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Light Water Reactor Status Monitoring During Accident Conditions

Prepared by J. vonHerrmann, R. Brown, A. Tome

EG&G Idaho, Inc.

Science Applications, Inc.

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FOREWORD

The accident at Three Mile Island in March, 1979, and the results of subsequent investigations have reemphasized the importance of reactor operators and the role they play in determining the level of safety associated with nuclear power. At the same time, the adequacy of some long-standing regulatory approaches to safety, such as design basis events and the single failure criterion, are being questioned. Alternate methods, some employing insights from probabilistic risk assessment, are being proposed in order to broaden our perspectives on reactor safety.

This report introduces some important new concepts and technical approaches which, if properly developed and applied, could make significant contributions to accident analysis. It emphasizes the perceptions of the operator, the needs for information and the alternative successful actions one might take given various combinations of component failures. The methods are potentially useful for determining instrumentation requirements, developing emergency procedures, generating training simulator exercises, and designing operational aids, including computerized diagnostic systems.

Among the purposes of this report are to expose these ideas to potential users, to solicit their comments, and to encourage others to utilize this or similar techniques so that they may generate additional insights toward improving reactor safety.

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ABSTRACT

A novel technical approach for systematically determining information needs during reactor accidents is proposed. The method is used to identify the necessary and sufficient set of light water reactor instrumentation needed by analyzing the appropriate operator response to specific plant states associated with risk significant accident sequences. The resultant set of measureable parameters is compared to the list of such parameters in Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs During and Following An Accident."

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LIST OF ACRONYMS

GENERAL

BWR - Boiling Water Reactor
ECCS - Emergency Core Cooling System
ESF - Engineered Safety Feature
EP - Electric Power
LOCA - Loss-of-Coolant Accident
PWR - Pressurized Water Reactor
RCS - Reactor Coolant System
RSS - Reactor Safety Study
RWST - Refueling Water Storage Tank

ESF FUNCTIONS

ECC - Emergency Core Cooling
ECI - Emergency Coolant Injection
ECR - Emergency Coolant Recirculation
RT - Reactor Trip

ESF Systems

PWR

ACC - Accumulators
AFWS - Auxiliary Feedwater System
CLCS - Consequence Limiting Control System
CHRS - Containment Heat Removal System
CSIS - Containment Spray Injection System
CSRS - Containment Spray Recirculation System
CST - Condensate Storage Tank
CVCS - Chemical Volume Control System
HPIS - High Pressure Injection System
HPRS - High Pressure Recirculation System
LPIS - Low Pressure Injection System
LPRS - Low Pressure Recirculation System
PCS - Power Conversion System
RPS - Reactor Protection System

BWR

ADS - Automatic Depressurization System
CSIS - Core Spray Injection System
CSRS - Core Spray Recirculation System
HPCIS - High Pressure Coolant Injection System
HPSWS - High Pressure Service Water System
LPCIS - Low Pressure Coolant Injection System
LPCRS - Low Pressure Coolant Recirculation System
RCICS - Reactor Core Isolation Cooling System
RHRS - Residual Heat Removal System
RPS - Reactor Protection System
SHA - Sodium Hydroxide Addition

1.0 INTRODUCTION

In recent years, increased attention has been focused on the performance of nuclear reactor operators and on the quality of the interface between the operator and the systems for which he is responsible. This emphasis has resulted in part from the recognition that the overall public risk associated with a nuclear plant is sensitive to the manner in which the human operators perform under both normal and accident conditions. While many plant safety functions are performed automatically, and numerous backup safety systems exist to protect the public, the reactor operator has a crucial role to play in both avoiding upset conditions and in bringing the plant to a safe shutdown condition following the initiation of a potential accident sequence.

In addition, recent experience has demonstrated that significant improvements are both necessary and possible in the quality of this man/machine interface. The accident at Three Mile Island brought national attention to this problem and most of the subsequent analyses of this incident recommended design and/or procedural changes aimed directly at enhancing the operator's capability to diagnose and respond to potential accident conditions. As a part of the Nuclear Regulatory Commission's safety research plan, an Enhanced Operator Capability Program has been initiated to comprehensively address the ability of reactor operators to respond to off-normal conditions. This report represents one of the initial efforts in that program.

The general task of improving the operator/plant interface involves many varied aspects of engineering, design, and operation. This problem has been, and continues to be, the subject of numerous studies by organizations throughout the nuclear industry. These groups have approached the problem from many different directions and on many levels, ranging from the determination of what color a flashing light should be to the initial attempts to design a totally computerized disturbance analysis system which could effectively remove the human from the problem.

The analysis reported here is based on two observations concerning the enhancement of operator capabilities:

1) The operator's capability to both diagnose and respond to accident conditions is very sensitive to the amount and quality of information available to him through the plant instrumentation. Accordingly, one of the primary objectives of this analysis was to systematically determine the necessary and sufficient set of plant instrumentation which would satisfy the operator's informational needs during accident conditions.

2) While there exist many diverse aspects of the general operator/plant interface problem, any efficacious changes to present designs and/or procedures must be based upon a foundation consisting of a thorough understanding of the plant response to accident events and a careful delineation of the specific responsibilities of the operator as the accident sequence progresses. Therefore, an additional objective of this analysis was to develop such a foundation upon which both this and additional analyses concerning enhanced operator capability could be performed.

In the following sections the specific goals of this initial analysis are more fully explained and the technical approach selected to accomplish these goals is discussed.

One of the key aspects of the selected technical approach was the judgment that the analysis should focus on those accident conditions which the operator is most likely to be confronted with and/or result in the most serious consequences should the operator fail to accomplish his required tasks. For this reason, a probabilistic risk criterion was adopted as the basis for identifying important accident conditions and required operation actions. This approach is discussed in much greater detail in Section 3.0.

Following the discussions of objectives and approach, the results and conclusions are presented and preliminary recommendations are made concerning the necessary and sufficient plant instrumentation. Since these results will include a listing of those specific parameters identified as necessary for monitoring plant behavior and a logical justification for their selection, the analysis can also provide input to the revision of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident." Reflecting the continuing nature of this analysis (of which this report represents the first step), recommendations will also be made concerning the need for and value of subsequent analyses.

2.0 OBJECTIVE

The ability of the operator to successfully respond to accident conditions is highly sensitive to the amount and quality of information he can obtain concerning the state of the plant. This information can only be provided to the operator by the plant instrumentation. Thus, the careful selection and design of the specific instrumentation intended to provide this information in an unambiguous manner to the operator is an effective contribution to increased operator performance and, thereby, plant safety. It was, accordingly, the objective of this analysis to systematically determine the plant instrumentation required to supply the operator with the necessary and sufficient information to allow him to unambiguously determine the status of the plant under accident conditions, and thereby allow him to take the most effective action to bring the plant to a successful shutdown.

The above statement of objective contains a few key words which significantly impacted the manner in which the work was performed: "systematic", "necessary and sufficient", and "unambiguous". One of the major problems an operator must contend with in responding to an upset condition is the fact that many different accident sequences requiring different operator responses often "look" the same to the operator if he confines his attention to only a few fundamental plant parameters. For example, a small LOCA, a steam generator tube rupture, and an overcooling transient are all characterized by an initial reduction in primary system pressure. Thus, sufficient additional information must be available to allow the operator to differentiate between these various events and unambiguously determine the state of the plant. However, an important aspect of this analysis was the recognition that merely listing an enormous number of potentially useful instruments does not adequately address the operator's problem; the quality of the information is as important as the amount. Because the operator cannot be expected to effectively assimilate the

information from a myriad of sources under stressful accident conditions, and because of the extreme costs associated with the installation of many instruments, the instrumentation must meet the requirements of being both necessary and sufficient. In order to satisfy these requirements, a systematic logical approach to the investigation was necessary.

The objective of the analysis was not, therefore, only to generate a list of required instrumentation, but to accomplish this task in a manner which would result in confidence that detailed justification exists for every member of the list and no necessary instrument is excluded from the list. Accordingly, an intermediate objective was to develop this required systematic approach and to confirm its effectiveness. Additionally, as discussed briefly in the Introduction, it was desired that the selected approach would be able to provide the framework upon which additional subsequent investigations aimed at the general goal of enhancing operator capability (e.g., developing criteria for a computerized disturbance analysis system) could be built. In the following sections the selected approach (which is based on the use of event trees to explicitly delineate important accident sequences and to define the information required by the operator to take action designed to terminate the sequence) is presented and an example sequence is discussed. In Section 5.0, conclusions regarding the value of this approach are presented.

3.0 TECHNICAL APPROACH

The method used in this analysis to accomplish the objectives outlined in the previous section was based on evaluating operator response in a logical progression of investigations. This approach can be succinctly summarized by addressing three fundamental questions:

1) What actions can (or must) the operator take in response to the accident condition?

2) What information is required by the operator to take this action?

3) What instrumentation is necessary and sufficient to provide this information?

By translating the general objective into these three interrelated questions which represent a logical progression of investigations, the analysis could be performed very systematically, producing maximum assurance that important operator informational needs will not be overlooked.

The first question listed above, which represents the foundation upon which the remainder of the analysis is constructed, focuses directly upon the role of the operator. Obviously, what an operator could or should do depends upon what specific accident sequence he is responding to. Similarly, the operator can act effectively only when he knows what he is trying to avoid, or, alternatively, what he is trying to obtain. This means he must know the potentially dangerous outcomes of proceeding along any particular accident sequence and also be aware of alternative pathways off this accident sequence which result in successful termination of the sequence. These observations suggest that event trees would provide a very effective logical framework for answering this first question.

In addition to the event trees which can explicitly delineate specific accident sequences, the need existed for some criterion by which the truly important accident sequences could be selected from the multitude of potential sequences produced by these event trees. The criterion used in this analysis was the relative amount of public risk associated with each accident sequence as calculated in the Reactor Safety Study.[1] In this way, the operator's informational needs can be justified on the basis of both the probability that the operator will actually be confronted with a particular accident condition and on the potential consequences should the operator not respond adequately to the situation.

Thus, a very significant aspect of the approach taken in this analysis was the decision that the justification for the need for any instrument should be based on an explicit identification of required operator responses to selected accident conditions and, further, that the principal criteria by which these accident conditions are selected should be public risk.

The decision to approach the problem in this way resulted in a number of tasks related to the development and use of these event tree models to answer the three basic questions listed above. These tasks and their interrelationships are illustrated on the flow chart seen in Figure 3.1 and are discussed in greater detail below.

The first step in the analysis was to select the important accident conditions which the operator must respond to and to develop the event trees associated with these accident situations. For this investigation, the accident sequence was terminated at either successful shutdown or at core melt. Operator actions beyond this point were not considered (e.g., monitoring of effluent release from containment was not included).

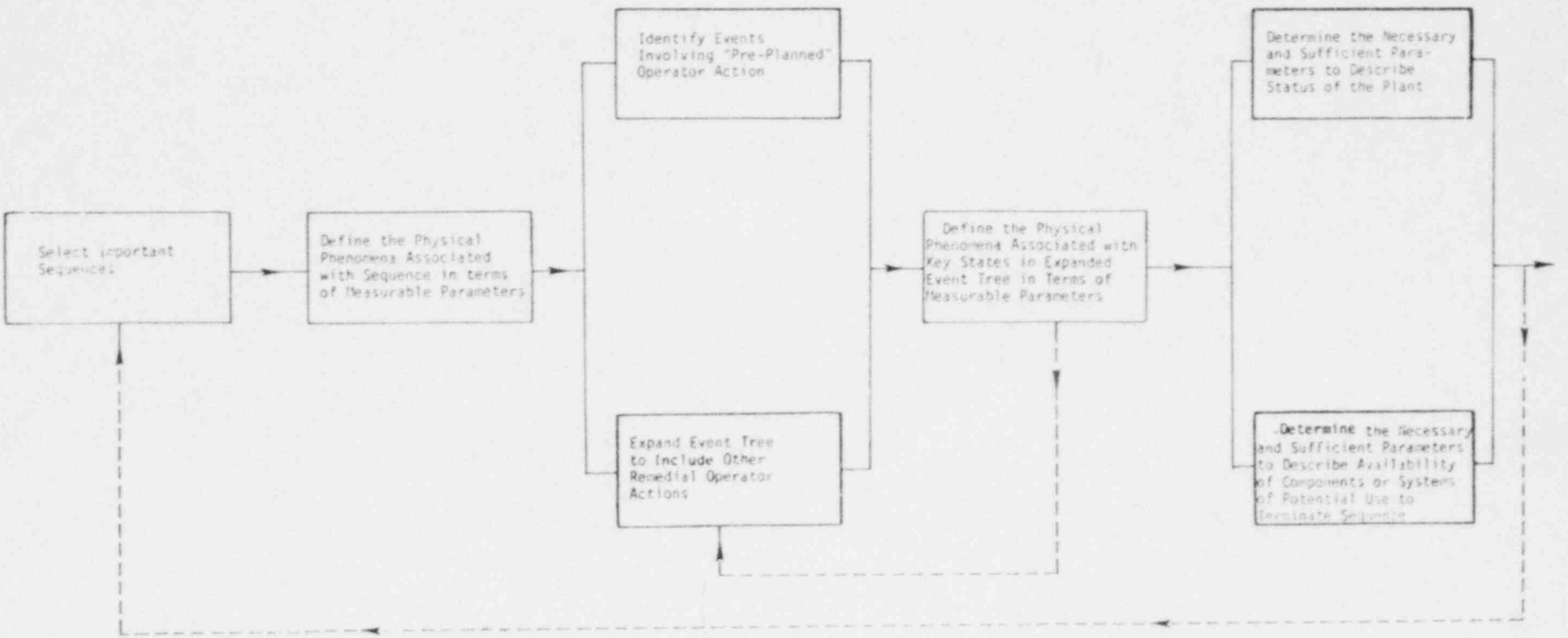

Figure 3.1. Task Flow Chart

Note: Dotted Lines Imply Possible Feedback in an Iterative Process

The decision was made early in the analysis that this selection of sequences and development of event trees should be based (to the greatest extent possible) on previously completed analyses. This would reduce, to a large degree, duplication of effort, and allow this investigation to concentrate its resources on the specific concerns of operator/plant interaction. By using existing accident analyses, these evaluations can be tied to a framework which is already familiar to the nuclear industry in order to facilitate review and comment. The sequences selected were those identified in the Reactor Safety Study (which examined the Surry PWR design and the Peachbottom BWR design) and the Sequoyah Reactor Safety Study[2] as being the dominant contributors to public risk. For the purposes of this initial investigation, a series of seven representative accident sequences was selected from the WASH-1400 and Sequoyah studies:

WASH-1400: V, S_2C, TMLB', TC

Sequoyah: S_1HF, S_2HF, TML

As can be seen from this list, this initial investigation focuses on PWR sequences. A representation BWR sequence, TC, was also included. Recommendations for extending this analysis, particularly with respect to BWR sequences, are discussed in Section 6.0.

Along with the advantages noted above of using the event tree analyses performed in these previous studies, it was necessary to accept the disadvantages associated with limiting this analysis and the resultant conclusions to three specific plants. The effects of this plant specificity are discussed in greater length in the conclusions of this report and recommendations are made which are aimed at mitigating these effects.

The next step is to define the physical phenomena associated with each sequence in terms of measurable parameters. The time dependent variations and the interrelationships of these parameters generate an

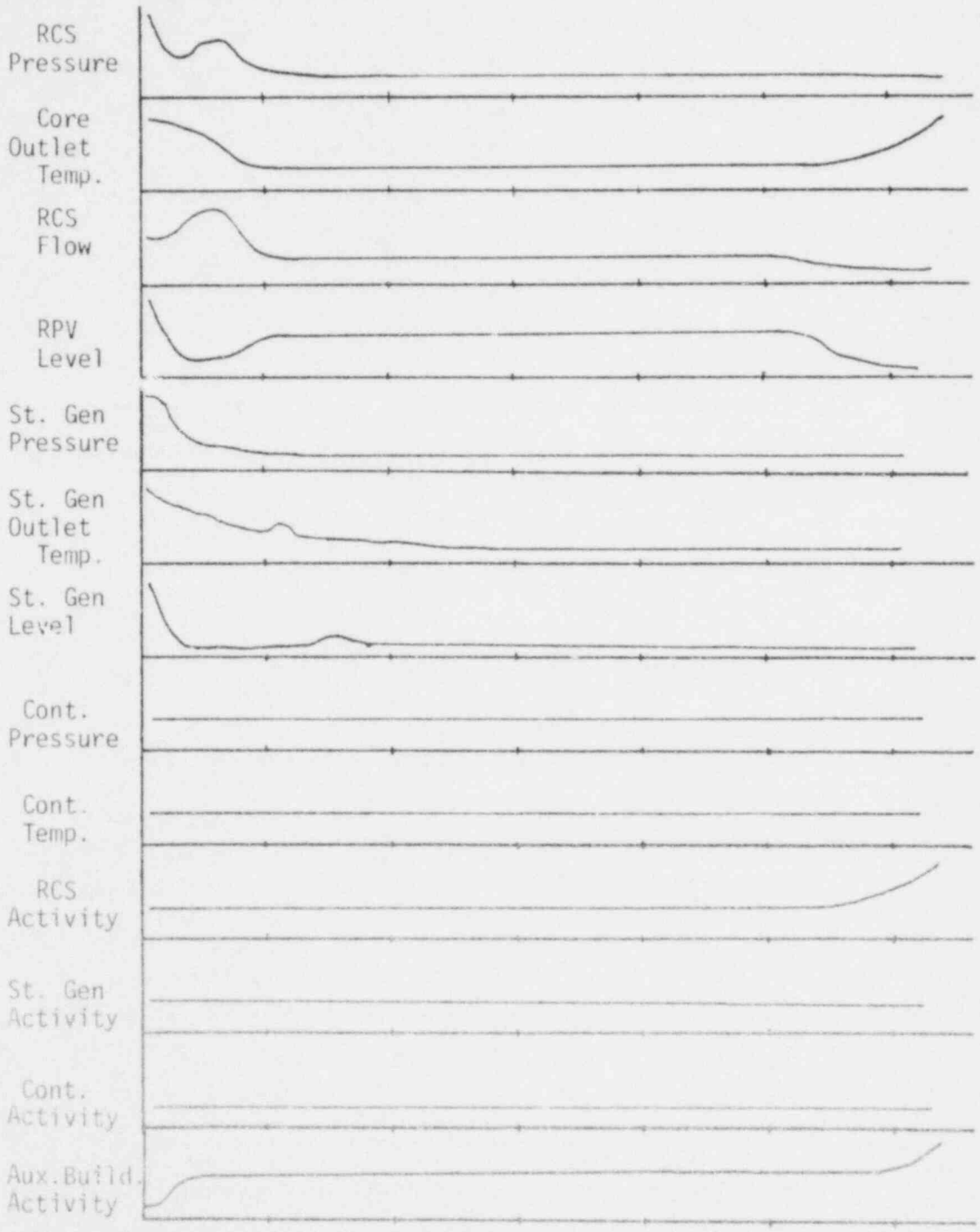

Figure 3.2. A Hypothetical Accident Signature; the time dependent variations and interrelationships of the parameters can uniquely characterize an accident sequence.

"accident signature" - a uniquely characteristic array which can be used to help the operator diagnose the status of the plant. Such an array might appear as shown in Figure 3.2. (This figure is intended only to illustrate the concept of an "accident signature"). This development of "accident signatures" in which each accident condition (or group of conditions which are similar with respect to required operator response) has associated with it a fundamental (and unique) set of parameter states necessitated gathering information from a variety of sources. The two major sources for this analysis were the investigations performed in support of the Reactor Safety Study and subsequent analyses performed for the NRC by Battelle Columbus Laboratories using the MARCH computer code package. As discussed in Section 5.0 below, this task of developing detailed accident signatures is a crucial element of the overall effort to enhance operator capability and should continue as a central part of investigations subsequent to this analysis.

The development of the event trees began with the trees as they appeared in the original reports. The events in each sequence which involved operator action were identified and in some cases broken down into additional events in order to separate out and highlight individual operator tasks. In addition, the sequences were expanded (events added to the event tree) to include additional operator actions which could be performed to prevent core melt, but were not taken credit for in the original analysis. These additional events usually took the form of "repair events" where the operator is given the opportunity to attempt to fix a component or system failure or "delay events" where the operator is called upon to delay an inevitable melt as long as possible, or to perform some other consequence mitigation action. The result of these efforts was an "operator action event tree" which logically displayed the role of the operator throughout the progression of the accident. Figure 3.3 presents a simple example of such a tree developed for the "V" sequence of WASH-1400. This sequence, which involves the interfacing systems LOCA, is discussed in detail in

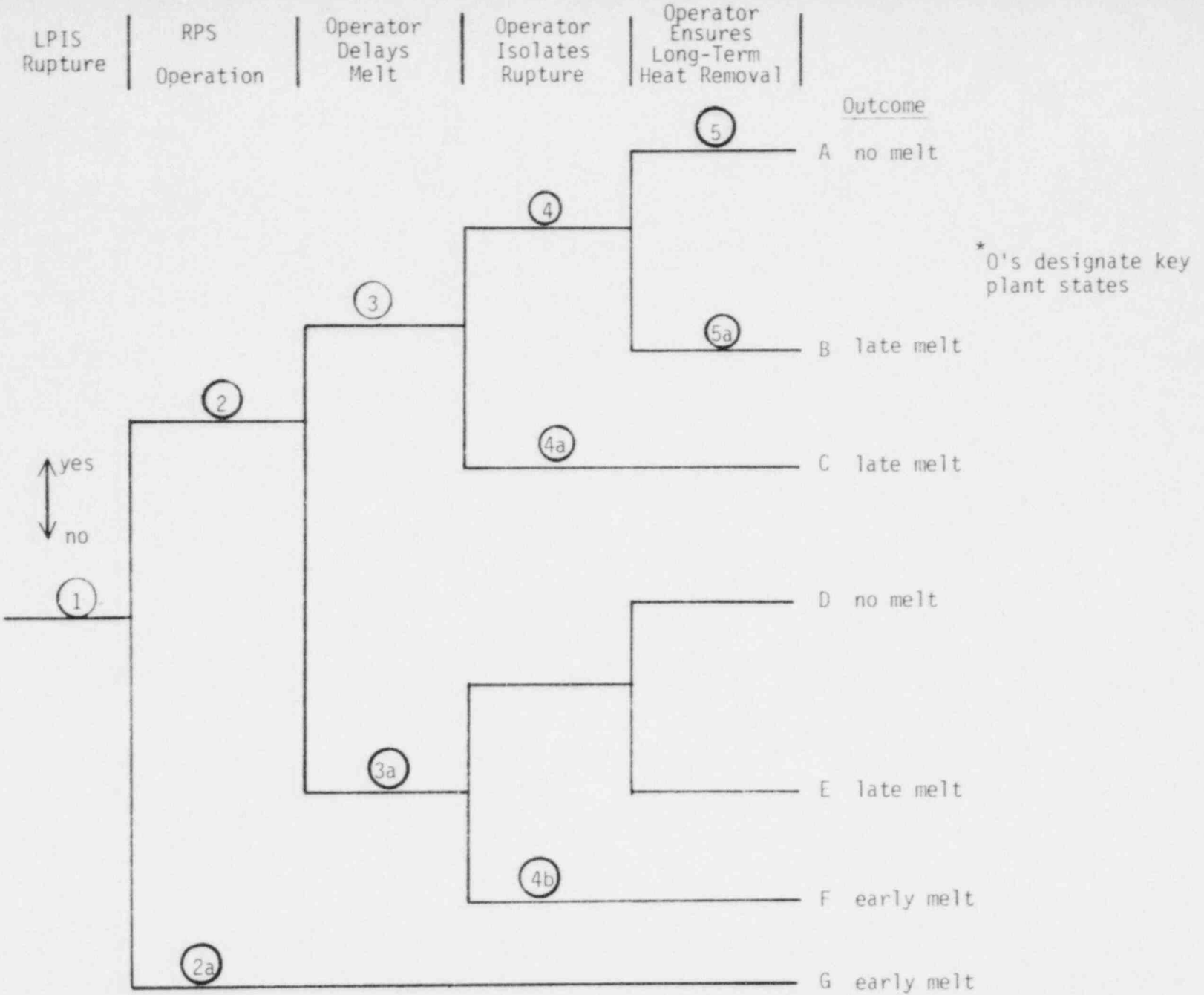

Figure 3.3. Interfacing Systems LOCA Operator Action Event Tree

Section 4.0. As can be seen from this figure, the headings for this tree involve the major operator tasks in response to the postulated failure event(s). In this case, these tasks include such actions as isolating the rupture, delaying core damage, establishing long-term heat removal, etc.

The event tree associated with these headings defines a series of key plant states that could evolve as the accident progresses and the operator attempts to respond. For example, the plant state designated as (2) in Figure 3.3 represents the situation which would exist following the initiating LOCA event and successful shutdown of the reactor.

The development of these operator action event trees and identification of the key plant states make it possible to address the second fundamental question listed at the beginning of this section. Associated with each plant state is an appropriate operator response. His informational needs are therefore fundamentally linked to these plant states: the operator must have sufficient information available to him to unambiguously determine the existence of the specific plant state and he must have sufficient information to allow him to efficiently take the action required of him at this point. He must be able to recognize that a situation exists which calls for his action, determine specifically what that action should be, and carry out these tasks. In terms of the event tree, he must be able to identify what tree he is on, to what branch point in the tree the accident has progressed, and how to move the progression of the incident to a pathway that terminates in successful shutdown.

The next step in the analysis was, therefore, to identify for each of these key plant states: 1) the information which would allow the operator to determine the existence of that specific state, and 2) the information necessary to take the action which has been determined to be appropriate at that state.

Since the only way that the operator can obtain information concerning the state of the plant is through the plant instrumentation, these informational needs described above have to be defined in terms of measurable plant parameters. For example, if the general informational need is the effectiveness of the High Pressure Injection System in cooling the core, this can be translated into the need to determine the amount of subcooling, which must in turn be translated into measuring temperatures and pressures. Translating these informational needs into parameters which can be physically measured defines the instrumentation necessary to supply this information, and therefore results in answering the third fundamental question associated with the objectives of this analysis.

After each key plant state has been addressed in the manner described above, a table was constructed for each accident sequence which presents the instrumentation required by the operator to respond to the plant conditions associated with that sequence (see, for example, Table 4.1).

The information contained in each of these tables was then integrated into a final table which summarizes the results of this analysis. This summary table, which lists each required measurable parameter along with the accident condition(s) and required operator action which necessitated the information provided by the parameter, is presented and discussed in Section 5.0.

Thus, the technical approach outlined above and based on the use of operator action event tree logic models allows the analysis to progressively answer the three key questions raised at the beginning of this section, and thereby accomplish the objectives detailed in the previous section. In summary, by focusing on individual accident sequences selected on a probabilistic risk basis, specific operator tasks could be identified and structured into a logic model which, in turn, identifies a series of

key plant states. These key plant states have associated with them certain informational needs. Finally, these informational needs are described in terms of measurable physical parameters, and thereby determine the necessary plant instrumentation.

The technical approach described above was designed to allow the systematic accomplishment of the goals described in Section 2.0. However, as discussed in the Introduction, it was also desired to approach this problem in such a way as to set the foundation upon which efforts to enhance operator capability extending well beyond the scope of this analysis could be constructed. It is believed that the approach described above is consistent with that desire. By structuring the investigation upon event trees which logically develop the basic functions which must be accomplished by the plant (e.g., reactivity control, primary inventory control, core cooling, etc.) and then focusing on the specific operator tasks necessitated by postulated equipment failure, this analysis can not only produce useful results concerning plant instrumentation which can stand by themselves, but can also be used as a starting point for a wide spectrum of subsequent analyses designed to further enhance operator capability. Further discussion concerning the value of this approach for other investigations can be found in Section 5.0.

4.0 EXAMPLE SEQUENCE TO ILLUSTRATE APPROACH

While the technical approach described in the previous section was designed to logically progress from one step to the next, it necessarily involved a relatively large number of interrelated steps. In order to facilitate an understanding of this approach, an application of this approach to an example accident sequence is presented in this section. The sequence selected for this purpose is the "V" sequence mentioned previously.

4.1 Sequence Description

In its evaluation of the Surry Pressurized Water Reactor, the Reactor Safety Study identified the Interfacing-Systems LOCA as the highest risk contributing accident sequence. This sequence, designated as sequence "V", is concerned with the failure of any one of three sets of two check valves which separate the Low Pressure Injection System (LPIS) and the primary coolant lines. These check valves, as they are configured for the Surry plant, are illustrated in Figure 4.1.

The significance of these check valves is that they separate a high pressure system (the primary coolant system) from a system which is not designed to withstand these high pressures and which passes outside of containment (the LPIS). Thus, failure of these valves would lead to an overpressure and subsequent rupture of the LPIS and provide a path for primary coolant loss outside of containment. Not only would the LPIS be unavailable to cope with the LOCA, but other emergency coolant injected into the primary system would flow out of the break. Emergency coolant recirculation would then be impossible due to the lack of water in the containment sump and core melt would occur soon after the supply of emergency coolant injection water in the Refueling Water Storage Tank (RWST) is depleted. Since this sequence involves a LOCA that breaches and

bypasses containment, the containment engineered safety features would be ineffective for this accident, and the status of the containment has little relevance to this sequence. Figures 4.2 through 4.6 provide information concerning the behavior of some key parameters which describe the primary system response following the interfacing systems LOCA assuming accumulator availability, but with all other ECCS components failed. As can be seen from Figure 4.6, core melt under these conditions would occur about 30 minutes after the LOCA. WASH-1400 indicates that core melt will be delayed for about another hour if more than one HPIS pump were to operate and for an additional 10 hours if only one HPIS pump were to operate.

The event tree developed for this Interfacing-Systems LOCA event in the RSS is presented in Figure 4.7 with the "V" sequence highlighted. In the following discussion it should be remembered that, for the "V" sequence, all plant safety systems successfully start-up and perform their designed functions (even though this is not sufficient to prevent core melt). Sequences which involve additional safety system failures (e.g., VD, VK, etc.) are probabilistically far less significant than the "V" sequence, and are not addressed here. Thus, in the analysis below, no additional safety system failures will be included with the exception of those attributed to incorrect operator action. In addition, it should be noted that we are concerned with operator action during and after an accident; events occurring prior to the initiating LOCA which might have contributed to the LOCA (e.g., failure of a check valve to reseat) are not explicitly addressed. Testing and surveillance procedures have been discussed in previous analyses[3] which involve the use of pressure monitors between the check valves. The information provided by these instruments should alert the operator to an abnormal condition in the LPIS line and shutdown procedures could be initiated before the LOCA occurs.

4.2 Operator Actions

As can be seen in Figure 4.7, the "V" sequence is different from virtually all other significant accident sequences because core melt will inevitably occur following the initiating event, even if all safety systems perform properly. Thus, the only barrier between the initiating event and core melt is extraordinary operator action over and above the normal pre-planned functions he must perform to allow successful safety system performance (e.g., the pre-planned action of re-aligning valves for recirculation mode). Since this type of action is called for, it is crucial that the operator immediately determine that this particular type of LOCA has occurred.

Following this important first step, the operator must initiate actions which could result in successful termination of this sequence short of core melt. The only action of this kind would be to isolate the low pressure system rupture from the primary system by closing the block valve upstream of the rupture. It is not clear, at this point, whether this block valve can be closed under the LOCA conditions; the possibility of isolation at least involves aspects of design that could vary from plant to plant.

Whether or not the rupture can be isolated, the operator must take action to delay core melt. This delaying action serves two purposes: 1) it provides additional time to attempt isolation, and 2) if isolation fails, delaying the melt will result in increased time for emergency actions such as evacuation of the site and surroundings, and transfer of water from an outside source to the RWST. Should melt occur, the delay will reduce the radioactivity release. The timing of the melt is determined by how long emergency coolant can be injected into the core to keep it covered before the coolant supply in the RWST is exhausted. The operator's job, therefore, is to ensure that sufficient coolant is delivered to the core through the HPCI system, and to also ensure that no other unnecessary draw on the RWST supply is made.

The remaining operator actions will depend on whether isolation is achieved or not. If not, the operator can only perform a monitoring role; determining approach to and commencement of core melt will be his major responsibility. If isolation is successful, the operator must ensure that water level is maintained in the core and long-term heat removal is achieved. This could involve not only his normal pre-planned actions to ensure these functions, but additional "recovery" actions made necessary by his previous actions to delay core melt which involved valving off emergency systems.

To summarize, the key operator actions given an Interfacing Systems LOCA are:

1. Determine occurrence of LOCA outside containment
2. Initiate attempt to isolate rupture (and to secure an outside water source to maintain water level in RWST).
3. Delay core melt by providing minimum necessary draw on RWST to keep core covered
4. If isolation fails, monitor approach to core melt. If isolation succeeds, ensure continued core covering and heat removal.

In Figure 4.8, a logic diagram is presented which displays the possible sequence progression. This figure can be viewed as a version of Figure 4.7 which develops the sequence logic in terms of operator action events.

At state (1) in the logic figure, the operator must unambiguously determine the occurrence of the LOCA. He must then make sure the reactor is shutdown. The "V" sequence in WASH-1400 assumes successful scram and the system moves into state (2) (scram failure would result in state (2a) which leads to core melt regardless of subsequent operator action). After

verifying scram, the operator then attempts to ensure minimum necessary coolant injection to prevent melt, isolate the rupture, and perform the necessary actions to ensure continued adequate water inventory in the core and long-term heat removal capability (i.e., move successively from state (2) to (3) to (4) to (5) and, thus, to successful termination of the accident).

Failure to isolate or to perform the necessary actions after isolation leads to core melt. If the operator fails to take any action to delay melt, it is not obvious that core melt will automatically follow. It is possible that rapid action to isolate the rupture could lead to a point similar to state (4). In fact, if this is accomplished, the operator could be in a better position to move to state (5) because he doesn't have to worry about re-aligning systems whose configuration was altered to delay melt. Of course, if he doesn't take the delaying action, the probability of successful isolation is reduced and the consequences of non-isolation are increased. The operating procedures specifying when to initiate delaying action will obviously be a function of the likelihood of successful isolation as a function of time, and therefore could vary from plant to plant. However, the important point to note here is that these considerations of specific operator procedures do not materially affect the conclusion relevant to this analysis, i.e., the operator must receive sufficient information to allow him to isolate the rupture and take delaying action (when and if he wishes). Thus, sequences D and E of Figure 4.8 are not significant here.

4.3 Key Parameters

In the preceding section, the significant actions required of the operator in response to the interfacing system LOCA were identified. In order for the operator to efficiently perform these actions, he must receive sufficient information via the plant instrumentation concerning the status of various plant systems and components. It is the purpose of

this section to identify the key parameters of the plant state whose measurement would provide the operator with the necessary and sufficient information to unambiguously determine the state of the plant as the accident sequence progresses and to take the corrective actions outlined above. Figure 4.8 will be used to provide the logical framework for this section.

In the WASH-1400 evaluation of this sequence, it was assumed that the LPIS check valve rupture would lead directly to overpressurization and rupture of the LPIS. It is not clear how much (if any) time exists for useful operator action to take place between the time of check valve rupture and LPIS rupture. If little or no time exists, then detecting the check valve rupture is not of great concern because 1) the operator wouldn't be able to do much about it, and 2) the effects of the LPIS rupture will be much more obvious and would provide a better basis for operator action. If, however, detecting the check valve rupture would provide a useful "early warning" to the operator, monitoring the pressure, temperature, and radiation levels in the LPIS would provide ample information. Large rapid increases in these parameters would indicate to the operator that the check valve rupture had occurred, reactor shutdown should take place, LPIS rupture can be expected, and LPIS isolation should be initiated.

As noted previously, the immediate concern following this type of LOCA is the rapid and unambiguous determination by the operator that this specific initiating event has occurred. That is, referring to Figure 4.8, the operator must determine that the plant is in state ①. This is especially important for this sequence because 1) if no operator action is taken, the core will melt even if all safety systems function properly, and 2) the operator action called for here is unusual in that making sure some safety systems do <u>not</u> operate is necessary to delay core melt.

Fortunately, the Interfacing Systems LOCA possesses a rather distinct accident "signature" that should be relatively easy to distinguish from others. The occurrence of the LPIS rupture will be characterized by a decrease in RCS temperature and pressurizer level and a rapid drop in RCS pressure and water inventory as the primary coolant flows out the rupture. The expected behavior of these primary system parameters is illustrated in Figures 4.2 through 4.6. However, unlike other LOCAs inside containment, a corresponding rise in containment pressure and temperature or radiation level will not be experienced. In addition, the radiation level and temperature within the Auxiliary building (where the rupture occurs) will increase.

Thus, measurements of the following parameters and their behavior should be sufficient to determine that the rupture has occurred and distinguish it from other LOCAs (i.e., that the plant is in state (1)):

- RCS pressure decrease
- RCS temperature decrease
- Pressurizer level decrease
- Containment pressure constant
- Containment temperature constant
- Containment radiation level constant
- Auxiliary building temperature increase
- Auxiliary building radiation level increase

As a minimum, a single parameter (temperature, pressure, or radiation level) in the RCS, Containment, and Auxiliary buildings must be monitored. Additional parameters can be used as diverse backups to ensure reliable determination.

In addition to the above parameters, increase in pressure, flow, temperature, and radiation in the LPIS piping between the RCS and the rupture can be monitored. These parameters could be utilized by the operator to differentiate this Interfacing Systems LOCA event from a steam line break outside containment or any other event which could potentially be confused with the "V" sequence initiator. These have been mentioned previously with regard to pre-rupture determination of check valve failure and will be discussed below in regard to isolation actions.

Once the determination has been made that the LPIS rupture has occurred based on the above mentioned measurable parameters, the operator must determine that the plant has moved into state (2) by confirming that reactor scram has occurred. Control rod position indication or measurements of neutron flux can provide this information. Secondary indications would be the RCS pressure and temperature which would be higher for the failure to scram state (2a) relative to state (2). The extent of the difference would depend upon the break size. More information is needed concerning the RCS pressure and temperature given failure to scram before this would be a reliable indication; however, the probability of this sequence makes this analysis a very low priority item.

Following operator determination that the reactor is shutdown, sufficient information must be provided to the operator to allow him to delay core melt. This entails two basic determinations: 1) emergency coolant injection is sufficient to keep the core covered and 2) no other unnecessary draws on the RWST are made.

In order to accomplish the first, the most direct measurement would be reactor vessel water level. Also, a combination of primary pressure and core temperature could provide the operator with an indication of the margin in core cooling and the need for emergency coolant injection. It is believed that all three parameters (coolant level, RCS pressure, and

core temperature) should be monitored to provide unambiguous determination of HPCI flow requirements. Measuring the radiation level in the primary coolant water would provide an indication of fuel cladding failure and thereby indicate that the delaying action was not adequate. Indication of emergency coolant flow into the reactor can be obtained by monitoring HPIS flow rates and accumulator flow rate and/or accumulator tank level.

While monitoring these flow rates will often indicate the source of problems should ECI not be adequate, they should not be relied upon by themselves to indicate successful ECI. This determination should be based on the more fundamental parameters noted above: primary pressure and core temperature, vessel water level, and primary coolant radiation level.

The second requirement for instrumentation for state (3) can be accomplished by monitoring the RWST level in conjunction with HPIS flow and by monitoring the flow rates from the RWST through lines connected to other systems drawing on the RWST. The operator is assumed to take the necessary action to ensure that these additional systems (such as LPIS and CSIS) do not draw from the RWST provided he is given sufficient information. Should the operator determine that the RWST level is decreasing at a rate higher than that consistent with the HPIS flow required, he must determine the source of the additional outflow and terminate the unneccesary draw on the RWST. Flow indications in the lines from the RWST other than HPI will indicate the reason for the excessive depletion of the RWST and indication of the position of isolation valves in the flow paths should provide the necessary information to allow corrective action. For the Surry plant, the important valves are the isolation valves in the LPIS and the CSIS.

Isolating the LPIS rupture will obviously be of major importance in this sequence. In order to do so and thereby move the plant into state (4) (as opposed to (4a) or (4b) which lead inevitably to core melt), the operator must identify the location of the rupture and close the isolation valve(s)

between the RCS and the rupture. The number and location of these valves will vary from plant to plant. The location of the rupture can be determined by monitoring the pressure, temperature, and/or radiation level in the interfacing systems. The probability of the rupture occurring in systems other than the LPIS was evaluated to be insignificant for the Surry plant in WASH-1400, and, therefore, the discussion here is limited to that system. Should evaluations of other plants determine that other interfacing system LOCAs are probabilistically significant, the monitoring of these systems in a manner analogous to the present discussion can be carried out. It is conservatively assumed that the rupture of the LPIS occurs in the common piping or header of the LPIS downstream from the LPIS pumps (in normal operation); and the locked-open valve must then be closed to accomplish isolation. (Refer to Figure 4.1.) The position of this valve must therefore be indicated in the control room. Confirmation of isolation will be available from the primary coolant system instrumentation: RCS pressure and pressurizer level will respond to the closing of the valve and begin to increase as the HPIS continues flow.

From system state(4), the operator must then take the plant into a safe shutdown state(5). At state(4), the primary system integrity has been re-established, the HPIS pump(s) are operating, the LPIS and CSIS are assumed to be isolated, and the containment is at, or very near, normal pressure and temperature. The operator must now take the necessary action to bring the plant to a stable condition by establishing long-term cooling.

In the initial stages of this phase, he must shut off the HPI pumps when they are no longer required. The information necessary to do this is provided by measurements of the coolant level supported by a combination of primary system temperature and pressure just as it was in the previous state. Pressurizer level could be used as a diverse backup measurement but should not be relied upon by itself.

At this point in the analysis, the necessary operator actions to establish long-term cooling are not exactly clear. It is necessary to identify the potential modes of cooling (e.g., steam generators with auxiliary feedwater), ascertain their effectiveness given the sequence of events resulting in plant state (4), and identify the key parameters whose measurement would allow the operator to utilize these cooling modes to arrive at state (5).

It is possible that difficulties could arise in establishing this cooling which could result in a rise in primary system pressure and temperature, which could then initiate a series of additional problems requiring operator action. Pressure rises which cause the primary system safety valves to open (and perhaps not reclose) could result in the need for re-starting the HPI pumps and the CSIS pumps. Sequences of events such as these will result in operator actions which are identical, with a few exceptions, to sequences which will be analyzed in other sections (e.g., small LOCA resulting from stuck-open relief valve). Therefore, in this section we will limit the discussion to that concerned with these exceptions. Note that the remaining actions required of the operator should these additional faults occur and initiate new accident sequences are addressed in the appropriate sections of this report pertaining to those accident sequences.

The exceptions noted above refer to the unique operator actions made necessary by the delaying actions undertaken by the operator at earlier stages of this accident. Specifically, the operator must ensure that all trains of the HPIS are available if needed, the CSIS is available, and sufficient water remains in the RWST for these injection systems to operate and provide enough water for recirculation cooling. However, the instrumentation which was identified above to allow the operator to take the initial delaying action is also sufficient to allow him to re-establish flow through these lines should it become necessary.

4.4 Summary and Conclusions

In the preceding sections, the interfacing LOCA "V" sequence was evaluated with the purpose of identifying the instrumentation which will provide the necessary and sufficient information to the operator to allow him to determine unambiguously the state of the plant and to efficiently take the required corrective action as this sequence progresses. Table 4.1 presents, in summary form, the results of this analysis. The presentation of these results is structured around the key plant states that could develop as the accident sequence progresses. These states are illustrated in Figure 4.8. For each plant state, the following information is summarized:

- the information required to unambiguously determine that the plant is in that specific state
- the appropriate operator action at that state
- the information required by the operator to take this action

Following this summary of results is a discussion of key assumptions that went into the analysis and the major areas where further work is necessary to answer specific questions, confirm assumptions, reduce uncertainties, etc.

The information contained in the summary table is based on a number of assumptions concerning plant performance and the feasibility and effectiveness of specific operator actions. Since many of these actions take place under plant conditions which have not been extensively analyzed in the past, there is necessarily some uncertainty associated with these assumptions. Summarized below are the key areas where further work could be beneficially performed to either confirm uncertain assumptions, answer key questions, or reduce uncertainties to a level to produce a reasonable level of confidence in the conclusions of this analysis:

- Is it reasonable to assume a time delay between check valve rupture and LPIS rupture of sufficient duration to allow operator action? if so, monitoring LPIS pressure, temperature, and flow becomes more important.

- A detailed examination of the possible draws on the RWST should be performed and the actuating signals for these systems should be identified and compared with the expected conditions during the core melt delay phase of the accident.

- The feasibility of successful isolation is uncertain and is at least sensitive to specific plant design. Can the valve(s) physically close under the pressure and flow conditions present? Are the valves remotely or locally controlled? (For the Surry plant, the isolation valve can be actuated from the control room.) Answering these questions would allow a more detailed evaluation of the operator actions required, but would not affect the remainder of the analysis.

- A better definition of the plant state following isolation is needed in order to identify the appropriate operator actions necessary to establish long-term cooling.

- More detailed information concerning plant states is necessary to establish the necessary ranges for the instrumentation.

- The appropriate operator actions once core melt is inevitable need to be better defined. Specically, what consequence mitigation actions can be performed?

Table 4.1. Summary of Key Operator Actions and Information Requirements for V-Sequence

PLANT STATE (See Figure 4.8)	DESCRIPTION OF PLANT STATE	INFORMATION REQUIRED TO IDENTIFY PLANT STATE	APPROPRIATE OPERATOR ACTION FOLLOWING STATE IDENTIFICATION	INFORMATION REQUIRED TO TAKE APPROPRIATE ACTION
①	Rupture of check valves results in LPIS overpressure and rupture	• RCS P,T • Pressurizer water level • Containment P,T,R • Aux. Building T,R • LPIS P,T,R,F	Prepare for actions illustrated in Fig.4.8	See states ②, ③, ④ and ⑤
②	Reactor scram; decay power level; RCS pressure rapidly decreasing to HPIS actuation level	Control Rod Position Neutron flux	Initiate core melt delay actions and isolation	RCS P,T Vessel water level HPIS flow Accumulator flow Accumulator Tank level LPIS flow from RWST CSIS flow from RWST RWST level Isolation valve(s) position
②a	Reactor not scrammed; power level above capacity of HPIS to remove heat; core melt assumed to follow	Control Rod Position Neutron Flux RCS P,T	Monitor approach to cladding failure; initiate consequence mitigation systems	Primary system radiation level Aux. Building R
③	Minimum sufficient flow from HPIS to keep core covered and prevent melt	RCS P,T Vessel water level RWST level LPIS flow from RWST CSIS flow from RWST	Initiate (or continue) isolation actions	Isolation valve(s) position
③a	Either insufficient HPIS flow or excessive draw on RWST	Same as ③	Same as ③	Same as ③

Table 4.1. (Continued)

PLANT STATE (See Figure 3)	DESCRIPTION OF PLANT STATE	INFORMATION REQUIRED TO IDENTIFY PLANT STATE	APPROPRIATE OPERATOR ACTION FOLLOWING STATE IDENTIFICATION	INFORMATION REQUIRED TO TAKE APPROPRIATE ACTION
④	LOCA successfully isolated before core melt occurs	Isolation valve position RCS P LPIS flow Pressurizer water level	Initiate long-term heat removal	RCS P,T Vessel water level Steam generator water level Auxiliary FW flow CST level Reactor power level
④a	Isolation fails after delaying action core melt occurs when RWST depleted	Same as ④	Monitor approach to core melt and initiate consequence mitigation actions	Primary system radiation level RWST level Aux. Building R
④b	Isolation fails; no delaying action has occurred; core melt occurs more quickly than 4a	Same as ④a	Same as ④a	Same as ④a
⑤	Long-term heat removal established	RCS P,T Steam gen. level Aux. FW flow		
⑤a	Long-term heat removal not established; no corrective action possible	RCS P,T Steam gen. level Aux. FW flow	Initiate consequence mitigation systems	

P = Pressure T = Temperature
R = Radiation Level F = Flow Rate

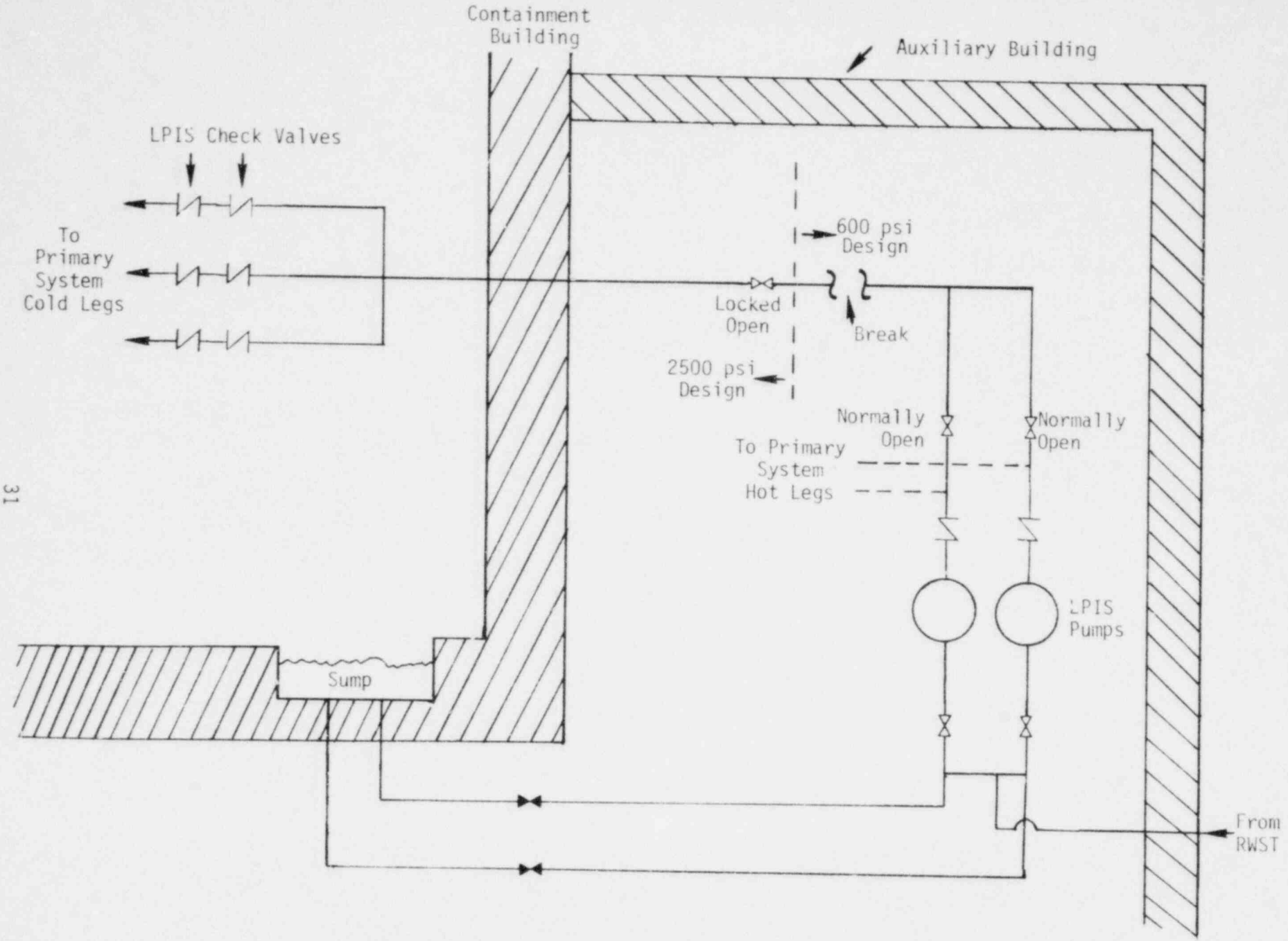

Figure 4.1. Low Pressure Injection System

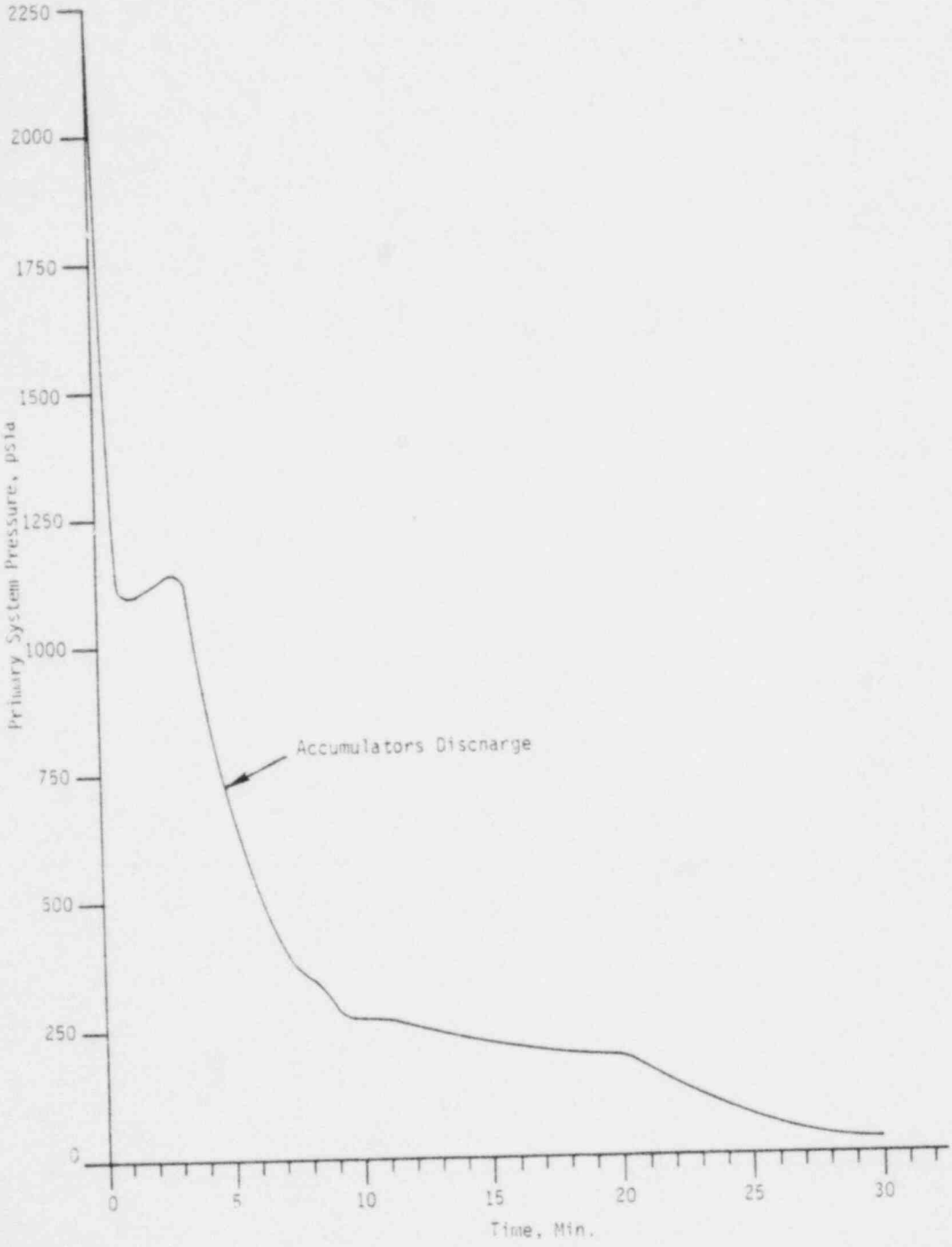

Figure 4.2. Interfacing System LOCA Primary System Pressure vs. Time

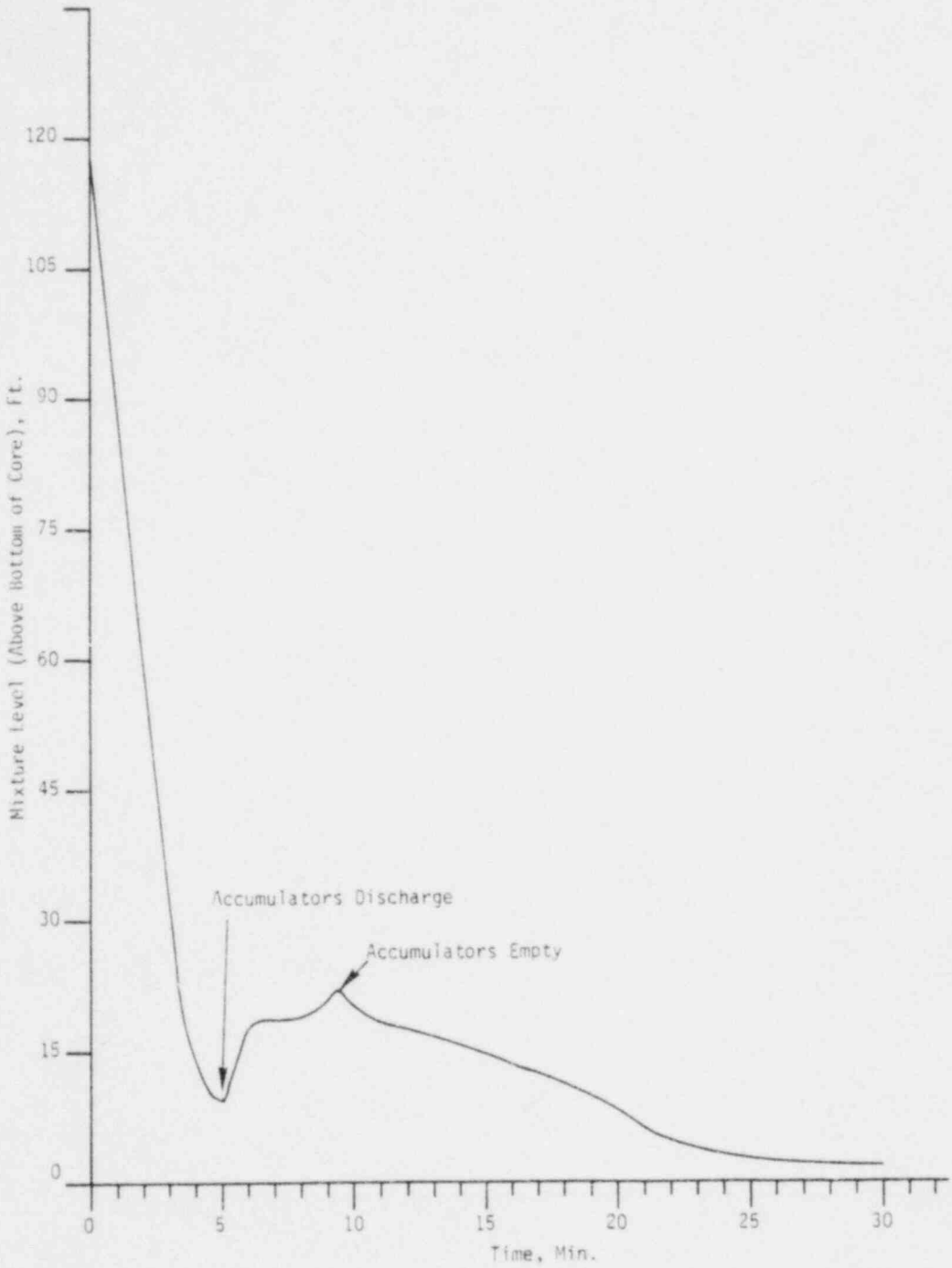

Figure 4.3. Interfacing System LOCA Mixture Level vs. Time

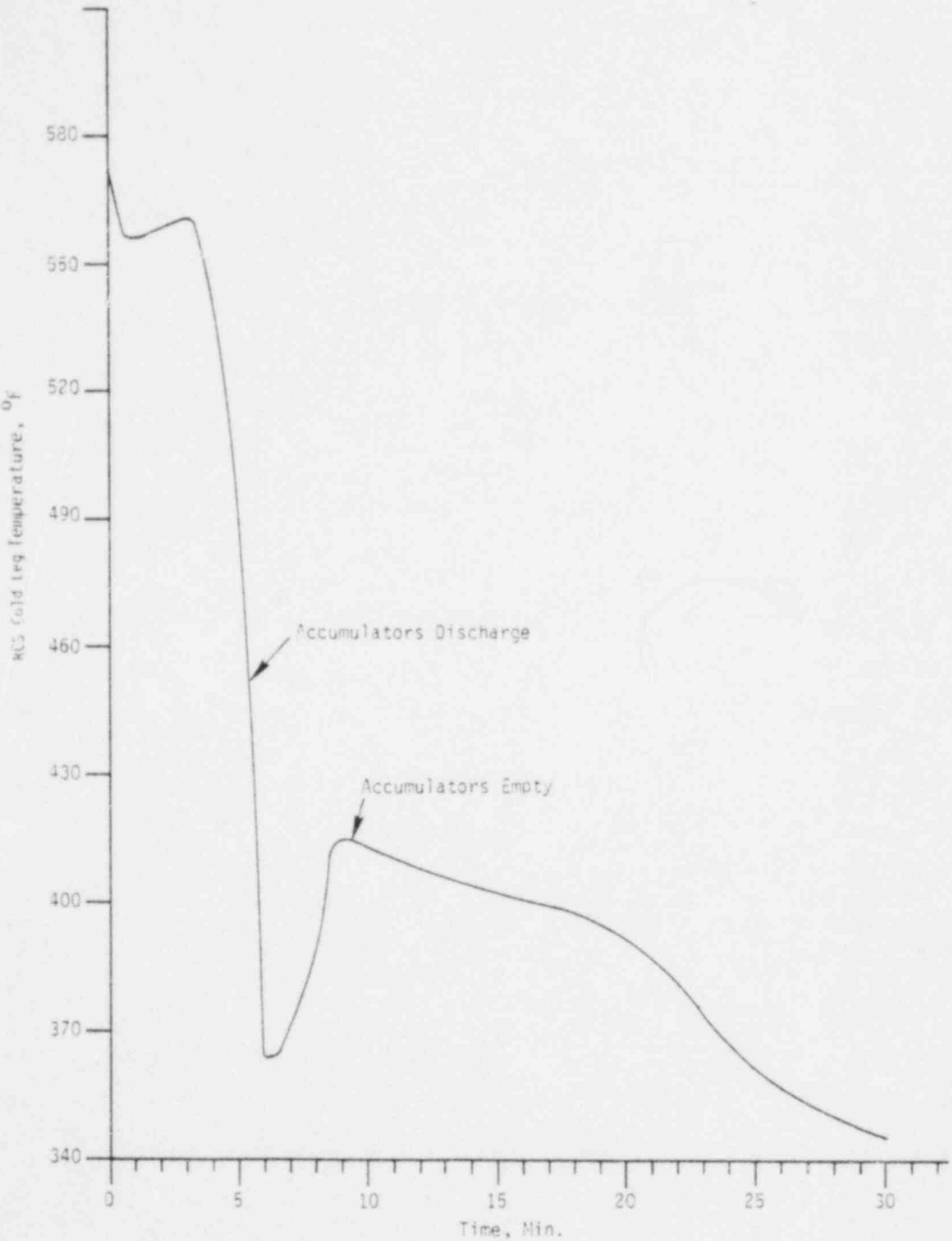

Figure 4.4. Interfacing System LOCA RCS Cold Leg Temperature vs. Time

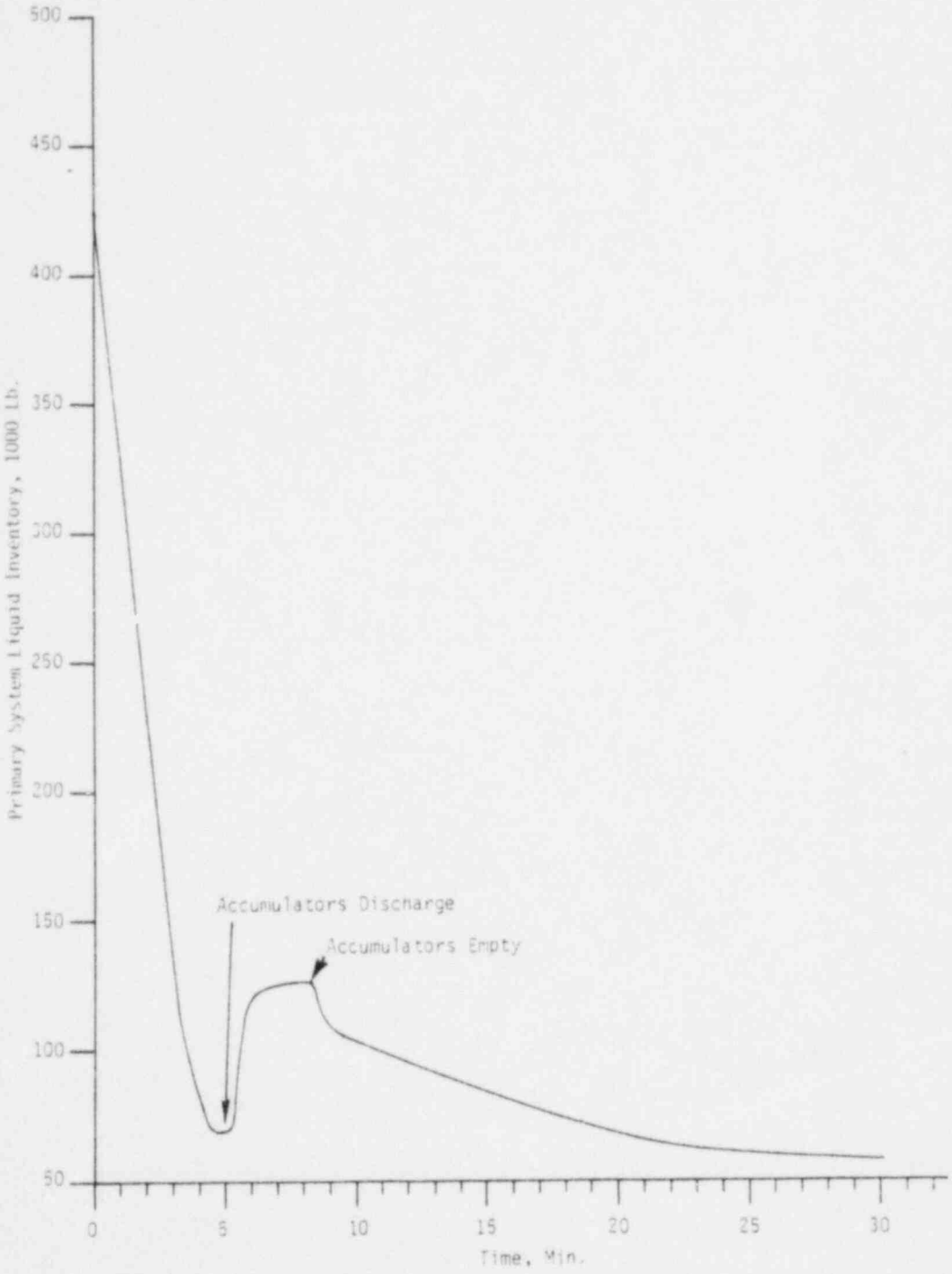

Figure 4.5. Interfacing System LOCA Primary System Liquid Inventory vs. Time

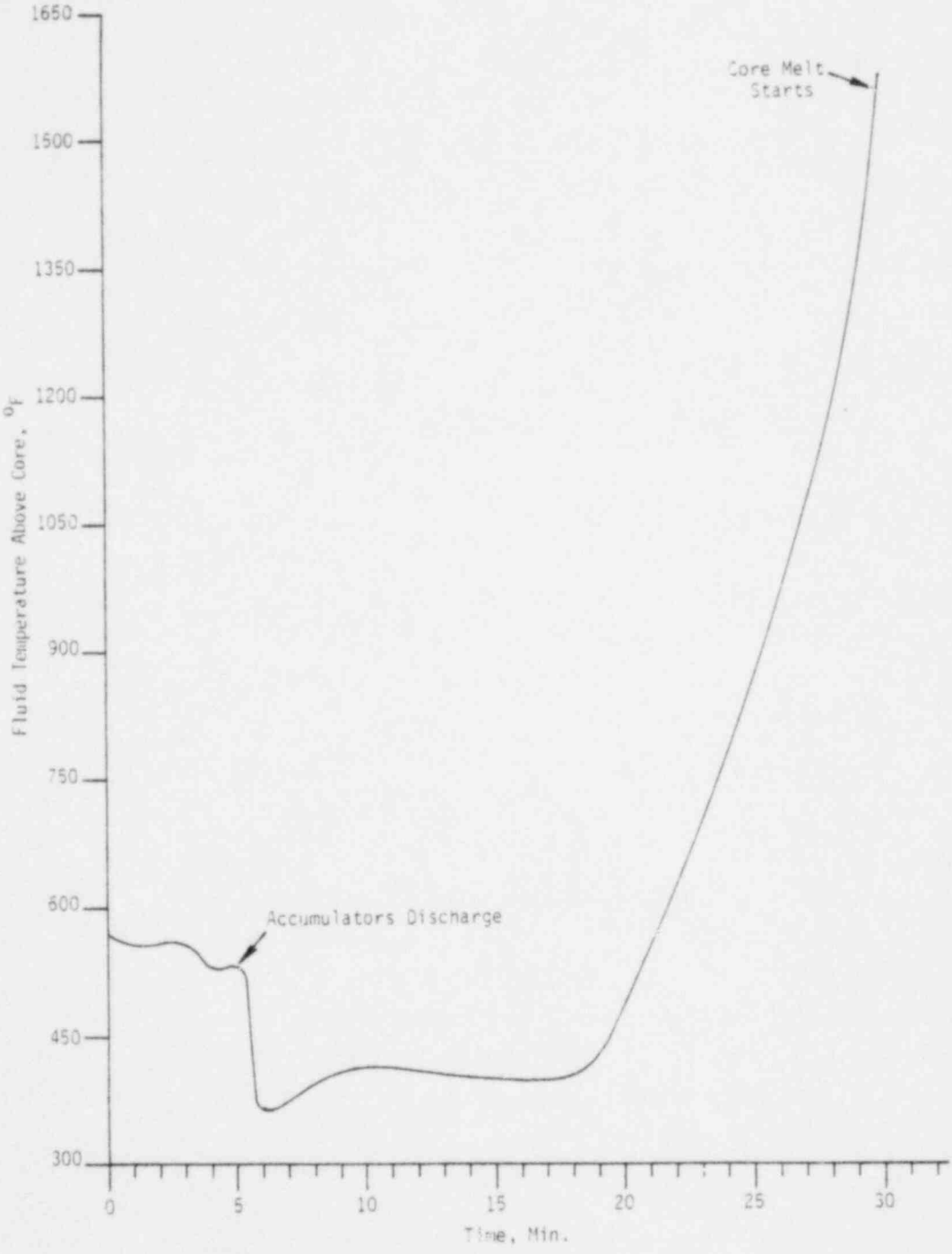

Figure 4.6. Interfacing System LOCA Fluid Temperature Above Core vs. Time

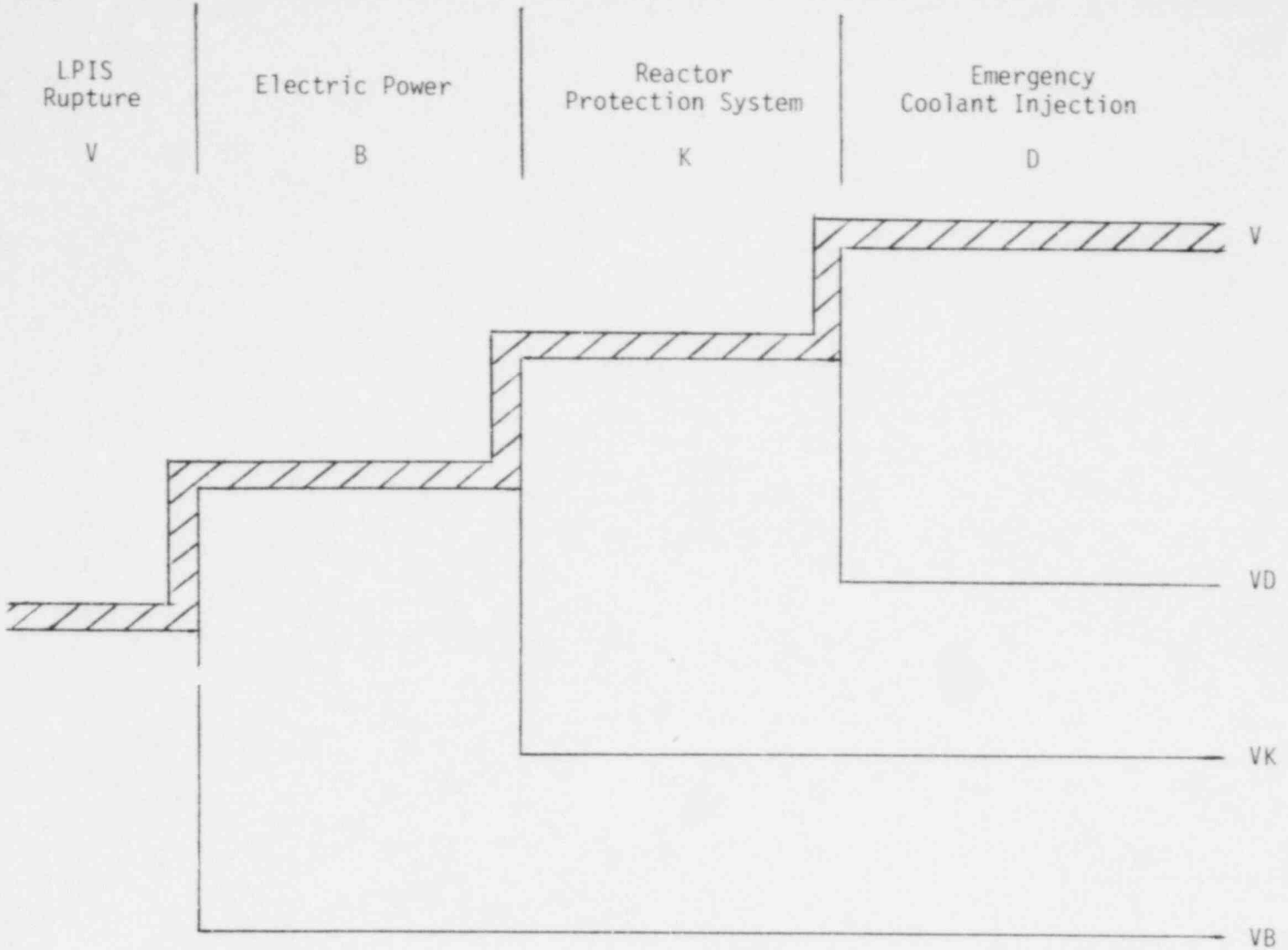

Figure 4.7. Interfacing Systems LOCA Event Tree

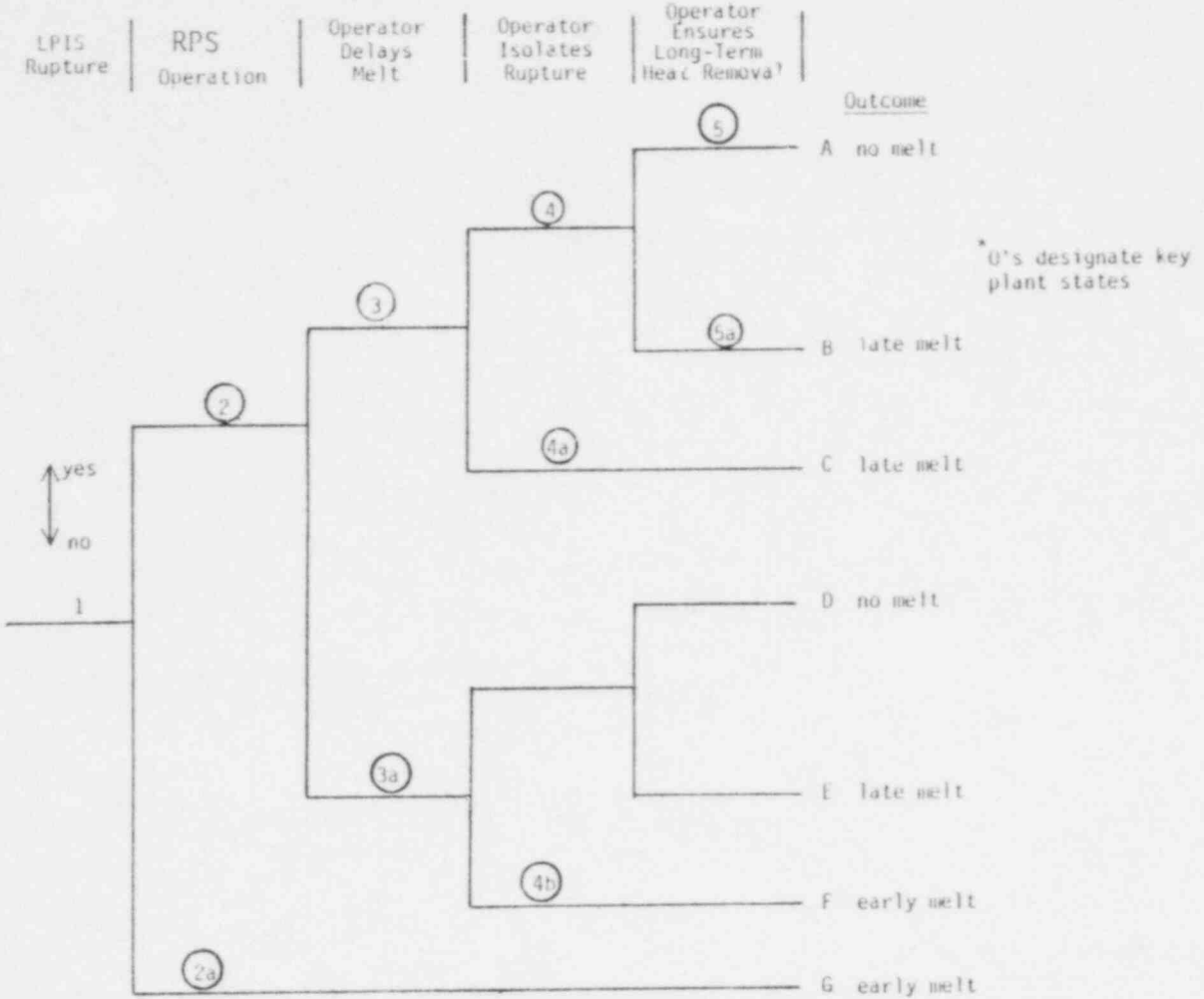

Figure 4.8. Interfacing Systems LOCA Operator Action Event Tree

5.0 RESULTS AND CONCLUSIONS

As noted previously, this analysis involved the investigation of seven accident sequences selected primarily on the basis of public risk. One of these sequences - the "V" sequence from WASH-1400 - has already been presented in the preceding section as an example intended to illustrate the technical approach utilized in this study. Similar discussions of the remaining six sequences are presented in the appendices to this report. In this section, the results obtained from the analysis of these seven sequences are summarized and discussed. In addition to delineating the key measureable parameters resulting from these analyses, conclusions concerning the validity of the technical approach in accomplishing the objectives of this investigation and its value with respect to subsequent related analyses are discussed.

5.1 Summary Table

The results obtained for each of the seven accident sequences were combined together to produce an integrated listing of instrumentation needs. This list is presented as the first column of Table 5.1. Included in this summary table are brief descriptions of the purpose of each parameter; this entails an identification of the specific accident condition and associated operator tasks which necessitate the information provided by the monitoring of the parameter. Remarks which are considered necessary to either amplify, clarify, or qualify the need for each parameter are also included.

As noted in the summary table, a number of parameters have been identified which are not included in the current version of Regulatory Guide 1.97. These additional parameters can be grouped into three general categories.

The first category consists of parameters which are necessary to provide crucial information to the operator and should be included in Reg. Guide 1.97. Examples of such parameters are: 1) the containment sump water temperature which provides the operator with critical information concerning the availability of an adequate NPSH for the emergency recirculation pumps in the S_2C, S_1HF, and S_2HF sequences; 2) various parameters concerned with the LPIS (e.g., LPIS pressure, isolation valve position, etc., which are necessary for unambiguous diagnosis of the interfacing system LOCA initiator; 3) boron concentration, which can provide shutdown margin information under accident conditions which adversely affect the reliability of neutron flux monitor measurements.

The second category consists of parameters for which additional supporting analysis is required before they can confidently be identified as necessary. These are primarily normally redundant parameters which are intended to provide information under accident conditions which could adversely affect the reliability of the prime sources of information. For example, in BWRs the vessel water level is normally considered the prime source of information concerning the effectiveness of core cooling (it is generally assumed that if the water level is adequate, the core is being sufficiently cooled). However, under some postulated severe accident conditions, the reliability of this level measurement might be significantly reduced and alternative measurements (such as core temperature and pressure) could be necessary to monitor core cooling. Further analysis is required to determine the likelihood of such severe core conditions and the reliability of the prime information source under these conditions. The vessel water level would be a comparable parameter in PWRs.

The parameters which are included in the third category are concerned with the status of individual components in safety systems (especially valve positions). These parameters are intended to provide the operator with the necessary information to take specifically identified system re-configuration actions (e.g., manual alignment of RHR and securing of HPIS in the BWR TC sequence), to verify system availability, or to diagnose system failure causes and to initiate repair actions (e.g., the upper compartment drain valves in the S_1HF and S_2HF sequences).

This last purpose (diagnosis of failure cause and initiation of repair) resulted in a number of parameters which were not specifically included in Reg. Guide 1.97, and might represent a fundamental difference of scope between this report and Reg. Guide 1.97. This analysis was based on an identification of required operator tasks which included not only a determination of the plant state, but also actions to bring the plant to a successful shutdown. While the determination of plant state can often be accomplished with a relatively few fundamental parameters (e.g., RCS temperature, RCS pressure, etc.), the diagnosis and repair of failures often necessitates much more detailed information concerning specific subsystems or components.

5.1.1 Completeness

The completeness of the list of measureable parameters in Table 5.1 is obviously dependent upon the completeness of the supporting analysis (i.e., by the completeness of the set of accident sequences addressed). In a probabilistic approach, the attainment of absolute completeness is not attempted, and the cut-off point is based on a determination that any sequences not considered are not significant contributors to risk (relative to those addressed) or would not affect the results of the analysis due to their similarity to included sequences. The precise meaning of "significant" usually entails some implicit estimation that the risks

associated with the excluded sequences are either acceptable or at least not high enough to justify the cost of implementing any safety improvements which might result from the inclusion of these additional sequences in the analysis. In some cases, an explicit determination of this type is performed in the form of a cost-benefit or risk-benefit analysis.

Due to the preliminary nature of this investigation, no attempt was made to achieve completeness. A number of accident sequences were selected which represent dominant risk contributors; however, this does not imply that additional sequences do not exist which are "significant" in the sense described above. While this indicates that additional parameters might be added to the list should subsequent analyses consider additional accident sequences, it should also be noted that those parameters which are included in Table 5.1 will not be removed due to the consideration of additional sequences. Recommendations for further investigations to address this completeness problem are discussed in Section 6.0.

5.1.2 Necessary vs. Redundant

In many instances, multiple parameters are included in Table 5.1 with the identical indicated purpose. For example, instruments to measure neutron flux and boron concentration are both indicated with the purpose of providing the operator with criticality information. In these cases, one or more of the instruments serves as a redundant or diverse back-up to ensure reliable information flow to the operator. How to determine whether a specific redundant instrument is necessary to the operator or merely a handy "extra" of little real value is a difficult problem. The ability to make this determination is also sensitive to the completeness of the analysis. This is because under one set of accident conditions monitoring a particular parameter might be necessary for back-up confirmatory information, while under another accident condition it might be the operator's primary or only source of this information. For example, under most

conditions, the neutron flux is a reliable indicator of criticality; however, under accident conditions which could result in voids in the core, the neutron flux monitors can become much less reliable and the ability to measure boron concentration becomes very important.

Therefore, in developing Table 5.1, multiple parameters were included for the same informational purpose if it was determined that the existing accident conditions could be expected to adversely affect the ability to reliably monitor one or more of these parameters. A reasonable justification therefore exists for normally redundant instruments based on an examination of specific accident conditions. For these cases where the effect of the accident on instrument reliability was not very clear, redundant parameters were included. This was done as both a conservative action under uncertain information and as a recognition that the examination of sequences beyond those considered here could result in increased importance of these seemingly redundant parameters.

It should also be noted that redundant and diverse instruments are also valuable for protecting against random instrument failure unrelated to extraordinary conditions imposed by the accident; however, this purpose was not considered in the determination of the parameters included in Table 5.1.

5.1.3 Plant Specificity

While most of the instrumentation requirements listed in Table 5.1 would apply to all nuclear plants, it should be remembered that this analysis was based on three specific plant designs. Design differences in other reactors could affect the results of an analysis of this type in three basic ways:

1) The risk significance of particular accident sequences varies from plant to plant; this was demonstrated by the results of the Sequoyah Reactor Safety Study.

2) The physical plant response to the postulated failure events can be significantly different (especially between plants of different vendors); this could affect the definition of appropriate operator tasks, the accident signature by which the operator diagnoses the plant condition, and the probability of the operator successfully accomplishing his tasks due to variations in required response time.

3) Details of the plant design can significantly affect the options available to the operator, especially with respect to repair tasks; for example, the number and position of block valves and whether they are locally or remotely controlled obviously affects the ability of the operator to isolate postulated breaks and can in some circumstances make this action impossible.

Many of these considerations are addressed in the discussions of the seven individual sequences included in this report and recommendations related to this problem of plant specificity are presented in Section 6.0.

5.2 Validity of Approach

It was recognized at the outset of this analysis that the proposed technical approach differed significantly from that taken in similar past investigations. In fact, a major purpose of this work was to determine the effectiveness and usefulness of the selected approach.

A key conclusion of this effort is that the technical approach outlined in Section 3.0, with a few minor reservations discussed below, was not only a very effective way of accomplishing the objectives of this

particular analysis but was also able to set a logical foundation upon which further analyses designed to enhance operator action can be based.

Probably the most important feature of this approach which allowed an efficient systematic investigation to be performed was that it forced the analyst to identify and focus on the specific tasks required of the operator under a variety of selected important accident sequences. In this way, specific informational needs could be identified and the required instrumentation determined.

The identification of these specific required tasks and associated operator informational needs was not only a key step in this analysis, but must necessarily be the starting point for many other analyses addressing additional aspects of the general operator/plant interface problem. For example, the logical foundation for a computerized disturbance analysis system must be a determination of the functions such a system will be expected to perform. These functions can be considered, in effect, to be the required tasks of a superhuman operator. Thus, the technical approach utilized in this study can provide this foundation. In general, any study with the objective of improving the operator's capability to perform his tasks -- whether the study concerns the optimum configuration of knobs and dials in the control room or the training of operators on simulators -- must begin with a systematic definition of these required tasks. Again, this approach is designed to provide this starting point.

The reservations mentioned above concern the amount of supporting analysis required by this approach. As discussed in Section 3.0, this approach is based on a detailed investigation of individual accident sequences which necessitates a thorough understanding of the plant response to key accident conditions. However, most of the available plant response analyses either address sequences which do not entail the multiple failures associated with the high risk accident sequences, or adopt very conservative assumptions which obscure important information.

Fortunately, the value of best-estimate codes for plant response modeling and the application of such codes to investigations of risk significant accident sequences is becoming recognized throughout the industry. With numerous groups increasing their capability to perform such analyses, the difficulties associated with the technical approach utilized here should significantly diminish in the near future. In fact, the technical approach described in Section 3.0 provides the most effective way of utilizing the flow of new information which should result from the expanded use of realistic plant models.

Table 5.1. Summary of Variables Identified in Sequence Evaluations

PWR Measured Variable	Major Purpose for Indicated PWR Accident Sequence				Comments
	V	S_2C	S_1HF	TML/TMLB′	
Control Rod Position	• Verification of scram	Same as V	Same as V	Same as V	Provides primary indication of successful scram
Neutron Flux	• Verification of scram	Same as V	Same as V	Same as V	Indicates shutdown margin; important after initial failure to scram; might be unreliable under voiding conditions
RCS Pressure	• Diagnosis of initiating LOCA event • Determination of need for and effectiveness of ECI • Provides, along with RCS temperature, degree of subcooling • Indication of break isolation	• Identification of initiating small break • Determination of need for and effectiveness of ECI and ECR • Provides, along with RCS temperature, degree of subcooling	Same as S_2C	• Indication of transient initiator • Indication of integrity of primary system • Provides, along with RCS temperature, degree of subcooling	
RCS Temperature	• Provides, along with RCS pressure, degree of subcooling	Same as V	Same as V	• Provides, along with RCS pressure, degree of subcooling • Indicator of natural circulation	Measurements of both hot and cold leg temperatures useful for natural circulation
Pressurizer Level	• Indication of initiating event • Indication of isolation of break	• Indication of initiating event • Diagnosis of size and location of break	Same as S_2C	• Indication of initiating event	

Table 5.1. Summary of Variables Identified in Sequence Evaluations

PWR Measured Variable	Major Purpose for Indicated PWR Accident Sequence				Comments
	V	S_2C	S_1HF	TML/TMLB′	
Pressurizer Relief Valve position, discharge line flow, or drain tank level				•Verification of pressurizer relief valve reclosure	Other parameters designed to indicate RCS integrity can be used as back-up to these direct indications
Vessel Water level	•Indication of need for and effectiveness of ECI •Indication of isolation of break	•Indication of initiating event •Indication of need for and effectiveness of ECI	Same as S_2C	•Indication of initiating event •Verification of relief valve closure and success of maintaining adequate liquid inventory	Not included in Reg. Guide 1.97. Other thermodynamic parameters (e.g. RCS pressure and temperature) can be used for most accident conditions. Further analysis is required to determine if these parameters are sufficient for all significant accident conditions
Primary System Radiation Level	•Indication of approach to core melt •Assessment of extent of core damage following restoration of core cooling	Same as V	Same as V	Same as V	Un-l[illegible] timely measurements are [illegible]ssary; system should rem[illegible] operable under all accident conditions including containment isolation
Boron Concentration	•Indication of shutdown margin	Same as V	Same as V	Same as V	Could be useful back-up if accident progresses to conditions which make neutron flux monitors unreliable

Table 5.1. Summary of Variables Identified in Sequence Evaluations

PWR Measured Variable	Major Purpose for Indicated PWR Accident Sequence				Comments
	V	S_2C	S_1HF	TML/TMLB′	
Containment Pressure	•Diagnosis of initiating LOCA	•Diagnosis of initiating break •Indication of CSIS failure, repair of CSIS, and effectiveness of CSRS •Provides, in combination with sump water temperature, indication of adequate NPSH for ECR pumps. •Indication of containment integrity	•Diagnosis of initiating break •Provides, in combination with sump water temperature, indication of adequate NPSH for ECR pumps •Indication of containment integrity •Indication of CSRS failure or effectiveness	•Verification of relief valve reclosure •Indication of containment integrity	
Containment Isolation Valve Position		•Verifies containment isolation to preclude transport of radioactive material through containment penetrations	Same as S_2C	Same as S_2C	

Table 5.1. Summary of Variables Identified in Sequence Evaluations

PWR Measured Variable	Major Purpose for Indicated PWR Accident Sequence				Comments
	V	S_2C	S_1HF	TML/TMLB′	
Containment Temperature	• Diagnosis of initiating LOCA	• Diagnosis of initiating break • Indication of CSIS failure, repair of CSIS, or effectiveness of CSRS	• Diagnosis of initiating break • Indication of CSRS failure or effectiveness	• Verification of relief valve reclosure	Containment humidity can be used as a highly reliable backup to containment pressure and temperature to indicate primary system integrity
Containment Radiation Level	• Diagnosis of initiating LOCA	Same as V	Same as V		Serves as backup to containment pressure and temperature for indication of loss of primary boundary integrity
Containment Sump Water Level		• Indicate availability of water for ECR and CSRS	• Indicate absence of coolant flow between upper and lower compartment and successful restoration of flow		Can also be used as indicator of initiating break
Containment Sump Water Temperature		• In conjunction with containment pressure, indicates adequate NPSH for CSRS and ECR pump operation	Same as S_2C		Not included in Reg. Guide 1.97

Table 5.1. Summary of Variables Identified in Sequence Evaluations

PWR Measured Variable	Major Purpose for Indicated PWR Accident Sequence				Comments
	V	S_2C	S_1HF	TML/TMLB′	
Upper Containment Compartment Water Level and Drain Valve (between upper and lower compartments) position			• Indication of major cause for ECCS recirculation failure • Indication of repair and restoration of flow		Not specifically identified in Reg. Guide 1.97 but only applicable to plants with similarly designed containment drain system
Steam Generator Level	• Indication of capability of long term decay heat removal	• Indication of feedwater system performance	Same as S_2C	• Indication of initiating transient • Indication of performance of auxiliary system	
Steam Generator Pressure	• Indication of capability of long term decay heat removal	• Indication of feedwater system performance • Indication of secondary system integrity	Same as S_2C	• Indication of performance of feedwater system • Indication of capability of using condensate pumps (TML)	
Steam Generator Safety/Relief Valve Positions		• Indications of secondary system integrity	Same as S_2C	Same as S_2C	
Main Feedwater Flow				• Indication of initiator, success of repair, or utilization of condensate pumps (for TML)	Pump discharge pressure (not included on Reg. Guide 1.97) could be used as backup indication and assist in specifying cause of failure for TML

Table 5.1. Summary of Variables Identified in Sequence Evaluations

PWR Measured Variable	Major Purpose for Indicated PWR Accident Sequence				Comments
	V	S_2C	S_1HF	TML/TMLB′	
Auxiliary Feedwater Flow	•Indication of adequate water flow to steam generators for long term decay heat removal	•Indication of adequate flow to steam generators to enhance heat removal	Same as S_2C	•Indication of AFWS failure and determination of restoration	Pump discharge pressure could be used as backup; flow control valve positions could be useful in determining cause of AFWS failure and in regulation of restored AFWS
Condensate Pump Flow or Discharge Pressure				•Potentially useful in diagnosis of initiating event •Indication of effectiveness of using condensate pumps to supply feedwater to steam generators for some TML initiators	Not included in Reg. Guide 1.97
Steam Supply to AFW turbine driven pump				•Diagnosis of AFW failure cause and subsequent repair	Not included in Reg. Guide 1.97
Accumulator Tank level, flow rate, and/or isolation valve position	•Indicate injection after initiator		Same as V		Passive system; indirect indication of performance can be obtained from other parameters
Condensate Storage Tank Level	•Indication of ability to use AFW as heat removal system	Same as V	Same as V	Same as V	

Table 5.1. Summary of Variables Identified in Sequence Evaluations

PWR Measured Variable	Major Purpose for Indicated PWR Accident Sequence				Comments
	V	S_2C	S_1HF	TML/TMLB′	
Refueling Water Storage Tank Level	• Indication of availability of water for ECI • Determination of optimum use of RWST water supply in core melt delaying actions	• Indication of availability of water for ECI	Same as S_2C		
HPIS Flow	• Indicates success of ECI for core melt delay actions	• Verification of ECI operation following initiator	Same as S_2C		Pump discharge pressure can be used as backup indication of system operation
LPIS pressure, temperature, radiation level, and/or flow	• Diagnosis of initiating event (differentiate from other events with similar RCS response) • Indication of isolation of break • Determination of break location				LPIS pressure, temperature, and radiation level not included in Reg. Guide 1.97
LPIS Isolation valve position	• Indication of success of isolation				Not included in Reg. Guide 1.97
Containment Spray flow (including CSIS and CSRS)	• Indication of need to isolate system for delaying actions	• Indication of failure of CSIS and subsequent repair	• Indication of operation containment heat removal		Pump discharge pressure can be used as backup indication of system operation

Table 5.1. Summary of Variables Identified in Sequence Evaluations

PWR Measured Variable	Major Purpose for Indicated PWR Accident Sequence				Comments
	V	S_2C	S_iHF	TML/TMLB′	
RHR Flow	• Indication of system operation for long term heat removal	Same as V	Same as V	Same as V	Pump discharge pressure can be used as backup indication of system operation
Positions of key valves in safety related systems (HPIS, LPIS, CSIS, CSRS, CHRS, RHR)	• Indication of capability of systems to operate when called upon • Diagnosis of failure	Same as V	Same as V	Same as V	Not specifically included in Reg. Guide 1.97
Component Cooling Water Flow in CHRS heat exchangers		• Indication of effectiveness of containment cooling using CSRS	Same as S_2C		
Component Cooling Water Flow to RHRS Heat Exchanges	• Indication of effectiveness of long-term heat removal	Same as V	Same as V	Same as V	
Auxiliary Building Temperature or Radiation level	• Diagnosis of initiating event • Determination of successful isolation of break				Auxiliary Building Temperature not included in Reg. Guide 1.97

Table 5.1. Summary of Variables Identified in Sequence Evaluations

PWR Measured Variable	Major Purpose for Indicated PWR Accident Sequence				Comments
	V	S_2C	S_1HF	TML/TMLB′	
Containment auxiliary heat removal fan discharge flow			• Indication of the amount of containment cooling which is being performed and the requirements for CSRS		Only applicable to plants with such a system
Status of Class-1E power supplies to key safety system components	• Verification of safety system availability	Same as V	Same as V	• Indication of safety system availability • Diagnosis of cause for AFWS failure	
Status of Non-Class-1E Power Supplies	• Verification of available power source	Same as V	Same as V	• Indication of initiating event for TMLB′ and determination of restoration	

Table 5.1. Summary of Variables Identified in Sequence Evaluations

PWR Measured Variable	Major Purpose for Indicated BWR Accident Sequence: TC	COMMENTS
Control Rod Position	• Indication of failure of automatic scram, and success/failure of manual insertion of rods	
Neutron Flux	• Indication of failure to scram and determination of effect of manual shutdown actions	
RCS Pressure	• Determination of effect of delayed scram • Need for and effectiveness of HPCI • Effectiveness of long term cooling • Secondary indication of reactor shutdown	
RCS Temperature	• Indication of effectiveness of core cooling (in combination with RCS pressure)	Location of instruments not yet determined; core exit temperature (as listed in Reg. Guide 1.97) does not seem to be best location. Intended for those accident conditions where coolant level measurement might be expected to be unreliable
Vessel Water Level	• Indication of initiating transient event • Indication of water inventory • Determination of need for and effectiveness of emergency core cooling • Determination of when to secure HPIS and rely on RCIC for long term cooling	
Main Steam Flow Isolation Position	• Indication of initiator • Determination of potential core cooling procedures	MSIV should automatically close following the initiating loss of feedwater transient event

Table 5.1. Summary of Variables Identified in Sequence Evaluations

BWR Measured Variable	Major Purpose for Indicated BWR Accident Sequence: TC	COMMENTS
Safety/Relief Valve Positions in Primary System (including ADS)	• Indication of effect of delayed shutdown • Indication of potential effectiveness of manual shutdown using SLCS • Indication of primary boundary integrity	
Radiation Level in Coolant	• Information for monitoring of core melt • Indication of amount of core damage	
Containment Pressure	• Indication of integrity of primary pressure boundary • Indication of containment integrity	
Containment Temperature	• Indication of integrity of primary pressure boundary • Indication of containment integrity	
Containment Radiation Level	• Indication of integrity of primary pressure boundary	
Suppression Pool Level	• Indication of primary coolant boundary integrity • Indication of availability of water for ECR	
Suppression Pool Temperature	• Indication of ability of cooling system to pump water	
Boron Tank Level	• Indication of Boron injection for shutdown	
SLCS flow or pump discharge pressure	• Indication of system operation	

Table 5.1. Summary of Variables Identified in Sequence Evaluations

BWR Measured Variable	Major Purpose for Indicated BWR Accident Sequence: TC	COMMENTS
Boron Concentration	• Determination of effectiveness of manual shutdown using SLCS; indication of shutdown margin	Not included in Reg. Guide 1.97. Could be useful backup under accident conditions which make neutron flux monitors less reliable
Feedwater flow	• Indication of initiating event	
Feedwater pump discharge pressure, current to pumps, or controller position	• Indication and diagnosis of cause of initiator	
RCIC valve positions	• Ensure availability of system	Not specifically included in Reg. Guide 1.97
Steam flow to RCIC turbine	• Indication of adequate flow to ensure system operation	
RCIC flow or pump discharge pressure	• Indication of successful system operation or cause of failure	
HPCS valve positions	• Ensure availability of system	Not specifically included in Reg. Guide 1.97
HPCS flow, pump discharge pressure, or current to pumps	• Indication of successful system operation or cause of failure	

Table 5.1. Summary of Variables Identified in Sequence Evaluations

BWR Measured Variable	Major Purpose for Indicated BWR Accident Sequence	COMMENTS
	TC	
RHR valve position (valves required for pre-warming and flushing and flow control valves)	• Allow startup of system and subsequent operator control of flow	Not included in Reg. Guide 1.97
RHR heat exchanger inlet/outlet temperature	• Information necessary for manual startup and indication of subsequent system performance	
HPSW valve position	• Indication of availability of system	
HPSW flow or pump discharge pressure	• Indication of system operation	

6.0 RECOMMENDATIONS

The following recommendations (many of which have been mentioned in the preceding sections) follow from the results of this investigation as summarized in Section 5.0 and are based on the conclusion that the efforts reported above provide a valuable tool for enhancing operator capabilities and should be actively pursued:

1) The instrumentation listed in Table 5.1 should be compared to that which exists in present plants or is called for in current regulatory documents (specifically, Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident"). Most importantly, instrumentation needs which are included in Table 5.1, but are not identified in Reg. Guide 1.97, should be noted (see Section 5.1) and justification provided for their absence from the latter list.

2) Additional accident sequences (particularly BWR sequences) which are considered to be risk significant should be identified and analyzed in the same manner as the seven sequences addressed here. Also, the analysis of the transient initiated sequences should be expanded to include additional specific transient events. The selection of these additional transient initiators should be based on the probability of occurrence and whether the occurrence of such an event would require different operator actions or affect the ability of the operator to gather the necessary information compared to the initiating transient events considered in this report. At some point "risk significant" must be defined by some sort of risk-benefit calculation.

3) Included in the discussions of each of the seven accident sequences is a list of areas where further information would be beneficial in either confirming key input assumptions and/or data to the analyses or reducing the uncertainties involved. The efforts required to address these important areas of uncertainty should be undertaken.

4) The present analysis should be extended to provide more specific information concerning the instrumentation listed in Table 5.1. For example, some aspects of the manner in which the information should be presented to the operator can be

addressed. This would entail a determination of the need for continuous readouts, recorders, trend information (e.g., rates of change), etc. In addition, the necessary ranges of these instruments and the environmental conditions for which they should be qualified should be determined. The definition of specific operator tasks and associated informational needs performed in this present analysis will provide the framework upon which these additional tasks can be efficiently performed.

5) The development of best-estimate computer codes and their application to modeling the plant response to risk significant accident sequences should be actively pursued. Only in this way can reliable accident signatures useful for diagnostic purposes be obtained. This should include identifying centers of expertise in this country and assessing their capability and availability to perform the required analyses.

6) In recognition of the possible variations in instrumentation needs associated with diverse plant designs (as discussed in Section 5.1.3), it would be desirable to extend this analysis to address additional plants. However, the supporting analysis required for this (including identification of high risk accident sequences and accompanying physical response modeling) would be quite extensive. Therefore, as a near-term recommendation, the major reactor types should be surveyed and important design features which could potentially affect the applicability of Table 5.1 to each plant type should be identified. As more supporting risk analyses and plant response modeling are performed, more detailed investigations of additional plants should be performed.

7) The efforts involved in this analysis should be utilized as the foundation for additional investigations of other aspects of the general operator/plant interface problem. One example mentioned previously involves the information generated in this report as the starting point in the development of a computerized disturbance analysis system. Other tasks which could be performed based on the contents of this report include the evaluation of current operating procedures, the development of effective training simulators, an estimation of the value of a "safety state vector" (or its constituent parameters) or other current recommendations resulting from post-TMI investigations, etc.

7.0 REFERENCES

1. Reactor Safety Study; An Assessment Of Accident Risks in U.S. Commercial Nuclear Power Plants, WASH-1400, U.S. NRC, October 1975.*

2. Report on Systems Analysis Task Reactor Safety Study Methodology Applications Program Sequoyah Unit 1 PWR Power Plant (To be published).

3. PWR Sensitivity to Alterations in the Interfacing-Systems LOCA, EPRI NP-262, Prepared by Science Applications, Inc., September 1976.**

*Available free upon written request to the Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.

**Available for purchase from the National Technical Information Service, Springfield, VA 22161.

APPENDIX

ACCIDENT SEQUENCE EVALUATIONS

The list of plant parameters presented in Table 5-1 was developed from an evaluation of dominant accident sequences identified in previous risk assessments. The technical approach used in this evaluation is discussed in Section 3, and the evaluation of the PWR V sequence is presented in Section 4 to illustrate the methodology. This appendix contains the sequence evaluations for the remaining six accidents considered in this study. These include the BWR TC sequence, and S_1HF-γ, S_2HF-γ, TML-γ, TMLB'-δ, and S_2C-δ PWR sequences. The latter two sequence evaluations and the TC accident were based on the Reactor Safety Study (WASH-1400) analysis. The remaining PWR sequences were based on the Sequoyah Reactor Safety Study.

The summaries in this appendix are organized in a similar manner to the V sequence discussion of Section 4. A description of the particular sequence as analyzed in the original risk assessment introduces each section. The sequence of events and plant response are summarized, and the key assumptions presented. Given the initiating event and the associated failure(s), the sequence is then evaluated to determine what actions the operator could take to successfully terminate the accident prior to core damage, or mitigate the consequences. The logic and event tree methodology employed are discussed in Section 3. Following identification of operator actions, the information required by the operator to perform these functions is determined. Measurable plant parameters are then identified which can provide the operator with the information necessary to identify the plant state during each stage of the sequence, take the appropriate action, and determine the success or failure of his response. Finally, a brief listing of the important conclusions which evolved from the sequence evaluation are summarized. These include denoting the critical areas where additional information or analysis would reduce uncertainties and verify important assumptions, thus allowing an improved assessment of operator response and instrumentation requirements.

A.1 TML-γ SEQUENCE

A.1.1 TML-γ Sequence Description

It is anticipated, in the Sequoyah Reactor Safety Study (RSS)[1], that approximately seven times each year a deviation from normal plant parameters will occur which requires shutdown of the reactor. These deviations are referred to as transient events (T). Under normal circumstances, with all systems functioning as designed, the operator would bring the reactor to an orderly hot or cold shutdown condition. Given the malfunction of various systems, heat imbalances could occur in the reactor system which result in a core meltdown and/or containment failure. A dominant transient event resulting in core melt, TML-γ, was identified in the Sequoyah Reactor Safety Study. Figure A.1-1 presents the transient event tree developed in the Sequoyah RSS with the TML sequence highlighted.

The initiating event for this sequence is a malfunction, failure, or fault in the plant equipment which leads to a demand on the Reactor Protection System (RPS) and requires operation of the plant normal or alternate heat removal systems to ensure cooling of the reactor core. In addition to this transient event initiator, the TML sequence postulates a failure of both the Main Feedwater System (MFWS) and the Auxiliary Feedwater System (AFWS).

Probabilistically, the most important sequences of this type involve transient events which make unavailable, or degrade the performance of the main feedwater system. This is easy to understand, since the conditional probability of failure of the MFWS would be higher for such initiators and equal unity for some of them.

These important initiators can be either faults or failures of components within the MFWS (e.g., pump failures, loss of condenser vacuum, etc.) or can involve failure events in supporting systems (e.g., loss of offsite power). While the loss of offsite power initiated sequence was a major contributor to risk in WASH-1400, the Sequoyah RSS determined that this sequence was not significant in comparison to sequences associated with failures directly involving MFWS components. This was primarily due to the additional redundancy in the emergency AC power supplies available to power the AFWS at the Sequoyah plant compared to those available at the Surry plant studied in WASH-1400. Accordingly, the initiating event selected for this analysis is the loss of MFWS due to causes other than the loss of power. The important WASH-1400 sequence initiated by loss of offsite power, TMLB', is analyzed in Section A.2 of this report.

Figure A.1-3 through A.1-7 illustrate the response of some important plant variables during the initial stage of the TML sequence.* Within one minute after the loss of main feedwater, the liquid level on the steam generator secondary side will drop to the low-low level setpoint (Figure A.1-6). This signal normally accomplishes two functions: 1) it initiates a reactor trip and 2) it generates a demand for the auxiliary feedwater system.

The reactor trip signal causes the control rods to be dropped into the core which reduces the heat generated by the fuel to approximately 6 percent of full power (see Figure A.1-7). This reduction in power causes the primary system pressure to drop and the liquid level to decrease as less heat is being generated in the core and the steam generator still contains some liquid to remove decay heat. This is clearly seen in Figures A.1-3 through A.1-6. The auxiliary feedwater system would normally

*This analysis is not specific to the Sequoyah plant, but the design evaluated here is similar and the parametric trends are representative.

start to deliver water to the secondary side of the steam generators at this time. The auxiliary feedwater pumps draw water from the condensate storage tank for delivery to the steam generators. The auxiliary feedwater system is comprised of two electrically driven and turbine driven feed pumps. Either one of the two electrical or the turbine driven pumps will supply sufficient liquid to the steam generators to remove decay heat.

This sequence assumes the failure of the auxiliary feedwater system. Without the main or auxiliary feedwater system, normal heat removal capability from the primary side is lost. As the remaining liquid in the steam generators is boiled off, the pressure on the primary side begins to increase until the pressurizer relief valve setpoint is reached (Figure A.1-3). The pressurizer relief valve will open and fluid will be discharged into the pressurizer relief tank. An uninterrupted discharge of steam and liquid into the pressurizer relief drain tank will eventually open the rupture disc on the tank and fluid will spill into the containment. This will activate the containment safety features which include containment spray injection system, ice condenser system, and air return fan system.

Steam venting through the pressurizer relief valves causes a gradual depletion in the primary coolant inventory. The charging pumps in the CVCS can be manually activated to provide some make-up. However, the maximum deliverable flow is insufficient to compensate for the fluid loss due to boiloff. The safety injection pumps are unavailable to restore inventory because the system pressure at the relief valve set point is above the pump shut-off head. Hence, the loss of coolant from the primary system will eventually cause the fuel to be uncovered and will subsequently lead to core melt. Containment failure is predicted to occur as a result of the combustion of hydrogen which would be generated during the accident progression.

A.1.2 Operator Response to TML Sequence

The most immediate action the operator must perform is to identify the occurrence of a transient event. The initiating transient for the TML seqence was assumed to be the loss of main feedwater to the steam generators. The instrumentation necessary and sufficient to allow the operator to unambiguously identify the transient is presented in Section A.1.3.

Figure A.1-2 displays, in event tree format, the appropriate actions the operator must take to terminate the accident and bring the reactor to a safe shutdown condition. This figure was developed by modifying the event tree for the TML sequence, as shown in Figure A.1-1, to reflect potential operator actions. The following sections examine the various operator actions and their consequences associated with the relevant sections of this event tree.

A.1.2.1 Response to Initiator (T)

Once the operator has identified the cause of the transient event, in this case loss of feedwater, he would need to identify primary or alternative systems needed to bring the system to a safe shutdown condition. The operator would have to ensure that these systems have functioned as designed when the appropriate signal required for their operation is generated. The first signal that the operator would need to recognize is the reactor trip signal. Once the reactor trip signal is received, the operator would verify that control rod insertion has occurred or manually scram the reactor, if necessary.

After interpreting the instrumentation and identifying the initiating transient as a loss of feedwater, the operator would next ascertain whether auxiliary feedwater system operation has initiated. It is at this time that the operator should discover that the auxiliary feedwater system has failed. It is critical that the operator recognize that this system has failed to function so that he can take appropriate corrective or consequence mitigation actions. Instrumentation needed by the operator to identify this failure is presented in Section A.1.3.

A.1.2.2 Heat Removal

Subsequent to identifying the failure of the AFWS, the principal option available to the operator is to identify the cause(s) of failure of either the main feedwater system or the auxiliary feedwater system, and take corrective action to restore heat removal capability. Restoring either of these systems requires that 1) the fault can be identified and 2) that corrective action can be taken. Although the identification of specific failure modes is not an objective of the current evaluations, some consideration must be given to the cause of the MFWS and AFWS failure because a primary operator action to prevent core melt for the TML sequence involves repair or restoration of one of these systems. In this regard, one particular and highly probable operator response (the use of condensate feed pumps), is examined in this section. Future evaluations should include a detailed review of the fault tree diagrams to identify the most likely modes of failure, particularly common mode failures, in both the main feedwater system and the auxiliary feedwater system.

Calculations performed for the Sequoyah RSS* indicate that the operator would have 60 minutes from the time of loss of feedwater until the steam generator secondary liquid would boil off entirely and three hours until the liquid level in the reactor coolant system reached the top of the fuel rods. After evaluating the failure modes, those failures which can be rectified within the time constraints the accident progression can then be identified and this information made available to the operators. For example, if the two main feedwater system turbine driven pumps cannot be restored to operation within 60 minutes, no steam will be available to drive these pumps. This would then necessitate the operator restoring the auxiliary feedwater system or using the condensate feed pumps, if available, to deliver water to the steam generator. The latter option represents an important operator action, which, if successful, could terminate the accident sequence prior to core melt, or significantly delay the onset of melt.

*This work was performed by Battelle Columbus Laboratories using the MARCH computer code package.

Restoration of cooling water to the steam generators using the condensate pumps requires that the failure(s) which disabled the power conversion system did not also preclude pump operation. The condensate pumps are electrically driven, but are not connected to the emergency AC power supply. Hence, operation of the condensate feed pumps requires the availability of off-site power. In addition, the condensate feed pump discharge pressure is much less than the secondary side pressure in the steam generator. Thus, in order to supply cooling water using only the condensate pumps, pressure in the steam generator must be reduced. Operator action is required to accomplish this. Manual operation of the power operated relief valve will vent steam from the secondary side of the steam generators, and reduce pressure to a level where the operator can activate the condensate feed pumps. In addition to control of the secondary side pressure, the operator must also take action to ensure a sufficient supply of cooling water is available in the event that this mode of operation is necessary for a long period. If steam dump is available and and condenser vacuum can be maintained, the cooling water can be recycled through the normal main feedwater piping. If steam dump is not available, the water inventory in the hot well will eventually become depleted and must be replenished. The condensate storage tank inventory, replenished by the service water or fire protection system sources if necessary, could be utilized to extend the period of heat removal of this mode.

Analysis of TML sequence has indicated that, if the operator cannot restore either main feedwater, condensate feed flow or auxiliary feedwater within three hours, the core will uncover and fuel melt will begin. An analysis which illustrates the system response given restoration of auxiliary feedwater is presented in Figures A.1-8[4] through A.1-12.* These transients show that, assuming auxiliary feedwater flow has been reestablished at 4100 seconds, the primary system pressure and temperature decrease and the core mixture level begins to recover. Soon the system pressure drops to where safety injection is initiated and system liquid inventory which was lost during the transient is replenished. In addition to restoring the auxiliary feedwater systems, the operator would have to ensure proper alignment of the HPIS prior to activation

*This analysis was not specific to the Sequoyah Nuclear Plant, but the parametric trends would be similar if this analysis were performed on Sequoyah.

and verify safety injection initiation. After 8000 seconds the primary pressure has leveled, the pressurizer water level is beginning to increase and indications are that the system has stabilized.

If restoration of main or auxiliary feedwater cannot be accomplished, the operator must find some other way to provide core cooling and stabilize the system. One possible way to do this may be through manual operation of the pressurizer power operated relief valves (PORV's). Figures A.1-13 through A.1-17 present an analysis of a loss of feedwater transient illustrating this operator action. Again, this analysis is not specific to the Sequoyah plant. However, the transients are representative and illustrate an operator action which may prevent core melt. As the steam generators boil dry (Figure A.1-17), and heat removal for the primary system decreases, the pressure in the primary system slowly begins to rise until the PORV setting is reached (Figure A.1-13). Venting through these valves will occur until the primary pressure drops below the closure set point, where the PORV's will automatically reclose. As the primary pressure builds up again, they will reopen. This cycle will continue until the primary inventory is depleted and core melting occurs.

For the TML sequence, the immediate operator action is to restore primary coolant inventory and maintain core cooling. One approach is to lower the primary system pressure to where the HPIS can be activated. If successful this action will provide a mechanism from heat removal and coolant inventory make-up. To accomplish this, the operator must take action to ensure that the PORV's do not automatically reclose as noted above. In the analysis presented in Figure A.1-13 through A.1-17, the operator opened the PORV's at 2500 seconds and maintained venting at the maximum rate. Subsequent to this action, the proper alignment of the HPIS must be checked and its operation verified when the actuation pressure is reached. The operator can then control HPI operation* and PORV valve venting to maintain adequate

*As the primary system depressurizes, the accumulators will automatically inject coolant to assist in inventory make-up.

core cooling. Figure A.1-16 illustrates that core mixture level recovers and no fuel damage is predicted for this scenario.

A.1.2.3 Long-Term Cooling

Once core cooling has been restored, the operator action is directed toward ensuring adequate long-term heat removal. The operator action will depend in part on the method used to restore cooling. If the AFWS has been restored, the operator must verify that the pressurizer PORV has reclosed, thus restoring primary system integrity. When the system pressure has been reduced, the primary liquid inventory must be adequately replenished by the HPIS. Once coolant inventory has been returned to normal, HPIS operation can be terminated and the CVCS utilized for make-up and letdown during plant cooldown.

For the case where the condensate feed pumps are utilized to restore heat removal, the operator actions would be similar to those above (i.e., ensuring primary system integrity and inventory). In addition, the secondary side must be operated in an abnormal mode for plant cooldown, Rather than using the AFWS,** the condensate pumps must supply coolant to the steam generators until the pressure and temperature are reduced to a level where the residual heat removal system can be activated. As noted in Section A.1.2.2, the operator must monitor the condenser hotwell inventory and supplement it if necessary during this process. The LPRS must be aligned for residual heat removal operation, and cooling water for the RHR heat exchangers provided.

If the operator is forced to restore primary heat removal by venting through the pressurizer PORV's (the last option discussed in Section A.1.2.2), he must ensure adequate primary coolant inventory is provided by HPIS. Once

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**This assumes that AFWS is not recovered.

this condition is achieved, the system must be brought to a condition where the RHRS can be activated (approximately 400 psig and 350°F primary conditions). This transition phase of the transient has not been analyzed in detail at this time. Hence, the specific operator actions can not be identified at this time. However, it appears that this can be accomplished by continued venting through the PORV's and operation of the high pressure ECCS.

In addition to providing for heat removal and an orderly cooldown of the primary system, the operator may also have to monitor the performance of the containment ESF's. Depending on the duration of venting through the PORV's, the pressure in the relief tank may increase sufficiently to burst the rupture disk, and release steam into containment. Should this occur the ice condenser and if necessary, the air return fan system and containment spray injection system will provide adequate heat removal capability. The performance of these systems is discussed in Section A.4.1.

A.1.3 Operator Information Requirements

In order for the operator to successfully respond to the events discussed in the previous section, he must be provided with necessary and sufficient instrumentation to allow him to unambigously determine the state of the plant as the accident progresses. Figure A.1-2 will again be utilized as a framework for this section.

The initial task of the operator is to recognize that the transient event has occurred and that the plant is in state 1. The appropriate indication of this state will depend upon the initiating transient. For the majority of the likely initiating transients identified in the Reactor Safety Study, it is not crucial that the operator immediately identify the cause of the transient, because these transients are not initiated as the result of the loss of equipment which would be crucial to termination of the event. Transient events which do

require the operator to identify, within a period of time determined by the assumed failures, the cause of the transient are 1) loss of offsite power or 2) loss of main feedwater. These transients require the use of backup systems (e.g., diesel generators and auxiliary feedwater) to bring the plant to a safe shutdown condition. It is important that the operator verify the successful operation of these backup systems. Therefore, the unambiguous determination of the specific initiating transient is considered essential here, but it is recognized that for some transients, information of a more general nature would be sufficient for the operator to take the required actions.

As already mentioned, the loss of feedwater transient is unambiguously identified by a decrease in the steam generator secondary side level with an increase in secondary side pressure and corresponding increases in primary side pressure, temperature and pressurizer water level. In addition, monitoring of the feedwater pumps and feedwater controllers should provide additional evidence of loss of feedwater. This includes measurements of feedwater and condensate pump discharge pressure and flow rate, feedwater flow controller position, and power and steam supply to the condensate and feedwater pumps.

The low-low steam generator water level signal will generate a reactor trip and a demand for auxiliary feedwater. Should the rods fail to insert, the reactor power will remain at a high level. The control rod position indicators and neutron flux will be sufficient to allow the operator to determine failure of reactor trip, and to take appropriate action to bring the plant to a subcritical condition. In addition, other plant parameter response indicative of successful reactor trip include a sudden decrease in reactor coolant system pressure, temperature, and pressurizer level.

A crucial step for the operator is to identify that the auxiliary feedwater system has failed to start (state 3). Indications of auxiliary feedwater failure would be a continuing decrease in the steam generator water level and an increase in RCS pressure and temperatures the steam generators remove less

energy. Additional indications are the status of components in the auxiliary feedwater system, such as auxiliary feedwater pump flow rate and discharge pressure, feedwater flow control valve position (these valves are normally closed), power supply to the electrically driven pumps, and steam supply to the turbine driven pump.

If it is found that the condensate feed pumps are available to supply flow to the steam generators, the operator will require indications of power supply to the condensate feed pumps and the steam generator power operated relief valves. To reduce steam generator secondary pressure and thereby allow the condensate pumps to supply flow requires the operator to manually open the steam generator power operated relief valves. Successful opening of the PORVs is indicated by a reduction in steam generator pressure. Additional indications would be valve position and discharge line flow. Successful operation of the condensate pumps would be steam generator water level and fluid temperature. Additional indications are condensate pump discharge flow and pressure. The restoration of steam generator cooling will be accompanied by an immediate reduction in primary system pressure and temperature (see Figures A.1-8 and A.1-9). For long-term operation in this heat removal mode, the operator would require knowledge of the condenser hotwell inventory, condensate storage tank level, and other parameters required to assure an adequate water supply for pump suction.

If the operator is unable to restore heat removal through the steam generators, the only other action which could potentially prevent core melt requires venting steam through the pressurizer PORVs and lowering primary pressure to where the HPIS can be activated to restore inventory. In order to take this action, the operator must know the position of the relief valves and the discharge line flow. The effectiveness of this action can be monitored by observing the RCS pressure. When the primary has depressurized to the HPIS activation level, the operator must verify successful operation of this system (or activate it manually). The primary effect of the addition of water from the HPIS will be a gradual recovery

in core water level (Figure A.1-16). Because of the many variables associated with the plant response to TML and this specific operator action, the primary pressure and core temperature may not give an immediate indication of HPIS activation (see Figures A.1-13 and 15). However, failure of the HPIS would soon result in rising core temperatures as well as a continuing decline in core water level. Confirmation of successful HPIS operation is also provided by the measurement of pump discharge pressure or flow.

The operator response after restoring heat removal is to bring the plant to a safe, cold shutdown condition. However, the subsequent operator actions are dependent in part, on the method utilized to arrive at state 4a. If heat removal through the steam generators has been restored, the operator must verify that all primary relief and safety valves have reclosed (state 5a).* This can be accomplished by monitoring the valve position or the discharge line flow. Should these valves fail to reseat (state 5b), the system would in effect have a small break LOCA. However, this event is probabilistically insignificant when combined with the multiple failures which initiated the TML accident, and therefore has not been considered in this analysis.

Because of the unique role of operator response to the TML, evaluations have not been performed to determine the specific steps required to bring the plant to a cold shutdown condition. For this reason, a list of operator actions and instrumentation requirements is subject to uncertainty at this time. The principal items are noted in the following discussion. Certainly, the primary system temperature and pressure would have to be monitored to ensure effective cooling. Long term inventory control would require the use of the CVCS. Indications of the CVCS component status necessary will be charging the pump flow rate and discharge pressure,

*However, if heat removal were available only through PORV venting, then continued operation of these valves would be required.

and volume control tank liquid level. Proper operation of the CVCS will be indicated by the response of the reactor coolant system pressure, temperature and liquid levels.

The effect the transition to RHR operation, the operator must ensure that the primary system temperature and pressure are reduced to the appropriate level. The correct alignment of the low pressure system for RHR operation requires knowledge of the valve positions. Measurement of flow and coolant temperatures in the RHR heat exchangers will establish that they are ready for operation, while pump discharge pressure or flow will verify that coolant is being delivered to the primary coolant system. The effectiveness of RHRS operation can be monitored by observation of the primary system pressure and temperature.

A.1.4 Conclusions

In the preceding sections, the TML-γ sequence was evaluated with the purpose of identifying operator actions and necessary instrumentation needed to terminate or mitigate the consequences of this sequence. Table A.1-1 presents a summary of the results of this evaluation.

The information presented in the summary table is based on a number of assumptions concerning the plant performance and response to the postulated sequence. Many of the plant conditions and proposed operator actions have not been analyzed in the past. Hence, there is some uncertainty and generality in these evaluations. The following list identifies areas where further information would be beneficial in either confirming the key assumptions used in this study, or reducing the level of uncertainty.

- The utilization of the condensate feed pumps to supply cooling water to the steam generators has been identified as a potentially important operator action. Plant response characteristics for the cases where the condensate feed pumps are utilized are required to provide a more definitive accident signature, and to facilitate the delineation and timing of operator actions in bringing the core to a stable condition. A particularly important parameter in this regard is the timing of feedwater restoration.

- The conditions under which the condensate pumps can be utilized need a more thorough investigation. Specifically, what additional components of the PCS are required for this action to be effective, and will they be available under the conditions associated with the more probable MFWS failure modes? What measurements are required by the operator to check the status of these

components and verify their correct operation? A more detailed evaluation of specific operator responses and systems capabilities is needed to establish the effectiveness of this mode of operation for extended periods. This includes ensuring adequate water supply for pump suction, and definition of the actions necessary to switch to RHRS operation.

- The use of the PORV's to remove heat, assuming a complete loss of cooling through the steam generators requires a more thorough investigation. For example, it is uncertain if this action would be effective for the Sequoyah plant.* Existing analysis indicate that prompt operator response (i.e. manually opening the PORV's to depressurize the primary system) is critical to preventing core damage. Hence, the effect of operator response time merits further study. Specifically, how long can primary depressurization be delayed before HPIS operation is no longer effective? In addition, if the action is effective in restoring primary heat removal and inventory, the system response for long periods requires further analysis to permit a better definition of the operator actions necessary to bring the system to a safe, cold shutdown in the absence of secondary system heat removal capability.

- The fault tree diagrams for the MFWS and the AFWS should be reviewed to assess the capability to restore these systems for the most probable failure modes (as discussed in Section A.1.3).

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*The analysis illustrated in Figures A.1-13 through 17 were performed for a different plant design.

- To complete the development of the accident signatures, analysis of the containment response should be performed for those sequences which release steam into containment.

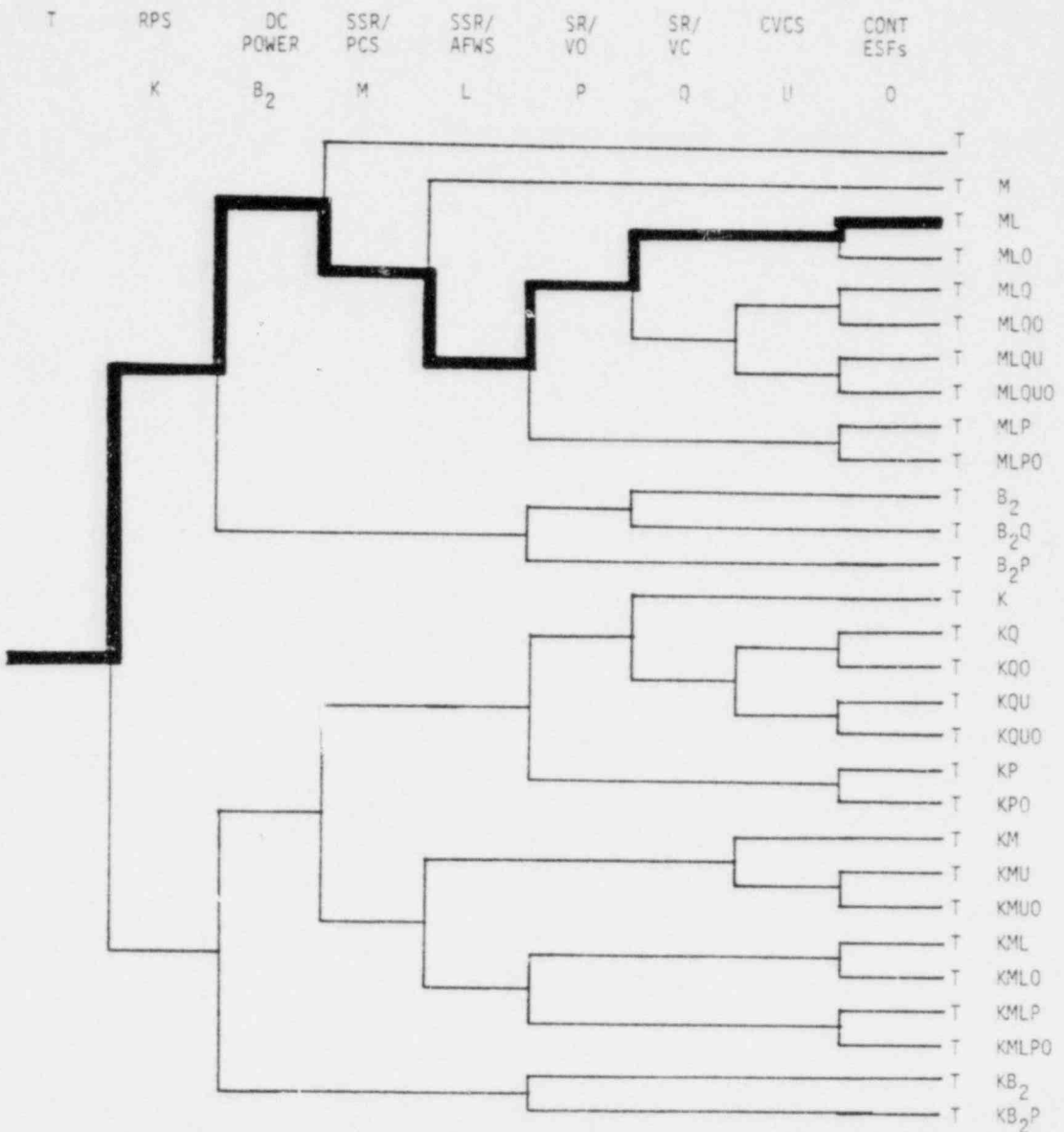

Figure A.1-1. Sequoyah Transient Event Tree

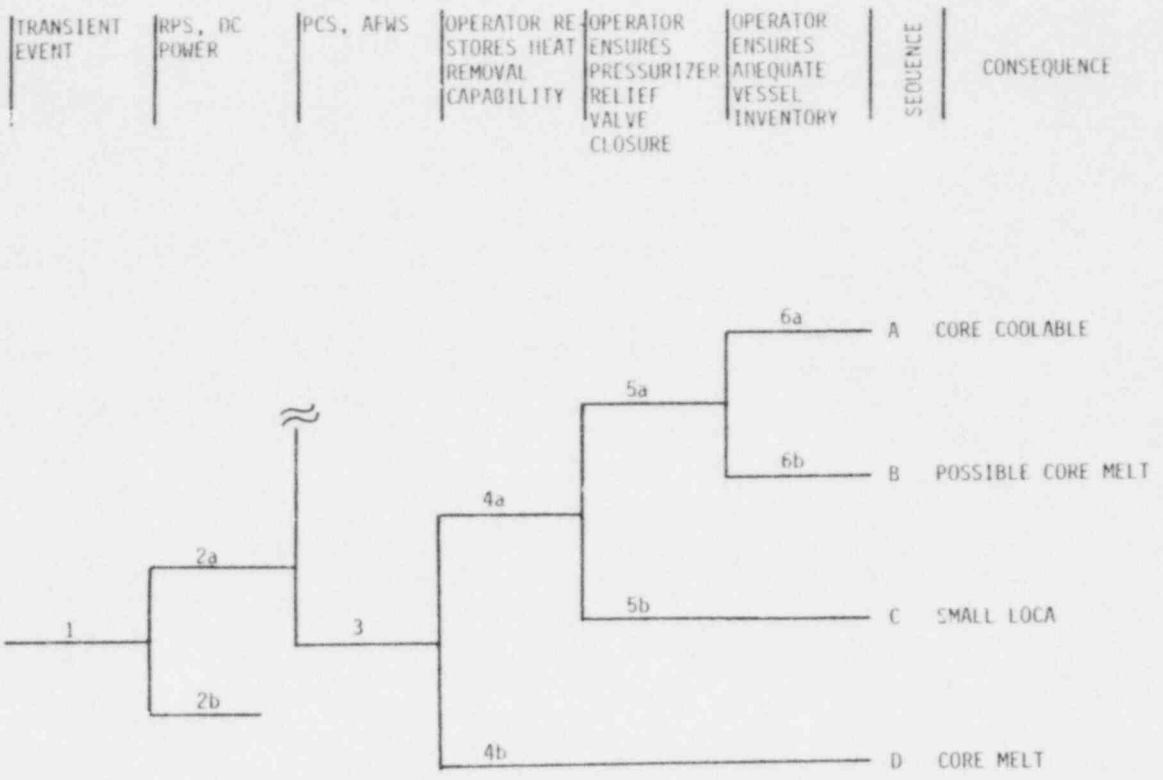

Figure A.1-2. TML Sequence Operator Action Event Tree

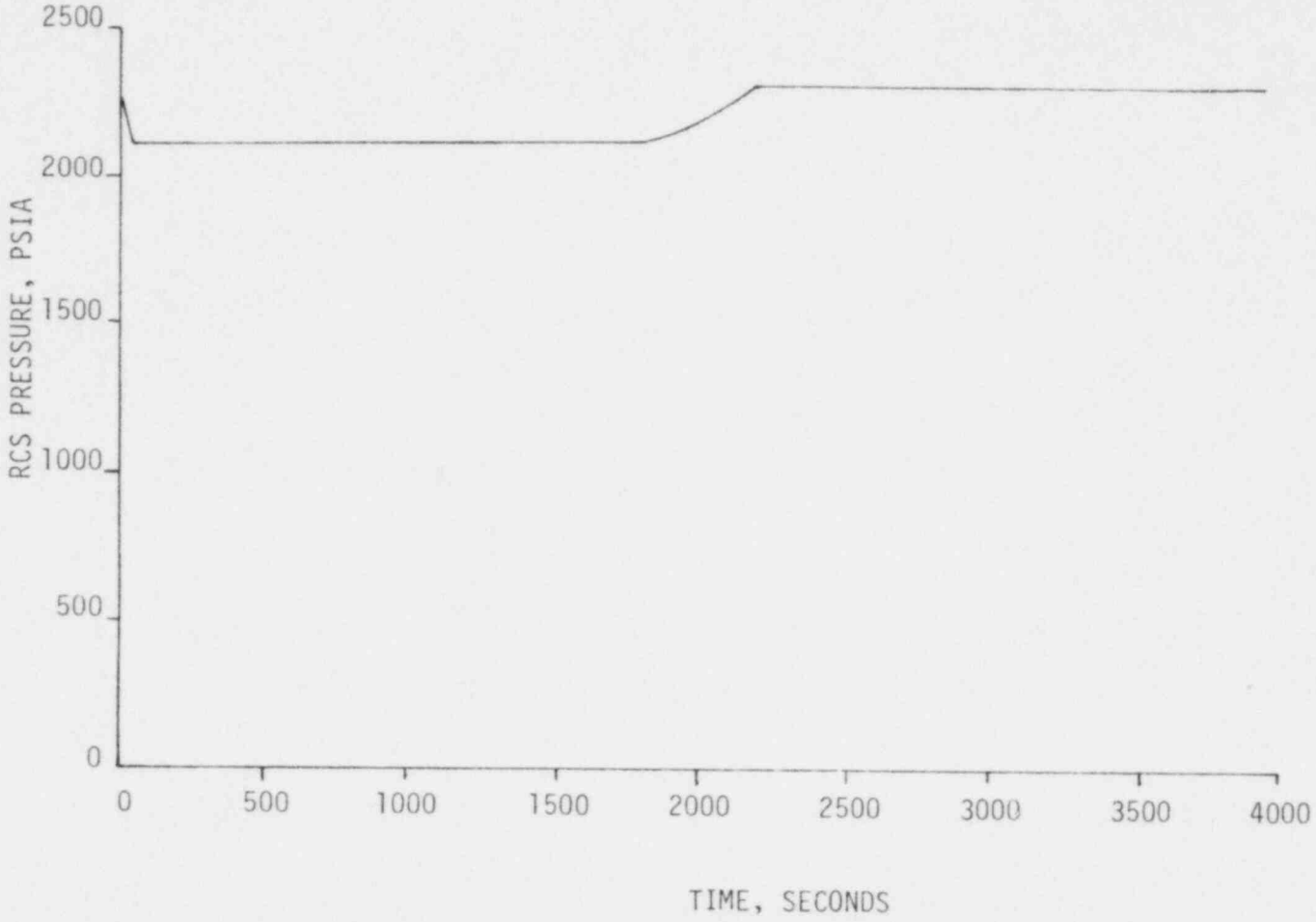

Figure A.1-3. Loss of Feedwater Transient: RCS Pressure vs. Time

Figure A.1-4. Loss of Feedwater Transient: Core Outlet Temperature vs. Time

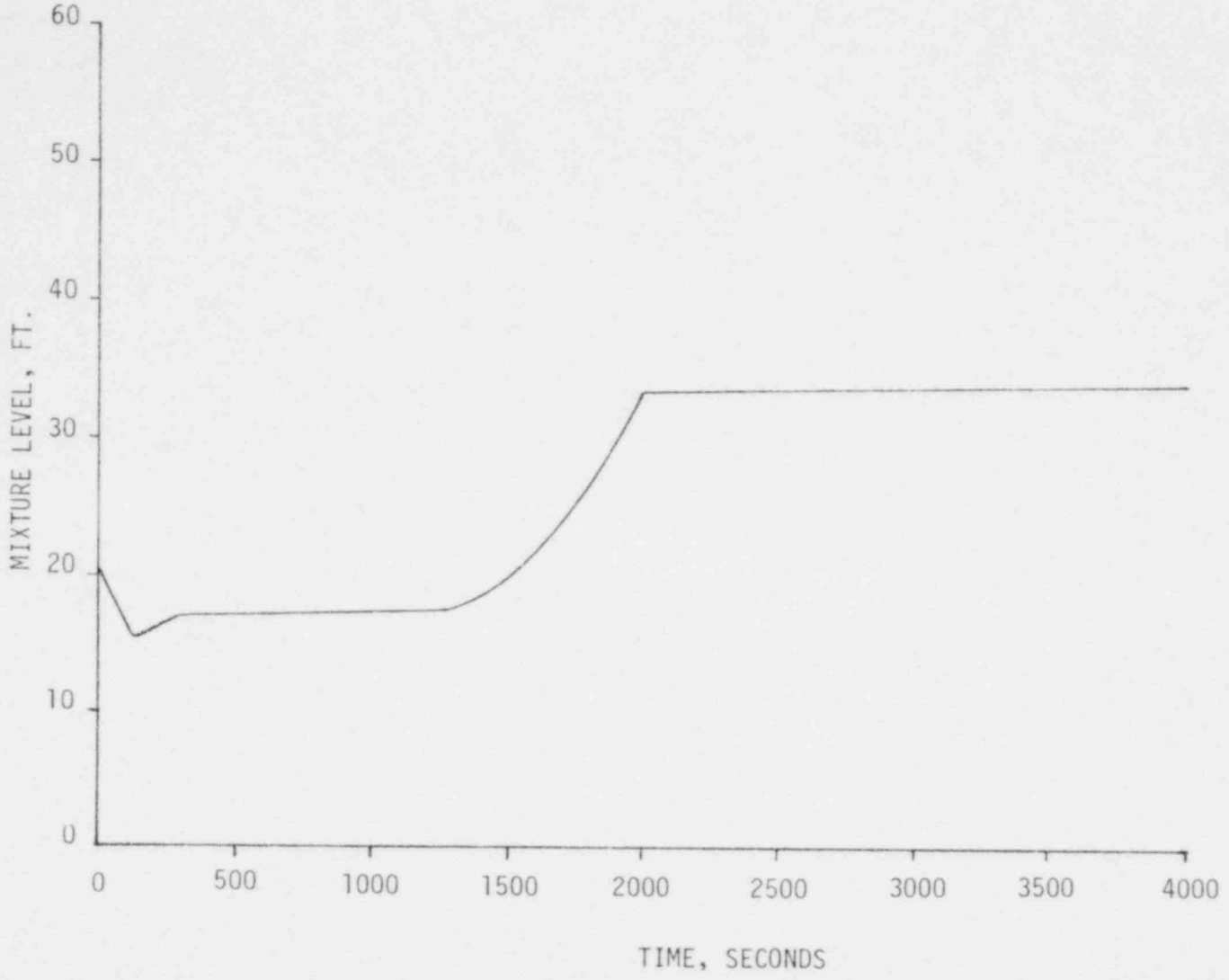

Figure A.1-5. Loss of Feedwater Transient: Pressurizer Level vs. Time

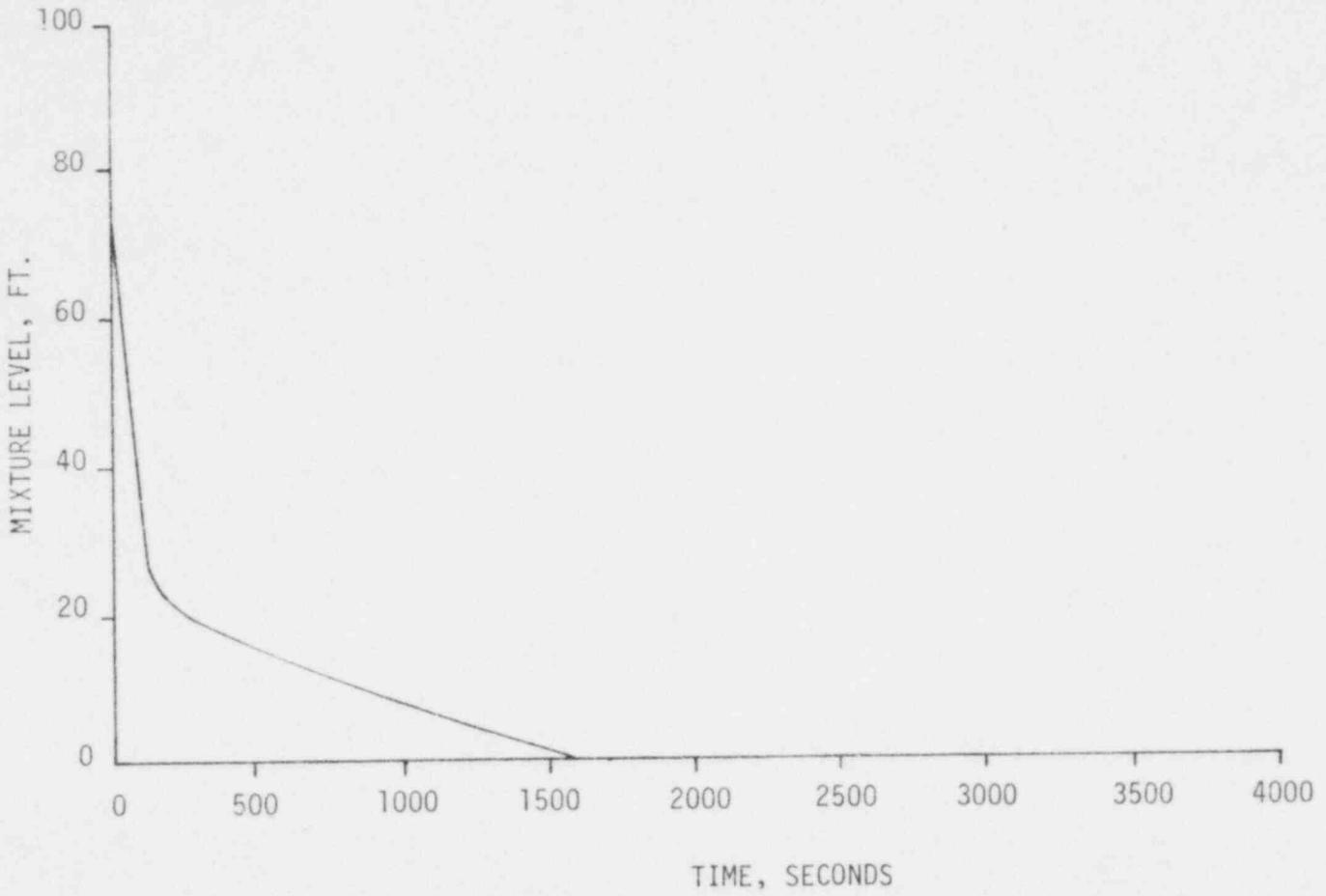

Figure A.1-6. Loss of Feedwater Transient: Steam Generator Secondary Side Mixture Level vs. Time

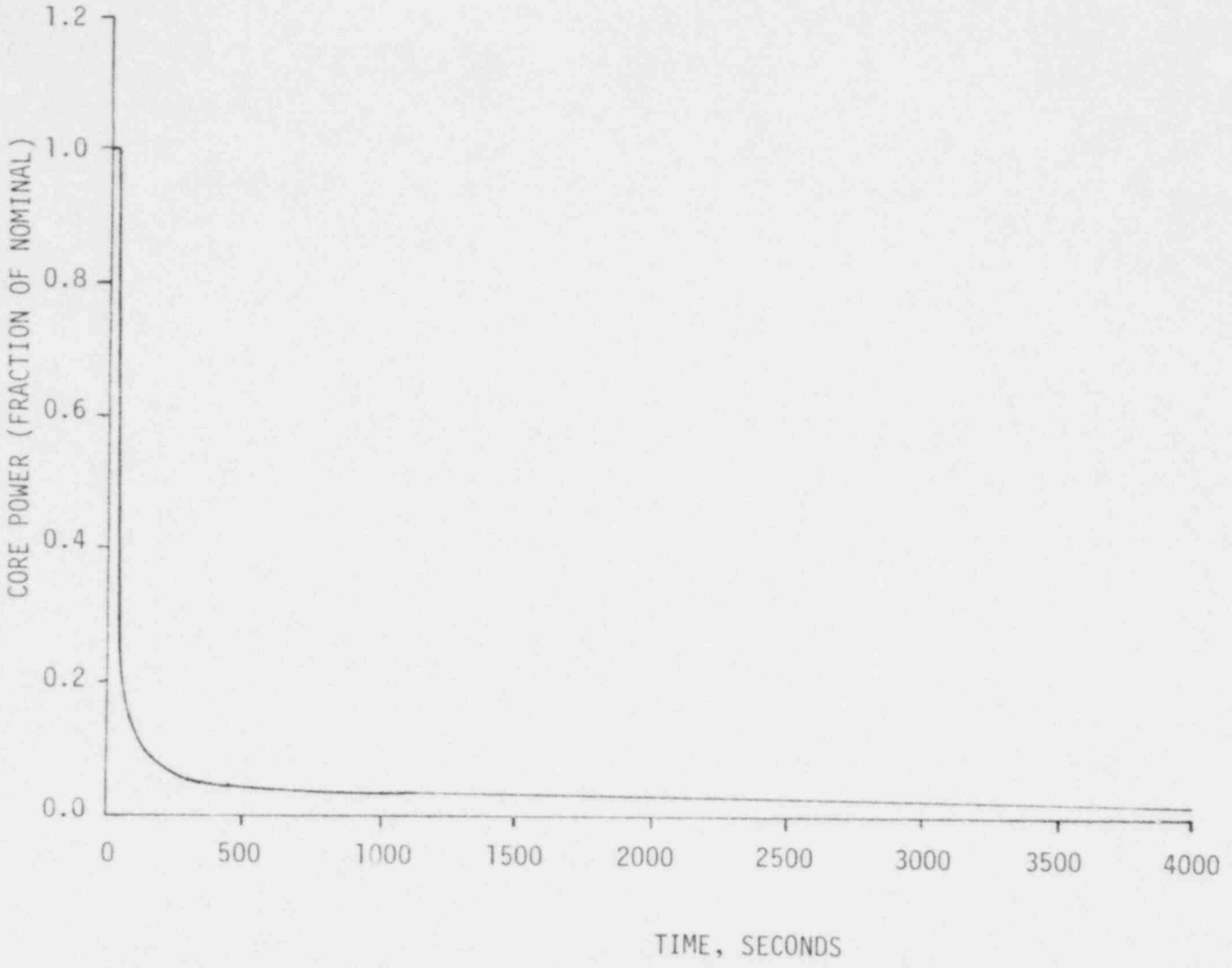

Figure A.1-7. Loss of Feedwater Transient: Core Power vs. Time

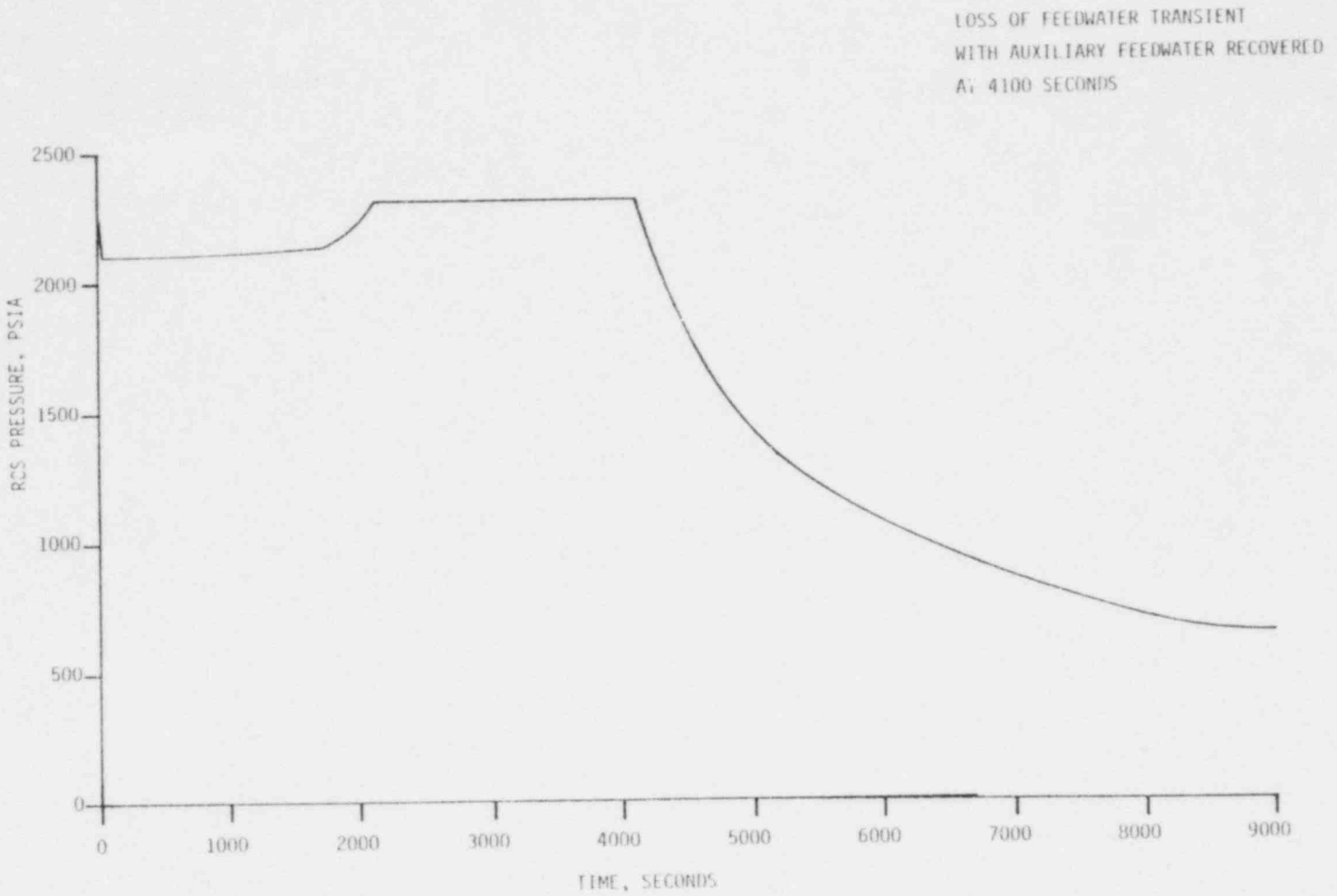

Figure A.1-8. Reactor Coolant System Pressure vs. Time

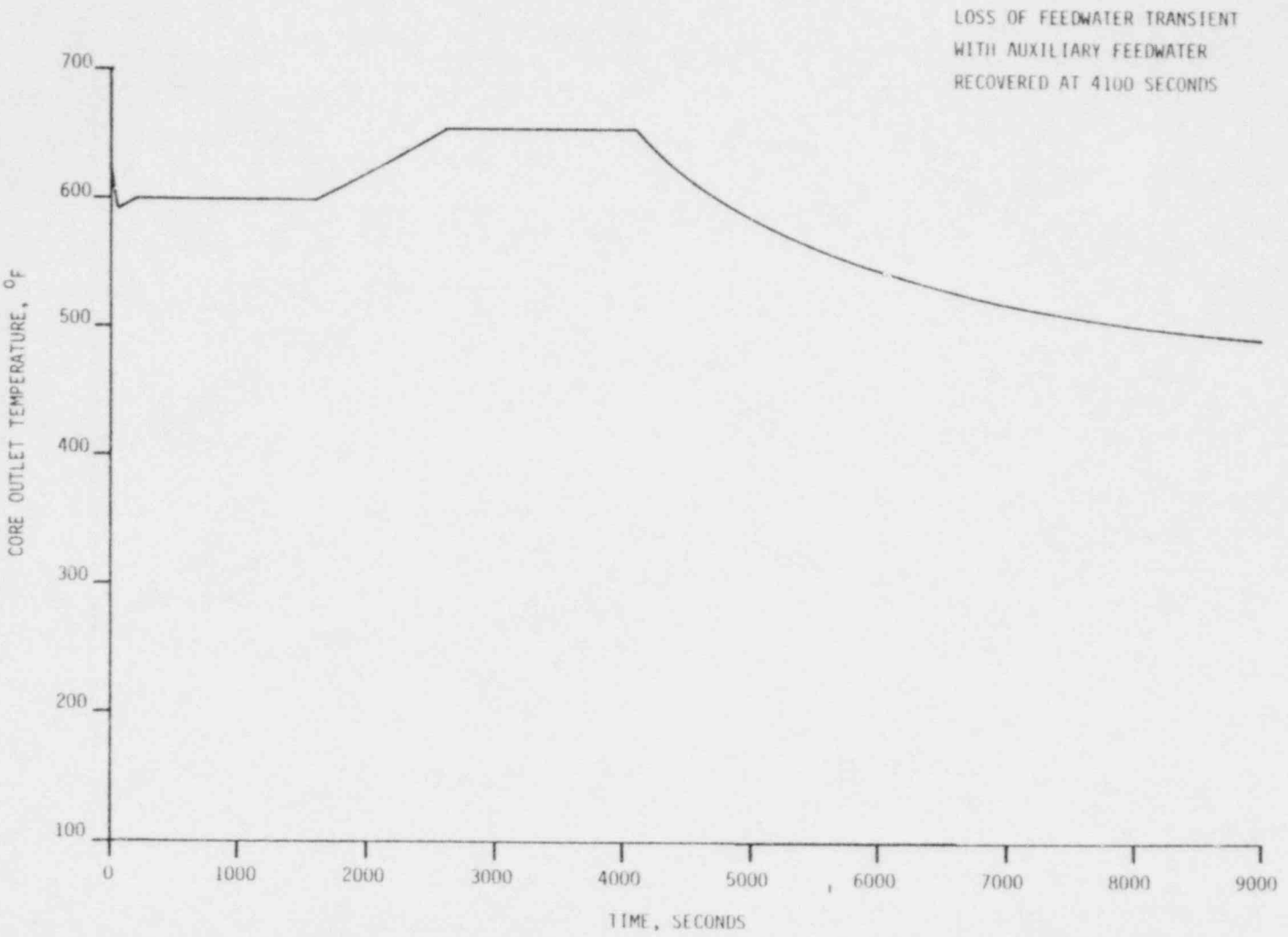

Figure A.1-9. Core Outlet Temperature vs. Time

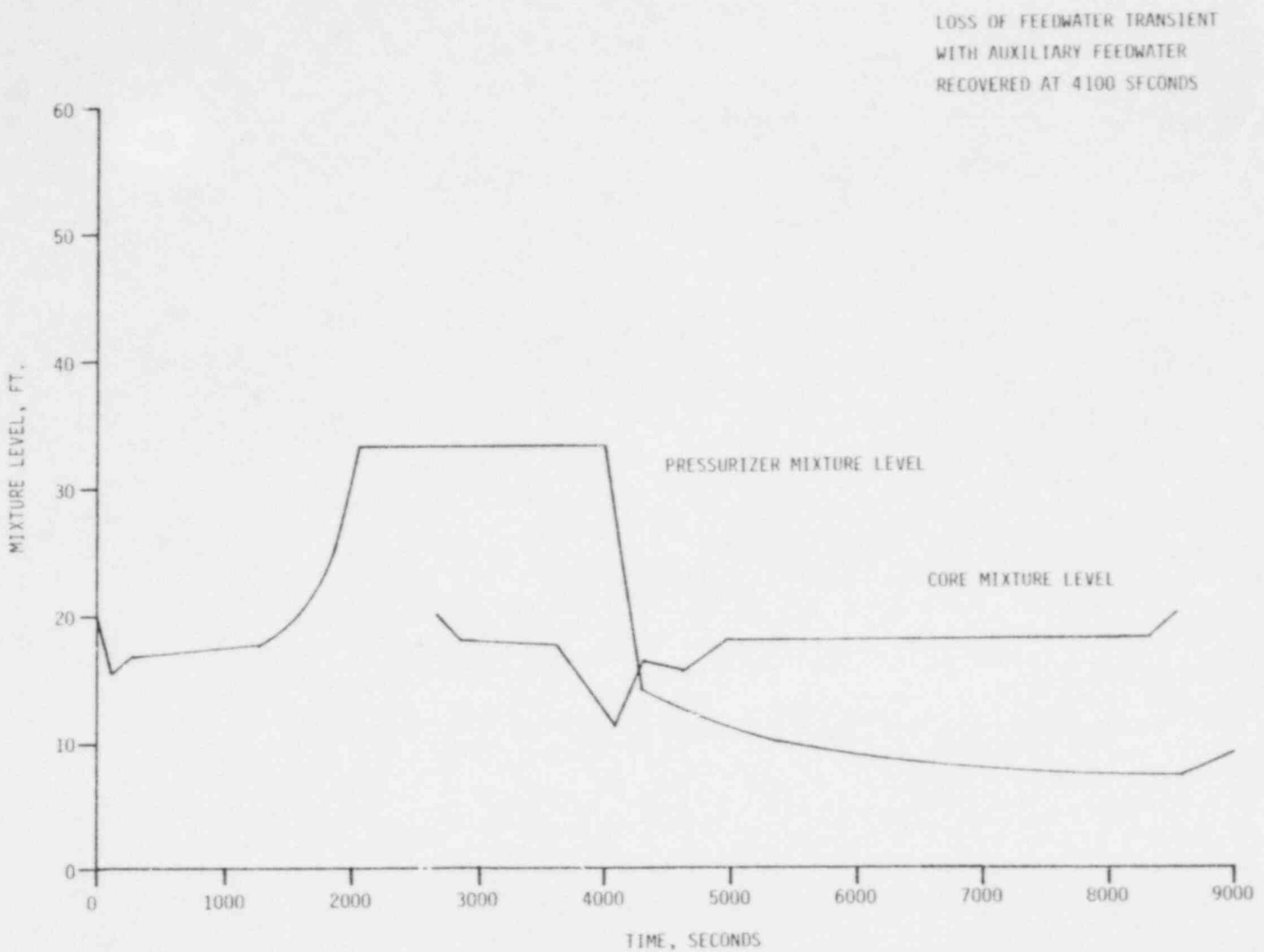

Figure A.1-10. Core and Pressurizer Mixture Level vs. Time

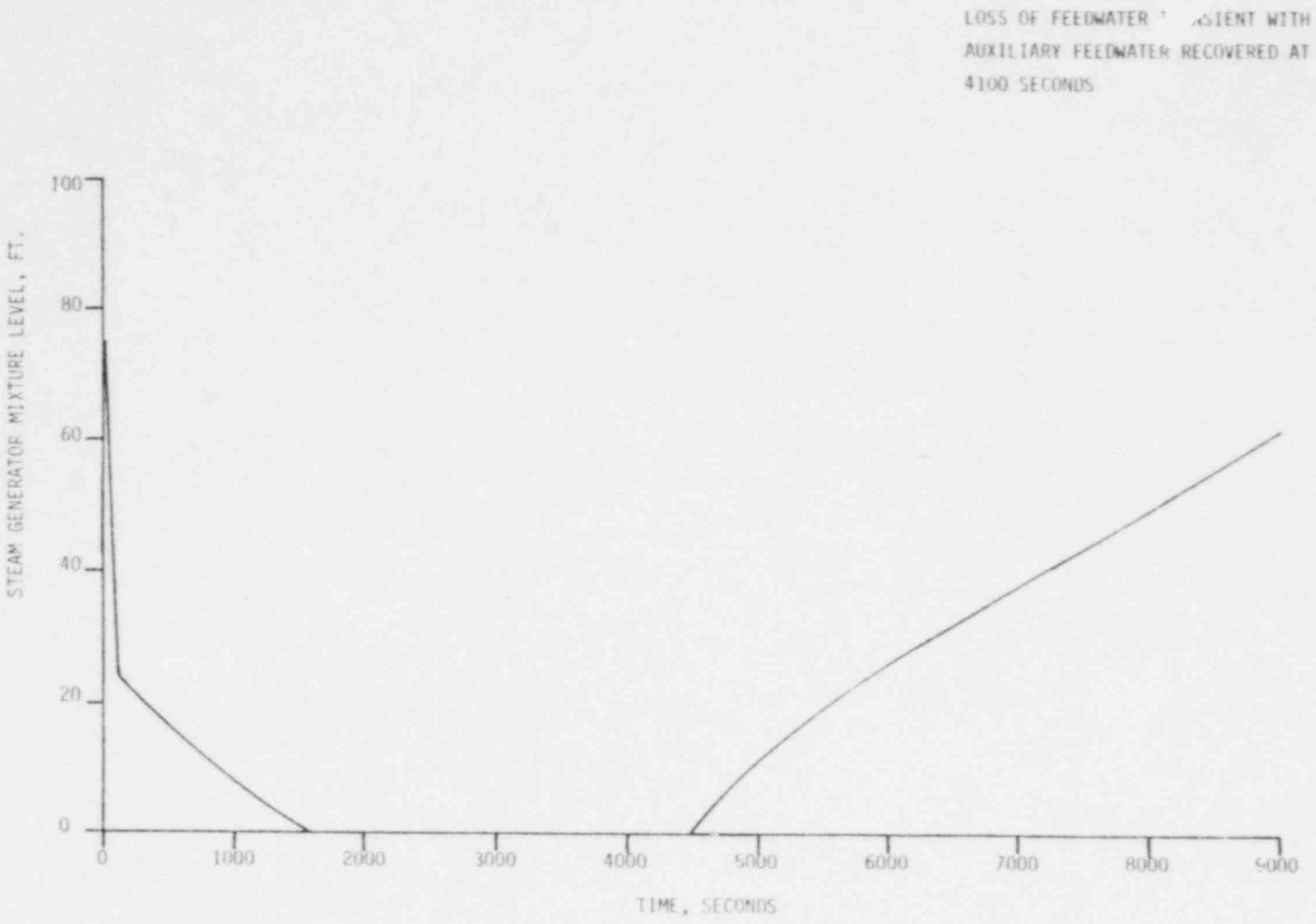

Figure A.1-11. Steam Generator Secondary Side Mixture Level vs. Time

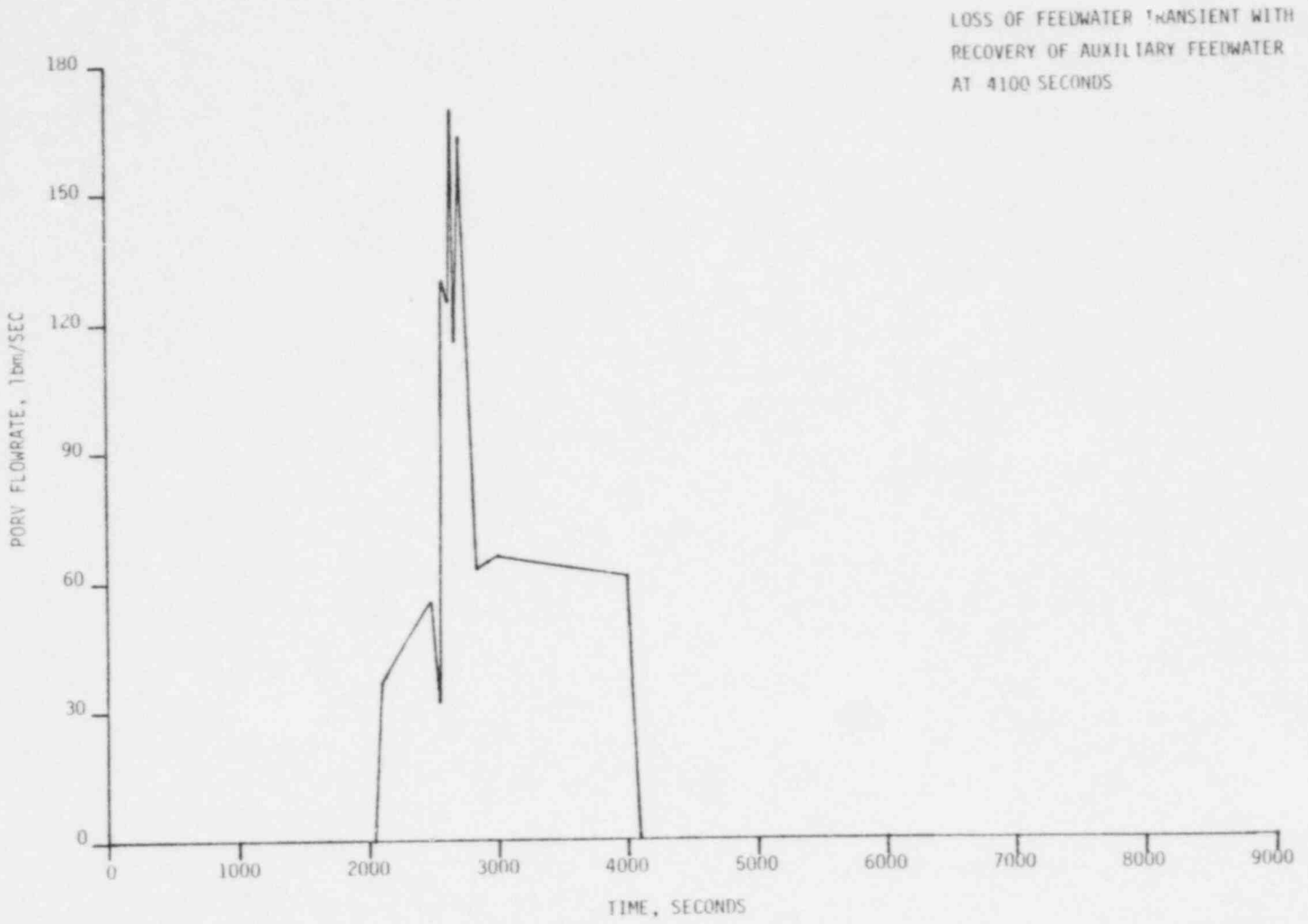

Figure A.1-12. Power Operated Relief Valve Flowrate vs. Time

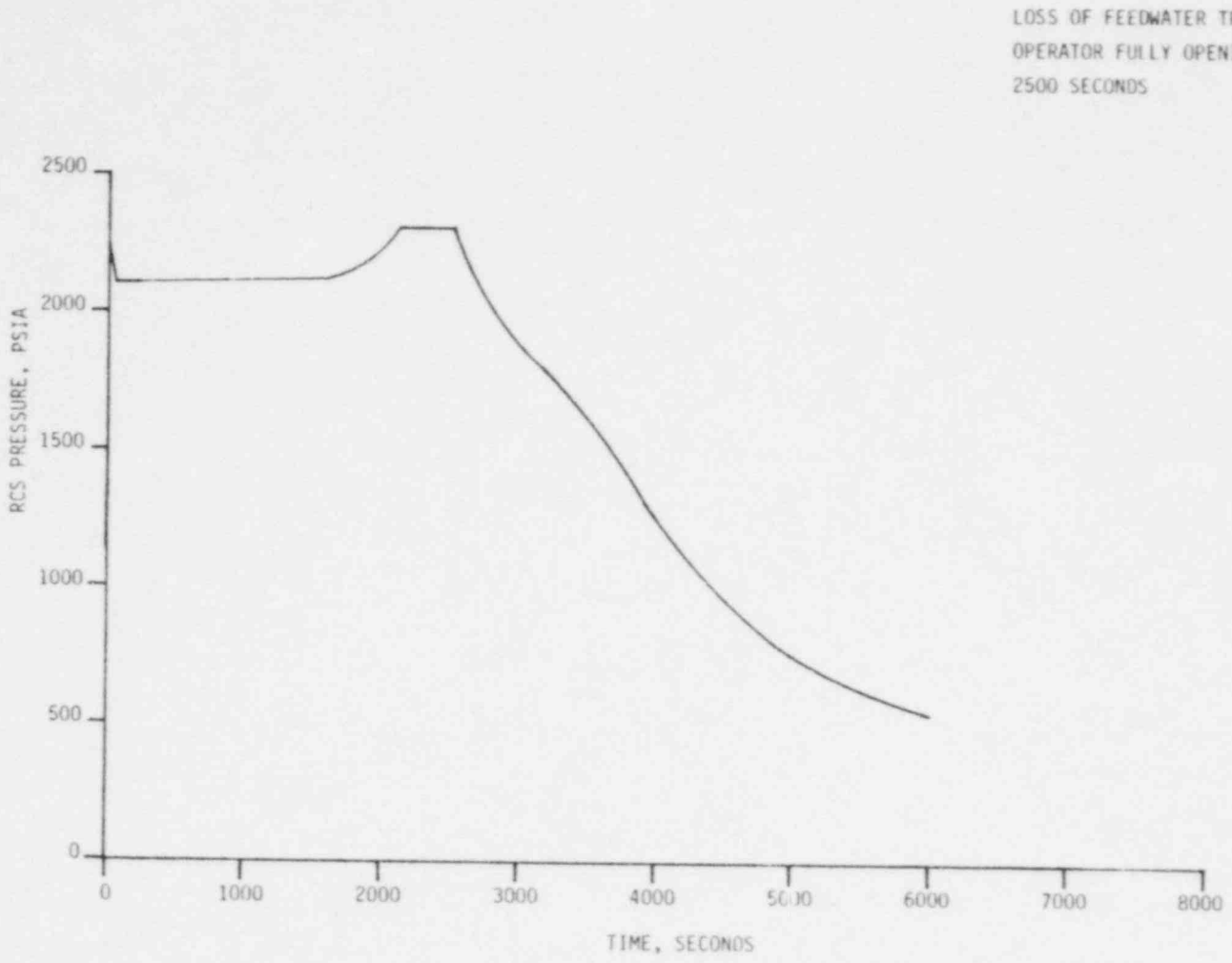

Figure A.1-13. Reactor Coolant System Pressure vs. Time

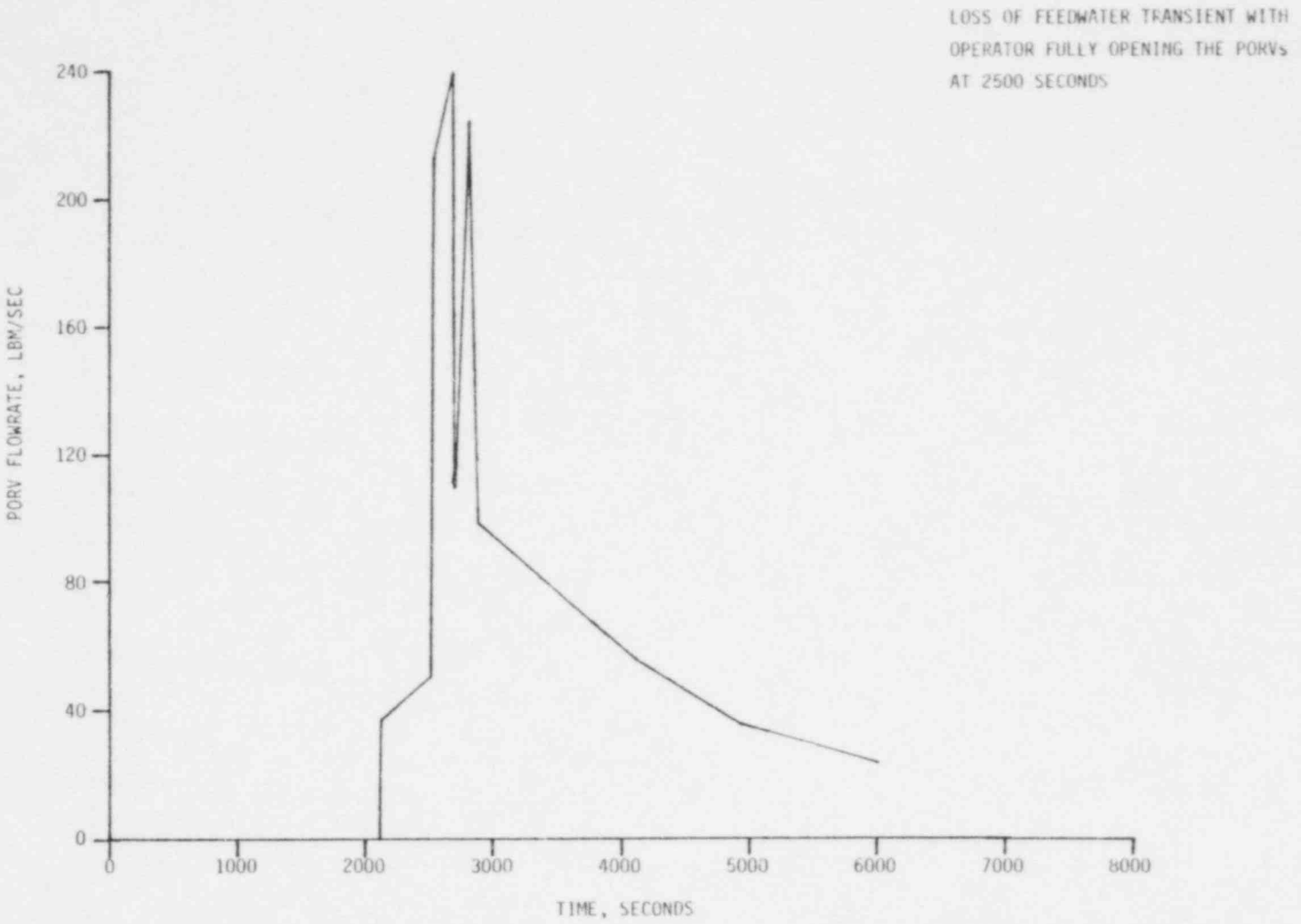

Figure A.1-14. Power Operated Relief Valve Flowrate vs. Time

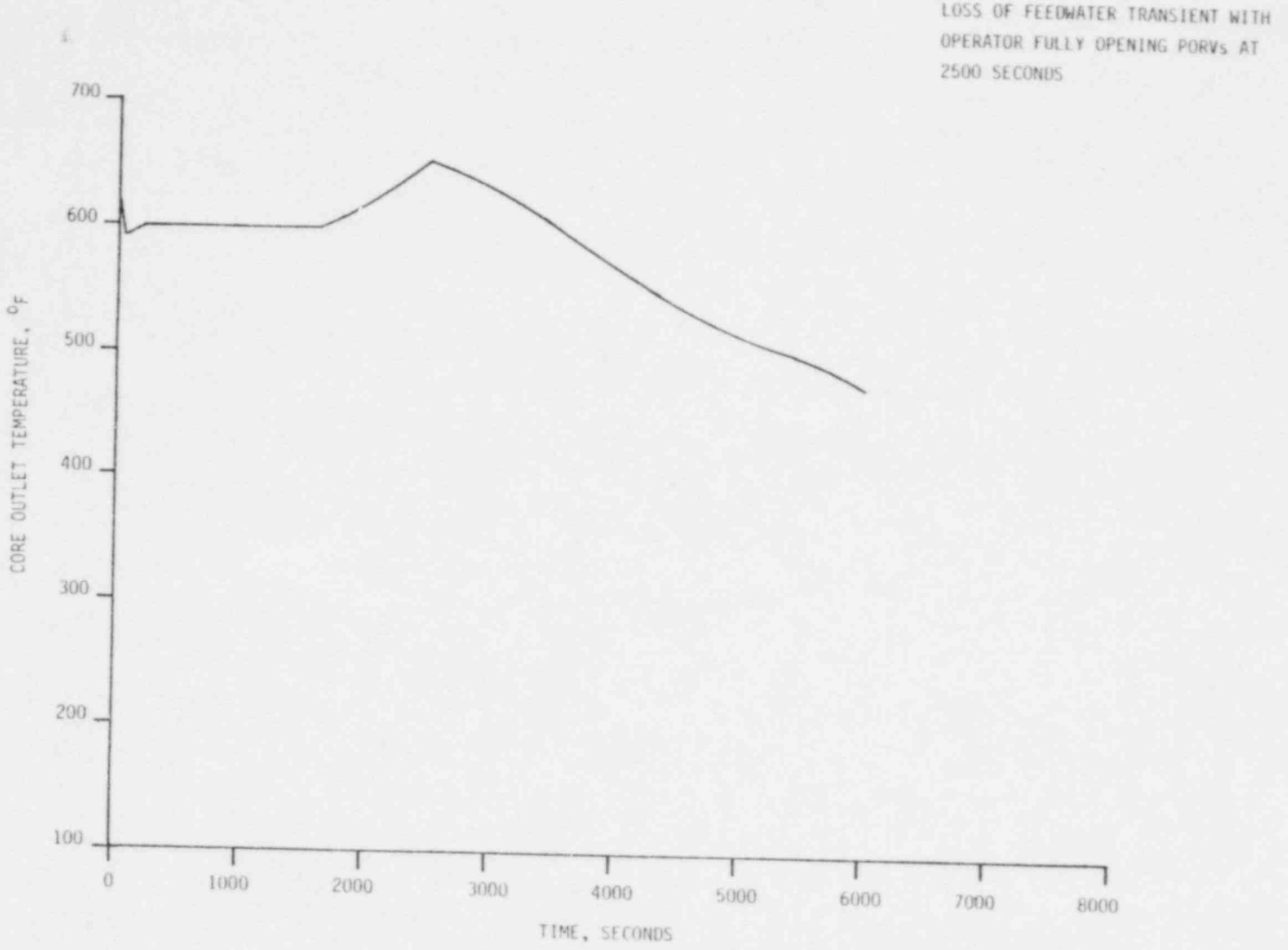

Figure A.1-15. Core Outlet Temperature vs. Time

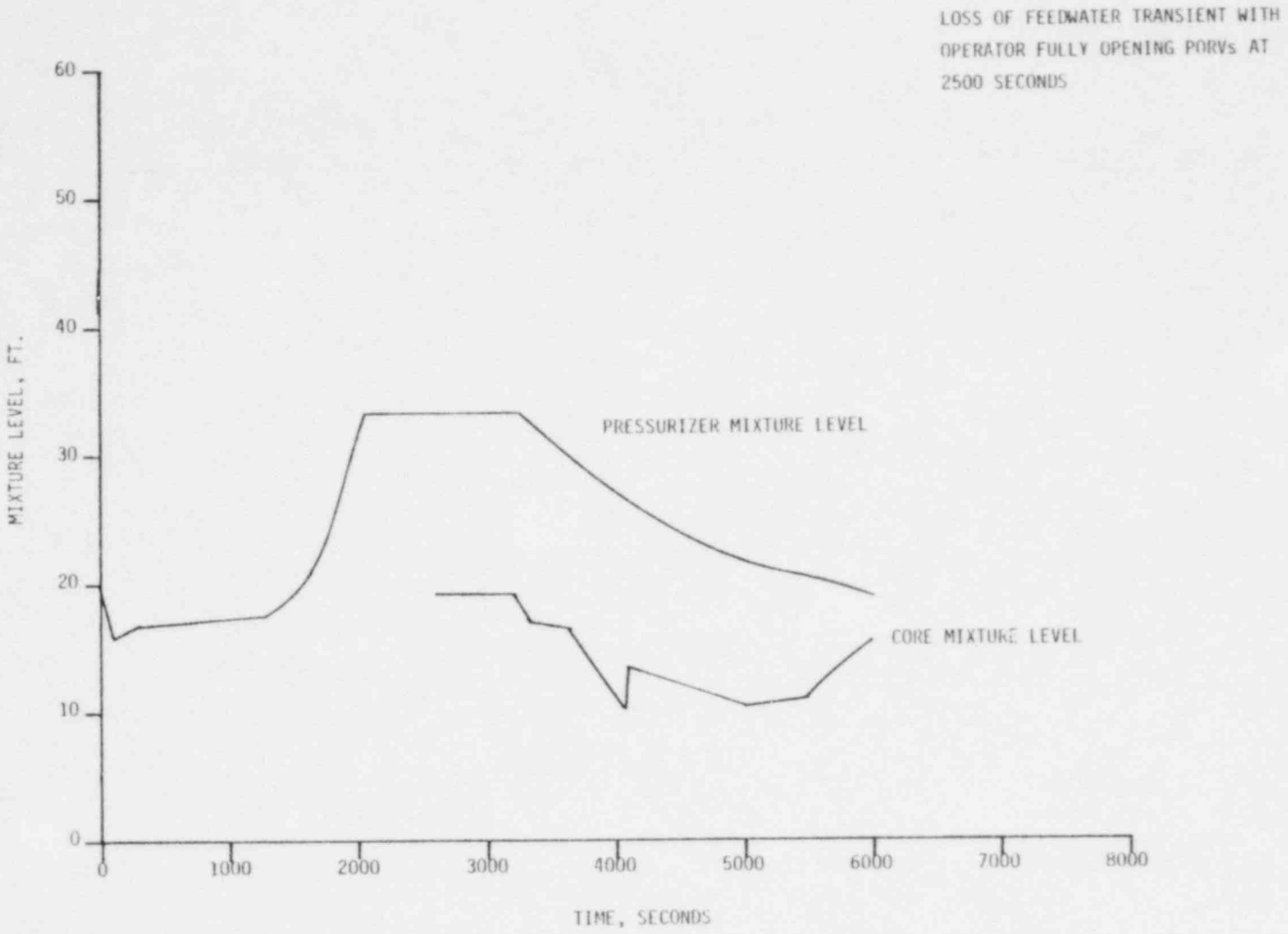

Figure A.1-16. Core & Pressurizer Mixture Level vs. Time

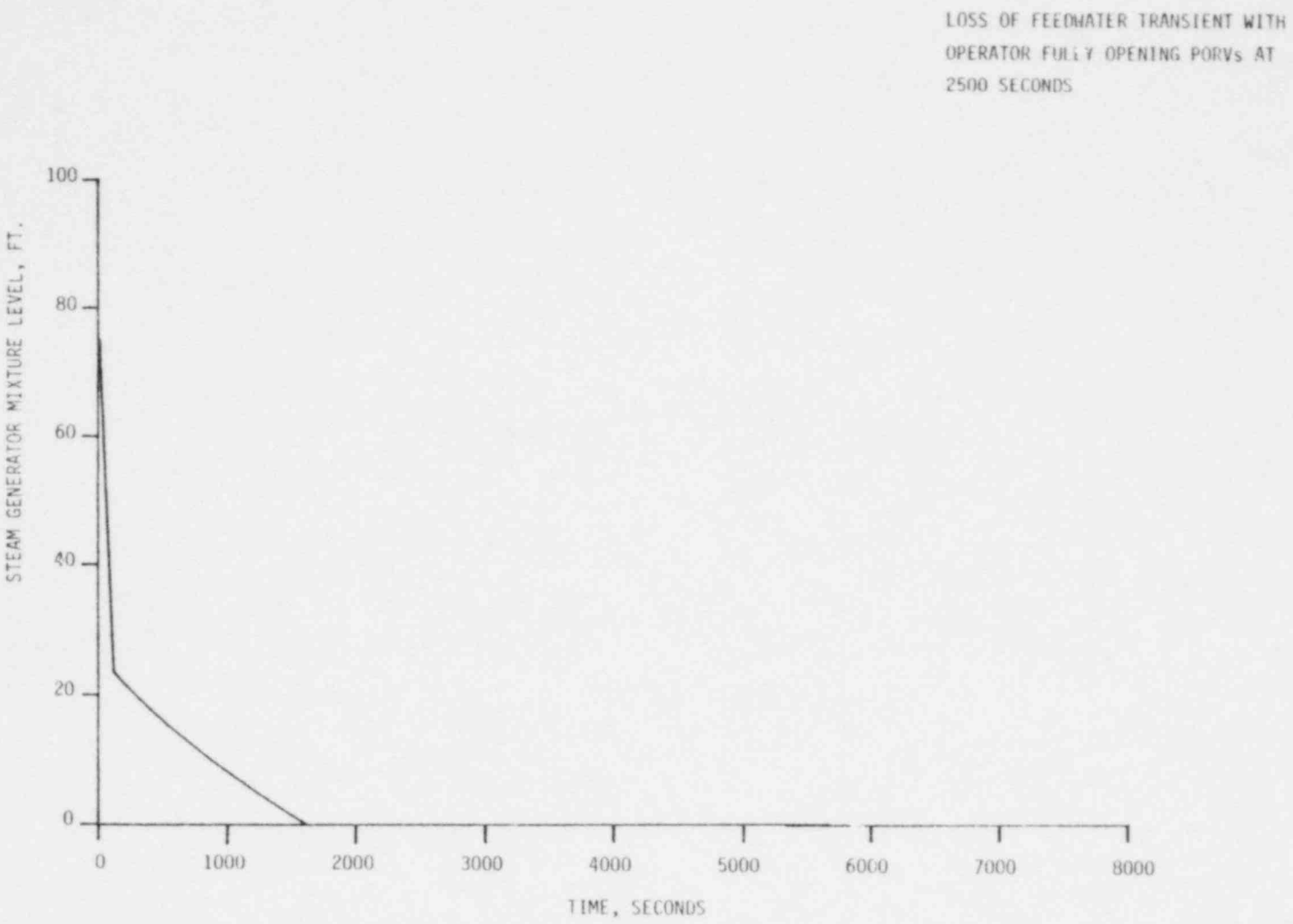

Figure A.1-17. Steam Generator Secondary Side Mixture Level vs. Time

Table A.1-1

SUMMARY OF KEY OPERATOR ACTIONS AND INFORMATION REQUIREMENTS FOR TMI-1 SEQUENCE

Plant State	Description of Plant State	Information Required for Plant State Identification	Operator Action Following Plant State Identification	Information Required to Take Appropriate Action
1	Loss of feedwater transient has occurred	o steam generator water level; o feedwater flow indication; o feedwater and condensate pump discharge pressure and temperature; o RCS pressure and temperature o pressurizer water level	Verify electric power availability Verify reactor shutdown	o See plant state 2a
2a	Reactor shutdown by reactor protection system, electric power supply available	o control rod position indicator, o neutron flux o RCS pressure and temperature o pressurizer water level o electric power supply to key plant switch gear	Prepare for plant shutdown using AFWS	o auxiliary feed pump flow rate and discharge pressure, o flow controller valve position, o power supply to electrically driven pumps; o steam supply to turbine driven pump o steam generator level
3	Failure of auxiliary feedwater system	o auxiliary feed pump flow rate and discharge pressure o steam generator level	Restoration of MFWS or AFWS; or manual operation of pressurizer PORV's to reduce primary system pressure and actuate HPIS	o Same as required for state 1 identification and to take action subsequent to state 2a identification* o MFW flow controller position indication o Main feed pump steam supply indication o Condensate pump power o Parameters defining cause of MFWS and/or AFWS failure o Steam generator PORV position or discharge line flow o PORV position or discharge line flow o Pressurizer relief tank pressure, temperature or level.

*The operator actions taken to restore heat removal will depend on which of these parameters are required. The first procedure for state 3 identification include those necessary to accomplish all actions called for in the operator response column.

Table A.1-1
(continued)

Plant State	Description of Plant State	Information Required for Plant State Identification	Operator Action Following Plant State Identification	Information Required to Take Appropriate Action
4a	Successful restoration of either main or auxiliary feedwater system or use of pressurizer PORVs supplemented by HPIS injection	o same as for states 1 and 3 identification o PORV valve position or discharge line flow o HPIS flow or pump discharge pressure	Verify that pressurizer relief valves have reclosed* Ensure adequate liquid inventory and core heat removal Ensure correct response of containment ESF's if actuated	o Pressurizer safety and relief valve discharge line flow o Pressurizer safety and relief valve position indicator o Core water level o Pressurizer drain tank level o RCS pressure and core temperature o Containment pressure and temperature
4b	Failure to restore operation of either main or auxiliary feedwater systems or use PORV's to remove heat	o same as state 4a	Monitor approach to core melt; take consequence mitigating actions Use CVCS to supply liquid to core	o Core outlet temperature o Core water level o Containment pressure and temperature o Containment radiation level o Coolant radiation level o CVCS flow and discharge pressure
5a	Pressurizer relief valve has properly reseated after system pressure has dropped below system setpoint*	o valve position indication; o discharge line flow	Maintain adequate vessel liquid inventory	o Charging pump flow rate and discharge pressure o Volume control tank level o RCS pressure and temperature o Core Water level
5b	Pressurizer relief valve fails to reseat properly*	o Same as for plant state 5a identification	Follow procedures for dealing with a small LOCA	o RCS pressure and temperature o Availability of appropriate ESFs

*These conditions may not apply in the short term for the case where heat removal through the steam generators is unavailable and the operator must vent through the PORV's.

Table A 1-1
(continued)

Plant State	Description of Plant State	Information Required for Plant State Identification	Operator Action Following Plant State Identification	Information Required to Take Appropriate Action
6a	Adequate vessel liquid inventory is maintained	o RCS pressure and temperature; o core water level	Monitor system parameters in preparation to bring plant to cold shutdown condition	o RCS pressure and temperature o RHR component status and alignment o RHR heat exchanger cooling water flow and temperature
6b	Adequate vessel liquid inventory not maintained	o Same as for plant state 6a identification	Monitor approach to core melt take appropriate consequence mitigation actions	o Same as for response to state 4b identification

A.2 TMLB' SEQUENCE

A.2.1 Sequence Description

The TMLB' sequence was identified in WASH-1400 as a significant risk contributor for the Surry PWR plant. This sequence is initiated by a loss of offsite power transient (T) and involves the subsequent failure of both the Main Feedwater System (M), and the Auxiliary Feedwater System (L). The loss of main feedwater is assumed to be caused by the initiating loss of offsite power and the failure to restore the power source within one hour. The failure of the auxiliary feedwater system is caused by the coincident loss of onsite emergency AC power and the failure of the steam turbine driven auxiliary feedwater pump. The unavailability of both offsite and onsite power would also preclude the use of emergency core cooling systems, containment heat removal systems, and power operated relief valves. In addition, non-recovery of either offsite or onsite AC power for the containment ESFs within a period of about one to three hours is postulated (B').

The response of the reactor coolant system parameters to the TMLB' sequence is similar to the response of these parameters for the TML sequence which was presented in Secion A.1. The analysis presented in Section A.1 was for a different reactor than the one analyzed in the RSS; however, the trends would be similar, and Figures A.1-3 through A.1-7 can be used in conjunction with a description of the TMLB' sequence. Analysis which would provide a representative accident "signature" of the TMLB' sequence was not available for inclusion in this report. A generalized description of the TMLB' sequence using a composite of information from the TML sequence and calculations done for the RSS is presented below.

Following the loss of both main and auxiliary feedwater, the pressure in the primary side of the reactor coolant system will increase until the pressurizer safety valves are opened. Fluid will be discharged through the safety valves into the pressurizer relief tank. Since all AC

power is lost, there exists no way to prevent an excessive coolant loss through the RCS safety and relief valves or provide heat removal. The system will continue to vent steam until the primary inventory is depleted, and core melt will occur. In addition, due to the loss of all AC power sources and the failure to restore these sources within an acceptable time period, containment ESFs could not operate to mitigate the effects of the core melt.

In WASH-1400, a number of significant containment failure modes were identified for the 'TMLB' sequence. The particular mode selected for this analysis is containment overpressure. Due to the failure of the containment ESFs, the containment pressure will rise uninterrupted until the containment burst pressure is exceeded, at which time rupture of the containment shell will occur. Calculations performed for the RSS indicate that the time frame for containment rupture is 200 minutes subsequent to the loss of power initiator, with fuel melt occurring between 170 to 220 minutes. Contributors to this pressure buildup are steam released from the RCS, noncondensible gases (H_2) generated during core melt and energy released during hydrogen burning.

In the following sections, the key operator actions in response to the sequence of events described above are delineated and the information necessary to allow the operator to efficiently take these actions is identified.

A.2.2 Operator Actions

The previous section described the sequence of events which were determined in WASH-1400 to lead to core melt, failure of containment, and release of radioactivity to the environment. The key operator actions in response to this sequence are centered around the attempt to restore feedwater before irreversible core damage occurs. The operator must efficiently recognize the occurrence and cause of the loss of all feedwater, initiate attempts to restore feedwater, and, if successful, bring the plant to a safe shutdown condition. Figure A.2-2 presents, in event tree format, these operator actions.

The operator's first responsibility is to recognize that the initiating loss of offsite power transient has occurred. He would then verify that a reactor trip has occurred and ascertain whether the emergency diesel generators have started and emergency systems are being loaded. He should at this time become cognizant of the failure of emergency AC power. The operator must then determine that the steam turbine driven auxiliary feedwater pump has also failed. Efficient diagnosis of the situation will allow maximum time for repair actions.

The next step for the operator would be to initiate attempts to restore either offsite or onsite power or to repair the steam driven AFWP. Since many of the potential causes for a loss of offsite power are beyond the ability of the operator to remedy, his actions are assumed to be concentrated on restoring onsite power or the turbine driven feed pump. The operator's first attempt at restoring onsite power would be to try to manually start the diesel generators to circumvent any logic failures which may have prevented automatic startup. Should this attempt to manually start the diesels fail, plant personnel must diagnose the cause of failure of either onsite power or the steam driven feed pump and initiate repair actions.

As stated above, feedwater restoration can be successfully accomplished by any one of three repair modes: 1) restoration of offsite power, 2) restoration

of onsite power, or 3) repair of the AFW steam driven pump. However, the time available for operator action to restore feedwater and the required operator tasks subsequent to feedwater restoration are strongly dependent upon the particular repair mode.

If the operator were to restore the steam driven turbine feed pump, he would need to accomplish the necessary repairs prior to the time that the steam generators have boiled dry. Calculations[4] indicate that the time to steam generator dryout is approximately one hour for a plant of the type analyzed in the RSS. If the steam driven pump is restored, it would then provide sufficient feed flow to the steam generators to remove decay heat. The steam generator water level would recover and the primary pressure would decrease. The operator would want to verify that the pressurizer safety valves have reseated to prevent additional loss of coolant from the primary. In this paricular sequence it is assumed that these valves successfully reclose.

Calculations[3] of loss of feedwater transients have indicated that the core mixture level will not have dropped below the top of the fuel rods before one hour from initiation of the transient. Therefore, even without AC power to provide makeup flow from the Emergency Core Cooling or Chemical Volume Control Systems, there should be sufficient liquid inventory remaining in the reactor coolant system to establish natural circulation. The system will be in either a two phase or reflux boiling mode of natural circulation.

If onsite power were restored within the first hour before core uncovery occurred, the electrically driven auxiliary feedwater pumps would supply water from the condensate storage tank to the steam generators. As steam generator level recovered, the primary side pressure would begin to decrease. Again, the operator would verify closure of the safety valves.

The important difference between this particular repair mode and the previous mode (repair of AFW steam driven pump) is the availability of electrical power to plant safety systems. Therefore, as the primary

system pressure is decreasing, automatic actuation of safety injection will occur or the operator may choose to manually initiate it at some earlier time. In either case, the operator should verify the alignment of the charging pumps in the injection mode of operation. Since the core was not uncovered, safety injection will be used to refill the reactor coolant system. Once proper liquid level is reached, the operator can maintain proper system inventory either through use of the safety injection system or by switchover to automatic makeup (CVCS) and preparations for normal plant cooldown can begin.

If onsite power is recovered within one to three hours, the coolant inventory in the reactor coolant system will have dropped to a level where partial or complete core uncovery has occurred and the containment pressure level is approaching the failure point. Further reactor specific analytical studies are needed to determine at what time onsite power can be recovered so that severe core damage will be prevented and a core coolable geometry maintained. The incorporation of steam cooling into these analyses may extend this time for operator action beyond the existing three hour limit.

If the operator is successful in restoring core cooling, his next actions are directed toward bringing the plant to a safe, cold shutdown condition. Because of the unique role of the operator discussed in response to TMLB', the various steps required to accomplish this have not been considered in previous safety evaluations. For this reason, specific operator actions have not been investigated. However, since electric power is required for RHRS operation, restoration of either onsite or offsite power is required before the plant can be safely shutdown. Thus, if the immediate operator action was to restore the steam turbine driven auxiliary feed pump, subsequent actions should be directed toward providing electric power.* Once electric power has been restored the operator actions are similar to those discussed for the TML sequence in Section A.1.2.2.

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*For an extended period without electrical power (on the order of eight hours in the Surry plant), the inventory in the condensate storage may become depleted. Hence, it may be necessary to replenish this source to maintain delivery of coolant to the steam generators.

A.2.3 Operator Information Requirements

In this section, the information concerning the state of the plant systems and components which is required by the operator to efficiently take the actions described in the previous section is described and the measurable plant parameters which can provide this information are identified.

The first operator task is to recognize the occurrence of the initiating loss of offsite power initiator. The most direct indication of this event is provided by monitoring the power supply to the major electrical buses fed from offsite. Indirect indication can be provided by monitoring the behavior of the numerous systems or subsystems which depend upon offsite power (e.g., monitoring main feedwater system flow or steam generator level). The simultaneous observation of multiple system abnormal behavior should indicate to the operator that a common link between all the systems (i.e., offsite power) has been lost. These indirect indications provided by observing the anticipated effects of a loss of offsite power will provide an extremely effective backup to the direct measurement of current flow.

Given the initiating event, the operator can verify reactor trip by monitoring the control rod positions or measuring the neutron flux.

The failure of the diesel operators to start and/or take load can be indicated in much the same way that the loss of offsite power was indicated. The current supplied to the major electrical buses fed by the diesels can be measured and the system effects of such a power failure can be monitored (e.g., the inability of the electrically driven auxiliary feedwater pumps to start and the resultant lack of flow from these pumps).

With the knowledge that a loss of offsite power has occurred (with the resultant loss of main feedwater), and that onsite power is unavailable (thereby precluding operation of the electrically driven auxiliary feed pumps), the operator would ascertain the status of the steam driven auxiliary feed pump. This can be indicated by feedwater flow

rate, pump discharge pressure, steam generator water level, or primary system pressure and temperature.

The operator's next step would be to initiate repair actions. Indications of the status of a variety of components associated with the operation of the turbine driven pump and the diesel generators would be useful to the operator in diagnosing the cause of failure. The selection of the specific parameters should be based upon an identification of the most probable failure causes. Examples of such specific parameters are diesel fuel oil tank level, lube oil pressure, steam flow to feedwater turbine, etc. Indications of successful repair of onsite power or the steam driven pump can be provided by monitoring the same parameters used to determine the initial failure.

If the steam driven auxiliary feed pump is recovered, the operator would need to monitor system pressure, temperature, steam generator level, and core water level, to determine if the system has entered into a stable mode of natural circulation. As mentioned, he will need to verify that the safety valves have reclosed when system pressure is reduced. For this verification, valve position indicators or discharge line flow measurements are needed. If the plant is in a stable cooling mode and the safety valves have reseated properly, the rise in containment pressure will be halted.

If the operator restores onsite power, he must monitor primary system pressure and temperature, and core water level. Coolant radioactivity should also be measured to determine the extent of fuel failure, if any, and deduce the core coolability. The operator would verify operation of the containment spray system by the reduction in containment pressure and temperature and spray pump flow rate and discharge pressure.

Actuation of the safety injection system on demand by either the appropriate signal or manual initiation would be indicated by monitoring system pressure and temperature. In addition, indications of valve position, pump power, flow rate and discharge pressure should be available.

A.2.4 Summary and Conclusions

In the preceding sections, the information that the operator needs to respond to the events of the TMLB' accident sequence have been identified and the measurable plant parameters which can provide this information have been delineated. Presented in Table A.2-1 is a summary of these results. Table A.2-1 includes, for each of the key plant states illustrated in Figure A.2-2:

- a brief description of the plant state
- the information (in terms of measurable plant parameters) required by the operator to unambiguously determine this plant state.
- the appropriate operator action
- the information necessary to perform this action.

The conclusions of this section (represented by the required plant parameters listed in Table A.2-1) were based on a number of input assumptions concerning the plant response to the failure events postulated in the TMLB' sequence. As noted previously, due to the unavailability at this time of adequate detailed analyses of the response of the Surry plant during the TMLB' sequence, many of these input assumptions have a relatively high uncertainty associated with them. Presented below are a few major areas where further analytical work would be beneficial in confirming or reducing the uncertainty of these assumptions:

- A better definition of plant behavior following a loss of all feedwater is required. From this a better definition of required repair times can be obtained and operator repair options can be identified. In addition, this will provide a better description of the state of the plant upon restoration of feedwater and

thereby determine the steps necessary to establish long term cooling at that time

- The natural circulation heat removal capability of the primary system given feedwater flow from the steam driven pump needs to be determined.
- What affect would the loss of all AC power have on the instrumentation system? Will DC power supplies be adequate for duration of sequence?
- The appropriate operator procedures to reestablish forced circulation given restoration of power need to be defined.

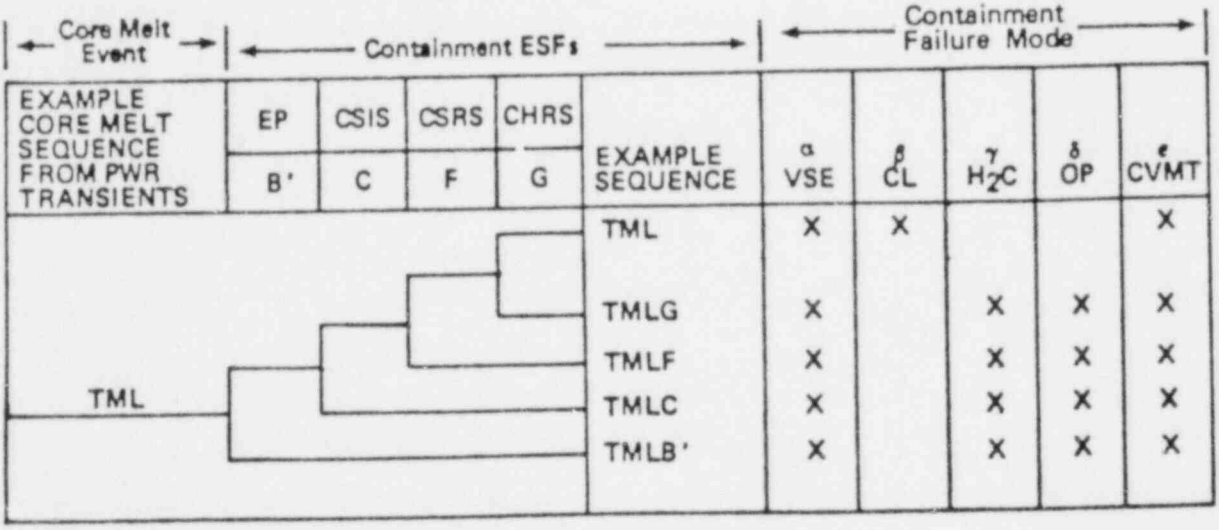

FIGURE A.2-1

SIMPLIFIED EVENT TREE FOR TRANSIENT SEQUENCES INVOLVING A CORE MELT

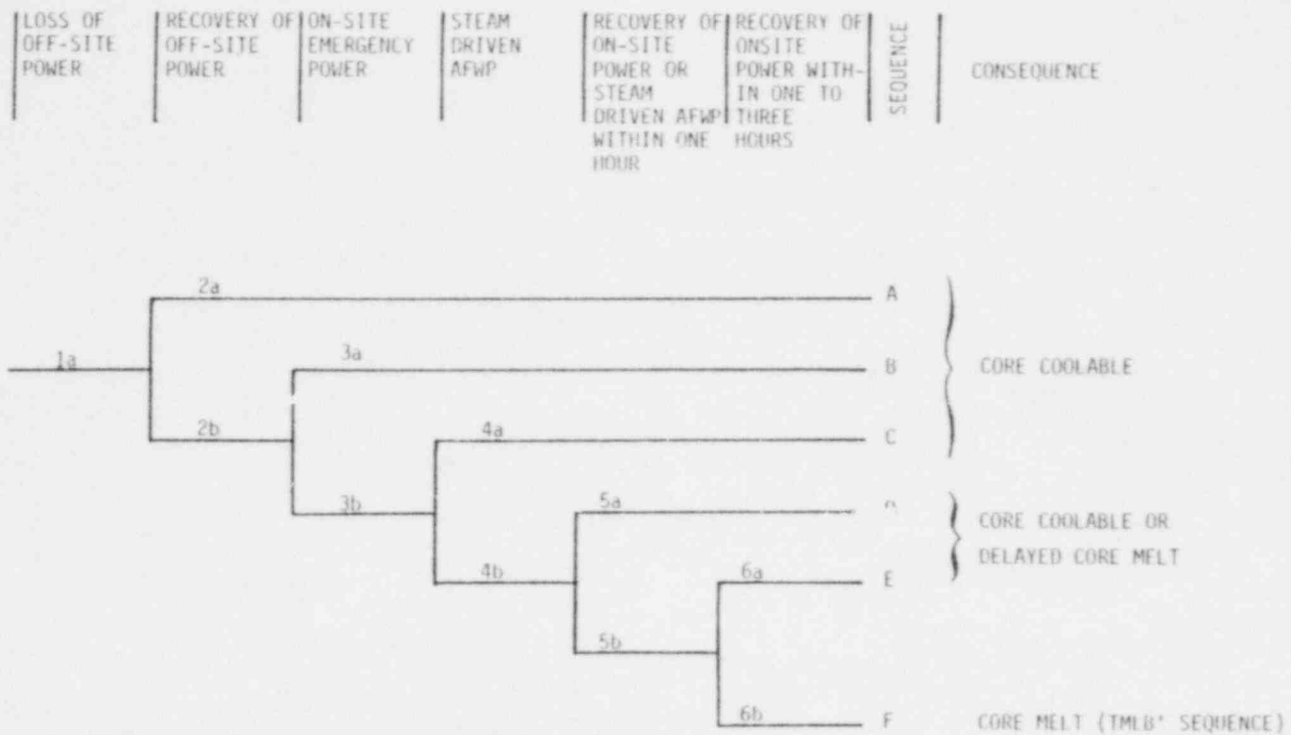

Figure A.2-2. Operator Action Event Tree TMLB' Sequence

Table A.2-1

SUMMARY OF KEY OPERATOR ACTIONS AND INFORMATION REQUIREMENTS FOR TMLB'-S SEQUENCE

Plant State	Description of Plant State	Information Required for Plant State Identification	Operator Action Following Plant State Identification	Information Require to Take Appropriate Action
1a, 2b	Loss of offsite power; non-recovery of offsite power within one hour	o power supply to electrical buses o main feedwater flow and discharge pressure o steam generator level o RCS pressure and temperature o pressurizer level	Verify reactor shutdown Determine emergency diesel generator availability	o RCS pressure and temperature o Neutron flux o Control rod position o Emergency power to electrical buses
2a	Offsite power restored	o same as for state 1a identification	Ensure system at stable condition, reestablish liquid inventory, prepare to start reactor shutdown if necessary	o RCS pressure and temperature o Core water level o Pressurizer water level o Steam generator water level o Safety valve positions
3a	Onsite emergency power established	o RCS pressure and temperature o emergency power supply to electrical buses o steam generator level o AFWS flow o pressurizer level	Ensure loading of ESFs on emergency power and proper actuation of safeguards equipment at the appropriate signal, reestablish liquid levels; prepare for plant shutdown	o AFWP flow and discharge pressure o Charging pump flow and discharge pressure o Containment spray pump flow and discharge pressure o RCS pressure and temperature o Pressurizer, core and steam generator level
3b	Failure of onsite power	o same as state 3a identification	Determine availability of steam driven auxiliary feed pump	o Steam supply to turbine pump o Pump flow and discharge pressure
4a	Success of steam driven AFWP	o RCS pressure and temperature o steam generator level o AFWP flow and discharge pressure o core water level o pressurizer water level	Ensure system at stable condition; prepare for plant shutdown	o Same as state 2a
4b	Failure of steam driven AFWP	o Same as for state 4 identification	Restore and repair steam driven AFWP and /or onsite power	o Parameters identifying possible causes of failure

Table A.2-1
(continued)

Plant State	Description of Plant State	Information Required for Plant State Identification	Operator Action Following Plant State Identification	Information Required to Take Appropriate Action
5a	Recovery of onsite power or steam driven AFWP within one hour	o same as for state 4a identification	Reestablish liquid inventory in onsite power is recovered, verify stable plant conditions if AFWP is restored	o Same as for state 2a o Coolant activity level o Safety valve position o Discharge line flow o AFWP flow o Safety injection availability o Containment spray availability o Containment temperature and pressure
5b	Non-recovery of steam driven AFWP or onsite power within one hour	o same as for state 4a identification	Continue attempts to restore onsite emergency power	o Potential causes of failure of onsite power system
6a	Onsite power restored within one hour to three hours	o sames as for state 3a identification	Ensure core is in coolable geometry	o Same as for state 5a
6b	Onsite power not restored within three hours	o same as for state 3a identification o coolant activity level o containment activity level		

A.3 S_2C-δ SEQUENCE

A.3.1 S_2C-δ Sequence Description

The S_2C-δ sequence is initiated by a small break in the Reactor Coolant System (RCS) followed by a failure of the Containment Spray Injection System (CSIS). In the Reactor Safety Study[4] analysis of the Surry PWR, the wide spectrum of postulated primary system breaches was categorized by the minimum Emergency Core Cooling System (ECCS) response necessary to prevent a core meltdown. For the S_2 event, a flow rate equivalent to the delivery from one of the High Pressure Injection System (HPIS) pumps satisfies this requirement. The corresponding break size is an effective diameter in the range of approximately 0.5 to 2 inches. The lower limit corresponds to a leak rate just slightly greater than that which can be replenished by the Chemical and Volume Control System (CVCS). The upper bound was selected based on the containment sump fill rate that will just fail to meet the minimum water supply requirements of the Containment Spray Recirculation System (CSRS) should a failure of the CSIS be postulated. This subject will be discussed in more detail in the following sequence description.

The event tree for the S_2 initiator is presented in Figure A.3-1 with the path of the S_2C sequence highlighted. With the loss of primary system integrity, reactor coolant system depressurizes and the coolant temperatures begin to rise. The water level in the pressurizer decreases as the make-up flow from the CVCS cannot compensate for the break flow. The leaking coolant causes a gradual increase in the containment temperature and pressure. The radiation levels in containment will also increase, the rate depending upon the system cleanliness. Reactor trip occurs when the low pressure or overtemperature ΔT trip settings are encountered.

It is assumed that electric power supply is adequate to meet the needs of all Engineered Safety Features (ESF). As part of the normal sequence of events, following reactor trip, the Main Feedwater System is isolated and the Auxiliary Feedwater System (AFWS) is activated for decay heat removal and primary system cooldown. In this mode of operation, water from the Condensate Storage Tank is transported by the AFWS pumps to the secondary side of the steam generators. The energy is then removed by venting steam through the power operated relief valves. Adequate heat removal is provided by venting through only two of three power operated relief valves. As a backup, venting can also be achieved through the mechanical safety valves.

As the primary coolant is discharged into containment, a slow rise in containment temperature and pressure occurs. As seen in Figure A.3-5, after approximately thirty minutes, the containment pressure will reach the 30 psig level which should actuate the CSIS. However, in this sequence, the CSIS is assumed to have failed.* As a result, heat removal from the containment atmosphere does not occur and the pressure continues to increase as coolant is discharged.

The HPIS of the ECCS is activated in response to the decreasing RCS pressure. The charging pumps are automatically realigned to take suction from the RWST and transport coolant through the boron injection tank and into the RCS cold legs. For the S_2 initiator, the flow from only one of the three charging pumps is required to maintain coolant inventory and provide adequate core cooling.

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*Failure of the CSIS is defined as failure to deliver water from the RWST to containment at a rate equivalent to full flow from one of the two containment spray pumps.

As depicted on the event tree (Figure A.3-1), the remaining ESFs required for successful accommodation of the small break (S_2) have little effect on the consequences of the accident. The reasons are summarized briefly in the following discussion.

The CSRS is designed to provide long-term heat removal capability within containment and to lower the atmospheric radionuclide concentration. This system recirculates water from the containment sump through the heat exchangers in the CHRS to the spray nozzles which disperse the water throughout the containment atmosphere. However, the CSRS requires a minimum water level in the containment sump or the pumps will cavitate and are assumed to fail. Because of the large quantity of water supplied by the CSIS pumps (3200 gpm each), the sump fills rapidly to a level where CSRS operation can be safely initiated when required. However, since this sequence assumes failure of the CSIS, this quantity of water is not available in the sump. The leakage from the break alone is insufficient to meet the fluid inventory demand of the CSRS. Hence, the system is assumed to be unavailable for heat removal or will fail if the CSRS pumps are started with insufficient water level. Since the CSRS does not operate, it makes no difference whether or not the CHRS (which supplies service water to cool the sump water for the CSRS) functions as designed, as sump water is not delivered to the CHRS heat exchangers.

After a period, the RWST supply will be consumed and the ECCS operation must be realigned to the recirculation mode for long-term cooling.* At this

*Because of the failure of the CSIS, the RWST depletion rate is greatly reduced. Consequently, the injection mode of ECCS operation can be extended for a much longer period than if the CSIS had functioned. This length of time depends on the break flow but is of the order of several hours. However, eventually recirculation capability will be required.

time the level in the containment sump may have reached a level which is safe for recirculation operation. However, due to a lack of heat removal in containment, the sump water temperature is very high. Furthermore, the rising pressures in containment will eventually lead to a loss of containment integrity and subsequent depressurization. Based on these considerations, it was assumed in the Reactor Safety Study that suction from the sump would produce cavitation in the low pressure pumps, thus eliminating ECCS recirculation operation. This failure would eventually lead to a meltdown of the reactor core.

The final ESF noted on the event tree - sodium hydroxide addition (SHA) - fails because the CSIS, which delivers the sodium hydroxide to the containment atmosphere, did not function.

As noted previously, the lack of containment heat removal leads to a continuing increase in containment pressure. Eventually the boundary will fail releasing radioactivity to the environment and depressurizing containment. This failure mode is designated as δ in the Reactor Safety Study.

The previous scenario describes the S_2C-δ sequence as considered in the Reactor Safety Study. For conservatism, the Reactor Safety Study assumed no effective operator action to respond to the system failures. The following sections examine what actions the operator might take to successfully terminate this accident prior to core meltdown or containment failure, or reduce the consequences of the event.

A.3.2 Operator Actions

The preceding section described the S_2C-6 sequence and the various assumptions utilized in the WASH-1400 analysis of this accident for the Surry reactor. Given that the accident occurs, the next concern is to determine what actions the operator might take to successfully terminate or mitigate the consequences of the transient. The various options which should be considered and the consequences of their implementation are discussed in this section. Once the potential operator actions and their effects on the transient have been assessed, the information required by the operator can be determined.

Figure A.3-2 illustrates the operator actions in response to the S_2C initiator. This diagram was developed by considering the S_2C sequence pathway in Figure A.3-1, and modifying it to reflect the potential operator actions in response to the event. The initial part of the original event tree prior to CSIS failure has been condensed in Figure A.3-2 for simplicity. The functions and systems which are operable and assumed to perform as designed are combined into a single event. These are electric power, reactor trip, emergency coolant injection, and the auxiliary feedwater system for heat removal. Subsequent to the failure of the CSIS, the remaining event tree headings list the potential operator actions in response to the S_2C failure sequence. The different branches of the event tree have been assigned alphanumeric identifiers for referencing the various plant states in future discussions. The following sections examine the various operator actions and their consequences associated with the relevant sections of this event tree.

A.3.2.1 Response to Initiator (S_2)

The initial operator action is to determine what event has occurred so that he will know what options are available to him. The primary system behavior following a small break is characterized by a decrease in the reactor coolant system pressure (as seen in Figure A.3-3) and pressurizer water level* with a decrease in coolant temperature (Figure A.3-4). In conjunction, containment pressure, (Figure A.3-5), temperature (Figure A.3-6) and humidity, begin to rise as the water/steam mixture is expelled into the containment atmosphere. The radiation level in containment will also become elevated. The magnitude and rate of these variations depend on many factors which include the break size and location, the reactor control and volume control systems response, the normal containment heat removal system efficiency and the contamination of the primary coolant.

After verifying that a breach in the primary coolant system has occurred, the operator's next action should be to identify the ESFs required to accommodate this event and mitigate its consequences. The specific systems are electrical power, the reactor protection system, the auxiliary feedwater system, and the high pressure injection system of the ECCS. The functions of these systems were discussed in Section A.3.1. The status of each of these systems should be checked to ensure that it is ready for operation upon demand. Any systems which were temporarily bypassed for periodic testing or deactivated for maintenance should be returned to their standby configuration if possible. Depending on the nature of the break and the response of these systems, the operator may have to manually control the HPIS to avoid repressurization of the primary system.

*For vapor space breaks (e.g., stuck-open pressurizer relief valve), the pressurizer level would increase, while the RCS pressure decreases.

As a result of post-TMI analysis, it has been determined that automatic trip of the reactor coolant pumps early in a small loss of coolant accident is desirable. The appropriate signals or parameters to accomplish this function have not been defined. The results of this study may aid in this decision. When this directive becomes part of the operating instructions, the operator must verify RCP trip on the appropriate signals.

A.3.2.2 Response to CSIS Failure (C)

Subsequent to the identification of the small break and verification of successful operation of the key ESF's, the plant is at state 2a on Figure A.3-2 Once the containment pressure reaches 30 psig, CSIS should be activated. Since this sequence assumes failure of the system the containment pressure will continue to increase beyond this level. It is critical that the operator recognizes that this system has not functioned, so he will know what are the potential consequences, and thus can take the appropriate action. In effect the operator must know the plant's state on the S_2 event tree. With this information, it is possible to determine how the sequence can progress from that state, and the options that are available to him.

A.3.2.2.1 Containment Heat Removal

Subsequent to failure of the CSIS, the operator has two options for future action. First, an attempt can be made to determine the cause(s) of the CSIS failure, and if possible restore the system to operation. Alternatively, the operator can try to find a different means of accomplishing the functions of the CSIS.

Restoring CSIS capability requires that (1) the fault can be identified, and (2) that corrective action can be taken. Upon verification that containment spray was not working, the first operator response should be to manually initiate the CSIS. This action could restore the CSIS if the failure were due to a failure of the Consequence Limiting Control System (CLCS) to automatically initiate containment spray. Identification of other specific faults which could cause CSIS failure is beyond the scope of this initial study. The approach which could be followed is summarized briefly. Since the S_2C-δ sequence is a slowly developing one, there may be time for some types of corrective action. However, if for example, the fix requires personnel to enter containment where the hot, radioactive coolant is leaking, then such an action is unacceptable. To determine if any of the potential CSIS failure modes are amenable to short-term corrective action, the fault tree diagrams can be reviewed to identify the various failure modes. After evaluating the causes of the system failure, the actions necessary to repair the system can be determined. Those failures which can be repaired within the time scale dictated by the accident progression and are feasible to effect in the accident environment can then be identified. Procedures to perform this corrective action can then be developed. Instrumentation requirements relative to CSIS failure identification and repair are addressed in Section A.3.3.

If the operator is successful in restoring the CSIS, the next action is to ensure that the remaining ESF's identified in Figure A.3-1 perform as designed. Operator actions to correct the CSIS malfunction, or any adverse environmental conditions which resulted from a delay in CSIS operation, may impact the operation of other systems. Hence, operation of these systems must be carefully

monitored to assure a safe cooldown of the plant (Sequence A of Figure A.3-2). The two primary objectives are maintaining containment integrity and in-place, coolable core geometry. Both of these objectives require the use of the CSRS/CHRS. Spray recirculation and cooling is required to keep the pressure in containment at an acceptable level and also to reduce sump water temperature. This latter condition is essential for long-term ECCS recirculation (Section A.3.2.2.2). Hence, even if the CSIS is repaired, CSRS must still perform or core melt will eventually occur (Sequences D and E). This may require additional operator action. The CSRS is automatically initiated on receipt of the containment hi-hi pressure signal with a time delay of two minutes on the internal spray pumps and five minutes on the external spray pumps. Once the internal pumps are started they cannot be stopped until the containment pressure returns to subatmospheric conditions. If the sump water level is inadequate to properly operate these pumps, they could be lost. Upon receipt of the hi-hi containment pressure signal, the operator should check the sump water level to determine if the CSRS can be operated properly. If sump water level is inadequate, then the operator should take steps to manually override the CSRS until adequate sump water level is ensured.

If short-term repair of the CSIS cannot be implemented (state 4b of Figure A.3-2) operator action must be directed toward finding an alternate means of cooling containment and providing the long-term heat removal capability for the reactor core. If no method of cooling containment can be effected, then

the rising pressure will ultimately lead to failure. Furthermore, without containment cooling the water in the containment sump will approach saturation and could result in pump cavitation when used as a source during the recirculation mode of ECCS operation. A loss of ECCS recirculation capability will eventually lead to core melting (Sequences I and J of Figure A.3-2). Figures A.3-3 through A.3-7 present some of the key parameters from the analysis of the S_2C-δ sequence.

One possible approach to providing containment heat removal is to utilize the CSRS and CHRS. Operator action would involve verifying alignment of the containment spray system for recirculation, ensuring cooling water supply to the CHRS heat exchangers and monitoring the water level in the sump. When a safe level is reached, CSRS pumps can be activated. The operator may have to override the automatic initiation of the CSRS pumps if the safe level is not reached when the containment pressure reaches the hi-hi set point. In addition to an adequate water level in the sump, the operator must also ensure that the thermodynamic state of this source is such that pump cavitation will not occur.

As noted in the S_2C-δ sequence description, one of the characteristics that defines the S_2 event in the Reactor Safety Study is that the water level in the sump is insufficient for CSRS operation. However, this may not be true for all small breaks. If the break location is such that the leaking coolant spills into the reactor cavity, then this volume must fill and overflow

before water enters the sump. Then it is poss ble that the containment pressure will be approaching the failure limit by the time an adequate inventory is available in the sump. However, calculations of the sump inventory for the S_2C sequence* have shown that an adequate water level is likely to be achieved while there is still substantial margin to containment failure (Figure A.3-7). These transient analyses predicted that the mass of liquid water in containment would completely fill the reactor cavity volume ($\sim$ 11,000 ft^3 in the Surry plant) while the containment pressure was less than 45 psia. Since containment failure does not occur until $\sim$ 100 psia, there is ample water inventory for suction from the containment sump.

Although the water leaking from the primary system may fill the sump to a safe level for pump suction, the fill rate may be insufficient to operate the CSRS at full capacity (4 pumps with a design flow of 3500 gpm in Surry). However, the fill rate may supply enough water to operate a single pump at full or reduced flow. In this case, manual control of the CSRS by the operator to regulate flow with respect to replenishment of the water supply may be required. As further leakage from the primary system occurs, containment spray flow can be increased by activating additional pumps. Even at a reduced flow rate, the containment pressure buildup would be lessened. This effectively buys time until the sump water level reaches a level where the minimum flow necessary to achieve a pressure reduction in containment can be supplied.

In the event that the water level, or its thermodynamic state, preclude CSRS operation, another source of water must be found to supplement the

*These calculations were performed by Battelle Columbus Laboratories using the MARCH computer code package.

break flow if containment spray is to be used to reduce pressure. Preliminary evaluations indicate that there are no practical means for deliberately introducing sizeable quantities of water into the containment sump.

If containment heat removal is unavailable (state 5d) because of an inadequate water supply in the sump, a core meltdown will eventually occur. Further operator actions from this point are discussed in the following section.

It is important to note that the previous discussion and assumptions are based on the Surry reactor design, as considered by the Reactor Safety Study. Other PWR designs include alternate containment heat removal systems which could be effectively utilized to limit containment temperatures and perhaps prevent a core meltdown accident. For example, some more recent designs utilize ice condensers as a passive heat removal system. An evaluation for the Sequoyah plant has shown that this feature prevents core meltdown for the S_2C sequence. Such additional features introduce different possibilities for operator response and would require a different operator action event tree.

A.3.2.2.2 Long-Term Cooling of the Core

If the CSRS/CHRS is operable and is effective in cooling containment (state 5c of Figure A.3-2), the remaining major concern is to assure long-term cooling of the core. This requires operation of the ECCS in the recirculation mode. The containment sump serves as the water source during this phase of operation. Hence, the same considerations noted in the previous section

regarding water level and its thermodynamic state apply. However, recirculation through the core is not initiated until the RWST inventory has been depleted. By this time the water level in the sump is adequate to supply both CSRS and ECR system needs. Furthermore operation of the CHRS has kept sump water conditions well below saturation. Hence, pump cavitation is highly unlikely. Operator action involves successfully changing the ECCS configuration from injection to recirculation operation. This places the plant in state 6c and long-term cooling can then be maintained (Sequence F).

Similar considerations apply to the case where the operator was successful in restoring the CSIS to operation (state 5a). In this instance, it is likely that any effects on the operation of the ESF's necessary to maintain the plant in a stable, coolable condition will be less severe than if there were no containment spray.

It is probable that failure to provide effective long-term core cooling, given that the plant is in states 5a or 5c, would require additional failures of equipment that are unrelated to the failure of the CSIS. Such additional failures, when compounded with the events which produced the S_2C sequence initially, are of extremely low probability and can be neglected for purposes of this assessment. Hence, the operator response to states 6b and 6d is not addressed. However, the operator action in such a postulated occurrence would be very similar to the response required should effective containment heat removal be absent (states 5b or 5d).

Assuming that the containment can not be effectively cooled, it will eventually fail from the increasing pressure. In this case, operator action

should be directed toward delaying a core meltdown and minimizing releases to the environment subsequent to containment breach. By delaying core melt, the consequences of containment failure are likely to be lessened. This action also "buys time" during which containment cooling may be restored, thus minimizing the accident consequences and preventing a large meltdown. The effect of successful delaying action is noted in the final column of the operator action event tree (Figure A.3-2).

Core melting can be delayed by making efficient use of the ECCS. For small break accidents (S_2) the injection phase of ECCS operation can be extended significantly. The RWST inventory depletion rate is greatly reduced as a result of the failure of the CSIS. Hence, the volume of water available for HPIS injection is much larger. Utilizing the HPIS to maintain primary system inventory, and the AFW/SSR system for heat removal, coolable geometry can be maintained for a considerable period. Eventually a transfer to the recirculation mode will be required as the RWST is emptied. By this time, the water level in the sump should be adequate to operate the low head injection pumps. Depending on the nature of the transient, it may be possible to initiate ECCS recirculation. Because of the failure of containment heat removal capability, the water temperature may be elevated to the point where pump cavitation would occur. If this happens, make-up water to compensate for the leakage through the break would be unavailable. A gradual melting of the core would ensue.

Due to an absence of sump water cooling, it is expected that sump conditions will eventually preclude operation in the recirculation mode. Therefore, it may be advantageous to extend the injection phase as long as possible. This could be accomplished by utilizing only the minimum HPIS flow to maintain inventory, rather than operating at full capacity. For the S_2 event, only one of three charging pumps is required to maintain coolability. Hence, utilizing only one pump could significantly increase the time prior to the onset of core melt.* In order to take this action, the operator would require knowledge of conditions in the core to ensure maintenance of a coolable geometry. This would allow the operator to regulate the make-up flow being provided by HPIS. The injection phase could be extended further by replenishing the RWST. Eventually, however, rising water levels in containment may result in other failures that would lead to core melt. The impact of this action has not been assessed in this study.

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*As decay heat load decreases, less than full flow from one pump would be required.

A.3.3 Operator Information Requirements

The preceding section addressed the operator action in response to the postulated S_2C accident. The principal actions are summarized below:

1) Identify occurrence of small break
2) Determine ESF's required and verify their status and successful operation
3) Identify CSIS failure
4) Repair or restore CSIS if possible
5) Provide containment heat removal using CSRS/CHRS
6) Ensure long-term cooling for core
7) If long-term core cooling cannot be provided, delay core meltdown

To take these actions and make the associated decisions, the operator must have a clear understanding of the plant state at all times, and know what options are available. This section addresses the information which will enable the operator to determine the plant condition during a postulated S_2C sequence and thus implement the above actions as necessary. A summary of the operator information requirements and appropriate actions for the relevant plant states in Figure A.3-2 is presented in Table A.3-1.

The first operator action is to determine that a rupture in the primary coolant boundary has occurred. The parameters which unambiguously indicate a small break are a decrease in the reactor coolant system pressure

in conjunction with a rise in containment pressure. The magnitude and rate of these variations depends on many factors which include the break size and location, the reactor control and volume control systems response, and the efficiency of the normal containment heat removal system. Additional confirmation of a primary system leak would be an elevation in radiation levels in containment. In addition to the previously noted variables, the increase in radioactivity depends on the contamination of the primary coolant. Other parameters, which could be utilized as diverse backup measurements, are the reactor coolant temperature, containment temperature and humidity, and sump water level. All of these parameters increase slowly subsequent to a small RCS break. Changes in the pressurizer water level would also accompany a small break. For most break locations, the level would decrease. However, if the coolant loss was through a "stuck-open" pressurizer relief of safety valve, the water level could increase. Additional indications of this event would be valve position, discharge line temperature, or pressurizer relief tank level, pressure, and temperature.

After verifying the presence of a primary coolant system breach, the next operator action is to identify the ESF's required to accomodate this event and mitigate its consequences. The specific systems were mentioned in Section A.3.1 and are illustrated on the S_2 event tree (Figure A.3-1). The status of each of these systems should be checked to ensure their readiness for operation. Once their actuation is required, verification of correct system response should be performed.

The availability of electrical power (EP) can be readily verified, as its absence would be indicated by numerous instruments and annunciators in the control room. Tripping of any circuit breakers feeding critical equipment will be annunciated. In the unlikely event of a total loss of offsite AC power, the operator must ensure that the diesel generators are actuated as designed. Similarly, a reactor trip will be easily recognized. If, for some reason, an automatic trip has not been initiated, the operator can manually scram the reactor. The operator can ensure an adequate margin for safe shutdown by monitoring the neutron flux.

Activation of the HPIS automatically isolates the main feedwater system and activates the auxiliary feedwater system. The effectiveness of high pressure coolant injection can be verified by monitoring the primary system temperature and pressure. The successful operation of the individual HPIS trains can be verified by measuring the respective flow rates or pump discharge pressures. Additionally the pressurizer and/or reactor water levels should respond to the addition of water from the ECCS. Similarly AFW flow or AFWP discharge pressure can be monitored to verify flow to the steam generators. The steam generator water level will indicate if the water supplied by the AFWP's is adequate.

If the containment spray were actuated, the immediate response would be a reduction in containment pressure and temperature. Since the CSIS is assumed

to fail in this sequence, these parameters will continue to increase. Additional variables which should indicate CSIS function under most conditions are the flow in the injection lines and the CSIS pump discharge pressure. These measurements could be utilized as diverse backups and to provide the operator with additional information which might assist in identifying the cause of the failure.

As discussed in Section A.3.2.2.1, the operator has two options for providing containment heat removal capability: the CSIS can be repaired, or alternate systems can be employed. If the CSIS is to be restored, the cause for its failure must be identified. Additionally, the corrective action must be feasible to implement under the accident conditions and within a limited time period. The specific failure modes which satisfy these criteria (if any) can be identified using fault tree analysis. Knowing the failures, it is then possible to specify instrumentation to detect these faults. However, before additional plant monitoring capability can be recommended, some consideration must be given to the probability of these CSIS failure modes. Their contribution to the overall CSIS should be evaluated. If these events are not significant contributors, then the addition of instrumentation to identify these faults is probably not warranted. If any are discovered to be important, then the decision to add the capability to detect this fault must consider if it is possible and practical to instrument the specific components such that their

failure can be reliably detected. Furthermore, it must be ensured that the additional instrumentation will be unambiguous and not likely to confuse the operator. The identification of the specific CSIS failure modes, and the instrumentation which might be utilized to detect these faults, is beyond the scope of this preliminary study.

If restoration of the CSIS is not feasible, the operator can attempt to utilize the CSRS and CHRS to cool containment. As discussed in Section A.3.2.2.1, the critical factor which determines if this option is available is the availability and thermodynamic state of the water in the containment sump. Measuring the water level in the sump will indicate if there is sufficient inventory for pump suction. This information is particularly important because the CSRS pumps are actuated automatically in a short time after the containment hi-hi pressure set point is reached (See Section A.3.2.2.1). If the water level in the sump is insufficient for pump suction at this time, the operator must manually override the automatic CSRS actuation. The variation in water level during CSRS operation will also provide the operator with the information required to regulate the flow in the system (initially the sump inventory may not be adequate for CSRS operation at full capacity).

Sufficient sump water subcooling must also be ensured. Otherwise the pumps could fail from cavitation. The margin for safe pump suction can be determined by measuring the temperature of the water in the sump and the

containment pressure, and comparing the resultant state to saturation conditions. Measurement of the containment atmosphere pressure will also provide verification of successful heat removal. If the CSRS must be operated at a reduced capacity initially, the containment pressure may not immediately decrease. However, its rate of increase would be lessened,and as CSRS flow is increased the pressure would eventually begin to fall.

Subsequent to providing containment heat removal, the operator action is directed toward preventing a core meltdown. Section A.3.2.2.2 discussed the provisions for long-term cooling of the reactor core, and the associated operator actions. The RWST water level indicator automatically alerts the operator when change-over to recirculation operation is required. Upon transferring to the recirculation mode, the operator must ensure an adequate sump water level for pump suction. However, since the RWST has been depleted, this criterion should be satisfied. As with the previous discussion, the sump water must be adequately subcooled. Assuming successful operation of the CHRS (state 5c), this condition will also be satisfied. Thus, assuming successful operation of the ECCS components, no additional measurements other than those required for CSRS actuation and regular ECCS control are needed.

If containment heat removal is unavailable (states 5b and 5d), a core meltdown will eventually occur. Operator action under these circumstances

wou'd be to delay the meltdown as long as possible by extending the injection phase of ECCS operation. The operator actions are to provide the minimum required make-up flow to keep the core covered and avoid DNB. In order to successfully regulate ECC flow, the RCS pressure, outlet temperature, and reactor vessel water level are needed. The pressure and temperature measurements should provide an indication of the margin to dryout in the core. Water level indication would warn the operator of potential core uncovering, even if the temperature and pressure indicated conditions in the core were acceptable. Additional considerations with respect to delaying core melting are addressed in the evaluation of the V sequence.

A.3.4 Summary and Conclusions

The preceding discussion considered the S_2C sequence and identified potential operator actions to interrupt this sequence or reduce its consequences, for the plant design evaluated by the Reactor Safety Study. The reactor and plant parameters which are necessary and sufficient to define the plant state during the accident have been identified with intent of providing the operator with clear information on which to take the proposed corrective actions. The results of this evaluation are summarized in Table A.3-1.

The information presented in the summary table is based on a number of assumptions concerning the plant performance and response to the postulated sequence. Many of the plant conditions and proposed operator actions have not been analyzed in the past. Hence, there is some uncertainty and generality in these evaluations. The following list identifies areas where further information would be beneficial in either confirming the key assumptions used in this study or reducing the level of uncertainty.

- The assumption of insufficient water level in the containment sump for CSRS operation given that the CSIS fails should be carefully examined. It appears that this assumption may not apply, or may be unduly conservative in many small break accidents. The relative variation in containment pressure, sump fill rate, and sump water temperature for different break sizes and locations merits further analysis.

- The possible use and effectiveness of alternate containment heat removal systems should be investigated. This study was performed assuming containment spray as the only effective system for heat rejection (WASH-1400 assumption). However, the normal containment heating and ventilating system may be capable of some heat removal. Unless containment isolation considerations preclude its use, it may provide sufficient cooling to prevent or delay containment failure and core melt.

- The effect of extending ECCS injection to delay core melt (assuming inability to remove heat fron containment) should be evaluated in more depth. In particular, does this action significantly delay the onset of melting? If so, the specific operator actions need to be better defined. One action which should be considered in this regard is replenishing the RWST to further extend the injection phase.

- Does a loss of containment heat removal inevitably lead to core melt? Are there some small break accidents where core melt can be prevented - even though containment integrity may be violated. Are there any mechanisms for cooling containment sump water if water cannot be delivered to the containment spray headers?

- More detailed information on the plant states for this accident is necessary to establish the necessary ranges for instrumentation for this sequence. This may require some sensitivity studies to examine the effects of different assumptions regarding plant systems response on key plant variables. The range required for each measured parameter would then be determined by integrating this type of information for all sequences in which measurement of a given variable is necessary.

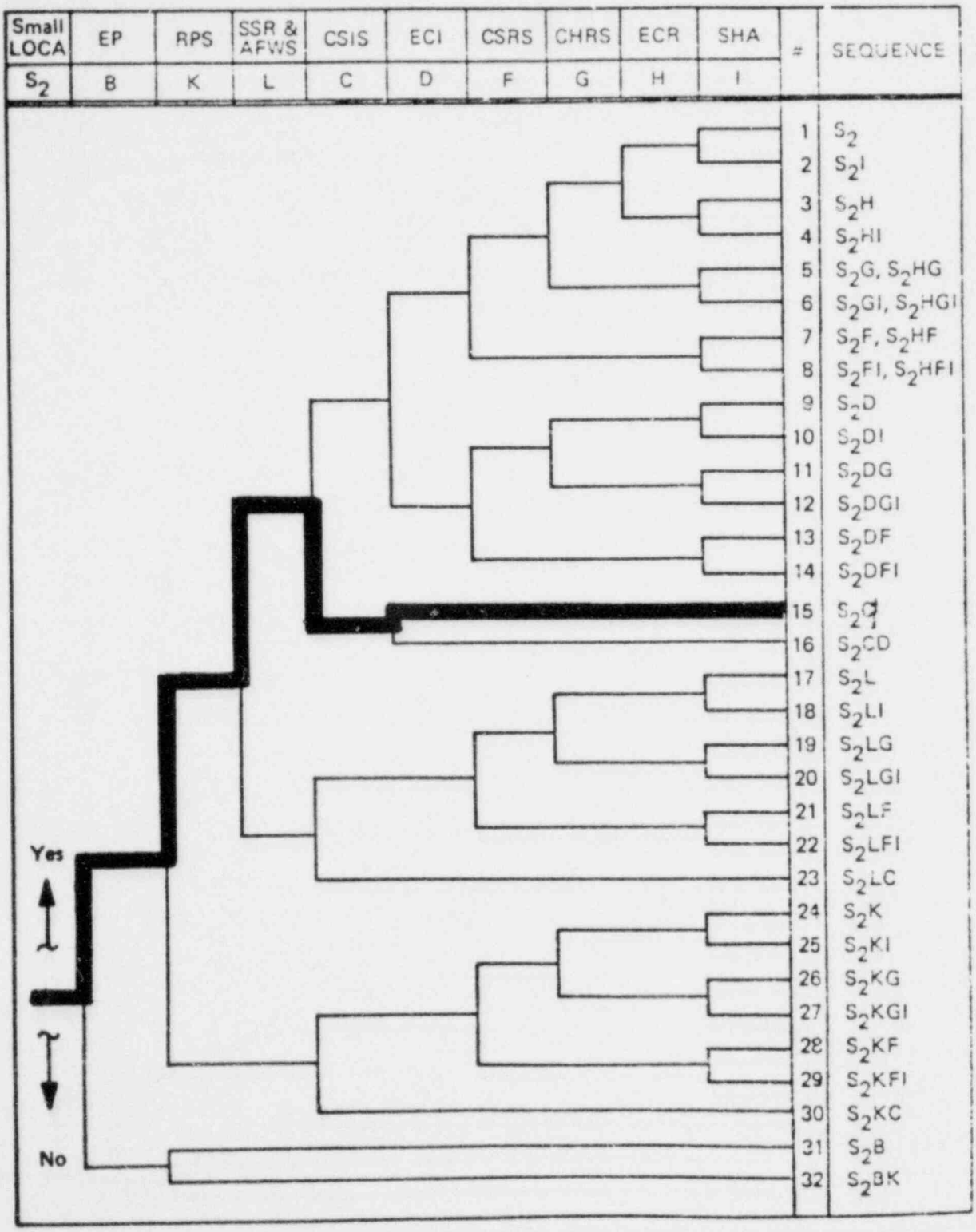

FIGURE A.3-1

PWR Small LOCA (S2, 1/2-2 inch diameter) in RCS Event Tree (Reference: Reactor Safety Study, WASH-1400, October 1975).

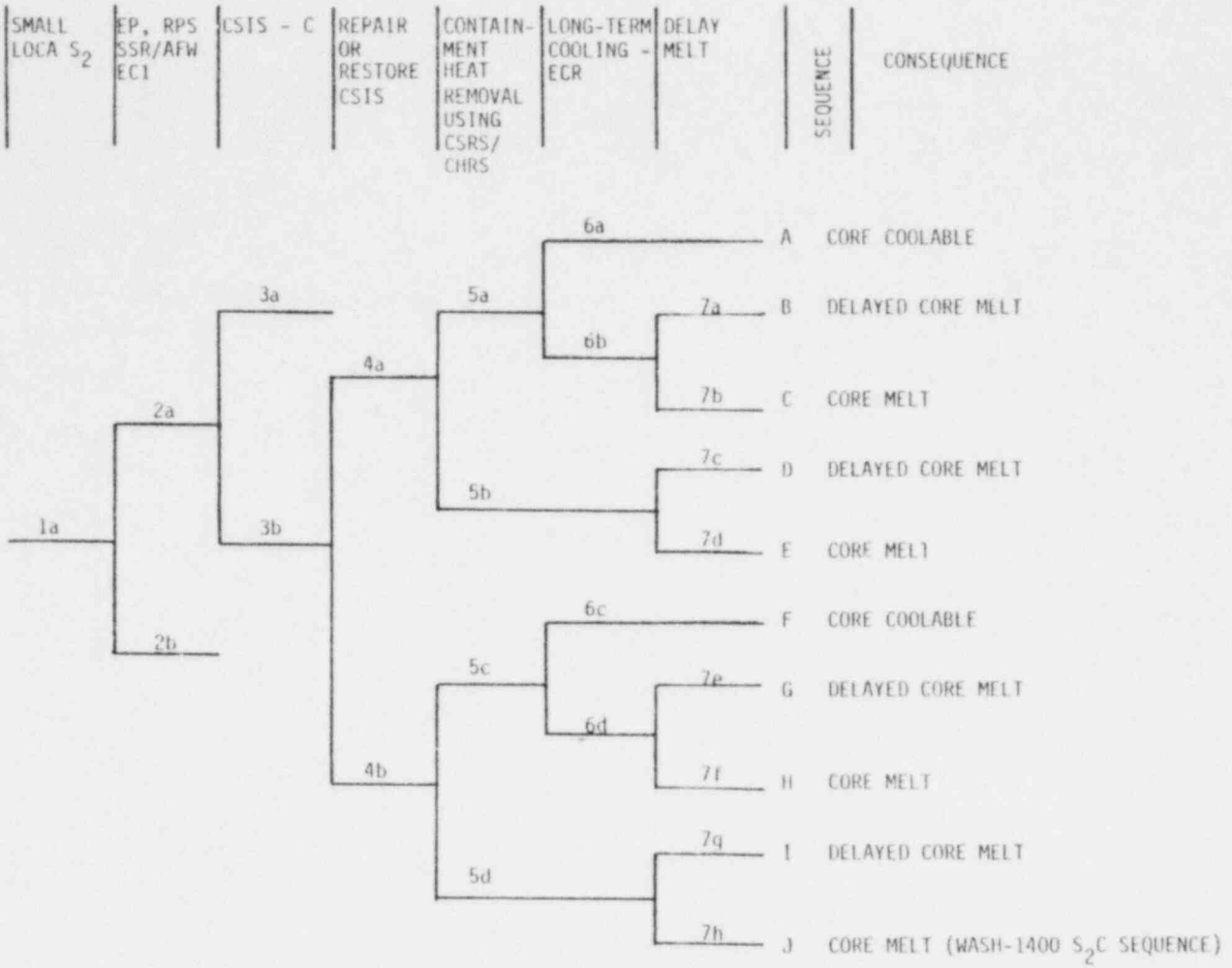

Figure A.3-2. S_2C Sequence Operator Action Event Tree

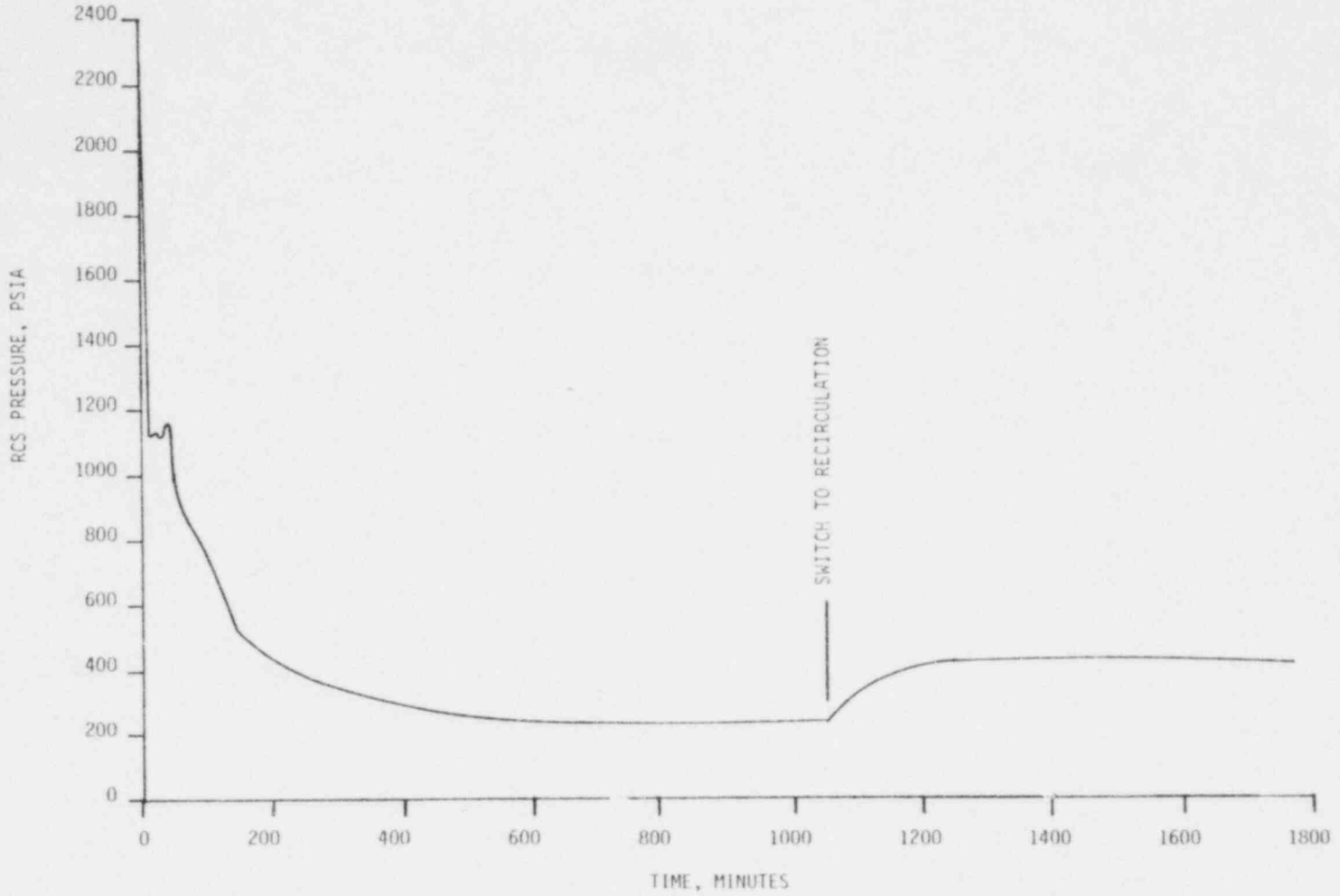

Figure A.3-3. Sequence S_2C-δ Reactor Coolant System Pressure vs. Time

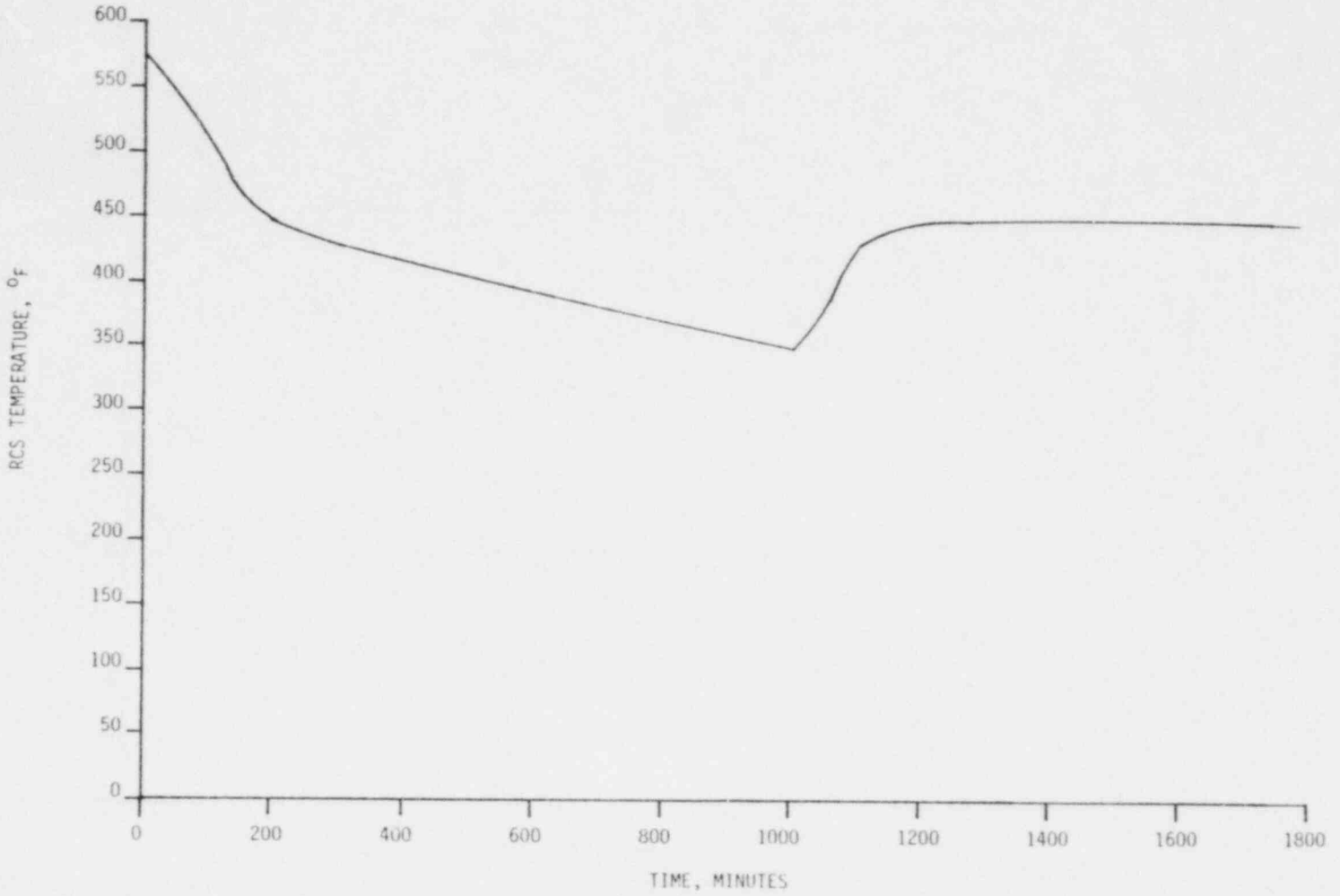

Figure A.3-4. Sequence S_2C-δ Reactor Coolant System Temperature vs. Time

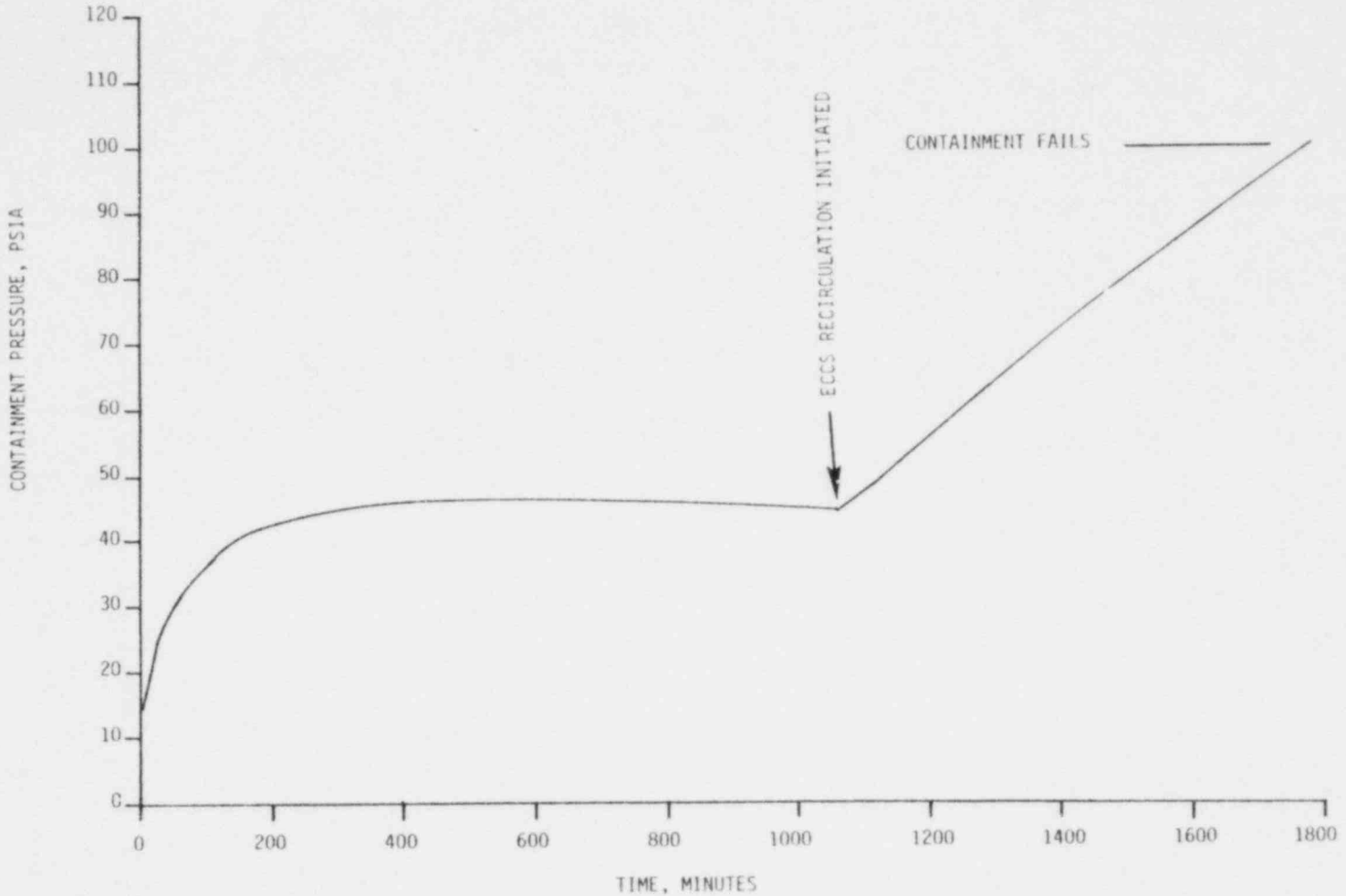

Figure A.3-5. Sequence S_2C-δ Containment Pressure vs. Time

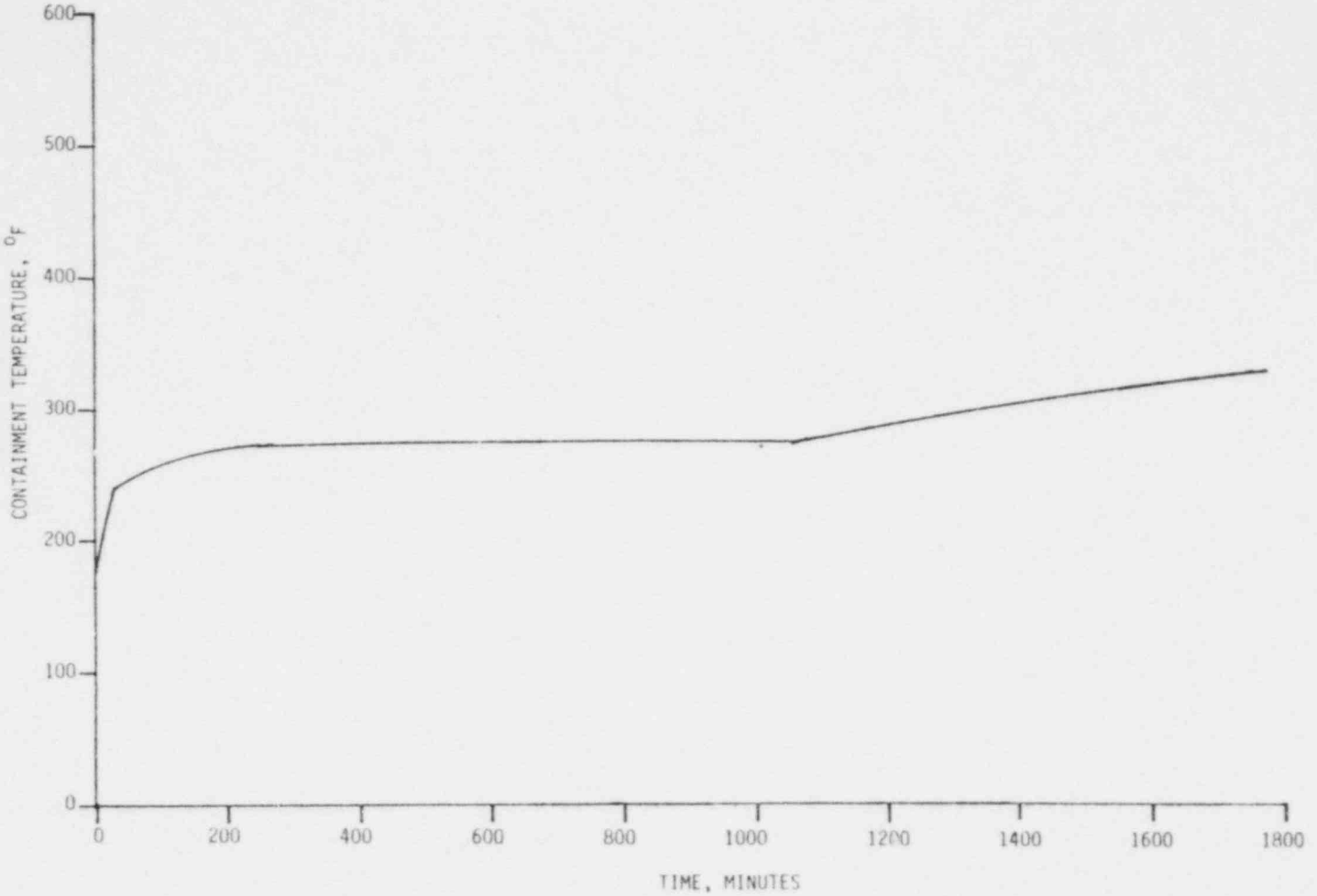

Figure A.3-6. Sequence S_2C-δ Containment Temperature vs. Time

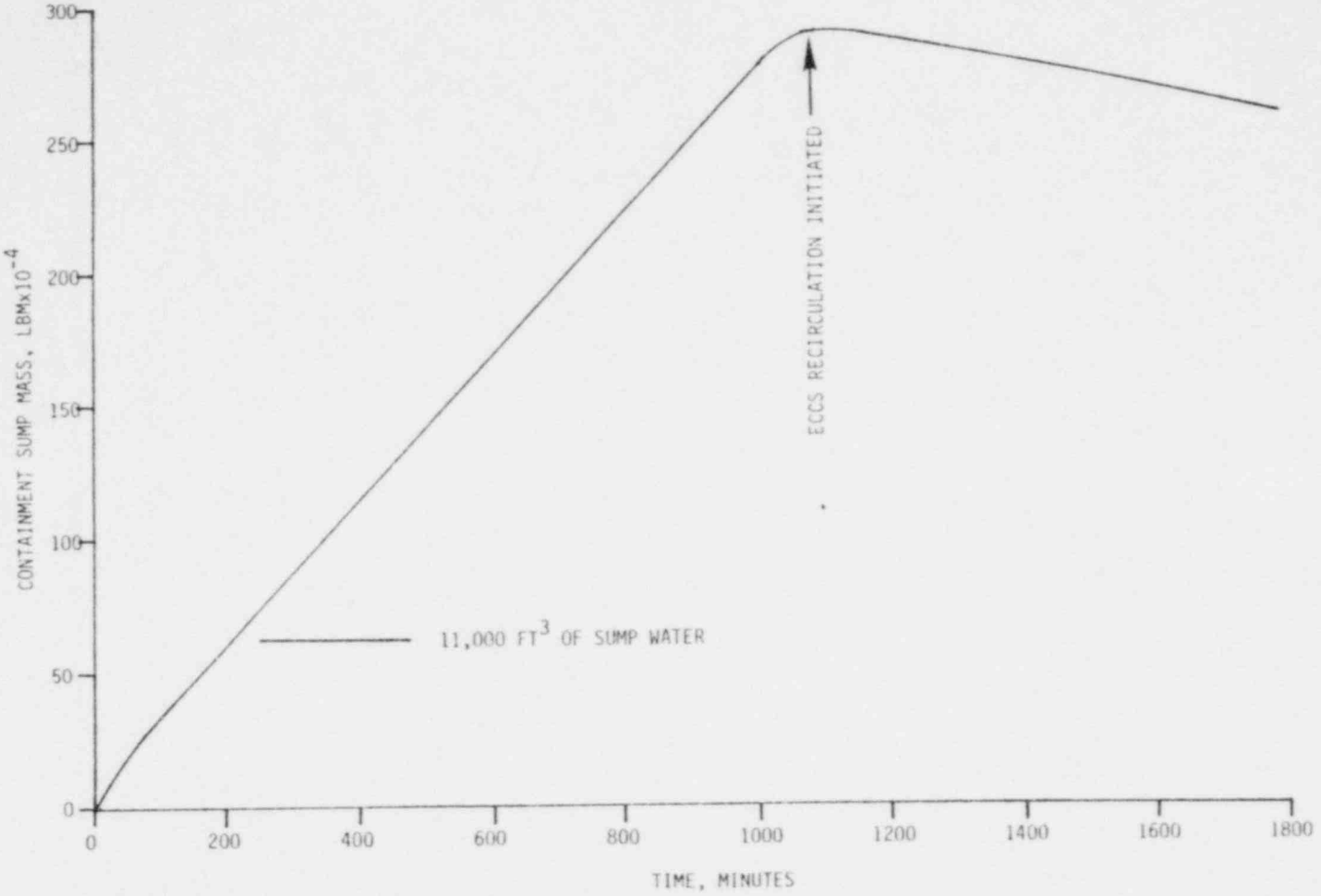

Figure A.3-7. Sequence S_2C-δ Containment Sump Mass vs. Time

Table A.3-1. Summary of Key Operator Actions and Information Requirements for S_2C-δ Sequence

Plant[1] State	Description of Plant State	Information Required for Plant State Identification	Operator Action Following Plant State Identification	Information Required to Take Appropriate Action
1a	Small break (½ to 2" dia.) in primary coolant system boundary.	o RCS Pressure, Temperature o Pressurizer Water level o Containment Pressure, Temperature, Radiation level, Humidity o Containment sump level	Identify ESF's required for S_2 accomodation, and ensure their readiness, and verify correct ESF response. Manually actuate or control any systems which do not function automatically.	o Status of key components in the appropriate ESF systems o Parameters for state 2a identification
2a	Reactor Trip, High Pressure ECI activated, AFWS activated for primary heat removal.	o Neutron Flux, Control Rod position o Core Temperature, RCS Pressure, Reactor Vessel Water Level o ECCS flow rate, HPIS pump discharge pressure o Steam generator water level, AFWS flow rate, pump discharge pressure	Control systems as required for effective ESF operation and accident accomodation.	o Same as required for state identification.
3b	CSIS failure	o Containment Pressure, Temperature o CSIS pump discharge pressure, flow rate o Containment sump level	Repair or Restore CSIS Override automatic CSRS actuation if sump water level is inadequate.	o Parameters defining the cause of CSIS failure o Sump water level
4a	CSIS function Restored. Containment Heat Removal Provided.	o Same as required for state 3b identification	Monitor containment heat removal, and primary system response. Check status of CSRS/CHRS and ECCS recirculation to ensure readiness for operation.	o Status of key components in CSRS/CHRS and ECR system o Cooling water flow and temperature in CHRS and RHRS heat exchangers.

Table A.3-1. Summary of Key Operator Actions and Information Requirements for S_2C-δ Sequence (Continued)

Plant State[1]	Description of Plant State	Information Required for Plant State Identification	Operator Action Following Plant State Identification	Information Required to Take Appropriate Action
4b	CSIS cannot be restored to operation. No containment heat removal.	o Same as required for state 3b identification	Provide containment heat removal using CSRS/CHRS.	o Status of components in CSRS. o Water level in containment sump o Sump water temperature and pressure (to define thermodynamic state).
5a 5c	Containment heat removal supplied by either restored CSIS and subsequently CSRS/CHRS (5a), or CSRS/CHRS alone (5c)	o Sames as required for state 3b as well as sump water conditions to ensure safe CSRS operation (identified under state 4a)	Provide for long-term heat removal from core using ECCS in recirculation mode.	o Sump water level o Sump water temperature and pressure o RCS pressure, temperature o Reactor Vessel Water Level o Other parameters necessary to control ECR components
5b 5d	No long-term containment heat removal capability, which eventually leads to a failure of ECR.	o Same as required for state 3b identification o RCS temperatures and pressure will detect failure of ECR o Reactor Vessel Water Level	Delay core melt as long as possible and take other consequence mitigation action to prepare for core melt.	o RCS pressure, core outlet temperature, reactor vessel water level, coolant activity level
6a[2] 6c	Long-term cooling and makeup water provided for core.	o Containment pressure temperature o RCS pressure, temperature o Reactor Vessel Water Level	Monitor and control CSRS/CHRS and ECCS as required to maintain coolable core geometry and containment integrity	Same as identified to initiate ECR for states 5a and 5c

NOTES: [1] Plant state as identified in the Operator Action Event Tree for S_2C-δ (Figure A.3-2)

[2] States 6b, 6d, and their consequent states are not addressed in this summary as they imply failures in addition to those of the S_2C sequence.

A.4 $S_1HF-\gamma$ AND $S_2HF-\gamma$ SEQUENCES

A.4.1 Sequence Descriptions

The draft report of the Sequoyah Reactor Safety Study[1] identified $S_1HF-\gamma$ and $S_2HF-\gamma$ as being dominant accident sequences. The event trees for these two sequences, as seen in Figures A.4-1 and A.4-2, differ only in the requirement for the auxiliary feedwater system. For breaks less than two inches in diameter (S_2 initiator), it is assumed in the Sequoyah RSS that the steam generators are necessary to remove some of the core decay heat being generated; for breaks greater than two inches (S_1 initiator), it is assumed that the energy removed by the break is greater than or equal to the core decay heat and the steam generators are not necessary for heat removal. During the course of this study it became evident that a better division of the small break, by both break size and break location, was needed to determine if operator actions would be dependent upon break size. Some of the break sizes and locations examined did reveal potential operator actions. Most did not require any operator action to be taken prior to the postulated failure of this sequence but did identify subsets of the Emergency Safeguards Features (ESF) needed to mitigate the effects of various break sizes. This was the criteria used to establish the range of break sizes which are described in Section A.4.2. The accident sequence for S_1HF and S_2HF, as shown in Figures A.4-1 and A.4-2, will be described in this section, and the sequences will hereafter be referred to as S_iHF. A detailed characterization of the S_i initiators will be given in Section A.4.2.

The initiating event in the $S_iHF-\gamma$ sequence is a small break which is located in either the liquid or vapor space of the Reactor Coolant System (RCS). The S_i initiator includes breaks whose equivalent diameters range from 0.5 inch to 6 inches. The event trees for the S_1 and S_2 initiators from the Sequoyah RSS are shown in Figures A.4-1 and A.4-2 with the S_iHF sequence highlighted in each. A generalized description of the S_iHF sequence follows.

Following the rupture of the reactor coolant system, the system pressure begins to decrease, with the rate of decrease being dependent upon the size of the break. The water level in the pressurizer begins to drop* as makeup flow from the

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*The exception to this would be the vapor space break or the inadvertent opening of a safety/relief valve in which case the pressurizer water level would increase.

chemical volume control system can no longer maintain adequate water level. The containment temperature and pressure will increase as high energy fluid is discharged from the rupture. This pressure increase produces a pressure drop across the ice condenser inlet doors and permits steam and air to flow through the ice condenser. The ice condenser system is a passive pressure suppression system containing metal baskets filled with borated ice flakes.

For this sequence it is assumed that the electrical power supply is adequate to meet the needs of all the Engineered Safety Features (ESF), i.e., either offsite or onsite AC power is available. As the system pressure continues to decrease, a reactor trip setpoint will be reached, e.g., overtemperature ΔT or pressurizer low pressure, which will cause the control rods to be dropped into the core. This will reduce the reactor power to approximately 6 percent of full power.

As more fluid is discharged from the rupture, the containment pressure will continue to increase. When the containment pressure reaches the high-high containment pressure setpoint, the Containment Spray Injection System (CSIS) and the Air Return Fan System (ARFS) are activated. The CSIS draws water from the Refueling Water Storage Tank (RWST) and sprays it into the containment upper compartment thereby reducing the containment pressure and removing radioactivity from the containment. Ten minutes after receipt of the high-high containment pressure setpoint, the ARFS is activated. The ARFS enhances the operation of the ice condenser system by circulating air from the lower compartment to the upper compartment in the containment through the ice condenser.

Upon receipt of the safety injection signal, the Emergency Coolant Injection (ECI) system is automatically aligned to deliver coolant from the RWST to the cold legs of the reactor coolant system. The minimum ECI required for the S_i initiator is the flow from one of two charging pumps and one of two high pressure injection pumps. The two centrifugal charging pumps,

normally part of the chemical volume control system, are aligned to take suction from the RWST on receipt of the safety injection signal. The high pressure injection pumps are components of a dedicated system which requires no alignment to inject into the reactor coolant system.

The safety injection signal also results in main feedwater system isolation and the auxiliary feedwater system activation. The auxiliary feedwater system draws water from the condensate storage tank and delivers it to the secondary side of the steam generators. The flow from the auxiliary feedwater system is sufficient to remove decay heat from the primary system. As mentioned previously, the auxiliary feedwater system is necessary during a small break transient where the break size is insufficient to remove decay heat. For breaks large enough to remove decay heat, the auxiliary feedwater system provides an additional heat sink which aids in reactor cooldown.

Two passive subsystems of the ECI system are available to inject additional cooling water into the RCS. These are the Upper Head Accumulator Injection System and the Cold Leg Accumulator Injection System. The actuation pressure of the Upper Head Accumulator is high enough that it will inject automatically for most size small breaks. The actuation pressure for the cold leg accumulator is low enough that injection will only occur automatically for the larger size small breaks, i.e., breaks sizes of approximately four inches equivalent diameter and above.

As the refueling water storage tank is depeleted, realignment of the safety injection system to the recirculation mode is automatically initiated when the RWST water level reaches the low level setpoint. Upon receipt of the low-low water level signal from the RWST, the system is completely realigned in the recirculation mode. At this time the containment spray system and the emergency cooling

system would normally take suction from the containment sump; however, in this sequence, a common mode failure is assumed to occur. The failure is the loss of flow communication between the upper and lower compartments by the drain lines being plugged or inadvertently left closed after refueling. During recirculation, the containment spray pumps would be removing liquid from the sump and spraying into the upper compartment. With no flow to the lower compartment to replace this lost inventory, the sump water would be depleted and failure of the RHR and CSRS pumps would occur. This would eventually result in the loss of both containment and core heat removal, followed by containment failure and core meltdown.

The previous scenario describes the S_1HF-γ sequence as found in the Sequoyah Reactor Safety Study[1]. This study assumed no effective operator action to respond to system failures. The following sections examine what actions the operator might take to successfully terminate this accident prior to core meltdown or containment failure, or to reduce the consequences of the event.

A.4.2 Operator Actions

The preceeding section described the S_iHF-γ sequence. This section details the actions the operator must take to successfully terminate or mitigate the consequences of the S_iHF transient. The various options which should be considered and the consequences of their implementation are discussed in this section. Once these potential operator actions have been identified and their effects have been assessed, the key parameters required by the operator to take action can be identified. Information needs of the operator are covered in Section A.4.3.

Figure A.4-3 illustrates the operator actions in response to the S_iHF initiator. This diagram was developed by considering the S_iHF sequence and modifying it to reflect potential operator actions that can be taken to respond to the event. The initial portion of the tree, prior to failure of the ECR and CSRS, has been simplified in Figure A.4-3. Those functions and systems which were assumed to function successfully are combined into one event. These are electrical power, reactor protection system, auxiliary feedwater system, air return fan system, containment spray system and emergency coolant injection. Subsequent to the failure of the ECR and CSRS, the event tree headings reflect the potential for operator action. The different branches of the event tree have been assigned alphanumeric identifiers for referencing the various plant states in future discussions.

A.4.2.1 Response to Initiator (S_i)

The initial operator action is to determine what type of event has occurred so that he will know what options are available and what equipment is needed to assist him in either successfully terminating the transient (i.e., preserving containment integrity and maintaining a coolable core geometry) or mitigating the consequences of the accident in the event of core meltdown and/or containment failure.

Unlike the large break loss of coolant accident where the accident "signature" is similar regardless of break size or location, the small break loss of coolant accident has a unique "signature" dependent on break size and location. To attempt to characterize the S_i initiator by break size and location, a number of assumptions will be made. These assumptions are consistent with the event sequence as shown in Figures A.4-1 and A.4-2 (e.g., maximum delivery of auxiliary feedwater to the steam generators and complete availability of engineered safeguards features).

Another important consideration is the availability of offsite electrical power. Ths loss of offsite electrical power will preclude operation of the steam dump system. In the event of loss of offsite power, the operator would have to ensure the start up of the emergency diesel generators and the proper loading of safeguards equipment. The loss of offsite power combined with a rupture of the reactor coolant system is a probablistically insignificant event. The reason it is considered here is that most of the analytical results available, and those used to support this study, were calculated with the conservative assumption of loss of offsite power. The difference between offsite and onsite power is in the operation of the steam dump system as already stated. More analytical work with "best estimate" assumptions is needed for small breaks. To illustrate the effects, and to describe the S_i initiators under more realistic conditions, the assumption of offsite power availability was also considered in this study.

The applicable range of break sizes being considered, 0.5 inch to 6 inches in diameter, was further subdivided to examine the effect of break size on potential operator action. It was found that three distinct ranges of break sizes could be defined. Each of these ranges required a distinct set of emergency safeguards equipment or operator actions which were essential for the successful termination of an accident in that range. The characteristics of the S_i initiator are generally applicable to all plants; however, specific plant designs, (e.g., ECI pump capacity), may shift the break ranges which are defined here.

A general characterization of the small break initiator (S_i), applicable to all break sizes and locations, is a gradual reduction in system pressure until the reactor trip setpoint is reached. The rate at which the pressure drops is dependent on the break size. After reactor trip occurs, there is a rapid reduction in system pressure which initiates the safety injection signal. It is at this point that the description for the selected break size ranges considered in the following sections will begin. The operator should verify that reactor trip has occurred and that safety injection has been initiated when the proper setpoints are reached. Verification of safety injection includes assuring that the charging pumps and associated valving are aligned in the injection mode to take suction from the RWST. In the event the reactor fails to trip automatically, the operator can scram it manually.

A.4.2.1.1 Cold Leg Breaks from 0.5 Inch to 1 Inch in Diameter

Cold leg breaks in the range of 0.5 inch to 1 inch in diameter are characterized by an eventual repressurization of the Reactor Coolant System after initiation of the high pressure injection system. Immediately after safety injection is initiated, there is a rapid reduction in the Reactor Coolant System (RCS) temperature and pressure as shown in Figures A.4-4[(2)] and A.4-5*. The break flow for the breaks in this range is subcooled liquid and the energy being removed is less than the energy being generated by core decay heat. This requires that the steam generators and auxiliary feedwater be available to remove heat from the primary system. The steam generator would be needed for approximately one day for the 1 inch diameter break to remove decay heat and for greater periods of time for smaller breaks. If the auxiliary feedwater system were unavailable, a backup means of removing decay heat would be the manual operation of the pressurizer power operated relief valves. This would increase the effective break area, and the accident signature subsequent to PORV operation would be similar to the vapor space break description of Section A.4.2.1.5.

In the event of loss of offsite power with subsequent loss of steam dump to the condenser, the steam generator secondary pressure would rise to the steam generator secondary side safety valve setpoint and steam would be discharged to maintain the secondary side pressure as seen in Figure A.4-6. With steam dump

*Figures A.4-4 through A.4-7 are not specific to the Sequoyah Plant but of a similar design which does not incorporate upper head injection. This analysis assumed loss of offsite power.

available, the continued addition of auxiliary feedwater to the steam generator would reduce the secondary side pressure and temperature.

The pressure in the primary would be governed by the equilibrium pressure established by the safety injection and break flow. The combined injection of two centrifugal charging pumps and two safety injection pumps is greater than the break flow at pressures near the safety injection signal setpoint. This results in the liquid inventory of the primary system increasing and an increase in system pressure as seen in Figures A.4-7 and A.4-5. When the flow from the high pressure injection system and the break flow are equal, the system will have reached a stable equilibrium pressure. This equilibrium pressure could exceed the setpoint at which the pressurizer relief valve opens (this is dependent on the shutoff head of the charging pumps). Action would be necessary by the operator to control safety injection flow to maintain the Reactor Coolant System pressure and level at an acceptable limit. Present NRC criteria require the high pressure injection to be terminated when the reactor coolant system temperature reaches 50^{o}F subcooled. The operator can maintain the reactor liquid inventory through judicious use of the makeup flow and safety injection flow.

If less than full emergency coolant injection were available, the pressure at which equilibrium is reached would be less than the pressurizer relief valve setpoint but above the pressure at which upper head accumulator injection will occur. Therefore, for this range of breaks, upper head accumulator injection will not occur early in the transient. Eventually, after the system has stabilized, with the operator controlling safety injection and after plant cooldown has been initiated, the system pressure will drop below the upper head accumulator pressure and injection will occur unless the system has previously been isolated. At no time during this transient will the liquid level drop below the top of the fuel rods.

A.4.2.1.2 Cold Leg Breaks From 1 Inch to 4 Inches in Diameter

Breaks in this range are characterized by the system pressure stabilizing at some pressure which is below the upper head accumulator injection pressure but above the cold leg accumulator injection pressure. Upper head accumulator injection will occur for this range of breaks and the steam generator is still required to aid in removing decay heat during the early portion of the transient. The time for which the steam generator is necessary to remove decay heat becomes shorter as the break size increases.

In the event that no steam dump to the condenser is available (i.e., loss of offsite power), the system pressure will initially remain above the steam generator safety valve setpoint so that core heat can be removed by the steam generators which are discharging steam through the safety valves. The break flow and injection flow are in non-equilibrium at this pressure with a higher break flow discharge rate. There is a net loss of mass from the system, and the system will continue draining until the break is uncovered. At this point steam will be relieved through the break, and the system continues to depressurize as seen in Figure A.4-9. The break will be removing more of the decay heat and the steam generator pressure will begin to drop. To remove sufficient mass from the system to uncover the break results in the liquid level falling below the top of the core (see Figure A.4-9) and partial core uncovery occurs from most of the breaks in this range. Eventually, the system will stabilize at a pressure which is between the upper head accumulator and the cold leg accumulator back pressures.

Figures A.4-8 through 10 show the transient beyond the time when the Refueling Water Storage Tank is exhausted and switchover to recirculation would have occurred for a 2 inch diameter cold leg break. When Emergency Core Cooling terminates, the system pressure increases and the core water level decreases until the volumetric flow of steam through the break exceeds the rate at which steam is generated in the core. The primary system depressurizes until the cold leg accumulators begin injecting and the core liquid level increases. This results in large volumes of steam being generated in the core and a cyclic oscillation of system pressure and core liquid level is established until the accumulators are empty. At this point, the core liquid will boil away and core melt follows.

With steam dump to the condenser available, the steam generator secondary pressure will not rise to the safety valve setpoint. This results in the primary system pressure being just above the secondary side pressure in order to maintain the temperature difference necessary to remove decay heat. As the secondary side cools, this results in a steady decrease in both secondary and primary pressure. Since the primary pressure decreases without the necessity of uncovering the break, the core liquid level never drops to the point where the fuel rods are uncovered. Eventually, the pressure will reach an equilibrium condition which is above the cold leg accumulator injection pressure as described previously.

A.4.2.1.3 Cold Leg Breaks from 4 Inches to 6 Inches in Diameter

Breaks in this range are characterized by a system pressure which stabilizes below the cold leg accumulator injection pressure but above the low pressure injection system pressure setpoint. Breaks of this size can remove all the heat generated by the core very early in the transient, therefore, the steam generator is not needed to aid in heat removal. The reason for this is that the rate of mass removal from the system is high, and the break is soon uncovered allowing steam to be relieved through the break.

Even with no steam dump available, the system pressure will rapidly drop below the steam generator safety valve setpoint pressure and the steam generator becomes a heat source. With steam flow through the break, the volumetric flow rate is higher than safety injection flow and equilibrium between safety injection and break flow will not be reached until the pressure drops below the cold leg accumulator pressure. The high break flow rates result in rather deep uncoveries of the fuel rods. Eventually, the coolant from the cold leg accumulator and safety injection will recover the core and an equilibrium pressure will be reached which is above the low pressure injection setpoint pressure.

With steam dump available, the scenario is only slightly different from above. Since the break is sufficiently large to remove all the heat generated in the core very early in the transient, heat removal through the steam generator will not be significant and the availability of steam dump will minimally affect the accident sequence.

A.4.2.1.4 Hot Leg Breaks from 0.5 Inch to 6 Inches in Diameter

Hot leg small breaks are very similar to cold leg small breaks in many respects. The equilibrium pressure reached between safety injection and break flow is similar, and the break sizes at which upper head and cold leg accumulator injection occur are similar.

The major difference between hot leg and cold leg breaks will be the core mixture level transient. Because the break is located in the hot leg, the steam generated by decay heat in the core has a direct path to the break. Hot leg break will vent steam or high quality fluid sooner in the transient than the corresponding cold leg break. This results in less mass being released through the break and the core remains covered with liquid for the complete range of break sizes, i.e., 0.5 inch to 6 inches in diameter. The best indication of the break size will be the pressure at which equilibrium is reached since this remains the same as the cold leg break for the same break size.

A.4.2.1.5 Vapor Space Breaks

In most respects the vapor space break is very similar to the cold leg and hot leg breaks already described. The distinctive feature of the vapor space break is the pressurizer mixture level transient. In the cold leg and hot leg breaks the pressurizer level would drop and not recover

back into the pressurizer. The exception would be the case of breaks less than 1 inch where repressurization occurs and the liquid level would return into the pressurizer. This is characterized by an increase in both pressure and level. In the case of the vapor space break, only the pressurizer level increases, but not the system pressure. This can be clearly seen in Figures A.4-11 and A.4-12[3]. Another indication would be a change in the pressurizer relief tank level, pressure and temperature, if the vapor space break is an inadvertently open relief or safety valve.

The pressurizer level during the vapor space break may never be low enough to activate the low pressurizer level signal. Until recently, the safety injection signal was activated on a coincident low pressurizer pressure and low level signal; thus, operator action would have been required to manually initiate injection. However, the NRC's Office of Inspection and Enforcement Bulletin 79-06A eliminates the coincident logic noted above for initiation of safety injection. Assuming these changes have been implemented no operator action (other than verifying that the safety injection has been automatically initiated) is necessary.

A.4.2.1.6 Actions Subsequent to the S_i Initiator

After verifying the existence of a break of the reactor coolant system, the operator's next action would be to identify the critical engineered safety features necessary to contain or mitigate this event. The preceding accident descriptions would assist the operator in identifying these critical systems. The systems identified as essential to mitigating the consequences of these events include the electrical power system, the reactor protection system, the auxiliary feedwater system, the high pressure injection system, the upper head accumulator injection system, the cold leg accumulator injection system, the air return fan system and the containment spray injection system. The functions of these systems have been discussed in Section 4.4.1. The status of each of these systems should be checked to ensure that it is ready to operate upon demand. Any system which is deactivated for testing or maintenance should be returned to a standby condition if it has not already been done automatically. It may also be necessary for the operator to terminate the operation of some equipment, e.g., high head safety injection for breaks of less than one inch, to lessen the consequences of the event.

As a result of post TMI analysis, it was determined that the reactor coolant pumps should be automatically tripped early in a small loss of coolant accident. It has not been determined which signals or parameters are appropriate to perform this function. When this directive is incorporated into plant operating instructions, the operator will have to check that the reactor coolant pumps have tripped on the appropriate signal.

A.4.2.2 Response to the Recirculation Failure (HF)

After identifying that a small rupture of the reactor coolant system has occurred and verifying the operation of engineered safeguards features, the operator, without taking any prior actions except those potential actions already identified for vapor space or small breaks less than one inch, is awaiting the signal to begin the switchover to the recirculation mode of operation. Switchover to recirculation is begun upon receipt of a low level signal from the refueling water storage tank in conjunction with a high water level indication from the containment sump. Since this sequence assumes failure of the drain line between the upper and lower compartment, the automatic switchover may not be initiated. The high containment sump water level signal may be generated for some of the break sizes but not for others. The length of time that the operator has to become aware that the drains are inoperative is dependent upon the break size. The higher the equilibrium pressure reached by the system, the lower the safety injection flow and the longer the time until the RWST is exhausted. Also, for very small breaks of less than 1 inch where the operator would be controlling the safety injection to prevent excessive repressurization, the time before the RWST is exhausted would be extended even more. Further analytical study is needed to determine the containment pressure transients, the time that the RWST low-low signal is generated, and the sump water level for representative small breaks. There may be some segment of breaks for which the S_iHF sequence does not present a problem. Section A.4.3 will indicate instrumentation which will possibly alert the operator to this malfunction prior to exhausting the RWST. The remainder of this section will deal with the supposition that the operator is unaware of the malfunction until switchover to recirculation is required.

As previously mentioned, with the containment spray system drawing water from the RWST, the break flow alone may provide sufficient inventory to the containment sump to actuate the sump high level signal. The initiation of the sump high water level signal in conjunction with the RWST low level alarm signal would begin automatic alignment of recirculation, and the operator would manually complete the alignment. The inherent danger would be that alignment of the CSRS and ECR were completed and the operator failed to recognize that the drain had malfunctioned. With the CSRS taking suction from the sump and spraying into the upper compartment, eventually the water in the sump would be depleted and insufficient suction for the RHR and CSRS pumps would result. This would lead to failure of both systems and a core melt would result (state 4b of Figure A.4-3).

A.4.2.2.1 Response to Drain Malfunction

Subsequent to the malfunction of the drain, the operator can either restore the operation of the drain and ensure the flow of water between the upper and lower compartment, or he can find an alternate means of supplying water to the core from a backup source. No such source has been identified at this point.

Restoring the flow between the upper and lower compartment requires that: (1) the fault can be identified; and (2) that corrective action can be taken. The potential causes of drain failure have been identified as: (1) the drains were isolated during refueling and not reopened; or (2) the drains are plugged by debris. It is not entirely clear how these drains are opened or closed; whether this can be accomplished remotely is not apparent. Any action to restore operation which requires personnel to enter the containment is unacceptable.

A.4.2.2.2 Long-Term Heat Removal

If the operator is successful in restoring flow to the containment sump (4a), then the next step is to ensure that the emergency coolant recirculation system and the containment spray recirculation system perform as designed (Sequence A of Figure A.4-3).

If the Emergency Coolant Recirculation System fails to operate (5b), the eventual result is the boiling of liquid from the core (exposing the fuel rods) and the clad and fuel begin to melt.

If the operator is unsuccessful in restoring flow to the containment sump (4b), then he must take steps to extend the time to core melt to minimize releases to the environment.

Actions available to the operator to extend the time to core melt are dependent upon the amount of water in the containment sump and the thermodynamic state of this water. One means to delay containment failure and core melt would be for the operator to manually control the operation of the containment spray pumps. This would result in a faster increase in containment pressure than if the containment spray pumps were continuously running, but it would reduce the depletion of the sump water. The operator would have to closely monitor the containment pressure and sump water temperature to ensure that thermodynamic conditions for proper operation of the recirculation system are maintained. The operator would also limit the containment pressure below the threshold where containment failure is likely to occur. With the loss of communication between the upper and lower compartments, only the RHR heat exchangers are available to remove heat from the sump water. Since the RHR heat exchangers have only one-third the heat removal capability of the CSRS heat exchangers, the sump water temperature may eventually reach a condition where RHR pump cavitation occurs prior to the time when sump water inventory is exhausted.

A second alternative would be to shut off the containment spray pumps and manually switch to RHR spray. This diverts a portion of the RHR to spray headers in the upper compartment. The RHR spray has one-half the heat removal capability of the CSRS. This will deplete the sump water inventory at a slower rate than alternative one and will retard the rise of containment pressure, but it will also reduce the amount of coolant being supplied to the high pressure injection pumps.

In conjunction with either alternative would be the control of the high pressure injection flow to provide the minimum amount of coolant to maintain acceptable core outlet temperatures and liquid levels to ensure adequate core cooling. Method two, above, could also be used to delay core melt if the containment spray system were to fail (6b). The danger of RHR pump failure due to improper sump thermodynamic conditions is also a possibility as discussed above.

A.4.3 Operator Information Requirements

The preceding section addressed the operator action in response to the postulated S_1HF sequence. The principle actions are summarized below:

1. Identify occurrence of small break.
2. Determine ESFs required and verify their status and successful operation.
3. Identify drain malfunction.*
4. Restore flow communication between upper and lower compartments if possible.
5. Ensure long-term containment and core heat removal.
6. If drain flow or long-term heat removal cannot be restored, delay core melt as long as possible.

To take these actions and make the associated decisions, the operator must have a clear understanding of the plant state at all times and know what options are available. This section addresses the information which will enable the operator to determine the plant condition during a postulated S_1HF sequence and thus implement the above actions as necessary. A summary of operator information requirements and appropriate actions for the relevant plant states in Figure A.4-3 is presented in Table A.4-1.

The first operator action is to determine that a rupture in the primary coolant boundary has occurred. The parameters which unambiguously indicate a small break are the reactor coolant system pressure decreasing to some equilibrium pressure which is a function of break size, an increase in containment pressure, temperature and radiation level and a decreasing vessel water level. Characterizations which are dependent on break size and location, as described in

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*This specific failure mode has been considered in the S_1HF evaluations because it has been determined to be the dominant risk contributer. In general, the operator action would be to identify the cause of ECR failure.

Section A.4.2, include decreasing core water level and decreasing pressurizer water level, the latter being a characteristic for all breaks except vapor space breaks. For the inadvertent opening of a pressurizer relief valve, an indication would be valve position or discharge line temperature. The pressurizer relief tank level could also be used if it does not accept flow from any other source than the pressurizer.

After verifying that a break of the reactor coolant system boundary has occurred, the next operator action is to identify the ESFs required to maintain both containment integrity and core coolable geometry. The specific systems are mentioned in Section A.4.1 and are illustrated on the S_1HF event trees (Figures A.4-1 and A.4-2). The status of each of these systems should be checked to ensure their readiness for operation and, once their actuation is required, to verify that correct system response has been performed.

In the event of loss of offsite power, which is annunciated within the control room, the operator must ensure that the diesel generators have operated as designed. The reactor trip signal will also be annunciated and the operator should check for successful insertion of all control rods into the core. Control rod position indicators and neutron flux measurements are available to ensure a safe shutdown margin. A manual trip of the control rods can be performed if necessary.

The low pressurizer pressure or low pressurizer level signal should actuate the safety injection signal. The operator should verify that the safety injection pumps have started and that the valves that align the charging pumps to the RWST are in proper position. Operation of the charging and safety injection pumps can be verified by discharge pressures and flow rates and valve positions by valve indicators. The operation of the passive injection systems, the upper head injection and cold leg accumulators, can be verified by monitoring the accumulator level and pressure indicator when their appropriate actuating pressure has been reached. The safety injection signal trips the main feedwater pump and initiates the auxiliary feedwater pump. Successful operation of auxiliary feedflow can be confirmed by the pump discharge pressure and flow rate. In addition, steam generator level will indicate if adequate auxiliary feedwater is being delivered. Auxiliary feedwater is required to remove decay heat for some breaks as previously discussed in Section A.4.2.

Successful operation of containment pressure reducing systems include the ice condenser system, the air return fan system and the containment spray system. The operator should check for actuation of these systems on the appropriate containment pressure signals. Containment pressure in the lower compartment will give some indication of successful operation of the ice condenser system. The discharge flow from the air return fan system and the containment spray system are indications that these systems have operated successfully. In addition, the air return fan system low flow alarm will annunciate if flow is below 20,000 cfm from either fan. Successful operation of these systems would also be indicated by a reduction in containment pressure.

For the operator to determine that the drain between the upper and lower comparment was inoperative, he would need to have an indication of 1) whether the drain line is open or closed, e.g., a valve position indicator if the drain line is isolated by a valve; 2) the rate at which the sump water level should be increasing given some knowledge of the break flow and containment spray rate and 3) water level indication for the upper containment compartment.

Knowledge that water level is increasing in the upper containment compartment would be the simplest method to employ to inform the operator that there is no flow communication between the upper and lower level compartments. Drain line valve position indicators are necessary but are not sufficient to indicate a drain line malfunction; level instrumentation would be required to [illegible]ine a malfunctioning drain line if the indicators should state that the [illegible]ns are open. The rate at which the sump is filling would be the most difficult indication of drain line malfunction. The operator would need detailed knowledge of the containment spray injection rate and the break flow rate.

If the flow through the drain is restored, then the operator must ensure that the safety injection system is aligned in the recirculation mode of operation. The operator must continually monitor containment sump water level to assure that proper suction is available to the RHR and CSRS pumps. If the water level is dropping, the operator must limit the operation of ESFs to maintain sufficient sump water level while monitoring con[illegible]inment

and reactor system parameters to ensure that they are within acceptable limits.

Sufficient water subcooling must be ensured, otherwise the RHR and CSRS pumps could fail from cavitation. The margin for safe pump suction can be determined by measuring sump water temperature and containment pressure and comparing the resultant state to saturated conditions.

If the drain flow cannot be restored, the operator action is directed toward delaying a core meltdown. As discussed in Section A.4.2, the operator would minimize the amount of sump water which is diverted to containment spray during the recirculation phase. The operator would use either the CSRS or the RHR spray to maintain the containment pressure at an acceptable level. Measurements needed would be containment pressure and temperature, sump water level and temperature, reactor coolant system pressure and temperature, and core water level.

A.4.4 Summary and Conclusions

The preceding discussion considered the S_1HF sequence and identified potential operator actions to accommodate this sequence or reduce its consequences for an upper head injection plant. The reactor and plant parameters which are necessary and sufficient to define the plant state during the accident and thereby provide the operator with clear information on which to take the proposed corrective actions have been identified. The results of this evaluation are shown in Table A.4-1.

The information presented in the summary table is based on a number of assumptions concerning the plant performance and response to the postulated sequence. Many of the plant conditions and proposed operator actions have not been analyzed in the past. Hence, there is some uncertainty and generality in these evaluations. The following list identifies areas where further information would be beneficial in either confirming the key assumptions used in this study or reducing the level of uncertainty:

- What indications are available to inform the operator that the drains between the upper and lower compartments are shut? Can these drains be remotely opened?

- For what break sizes will there be sufficient water in the sump to generate the high sump water signal to begin switchover to recirculation? What are the break sizes for which neither the CSRS nor RHR spray is needed to maintain containment integrity?
- Analysis would be needed to determine what is the optimum method of those suggested in Section A.4.2.2.2 to delay the time to core melt. Is heat removal by the RHR heat exchangers alone sufficient to maintain sump water thermodynamic conditions?
- Are there alternate sources of backup water available to replenish sump water?

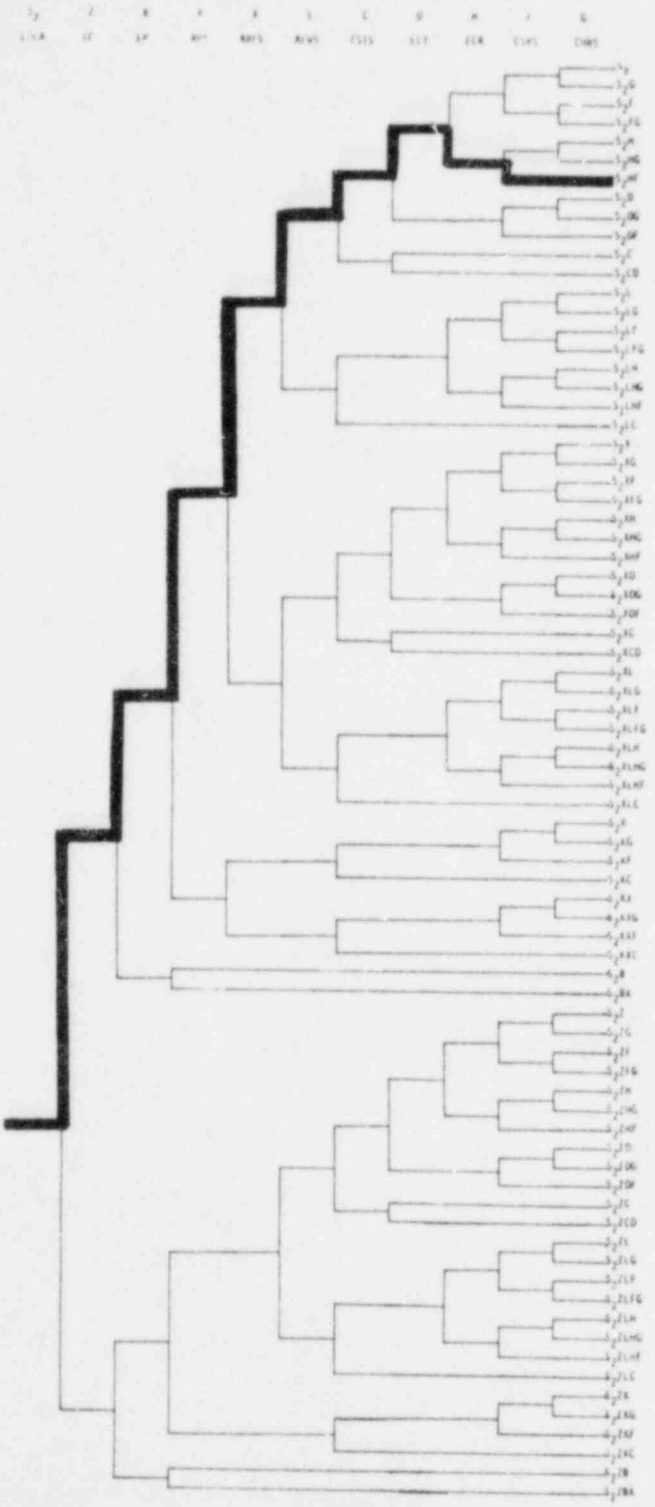

Figure A.4-1

SMALL LOCA (S_2) EVENT TREE (0.5"-2.0" DIAMETER) SEQUOYAH PLANT

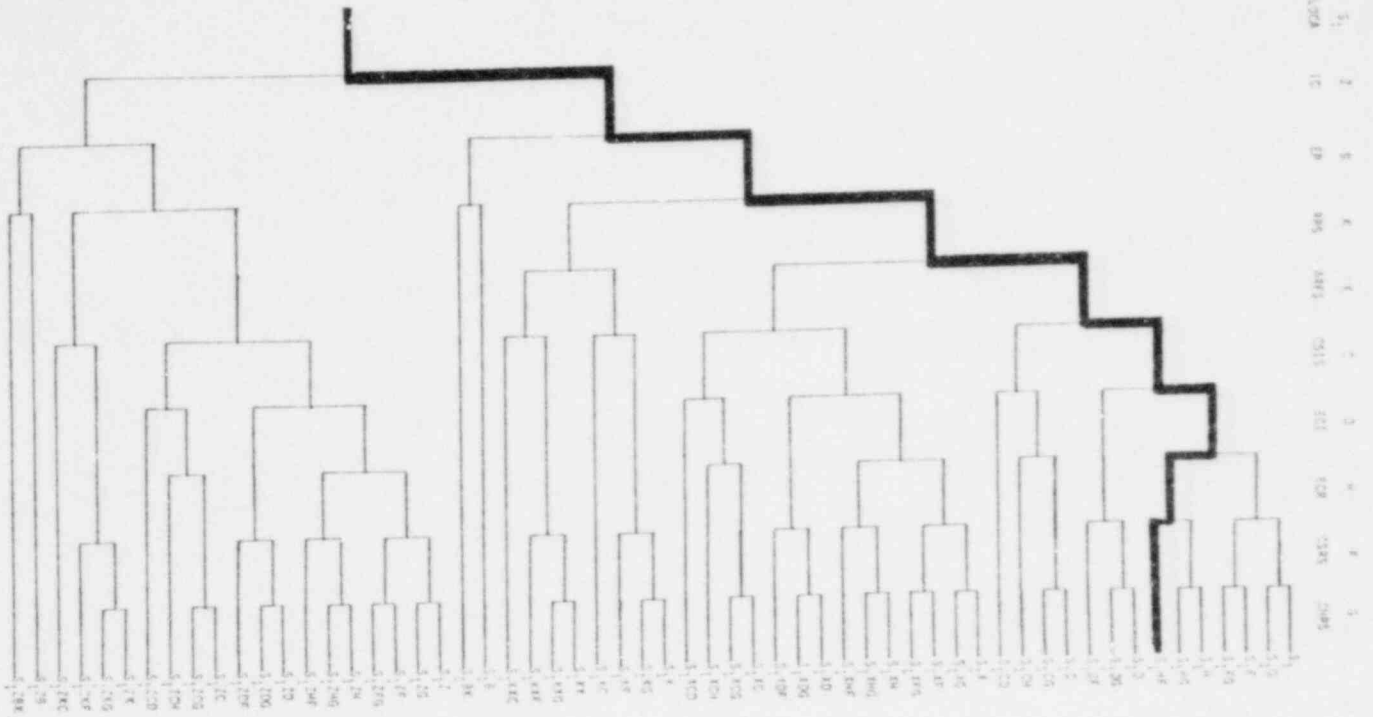

Figure A.4-2. Small LOCA (S_1, 2"-6" Diameter) Event Tree, Sequoyah Plant

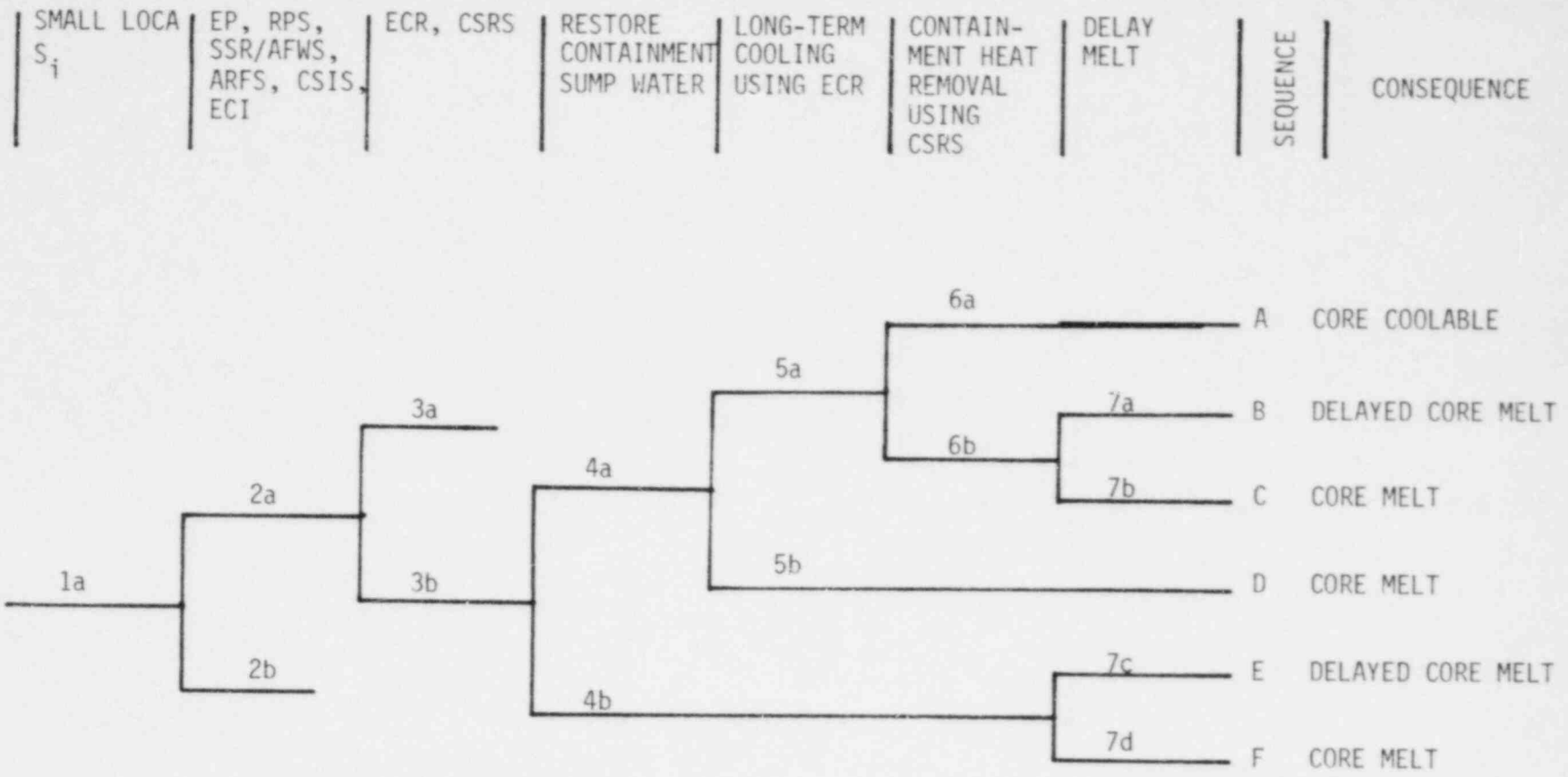

Figure A.4-3. S_iHF SEQUENCE OPERATOR ACTION EVENT TREE

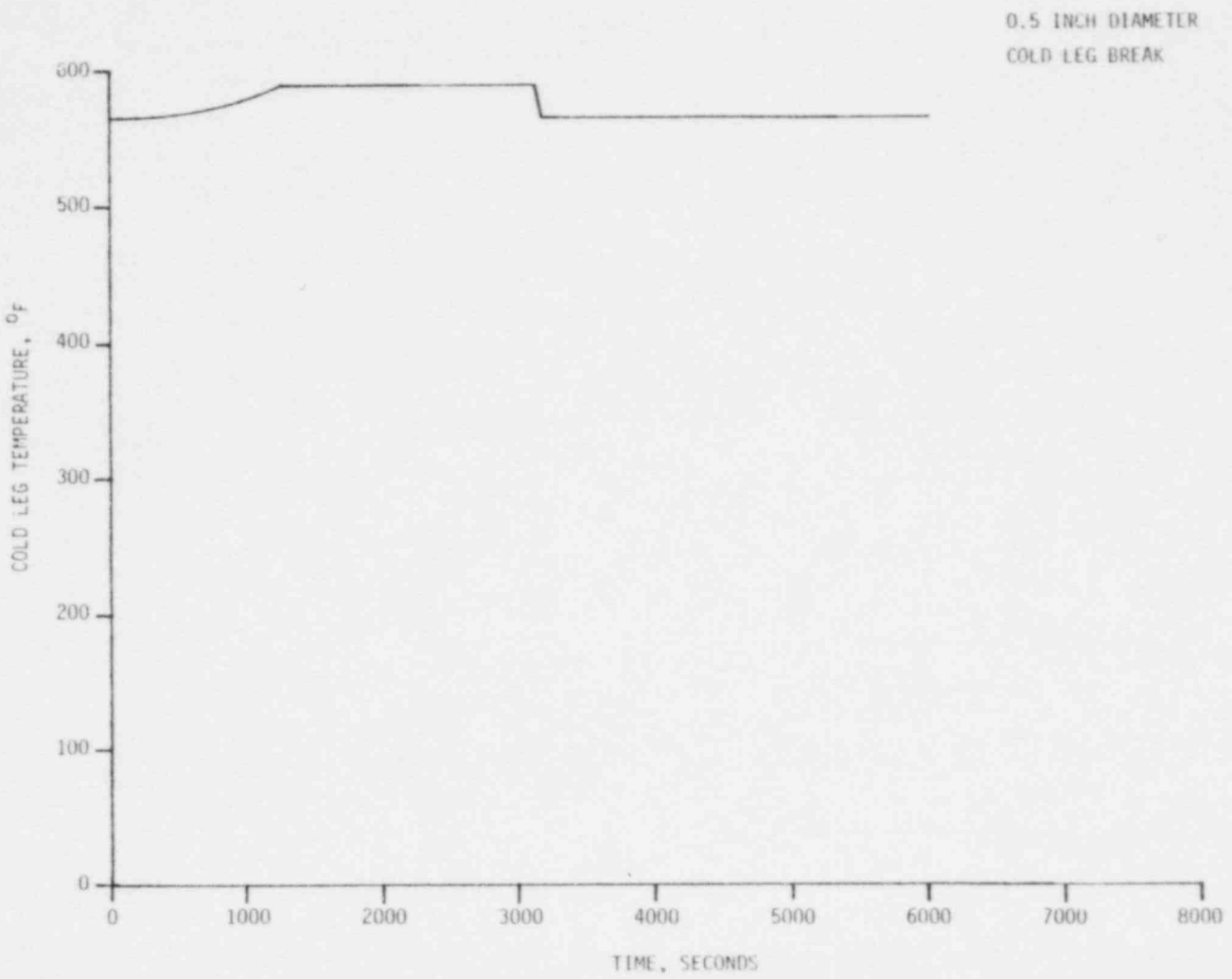

Figure A.4-4. Cold Leg Temperature vs. Time

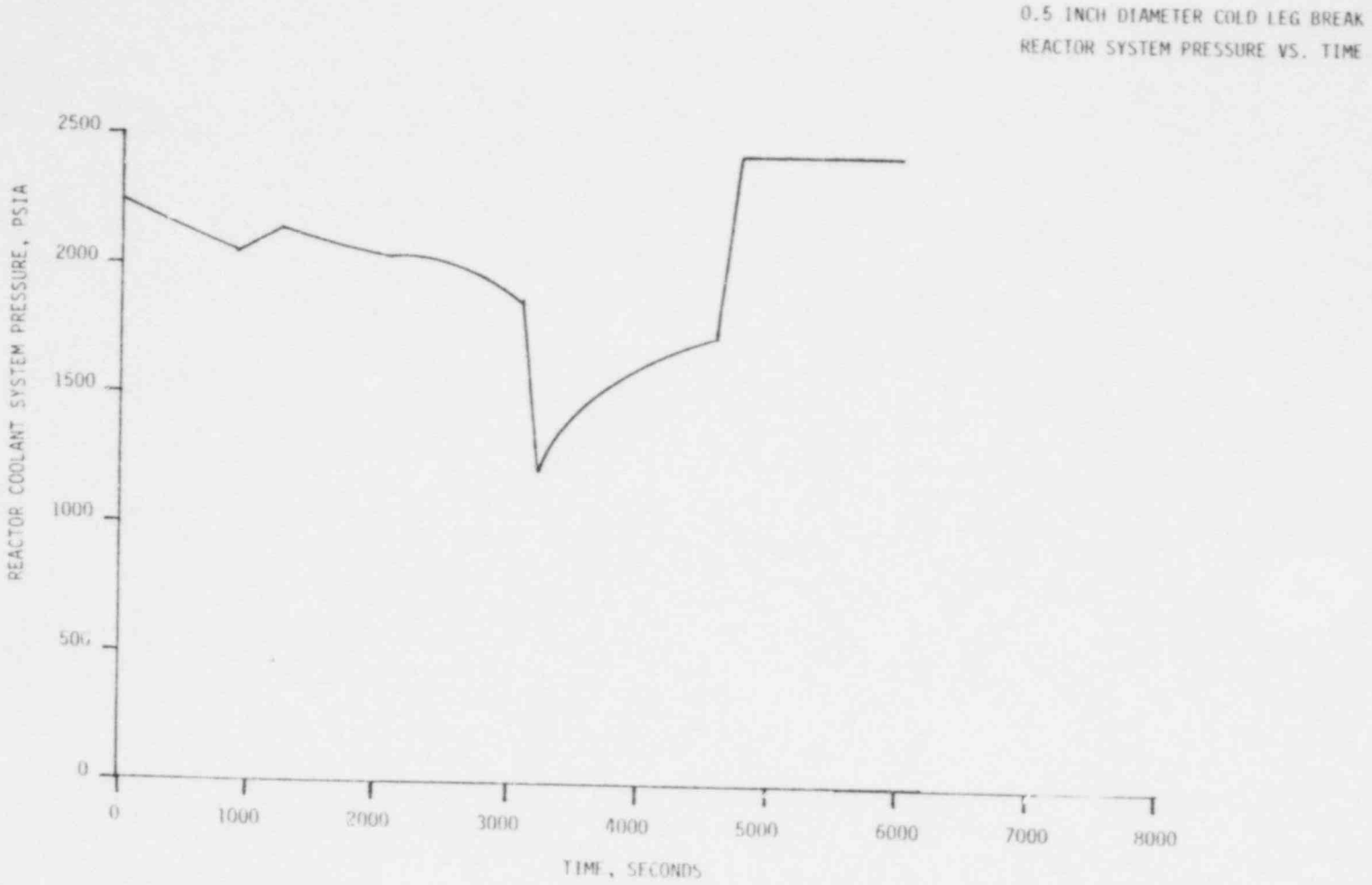

Figure A.4-5. 0.5 inch Diameter Cold Leg Break Reactor System Pressure vs. Time

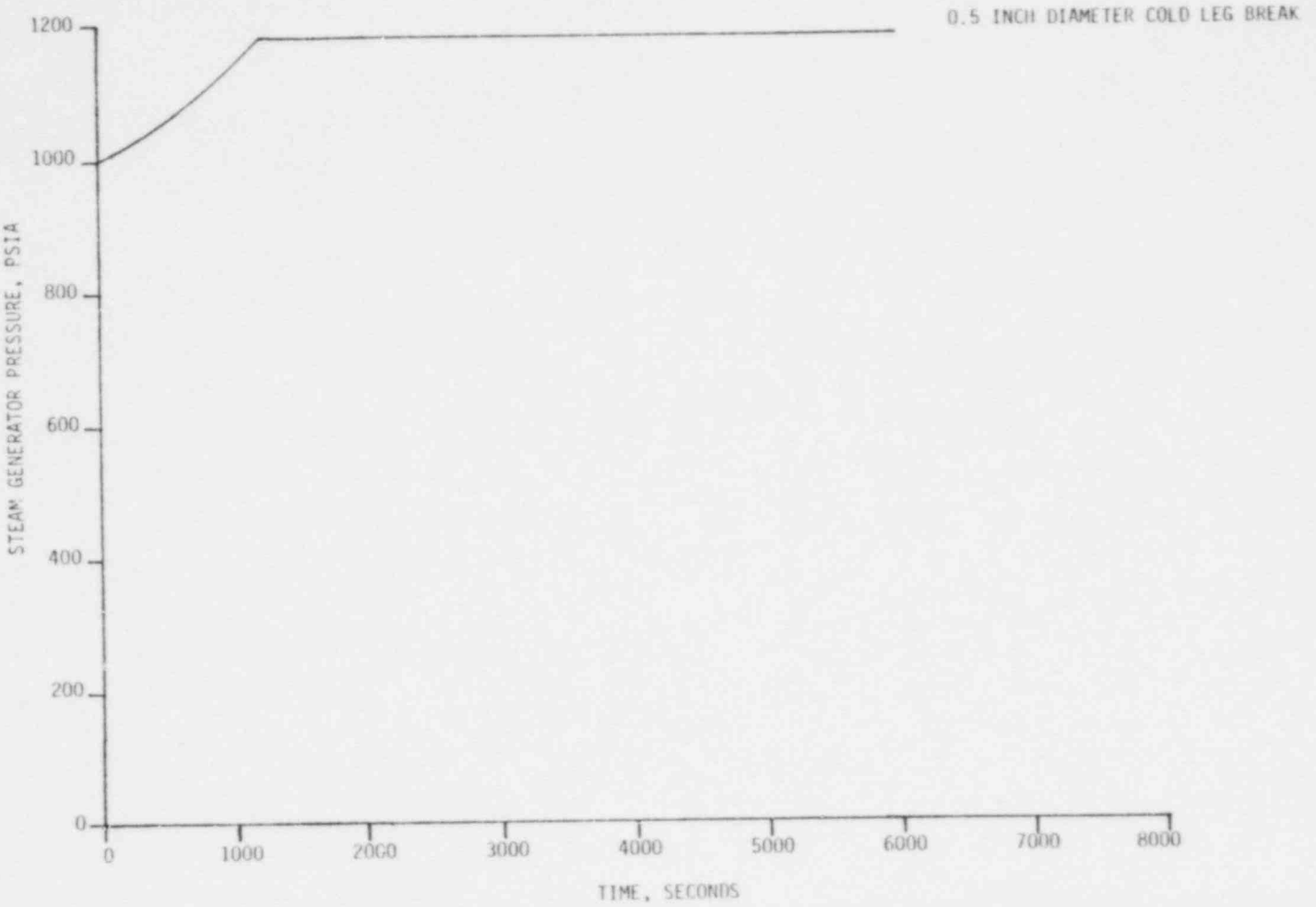

Figure A.4-6. Steam Generator Secondary Side Pressure vs. Time

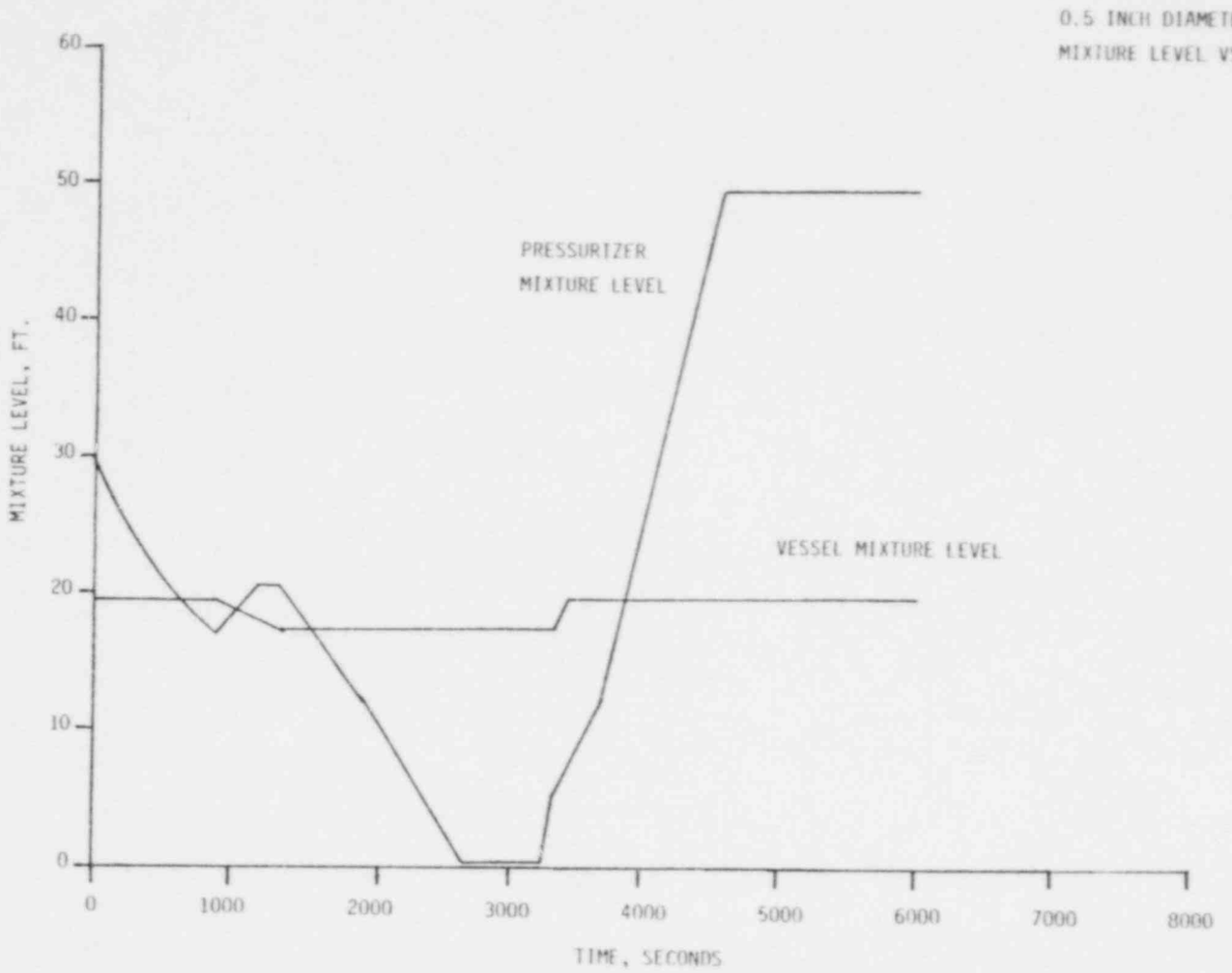

Figure A.4-7. Pressurizer and Vessel Mixture Level vs. Time

Figure A.4-8. Sequence S_2HF Reactor Coolant System Pressure vs. Time (2 inch Cold Leg Break)

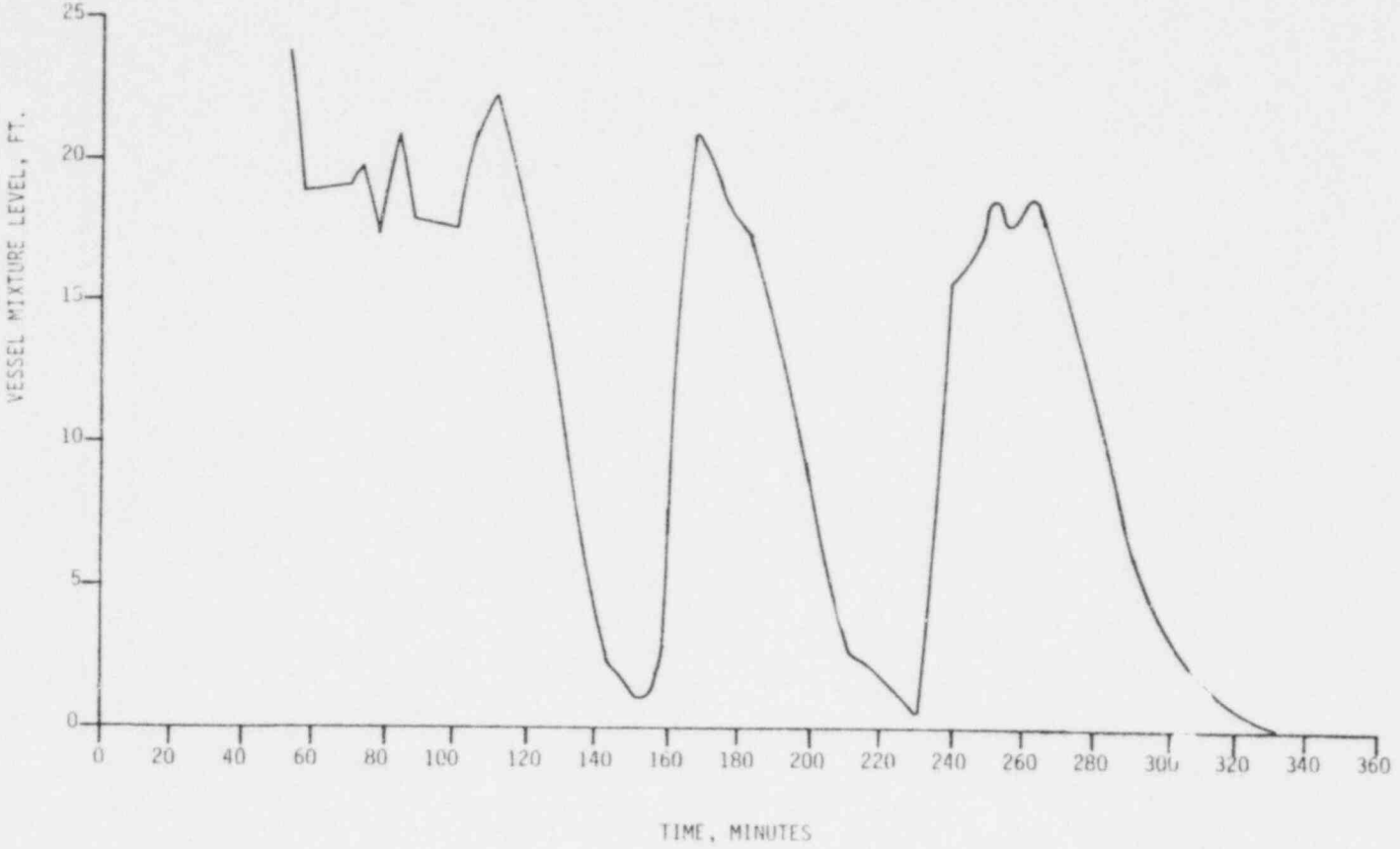

Figure A.4-9. Sequence S_2HF Vessel Mixture Level vs. Time (2 inch Cold Leg Break)

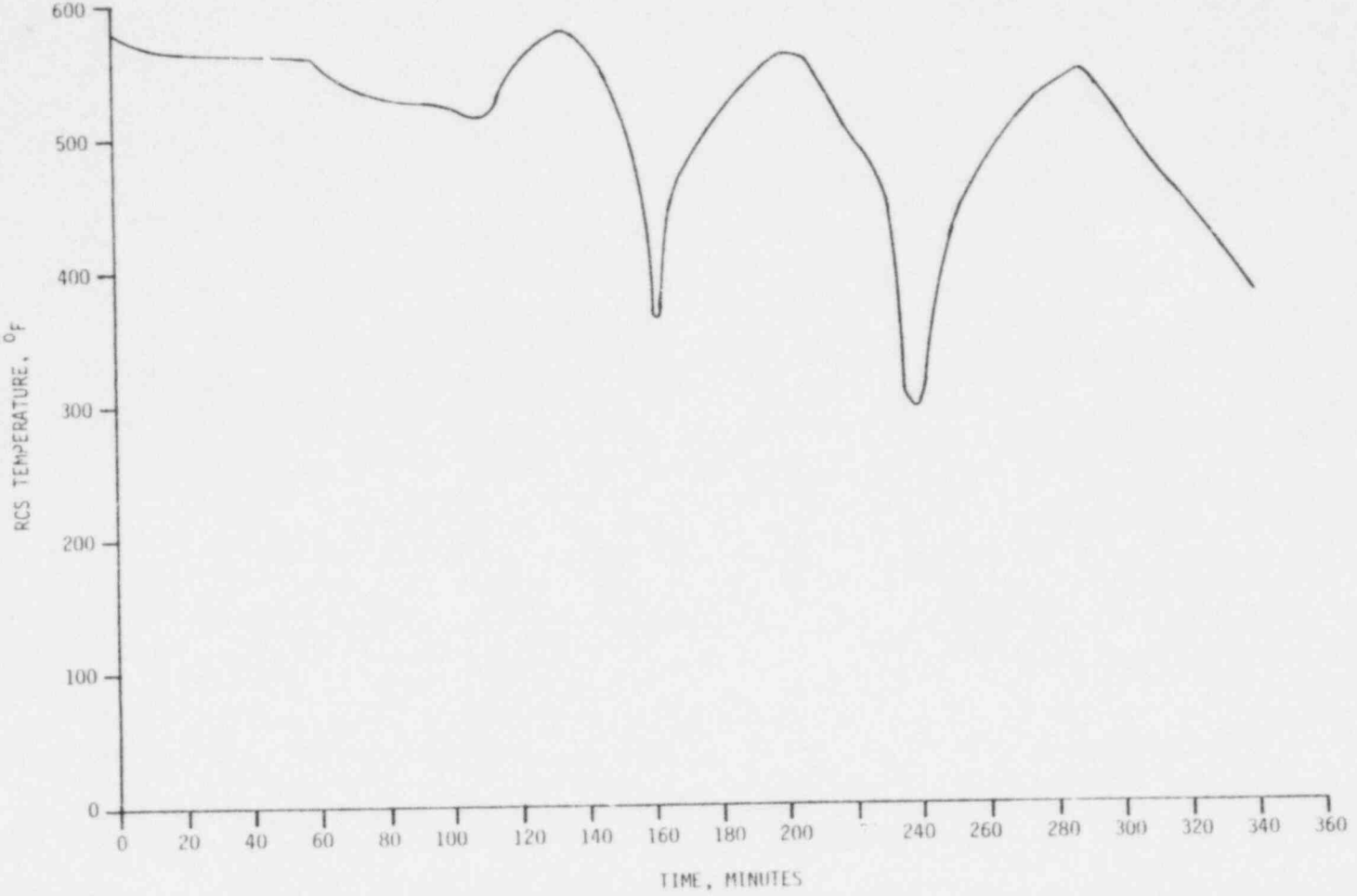

Figure A.4-10. Sequence S_2HF Reactor Coolant System Temperature vs. Time (2 inch Cold Leg Break)

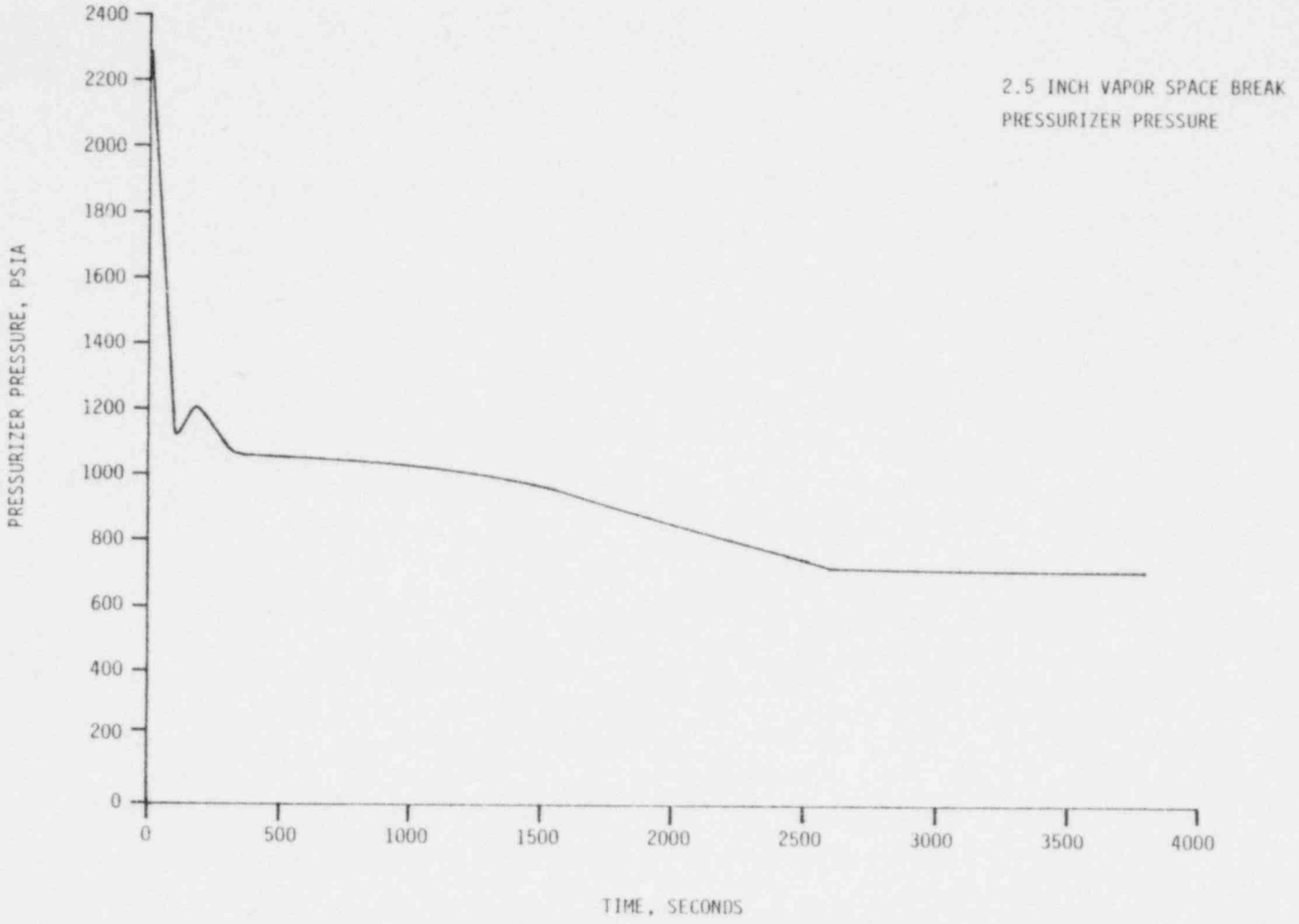

Figure A.4-11. 2.5 inch Vapor Space Break Pressurizer Pressure

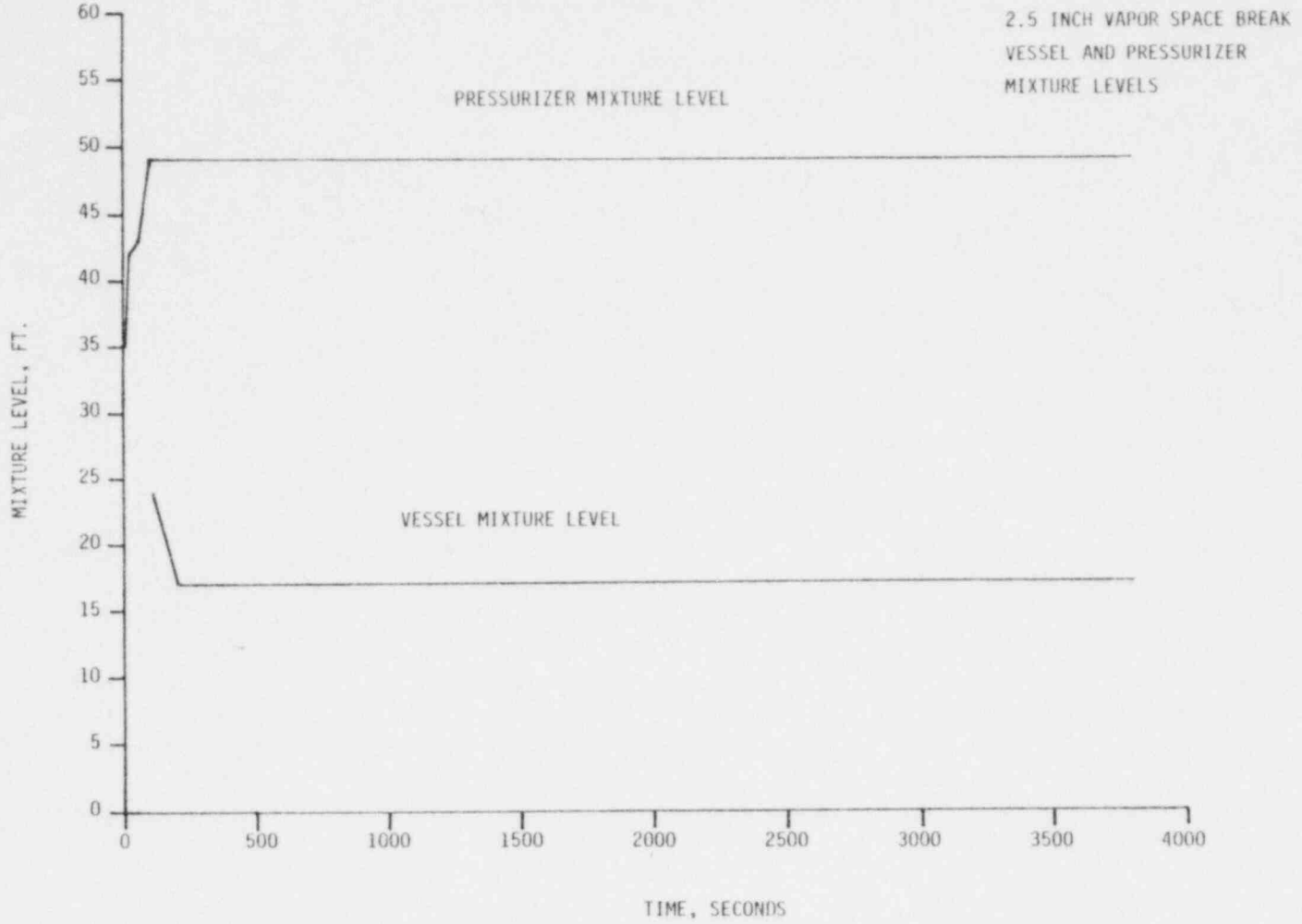

Figure A.4-12. 2.5 inch Vapor Space Break Vessel and Pressurizer Mixture Levels

Table A 4-1

SUMMARY OF KEY OPERATOR ACTIONS AND INFORMATION REQUIREMENTS FOR $S_1HF-\gamma$ SEQUENCE

Plant State	Description of Plant State	Information Required for Plant State Identification	Operator Action Following Plant State Identification	Information Required to Take Appropriate Action
1a	Small cold leg break (0.5" to 1" dia.)	o RCS pressure and temperature o Pressurizer and core water level o Containment pressure, temperature, humidity and radiation level o Charging pump flow/safety injection flow o Pressurizer relief tank pressure, temperature and level o Auxiliary feedwater flow rate and discharge pressure	Identify ESFs required for small break accommodation, ensure their readiness and verify correct ESF response. Manually actuate or control any system which does not function automatically. Possible termination of safety injection to prevent over pressurization.	o Status of key components in ESF systems o Parameters for state 2a identification
	Small cold leg break (1" to 4" dia.)	o Same as above o Upper head injection accumulator level and pressure		
	Small cold leg break (4" to 6" dia.)	o Same as above o Cold leg accumulator level and pressure		
	Vapor space break (Inadvertent opening of relief/safety valve)	o Same as above o Relief tank pressure, temperature and level o Relief/safety valve position indicators o Discharge line flow rate		

Table A.4-1
(continued)

Plant State	Description of Plant State	Information Required for Plant State Identification	Operator Action Following Plant State Identification	Information Required to Take Appropriate Action
2a	Reactor trip, high pressure ECI activated, AFWS activated for primary heat removal, ARFS activated for containment heat removal, reactor coolant pumps tripped	o Neutron flux, control rod position o RCS pressure and temperature o Steam generator water level, AFWS flow rate, pump discharge pressure o ECI flow rate, valve positions, pump discharge pressure o Fan discharge flow and differential pressure o RCP power	Control systems as required for effective ESF operation and accident accommodation	o Same as required for state identification
3b	Drain valve between upper and lower containment compartment left closed or plugged	o Containment sump water level o Upper compartment water level o Drain valve position indication	Restore flow between upper and lower compartments	o Same as for state identification
4a	Restore communication between upper and lower compartment	o Containment sump water level o Drain valve position indication o Upper compartment water level	Ensure proper alignment of valves and pumps for long-term heat removal	o Status of components in ECR and CSRS o Cooling water flow to the CHRS and LPRS heat exchangers o Sump water level o Containment pressure and sump water temperature
4b	Communication cannot be restored between upper and lower compartments	o Same as for state 4a	Delay core melt as long as possible and take other consequence mitigation actions to prepare for core melt	o RCS pressure and temperature o Core water level o Containment pressure and radioactivity level o RWST water level

Table A.4-1

(continued)

Plant State	Description of Plant State	Information Required for Plant State Identification	Operator Action Following Plant State Identification	Information Required to Take Appropriate Action
5a	Establish long-term cooling mode using ECR	o RCS pressure o Core outlet temperature o Core water level	Ensure proper alignment of valves and pumps for containment heat removal	o Status of key components in CSRS o Component cooling flow to CHRS heat exchangers
5b	Long-term cooling not established; eventual core melt	o RCS pressure and temperature o Core water level o Containment pressure, temperature and radioactivity level o HPIS and charging pump flow and discharge pressure o Primary coolant radioactivity level o RHR flow and discharge pressure	Monitor approach to core melt and initiate consequence mitigating actions	o Same as required for state identification
6a	Containment heat removal established using CSRS	o Containment pressure, temperature and radioactivity level o Sump water temperature o CSRS pump discharge flow and pressure	Monitor and control ECR and CSRS as required to maintain core coolable geometry and containment integrity	o Sump water level o Sump temperature and pressure o RCS temperature and pressure o Core water level o Status of components in ECR and CSRS
6b	Containment heat removal not established, ultimately results in containment failure and core meltdown	o Containment pressure and temperature o CSRS pump flow and discharge pressure	Delay core melt as long as possible and take necessary actions to mitigate consequences	o RCS temperature and pressure o Coolant activity level o Containment pressure, temperature and radioactivity level o Core water level

A.5 BWR TC SEQUENCE

A.5.1 Sequence Description

It is anticipated that a few times each operating year deviations of process parameters from normal values will occur that require rapid shutdown of the reactor to prevent fuel heat imbalances. The accident sequence to be addressed here is concerned with a failure to make the reactor subcritical (designated as event "C") following one of these anticipated transients (event "T"). Figure A.5.1 presents the BWR transient event developed in WASH-1400[4] for the Peachbottom BWR with the "TC" sequence highlighted.

A number of likely BWR Transient initiating events have been identified (WASH-1400 listed 15 such events) that would be applicable here. For this analysis, the "Loss of Feedwater Flow" initiating event has been selected. This particular event was chosen primarily because 1) it is probabilistically important, 2) most of the operator action required in response to this event would be identical to other transient events, and 3) the amount and quality of information available concerning the plant response to this event is greater relative to most other events.

A loss of feedwater flow could occur because of pump failures, feedwater controller, operator errors, or reactor system variables such as high vessel water level trip signal. Upon loss of feedwater, the vessel water level will begin to drop. Within a few seconds, the water level will be reduced to a point where a low level Scram actuation signal will be sent. Main steam line isolation will also be initiated on low water level. The water level will continue to drop to a low-low level at which point the Recirculation pumps trip and the High Pressure Core Spray (HPCS) and Reactor Core Isolation Cooling (RCIC) systems are initiated.

Failure to make the reactor subcritical would result in the following sequence of events, as described in the RSS:

...the reactor would tend to remain at relatively high power immediately following the transient. After steam flow to the turbine would be terminated due to the closure of the turbine stop valve or the main steam isolation valve, the reactor pressure would increase. This pressure increase would lead to a rise in power which, in turn, would further increase the primary coolant system pressure. The opening of the primary system relief and safety valves would limit the pressure increase; the initial peak pressure attained will be a function of the transient power history and the setpoints and capacities of the safety and relief valves. Recirculation pump trip combined with the loss of moderator through the relief and safety valves would tend to reduce the reactor power level. The power level would be expected to stabilize at about 30 percent of nominal. The HPCI system would start to add water to the primary system shortly after the initial pressure surge subsides. However, at power levels that are significantly above decay heating, the boiloff rate would be greater than the capacity of the HPCI; thus, the water level in the primary system would decrease and eventual core meltdown could be expected.

In the sections below the key operator actions associated with this sequence are delineated and the instrumentation which would provide the operator with the necessary and sufficient information to efficiently take these actions is identified. Actions designated to "fix" the initiating event (e.g., repair of feedwater pumps or use of condensate pumps for feedwater injection) were not specifically addressed because 1) this portion of the report was intended to focus on the failure to scram event and the associated operator actions and 2) the feasibility of such fixes is very uncertain and many aspects of the procedures would vary from plant to plant.

A.5.2 Operator Actions

Should the Reactor Protection System (RPS) fail to automatically make the reactor subcritical following the initiating transient event, the only remaining barrier to core melt is operator action. The operator must perform three basic tasks in order to prevent core melt: 1) Recognize the occurrence of the transient and the failure of the RPS, 2) Rapidly act to make the reactor subcritical, and 3) Ensure adequate vessel water inventory and heat transfer to the environment to bring sequence to successful termination.

Figure A.5-2 displays in a logic diagram format the relevant operator action events. This figure can be viewed as a version of the transient event tree (Figure A.5-1) which focuses on the key operator actions necessary to recover from the postulated failure events and bring the reactor to a safe shutdown condition. The important states to which the plant can evolve as the accident sequence progresses are enumerated on the logic diagram.

As seen in Figure A.5-2, system state 1 corresponds to the state of the plant immediately following the transient initiating event and state 2 indicates that the RPS has failed to automatically respond to the transient event. Since the reactor trip signal will be actuated on low water level within a few seconds following the occurrence of the transient event, there is no need to consider system states 1 and 2 separately with respect to operator action. The operator must at this point determine that a transient event has occurred which necessitates reactor shutdown and that the RPS has failed to automatically respond (i.e., he must determine that the plant is in state 2).

The operator must then initiate actions to manually shutdown the reactor and move the plant into state 3 .

There are two methods of Scram available to the operator should rapid automatic insertion of the rods fail: 1) Manual insertion of rods not successfully inserted automatically and 2) Operation of the Standby Liquid Control System (SLCS) in conjunction with tripped recirculation pumps. Operation of the SLCS must begin within 10 minutes after receipt of Scram signal and be complete (reactor subcritical) within 38 minutes. In order to utilize the SLCS, the operator must insert a key into a key switch to open all system valves and to start one of the pumps.

If he fails to accomplish this task, the plant will move into state 3a and core melt will inevitably occur. Should the plant be in state 3a the only useful action would be to monitor the approach to core melt and take the appropriate consequence mitigation actions.

Assuming successful attainment of state 3 , the operator must ensure adequate water inventory and heat removal capability to move into state 4 and state 5 and thus to successful termination of the accident. Failure to perform either task will result in a plant state (4a or 5a) which leads to coremelt.

For this loss of feedwater transient (and for many other transients) the HPCS and RCIC will normally be utilized initially to provide sufficient water inventory to the vessel and remove heat. Both the HPCS and RCIC systems will be automatically started upon receipt of an initiation signal from reactor low water level. The operator's role at this point is to verify that the systems are properly aligned for injection, sufficient water is available in the CST, power is available to the system, and the pumps properly start up.

When the normal water level is again reached, the HPCS system may be manually tripped and the RCIC system flow controller adjusted and switched to manual operation. The RCIC system will continue operation until the decay heat diminishes to a point where the RHR system can be

put into service. At this point, the operator will manually trip the RCIC system, turn the flow controller back to automatic, and close the steam supply valve to the turbine.

Initiation of the RHR in the shutdown cooling mode is performed manually. The system is initially flushed by opening local manually operated valves and prewarmed by opening vessel suction valves from the control room. Effluent from both the flushing and prewarming are directed to the radwaste system via valves all operated from the control room. When increasing temperature is noted at the RHR heat exchanger inlet, the radwaste effluent valves are closed, the RHR pump is started, and the service water flow is started. The cooldown rate is subsequently controlled via control valves in the main line and heat exchanger bypass line.

It is assumed in this analysis that the systems normally necessary to ensure adequate water inventory and heat removal following a transient event will be available when called upon. Sequences which involve independent failures of these systems are not considered to be probabilistically significant when combined with the failure-to-scram event. However, system states 4a and 5a can result from either 1) failure of the operator to take the necessary actions involved in the use of these systems, or 2) abnormal demands imposed on these systems because of a delayed scram which are not adequately handled by the operator or which simply exceed the capability of these systems regardless of operator action. Thus, the operator, in order to move the plant into states 4 and 5, must assess the state of the plant in state 3, translate these conditions into systems requirements for water inventory and heat removal, and take any necessary action to successfully operate these systems.

It is not clear at this stage of the analysis whether a delayed scram will impose demands upon the safety systems greater than those which exist following an immediate scram. If not, the appropriate operator actions at state 3 would be identical to the normal actions required of the operator following a transient and successful scram as described above.

A.5.3 Operator Information Requirements

In order for the operator to efficiently accomplish the tasks discussed in the previous section, he must be provided with the necessary and sufficient information to unambiguously determine the state of the plant as the accident progresses. With this information, he can identify the need for specific actions and be able to confirm the successful accomplishment of these required tasks. This information will be supplied to the operator by the plant instrumentation. It is the purpose of this section to identify the key plant parameters which can and must be measured to provide the operator with his informational needs.

Figure A.5-2 will again be utilized as a framework for this section. For each plant state enumerated in Figure A.5-2, the operator must be provided with the necessary and sufficient information to allow him to determine unambiguously that the specific state exists and to take the appropriate action corresponding to that state.

The first task of the operator is to recognize that the transient event has occurred and that the plant is (or has just passed through) state 1. The appropriate indication of this state will obviously depend upon the specific transient event. In addition, for many anticipated transients, it is not crucial that the operator be able to determine the exact cause of the transient; simply knowing that some type of abnormal event has occurred which necessitates a plant shutdown will be sufficient in many cases. However, knowing the specific nature of the initiating transient event could affect the efficiency of subsequent actions if the event involved systems which would be expected to respond to the initiating event (e.g., loss of power to safety systems). Therefore, the unambiguous determination of the specific initiating transient is considered necessary in this evaluation, although it is recognized that for some transient events information of a more general nature would be sufficient for the operator to take his required actions.

For the loss of feedwater transient, the most obvious indication would be a reduction in the vessel water level. Measurements of this level are, in fact, expected to initiate the scram signal. However, there are other transients which will also result in this level reduction, such as MSIV closure, steam line break, LOCA, etc. In order to differentiate the loss of all feedwater initiator from other events which result in reduction in vessel water level, additional information is necessary. One method to differentiate would be to measure the level reduction as a function of time. Since the reduction in water level over time will be somewhat different depending upon the specific initiator, each transient event has associated with it a unique level vs. time "signature" which could be used to identify the initiator. This method would only require measurements of vessel water level but is not considered to be totally adequate for the following reasons:

1) It would be necessary to have a high degree of confidence in the calculated level vs. time for all anticipated transients (many of which would be very similar)

2) Faults in the water level instrumentation could have contributed to the existence of state 2.

3) More direct indication of the cause of the transient is available by monitoring the status of components and systems associated with the anticipated transient.

For these reasons, an indication of the status of the feedwater pumps or feedwater controller should be sufficient to unambiguously identify the occurrence of a loss of feedwater transient when coupled with an indication of the rapid reduction in vessel water level. Causes for a loss of feedwater which might not be indicated by the status of the pumps or controller are either probabilistically not significant or are included in other identified transients which are not addressed here (e.g., feedwater LOCAs).

The loss of all feedwater will result in scram signal on low vessel water level within a few seconds. Should the rods fail to rapidly and automatically insert (state 2), the reactor will remain at a relatively high power level after the transient. Indication of the control rod position and neutron flux will be sufficient to allow the operator to determine that the plant is in this state and initiate manual actions to bring the plant to a subcritical state.

The operator would then attempt to insert the rods manually. Again, indication of rod position and neutron flux would be sufficient to allow him to take this action and determine the success or failure of his efforts. There is a significant probability that the cause of the failure to automatically shutdown will also prevent the operator from inserting the rods. In this case, the appropriate operator response is to initiate poison injection through the SLCS and trip the recirculation pumps (these pumps will automatically trip on low-low vessel water level at about 30 seconds after the initiator for most new plants).

The SLCS is typically designed to pump sufficient neutron absorber (boron) solution from a storage tank through either of two independent lines to shutdown the reactor and keep the reactor from going critical again as it cools. The SLCS is actuated by either of two key-locked switches on the control room console. Changing either switch to "run" starts an injection pump, actuates an explosive valve, opens a storage tank outlet valve, and closes reactor cleanup system isolation valves to prevent loss or dilution of boron. Indications of the storage tank liquid level, valve positions, and pump discharge pressure will provide the operator with sufficient information to determine the performance of the system. If any of these items indicates that the liquid may not be flowing, the operator may immediately change the other switch position to "run" thereby activating the redundant train of the SLCS. Measurements of the boron concentration in the core will indicate whether the solution being delivered is adequate to shutdown the reactor and indication of neutron flux will allow the operator to determine the success or failure of his actions and whether the plant has moved into state 3 or 3a .

It is possible that conditions could exist in the core as a result of this accident sequence (e.g., voids) which could produce unreliable neutron flux measurements. Therefore, the measurement of boron concentration takes on increased importance, and instrumentation which would allow a more rapid indication of boron concentration than that afforded by periodic sampling and analysis would be necessary.

State 3a will lead inevitably to core melt and the only beneficial action left to the operator at this point would be to delay melt as long as possible, monitor the approach to melt, and take any other consequence mitigation actions available. As discussed in Section A.5.1, state 3a is accompanied by a rise in reactor pressure (the MSIV is closed upon low water level) which would require the opening of the primary system relief and safety valves to prevent system overpressure. Loss of coolant through these relief and safety valves would be partially compensated for by high pressure coolant injection (which is initiated as low water level) although the boiloff rate would exceed the capacity of HPCI and core melt would eventually follow.

Information necessary to prevent primary system overpressure can be provided by indications of the safety and relief valve positions together with measurement of the RCS pressure. Monitoring the effectiveness of the HPCI system in cooling the core and delaying core melt can be accomplished by measuring the vessel coolant level. As a diverse backup, the pressure and temperature at appropriate positions in the core could be measured. Measurements of the radiation level in the coolant system would indicate the onset of fuel damage. Direct measurements of the HPCI status to enable the operator to ensure adequate injection can be obtained by monitoring the fluid flowrate, valve positions, current supplied to the pumps, or pump discharge pressure.

As noted in WASH-1400, following state 3a the containment would overpressurize due to steam generated during the boiloff phase and noncondensable gases generated during the melting phase. The status of the containment during the course of this accident can be determined by measuring the containment pressure. Containment temperature, radiation level and

hydrogen concentration can also be measured to assist the operation in monitoring the approach to containment failure. The Containment Spray Cooling System (CSCS) can be used to a limited degree of effectiveness in slowing down the containment pressure rise. Measuring the temperature of the water in the suppression pool, containment pressure, and the suppression pool level will provide the necessary information to the operator to determine if the CSCS will function under the given conditions. Measurements of the coolant flowrate or current to the pumps will indicate the operational state of the system and the containment pressure will indicate the effectiveness of the system.

If state 3 is achieved, through successful operator action, the operator must then bring the plant to a safe shutdown condition. If the delayed scram does not result in any abnormal plant conditions which would affect the performance of this task, the relevant procedure will be the use of the RCIC to maintain the necessary reactor water inventory to cool the core until the reactor vessel is depressurized sufficiently to allow the operation of the shutdown cooling function of the residual heat removal system (RHRS). This is equivalent to following sequence TQ in Figure A.5-1.

Although the RCIC system will automatically start, the operator must verify successful operation. Reactor water level, temperature, and pressure will indicate the effectiveness of the system in cooling the core. Direct indication of the status of the RCIC system can be obtained by monitoring the system valve positions (steam isolation valves, turbine exhaust isolation valves, flow controller, turbine throttle valve), steam flow to turbine, and pump discharge pressure.

A measurement of the reactor pressure will indicate when the RHRS can assume the heat removal function. Position indication of the valves required for flushing and prewarming the system will allow the operator to perform these start-up tasks. An indication of the RHR heat exchanger inlet temperature will provide the operator with the information required to start the RHR and service water pumps. Reactor pressure and temperature combined with indications of RHR control valve position will allow the operator to control the cooldown rate.

In this process of moving to state 4 and state 5 , it is assumed that the necessary plant systems will operate successfully. Sequences which involve failure of these systems combined with failure to scram are considered probabilistically insignificant.

A.5.4 Summary and Conclusions

In the preceding sections, the BWR "TC" sequence was evaluated with the purpose of identifying the instrumentation which will provide the necessary and sufficient information to the operator to allow him to determine unambiguously the state of the plant and to efficiently take the required corrective action as this sequence progresses. Presented in Table A.5-1, in summary form, are the results of this analysis. The presentation of these results is structured around the key plant states that could develop as the accident sequence progresses. These states are illustrated in Figure A.5-2. For each plant state, the following information is summarized:

- the information required to unambiguously determine that the plant is in that specific state
- the appropriate operator action at that state
- the information required by the operator to take this action

Following is a discussion of the key assumptions that went into the analysis and the major areas where further work is necessary to answer specific questions, confirm assumptions, reduce uncertainties, etc.

The information contained in the summary table is based on a number of assumptions concerning plant performance and the feasibility and effectiveness of specific operator actions. Since many of these actions take place under plant conditions which have not been extensively analyzed in the past, there is necessarily some uncertainty associated with these assumptions. Summarized below are the key areas where further work could be beneficially performed to either confirm uncertain assumptions, answer key questions, or reduce uncertainties to a level to produce a reasonable level of confidence in the conclusions of this analysis:

- The effect of a delayed scram following the initiating failure is very uncertain. The assumption that the demands upon the cooling systems under the conditions which would exist following a loss of feedwater and high power level for an extended period of time would be the same as the demands given rapid shutdown is difficult to accept without more supporting analysis. Different demands imply different operator actions and perhaps different information.

- The effectiveness of the SLCS in quickly reducing the power level is somewhat uncertain. This would depend upon the mixing capacity of the coolant in the core and could be affected by the specific initiator and timing of the SLCS initiation. The important effects that this question has on the present analysis are 1) the time allowed for operation initiation, 2) the reliability of SLCS status monitors as indications of shutdown; if the effectiveness of SLCS is highly uncertain, the operator is limited to flux monitors for inidcation of shutdown, and 3) the effectiveness of using boron concentration measurements; if this must be done by sampling, the time allowed might not be sufficient.

- In many instances it was stated that reactor pressure, temperature, and water level would provide sufficient information to the operator. However, the reliability and usefulness of this information often depends upon the location of the instrumentation in the core. This is especially important with regard to in-core temperature monitors for a BWR which operates under saturated conditions. Additional analysis which would provide a more detailed picture of the core as the accident progresses is needed before the significance of the instrument location can be determined and the optimum locations identified.

- More detailed information concerning plant states is necessary to establish the necessary ranges for the instrumentation.

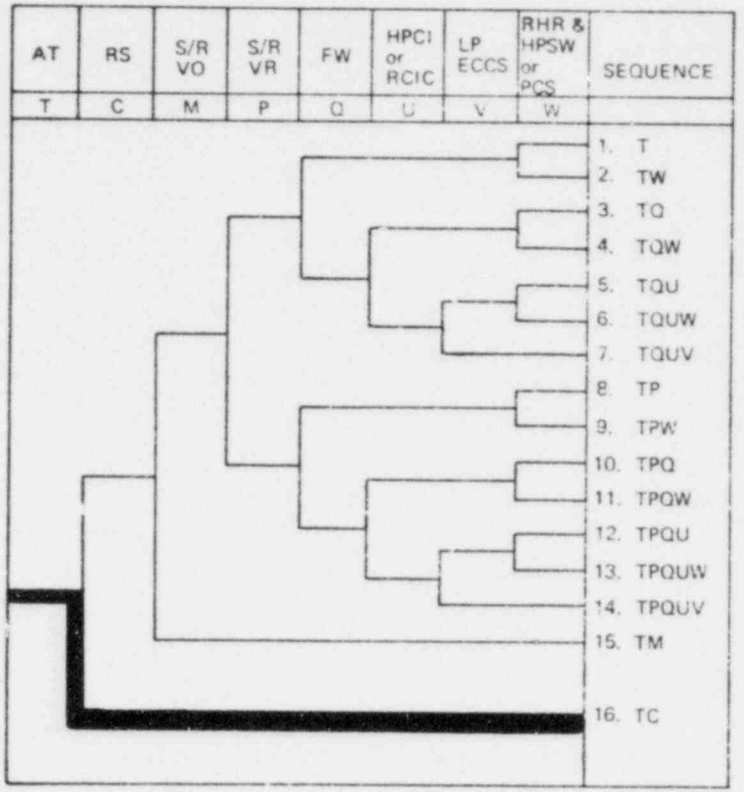

Figure A.5-1. BWR Transient Event Tree (from WASH-1400)

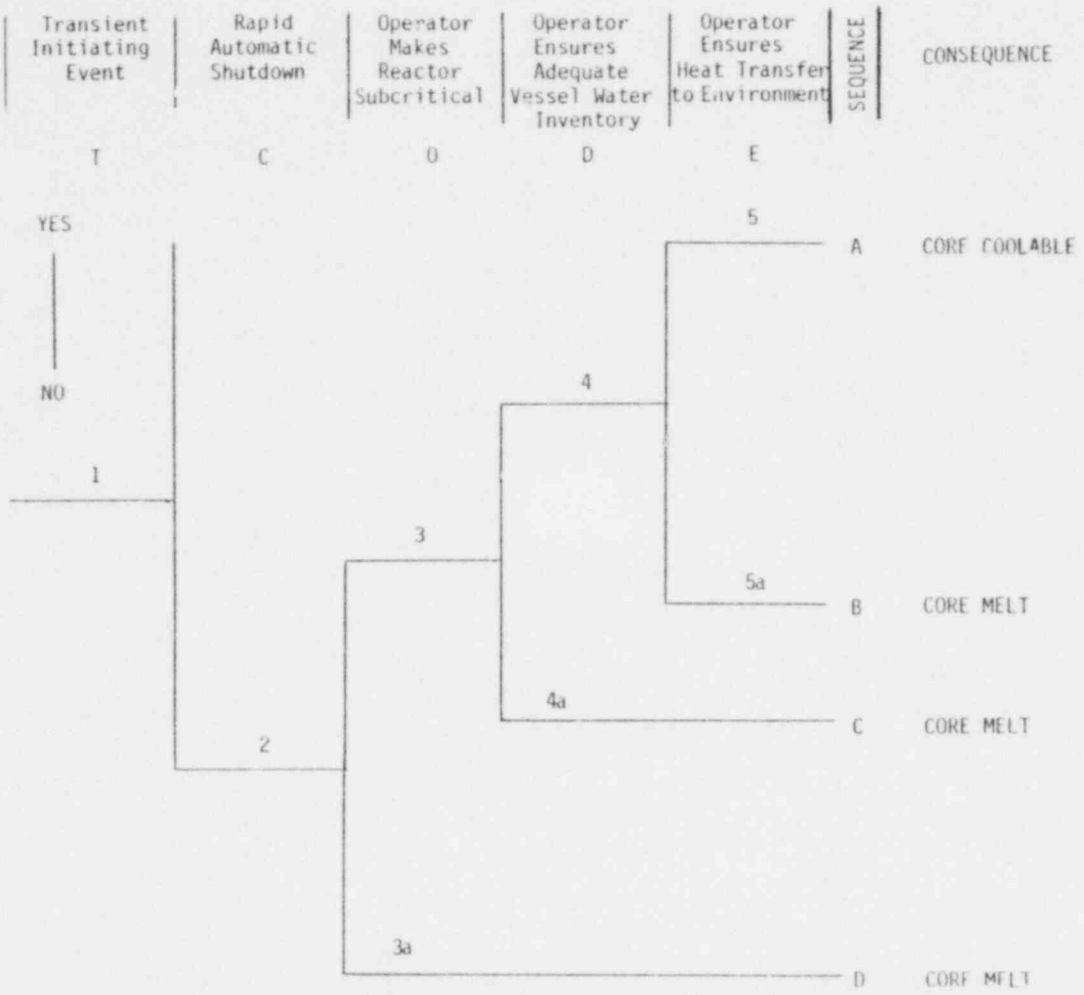

Figure A.5-2. Operator Action Event Tree

Table A.5-1

SUMMARY OF KEY OPERATOR ACTIONS AND INFORMATION REQUIREMENTS FOR TC SEQUENCE

Plant State (See Fig. A.3-2)	Description of Plant State	Information Required for Plant State Identification	Appropriate Operator Action Following State Identification	Information Required to Take Appropriate Action
1	Loss of all Feedwater Transient event has occurred. Water Level in vessel dropping. Low water level initiates scram MSIV closure.	Vessel water level MSIV position Feedwater flowrate Current to FW pumps Feedwater controller position	Prepare for actio s illustrated in Fig. A.5-2	See states 2, 3, 4, and 5
2	Failure of the RPS to automatically shutdown reactor upon receipt of low water level signal	Control rod position Neutron flux	Manually shutdown reactor	Control rod position Neutron flux Boron tank level SLCS valve position SLCS pump discharge pressure Boron concentration
3	Reactor Manually Shutdown Delay could result in RCS pressure rise limited by safety and relief valves; HPCI and RCIC initiated on low-low vessel water level	Neutron flux RCS p,T Vessel water level Safety/Relief valve position	Ensure HPCI and/or RCIC operation until RHR is capable of long-term heat removal	See states 4 and 5
3a	Failure to Manually Shutdown Reactor. Reactor stabilized at ~ 30% power. Boiloff rate exceeds HPCI	Same as 3	Monitor Approach to core melt; delay melt and other consequence mitigation action	RCS P,T,R Vessel water level Containment P,T,R Suppression pool T
4	Reactor successfully shutdown with HPCS and/or RCIC providing cooling water.	Vessel water level RCS P,T RCIC Valve positions Steam flow to RCIC turbine RCIC pump discharge P HPCS valve positions HPCS pump discharge P Current to HPCS pump(s)	Monitor reduction in decay heat level in anticipation of securing first the HPCS and then the RCIC when RHRS can provide long-term cooling	RCS P,T Vessel water level
4a	Failure to provide adequate water inventor to cool core after shutdown	Sames as 4	Same as 3a	Same as 3a
5	Successful Transition to long-term heat transfer to the environment via RHRS; successful termination of accident sequence	RCS P,T Vessel water level Position of RHR valves required for flushing and prewarming RHR heat exchanger inlet/outlet temperature RHR control valves positions HPSW valve position HPSW pump discharge pressure		
5a	Failure to provide long-term heat removal	Sames as 5	Same as 3a	Sames as 3a

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*Available free upon written request to the Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.

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A novel technical approach for systematically determining information needs during reactor accidents is proposed. The method is used to identify the necessary and sufficient set of Light Water Reactor instrumentation by analyzing the appropriate operator response to specific plant states associated with risk significant accident sequences. The resultant set of measurable parameters is compared to the list of such parameters in Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident."

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