.
High high RB Radiation	by Oconee	- Stop hydrogen purge. - Check for ICC
High RB Hydrogen	$\geq$3.5 vol%	- Purge RB hydrogen but only if RB radiation less than high-high RB radiation limit. - Run all RB cooler fans continuously.

TABLE 8 (Cont'd)

PARAMETER	LIMIT	CONTROL ACTION
High high RB Hydrogen	> 4.0 vol%	- Start 2 RB spray systems - Stop spray if hydrogen RB concentration drops to 3.5 vol% (if RB pressure, temperature, & radiation levels permit)
RB Emergency Sump pH	< 7.0	- Add NaOH 30 minutes after switch over to recirculation flow to raise pH to between 7.0 and 8.0.
RB Emergency Sump Boron	< by Oconee (ppm vs burnup)	- Add boron. - Check for non-borated fluid system leaks into RB, i.e.: • RB cooler cooling water • Main or aux feedwater • Component cooling water • Steam line - Check for boron precipitation in RCS
High Level RB Emergency Sump	> by Oconee elevation of equipment not to be submerged	- Lower sump level if core cooling is assured and radiation levels of sump water are less than (by Oconee) - Check for other fluid systems leaking into RB.
Low Level RB Sump When on LPI Récirculation	< by Oconee	- Raise level to between low and high RB emergency sump level limits with borated water - Check for steam generator tube rupture.

TABLE 9

PENETRATIONS OF NON-SAFETY SYSTEMS USED IN POWER OPERATION
(AUTOMATIC ISOLATION VALVES)

PENETRATION	ES CHANNEL 1	ES CHANNEL 2	ES CHANNEL 5	ES CHANNEL 6
RC Letdown	HP 3, 4	HP-5		
RCP Seal Return	HP 20	HP 21		
Quench Tank Vent	GWD-12	BWD-13		
RB Normal Sump	LWD-1	LWD-2		
Quench Tank Suction	CS-5	CS-6		
RB Purge	PR-1	PR-2,3,4,5		
RB RIAs	PR-7, 9	PR-8, 10		
PZR. Sample	RC-5, 6	RC-7		
SG Sample	FWD-105, 107	FDW-106, 108		
SG Drain		FDW-103, 104		
Component Cooling - RC Pumps - Letdown cooler - Quench tank cooler - CRD service structure			CC-7	CC-8
LPSW to RCP Motors			LPSW-6, 15	LPSW-6, 15

NOTES:
1. All ES channel 1 valves are electric operated and inside RB
2. All ES channel 2 valves are air operated and outside RB
3. ES channels 1 and 2 auto actuate on low RCS pressure and high RB Pressure
4. ES channels 5 and 6 auto actuate on high RB pressure

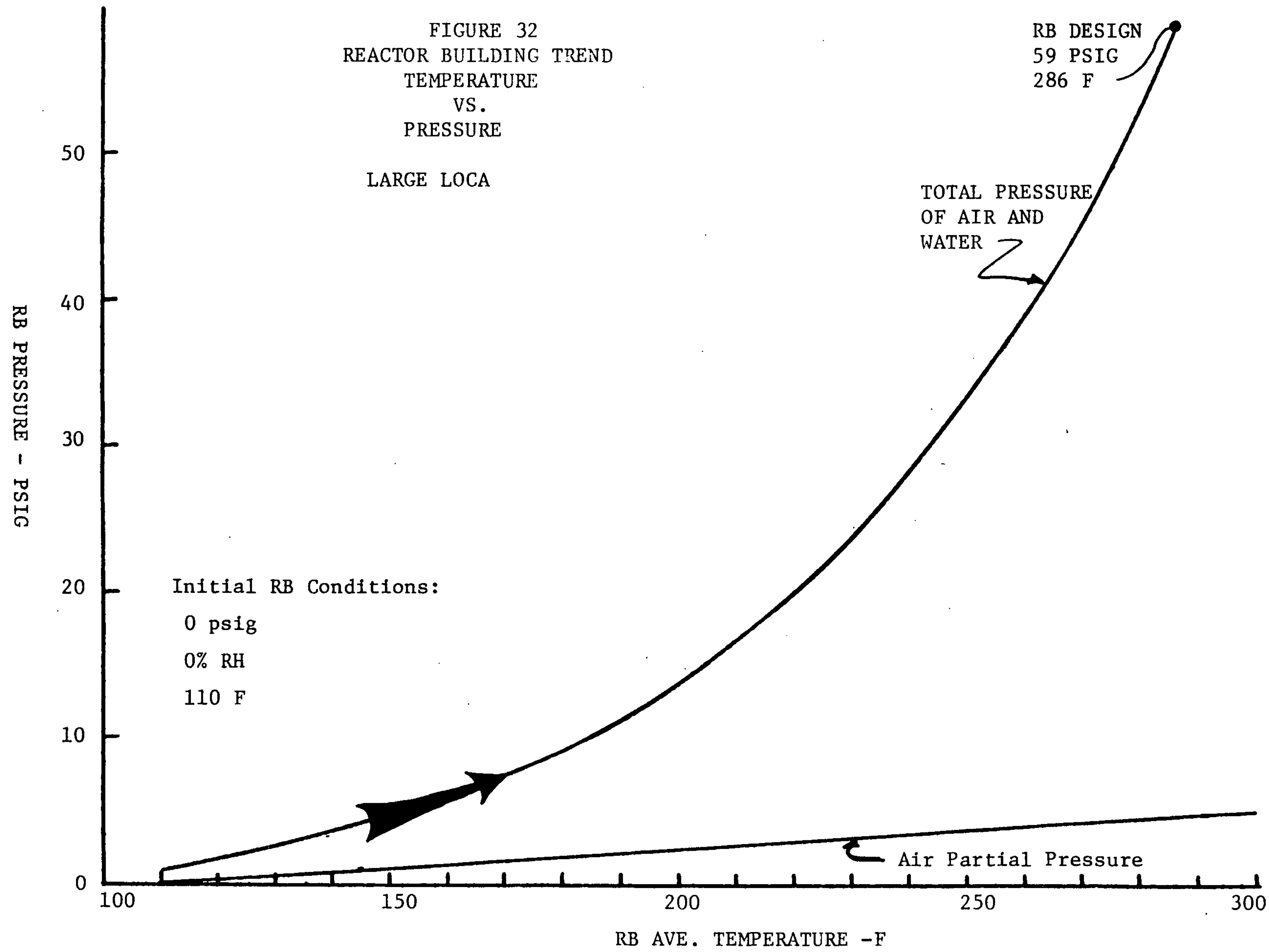


FIGURE 32
REACTOR BUILDING TREND
TEMPERATURE
VS.
PRESSURE
LARGE LOCA
RB DESIGN
59 PSIG
286 F
TOTAL PRESSURE
OF AIR AND
WATER
Initial RB Conditions:
0 psig
0% RH
110 F
Air Partial Pressure
RB PRESSURE - PSIG
0
10
20
30
40
50
100
150
200
250
300
RB AVE. TEMPERATURE -F

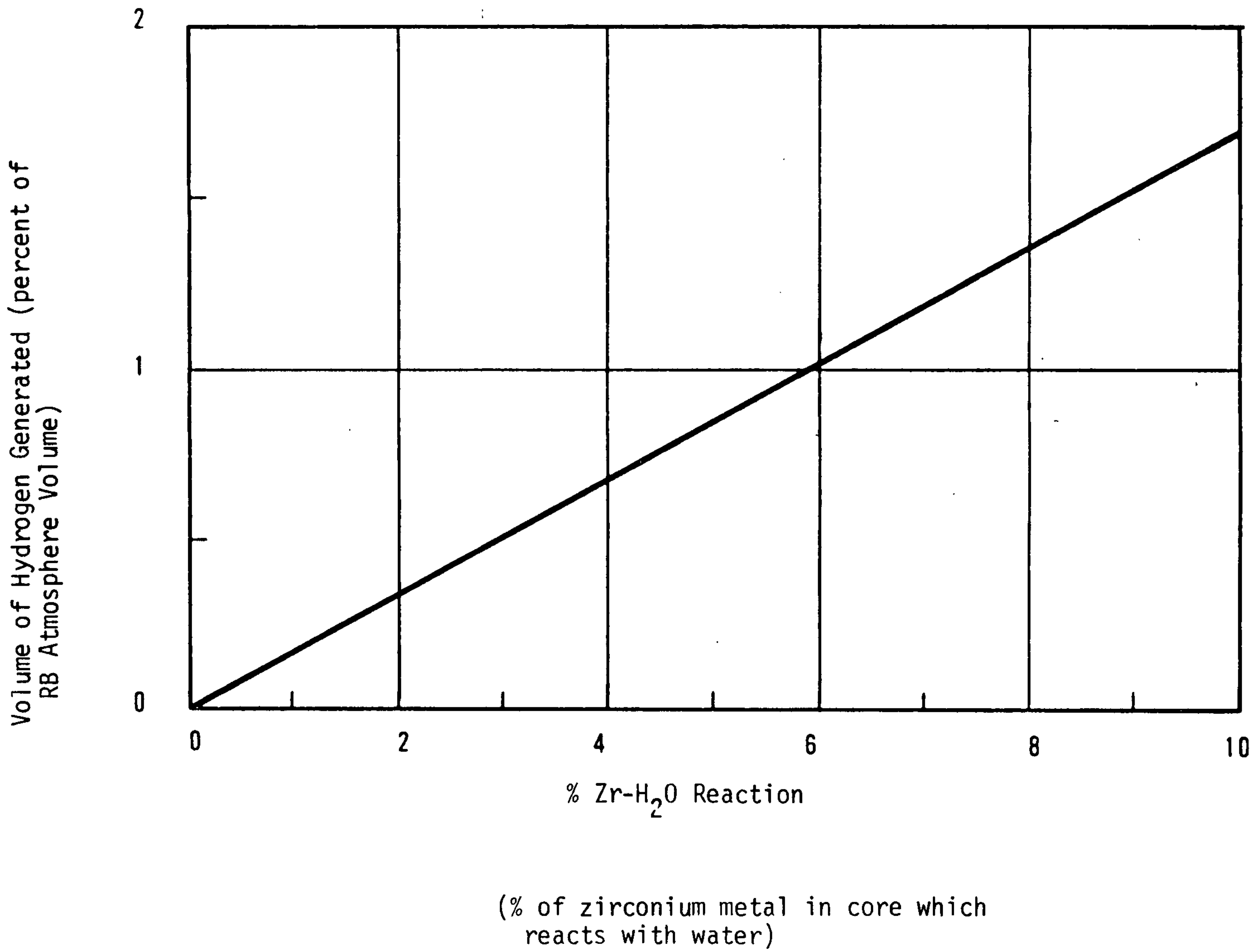


FIGURE 33

HYDROGEN GENERATION FROM ZR-H_2O
REAC

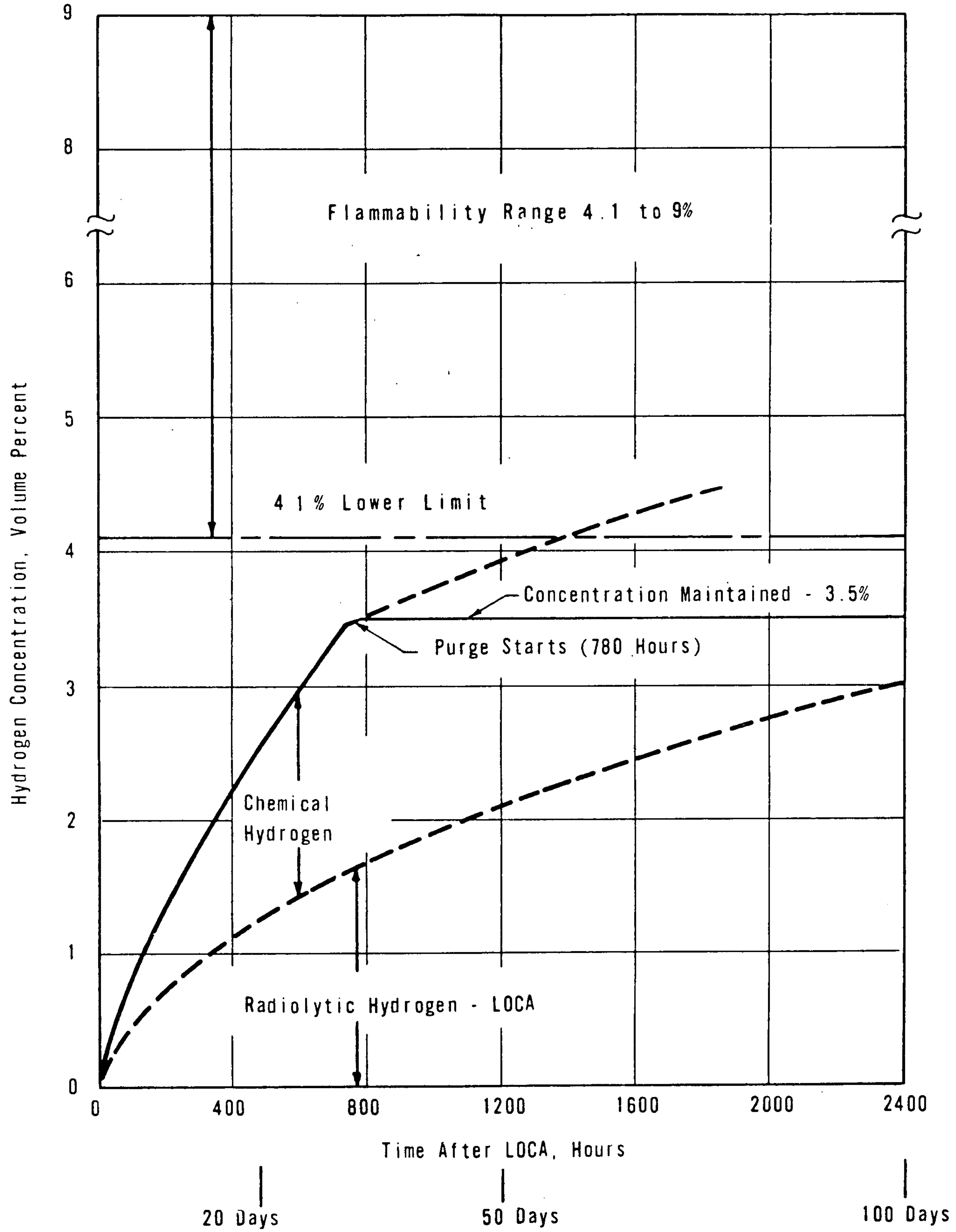


FIGURE 34

REACTOR BUILDING HYDROGEN CONCENTRATION FOLLOWING A LOCA

FIGURE 35

ESTIMATE OF TOTAL HYDROGEN
VOLUME IN RCS OF TMI-2

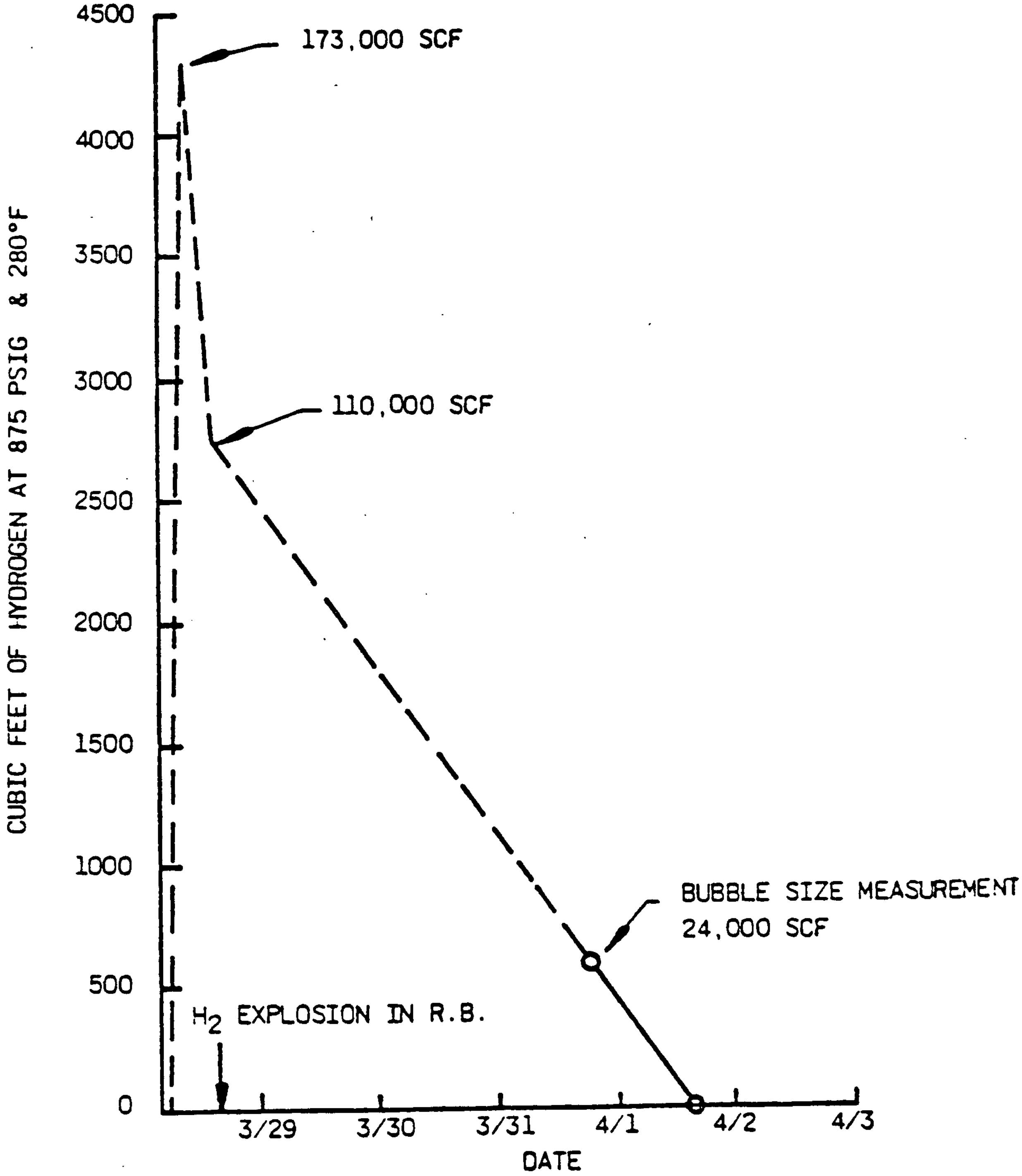


FIGURE 36

REACTOR BUILDING RADIATION MONITOR
RESPONSE FOLLOWING 2-25-80
LOSS OF NON-NUCLEAR INSTRUMENTATION WITH
HPI COOLING EVENT

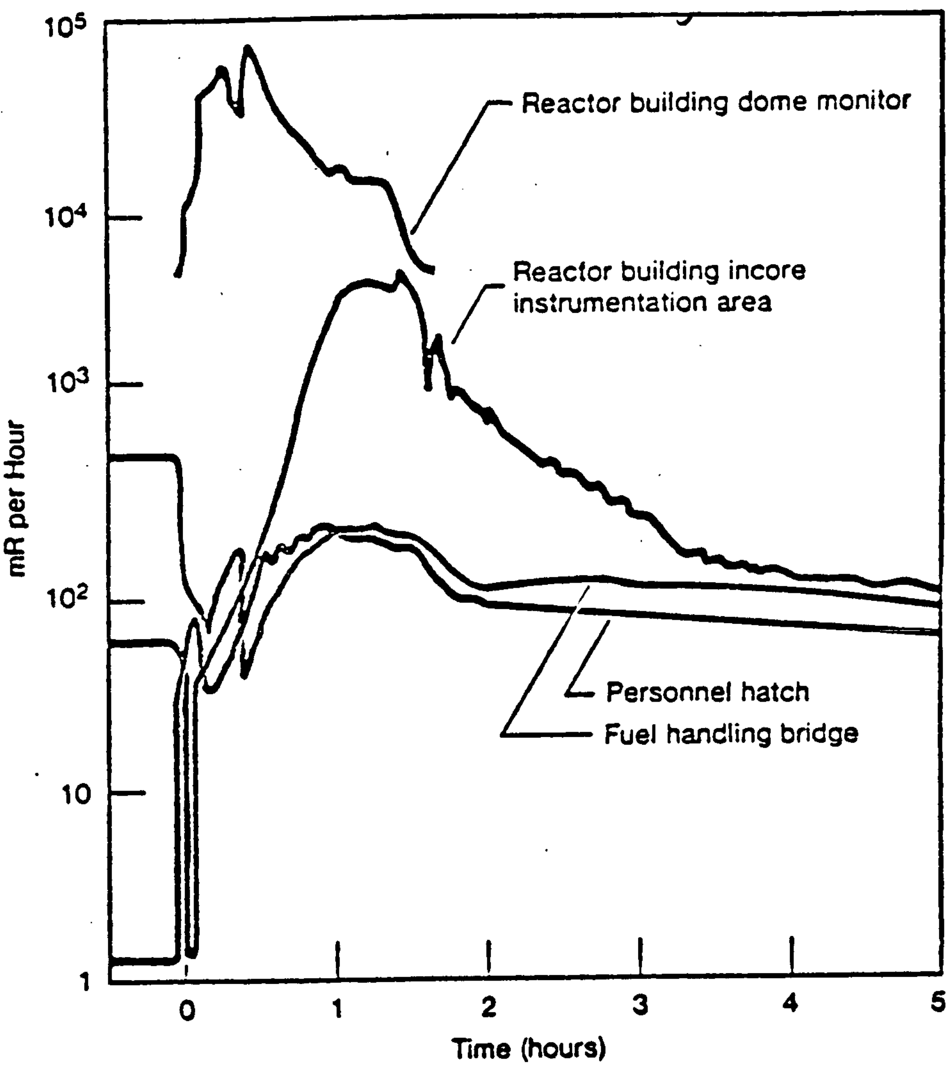
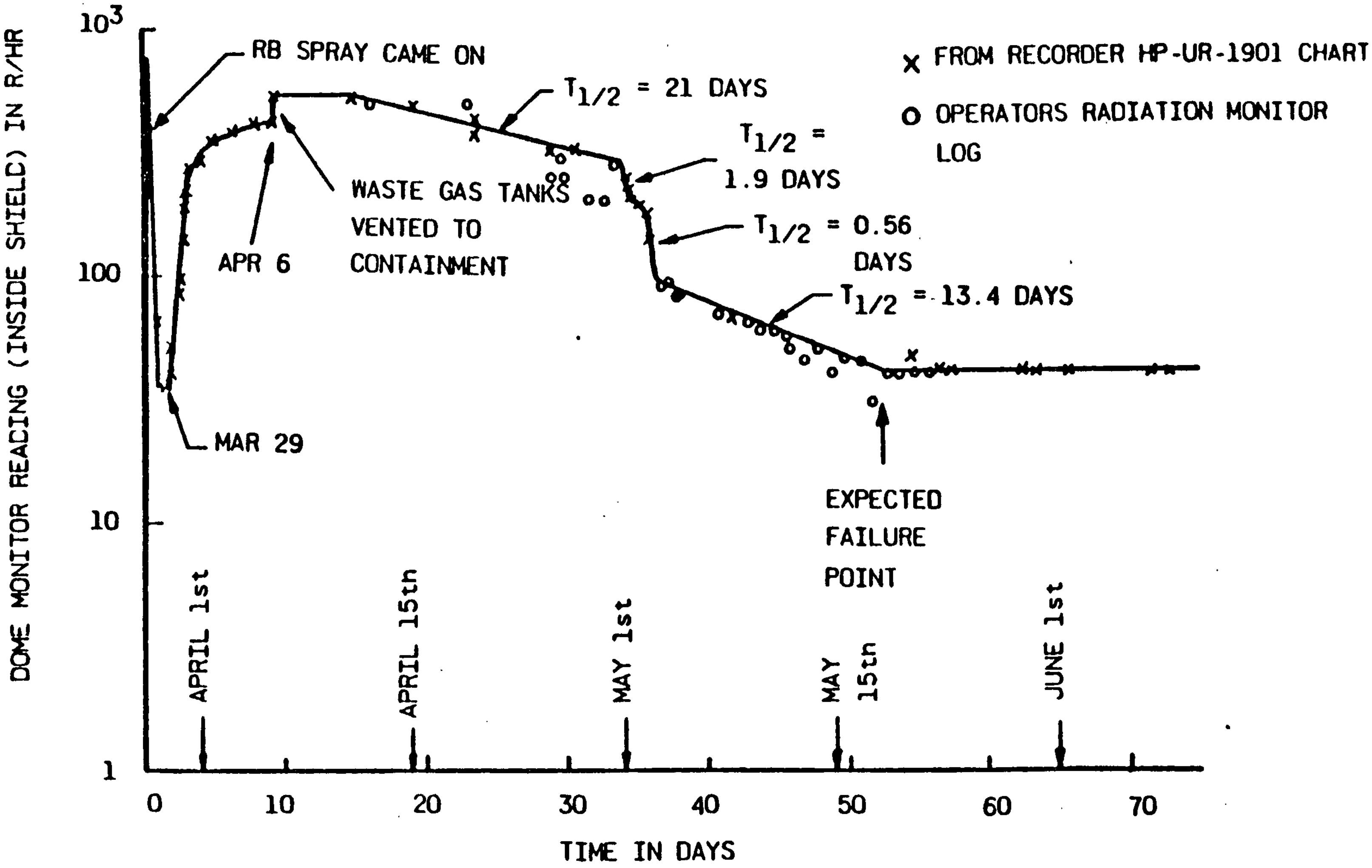


FIGURE 37

TMI-2 DOME MONITOR READINGS (INSIDE SHIELDED HOUSING)



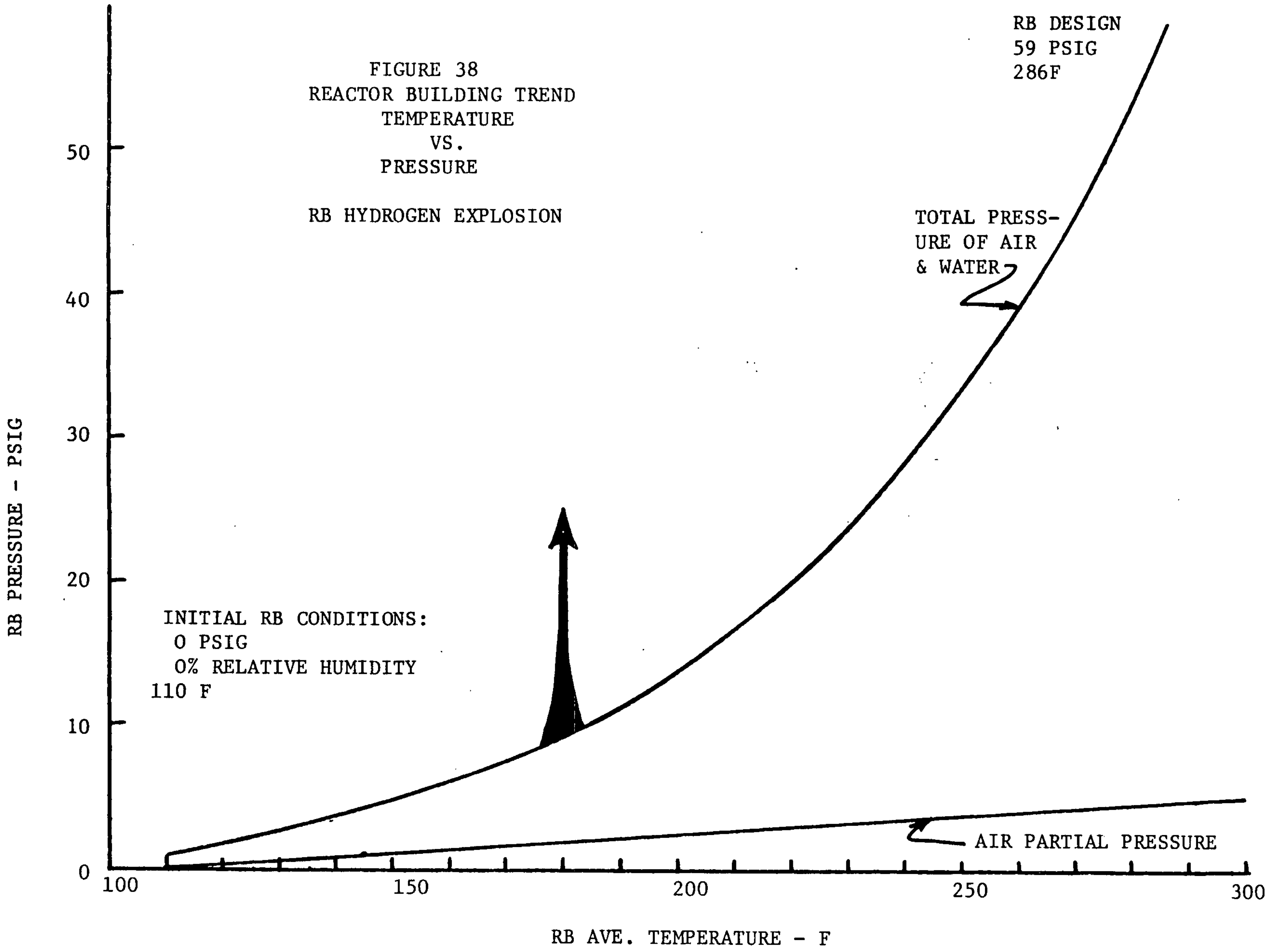


FIGURE 38
REACTOR BUILDING TREND
TEMPERATURE
VS.
PRESSURE
RB HYDROGEN EXPLOSION
RB DESIGN
59 PSIG
286F
TOTAL PRESSURE OF AIR & WATER
INITIAL RB CONDITIONS:
0 PSIG
0% RELATIVE HUMIDITY
110 F
AIR PARTIAL PRESSURE
RB PRESSURE - PSIG
RB AVE. TEMPERATURE - F
0
10
20
30
40
50
100
150
200
250
300

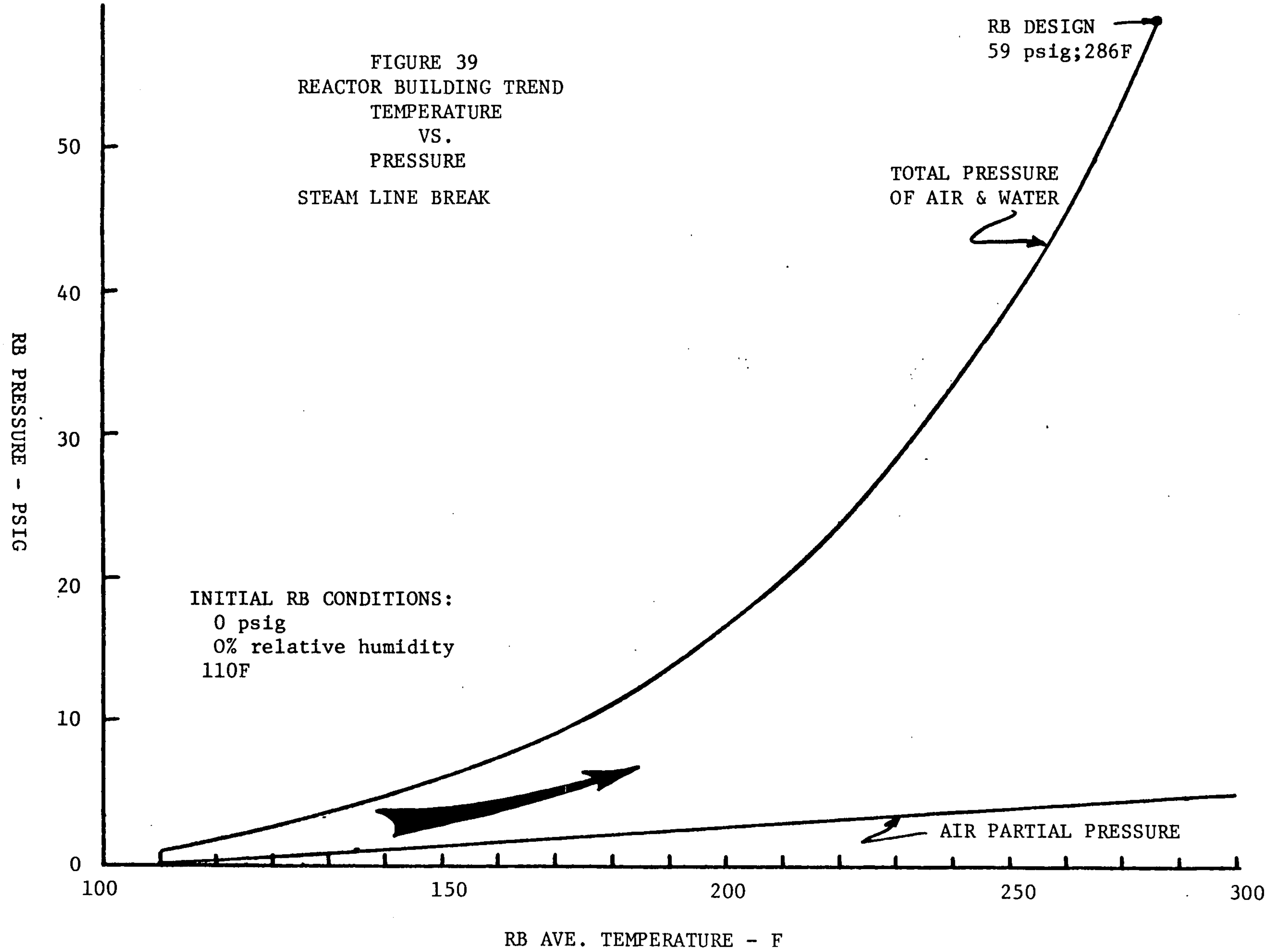


FIGURE 39
REACTOR BUILDING TREND
TEMPERATURE
VS.
PRESSURE
STEAM LINE BREAK
RB DESIGN
59 psig;286F
TOTAL PRESSURE
OF AIR & WATER
INITIAL RB CONDITIONS:
0 psig
0% relative humidity
110F
AIR PARTIAL PRESSURE
RB PRESSURE - PSIG
0
10
20
30
40
50
100
150
200
250
300
RB AVE. TEMPERATURE - F

Figure 40 GENERAL APPROACH FOR CONTROL OF THE REACTOR BUILDING

Abnormal Transient Changing Reactor Building Environment

i.e., LOCA or Steam/Feed Line Break releasing, steam water hydrogen boric acid and radiation

↓

First Line of Action - Limit Radiation Release

Assure the reactor building leak paths are sealed

- manual isolation of penetrations
- automatic isolation of penetrations
- start of penetration ventilation system (if high RB press.)

↓

Second Line of Action - Control Reactor Building Internal Environment to Protect the Reactor Building

Operate reactor building control equipment to prevent RB design conditions from being exceeded

- reduce RB pressure/temperature
- prevent hydrogen from exploding

↓

Third Line of Action - Eliminate or Reduce the Cause for the Change in the Reactor Building Environment

i.e., isolate feedwater to leaking SG
adequate core cooling

↓

Fourth Line of Action - Control Reactor Building Internal Environment to Protect RCS Heat Removal Equipment/Inst. and to Protect RB Control Equipment/Inst.

- control RB pressure and temperature
- control RB hydrogen to prevent hydrogen burn
- control RB emergency sump level for sufficient LPI recirculation flow and to prevent submerging equipment
- control RB emergency sump chemistry to make less acidic and to retain adequate boron concentration
- control RB radiation to prevent spread of radioactive fluid to other systems and reduce RB airborn radiation

CHAPTER H

USE OF THE GUIDELINES

Philosophy of Part I Organization

Part I is designed for use following any reactor trip or forced shutdown. Its primary purpose is to maintain core cooling and ensure plant stability. A reactor trip, depending on the cause and initial plant conditions, can result in demands on various systems and components (MSSV's, TBV's, EFW, etc.). These demands, coupled with the cause of the trip or forced shutdown, are occurrences that have a higher probability for abnormal conditions to develop.

When equipment or system failures occur resulting in an abnormal plant response following a trip, it is not so important to immediately identify the cause as it is to restore stable, controlled conditions. Once the plant has been stabilized, then time exists for failure identification and the decision for future operations (i.e., return to power, remain at existing conditions, or begin controlled cooldown). The main thrust of ATOG and certainly the most important aspect of dealing with any transient or accident is to maintain adequate core cooling. The most expeditious and positive approach to accomplish this objective is to recognize abnormal conditions when they develop and take appropriate actions to restore stability.

Part I of ATOG contains four basic sections that flow in a logical sequence based on this philosophy. The four sections, in order, are:

a. Section I: Immediate Actions

b. Section II: Vital System Status Verification

c. Section III (A-D): Follow-up Actions for Key Symptoms

d. Cooldown Procedures (including Inadequate Core Cooling)

The immediate actions in Section I are those actions taken after every reactor trip regardless of cause (e.g., manually tripping the reactor and turbine). Once the immediate actions have been performed, the next priority is to verify that the reactor is shutdown and that key systems and equipment are available and functioning properly (e.g., NNI/ICS power, turbine stop valves shut, etc.). These items are included in Section II. One abnormal transient (excessive MFW), should it occur, can requi prompt recognition and response by the operator to prevent the possibly severe consequences of water spillover into the steam lines. Therefore, one of the first checks in Section II is the verification that MFW flow has runback. The important items to check early are reactor/turbine trip and MFW flow status. No particular importance should be placed on the sequencing of the other verification steps as long as all of them ar performed.

Section II also includes obtaining a status of the four main symptoms of abnormal transients (the three basic heat transfer symptoms and the special case for tube rupture). However, operator training should emphasize continuous surveillance for indications of off-normal conditions. Four main symptoms are:

1. Lack of adequate subcooled margin
2. Lack of primary to secondary heat transfer (overheating)
3. Excessive primary to secondary heat transfer (overcooling)
4. Indications of steam generator tube rupture

Recognition of these symptoms is covered in more detail in the "P-T Diagram" and "Abnormal Transient Diagnosis and Mitigation" chapters of this volume. Should any of these symptoms occur, Section II references the operator to the appropriate follow-up action in Section III of Part I.

Philosophy on the Use of Section III

The order that the symptoms are listed in Section II corresponds with the order of the respective follow-up actions in Section III and is based on the relative priorities of the corrective actions. Lack of adequate subcooling margin is of course top priority because core cooling cannot be assured until certain actions are performed. If a lack of adequate subcooling margin occurs, tripping the RC pumps and initiating full HPI flow must be performed quickly. While ICC is the most severe condition of the RCS, it follows after a lack of adequate subcooling margin and the operator would be directed to the ICC section if it should occur. However, if the actions in the lack of adequate subcooling margin can be performed, ICC will not occur. Lack of heat transfer and excessive heat transfer are both second priority symptoms. These two symptoms are equal

in priority because they are mutually exclusive conditions. However, excessive heat transfer will require the quickest response by the operators. The SG Tube Rupture is the last priority symptom. The order of the priority for these symptoms can be understood by focusing on the objectives for treating any abnormal transient, which is to maintain adequ core cooling and minimize radiation release. For example, even though a SGTR occurs which can release large amounts of radiation if it is not quickly treated, the lack of adequate subcooling margin and the lack of heat transfer take precedence. This is true because, if adequate core cooling is not maintained, the radiation release would be much greater.

Again, even though it is important to diagnose and treat a steam genera tube rupture as soon as possible, termination of a concurrent overcooling transient takes precedence. This is true because:

1. The overcooling could be the result of a steam leak to atmosphere on the steam generator with the tube rupture, resulting in higher offsite releases.
2. The overcooling increases the tensile stresses on the steam generator tubes which could result in a larger leak size.
3. The contraction of the RCS liquid volume due to the overcooling, especially when compounded by the inventory loss through the tube leak, could result in draining the pressurizer and saturation of the RC loops. Subsequent voiding in the loops can significantly delay the cooldown and thus lead to increased offsite releases.

This discussion of priorities is given to show the logic behind the development of Part I. The important point to remember is that the operator should always be alert for the presence of these symptoms and should always proceed directly to the appropriate section for follow-up actions for a symptom without necessarily waiting to see if a "higher priority" symptom develops. This constant operator surveillance should continue during and after the stabilization of a transient. Some symptoms can mask the presence of others. For example, an overcooling transient can mask the presence of a small LOCA. However, once the overcooling transient is terminated, the small LOCA should quickly become evident. Abnormal transients can occur at any time and will not always oblige by beginning immediately after a reactor trip. Therefore, continuous surveillance is warranted.

One symptom, lack of adequate subcooling margin, always requires immediate attention. The operator must trip reactor coolant pumps and initiate HPI flow immediately whenever the subcooled margin is lost regardless of which part of Section III he is currently following. For this reason, the other parts of Section III will either a) reference the operator to Section III.A or b) reiterate the required actions given in Section III.A for loss of subcooling margin.

It can be seen that these symptoms can occur in various combinations and virtually at any time during a transient or subsequent cooldown. Thus it would be very difficult to write a comprehensvive procedure that will lead

the operator through any possible sequence of events. Where it is known that multiple symptoms are more likely to occur the guidelines will specifically address the possibility. However, to ensure maximum coverage this approach is supplemented by a combination of operator training and procedural guidelines. Recognition of the four basic symptoms and following appropriate actions of Section III will provide plant stabilization for all single events regardless of cause. Operator training is the key in the recognition and understanding of the four basic symptoms. When events become more complex, either due to additional failures or due to an event progressing to the point of inducing other symptoms (e.g., loss of subcooling margin), operator training is again the key in successful mitigation. The procedural guidelines contain all the necessary information but the operator must know when to implement the appropriate sections. He does this by careful surveillance of the plant conditions during and following a transient, recognizing the symptoms whenever they occur and going to the appropriate section. In this respect ATOG is somewhat similar to event-oriented procedures in that the operator must recognize a condition and react to it. The difference is that the symptom will <u>always</u> be evident when the failure occurs. The cause (event) will not always be evident. Also, there are just four symptoms to recognize as opposed to numerous events.

In addition to recognizing a symptom and implementing the appropriate part of Section III, the operator must also know when to transfer between parts

of Section III (for multiple symptoms or incorrect diagnosis) and when to terminate his actions if the problem is corrected. Some basic instructions for the use of Part I can be summarized as follows:

1. Priorities between the parts of Section III:
   a. Loss of subcooling margin always requires immediate attention regardless of which part of Section III is being following (with single exception of raising SG levels noted in 1.c below).
   b. Lack of heat transfer or excessive heat transfer must be corrected before, or at least concurrently, with actions for SGTR.
   c. Excessive heat transfer (overcooling transients) must always be terminated as soon as possible, and before raising SG levels to high level for loss of subcooled margin.
2. Follow the appropriate part of Section III for the dominant symptom, unless a "higher priority" symptom (in item 1 above) appears, in which case recycle to the part of Section III for the higher priority symptom.
3. If a reactor trip occurs during a forced shutdown, recycle to Section I.
4. If a major change in equipment status occurs during the performance of a part of Section III or subsequent cooldown, carry out the appropriate actions of Section II (i.e., loss of NNI/ICS power, loss of offsite power, safeguards actuation, etc.). This can be accomplished in parallel with Section III.

5. If it is discovered that an incorrect diagnosis has been made and:

   a. no other symptom exists (e.g., inadvertent entry into Section III because an overshoot after trip misinterpreted as overcooling), then stabilize plant (e.g., restore heat transfer; don't exit III.C immediately after isolation both SG's) and recycle to Section II; or

   b. a different symptom exists, then recycle to the appropriate part of Section III.

6. If, during the performance of follow-up actions in Section III, the cause of the transient becomes evident and is corrected, then hold at that point and allow the plant to stabilize while checking for other symptoms/problems (e.g., if in the process of isolating both SG's for excessive heat transfer and closing of the TBV's on one SG stops the transient, there is no need to continue isolation of steam and feedwater on both SG's). Similarly, if the intent of a group of actions is satisfied, then continuation of those actions may not be necesary (e.g., HPI flow can be throttled and SG levels do not need to be raised to the high level once subcooling margin is restored).

7. All normal limits and precautions are applicable during the performance of Part I unless specifically superceded by the ATOG procedure (e.g., the use of pump bumps regardless of NPSH requirements when saturated with SG level). Whenever a step appears in Part I that supersedes a normal limit or precaution,

it has been carefully considered and deemed acceptable for plant conditions existing at the point in the procedure (e.g., violation of fuel pin compression limits during a large SGTR). One should not infer, however, that since it is acceptable at one point in the procedure then it is always acceptable to violate the limit or precaution.

As an example on the use of Part I, assume an overcooling transient occurs following a reactor trip that leads to drainage of the pressurizer and a loss of the subcooling margin. When the subcooling margin is lost the operator must quickly perform the actions of III.A which include tripping the reactor coolant pumps and initiating full HPI flow (1.a above). The operator is also required to raise SG levels. These actions are required in the event of a small break LOCA. In this case the LOCA could be masked by the overcooling or induced by the overcooling transient. The overcooling will be caused by either excessive feedwater flow or loss of steam pressure in one or both steam generators. The operator must either regain control of feedwater or isolate the loss of steam pressure (may require isolation of one SG) before proceeding with raising SG levels (1.c above). The intent is to terminate the overcooling first as it could otherwise continue to mask the presence of a small break and, if the loss of subcooling margin was due to <u>just</u> an overcooling event (i.e., no LOCA), then filling the SG's would be exactly the <u>wrong</u> action (it would aggravate the overcooling). Similarly, if the subcooled margin is regained while the

operator is filling the SG's, he need not continue to the high level required for small breaks (6 above). The intent of the actions (restore subcooled margin and ensure SG's available for heat removal) has been satisfied. The operator should throttle HPI flow and adjust SG levels to the appropriate setpoint for natural circulation or forced circulation if he has restarted RC pumps. These variations on the use of the guidelines are examples of situations that, if Part I attempted to cover them all, would result in a procedure of such bulk and complexity as to render it useless. However, by training the operator so that he understands what the symptoms mean in terms of heat transfer and what is really occurring in the plant, and so that he understands what the guidelines are intended to accomplish, then his educated use of the guidelines will ensure transient termination and plant stability for any event or combination of events.

Another example of proper operator judgement can be shown by again assuming an overcooling transient. The guidelines may tell the operator to completely isolate both SG's (steam and feedwater). This is because Part I is designed to restore plant stability assuming worst case accidents, which, in the case of overcooling, would be an unisolable steam line break. If the operator does not know the cause of the overcooling or even which SG is causing the overcooling then the proper response is to isolate both SG's to stop the transient. With the transient terminated he should then be able to monitor secondary conditions and isolate the problem to

one SG so that controlled decay heat removal can be restored using the unaffected SG. However, if he should discover the cause of the transient while he is isolating the SG's (e.g., stuck open TBV's) and isolation of this cause does indeed stop the overcooling, then there is no reason for him to complete the isolation of both SG's. If he understands the purpose of the steps he is taking in following the guidelines, then he will understand when that intent is satisfied and may be able to exit the procedure at that point rather than arbitrarily following it through to completion. Of course, should he fail to realize that isolation of the TBV's terminated the transient, continuing through the guidelines, including isolation of both SG's for this example, will not cause any significant problem; it is merely unnecessary. Continuing through the procedure will lead to restoration of core cooling using the unaffected SG(s).

Objectives of Section III

In order to promote understanding of the procedural guidelines, this section will address each subsection of Part I, Section III in terms of what the plant conditions are, what are the possible causes, and why the operator is directed to take the specified actions (i.e., what the actions are intended to accomplish). The referenced section of Part I should be followed while reading this section.

Section III.A, Follow-up Actions for Treatment of Lack of Adequate Subcooled Margin

Whenever plant conditions reach or exceed the subcooled margin curve the

assumption is that the RCS is saturated. The RCS can become saturated as a result of three basic causes:

i. loss of coolant inventory (LOCA)

ii. overcooling that results in sufficient coolant contraction to drain the pressurizer, or

iii. prolonged loss of heat transfer that allows the RCS to overheat to saturation at high pressure (this cause would be recognized as lack of heat transfer before loss of subcooling and handled in accordance with Section III.B).

The primary objectives of this section are to 1) restore subcooled margin and 2) maintain or restore core cooling. To accomplish these objectives, HPI must be initiated and either secondary heat transfer or HPI cooling must be established.

Two actions are always required whenever the subcooling margin is lost:

1. trip all reactor coolant pumps and
2. initiate full, balanced HPI flow.

These actions are necessary in the event the loss of subcooled margin is due to a small break LOCA. Tripping the RC pumps must be done immediately following the loss of subcooled margin to minimize inventory loss if a small break exists. In addition, if the loss of subcooled margin was the result of an overcooling transient, HPI will compensate for the coolant contraction and restore subcooling. Re-establishing controlled secondary

heat removal should then be possible. If the loss of subcooled margin was due to a total loss of feedwater (main and emergency) then full HPI will be needed to establish HPI cooling (in Section III.B). In this case subcooling should also be restored but it may take some additional time until the cooling capacity of the HPI flow exceeds the core heat generation rate.

Raising SG levels should also be done in the event of a small break LOCA. High SG levels will allow condensation of steam voids in the RCS side of the upper tube region to establish boiler-condenser cooling. If the transient was initiated by total loss of feedwater then the operator may not be able to restore feedwater and raise SG levels but after he has established HPI cooling he should continue efforts to restore feedwater. If the transient indicates overcooling then he should not raise level in the affected SG(s) until the overcooling is corrected. This is to prevent further uncontrolled plant cooldown. Actions are also included to isolate possible causes for the loss of primary pressure.

At this point the operator has full HPI flow, the RC pumps are off, he is raising or attempting to raise SG levels, and he has isolated possible causes of the loss of subcooling. Further actions will be determined by the plant response to the actions already taken. The subcooling margin may be restored by HPI or the plant may remain at saturation. In either case, primary to secondary heat transfer may or may not exist.

Subcooled Margin Restored

If the subcooled margin is restored, the operator should restart reactor coolant pumps. This will aid in establishing primary to secondary heat removal if it does not already exist and will provide better mixing of the cold HPI flow to alleviate the brittle fracture concern. He will also throttle HPI flow to prevent excessive RCS pressure that could lead to NDT violation and/or unnecessary lifting of the pressurizer safety valves.

If primary to secondary heat transfer does not exist even after the subcooled margin is restored, it is probably due to a lack of feedwater or a blockage of RC flow due to steam voids in the hot legs. The operator will proceed to Section III.B which will provide for restoration of feedwater and primary to secondary heat transfer or establishment of HPI cooling.

If excessive heat transfer exists, the operator will proceed to Section III.C. The operator should also be alert for indications of a small break (refer to Appendix F in Volume 2) as a small break could exist that is within the capacity of the HPI system.

If controlled primary to secondary heat transfer exists then the operator should regulate feedwater flow to establish SG levels at the appropriate setpoint (dependent upon whether RC pumps were restarted). The operator

should also control steam pressure to prevent RCS reheating and swell. If the RCS were allowed to reheat, the added inventory from HPI could result in a large pressure increase and possibly a full pressurizer.

Subcooled Margin not Restored

If full HPI flow does not restore the subcooled margin even with heat transfer to the SG's, it is a LOCA. If heat transfer deos not exist or exists in only one SG, the operator will proceed to Section III.B to attempt restoration of heat transfer to both SG's. The prolonged period at saturation may be due to a total loss of feedwater or a small break. In either case, restoration of feedwater flow in III.B will aid primary cooling. If, however, the CFT's begin to empty, a large break exists and primary to secondary heat transfer cannot be regained. In this case the operator will go to CP-101 for long term cooling following a major LOCA.

NOTE: Whenever adequate subcooling margin does not exist, the operator should be alert to indications of superheat in the RCS (The RTD's and incore thermocouples read higher than saturation temperature for the existing RCS pressure). If indications of superheat occur, the operator should proceed to the Inadequate Core Cooling (ICC) guidelines.

Summary

The bases for Section III.A can be summarized as follows:

Symptom: RCS pressure-temperature to the right and/or below the subcooled margin curve.

Problems: a) Possible LOCA

b) Void formation in RCS at saturation can interrupt core cooling

Objectives: a) Restore subcooled margin

b) Maintain or restore core cooling:

i. preferably with SG's

ii. with HPI cooling (after transfer to Section III.B) if SG cooling unavailable

Key Points: a) Be alert for indications of ICC

b) Lack of subcooled margin can severely hamper primary to secondary heat transfer

c) HPI cooling, while adequate for interim core cooling, is not a stable long-term cooling mode. Cooling with one or both SG's must be restored as soon as possible. (HPI cooling is discussed in more detail in the "Backup Cooling Methods" chapter).

Section III.B: Follow-up Actions for Treatment of Lack of Primary to Secondary Heat Transfer in Either OTSG

If adequate subcooled margin exists, the most likely cause for lack of primary to secondary heat transfer is no heat sink (loss of feedwater). The operator will take steps to restore feedwater. If he cannot restore feedwater he will establish HPI cooling and he should reduce the number of running RC pumps to one to minimize heat input to the RCS. This will provide adequate core cooling while he continues efforts to restore feedwater to at least one SG.

Lack of Adequate Subcooling Margin

If, however, adequate subcooled margin does not exist, then the lack of heat transfer could be due to no heat sink (loss of feedwater) and/or no reactor coolant flow (hot leg voiding). The primary objective is to restore core cooling. The preferred method of core cooling is with primary to secondary heat transfer.

As in Section III.A, the operator will trip the RCP's and initiate full HPI flow when the subcooling margin is lost. If the operator has feedwater flow and a level in at least one SG, then the lack of heat transfer is due to lack of RC flow. If the CFT's are emptying, a major LOCA has occurred and there is no benefit in restoring primary to secondary heat transfer. In this case the operator will proceed to the procedure for long term cooling following a major LOCA. If a major LOCA has not occurred, he will attempt to induce natural circulation flow by raising

his SG level(s) and lowering SG pressure. If this fails, and RCP's are operable, he will attempt to induce natural circulation flow by bumping an RC pump. With the SG available as a heat sink, bumping an RC pump will force steam voids in the RCS into the SG tubes where they can be collapsed. Fifteen minutes should be allotted between successive pump bum to allow natural circulation flow to build. If natural circulation flow is still not established after all operable RCP's have been bumped and one hour has passed since the reactor trip, a pump should be started and run, if possible, in a loop with the SG available as a heat sink. The hour limitation is based on allowing delay heat to decrease to a level that the HPI flow can accommodate so that additional inventory loss through the break due to forced flow is no longer a concern.

If the RCP's are not operable, he must cool the core with HPI and he will go to CP-104 for HPI cooling. If he is successful in establishing primary to secondary heat transfer then he will go to the appropriate cooldown procedure depending on the degree of subcooling. If the subcooled margin is not restored then a small break probably exists.

Summary

The bases for Section III.B can be summarized as follows:

Symptoms: a) With subcooled margin, symptoms are those indicative of loss of feedwater:

i. RCS reheating and repressurizing after normal post-trip cooldown.

ii. Low or non-existent SG levels and feedwater flowrates.

b) With lack of adequate subcooled margin, symptoms could be as above for loss of feedwater and/or small break symptoms (see Appendix F in Volume 2).

Problems:

a) Lack of core cooling

b) Extended loss of feedwater will lead to saturation and loop voiding-necessitates HPI cooling.

c) Possible LOCA.

Objectives:

a) Maintain or restore subcooling margin.

b) Restore core cooling, preferably using the SG's.

Key Points:

a) Lack of heat transfer with adequate subcooling is more than likely due to total loss of feedwater or insufficient feedwater to induce natural circulation.

b) Every effort must be made to restore primary to secondary heat transfer (unless a Major LOCA occurred). HPI cooling will be adequate for short-term core cooling but is not a stable long-term cooling mode (see HPI cooling section in "Backup Cooling Methods" chapter).

Section III.C: Followup Actions for Treatment of Too Much Primary to Secondary Heat Transfer

Excessive heat transfer is always caused by a failure in the control of secondary side parameters, resulting in a loss of steam pressure or excessive feedwater flow or a combination of both. The overcooling places large thermal stresses on the RCS piping and components and on the steam generators. It can also lead to saturation of the primary system if the RCS contraction is large enough to drain the pressurizer. The primary objective of this section is to terminate the overcooling transient and then to restore controlled decay heat removal.

HPI is initiated if pressurizer level is low and RCS pressure decreasing in an effort to prevent drainage of the pressurizer and loss of subcooling. The operator will then check to see if the SG causing the overcooling can be identified. The best method to identify the affected SG is to compare T_{cold} temperatures. The loop with a significantly lower T_{cold} is the loop with the affected SG. However, T_{cold} temperatures can be fairly close together even when only one SG is causing the overcooling (refer to the discussion on overcooling transients in the "Diagnosis and Mitigation" chapter). Since the primary objective is to first terminate the transient, both steam generators should be isolated if there is any doubt which SG is affected.

In either case (one or both SG's isolated), the affected SG will either:

a) stabilize in level and pressure if the overcooling was due to excessive feedwater or isolable steam leak, or

b) continue to lose pressure and level due to an unisolable steam leak.

If a), then controlled decay heat removal can be restored using both SG's (being careful not to unisolate a steam leak). If b), then the SG with the steam leak should be allowed to boil dry while decay heat removal is established with the intact SG.

Once controlled decay heat removal is established, the operator should check for indications of a tube rupture (since the tubes have been stressed by the overcooling) and verify adequate subcooling margin exists. If the affected SG was returned to service, then the check for tube rupture can be made using steam line monitors. If the affected SG was left isolated, then the operator should check for indications of a small break LOCA and continued cooling of the RCS by the isolated SG (due to boiloff of tube leakage). If adequate subcooling margin does not exist he should recycle to Section III.A.

Whenever an overcooling transient has been terminated, the operator should hold RCS temperatures at the existing values. If the RCS were allowed to reheat, the added inventory from HPI could result in a large pressure increase and possibly a full pressurizer.

Summary

The bases for Section III.C can be summarized as follows:

Symptoms:

a) Decreasing T_{colds} and/or SG pressures significantly below the normal post-trip cooldown.

b) Possibly high SG level(s) and feedwater flowrate(s).

c) Low RCS pressure and pressurizer level.

Problems:

a) Thermal stresses on RCS components and SG's (tubes).

b) Possible RCS saturation due to pressurizer drainage.

c) Excessive feedwater flow could result in water carryover into the main steam lines.

Objectives:

a) Prevent loss of subcooling margin due to RCS contraction and drainage of the pressurizer.

b) Terminate the overcooling.

c) Re-establish controlled primary to secondary heat transfer.

Key Points:

a) Comparison of loop T_{cold} temperatures is best method for identifying affected SG before SG isolation.

b) Comparison of SG levels and pressures is best method for identifying affected SG after both SG's are isolated.

c) Severe overcooling can induce tube leaks.

d) The RCS should not be allowed to reheat after the transient is terminated.

e) Unisolable steam leaks require boiling the affected SG dry to stop the overcooling.

Section III.D: Followup Actions for OTSG Tube Rupture

Several concerns exist whenever indications of a steam generator tube rupture become evident. In addition to being a LOCA, the primary inventory lost through the tube leak cannot be recovered for sump recirculation as it would for other LOCA's. Thus, it is important to cooldown and stop the tube leak before makeup capacity (BWST) is lost. However, it is also important to minimize offsite releases. Therefore, if at all possible, it is desirable to perform a controlled power runback and reactor shutdown at a power level less than the capacity of the TBV's rather than trip the reactor at high power. This prevents lifting of the safety valves on the SG with the tube leak which helps reduce overall releases to the atmosphere. Thus the primary objectives in mitigating a tube rupture are to minimize offsite releases and total tube leakage by performing an orderly but expedient shutdown and cooldown. Opening the TBV's before tripping the turbine and reactor (when power is less than total TBV capacity) will prevent lifting of the main steam safety valves.

Reactor Trip

If a reactor trip should occur or be required because the tube leakage exceeds HPI capacity, then it is important to ensure proper plant response, particularly with respect to steam and feedwater control. If a loss of subcooled margin occurs the RC pumps must be tripped and full HPI initiated. These actions are required for the same reason as in any small

break but, in addition, it is very important with a tube rupture to prevent void formation in the hot legs which could block coolant flow and severely hamper the cooldown. It is also very important to terminate excessive heat transfer should it occur. Uncontrolled cooldown could also result in void formation in the hot legs and could overstress the SG tubes resulting in larger leak rates. Thus, if excessive heat transfer is indicated, the operator is directed to Section III.C to correct the condition before continuing in Section III.D.

During the period that the operator is performing the shutdown or stabilizing the plant after a trip, a survey of the steam lines should be made to verify which SG has the tube rupture. It is desirable to isolate the affected SG as soon as possible after primary temperature is low enough to preclude lifting of the main steam safety valves.

Cooldown Methods

Two basic cooldown methods are provided for tube ruptures, designated here as "normal" and "emergency". The method to be used is determined by the tube leak rate and the existing plant status. The differences between the methods are as follows:

	Normal	Emergency
1) Cooldown rate	100F/hr	240F/hr to 500F
2) Tube/shell ΔT limit	100F	150F
3) Fuel pin compression limit	Applies	May be violated, if necessary

The factors determining use of the "normal" cooldown method are:

1) Tube leak rate within capacity of makeup system (minimal rate of BWST depletion).

2) Condenser available

3) RC pumps available

All three conditions must exist to use the normal cooldown. If any one condition is not met, the emergency cooldown method must be used. If, during the performance of the emergency cooldown, all three conditions are satisfied then the normal cooldown can be used. Conversely, if during the performance of the normal cooldown, any one of the conditions is no longer satisfied, then the cooldown must be switched to the emergency method. In either case, the SG with the tube rupture should be isolated as soon as it is identified and T_{hot} is < 540F (to prevent lifting the steam safeties on the SG with the tube leak).

Loss of Offsite Power/RC Pumps Not Running

A loss of offsite power can significantly impact the mitigation of a tube rupture and therefore power should be restored as quickly as possible. While power is unavailable a natural circulation cooldown will be required. It will be necessary to periodically steam the SG with the tube leak during a natural circulation cooldown to induce loop circulation and avoid hot leg flashing in that loop. Void formation in the hot leg would hamper the cooldown due to the inability to depressurize the RCS (the hot leg would act as a pressurizer). If the condenser were not available

steaming of the affected SG would have to be done directly to atmosphere thus increasing offsite releases. However, at the Oconee Nuclear Station the condenser will be available even during a loss of offsite power due to gravity feed of cooling water to the condenser.

In addition, while RC pumps are not available, RCS pressure reduction must be accomplished using the pressurizer relief. This is a less desirable method since it repeatedly challenges the relief valve, results in additional inventory loss, and may degrade the reactor building environment. The cyclic operation between 50F subcooling and the minimum subcooling margin is a compromise between the need to limit the cycles on the relief valve and the need to maintain as low a pressure as possible to minimi[ze] the tube lekage rate.

RC Pumps Running

With RC pumps available, steaming of the affected SG can be reduced to the minimum necessary to keep the steam pressure less than 1000 psig and the level less than 95% on the operate range. This is required to preven[t] lifting of the steam safeties (additional release to atmosphere) and to prevent water spillover into the steam lines. Forced circulation will prevent the formation of steam voids in the idle (non-steaming) loop.

RCS pressure control is better with RC pumps and spray available. Therefore, spray should be used as necessary to maintain the RCS pressure as close as possible to the minimum subcooling margin to minimize the leak rate.

Continued Cooldown and Isolation of the Affected SG

The plant is not stable after a tube rupture until the tube leakage has been stopped. This will require cooldown and depressurization and DHRS operation to the point where the RCS can be drained to below the elevation of the tube leak. Therefore, the cooldown should progress as expediently as possible.

Once the affected SG has been isolated, it should only be fed and/or steamed as necessary to maintain steam pressure < 1000 psig and level <95% on the operate range and to maintain the tube/shell ΔT within the limits. Excessive tube/shell ΔT could result in a higher leak rate due to the increased tensile stress on the failed tube. In addition, as previously stated, steaming of the isolated SG may be required during a natural circulation cooldown to prevent void formation in the hot leg.

Summary

Symptoms: a) high radiation in steam lines and/or condenser

b) LOCA symptoms (decreasing pressure, unaccountable RCS inventory loss, etc.)

Problems: a) SG tube rupture/LOCA

b) unrecoverable RCS inventory loss (i.e., not available for sump recirculation)

c) offsite releases

Objectives: a) minimize offsite releases

b) terminate leakage before BWST depletion

c) maintain core cooling/expedient cooldown and depressurization

Key Points: a) transient (leakage) not terminated until RCS cooled, depressurized, and drained below tube leak elevation

b) LOOP/natural circulation cooldown will probably re in higher offsite releases due to greater need to steam affected SG

c) natural circulation cooldown necessitates use of pressurizer relief to reduce RCS pressure

Cooldown Procedures/Inadequate Core Cooling

The objective of these guidelines is to maintain adequate core cooling terminating transients and stabilizing the plants with controlled decay heat removal. Once stable conditions are achieved, further plant cooldown can be accomplished by existing plant procedures. However, the end conditions at stabilization following the execution of the guidelines will not necessarily coincide with the entry conditions for plant cooldown procedures. Therefore, procedures are provided in Part I to accomplish t transition from the guidelines to the plant procedures. Five cooldow procedures are provided to cover the five possible end conditions of the guidelines:

1) Cooldown following a large LOCA

2) Normal cooldown

3) Saturated cooldown with primary to secondary heat transfer

4) HPI cooling

5) Solid plant cooldown/recovery from solid plant.

A sixth procedure is provided for the special case of Inadequate Core Cooling (ICC). The philosophy and the objectives of the actions for ICC are discussed in detail in the "Backup Cooling Methods" chapter in Volume 1 of Part II.

At the end of the section containing the five cooldown procedures (immediately before the ICC section) are specific rules. These rules are provided in a separate section to avoid repetition throughout Section III of the guidelines. These rules apply wherever they are referenced in Section III. Four specific rules are provided to cover:

1) Initiation of HPI
2) HPI flow control
3) Feedwater throttling methods
4) SG level setpoint.

In addition, three figures are provided at the back of Part I for easy reference during the use of the guidelines. These figures provide:

1) HPI flow vs. RCS pressure
2) RCS pressure-temperature limits for brittle fracture/NDT
3) Core exit thermocouple temperature for ICC

Operator Aids

In addition to the guidelines in Part I and the training material in Part

II, two other developments of the ATOG program can provide significant assistance to the operator during the mitigation of abnormal transients: the pressure-temperature (P-T) display and the System Auxiliary Diagrams (SADs). The ATOG user should provide for full utilization of these aids in the implementation of the guidelines.

P-T Display

The information required to identify and track the symptoms discussed previously is already available in the control room. However, it consists of discrete displays for reactor coolant system hot and cold leg temperatures, reactor coolant system pressure and steam generator pressure. This format requires mental integration on the part of the operator to quick assess plant cooling status using individual displays and the steam tables. Thus, the problem is how these variables could best be displayed in real time to give the operator a simple and logical method of monitoring the symptoms of interest. The solution developed in ATOG was the use of a P-T display on a cathode ray tube (CRT). The display continuously shows the primary subcooling margin and the dynamic relationship the primary to secondary heat transfer. The particulars of the display format and identification of the symptoms is discussed in detail in the "P-T Diagram" chapter. This section will discuss the various functions the display can perform to aid the operator.

Symptom Identification

The primary purpose for developing the P-T display is to provide the means

for the operator to monitor the plant response following a reactor trip or during a forced shutdown to verify normal response and to quickly identify abnormal response should it occur. The "P-T Diagram" chapter describes the normal post-trip cooldown of the RCS and the various trends that can develop when the response is abnormal. With the P-T display the operator can quickly recognize the loss of subcooled margin, lack of heat transfer or excessive heat transfer and proceed directly to the appropriate section of Part I to restore plant stability and core cooling.

Response Verification

The P-T display also provides positive feedback to the operator on the response of the plant to his actions. For example, after the loss of subcooling margin, the operator can easily determine the effectiveness of full HPI flow by monitoring the P-T display and determine when to throttle HPI flow if the subcooling margin is restored.

Controlling Within Limits

Certain operations require that the operator control the primary system within specified P-T regions that can readily be displayed on the P-T diagram. For example, the guidelines for a natural circulation cooldown with a tube rupture require maintaining RCS pressure between 50F subcooled and the subcooled margin line. If the operator has the capability to select a 50F subcooled curve for display on the CRT (the subcooled margin line is already displayed) then he has a simple, convenient format for monitoring

the cooldown and ensuring compliance with the limits. Many other examples exists, including uses during normal plant cooldowns and heatups (e.g., fuel pin compression limits, NDT limits, etc.).

Power Operation

The CRT format is readily adaptable for displaying plant status information during normal power operation. Some examples for such usage are the reactor protection system pressure-temperature trip envelope (shown in the "P-T Diagram" chapter) and power imbalance envelope. However, if the ATOG CRT is used for these displays, they should be of a secondary nature with the CRT automatically reverting to the ATOG P-T display on reactor trip

Backup for the CRT

It can readily be seen that the availability of a P-T display improves the flow of information to the operator and enhances the use of Part I. Effort should be made in the implementation of CRT displays to provide high reliability.

However, control room personnel should allow for the possibility that the CRT displays are unavailable when needed. Provisions should be made to facilitate hand plotting of the parameters on a P-T diagram similar to the diagrams depicted herein. Hand plotting is quick enough to provide data which can be used for plant control. The format of the diagram for hand plotting (with the saturation and subcooled margin lines pre-drawn) would allow for trend diagnosis and still be a significant improvement over mental assimilation of discrete data displays.

System Auxiliary Diagrams

System Auxiliary Diagrams (SADs) were developed in the ATOG program to identify supporting systems essential to the operation of systems having direct input to plant response. They also identify instrumentation required to verify proper operation of supporting systems.

The SADs serve as a useful aid in the event that a critical system fails. For example, the operator may be required to initiate HPI. In the highly unlikely event that a total loss of HPI occurs, the associated SAD can be used as a rapid troubleshooting aid to restore HPI operation. The SAD for HPI shows HPI in the center and various arrows pointing toward HPI which identify everything required to make HPI initiation successful. Pump power supplies, required cooling/lube oil sources, major inline valve positions, ventilation cooling, etc. are all identified along with available instrumentation to verify proper operation of the HPI system. Only those items that are within the operators ability to control and can be accomplished quickly are included. Corrections that are longer term (e.g., replacing a pump impeller) are omitted.

Since troubleshooting will be performed by roving operators or maintenance groups, the SADs are packaged separately as opposed to being contained in the ATOG volumes. However, the appropriate SADs are reference in Part I where applicable. Station management will determine the availability and use of the SADs.

SADs have been developed for the following systems and components:

1. Main feedwater system (loss of flow).
2. Emergency Feedwater System (loss of flow).
3. Steam line components (loss of steam pressure).
    a. Turbine bypass valves
    b. Main steam safety valves
    c. Atmospheric dump valves
    d. Turbine controls
4. ECCS Systems (failure to deliver water).
    a. Makeup
    b. HPI
    c. LPI
5. Containment cooling systems (failure to depressurize containment)
    a. Building spray
    b. Building coolers
6. Containment isolation (failure to isolate).
7. Boron addition (inability to add boron).
8. Components for RC pressure control
    a. Pressurizer heaters
    b. Pressurizer spray