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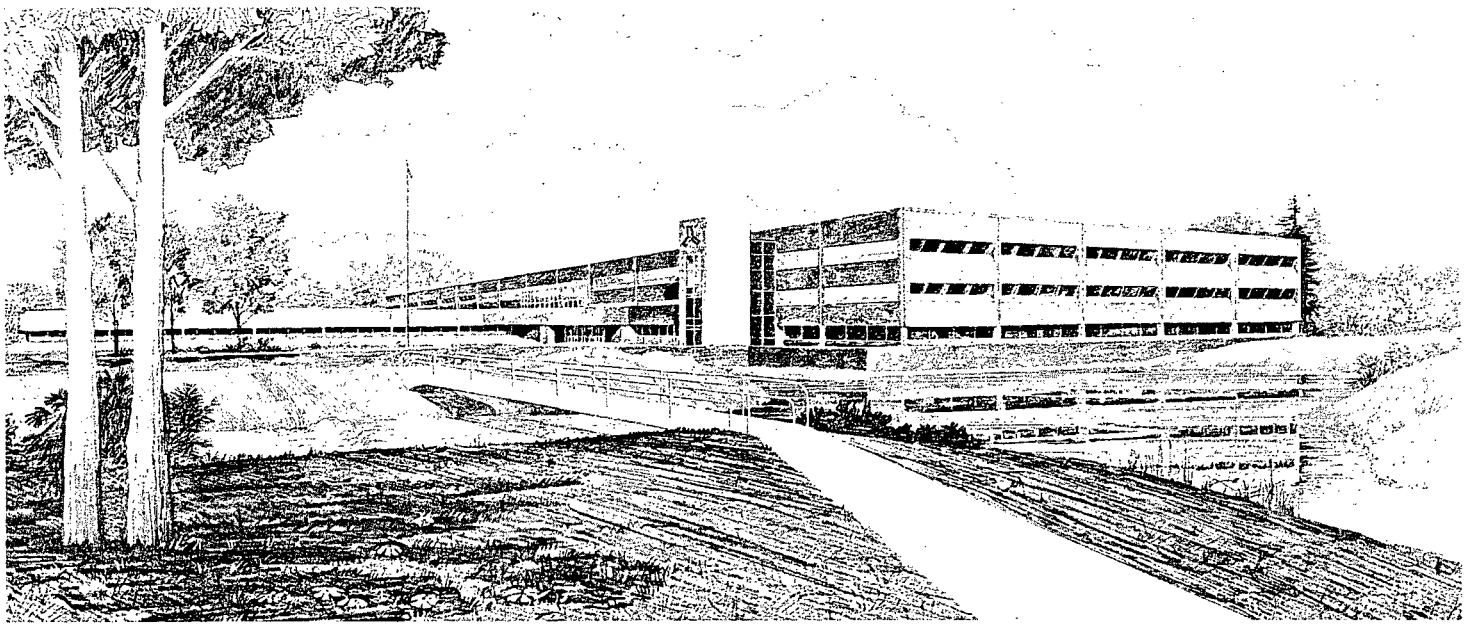

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Methods For Review and Evaluation of Emergency Procedure Guidelines Volume III: Applications to General Electric Plants

Wesley A. Brinsfield
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METHODS FOR REVIEW AND EVALUATION OF EMERGENCY PROCEDURE GUIDELINES VOLUME III: APPLICATIONS TO GENERAL ELECTRIC PLANTS

Wesley A. Brinsfield
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ABSTRACT

Systematic methods for finalizing, reviewing, or developing improved emergency procedure guidelines are applied to a representative General Electric BWR plant. The methods are based on the use of operator action event trees (OAETS) which document the key operator actions and plant symptoms associated with the various stages of risk significant multiple failure accident sequences. The application of the methodology utilizes OAETS developed for a BWR-4 and Mark I containment and the BWR Owners' Group Emergency Procedure Guidelines (Revision 2). Those aspects of General Electric plant design, operation, or response to multiple failure accident sequences which could result in incomplete, ambiguous, or incorrect guidance to the operator if not carefully addressed in the plant-specific guideline development or utilization process are identified.

SUMMARY

In previous projects performed under the Nuclear Regulatory Commission's Plant Status Monitoring Program (Refs 1,16), it has been demonstrated that Operator Action Event Trees (OAETs) can provide a systematic tabulation of the key operator actions and plant symptoms associated with the various stages of risk significant multiple failure accident sequences. Volume 1 of this report presented methodologies by which the information documented in these OAETs can be used to systematically review and evaluate functional emergency procedure guidelines and ensure that they provide unambiguous guidance under all important accident conditions. Volume II presented the results of such a review and evaluation as applied to a Westinghouse PWR plant design.

In this volume, the ability of these OAET-based methodologies to review, evaluate and produce effective guidelines applicable to a General Electric BWR plant design is investigated. This investigation takes the form of a review and evaluation of the BWR Owners Groups's Emergency Procedure Guidelines (Revision 2). The primary goal of this methodology application is the identification of those aspects of General Electric plant design, operation, or response to selected multiple failure accident sequences which could result in incomplete. ambiguous, or incorrect guidance to the operator if not carefully addressed in the plant-specific procedure development or utilization process.

The OAET techniques demonstrate the feasibility of the methodology as applied to the BWR symptom-based procedures and establish the adequacy of the EPGs for all states examined. However, while this evaluation covers a broad spectrum of potential BWR risk-producing sequences, the analysis is not intended as a complete evaluation of all possible accident sequences. The end states of the analysis included stable, hot shutdown and various potential core or containment challenges. The examination of the adequacy of the EPGs beyond core melt or containment failure are not performed.

While it was not the purpose of this analysis to pass judgement on the guidelines developed by the BWR Owners Group, it is appropriate to note that the results of the application of the guideline review methodology suggest that these guidelines can provide an effective model for the development of plant-specific technical guidelines and emergency operating procedures. Specific technical points which should be considered in the development of such plant-specific procedures ensure that BWR operators are provided with efficient unambiguous guidance under important accident conditions as delineated in this study.

CONTENTS

LIST OF ACRONYMS

AC	Alternating Current
ADS	Automatic Depressurization System
ARI	Automatic Rod Insertion
ATWS	Anticipated Transient without Scram
BWR	Boiling Water Reactor
CRD(S)	Control Rod Drive (System)
CS	Core Spray
CST	Condensate Storage Tank
DC	Direct Current
ECCS	Emergency Core Cooling System(s)
ECI	Emergency Coolant Injection
EPG	Emergency Procedure Guidelines
FW	Feedwater
GE	General Electric
HP	High Pressure
HPCI(S)	High Pressure Coolant Injection (System)
HPSW	High Pressure Service Water
LOCA	Loss of Coolant Accident
LOFW	Loss of Feedwater
LOMC	Loss of Main Condenser
LOSP	Loss of Offsite Power
LP	Low Pressure
LPCI(S)	Low Pressure Coolant Injection (System)
LPCR(S)	Low Pressure Coolant Recirculation (System)

MSIV	Main Steam Isolation Valve
OAET	Operator Action Event Tree
PCS	Power Conversion System
RCIC(S)	Reactor Core Isolation Coolant (System)
RHR(S)	Residual Heat Removal (System)
RPV	Reactor Pressure Vessel
SBO	Station Blackout
SASA	Severe Accident Sequence Analysis (Program)
SLC(S)	Standby Liquid Control (System)
SORV	Stuck Open Safety/Relief Valve
SRV	Safety/Relief Valve
SW	Service Water
TMI	Three Mile Island
RWCU(S)	Reactor Water Clean-Up (System)

Section 1

INTRODUCTION AND BACKGROUND

1.1 BACKGROUND

In Volume 1 of this report, a review of the post-TMI industry directions in the development of emergency procedures was presented. A discussion of the approaches currently being taken by groups representing each of the four major U.S. vendors in the development of improved procedures was provided. Each of these groups has recognized inherent deficiencies in the pre-TMI procedures which required operator diagnosis of specific events. These groups have all turned to function- or symptom - oriented approaches which focus on only a few key symptoms of critical safety functions as the bases for operator guidance under emergency conditions.

In that volume, it was concluded that these functional or symptom based approaches are effective in avoiding many of the problems inherent to the pre-TMI event specific procedures. However, it was also pointed out that the complex interactions of realistic plant response to multiple failure accident sequences often result in many different accident conditions looking the same to the operator, especially if his attention is focused on a relatively limited set of symptoms. This situation could give rise to ambiguous guidance, operator confusion, and aggravation of the upset condition. It was this concern that led to the conclusion that there is a need to 1) identify those accident conditions where this potential ambiguity exists and 2) develop and review emergency procedures in such a way that such potential problems are systematically addressed.

Volume 1 presents a methodology to systematically identify those diverse accident conditions which, because they exhibit common or similar symptoms, may result in ambiguous operator diagnosis and ineffective response. Methodologies are presented to review and evaluate function based emergency procedures guidelines and ensure that they provide unambiguous guidance under all important accident conditions.

These methods are based on the use of Operator Action Event Trees (OAETs) which systematically delineate the required operator actions and key plant symptoms throughout the progression of important accident sequences. These techniques are applicable to any accident scenario whether high risk or high frequency. (See Reference 1 for a basic description of OAETs).

1.2 SCOPE AND LIMITATIONS

In this volume, these OAET-based methods are used to examine the BWR Emergency Procedure Guidelines developed by the BWR Owners' Group.[2] There are three general objectives to this analysis:

1) To demonstrate that the OAET-based methodology is feasible to systematically examine existing procedure guidelines when applied to guidelines of the type and style of the BWR Owners' Group Emergency Procedure Guidelines.

2) To identify and highlight some important aspects of BWR plant response to multiple failure accident sequences which might give rise to incomplete or ambiguous guidance (areas of concern) if not carefully addressed in the process of producing plant specific procedures from the generic guidelines.

3) To suggest general methods of how a specific utility might effectively address any identified "areas of concern" in the generic guidelines and remove the potential ambiguities from their plant specific procedures.

The Operator Action Event Trees which will be used to achieve the above goals are developed for a representative BWR-4 with a Mark I containment. The specific accident conditions addressed in this analysis and the basis for their selection are described in Section 4. Best-estimate computer analyses generated under the Nuclear Regulatory Commission's Severe Accident Sequence Analysis (SASA) Program (References 3, 4 and 5) are used to determine system response to some accident conditions. Plant response calculations generated by General Electric and submitted to the NRC staff following TMI are also used to determine plant symptoms at many states (Reference 6). In addition, actual operating experience (Reference 7) is used when appropriate to determine plant response.

In most evaluations and analysis there are limitations on the completeness and depth of the analysis which can be performed. The systematic application of OAET techniques provides a valuable exercise of the BWR EPGs for explicitly defined risk-significant sequences. While this is useful and important, it does present some limitations on the conclusions which can be drawn. The following groundrules and limitations should be clearly understood in order to correctly interpret the scope of the Operator Action Event Tree Analysis presented here.

- The OAET evaluation is based upon a selected group of potential risk-significant accident sequences for BWRs. As such, this evaluation can demonstrate the feasibility of the OAET methodology as applied to BWR symptom-based procedures and can establish the adequacy of the BWR EPGs for all states examined. While this is intended to cover a large portion of the spectrum of BWR risk-producing accidents, this report is not intended as a complete evaluation of all possible accident sequences.

- The operator actions, symptoms, and emergency procedures are carried through the analysis for all identified accident sequences to one of the following states:

  a) stable, hot shutdown

  <u>OR</u>

  b) a core damage condition

  <u>OR</u>

  c) a containment failure state

  The examination of the adequacy of the EPGs beyond these states is not included in the current analysis.

- Revision 2 of the BWR OWNER'S GROUP EPG (Reference 2) is used as the basis for the evaluation of transition from generic to plant-specific. Revision 2 does not include a Secondary Containment Guideline; therefore, unique sequences affecting secondary containment are excluded from the OAET analysis presented here.

- The duration of the mission time for the identified risk-significant sequences is the time required for the operator to address all operator actions necessary for a safe plant shutdown.

- Operator errors of commission in the use of the EPGs are not included unless they are induced by ambiguities in the EPGs, the accident sequence, or a combination of both.

- Most of the dominant accident sequences identified in current PRAs and examined in this analysis involve states in which adequate instrumentation and control-room indication of key parameters are available to the operator. Lower-frequency accident sequences have been postulated which may obscure operator perception in the control room. However, these sequences have not been identified as dominant and are not specifically addressed in this analysis.

- The analysis is performed for a "typical" plant, and therefore there may be unique plant features which are not specifically addressed in this analysis.

1.3 REPORT ORGANIZATION

In the following section, the OAET-based methodologies to systematically examine existing emergency procedure guidelines are summarized.

In Section 3, the BWR Emergency Procedure Guidelines are discussed and represented as instructions which associate a particular set of actions with the observance of a specific set of symptoms. The version of the BWR Emergency Procedure Guidelines utilized in this assessment was Revision 2, submitted to the NRC in May 1982. Much of the information provided in these guidelines is summarized in Section 3.

In Section 4, the OAETs are developed for a variety of postulated accident conditions. The accident sequences selected and the basis for their selection are described. These OAETs provide the logical framework for delineating the required operator actions and plant symptoms throughout the progression of the accident sequences.

In Section 5, the OAET-based guideline examination methodology is applied to the BWR Owners' Group Emergency Procedure Guidelines (Revision 2).

In Section 6, a summary discussion is provided of the important aspects of BWR physical response to multiple failure accident sequences which were identified by the OAET method as possible areas of ambiguity and, as such, should be carefully addressed in the process of producing plant specific procedures from the generic guidelines. Additionally, mechanisms by which these identified areas of concern can be (and have been) effectively addressed are discussed.

Section 2

SUMMARY OF METHODOLOGY

Volume 1 of this report presented systematic methodologies to examine emergency procedure guidelines using Operator Action Event Trees. In this section, these methodologies will be briefly summarized and presented in terms of the BWR Owners' Group Emergency Procedure Guidelines and General Electric plant design. Volume 1 should be referenced for more detailed discussion of these methodologies.

Emergency procedure guidelines can be viewed as a collection of instructions, each of which relates a "symptom set" to an "action set". For example, one instruction might be in the form:

> "when you observe Symptom Set A (comprised of symptoms a_1, a_2, a_3), take Action Set P (comprised of actions p_1, p_2, p_3)."

The review process entails asking four basic questions regarding these instructions:

1) Is the Guidelines' collection of symptom sets complete? That is, are there risk significant states requiring operator action which could occur but for which entry into the EPGs is not accomplished?

2) Are the instructions in the EPGs always right? That is, if the EPG says "when you see symptom set A take action set P" are the actions associated with action set P always appropriate for every situation that can produce symptom set A?

3) Are the EPG action sets always complete? That is, are there important actions which should be carried out at a particular state which are not included in the EPG steps indicated at that state?

4) Are the instructions in the EPGs always unambiguous? Are there plant states which produce symptoms sets which the operator might confuse with EPG symptom sets that guide the operator to inappropriate actions?

These four questions can be answered by performing the systematic OAET-based symptoms comparison outlined in Figure 2.1.

As depicted in Figure 2.1, the input information is a description, for each key plant state identified in the OAETs, of the symptoms exhibited by the plant at that state and the necessary operator actions associated with that state. These plant states address both the relatively high frequency initiating events and the much less probable (but higher risk) multiple failure accidents.

In Step #1, attention is focused on the BWR Owners' Group's Guidelines and the goal is to translate these Guidelines into a collection of instructions, each of which relates a well defined symptom set to an action set. This step is discussed in detail in Section 3.

In Step #2, attention is focused on the OAETs. For each OAET state, the behavior of each of the EPG parameters listed in Step #1 should be tabulated. For some of these states, the behavior of some of the parameters may be uncertain. In these cases, several different symptoms (e.g. pressure rising, pressure stable) may be assigned to the same state. If any of these symptoms is later found to result in potential ambiguities, that particular state and symptom can be looked at more closely. These tabulated symptoms will be used as input to the systematic comparison steps discussed below. This step is documented in Section 4.

In Step #3, the comparison process is performed. For each OAET state, any and all EPG symptom sets listed in Step #1 which are completely produced are identified. The EPG action sets associated with these symptom sets are also identified.

In Step #4, any OAET states which do not completely produce any Guideline symptom set are identified.

The information generated in the above four steps can be used to systematically address the four basic questions listed above.

The first question - Is the collection of EPG symptom sets complete? - can be directly addressed using the results of Step #4 which identifies any OAET state which does not produce any symptom set in the collection.

The second question - Are the EPG instruction always right? - can be addressed using the results of Step #3. For each OAET state, there will be one or more EPG action sets identified in Step #3. The indicated EPG action sets must be compatible with each other and they must be compatible with the actions associated with that OAET state.

The third question - Are the EPG action sets always complete? - can also be addressed using the results of Step #3. There may be important actions which must take place which are indicated in the OAET but are not included in the indicated EPG action set. It should be recognized here that functional guidelines are not necessarily intended to provide all the detailed steps required to bring the plant to a safe shutdown condition, but, rather, are focused on those actions which will restore the critical safety function. Accordingly, this third question should also focus on actions related to restoration of critical safety functions.

The fourth question - Are the EPGs always unambiguous? - can also be addressed using the results of Step #3. The symptoms associated with each OAET state (in Step #2) should include all symptoms which the operator might observe at each state. If a parameter is not precisely known at a state, then multiple (even contradictory) behaviors should be indicated. If a parameter is going through a transition at a particular state, all possible behaviors of that parameter should be indicated at that state. If it is believed that the operator will not be clearly able to differentiate between two (or more) symptoms at a particular state then both (or all) symptoms should be listed for that state. The result of tabulating symptoms in this way is a listing of all conceivably observed symptoms at each state. Each OAET state will then have associated with it (as a result of Step #3) all action sets which the operator could conceivably think that he should perform. The appropriateness of each of these action sets can then be examined. If the action is found to be inappropriate at that state, or incompatible with other actions at that state, then the symptom set which directed the operator to the inappropriate actions

can be more carefully examined. If the postulated plant state can, indeed, realistically produce symptoms which the operator could "observe" and thereby take inappropriate action, this becomes an "area of concern".

The comparison process and the associated examination of the completeness and clarity of the BWR EPGs are presented in Section 5.

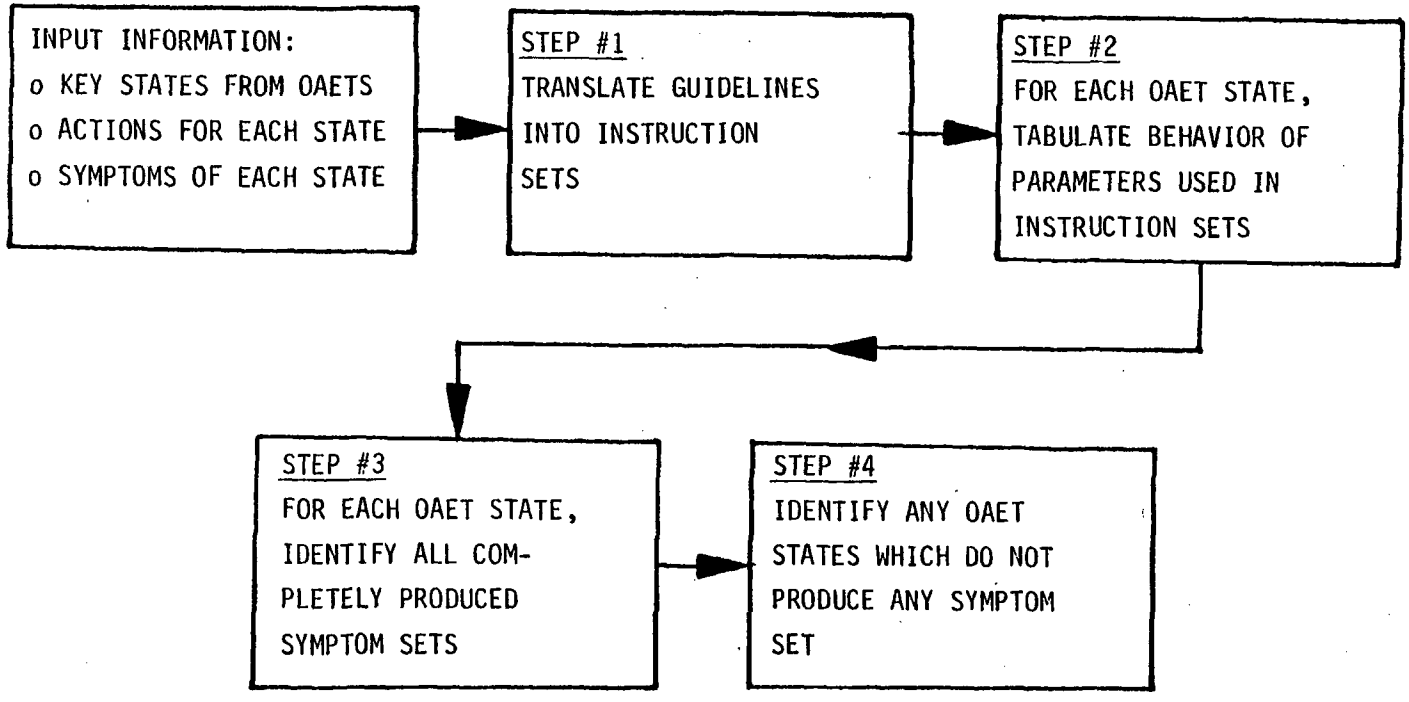

Figure 2.1. Emergency Procedure Review Flowchart for PSM Approach

Section 3

BWR EMERGENCY PROCEDURE GUIDELINES

In this section, the BWR Emergency Procedure Guidelines produced by the General Electric Owners Group are described and summarized.

The general structure of the guidelines is presented below. These guidelines are then translated into descrete instruction sets which associate one or more actions (an "action set") with the observance of a particular set of parameter behaviors (a "symptom set"). These action sets and symptom sets will be used in Sections 4 and 5 to examine and evaluate the guidance provided by the BWR EPGs under a variety of postulated accident conditions

3.1. General Outline of Guideline Structure

The BWR emergency procedure guidelines (Revision 2) are organized into two sections with separate entry conditions:

- o RPV Control Guideline
- o Containment Control Guideline

The purpose of the RPV Control Guideline is to provide guidance in those situations in which RPV water level must be restored and maintained within a satisfactory range, RPV pressure must be controlled, the reactor must be shutdown, and the RPV must be cooled down to cold shutdown conditions. The conditions for entry to this guideline are 1) low RPV water level, 2) high drywell pressure, 3) isolation, or 4) any situation requiring scram in which reactor power is either above the APRM downscale trip power (3%) or unknown.

The RPV Control Guideline is divided into three sections which are to be entered simultaneously if any of the entry conditions exist. The three sections include steps that address control of RPV water level, RPV pressure, and reactor power, respectively. Steps within these sections of the RPV Control Guideline are labelled with the letters RC/L, RC/P, and RC/Q, respectively. The

first step in the RPV level control section, for example, is designated as step RC/L-1.

The Containment Control Guideline provides guidance in controlling primary containment temperatures, pressure, and level. The conditions for entry to this guideline are 1) high suppression pool temperature, 2) high drywell temperature, 3) high containment temperature (relevant to Mark III containment only) 4) high drywell pressure, 5) high suppression pool water level, or 6) low suppression pool water level.

The Containment Control Guideline is divided into five sections to be entered simultaneously upon occurrence of any of the entry conditions. These five sections include steps that address control of suppression pool temperature, drywell temperature, containment temperature (Mark III only), primary containment pressure, and suppression pool water level, respectively. The steps within the sections of the Containment Control Guideline are labelled with the letters SP/T, DW/T, CN/T, PC/P and SP/L, respectively. The first step in the suppression pool temperature control section, for example, is designated as step SP/T-1.

Within the guideline sections, the steps call for different actions when plant parameters behave in certain ways (i.e., when the plant displays certain symptoms). Additionally, the emergency procedure guidelines include seven contingency procedures. Various steps in the RPV Control Guideline and the Containment Control Guideline call for branching to one of these contingency procedures. Steps within one contingency procedure may call for transfer to another contingency procedure. The seven contingency procedures are:

1. Level Restoration
2. Emergency RPV Depressurization
3. Steam Cooling
4. Core Cooling Without Level Restoration
5. Alternate Shutdown Cooling
6. RPV Flooding
7. Level/Power Control

Steps in the contingency procedures are designated by the contingency number and step number. The first step of contingency #2, for example, is designated as step C2-1.

The guideline steps are intended to provide appropriate guidance for BWR-1 through BWR-6 designs. The introduction to the guidelines states that the guidelines address all major systems that may be used in an emergency, but no one plant includes all of the systems in the guidelines. Those portions of the guidelines that call on systems not included in a particular plant may be omitted. The remaining portions are intended to comprise valid guidelines for that plant.

3.2 Concept of Symptom/Action Sets

The first step in this emergency procedure guidelines (EPG) evaluation was the representation of the EPGs as a collection of instructions. The instructions consist of a set of actions and a set of symptoms. To represent the EPGs as a collection of action sets and symptom sets, we put each EPG step (or a group of steps) in the form:

> If symptoms S_1, S_2, ..., S_m are exhibited, perform actions A_1, A_2, ..., A_n.

As an example, consider step DW/T-1 of the Containment Control Guideline, which reads:

> "When drywell temperature exceeds [135°F (drywell temperature LCO or maximum normal operating temperature, whichever is higher)], operate available drywell cooling.

This step translates into a simple instruction of the form (S_1, ..., S_m; A_1, ..., A_n) with a single symptom and a single action:

> S_1 = "Drywell temperature exceeds drywell temperature LCO or maximum normal operating temperature, whichever is higher."
>
> A_1 = "Operate available drywell cooling."

The task of translating the entire EPG into symptom and action sets was not as simple as this example might indicate. Often, the EPG steps direct the operator to a series of subsequent steps upon occurrence of a set of symptoms. Some of these subsequent steps may involve branching to another series of steps, and so on. In order to untangle this web of paths and separate the guidelines into symptom and action sets, the "Operator Actions Flowchart" was used, which was prepared by GE to illustrate the flow of operator actions within the EPGs. The flowchart allowed grouping of several EPG steps into a single action set, and grouping of symptoms mentioned in different EPG steps in a single symptom set.

For example, step RP/L-2 of the RPV Control Guideline directs the operator to branch to contingency 1 (Level Restoration) if RPV water level cannot be maintained above the top of active fuel. The third step (C1-3) of the contingency calls for different actions depending on whether the RPV level is increasing or decreasing and whether RPV pressure is high (J425 psig), intermediate (between 100 psig and 425 psig), or low (0100 psig). If RPV level is decreasing and RPV pressure is low, step C1-3 directs the operator to perform the actions of step C1-8. Part of step C1-8 states:

> [If no HPCS or LPCS subsystem is operating, start pumps in alternate injection subsystems which are lined up for injection.]

We may translate this path of EPG steps into the following symptom set and action set:

S_1 = RPV level below top of active fuel

S_2 = RPV level decreasing

S_3 = RPV pressure less than 100 psig

S_4 = No HPCS or LPCS operating

A_1 = Initiate IC (C1-1).

A_2 = Line up for injection and start pumps in 2 or more of the injection systems: condensate, HPCS, LPCI, LPCS (C1-2)

A_3 = Monitor pressure and water level (C1-3).

A_4 = Start pump in alternate injection subsystems which are lined up for injection (C1-8).

The EPG steps which specify each of the four actions are indicated in parentheses above.

3.3 Parameters and Symptoms

Table 3.1 lists the symptoms used to define the EPG symptom sets for comparison with OAET state symptom sets. The measurement of a plant parameter, such as level, temperature, or pressure, defines many symptoms. The status of plant systems, subsystems, and components defines other symptoms. The project analysts compiled an initial list of symptoms and parameters by listing every reference in the EPGs to a plant parameter or system, subsystem or component status. This initial list was pruned to remove symptoms not relevant to plants covered by the OAETs. They also removed symptoms if the EPGs did not direct significant actions when the symptoms existed. This revised list was used for the OAET comparisons in Section 4.

3.4 EPG Symptom/Action Sets

The matching symptoms and actions sets from the EPG steps were created as described above. Table 3.2 lists the associated symptom and action sets, along with the EPG steps that comprise each action set. For example, the observance of symptom (2), RPV Level below Low Level Scram Setpoint (from Table 3.1), will direct the operator to take EPG steps RC-1, RC/L-1, RC/L-2, RC/P-1, and RC/P-2a, which are collectively called Action Set A1. Similarly, the observance of Symptoms 7, 10, and 12 will direct the operator to take Action Set A1. As the table shows, more than one symptom set is often associated with a single action set. Occasionally, a single symptom set is associated with more than one action set. The symptom numbers correspond with those in Table 3.1. The action sets are labelled with a letter and a number. Those labelled with the letter "A" consist of EPG steps predominantly from the RPV Control Guideline. Those labelled with the letter "B" consist of steps predominantly from the Containment Control Guideline. Those labelled with the letter "X" consist of steps predominantly from the contingency procedures.

Table 3.3 summarizes the EPG steps that make up the action sets. The steps are grouped together in the action sets of Table 3.2 to show the sequence of actions comprising each set.

Table 3.1

BWR EPG

Parameters & Symptoms

Parameter	Symptom
RPV Level	1. Level above High Level Trip Setpoint (+ 58* in.)
	2. Level below Low Level Scram Setpoint (+ 12 in.)
	3. Level below ADS Initiation Setpoint/Top of Active Fuel (- 164 in.)
	4. Level below Minimum Zero-Injection RPV Level (- 272 in.)
RPV Pressure	5. Pressure above lowest SRV Lifting Pressure (1090 psig)
	6. Pressure below lowest SRV Lifting Pressure (1090 psig)
	7. Pressure above high RPV Pressure Scram Setpoint (1045 psig)
	8. Pressure below Pressure at which all turbine bypass valves are fully open (935 psig)
	9. Pressure below Minimum Single SRV Steam Cooling Pressure (700 psig)
	10. Pressure exceeds LPCS Shutoff Head (425 psig)
	11. Pressure below LPCS Shutoff Head (425 psig)
	12. Pressure above higher of HPCI or RCIC Low Pressure Isolation Setpoint (100 psig)
	13. Pressure above Minimum Alternate RPV Flooding Pressure (77 psig) plus Suppression Chamber Pressure
RPV Temperature	14. Cooldown Rate Exceeds Maximum RPV Cooldown Rate LCO (100°F/hr).
	15. Temperature below higher of RPV NDTT or Head Tensioning Limit (70°F)

*Values in parentheses refer to "generic" plant values contained in EPGs. Plant-specific values may be different.

Table 3.1 (continued)

BWR EPG

Parameters & Symptoms

Parameter	Symptom
Reactor Power	16. Power above APRM Downscale Trip (3%)
	17. If power above 3% power cannot be maintained above Reactor Flow Stagnation Power (8%)
Control Rod Position	18. Rods inserted beyond Maximum Subcritical Banked Withdrawal Position (06)
Injected Boron Quantity	19. Boron Quantity equal to Hot Shutdown Boron Weight (204 lbs).
Suppression Pool Level	20. Level above Suppression Pool Load Limit (varies with RPV Pressure)
	21. Level above Elevation of Suppression Pool Spray Nozzles (24'6")
	22. Level below Elevation of Suppression Pool Spray Nozzles (24'6")
	23. Level approaches/exceeds Elevation of Bottom Of Mark I Internal Suppression Chamber to Drywell Vacuum Breakers less Vacuum Breaker Opening Pressure in feet of water (17'2")
	24. Level below Elevation of Bottom of Mark I Internal Suppression Chamber to Drywell Vacuum Breakers less Vacuum Breaker Opening Pressure in feet of water (17'2")
	25. Level above Maximum Suppression Pool Water Level LCO (12'6")
	26. Level below Minimum Suppression Pool Water Level LCO (12'2")
	27. Level below heat capacity level limit (8'2" - 12'2")
	28. Level above Elevation of Top of SRV Discharge Device (4'9")
	29. Level below Elevation of Top of SRV Discharge Device (4'9")

Table 3.1 (continued)

BWR EPG

Parameters & Symptoms

Parameter	Symptom
Suppression Chamber Pressure	30. Pressure exceeds Primary Containment Pressure Limit (varies with primary containment level)
	31. Pressure exceeds Primary Containment Design Pressure (46.5 - 56.0 psig)
	32. Pressure exceeds Pressure Suppression Pressure (34.8 - 42.5 psig)
	33. Pressure approaches/exceeds Suppression Chamber Spray Initiation Pressure (17.4 psig)
	34. Pressure below Suppression Chamber Spray Initiation Pressure (17.4 psig)
Suppression Pool Temperature	35. Temperature exceeds Heat Capacity Temperature Limit (varies with RPV Pressure)
	36. Temperature exceeds Boron Injection Initiation Temperature (110°F)
	37. Temperature exceeds Most Limiting Suppression Pool Temperature LCO (95°F)
	38. Temperature exceeds Drywell Spray Initiation Limit (varies with Drywell Pressure)
	39. Temperature below Drywell Spray Initiation Limit (varies with Drywell Pressure)
Drywell Pressure	40. Pressure exceeds High Drywell Pressure Scram Setpoint (2.0 psig)
Drywell Temperature	41. Temperature near Cold Reference Leg Instrument Vertical Runs approaches/exceeds RPV Saturation Temperature (varies with RPV Pressure)
	42. Temperature approaches/exceeds lower of Maximum Temperature at which ADS is Qualified and Drywell Design Temperature (340°F)
	43. Temperature below lower of Maximum Temperature at which ADS is qualified and Drywell Design Temperature (340°F)
	44. Temperature exceeds higher of Drywell Temperature LCO and Maximum Normal Operating Temperature (135°F)

Table 3.1 (continued)

BWR EPG

Parameters & Symptoms

Parameter	Symptom
Containment Level	45. Level exceeds Maximum Allowable Primary Level Containment Level (104')
Primary Containment Temperature	46. Primary Containment Temperature exceeds Maximum Non-condensible Evacuation Temperature (212 °F) (not applicable to Mark I containments)
HPCI Status	47. In Service/Available
	48. Unavailable
RCIC Status	49. In Service/Available
	50. Unavailable
LPCI Status	51. In Service/Available
	52. Unavailable
CS Status	53. In Service/Available
	54. Unavailable
Diesel Generators Status	55. In Service/Available
	56. Unavailable
RHR Status	57. In Service/Available
	58. Unavailable
ADS Timer Status	59. Timed out to ADS
	60. Timing to ADS
	61. Not Timing
Main Condenser Status	62. In Service/Available
	63. Unavailable
CRD Status	64. In Service/Available
	65. Unavailable
SLC Status	66. In Service/Available
	67. Unavailable

Table 3.1 (continued)

BWR EPG

Parameters & Symptoms

Parameter	Symptom
RWCU Status	68. In Service/Available
	69. Unavailable
Alternate Injection Systems Status	70. In Service/Available
	71. Unavailable
Condensate/Feedwater Status	72. In Service/Available
	73. Unavailable
Recirc Pumps Status	74. In Service/Available
	75. Unavailable
MSIV Status	76. Open
	77. Closed
RHR Shutdown Cooling Interlocks	78. Clear
	79. Not Clear

Table 3.2
Symptom/Action Sets

Symptom Sets	Action Set Associated With Symptom Set	EPG Step(s) Comprising Action Set
(2)	A1	RC-1
(7,10,12)		RC/L-1
(40)		RC/L-2
(17)		RC/P-1
(77)		RC/P-2a
		RC/Q-1
(18,57,79)	A2	RC/P-3
(18,58,79)		
(18,57,78)	A3	RC/P-4
		RC/P-5
		RC/L-3
(18,58,78)	A4	C5-1
		C5-2
		C5-3
		C5-5
		C5-6
		C5-7
		C5-8

Table 3.2 (Continued)
Symptom/Action Sets

Symptom Sets	Action Set Associated With Symptom Set	EPG Step(s) Comprising Action Set
(2,10,18)	A5	RC/L-1
(2,11,12,18,48,50,6,8,9)		RC/L-2
(2,6,8,9,11)		RC/P-1
(6,8,9,11,12,18,47,49)		RC/P-2a
		C1-1
		C1-2
		C1-3
(35)	A6	RC-1
(42)		RC/L-1
(27)		RC/L-2
(20)		RC/P-1
		RC/P-2a
		RC/Q-1
		C2-1
		RC/P-4
		RC/P-5
(41)	A7	RC-1
		RC/L-1
		RC/Q-1
		C6-1
		C6-3
(16,36,51)	A8	RC/P-2

Table 3.2 (Continued)

Symptom/Action Sets

Symptom Sets	Action Set Associated With Symptom Set	EPG Step(s) Comprising Action Set
(2,76) (7,76) (16,76) (40,76)	A9	RC/Q-2
(16)	A10	RC/Q-3 RC/Q-5
(16,36,66)	A11	RC/Q-4
(16,36,67)	A12	RC/Q-4.1
(16,36)	A13	RC/Q-4.2 RC/Q-4.3 RC/Q-4.4
(37)	B1	SP/T-1 SP/T-2 SP/T-3 SP/T-4 SP/L-1
(44)	B2	SP/T-1 DW/T-1 SP/L-1
(39,43,44) (37,39,43,44)	B3	DW/T-3
(37,46) (40,46) (44,46) (25,46) (26,46)	B4	SP/T-1 PC/P-1 SP/L-1

Table 3.2 (Continued)

Symptom/Action Sets

Symptom Sets	Action Set Associated With Symptom Set	EPG Step(s) Comprising Action Set
(22,34) (22,30)	B5	PC/P-2 (PC/P-6)
(33,39) (30,39)	B6	PC/P-3 (PC/P-6)
(30)	B7	PC/P-7
(26)	B8	SP/L-1 SP/T-1
(25)	B9	SP/L-1 SP/L-3.1 SP/L-3.2 SP/T-1
(24,39)	B10	SP/L-3.3
(23)	B11	SP/L-3.4
(45)	B12	SP/L-3.5
(2,3,12,51) (2,3,12,53) (2,3,12,70) (2,3,12,72)	X1	C1-1 C1-2 C1-3 C1-7 C2-1 RC/P-4 RC/P-5

Table 3.2 (Continued)

Symptom/Action Sets

Symptom Sets	Action Set Associated With Symptom Set	EPG Step(s) Comprising Action Set
(2,3,12,48,50,52,54,65, 71,73)	X2	C1-1 C1-2 C1-3 C1-7 C3-1a
(2,3,11)	X3	C1-1 C1-2 C1-3 C1-8 C4-1 C4-2
(2,6,8,9,11,12,48,50) (2,6,8,9,11) (2,3,12,47) (2,3,12,49)	X4	C1-1 C1-2 C1-3 C2-1
(16,17) (31)	X5	C6-1 C6-3
(32) (2,3,9,12,48,50,52,54,65,71, 73)	X6	C2-1 RC/P-4 RC/P-5
(2,3,4,12,48,50,52,54,65,71, 73)	X7	C3-1b

Table 3.2 (Continued)

Symptom/Action Sets

Symptom Sets	Action Set Associated With Symptom Set	EPG Step(s) Comprising Action Set
(5,17,36)	X8	C7-1
(5,16,36)		C7-2
(16,36,40)		
(17,36,40)		
(2,3)	X9	C7-2.1 C7-2.2
(18,19)	X10	C7-3 C7-4

Table 3.3

Action Set Summary

Action Set	EPG Steps Comprising Action Set	
A1	(RC-1)	Scram.
	(RC/L-1)	Confirm/Initiate Isolation, ECCS, DG Start.
	(RC/L-2)	Restore and Maintain RPV Level (using Cond./FW, HPCI, RCIC, CS, LPCI, or CRD) and reset ADS timer if timing.
	(RC/P-1)	Minimize SRV cycling; Blowdown to TBVs 100% open pressure; Maximize heat load to MC.
	(RC/P-2a)	Control RPV pressure below SRV setpoint with TBVs augmented by IC, SRVs, HPCI, RCIC, RWCU, etc; Maintain pressure below S.P. temperature and load limits to assure containment available for ADS.
	(RC/Q-1)	Confirm/place reactor mode switch in shutdown.
A2	(RC/P-3)	Depressurize RPV and maintain cooldown rate below cooldown rate LCO (100°F/hr.).
A3	(RC/P-4)	Initiate RHR shutdown cooling. If RHR shutdown cooling cannot be established, cooldown using system(s) used for depressurization.
	(RC/P-5)	Proceed to Cold Shutdown.
	(RC/L-3)	Same as (RC/P-5).
A4	(C5-1)	Initiate S.P. cooling.
	(C5-2)	Close RPV head vents, MSIVs, main steam drain valves, HPCI and RCIC isolation valves.
	(C5-3)	Establish flow path through an SRV to SP.
	(C5-5)	Start CS or LPCI pump with suction from SP.
	(C5-6)	Stabilize RPV pressure, opening another SRV or starting additional LPCI or CS pumps if necessary. Maintain cooldown rate below LCO.
	(C5-7)	Maintain RPV temperature above RPV NDTT and RPV head tensioning limit.
	(C5-8)	Proceed to Cold Shutdown.

Table 3.3 (Continued)

Action Set Summary

Action Set	EPG Steps Comprising Action Set	
A5	(RC/L-1)	Confirm/Initiate Isolation, ECCS, DG. Start.
	(RC/L-2)	Restore and Maintain RPV Level (using Cond./FW, HPCI, RCIC, CS, LPCI, or CRD) and reset ADS timer if timing.
	(RC/P-1)	Minimize SRV cycling; Blowdown to TBVs 100% open pressure; Maximize heat load to MC.
	(RC/P-2a)	Control RPV pressure below SRV setpoint with TBVs augmented by IC, SRVs, HPCI, RCIC, RWCU, etc; Maintain pressure below S.P. temperature and load limits to assure containment available for ADS.
	(C1-2)	Line up and start pumps in 2 or more injection trains (Cond., CS, LPCI); If less than 2 can be lined up, line up as many alternate injection systems as possible (RHR SW crosstie, Fire, SLC, etc.)
	(C1-3)	Monitor RPV pressure and water level.
A6	(RC-1)	Scram.
	(RC/L-1)	Confirm/Initiate Isolation, DG Start.
	(RC/L-2)	Restore and Maintain RPV Level (using Cond./FW, HPCI, RCIC, CS, LPCI, or CRD) and reset ADS if timing.
	(RC/P-1)	Minimize SRV cycling; Blowdown to TBVs 100% open pressure; Maximize heat load to MC.
	(RC/P-2a)	Control RPV pressure below SRV setpoint with TBVs augmented by IC, SRVs, HPCI, RCIC, RWCU, etc; Maintain pressure below S.P. temperature and load limits to assure containment available for ADS.
	(RC/Q-1)	Confirm/place reactor mode switch in shutdown.
	(C2-1)	Depressurize RPV using ADS valves augmented by other SRVs and other systems (MC, RHR, Main steam drain valves, HPCI, RCIC, head vent). Use systems in order which will minimize release to environment.
	(RC/P-4)	Initiate RHR shutdown cooling. If RHR shutdown cooling cannot be established, cooldown using system(s) used for depressurization.
	(RC/P-5)	Proceed to Cold Shutdown.

Table 3.3 (Continued)

Action Set Summary

Action Set	EPG Steps Comprising Action Set	
A7	(RC-1)	Scram.
	(RC/L-1)	Confirm/Initiate Isolation, ECCS, DG Start.
	(RC/Q-1)	Confirm/place reactor mode switch in shutdown.
	(C6-1)	If enough SRVs for emergency depressurization can be open or CS or motor-driven FW pumps are available, close MSIVs, main steam line drain valves, IC, HPCI, RCIC and RHR steam condensing isolation valves.
	(C6-3)	Inject into RPV with CS, Motor-driven FW pumps, LPCI, Condensate pumps, CRD, and alternate injection systems until enough SRVs for emergency depressurization are open and RPV pressure is stable about minimum RPV flooding pressure or until water level is increasing.
A8	(RC/P-2b)	Open MSIVs to re-establish main condenser as heat sink.
A9	(RC/Q-2)	If main turbine-generator is on-line, confirm/initiate recirculation flow runback to minimum.
A10	(RC/Q-3)	Trip recirculation pumps.
	(RC/Q-5)	Insert control rods.
A11	(RC/Q-4)	Inject boron into RPV with SLC. Prevent automatic initiation of ADS.
A12	(RC/Q-4.1)	Inject boron into RPV by one or more of CRD, RWCU, HPCS, FW, HPCI, RCIC, Hydro Pump.
A13	(RC/Q-4.2)	If RWCU not injecting boron, confirm isolation or manually isolate RWCU.
	(RC/Q-4.3)	Inject boron until Cold Shutdown Boron Weight (280 lbs.) has been injected.
	(RC/Q-4.4)	Enter scram procedure.

Table 3.3 (Continued)

Action Set Summary

Action Set	EPG Steps Comprising Action Set	
B1	(SP/T-1)	Close all SORV; If any SORV cannot be closed, scram reactor.
	(SP/T-2)	Operate available S.P. cooling.
	(SP/T-3)	Before S.P. temperature reaches boron injection temperature, scram reactor.
	(SP/T-4)	Maintain S.P. temperature below heat capacity temperature limit.
	(SP/L-1)	Maintain S.P. level between min. and max. LCOs. Refer to sampling procedure prior to discharging water.
B2	(SP/T-1)	Close all SORV; If any SORV cannot be closed, scram reactor.
	(DW/T-1)	Operate available DW cooling.
	(SP/L-1)	Maintain S.P. level between min. and max. LCOs. Refer to sampling procedure prior to discharging water.
B3	(DW/T-3)	Shutdown recirc. pumps and drywell fans and initiate drywell sprays (flow < max. flow rate limit).
B4	(SP/T-1)	Close all SORV; If any SORV cannot be closed, scram reactor.
	(PC/P-1)	Operate SBGT and drywell purge.
	(SP/L-1)	Maintain S.P. Level between min. and max. LCOs. Refer to sampling procedure prior to discharging water.
B5	(PC/P-2)	Initiate S.P. sprays.
B6	(PC/P-3)	Shutdown recirc. pumps, drywell cooling fans; Initiate drywell sprays (flow < max. flow rate limit).
B7	(PC/P-7)	Vent primary containment to reduce and maintain pressure below primary containment pressure limit.

Table 3.3 (Continued)

Action Set Summary

Action Set	EPG Steps Comprising Action Set	
B8	(SP/L-1)	Maintain S.P. level between min. and max. LCOs. Refer to sampling procedure prior to discharging water.
	(SP/T-1)	Close all SORV; If any SORV cannot be closed, scram reactor.
B9	(SP/L-1)	Maintain S.P. level between min. and max. LCOs. Refer to sampling procedure prior to discharging water.
	(SP/L-3.1)	If adequate core cooling assured, terminate injection into RPV from sources outside primary containment.
	(SP/L-3.2)	Maintain RPV pressure below SP load limit.
	(SP/T-1)	Close all SORV; If any SORV cannot be closed, scram reactor.
B10	(SP/L-3.3)	Shutdown recirc. pumps and drywell cooling fans; Initiate drywell sprays (flow < max. flow rate limit).
B11	(SP/L-3.4)	Continue drywell sprays.
B12	(SP/L-3.5)	Terminate injection into RPV from sources outside primary containment regardless of adequate core cooling.
X1	(C1-2)	Line up and start pumps in 2 or more injection trains (Cond., CS, LPCI); If less than 2 can be lined up as many alternate injection systems as possible (RHR SW crosstie, Fire, SLC, etc.)
	(C1-3)	Monitor RPV pressure and water level.
	(C1-7)	Restart HPCI and RCIC if not operating.
	(C2-1)	Depressurize RPV using ADS valves augmented by other SRVs and other systems (MC, RHR, Main steam drain valves, HPCI, RCIC, head vent). Use systems in order which will minimize release to environment.

Table 3.3 (Continued)

Action Set Summary

Action Set	EPG Steps Comprising Action Set	
X1 (Con't)	(RC/P-4)	Initiate RHR shutdown cooling. If RHR shutdown cooling cannot be established, cooldown using system(s) used for depressurization.
	(RC/P-5)	Proceed to Cold Shutdown.
X2	(C1-2)	Line up and start pumps in 2 or more injection trains (Cond., CS, LPCI); If less than 2 can be lined up, line up as many alternate injection systems as possible (RHR SW crosstie, Fire, SLC, etc..)
	(C1-3)	Monitor RPV pressure and water level.
	(C1-7)	Restart HPCI and RCIC if not operating.
X3	(C1-2)	Line up and start pumps in 2 or more injection trains (Cond., CS, LPCI); If less than 2 can be lined up, line up as many alternate injection systems as possible (RHR SW crosstie, Fire, SLC, etc.)
	(C1-3)	Monitor RPV pressure and water level.
	(C1-8)	Start pumps in alternate injection systems.
	(C4-1)	Open ADS valves and other SRVs if required.
	(C4-2)	Operate CS with suction from SP. When RPV pressure is below CS rated flow pressure, terminate injection from outside primary containment.
X4	(C1-2)	Line up and start pumps in 2 or more injection trains (Cond., CS, LPCI); If less than 2 can be lined up, line up as many alternate injection systems as possible (RHR SW crosstie, Fire, SLC, etc.)

Table 3.3 (Continued)

Action Set Summary

Action Set	EPG Steps Comprising Action Set	
X4 (Con't)	(C1-3)	Monitor RPV pressure and water level.
	(C2-1)	Depressurize RPV using ADS valves augmented by other SRVS and other systems (MC, RHR, Main steam drain valves, HPCI, RCIC, head vent). Use systems in order which will minimize release to environment.
X5	(C6-1)	If enough SRVs for emergency depressurization can be opened or CS or motor-driven FW pumps are available, close MSIVs, main steam line drain valves, IC, HPCI, RCIC and RHR steam condensing isolation valves.
	(C6-3)	Inject into RPV with CS, Motor-driven FW pumps, LPCI, Condensate pumps, CRD, and alternate injection systems until enough SRVS for emergency depressurization are open and RPV pressure is stable above minimum RPV flooding pressure or until water level is increasing.
X6	(C2-1)	Depressurize RPV using ADS valves augmented by other SRVs and other systems (MC, RHR, Main steam drain valves, HPCI, RCIC, head vent). Use systems in order which will minimize release to environment.
	(RC/P-4)	Initiate RHR shutdown cooling. If RHR shutdown cooling cannot be established, cooldown using system(s) used for depressurization.
	(RC/P-5)	Proceed to Cold Shutdown.
X7	(C3-1b)	Open one SRV.
X8	(C7-1)	Lower RPV water level by terminating/preventing all injection into RPV except from boron injection systems and CRD.
	(C7-2)	When reactor power drops below 3% or RPV level drops below TAF or SP temperature drops below 110°F or all SRVs remain closed and drywell pressure remains below 2.0 psig, maintain RPV water level using C/FW, CRD, RCIC, HPCI, or LPCI.

Table 3.3 (Continued)

Action Set Summary

Action Set	EPG Steps Comprising Action Set	
X9	(C7-2.1)	Terminate/prevent all injection into RPV except from boron injection systems and CRD.
	(C7-2.2)	When RPV pressure is below Minimum Alternate RPV Flooding Pressure, commence and slowly increase injection into RPV using C/FW, CRD, RCIC, HPCI, LPCI to restore and maintain RPV level above TAF. If level cannot be restored and maintained with these systems, use CS, RHR SW crosstie, Fire system, or other alternate injection systems.
X10	(C7-3)	Restore and maintain RPV water level between high and low level scram setpoints, or above TAF.
	(C7-4)	Proceed to cold shutdown.

Section 4

OPERATOR ACTION EVENT TREES AND ACCIDENT ANALYSES

Presented in this section are the operator action event trees (OAET) which have been developed for a selection of potential accident sequences at a representative BWR 4 with a Mark I containment. These OAETs address a wide spectrum of accident conditions ranging from relatively high frequency anticipated transient events to risk significant multiple failure accident sequences. The process by which the accident conditions were selected for analysis is presented below in Section 4.1. The OAETs presented in this section were developed in accordance with the methodology described in Volume 1 of this report (see also Reference 1) and are based upon best-estimate computer analyses (References 3, 4, 5, and 6) and actual operating experience (Reference 7). The OAETs presented in this report address the role of the operator in his attempt to prevent core damage and do not explicitly address his role subsequent to core damage. Thus, while symptoms indicative of operator success or failure in preventing core damage are documented, the actions required after core damage has occurred are not addressed.

The accident conditions addressed by these OAETs can be grouped into two very general categories:

1) Those initiated by faults or failures (other than coolant system breaks) which require reactor trip, and

2) Those initiated by a break of the reactor coolant pressure boundary

The first category of accidents, which is comprised of "transient" initiated events, is addressed in Section 4.2. The second category, referred to as loss-of-coolant accidents (LOCAs), is addressed in Section 4.3. It should be noted that this latter category includes transient initiated sequences which subsequently result in breaches of the reactor coolant pressure boundary.

4.1 Selection of Accident Conditions

The goal of this EPG review process is to systematically examine the ability of the BWR EPGs to provide effective and unambigious guidance under a pre-selected set of potential accident conditions. The amount of confidence in the EPGs which can be generated by this systematic evaluation process is in direct proportion to the breadth and depth of the accident conditions examined. Accordingly, a broad spectrum of accident conditions was selected as the basis for this EPG review.

The first goal of the accident condition selection process was to ensure that all functional types of accidents (or potential accidents) are addressed. The definition of these functional types is based upon the concept that there is a set of critical safety functions whose successful performance will ensure plant safety. The critical safety functions for a BWR can be described as:

- Limitation of reactor power to a level commensurate with heat removal capability
- Maintenance of adequate coolant inventory
- Removal of heat

During normal operation, failure to perform these function will necessitate reactor shutdown. Following shutdown, these functions must be performed to ensure a successful cooldown process.

Accident conditions associated with the failure of or which pose a threat to each of these critical safety functions were selected. The inclusion of each functional type of accident condition ensures a broad base for the EPG evaluation.

Within each type, it was desired to select these particular accident conditions which represented both anticipated relatively high frequency events and lower probability multiple failure events. In order to ensure that the most important accident conditions from both a probability and consequence view were selected, the results of probabilistic risk assessments (PRAs) of BWRs were used

to guide the selection process. The accident sequences which were determined to dominate the risk (or core damage frequency) in these PRAs can be used to define important initiating events as well as the probabilistically significant multiple failure states. In this way, the criteria used to select the postulated accident conditions upon which the EPG evaluation was based included both the probability that the operator will actually be confronted with the particular accident condition and the potential consequences should the operator not respond adequately to the situation.

Table 4.1, which uses information produced by the NRC's Accident Sequence Evaluation Program (Reference 8) and the individual PRAs, summarizes these dominant sequences for six different PRAs. The Browns Ferry #1 PRA[9] was performed as part of the Interim Reliability Evaluation Program (IREP) by NRC contractors. The Peach Bottom #2 PRA was part of the Reactor Safety Study.[10] The Grand Gulf #1 PRA[11] was performed by Sandia National Laboratories under contract to the NRC as part of the Reactor Safety Study Methodology Application Program (RSSMAP). The Limerick PRA[12] was sponsored by the Philadelphia Electric Company and the Shoreham PRA[13] was sponsored by the Long Island Lighting Company.

Based on the dominant accident sequences presented in Table 4.1, the following accident conditions were selected for examination in this report:

1) Transient results in a loss of the Power Conversion System (PCS) followed by failure of the Residual Heat Removal (RHR) System to remove decay heat

2) Station Blackout results in a loss of the Power Conversion System (PCS) followed by failure of the Residual Heat Removal (RHR) System to remove decay heat

3) Transient results in a loss of Feedwater combined with failure of High Pressure and Low Pressure Coolant Injection Systems to maintain coolant inventory

4) Station Blackout results in a loss of Feedwater combined with failure of High Pressure and Low Pressure Coolant Injection Systems to maintain coolant inventory

5) Transient results in loss of PCS followed by failure of Reactor Shutdown (ATWS).

6) Transient results in stuck-open relief valve

NRC IREP (Reference 9) Browns Ferry #1 BWR 4/MARK 1	NRC RSSMAP (Reference 11) Grand Gulf #1 BWR 6/MARK III	PHILADELPHIA ELECTRIC CO. (Reference 12) Limerick BWR 4/MARK II
1. Transient results in Loss of Power Conversion System (PCS) followed by Failure of Residual Heat Removal (RHR) System.	1. Loss of Offsite Power results in Loss of PCS combined with Failure of PORV to reseat and Failure of RHR.	1. Loss of Offsite Power results in Loss of Feedwater combined with Failure of High Pressure Coolant Injection (HPCI) and Low Pressure Coolant Injection (LPCI).
2. Transient Results in Loss of PCS followed by Failure of Reactor Shutdown.	2. Transient results in Loss of PCS combined with Failure of PORV to reseat and Failure of RHR.	2. Main Steam Isolation Valve Closure results in Loss of Feedwater combined with Failure of HPCI and Failure to Depressurize.
3. Loss of Offsite Power results in Loss of PCS followed by Failure of RHR.	3. Loss of Offsite Power results in Loss of PCS combined with Failure of PORV to reseat and Failure of Emergency Core Cooling.	3. Loss of Offsite Power results in Loss of Feedwater, followed by Failure of HPCI and Failure to Depressurize
4. Transient results in Loss of PCS and stuck-open PORV combined with Failure of RHR.	4. Transient results in Loss of PCS combined with Failure of PORV to reseat and Failure of Emergency Core Cooling.	4. Turbine Trip results in Loss of Feedwater, followed by Failure of HPCI and Failure to Depressurize.
5. Transient results in Loss of PCS combined with Failure of Reactor Core Isolation Cooling (RCIC) System and then RHR.	5. Small LOCA ($01ft^2$) followed by Failure of RHR.	5. Loss of Offsite Power results in Loss of PCS followed by Failure of RHR.

Table 4.1

Summary of BWR Dominant Sequences Leading to Possible Core Melt
Based upon Published Probabilistic Analyses

Reactor Safety Study (Reference 10)	Long Island Lighting Company (Reference 13)
Peach Bottom #2 BWR 4/MARK I	Shoreham BWR 4/MARK II
1. Transient results in Loss of PCS combined with Failure of RHR.	1. Loss of offsite power initiator coupled with unavailability of coolant injection
2. Transient results in Loss of PCS coupled with failure of Reactor Shutdown	2. Transient results in loss of PCS coupled with failure of reactor shutdown
3. Transient results in Loss of Feedwater combined with Failure of emergency coolant injection	3. Loss of condenser vacuum causing loss of feedwater coupled with failure of HPCI and RCIC plus failure to depressurize
4. Small LOCA combined with Failure of Emergency Coolant Injection.	4. Turbine trip transient coupled with the unavailability of coolant injection
5. Small LOCA combined with Failure of Low Pressure Coolant Recirculation.	5. Loss of DC power and normal PCS
	6. Reactor building flooding initiated sequences coupled with failure of coolant injection
	7. Transient followed by loss of containment heat removal

Table 4.1 (Continued)

Summary of BWR Dominant Sequences Leading to Possible Core Melt
Based Upon Published Probabilistic Analyses

7) Small LOCA followed by failure to remove decay heat

8) Small LOCA followed by failure to maintain coolant inventory.

Each of these eight types of accident conditions is described in significantly more detail in Sections 4.2 and 4.3 below.

4.2 *Transient Initiated Accident Conditions*

The operator action event trees and supporting documentation presented in this section address the role of the operator in responding to accident conditions initiated by transient events. A transient event is defined to be a fault or failure (other than primary coolant boundary breaks) which results in a sufficient deviation of plant process parameters from normal values that a shutdown is required.

Once a transient event has occurred, four interrelated critical safety functions must be performed to ensure a cooldown process.

1) The reactor must be made subcritical and maintained in that state

2) Adequate coolant inventory must be maintained

3) Decay heat must be removed

Operator action event trees have been developed which address accident conditions associated with the failure to perform each of these functions.

In Section 4.2.1, an OAET is presented and documented which addresses operator actions in response to a transient event followed by a failure of plant systems to automatically maintain adequate coolant inventory. In Section 4.2.2, an OAET is presented and documented which addresses operator actions in response to transient event followed by a failure to adequately remove decay heat. Due to the special nature of the initiating event, the operator actions associated with maintaining the critical functions following a station blackout event are addressed in Section 4.2.3. Finally, a few important aspects of the opera-

tor's response to an anticipated transient with a failure to scram (ATWS) are addressed in Section 4.2.4

In all of the above transient sequences, the success or failure of the safety/relief valves to limit the RPV pressure is explicitly addressed. Transient initiated accident sequences which result in stuck-open relief valves are discussed in Section 4.3, Loss of Coolant Accidents.

4.2.1 Failure to Maintain Inventory after Transient Event (TQUV)

In this section, the physical plant response and operator actions associated with postulated accident sequences involving an inability to maintain adequate coolant inventory following a transient-induced shutdown are presented and discussed. A summary of the accident sequence is provided below in Section 4.2.1.1. The Operator Action Event Tree for this sequence is provided in Section 4.2.1.2 and used as the framework for a description of the role of the operator throughout the progression of this sequence. In Section 4.2.1.3, the physical plant response at each state in the OAET is presented in terms of the behavior of the twenty-nine key BWR parameters discussed in Section 3.

4.2.1.1 Sequence Description

Accident sequences initiated by a transient event and involving subsequent failure to maintain adequate coolant inventory have represented a dominant contributor to risk in most of the BWR PRAs performed to date. In the Reactor Safety Study, this sequence was designated TQUV and entailed a transient event (T) followed by failure of the Feedwater System (Q), and the unavailability of high pressure sources of water (U), as well as low pressure sources of water (V). A number of specific transient initiating events have been identified and assessed in the BWR PRAs discussed in Section 4.1 as well as supporting studies such as Reference 14. For purposes of this evaluation, the loss of feedwater initiator was considered. A major reason for this selection was the determination that the loss of feedwater is probabilistically an important initiator. Reference 14 reports that a BWR will experience between 8 and 9 transient events each year. Approximately ten percent of these initiators can be directly attributed to Feedwater System faults or failures. An even larger percentage is due to other power conversion system failures which will quickly lead to feedwater pump trip. Furthermore, initiating events which result in a loss of feedwater are most significant for TQUV sequences because the "Q" event involves unavailability of feedwater. The conditional probability of this event would obviously be much higher for loss of feedwater initiators than for these initiators which have no impact on feedwater availability (in fact, this conditional probability would be equal to unity for some loss of feedwater events which preclude recovery).

A loss of feedwater can be caused by a number of events. These include: equipment failures, such as the loss of the condensate or feedwater pumps; malfunctions of the feedwater control systems; disturbances in the power conversion system, such as a loss of condenser vacuum; or operator error. The specific cause of the loss of feedwater could impact the course of this sequence in two ways: 1) it could impact the operator's capability to restore feedwater should ensuing events require this action for accident termination, and 2) causes which impact the ability to isolate the core from the condenser could affect the rate at which the vessel water declines and the concurrent heat-up rate of containment. These considerations are discussed further below.

Table 4-2 is a compilation of key events which may occur during a loss of feedwater initiated sequence with failure to maintain coolant inventory. The timing for the key events is based upon three sources (Reference 3, 6, 7). Figure 4.1 presents the system event tree developed in the Reactor Safety Study for transient initiating events in BWRs. The TQUV sequence is highlighted and depicts the general progression of the sequence in terms of system failures. In Section 4.2.1.2, a detailed logic tree is developed to display the operator actions in conjunction with the key plant states.

Table 4.2

Key Sequence of Events in "TQUV" Sequence

Time	Event
0	Initiating Feedwater pump trip
0-2 sec	Main turbine trip
0-2 sec	Reactor trip
0-2 sec	Recirculation pump trip
4 sec	Relief valves set points activated
30-40 sec	HPCI/RCIC initiation signal are low low level MSIVs close if still open HPCI/RCIC fail to start
1-15 min	Operator attempts to restore high pressure systems
5-15 min	Operator initiates depressurization
30 min	If no depressurization, level reaches TAF If successful depressurization, but NO LP ECCS, level reaches TAF

The immediate response to a loss of feedwater is a reduction in the water level in the reactor vessel since no makeup is available to replace the steam generated in the core. Upon sensing the loss of flow in the feedwater lines, the control system automatically signals the flow control valves governing recirculation flow to reduce flow in those loops. The reduction in core flow increases the void fraction and reactor power decreases. The low vessel water level scram setting is reached within a few seconds. At this point the control rods are automatically inserted into the core terminating full power operation, and the turbine/generator is tripped.

The reactor pressure will momentarily rise in response to the turbine/generator trip. Safety valve operation combined with operation of the turbine steam bypass system will then bring the pressure back down. With decreased pressure, increased void formation will cause the reactor water level to momentarily rise (perhaps above the low vessel level setpoint-Level 3). However, the stored heat and fission product decay heat will continue to produce steam. The brief rise will be terminated and level will begin a steady reduction to the low-low level setpoint (Level 2).

Following the initial transients in pressure and level in the first few seconds after the loss of feedwater, the plant behavior can be characterized in terms of a steadily declining level with the pressure being maintained relatively high. If the MSIVs close, the pressure will quickly rise to and cycle about the safety/relief value setpoints. If the MSIVs do not close, the pressure will be controlled by the pressure regulator until the MSIVs do close. In either case, the collapse in voids due to the abrupt power reduction at high pressure together with the continued production of steam will lower the vessel water level to the low-low (Level 2) setpoint within 15-45 seconds (depending upon the feedwater coast down rate).

The low-low (Level 2) setting will close the MSIVs if they are open* and generate a demand for the Reactor Core Isolation Cooling (RCIC) System and the High Pressure Coolant Injection (HPCI) System. The RCIC System is designed

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*on some new plants MSIV closure does not occur at Level 2

to provide makeup to the core during a reactor shutdown in which feedwater flow is not available. The RCIC consists of one steam driven turbine pump with associated piping, valves, and controls. Reactor steam drives the turbine and steam exhaust from the turbine discharges to the suppression pool. Suction for RCIC comes initially from the Condensate Storage Tank (CST). Suction can be manually transferred to the suppression pool. The RCIC is capable of delivering approximately 400-600 gpm to the core via a feedwater line.

The HPCI System is a part of the Emergency Core Cooling System (ECCS) designed to supply water to the core during small break accidents which do not produce a rapid system depressurization. The HPCI system also consists of a single reactor steam driven turbine pump. The system is capable of delivering approximately 5000 gpm to the core. In the normal configuration, water is taken from the CST and discharged into a feedwater line (usually the opposite feedwater line used by RCIC; in some cases, HPCI, or HPCS on later BWR models may have its own injection line). If the CST level drops to a low level setpoint or the suppression pool level reaches the high level set point, automatic transfer of suction is made to the suppression pool.

In this accident sequence, it is postulated that both the RCIC and HPCI fail to maintain adequate inventory. Reference 15 reports that "a large number of BWR plants are not meeting the performance goals established for HPCI and RCIC reliability. Specifically, there have been four instances [in less than three years] when, upon demand, neither HPCI or RCIC would operate." The simultaneous unavailability of all high pressure makeup systems including feedwater is therefore a potential sequence of events that may realistically occur at a BWR.

The failure of HPCI and RCIC following the loss of feedwater initiator results in the absence of any means to replenish and adequately maintain vessel inventory at high pressure. The Control Rod Drive (CRD) hydraulic supply pumps can help the situation, but the steaming rate will exceed the makeup capability of the CRD pumps for more than five hours into the transient.

It should also be noted that the main steam line isolation will shut off the steam supply to the main feedwater pump turbines. This will inhibit recovery by means of these pumps. In fact, the MSIVs will be interlocked closed because of the low level conditions.

At this point, the only option to prevent core damage is to depressurize the system and attempt to utilize the low pressure ECCS. For a LOCA, this depressurization should be accomplished automatically. However, for the sequence of events postulated here, depressurization will have to be performed manually as the system response will not cause automatic actuation soon enough to prevent core uncovery.

The Automatic Depressurization System (ADS) is designed to rapidly reduce reactor system pressure. Several of the safety/relief valves are designated as ADS valves. For ADS operation, these valves are actuated by a separate logic which requires the following conditions:

(1) low vessel water level (Level 1) for a period of at least two minutes*
(2) high drywell pressure
(3) permissive signal for low water level (Level 3)
(4) adequate discharge pressure from any LPCIS pump or two CSIS pumps.

Once all of the criteria are satisfied, the ADS valves automatically open and dump steam into the suppression pool. This causes a rapid drop in the coolant system pressure. The ADS valves or any of the other SRVs can also be activated manually.

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*This period provides sufficient time for the RCIC or HPCIS to reflood the core in the event of a small break LOCA.

For the sequence of events described above, the pressure in the drywell would be unaffected in the short term by the loss of feedwater and subsequent failure events. The suppression pool is designed to quench large quantities of steam without elevating the pressure in the drywell. Thus, during the initial period of venting through the safety/relief valves for this sequence, the drywell pressure would remain constant. Of course, eventually the continued discharge of steam will cause the suppression pool temperature and primary containment pressure to rise. However, it is unlikely that the release into the pool resulting from the loss of feedwater and high pressure ECCS will elevate the drywell pressure to the ADS set point within 30 minutes. Hence, operator actuation of the ADS may be required to prevent core damage.

If automatic depressurization does not occur and the operator fails to either diagnose the need for or fails to implement a manual depressurization, the water level will continue to decline and reach the top of active fuel (TAF) within about 30-40 minutes. Soon after this point, the fuel will be in jeopardy of melting with the vessel at an elevated pressure.

If depressurization is successful, then the low pressure ECCS or the condensate system can be utilized to replenish coolant and remove decay heat. Two low pressure ECC systems are available. The Low Pressure Coolant Injection System (LPCIS) is one of the three operating modes of the Residual Heat Removal System (RHRS). The LPCIS consists of two independent loops, each with two pumps in parallel. The pumps take suction from the suppression pool and discharge water into the two recirculation loops. The LPCIS pumps are automatically started on low vessel water level (Level 1) and can deliver adequate flow to the core when the system pressure drops below 300 psi.

The other low pressure ECCS is the Core Spray Injection System (CSIS). The CSIS has two trains, each having one or two AC motor driven pumps. Water is drawn from the suppression pool and discharged into the vessel via separate core spray spargers. These two loops are not interconnected.

In addition to the low pressure ECCS systems, the operator may also make use of the condensate system which draws water from the condenser hotwell. The hotwell inventory can be replenished from the CST.

Should these low pressure systems subsequently prove unable to maintain adequate inventory and the operator is unable to restore these systems or obtain adequate inventory delivery from other sources, the vessel level will drop to below the top of the active fuel and the core will be in jeopardy of melting with the vessel depressurized.

4.2.1.2 Operator Response

The preceding section focused on the physical sequence of events associated with a failure to maintain adequate coolant inventory following a transient-induced shutdown. In this section, the role of the operator throughout the progression of this sequence will be examined.

Figure 4.2 presents an operator action event tree (OAET) developed for the TQUV sequence. This diagram will provide the framework for the discussion of operator response in this section. This OAET is similar to the system event tree in Figure 4.1 with modifications to reflect operator responses to the key events. For simplicity, several successful events (including scram and safety valve operation) are combined into a single event tree heading. The different branches of the event tree have been assigned alphanumeric identifiers for referencing the plant status in the following discussion.

As depicted in the OAET and as will be discussed below, the principal operator actions can be categorized as follows:

(1) identify loss of feedwater (FW)

(2) verify occurrence of automatic system responses to loss of FW (e.g., scram, recirculation pump trip, MSIV closure, etc.)

(3) identify unavailability of HPCI and RCIC

(4) restore HPCI, RCIC, or FW

(5) manually depressurize system

(6) identify failure of LP ECCS

(7) restore LP ECCS or FW

(8) provide long term cooling using RHRS or PCS

The initial operator action is to identify the initiating transient so that the correct actions to bring the plant to a safe shutdown can be implemented. The loss of feedwater will be immediately characterized by a reduction in vessel water level and a decrease in flow through the feedwater piping. To distinguish this event from a feedwater line break, the drywell or containment atmosphere conditions can be checked as they will be unaffected by the loss of feedwater.

As noted in Section 4.2.1.1, the plant instrumentation and control system should detect the changes in vessel level and feed flow and automatically reduce recirculation flow. Within a few seconds after a total loss of feedwater, the vessel low water level setting (Level 3) is reached and the reactor is scrammed. The vessel level will continue to decrease and will shortly reach the low-low level set point (Level 2) which initiates a recirculation pump trip, actuation of the HPCIS and RCIC, and in some plants, the closure of the MSIVs. In this analysis, MSIV closure at Level 2 is assumed.

For the TQUV sequence it is assumed that all of these events occur as expected (states 1 and 2 of Figure 4.2) with the exception that the HPCI and RCIC fail to provide adequate flow to the reactor. The operator must verify the reactor scram by checking the flux and control rod position, verify the rundown and subsequent termination of the recirculation pumps, and verify MSIV closure. MSIV closure will cause the pressure to increase and within several seconds the safety/relief valve (SRV) set points will be reached. Successful operation of one or more of these valves will limit the system pressure rise. The operator should take steps to minimize SRV cycling. This would include holding one valve open until the reactor pressure drops to well below the SRV setpoint. The operator should monitor the reactor pressure to ensure that the valves are performing their pressure relief function (successful opening and reclosing of the safety/relief valves is assumed in the TQUV sequence; see section 4.3.3 for a discussion of stuck-open relief valves).

Another operator response to the loss of feedwater transient is to verify automatic actuation of the HPCI and RCIC. For the postulated TQUV sequence, both HPCI and RCIC are unavailable for use upon demand (state 3).

Thus the normal means of inventory replenishment at high pressure are completely absent. The operator has the following options* when this occurs: (1) restore feedwater, HPCI or RCIC, or (2) depressurize the system and utilize the low pressure ECCS for makeup and cooling. Each of these actions will be discussed briefly.

Since there is less than 30 minutes to restore cooling water to the vessel to prevent core uncovery, there is probably not time to repair any failed components. However, recent investigations of HPCI and RCIC failures (References 7, 15) reveal that many failures are due to spurious trip or isolation signals. In these cases, the HPCI/RCIC pumps and valves remain physically capable of providing adequate flow and manual restart may prove successful. Reference 7 describes the actual occurrence at Hatch-1 (a BWR-4 operated by Georgia Power) in which a loss of feedwater initiated event was followed by failure of both HPCI and RCIC to start on demand. The HPCI failure was due to a false trip signal and repeated operator efforts to restart HPCI succeeded after five minutes.

If the operator successfully restores HPCI or RCIC (state 4), the next action is to bring the plant to a safe cold shutdown condition. Assuming unavailability of the feedwater and power conversion systems, plant cooldown would be accomplished using the restored high pressure system. The water level in the condensate storage tank should be monitored and suction changed to the suppression pool if the supply is depleted. This switchover should occur automatically due to either low CST level or high suppression pool level. Since the reactor is venting steam to the suppression pool, the water temperature of this heat sink should be observed. Before the temperature rise becomes excessive, the suppression pool cooling mode of the residual heat removal system (RHRS) should be actuated. Once the system temperatures and pressure are reduced, a transition to the normal RHRS operation can be effected (state 9). It is assumed that RHR operation in this mode is successful as none of the preceding events should impact its capability to perform on demand. For

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*For some cases with low decay heat levels there may be adequate coolant makeup capability from available small capacity high pressure pumps such as the CRD pumps or the Reactor Water Clean-up pumps.

this reason, state 9a of Figure 4.2 is not addressed and is shown only for completeness. Failure of long term cooling is considered in the TW accident sequence summary (Section 4.2.2).

If the operating staff cannot restore HPCI or RCIC, they might attempt to restore the feedwater system. However, due to closure of the MSIVs, the steam required to drive the feedwater turbines will not be available. A few minutes of flow may be possible if the initiating event did not preclude operation of the pump, but prolonged flow will not be possible without reopening or bypassing the steam isolation valves. As noted previously, these valves will be interlocked closed due to the low level condition. Since water level has not been restored, re-opening the MSIVs would entail bypassing the low level interlock.

In addition to the feedwater system, HPCI, and RCIC, there are other systems which can deliver water to the vessel at operating pressures. One of these is the control rod drive hydraulic supply system. This system contains two 90 gpm pumps. Only one of these is normally operating. It would remain operating during the events discussed at this point in the TQUV sequence. The second pump should start automatically following receipt of a scram signal. The operator must verify or actuate operation of the second pump to produce maximum makeup capability. There is some uncertainty that this source (even with quick actuation of the redundant pump) would be sufficient to prevent core damage for a sudden, total loss of all feedwater in which the water level drops rapidly. However, for some less severe transients, this system could provide adequate makeup and thus avoid the need for system depressurization.

Similarly, the Standby Liquid Control System contains two high head injection pumps which are capable of injecting fluid from the Standby Liquid Control Tank at normal system operating pressure. The pumps can only be operated one at a time*. These pumps have a limited capacity (approximately the same as the control rod drive pumps) and require operator actuation with a

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*ATWS mitigation requires SLC initiation (see Section 4.2.4). The NRC is currently reviewing the recommendation of BWR utilities to assess the adequacy of the SLC capacity and operating mode.

key locked switch. In addition, the inventory in the tank would limit the time this system could be utilized. For this reason, this system would only provide short term makeup. As with the control rod drive pumps, their ability to maintain inventory after a total loss of feedwater is questionable (even temporarily). In addition, and perhaps most importantly, the operator would be very reluctant to inject boron into the core except when absolutely necessary to produce reactor shutdown. Hence, use of this system has been omitted from the operator action event tree.

If quick restoration of HPCI or RCIC (or use of an alternate system) is not possible, the operator must manually actuate the ADS or other relief valves with the intent of using the low pressure ECCS or condensate system. As discussed in Section 4.2.1.1, manual action is considered necessary because the conditions for automatic actuation will probably not be reached for the loss of feedwater initiator before the level drops to the top of the fuel. The decreasing vessel water level and lack of makeup system flow should dictate the need for depressurization to the operator. In addition to the safety relief valves, the operator has some other systems available which can be used to reduce system pressure which are discussed below.

The final failure postulated for the TQUV sequence is the failure to provide sufficient cooling at low pressure. This can occur by either a failure to depressurize the system, or a failure of the LPCI, CS, or condensate system to supply adequate water to prevent core melt*. During the depressurization and low pressure injection process, there may be some difficulty in identifying the reactor vessel water level. Two major scenarios could occur:

1) The operator controls the blowdown and the injection valve opening, and throttles the low pressure injection to control water level in the range of Level 5-6.

2) The operator rapidly blows down the primary system and all of the low pressure systems inject without operator control, thus

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*There are additional low pressure systems available to the operator in extreme conditions, including: service water, fire pumps, condensate transfer pumps, etc.

flooding the RPV with the excess flow circulating through the open SRVs to the suppression pool.The operator should then either continue to circulate in this manner or attempt to recover water level indication by terminating flow and waiting for level to return to control room readout range.

The failure to depressurize through the ADS valves can be caused by a failure of the valves to function automatically, or a failure of the operator to manually open the valves. Failure to automatically depressurize may be due to one of three reasons:

1) no drywell pressure signal
2) failure of the ADS logic or valves, or
3) inadequate discharge pressure from the low pressure ECCS's.

If the ADS valves fail to open, there are other methods to depressurize the system which could be employed. In addition to the ADS valves, there are the other safety/relief valves which the operator could manually operate to reduce reactor pressure. The operator would have to ensure the availability of low pressure coolant injection systems, and then override the automatic reclosure function and hold the valves open until blowdown is complete or the valves reclose due to insufficient air pressure for the valves. To initiate blowdown through the valves, the operator may also have to restore the air supply which controls them. Containment isolation, which occurs on low vessel water level, may isolate the air supply to these valves on some plants. Hence, the operator would have to override containment isolation to take this action.

Another way to depressurize the system would be to depressurize using the pressure regulator directing steam into the condenser. This too would require the operator to override containment isolation. By manually reopening the MSIVs and bypass valve, steam could be vented through main steam lines using the turbine bypass system. In addition the Reactor Water Cleanup (RWCU) System can be aligned to blowdown to the main condenser. However, since the system isolates on low vessel water level, operator action would be required to override isolation as well as realign the system from the normal recirculation cleanup configuration to the blowdown mode.

Depending on the failure mode of the high pressure injection systems, it may also be possible to utilize these systems to depressurize. The operator could release steam through the RCIC and/or HPCI turbines even though they may not be providing coolant to the core. The rate of pressure relief provided by this action is uncertain, but over a period of minutes could be effective in lowering pressure below 300 psi.

If the operator is unsuccessful in reducing system pressure (state 5a), the core will uncover and eventually begin to melt. The onset of melting is predicted to occur within approximately one half hour after the beginning of the transient. Operator response would then be to ensure operation of containment systems and monitor the progression of the melt as needed to effectively mitigate the consequences.

As *noted earlier, the "V" failure event in this sequence could be* either a failure to depressurize or a failure of the low pressure injection systems to provide adequate flow following depressurization. In the former case, any of the previously discussed actions, if successful, would put the plant in state 5. From this condition, it is assumed that LP ECCS functions as designed and meets or exceeds the minimum makeup requirements. The delay in depressurizing the system as a result of the extraordinary operator action would not affect the operation of these systems significantly. The operator response for this scenario would be to monitor the operation of the LPCIS and CSIS, and control these systems as required to keep the core cooled. Once conditions have stabilized the RHRS would need to be realigned for long term heat removal.

If the ADS or operator action successfully reduces the system pressure (state 5), but the low pressure ECCS fails (state 6a) the operator must find a way to deliver water to the core. It is possible that one of these systems could be temporarily bypassed, but not functionally disabled or spurious trip signals have resulted in the unavailability of the pumps. In such cases, alert operator response could return the system to operation in time to prevent core melt (state 7). This action naturally depends on a specific LP ECCS failure mode and would require that the operator quickly recognize the unavailability of the system and be capable of quickly restoring the system.

Another potential action at this point would be to restore flow from the feedwater/condensate system. With the system at low pressure, the main feed pumps will not be required and the condensate booster pumps can provide flow to the core from the hotwell. If the operator successfully restores feedwater (state 8), then an adequate source of water must be continuously available. If the other components of the power conversion system (PCS) can be utilized, water can be drawn from the condenser hotwell and heat rejected via the the turbine bypass system. Condenser vacuum would have to be re-established to remove heat in this manner. If PCS can not rapidly be activated the operator would have to monitor the water level in the hotwell and supplement it as necessary using the condensate storage tank inventory.

4.2.1.3 Symptom Matrix

The symptoms exhibited by the plant throughout the progression of the TQUV sequence are tabulated in Table 4.3. The behavior of each of the key BWR parameters (delineated and discussed in Section 3) is described in Table 4.3 for each state appearing in the operator action event tree (Figure 4.2). To illustrate how to read Table 4.3, consider the first row, which addresses the plant symptoms exhibited at OAET states TQUV 1,2. A "1" in the Table indicates the presence of the corresponding symptoms listed across the top of the Table. A "0" in the Table indicates that that symptom is not expected to be observed at that state. For example, at OAET states TQUV 1,2, the RPV Level is expected to be below the Low Level Scram Setpoint (Symptom #2, as listed in Table 3.1, BWR EPG Parameters and Symptoms). The RPV Level is not expected, however, to be below ADS Initiation Setpoint or Top of Active Fuel (Symptom #3).

As discussed in Section 2, multiple symptoms involving the same parameter may be indicated in Table 4.3. These multiple (and sometimes contradictory) symptoms are due to either:

1) a state where the parameter is going through a transition from one trend to another; in these cases the operator may actually observe either or both symptoms.

2) a state where the physical plant response (e.g. rapid depressurization) may affect the symptoms observed by the operator (e.g. reactor water level); in these cases indications of these symptoms may be unreliable.

3) a state where the behavior of a certain parameter is simply unknown or highly uncertain.

If potential ambiguities are identified involving OAET states with such multiple symptoms, these few cases can be examined individually in more detail. The effects of the use of such multiple symptoms in Table 4.3 on the symptoms comparison process are discussed further in Section 2.

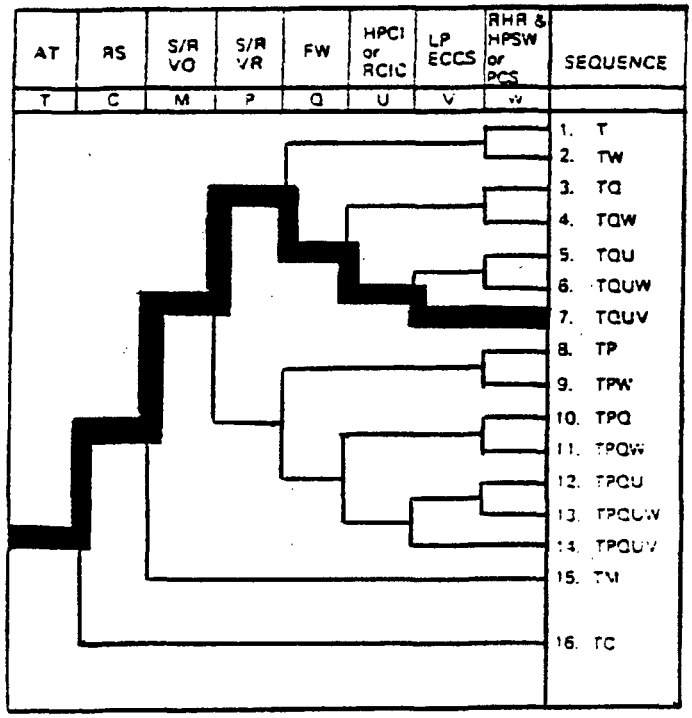
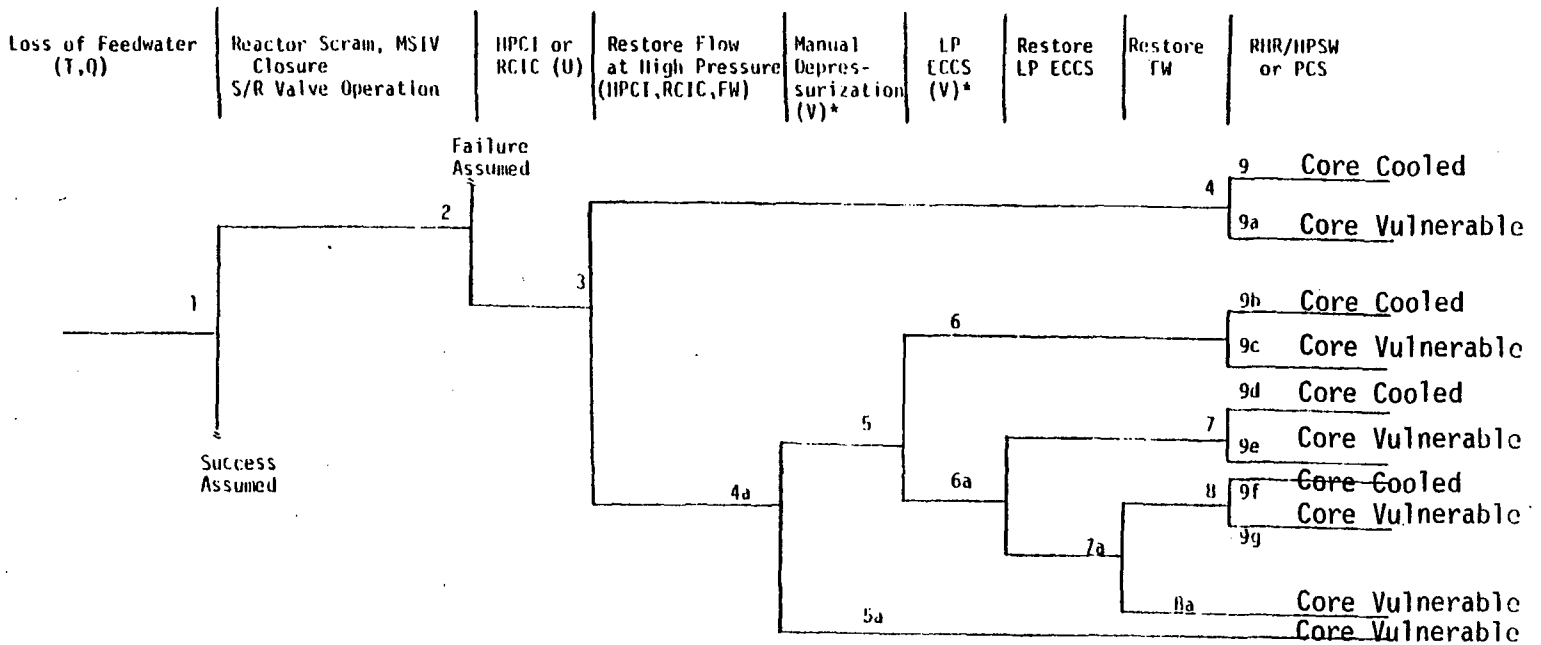


Figure 4.1 Reactor Safety Study Event Tree for BWR Transient Events (Event T): TQUV Sequence Highlighted.



* V Failure event can result from either a failure to depressurize or a failure of the low pressure ECCS's.

Figure 4.2 TQUV Sequence - Operator Action Event Tree

Table 4.3

Sequence: TQUV

BWR OAET Symptom Matrix Sheet 1

Symptoms 1-19

(See Section 4.2.1.3 for explanation of Symbols)

Symptoms	RPV Level				RPV Press.									RPV Temp		Reactor Pwr.		Rod Pos.	Boron
OAET State	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19
TQUV-1,2	0	1	0	0	1	0	1	0	0	1	0	1	1	0	0	1	0	1	0
TQUV-3	0	1	1	0	1	0	1	0	0	1	0	1	1	0	0	1	0	1	0
TQUV-4	1	1	1	0	1	0	1	0	0	1	0	1	1	0	0	1	0	1	0
TQUV-4a	0	1	1	0	1	0	1	0	0	1	0	1	1	0	0	1	0	1	0
TQUV-5	0	1	1	1	0	1	0	1	1	0	1	1	1	1	0	1	0	1	0
TQUV-5a	0	1	1	1	1	0	1	0	0	1	0	1	1	0	0	1	0	1	0
TQUV-6	1	1	1	0	0	1	0	1	1	0	1	1	1	0	0	1	0	1	0
TQUV-6a	0	1	1	1	0	1	0	1	1	0	1	1	1	0	0	1	0	1	0
TQUV-7	1	1	1	0	0	1	0	1	1	0	1	1	1	0	0	1	0	1	0
TQUV-7a	0	1	1	1	0	1	0	1	1	0	1	1	1	0	0	1	0	1	0
TQUV-8	1	1	1	0	0	1	0	1	1	0	1	1	1	0	0	1	0	1	0
TQUV-8a	0	1	1	1	0	1	0	1	1	0	1	1	1	0	0	1	0	1	0

Table 4.3 (Continued)

Sequence: TQUV

BWR OAET Symptom Matrix Sheet 2

Symptoms 20-44

Symptoms	S.P. Level										S.C. Press					S.P. Temp					DW Prs	Drywell Temp.			
OAET State	20	21	22	23	24	25	26	27	28	29	30	31	32	33	34	35	36	37	38	39	40	41	42	43	44
TQUV-1,2	0	0	1	0	1	0	0	0	1	0	0	0	0	0	1	0	1	1	0	1	0	0	0	1	0
TQUV-3	0	0	1	0	1	1	0	0	1	0	0	0	0	0	1	0	1	1	0	1	1	0	0	1	1
TQUV-4	0	0	1	0	1	1	0	0	1	0	0	0	0	0	1	0	1	1	0	1	1	0	0	1	0
TQUV-4a	0	0	1	0	1	1	0	0	1	0	0	0	0	0	1	0	1	1	0	1	1	0	0	1	1
TQUV-5	0	0	1	0	1	1	0	0	1	0	0	0	0	0	1	0	1	1	0	1	1	1	0	1	1
TQUV-5a	0	0	1	0	1	1	0	0	1	0	0	0	0	0	1	0	1	1	0	1	1	0	0	1	1
TQUV-6	0	0	1	0	1	1	0	0	1	0	0	0	0	0	1	0	1	1	0	1	1	1	0	1	1
TQUV-6a	0	0	1	0	1	1	0	0	1	0	0	0	0	0	1	0	1	1	0	1	1	1	0	1	1
TQUV-7	0	0	1	0	1	1	0	0	1	0	0	0	0	0	1	0	1	1	0	1	1	1	0	1	1
TQUV-7a	0	0	1	0	1	1	0	0	1	0	0	0	0	0	1	0	1	1	0	1	1	1	0	1	1
TQUV-8	0	0	1	0	1	1	0	0	1	0	0	0	0	0	1	0	1	1	0	1	1	1	0	1	1
TQUV-8a	0	0	1	0	1	1	0	0	1	0	0	0	0	0	1	0	1	1	0	1	1	1	0	1	1

Table 4.3 (Continued)

Sequence: TQUV

BWR OAET Symptom Matrix Sheet 3

Symptoms 45-65

Symptoms	PC Level	PC Temp	HPCI		RCIC		LPCI		CS		DG		RHR		ADS			MC		CRD	
OAET State	45	46	47	48	49	50	51	52	53	54	55	56	57	58	59	60	61	62	63	64	65
TQUV-1,2	0	0	1	1	1	1	1	1	1	1	1	0	1	0	0	0	1	0	1	1	0
TQUV-3	0	0	0	1	0	1	1	1	1	1	1	0	1	0	0	1	0	0	1	1	0
TQUV-4	0	0	1	0	1	0	1	1	1	1	1	0	1	0	0	1	1	1	1	1	0
TQUV-4a	0	0	0	1	0	1	1	1	1	1	1	0	1	0	1	1	0	0	1	1	0
TQUV-5	0	0	0	1	0	1	1	1	1	1	1	0	1	0	1	0	0	0	1	1	0
TQUV-5a	0	0	0	1	0	1	1	1	1	1	1	0	1	0	1	1	0	0	1	1	0
TQUV-6	0	0	0	1	0	1	1	0	1	0	1	0	1	0	1	0	0	0	1	1	0
TQUV-6a	0	0	0	1	0	1	0	1	0	1	1	0	1	0	1	0	0	0	1	1	0
TQUV-7	0	0	0	1	0	1	1	0	1	1	1	0	1	0	1	0	0	1	1	1	0
TQUV-7a	0	0	0	1	0	1	0	1	0	1	1	0	1	0	1	0	0	0	1	1	0
TQUV-8	0	0	0	1	0	1	0	1	0	1	1	0	1	0	1	0	0	0	1	1	0
TQUV-8a	0	0	0	1	0	1	0	1	0	1	1	0	1	0	1	0	0	0	1	1	0

Table 4.3 (Continued)

Sequence: TQUV

BWR OAET Symptom Matrix Sheet 4

Symptoms 66-79

Symptoms	SLC		RWCU		Alternate Inj.		C/F		Recirc. Pumps		MSIV		RHR Interlock	
OAET State	66	67	68	69	70	71	72	73	74	75	76	77	78	79
TQUV-1,2	1	0	0	1	1	0	0	1	1	0	0	1	0	1
TQUV-3	1	0	0	1	1	0	0	1	1	0	0	1	0	1
TQUV-4	1	0	0	1	1	0	1	1	1	0	0	1	0	1
TQUV-4a	1	0	0	1	1	0	0	1	1	0	0	1	0	1
TQUV-5	1	0	0	1	1	0	1	1	1	0	0	1	0	1
TQUV-5a	1	0	0	1	1	0	0	1	1	0	0	1	0	1
TQUV-6	1	0	0	1	1	0	1	1	1	0	0	1	0	1
TQUV-6a	1	0	0	1	1	0	1	1	1	0	0	1	0	1
TQUV-7	1	0	0	1	1	0	1	1	1	0	0	1	0	1
TQUV-7a	1	0	0	1	1	0	1	1	1	0	0	1	0	1
TQUV-8	1	0	0	1	1	0	1	1	1	0	0	1	0	1
TQUV-8a	0	0	0	1	1	0	1	1	1	0	0	1	0	1

4.2.2 Failure to Remove Decay Heat after Transient Event (TW)

The previous section discussed the physical plant response and operator actions associated with postulated accident sequences involving an inability to maintain adequate coolant inventory following a transient-induced shutdown. The present section presents similar discussions for a transient-induced shutdown in which inventory is maintained, but postulated failures occur which result in an inability to remove decay heat from the containment and suppression pool. This sequence was designated TW in the Reactor Safety Study. A summary of the accident sequence is provided in Section 4.2.2.1. The Operator Action Event Tree constructed for this sequence is provided in Section 4.2.2.2. The physical plant response at each state in the OAET is presented in Section 4.2.2.3 in terms of the key BWR parameters discussed in Section 3.

4.2.2.1 Sequence Description

As for the TQUV sequence discussed in Section 4.2.1, the initiator for this sequence is postulated to be a loss of all feedwater flow to the vessel. The initiator could be any transient event which results in a need for reactor shutdown. A loss of feedwater initiator is chosen because of its probabilistic importance as well as the immediate requirements it places upon the high pressure injection systems.

As described in the Reactor Safety Study, the TW sequence is initiated by a transient event (T) which requires a plant shutdown. The emergency coolant injection systems operate as required to maintain inventory, but the RHR fails to remove decay heat from the suppression pool (event W). Due to this failure, the suppression pool temperature increases, resulting in a steady rise in containment pressure until eventually the design pressure is reached and then exceeded. A failure of the containment is assumed. The rapid depressurization causes the water in the suppression pool to flash, and cavitation occurs in the pumps which are drawing suction from the pool. If coolant inventory in the core is not maintained by some source outside of containment, the core will uncover and melt.

Table 4.4 is a compilation of key events which may occur during the TW sequence. Figure 4.3 is the system event tree developed in the RSS for transient events. The TW sequence is highlighted. A more detailed event tree which displays the operator actions in conjunction with the key plant states is developed in Section 4.2.2.2.

The initial response to a loss of feedwater event is described in some detail in Section 4.2.1.1 for the TQUV sequence. As the feedwater flow ramps to zero, the level in the vessel drops to the low level setpoint (level 3), at which time a scram signal is generated. The level continues to fall to level 2, at which time the MSIVs are closed and the HPCI and RCIC systems are signalled to start. For the TQUV sequence described previously, it was postulated that these systems were unavailable. However, for the TW sequence it is postulated that the HPCI and RCIC systems maintain inventory in the vessel. The primary system begins to cooldown and depressurize.

When the vessel pressure drops below the shutoff head of the low pressure ECCS's (around 450 psig), these systems begin to provide coolant to the vessel. As the pressure continues to fall, the high pressure systems eventually trip (at approximately 100 psig). Coolant inventory control is then dependent upon the operation of the LPCI and core spray systems. Throughout this process, decay heat is transferred directly to the suppression pool, and the temperature of this pool rises.

For the TW sequence it is postulated that the Low Pressure Coolant Recirculation System (LPCRS) fails, thus resulting in a failure to cool the torus. The LPCRS normally circulates water from the suppression pool through RHR heat exchangers where the pool water is cooled by plant service water and then returned to the pool. Failure to provide this circulation and/or heat removal results in a continuing rise in suppression pool temperature and chamber pressure. If torus cooling is not reestablished, the temperature and pressure in the drywell will also increase. Without torus cooling it is postulated that the containment structure will eventually fail. As a result, injection of coolant into the vessel from the suppression pool will fail, and the core could uncover unless another source of water (from outside of the containment) is established.

Table 4.4

Key Sequence of Events in "TW" Sequence

Time*	Event
0.0	Initial vessel level = 46.75 ft., TAF = 30.03 ft. Trip of all feedwater pumps initiated.
5.0	Feedwater flow decays to zero.
15	Narrow range sensed water level reaches low level scram (Level 3). Reactor scram is initiated. All primary system isolation valves except the main steam line isolation valves (e.g., RHR shutdown isolation valves, RWCU isolation valves, and containment) are initiated to close. Automatic depressurization permissive on level.
34.0	Wide range sensed water level reaches low low water level (Level 2). Recirculation pumps are tripped. The main steam isolation valves are closed. RCIC and HPCI systems are initiated.
39.0	MSIV's fully closed.
64.0	HPCI and RCIC flow starts to enter the vessel, in shroud level bottoms out 12.6 ft above TAF and starts to rise.
93	Group 1 safety/relief valves start to open.
97	Group 1 safety/relief valves completely closed.
120	Group 1 safety/relief valves start to open again. Valves continue to cycle on and off on setpoints unless operator takes action to prevent cycling.
230	Group 1 safety/relief valves cease cycling due to HPCI and RCIC steam extraction.
380	Water level approaches normal water level. The operator takes manual control of HPCI and RCIC to maintain normal water level.
1000	Vessel depressurized, LPCI injection maintains water level.
1200	Failure of pool cooling and shutdown cooling modes of RHRS.
2 hours	Pool temperature exceeds 49°C (120°F).
5-7 hours	Suppression pool temperature exceeds 212°F.

Table 4.4 (continued)

Time*	Event
18 hours	Drywell coolers fail due to high drywell temperature and pressure.
23 hours	Containment pressure ∿90 psia. Suppression pool temperature ∿320°F.
23-27 hours	RPV repressurizes.
28-35 hours	Drywell fails when internal pressure exceeds failure limit (132 psia for Mark I)**

* Seconds unless otherwise noted.

** The Reactor Safety Study used a failure pressure of 177 psia, estimated to be reached in 27 hours.

In addition to threatening the containment integrity, the rising drywell pressure has one other effect. When the drywell pressure increases sufficiently, it is postulated that any of the safety/relief valves which are being held open by the operator will be forced closed due to backpressure, insufficient differential pressure across the valves, or failure of the air supply. The vessel pressure will begin to rise until it exceeds the shutoff head of the low pressure pumps. Flow from the low pressure ECCS's would then halt until the operator establishes another method of depressurization. With the core "bottled up", the vessel pressure would continue to rise until it once again reaches the safety setpoint of the SRVs, at which time it would cycle about this point. With the operator unable to depressurize the vessel at this time, a high pressure system could be required to replace coolant flowing through the SRVs. If coolant inventory is not maintained, the core will eventually uncover and fuel damage will occur. Hence, the core could be damaged even before the containment fails.

4.2.2.2 Operator Response

The discussion in Section 4.2.2.1 focused on a "hands off" description of the TW sequence following postulated system failures. This section will present operator actions which can be taken in response to the system failures. A detailed operator action event tree is included as Figure 4.4 for the TW sequence.

As depicted in the OAET and as is discussed below, the principal operator actions can be categorized as follows:

1) Identify loss of feedwater (FW)

2) Verify occurrence of automatic system responses to loss of FW (e.g. scram, recirculation pump trip, HPCIS initiation, etc.)

3) Verify inventory control with HPCI, RCIC, LPCI, CSIS

4) Identify failure of torus cooling (LPCRS failure)

5) Lower vessel pressure, maintain or restore vessel coolant inventory

6) Restore torus cooling with LPCRS or establish decay heat removal through PCS

The initial operator action is to identify the initiating transient so that the correct actions to bring the plant to a safe shutdown can be implemented. The loss of feedwater will be immediately characterized by a reduction in vessel water level and a decrease in flow through the feedwater piping. To distinguish this event from a feedwater line break, the drywell or containment atmosphere conditions can be checked as they will be unaffected by the loss of feedwater.

As noted in Section 4.2.1.1, the plant instrumentation and control system should detect the changes in vessel level and feed flow and automatically reduce recirculation flow. Within a few seconds after a total loss of feedwater, the vessel low water level setting (Level 3) is reached and the reactor is scrammed. The vessel level will continue to decrease and will shortly reach the low-low level set point (Level 2) which initiates a recirculation pump trip, actuation of the HPCIS and RCIC, and in some plants, the closure of the MSIVs. In this analysis, MSIV closure at Level 2 is assumed.

For the TW sequence it is assumed that all of these events occur as expected (states 1 and 2 of Figure 4.4). The operator must verify the reactor scram by checking the flux and control rod position, verify the rundown and subsequent termination of the recirculation pumps, and verify MSIV closure. MSIV closure will cause the pressure to increase and within several seconds the safety/relief valve set point will be reached. Successful operation of one or more of these valves will limit the system pressure rise. The operator should minimize safety valve cycling by holding open one of the valves until the system pressure is below the safety valve setpoints. The operator should monitor the reactor pressure to ensure that the valves are performing their function (successful opening and reclosing of the safety relief valves is assumed in the TW sequence).

Another operator response to the loss of feedwater transient is to verify automatic actuation of the HPCI and RCIC. For the postulated TW sequence, both HPCI and RCIC are available and operate as required to maintain inventory.

While the HPCIS is maintaining level, the vessel pressure will be decreasing. Successful inventory maintenance is indicated by state 3 of the OAET for this sequence. With the RCIC and/or HPCI systems maintaining vessel level, the fission product decay heat transported to and stored in the suppression pool via the safety/relief valves and HPCI/RCIC discharge will result in a gradual increase in the pressure in the primary containment. The containment cooling mode of the RHR system is designed to limit the suppression pool water temperature. This mode of the RHR system, which is manually initiated, takes suction from the suppression pool, pumps the fluid with any of four low pressure pumps through the RHR system heat exchangers, and returns the cooled fluid back to the suppression pool. The Service Water (SW) system is used to transport the heat from the RHR heat exchangers to the environment.

In the TW sequence it is postulated that suppression cooling fails and the operator is unable to immediately restore it to operation (state 4). Therefore, the suppression pool temperature continues to rise. An important point to note in considering operator actions at this stage in the accident sequence is that there is a maximum suppression pool temperature above which complete condensation of blowdown steam may not occur. The operator should be aware that once this temperature limit is reached, primary system blowdown could subject the primary containment to a pressure transient which could threaten its structural integrity. In order to prevent this situation, the operator should reduce primary system pressure to approximately 100-200 psi when the suppression pool temperature rises above 120°F.* It is assumed for this analysis that the operator is successful in these actions (state 5).

*Because the system has been depressurizing all along due to the successful maintenance of coolant inventory and the continued discharge through the HPCI/RCIC and SRV discharge, the pressure drop at state 5 due to depressurization to 100-200 psi may not be severe.

Following reactor depressurization the operator should verify the operation of the inventory control systems (LPCI, RCIC, etc.). If the LPCI pumps are not available (their failure may have resulted in the suppression pool cooling failure), the preferred course of action would be to maintain coolant injection with either:

(a) the RCIC aligned to the CST to maintain adequate lube oil cooling despite potentially high suppression pool levels; or

(b) the CRD system also aligned to the CST*

Once inventory maintenance is assured (state 6), the operator must continue his attempts to restore torus cooling with the RHR or PCS. Once again it is postulated that the operator cannot restore suppression pool cooling. If suppression pool cooling has been unsuccessful, one of the operator's first responses would be to attempt cooling with the PCS.

In order to restore the PCS, the operator would be required to reopen the MSIVs that closed on low level, take the required actions to verify or re-establish vacuum in the condenser and perform the control actions to establish steam flow from the vessel through the bypass valves to the condenser. Many of these actions will be similar to those involved in a start-up procedure. Perhaps one of the more difficult problems the operator might encounter in restoring PCS is starting up the main feedwater pumps under the conditions of low steam flow to the turbine. The electrically driven condensate pumps can be used for returning water to the vessel but the primary system pressure must be appreciably reduced for these pumps to deliver adequate flow. Therefore, this option may not be available if the operator cannot maintain depressurization of the system. Opening the MSIVs may provide enough initial depressurization for this method to be successful. Depressurization through the pressure regulators may also be useful. (Due to high containment back pressure, the operator may be unable to open the MSIVs, as described below.) At state 7 it is also postulated that the operator is unsuccessful in completely restoring the PCS to service. The MSIVs remain closed.

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*If the low pressure systems are available they may be used unless the system repressurizes. Note also that while HPCI operation can continue the operation may be somewhat degraded if the suppression pool temperature exceeds the recommended design lube oil cooling temperature of 170°F.

Without cooling, the temperature of the suppression pool continues to slowly rise. The pressure of the wetwell air space also rises until it exceeds the pressure of the drywell and the vacuum breakers open. At that time the drywell containment pressure will then begin to slowly rise. Eventually, the pressure will cause the high back pressure trip of both the RCIC (around 25 psig at approximately 14-17 hours, depending upon the drywell cooler status and the effectiveness of the sprays*) and the HPCI (about 65 psig at approximately 20-26 hours). If this occurs, only the control rod drive pumps (a high pressure injection system) and the low pressure systems would be available for inventory control.

Up until state 7 the reactor system has been depressurizing through the HPCI/RCIC discharge and the safety valves. The suppression pool is continuing to act as a heat sink and the containment temperature and pressure are rising. At this point the ability to maintain depressurization through the safety/relief valves becomes questionable for one of several reasons. The air-operated safety valves may close due to failures of the air supply, failure to maintain sufficient differential pressure across the valves, or failure of the valve components themselves, as follows.

To maintain depressurization, the operator is instructed to hold one SRV in the open position. To do this requires that there be adequate air pressure from the accumulators or air compressors to actuate the pilot-operated solenoids which are used for the "relief" portion of the safety/ relief valves. If the differential pressure across the valve actuators is less than approximately 30-50 pounds (between the air system and the containment), the valves will close and cannot be re-opened manually. With typical accumulator pressures of 100 psia, containment pressures in excess of about 70 psia are likely to compromise the ADS air supply effectiveness. Since the containment pressure without RHR or PCS recovery will rise to above this

*The operator should initiate suppression spray operation after noticing the continuing increase in suppression chamber pressure. He should initiate drywell sprays if the drywell temperature approaches 340°F.

"threshold", it is very possible that the air supply will be inadequate to maintain the valves in an open position. The high containment temperature and humidity may also quickly exceed the environmental qualification envelope for the air operated SRVs, resulting in failures of components necessary for the valves to remain open (see Reference 5).

For the above reasons it is postulated that the safety valves fail to remain open at state 7. The operator must either establish an alternate pathway to relieve reactor pressure or restore a high pressure system to operation. The possible alternative depressurization paths are to the main condenser or through the HPCI and RCIC system turbines, while the high pressure systems capable of maintaining inventory are the HPCIS, RCICS, CRD, and the feedwater system.

The most likely response at state 7 will be for the operator to attempt to open the MSIVs or the drain valves which bypass the MSIVs, even if he cannot restore the main condenser to operation. If the condenser cannot be placed in service before these valves are opened, the seals on the condenser may burst and create a leak path outside of containment. However, if the operator does not depressurize and remove heat through the MSIVs, the vessel pressure will rise to the safety valve setpoints, and these valves will open. Energy will then be added to an already hot and pressurized containment, and the likelihood of containment failure increases. The operator should prevent this, and should therefore attempt depressurization through the MSIVs or the drain valves. However, the same problem which caused the safety valves to close in the first place (high differential pressure) will also affect the ability to open the MSIVs. These valves also require air pressure to open, and may not open if there is a high containment backpressure.

If the main steam valves or drain valves cannot be opened, the operator can attempt depressurization through the HPCI or RCIC steam lines. This may require some rather complex actions. The HPCI and RCIC lines may be isolated due to the high temperature which exists in the areas containing the pumps. Thus the operator may need to bypass some isolation valve interlocks before he can release steam through these lines. Since defeating the interlocks

may mean entering these same high temperature areas, it may not be possible to bypass them. Even if possible, the operator may be very hesitant to bypass these lines and increase the potential for release outside of containment.

If the operator has been successful in depressurizing the vessel, as indicated by state 8 of the OAET, he must still maintain coolant inventory. This can be provided by the low pressure ECC systems, if available, or the condensate system. The condensate system draws suction from the hotwell and provides water to the vessel through the feedwater lines. The hotwell inventory can be maintained by the CST, which in turn can be provided with makeup water by the fire water system. It is postulated that given the operator's success in maintaining inventory up to this point, continued inventory makeup will also be successful (state 9). However, the operator must still restore the RHR or PCS to operation to ensure continued core coolability.

It may be possible that depressurization through the MSIVs, if this method was used at state 8, was accompanied by restoration of the main condenser. The plant would then be in state 10 as a direct result of the actions taken to reach state 8. If the PCS is in service, the majority of decay heat will bypass the suppression pool and be removed by the main condenser instead. The rate of increase in suppression pool temperature and containment pressure should decline. Some form of torus cooling may eventually be required, but it is postulated here that if the operator successfully re-establishes heat removal with the PCS, this will be sufficient to prevent over-pressurization of the containment. One caution must also be included here. If cooling of the core is being accomplished with the addition of water from an outside source (i.e., no recirculation of suppression pool water is occurring), the suppression pool water level may continue to rise due to HPCI/RCIC discharge or spray operation. Eventually it will rise to the point where structural integrity may be threatened. If recirculation cannot be restored, the containment may leak or fail. If cooling is not continued to the core, fuel damage may occur. One possible solution if this situation occurs is for the operator to continue to cool the core, but at a minimum injection rate, while also continuing to repair a low pressure recirculation system. Recirculation of suppression pool water should be implemented as soon as possible.

In the unlikely event that the RHR is not also restored by the time that the PCS has reduced primary pressure to below 50 psia, the normal switch-over to RHR cool down could not be accomplished (see discussion which follows concerning RHR repair). Continued use of the main condenser below this pressure would result in turbine shaft gland seal leakage, loss of condenser vacuum, and eventual blowout of the condenser rupture disks. Cooling could continue in this mode but a pathway would exist between the primary system and the turbine building enclosure, as described previously for state 8.

If the operator has not restored the PCS to service at states 8 or 9, the other principal alternative for restoring containment heat removal is the repair of the RHR system. There are however some risks associated with the RHR restoration if it occurs very late in the accident sequence. For example, with the suppression pool at elevated temperatures, i.e., in the range of 300°F for times greater than 20 hours, there may be some difficulty in the initiation of effective RHR pool cooling at this point. Specifically, as the suppression pool fluid is pumped by the RHR pumps through the RHR heat exchangers there may be some flashing (this is possible but unlikely if the RHR system is maintained at sufficiently high pressure). If flashing does occur it may come in contact with cool water present in the RHR piping or in the RHR heat exchangers. This contact could result in violent condensation and, in the extreme, damage to the RHR system. Pressures and temperatures within the RHR system may be extremely important in this operation of RHR restoration under elevated containment temperatures.

Successful restoration of PCS or RHR operation is indicated by state 10 on the OAET. At this state the operator must monitor the approach to cold shutdown conditions. At state 10a the operator has not been able to establish containment cooling. Although reactor coolant inventory is being maintained, the core is vulnerable to damage due to the potential for containment failure. In addition, the containment conditions could affect the ability to maintain inventory, depending upon the method being used to provide coolant to the core. If suction is being taken from the pool, these pumps could cavitate due to a loss of net positive suction head. Other components (instrumentation and control) could fail if their environmental qualification limits are exceeded. If core damage occurs, it will be at low pressure in state 10a.

Failure of the operator to maintain the vessel at low pressure is indicated by state 8a. At this state the operator did not or could not depressurize the vessel through the MSIVs, the drain valves, or the HPCI/RCIC turbines. With the MSIVs closed the reactor vessel has repressurized to above the shutoff head of the low pressure ECCS pumps. Eventually the vessel pressure will rise to the safety valve setpoints, and these valves will open to release pressure (and steam). The vessel water level will then drop. The operator must therefore maintain inventory by placing a high pressure system into service. The HPCIS pumps have previously tripped due to high backpressure or high temperature in the pump areas. The feedwater system may not be restored yet following the initiating trip, or, if repaired, may not be useful due to closure of the MSIVs. It is unlikely that the operator will be able to restore one of these systems to operation at this time. However, he should continue his attempts to restore these systems, while also supplying whatever coolant possible with the CRD and RCIC pumps (these latter two systems should be capable of maintaining inventory at the low decay heat levels present at this state). If coolant inventory maintenance is unsuccessful, the core may eventually uncover, resulting in fuel damage.

It is postulated that the operator does provide sufficient high pressure coolant makeup to keep the core covered (state 9a). The source of coolant is most likely the CRD system. Actions taken from this point on are analogous to those taken at state 9. Failure to restore some form of containment cooling may eventually result in containment failure, and possible core damage. Core damage at state 10c, if it occurs, will be at high pressure.

4.2.2.3 Symptom Matrix

The symptoms exhibited by the plant at each of the states on the OAET for the TW sequence are tabulated in Table 4.5. The tabulation is presented as a function of the key parameters discussed in Section 3.

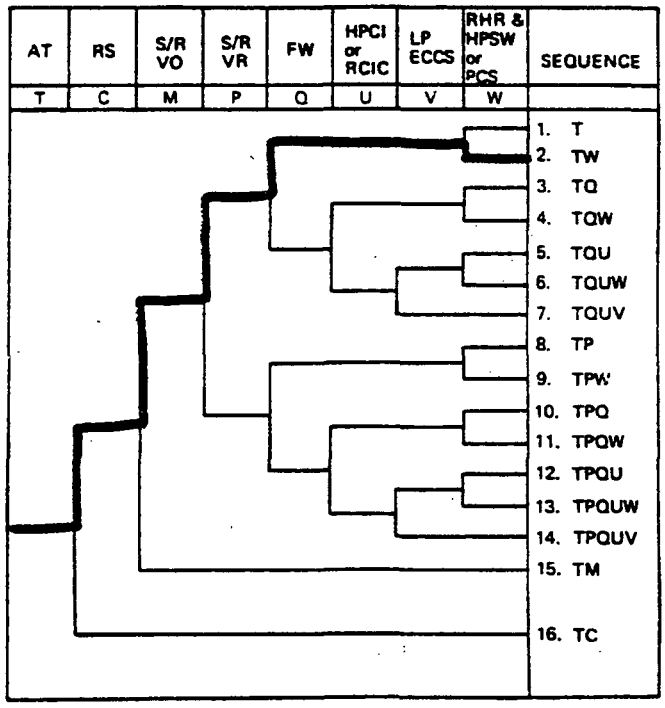

Figure 4.3 Reactor Safety Study Event Tree for BWR Transient Event : TW Sequence Highlighted

Figure 4.4 TW Sequence - Operator Action Event Tree

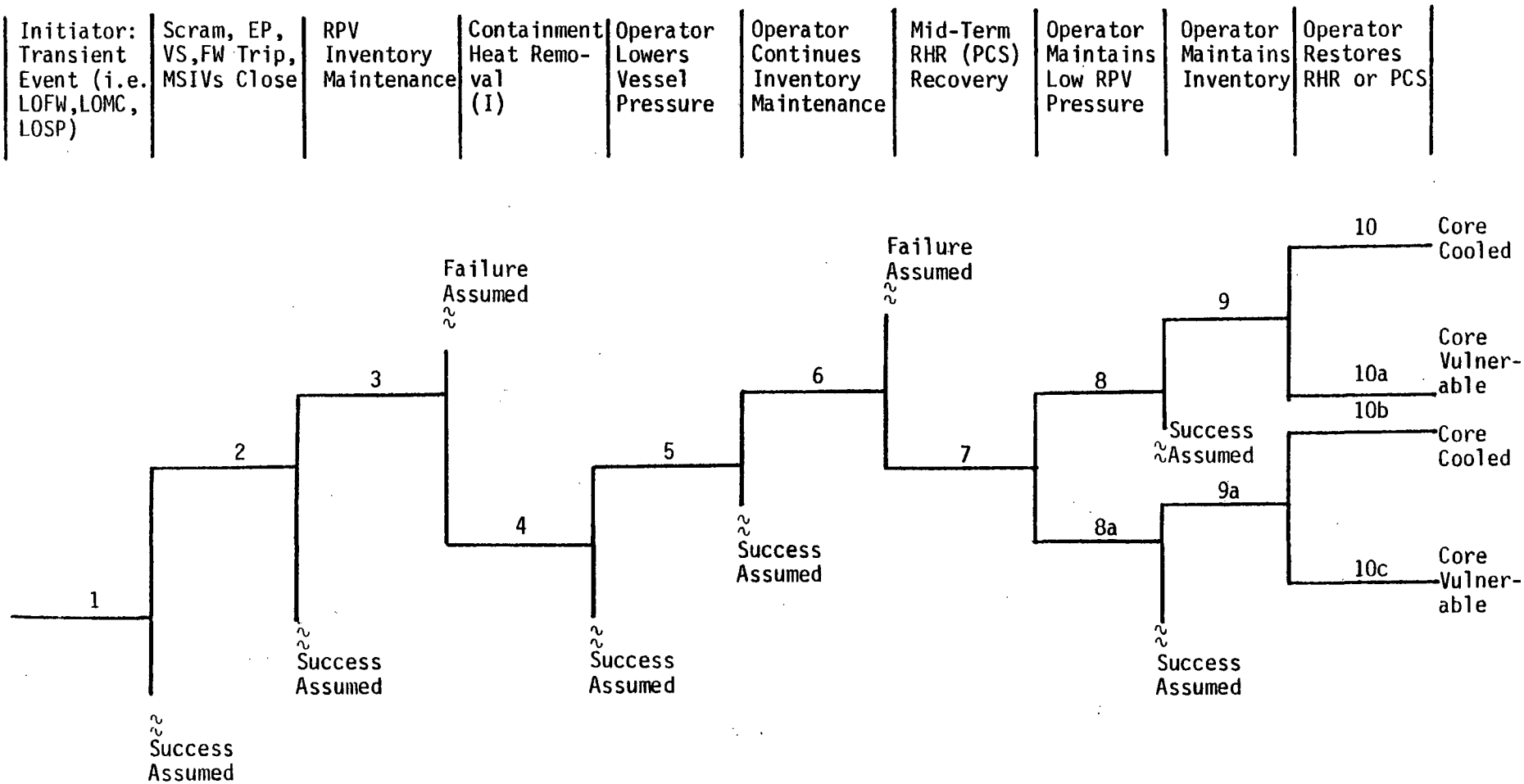

Table 4.5

Sequence: TW

BWR OAET Symptom Matrix Sheet 1

Symptoms 1-19

(See Section 4.2.1.3 for explanation of Symbols

Symptoms	RPV Level				RPV Press.									RPV Temp		Re-actor Pwr.		Rod Pos.	Bo-ron
OAET State	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19
TW-1,2	0	1	0	0	0	1	0	1	0	1	0	1	1	0	0	1	0	1	0
TW-3	0	0	0	0	1	1	1	0	0	1	0	1	1	0	0	1	0	1	0
TW-4	0	0	0	0	1	1	1	0	0	1	0	1	1	0	0	1	0	1	0
TW-5	0	0	0	0	0	1	0	1	1	0	1	1	1	1	0	1	0	1	0
TW-6	0	0	0	0	0	1	0	1	1	0	1	1	1	0	0	1	0	1	0
TW-7	0	0	0	0	0	1	0	1	1	0	1	1	1	0	0	1	0	1	0
TW-8	0	1	0	0	0	1	0	1	1	0	1	1	1	0	0	1	0	1	0
TW-8a	0	1	0	0	1	1	1	0	0	1	0	1	1	0	0	1	0	1	0
TW-9	0	0	0	0	0	1	0	1	1	0	1	1	1	0	0	1	0	1	0
TW-9a	0	0	0	0	1	1	1	0	0	1	0	1	1	0	0	1	0	1	0
TW-10	0	0	0	0	0	1	0	1	1	0	1	1	1	0	0	1	0	1	0
TW-10a	0	1	1	0	0	1	0	1	1	0	1	1	1	0	0	1	0	1	0
TW-10b	0	0	0	0	1	1	1	0	0	1	0	1	1	0	0	1	0	1	0
TW-10c	0	1	1	0	1	1	1	0	0	1	0	1	1	0	0	1	0	1	0

Table 4.5 (Continued)

Sequence: TW

BWR OAET Symptom Matrix Sheet 2

Symptoms 20-44

Symptoms	S.P. Level										S.C. Press					S.P. Temp					DW Prs	Drywell Temp.			
OAET State	20	21	22	23	24	25	26	27	28	29	30	31	32	33	34	35	36	37	38	39	40	41	42	43	44
TW-1,2	0	0	1	0	1	0	0	0	1	0	0	0	0	0	1	1	0	1	0	1	0	0	0	1	1
TW-3	1	0	1	0	1	1	0	0	1	0	0	0	0	0	1	1	1	1	0	1	0	0	0	1	1
TW-4	1	0	1	0	1	1	0	0	1	0	0	0	1	1	1	1	1	1	0	1	0	0	0	1	1
TW-5	0	0	1	0	1	1	0	0	1	0	0	0	1	1	0	1	1	1	0	1	1	0	0	1	1
TW-6	0	0	1	0	1	1	0	0	1	0	0	0	1	1	0	1	1	1	0	1	1	0	0	1	1
TW-7	0	0	1	0	1	1	0	0	1	0	0	0	1	1	0	1	1	1	1	1	1	0	1	1	1
TW-8	0	0	1	0	1	1	0	0	1	0	0	0	1	1	0	1	1	1	1	1	1	0	1	1	1
TW-8a	1	0	1	0	1	1	0	0	1	0	0	0	1	1	0	1	1	1	1	1	1	0	1	1	1
TW-9	0	1	1	1	1	1	0	0	1	0	0	0	1	1	0	1	1	1	1	1	1	0	1	1	1
TW-9a	1	1	1	1	1	1	0	0	1	0	0	0	1	1	0	1	1	1	1	1	1	0	1	1	1
TW-10	1	1	1	1	1	1	0	0	1	0	0	0	1	1	0	1	1	1	1	1	1	0	1	1	1
TW-10a	1	1	1	1	1	1	0	0	1	0	1	1	1	1	0	1	1	1	1	1	1	0	1	1	1
TW-10b	0	1	1	1	1	1	0	0	1	0	0	0	1	1	0	1	1	1	1	1	1	0	1	1	1
TW-10c	0	1	1	1	1	1	0	0	1	0	1	1	1	1	0	1	1	1	1	1	1	0	1	1	1

Table 4.5 (Continued)

Sequence: TW

BWR OAET Symptom Matrix Sheet 3

Symptoms 45-65

Symptoms	PC Level	PC Temp	HPCI		RCIC		LPCI		CS		DG		RHR		ADS			MC		CRD	
OAET State	45	46	47	48	49	50	51	52	53	54	55	56	57	58	59	60	61	62	63	64	65
TW-1,2	0	0	1	0	1	0	1	1	1	1	1	0	0	1	0	0	1	1	1	1	1
TW-3	0	0	1	0	1	0	1	1	1	1	1	0	0	1	0	0	1	1	1	1	1
TW-4	0	0	1	1	1	1	1	1	1	1	1	0	0	1	0	0	1	1	1	1	1
TW-5	0	0	1	1	1	1	1	1	1	1	1	0	0	1	0	0	1	1	1	1	1
TW-6	0	0	1	1	1	1	1	1	1	1	1	0	0	1	0	0	1	1	1	1	1
TW-7	0	0	1	0	1	1	1	1	1	1	1	0	0	1	0	0	1	1	1	1	1
TW-8	0	0	1	1	1	1	1	1	1	1	1	0	0	1	0	0	1	1	1	1	1
TW-8a	0	0	1	1	1	1	1	1	1	1	1	0	0	1	0	0	1	1	1	1	1
TW-9	1	0	1	1	1	1	1	1	1	1	1	0	0	1	0	0	1	1	1	1	1
TW-9a	1	0	1	1	1	1	1	1	1	1	1	0	0	1	0	0	1	1	1	1	1
TW-10	0	0	1	1	1	1	1	1	1	1	1	0	1	1	0	0	1	1	1	1	1
TW-10a	0	0	1	1	1	1	1	1	1	1	1	0	0	1	0	0	1	0	1	1	1
TW-10b	0	0	1	1	1	1	1	1	1	1	1	0	1	1	0	0	1	1	1	1	1
TW-10c	0	0	1	1	1	1	1	1	1	1	1	0	0	1	0	0	1	0	1	1	1

Table 4.5 (Continued)

Sequence: TW

BWR OAET Symptom Matrix Sheet 4

Symptoms 66-79

Symptoms	SLC		RWCU		Alternate Inj.		C/F		Recirc. Pumps		MSIV		RHR Interlock	
OAET State	66	67	68	69	70	71	72	73	74	75	76	77	78	79
TW-1,2	1	0	1	0	1	0	0	1	1	0	0	1	0	1
TW-3	1	0	1	0	1	0	0	1	1	0	0	1	0	1
TW-4	1	0	1	0	1	0	0	1	1	0	0	1	0	1
TW-5	1	0	1	0	1	0	0	1	1	0	0	1	0	1
TW-6	1	0	1	0	1	0	0	1	1	0	0	1	0	1
TW-7	1	0	1	0	1	0	0	1	1	0	0	1	0	1
TW-8	1	0	1	0	1	0	1	1	1	0	1	1	0	1
TW-8a	1	0	1	0	1	0	1	1	1	0	0	1	0	1
TW-9	1	0	1	0	1	0	1	1	1	0	1	1	0	1
TW-9a	1	0	1	0	1	0	1	1	1	0	0	1	0	1
TW-10	1	0	1	0	1	0	1	1	1	0	1	1	1	1
TW-10a	1	0	1	0	1	0	0	1	1	0	1	1	0	1
TW-10b	1	0	1	0	1	0	1	1	1	0	1	1	1	1
TW-10c	1	0	1	0	1	0	0	1	1	0	1	1	0	1

4.2.3 Station Blackout Event

This section addresses the physical plant response and operator actions associated with a station blackout event. Section 4.2.3.1 summarizes the sequence of events. Section 4.2.3.2 contains the Operator Action Event Tree for this sequence along with a description of the operator's role throughout the sequence. Section 4.2.3.3 details the physical plant response at each state in the OAET in terms of the twenty-nine key BWR parameters discussed in Section 3. Since many of the events in the Station Blackout sequence are similar to those in the previously discussed TQUV and TW sequences, these discussions (Sections 4.2.1 and 4.2.2, respectively) should be referenced for additional details.

4.2.3.1 Sequence Description

Station blackout occurs when a plant experiences a complete loss of offsite power followed by a failure of the emergency diesel generators to supply power to their selected loads. Therefore, only those systems which do not require AC power for operation are available to mitigate the effects of this event until such time as AC power is restored. The Reactor Safety Study (WASH-1400) did not explicitly model the station blackout scenario as an initiating event for BWRs. Rather, the report considered failure of emergency diesel generators as part of the failure probabilities for the mitigating systems of the BWR Transient Event Tree. Given a station blackout scenario there are two sequences of importance from the WASH-1400 BWR Transient Event Tree, sequences TQUV and TQW. These sequences are important because the station blackout event causes the conditional probability of events Q, V, and W to be unity since the systems included in each of these events require AC power to function.

For both of these sequences core damage is expected to occur unless AC power is restored. The time available to restore AC power depends on the availability of DC-powered systems and actions by the operator to recover power. Tables 4.6 and 4.7 describe the key events for these two sequences. Figure 4.5 is the BWR Transient Event Tree from the WASH-1400 report. The two sequences of interest are highlighted. Section 4.2.3.2 contains the detailed OAET for the station blackout event.

The immediate response to a complete loss of offsite power is a turbine trip and reactor scram. Because the pumps providing cooling to the main condenser are powered from offsite power, a loss of condenser vacuum will occur causing the MSIVs to shut. Isolation of the reactor from its heat sink will cause pressure to rise to the relief valve set point and decay heat will be transferred by relief valve action to the suppression pool. The feedwater pumps will ramp down to zero flow (event Q). This and the discharge through the safety valves will cause the vessel level to drop. The decreasing reactor vessel water level will quickly reach the HPCI/RCIC initiation set points.

Operation of HPCI/RCIC will ensure that reactor water level remains adequate while decay heat is transferred to the suppression pool for some time period. The length of time HPCI/RCIC is operational depends on many plant specific features such as battery capacity, water inventory, HPCI/RCIC cooling requirements, etc. However, without restoration of AC power HPCI/RCIC will eventually cease to operate (event U). Since the low pressure systems all require AC power to provide water to the core, they will also fail to operate (event V). The core will therefore eventually uncover. Failure to restore HPCI/RCIC and failure to restore AC power will result in a fairly rapid (∿ 30 minutes) core uncovery.

If AC power can be restored before HPCI/RCIC failure then the operator can continue to maintain reactor vessel level with these systems while removing decay heat from the suppression pool using the RHR system or from the reactor directly using the main condenser. Failure to remove decay heat (event W) will result in containment pressurization and possible structural failure.

4.2.3.2 Operator Response

This section examines the role of the operator throughout the progression of a TW or TQUV type sequence initiated by a station blackout event. Figure 4.6 is an operator action event tree for this scenario. It provides the framework for discussion of the operator response to a station blackout event. It is similar to the event tree in Figure 4.5 with modifications to reflect the operator's influence on key events. For simplicity, several events including scram and relief valve operation are included as a single event tree heading. Branches on the OAET have alphanumeric designators for referencing plant states inthe following discussion. The principal operator actions from the OAET are:

Table 4.6

KEY SEQUENCE OF EVENTS IN "TQUV" STATION BLACKOUT SEQUENCE

Time	Event
0	Loss of all offsite power and emergency diesel generators
0 - 1 sec	Main turbine trip
0 - 4 sec	Reactor trip
4 sec	Relief valves set point reached
8 sec	MSIVs close
30-40 sec	HPCI/RCIC fail to start
1-30 min	Operator attempts to restore HPCI/RCIC or AC power
30 min	Level reaches top of jet pump suction

Table 4.7

KEY SEQUENCE OF EVENTS IN "TQW" STATION BLACKOUT SEQUENCE

Time	Event
0	Loss of all offsite power and emergency diesel generators
0 - 1 sec	Main turbine trip
0 - 4 sec	Reactor trip
4 sec	Relief valves set point reached
8 sec	MSIVs close
30-40 sec	HPCI/RCIC start and provide water to core
4-8 hr (dependent)	Operator fails to recover AC power. Battery power supplies depleted. HPCI/RCIC ceases to operate. Level decreases and core uncovers.

1) Identify station blackout occurrence

2) Verify automatic actions such as scram, MSIV closure, and proper relief valve actuation

3) Verify inventory control with HPCI/RCIC

4) Identify HPCI/RCIC failure

5) Restore HPCI/RCIC operation

6) Restore AC power

7) Provide long term cooling using RHR or PCS

Initial operator action involves identifying the station blackout event so that the operator can take the correct actions to bring the plant to a safe shutdown condition. A station blackout event causes a scram, turbine trip, MSIV closure and loss of any AC powered equipment (including the motor driven condensate pumps). Loss of normal ventilation and lighting along with loss of non-emergency (AC-powered) instrumentation in the control room are likely to be the events which most easily allow the operator to recognize that a station blackout has occurred. For a station blackout event it is assumed that the plant and reactor trips, the MSIVs close, relief valve operation is successful and other automatic actions occur as expected (states 1 and 2 of the OAET).

State 3 of the OAET occurs if the HPCI/RCIC systems operate to maintain reactor water level. The operator should verify that system actuation has occurred and that vessel water level is not continuing to fall. Depending on the rate of increase in reactor water level, the operator may return one of the two systems to a standby readiness and manually control level. Without operator action vessel level should reach the high level trip point and cause a trip of both HPCI and RCIC. Only HPCI will automatically restart if level drops to the actuation point again. The RCIC actuation logic must be reset by the operator following each high level trip if automatic restart is desired.

Failure of HPCI/RCIC to operate appears as state 4 on the OAET. Studies indicate that following an isolation from the PCS the absence of a supply of water to the reactor will result in core uncovery in approximately 30 to 40 minutes (reference 3,7). To prevent this the operator must identify HPCI/RCIC failure by noticing a steadily declining vessel level and lack of flow

from the HPCI/RCIC pumps. He should try to restore operation of these systems. At this point the operator would probably try to restore the HPCI/ RCIC flow to the core by attempting to manually start the system either from the control room or locally. The short time period before core uncovery precludes any operator action involving repair of components. However, other studies have shown that many HPCI/RCIC failures have historically been due to control or initiation faults which can be bypassed by the operators (see reference 7). If the operator is successful in restoring HPCI/RCIC flow to the core then his subsequent actions are analogous to those discussed for state 3.

It is likely that operator efforts to restore AC power (state 5) will be occurring independently of the status of the HPCI/RCIC actions noted above. The two basic efforts would likely be attempting to restore power from the offsite grid, if available, and starting the emergency diesel generators which had failed to start initially. As with recovery of HPCI/RCIC it is unlikely that AC power can be restored quickly if repair work is required to fix broken components. If, however, the cause of the station blackout is due to spurious trips or faulty actuation signals, operator action such as a manual diesel generator start may be possible before core uncovery occurs. If AC power is restored then this sequence becomes part of the TQUV sequence previously discussed in Section 4.2.1. Failure to restore power following HPCI/RCIC failure combined with failure to restore HPCI/RCIC (state 6) will lead directly to core uncovery and core damage in much the same way as described for the TQUV sequence.

Given successful HPCI/RCIC operation (state 3) the operator can maintain reactor vessel level for some time, but not indefinitely. Recovery of AC power is necessary for decay heat removal from the containment as well as continued operation of HPCI/RCIC. There are several factors which effect the ability of the operator to maintain HPCI/RCIC flow to the core (state 7) for the longest period possible. These include battery capacity and discharge rate, available inventory for injection to the core, cooling for the HPCI/RCIC systems, and ability to discharge turbine steam exhaust to the suppression pool. These factors vary from plant-to-plant. The most limiting factor determines the time available for recovery of AC power which in turn influences the probability of recovery.

Since the HPCI/RCIC systems are DC powered, their operation time is proportional to the battery capacity. There are also other DC loads which would be receiving power from the batteries during a station blackout condition including vital instrumentation for the control room, emergency lighting, and communications. The operator can extend the time to battery depletion by turning off non-essential or redundant features once proper operation of required systems is verified.

The amount of water available for injection to the reactor can influence the maximum time that HPCI/RCIC operation can be maintained before AC power must be restored. Usually this is not the limiting factor since the CST or the suppression pool contain more than enough water to provide makeup for the duration of the battery capacity.

During state 7, the HPCI/RCIC turbines and the relief valves will be transferring decay heat to the suppression pool. Due to the loss of AC power, no heat removal (other than ambient losses) can take place. As the suppression pool heats up, the pressure inside the wetwell will also increase. If it reaches the high exhaust pressure limits, automatic HPCI/RCIC system trips will shut these systems down. Again, it is not likely that this condition will occur before battery failure.

Both turbines have lube oil systems which are cooled by the water being pumped. If the water temperature exceeds approximately 130°F then lube oil breakdown may occur resulting in turbine failure. As long as CST water is the source for the pumps, this is an unlikely occurrence. However, if suppression pool water is the source for the pumps, this temperature limit could be reached within a few hours due to the absence of suppression pool heat removal. Changeover from CST suction to suppression pool suction is a manual operation for the RCIC system. However, HPCI will automatically transfer suction if suppression pool level is high. If this condition occurs and suppression pool water temperature is also high, the operator would have to manually defeat this transfer signal and return suction to the CST to ensure continued HPCI operation. This may not be possible at some plants.

In addition to lube oil cooling, HPCI/RCIC system compartments normally receive room cooling from an AC system. Although loss of this cooling is unlikely to cause equipment failure during the time period in question, it will cause the rooms to become quite hot. Most designs have temperature sensors in these rooms to isolate the system on high room temperature. The function of this trip is to isolate a potential steam leak (LOCA) outside containment. Therefore, it is possible that isolation of HPCI/RCIC may occur before battery depletion even though no steam leak exists. The operator may also have to defeat this trip signal to continue HPCI/RCIC operation.

If the operators are unable to restore AC power (state 8 on the OAET) then HPCI/RCIC operation will eventually fail from one of the causes just described. At this point vessel water level will continue to decrease due to relief valve action and core uncovery will occur. This scenario is similar to the TQUV sequence of Section 4.2.1 and state 6 of this OAET except that the onset of core damage will occur later due to the grace time provided by the limited HPCI/RCIC operation.

Restoration of AC power from either the offsite grid or emergency generators (state 9 of the OAET) prior to HPCI/RCIC failure from one or more of the causes noted previously makes this sequence similar to the TW sequences discussed in Section 4.2.2. At this point the operator will attempt to remove the reactor decay heat in one of two ways. If offsite power is restored, the operator may reopen the MSIVs and use the main condenser as a heat sink and provide make-up water via the condensate and feed systems. Otherwise the operator can initiate suppression pool cooling via the RHR system. Successful operation of either means of decay heat removal results in core cooling (state 10a on the OAET).

Failure to establish decay heat removal via the PCS or RHR system will eventually result in containment overpressurization and structural failure (state 10b on the OAET). As noted previously in Section 4.2.2 for the TW sequence, this condition may or may not result in core damage. Those system requiring the suppression pool as a water source will not have adequate NPSH for operation. However, systems such as the CRD pumps or the condensate booster

pumps (if pressure is low enough) can still provide water to the core. Therefore, state 10b represents the case where the core may be vulnerable to uncovery but may still be cooled even though containment failure has occurred.

4.2.3.3 Symptom Matrix

The symptoms exhibited by the plant throughout the progression of the station blackout sequence are tabulated in Table 4.8. The behavior of each of the key BWR parameters (delineated and discussed in Section 3) is described in Table 4.8 for each state appearing in the operator action event tree (Figure 4.6).

As discussed in Section 2, multiple symptoms involving the same parameter may be indicated in Table 4.8. These multiple (and sometimes contradictory symptoms are due to either:

1) A state where the parameter is going through a transition from one trend to another; in these cases the operator may actually observe either or both symptoms.

2) A state where the physical plant response (e.g., rapid depressurization) may affect the symptoms observed by the operator (e.g., reactor water level); in these cases indications of these symptoms may be unreliable.

3) A state where the behavior of a certain parameter is simply unknown or highly uncertain.

If potential ambiguities are identified involving OAET states with such multiple symptoms, these few cases can be examined individually in more detail.

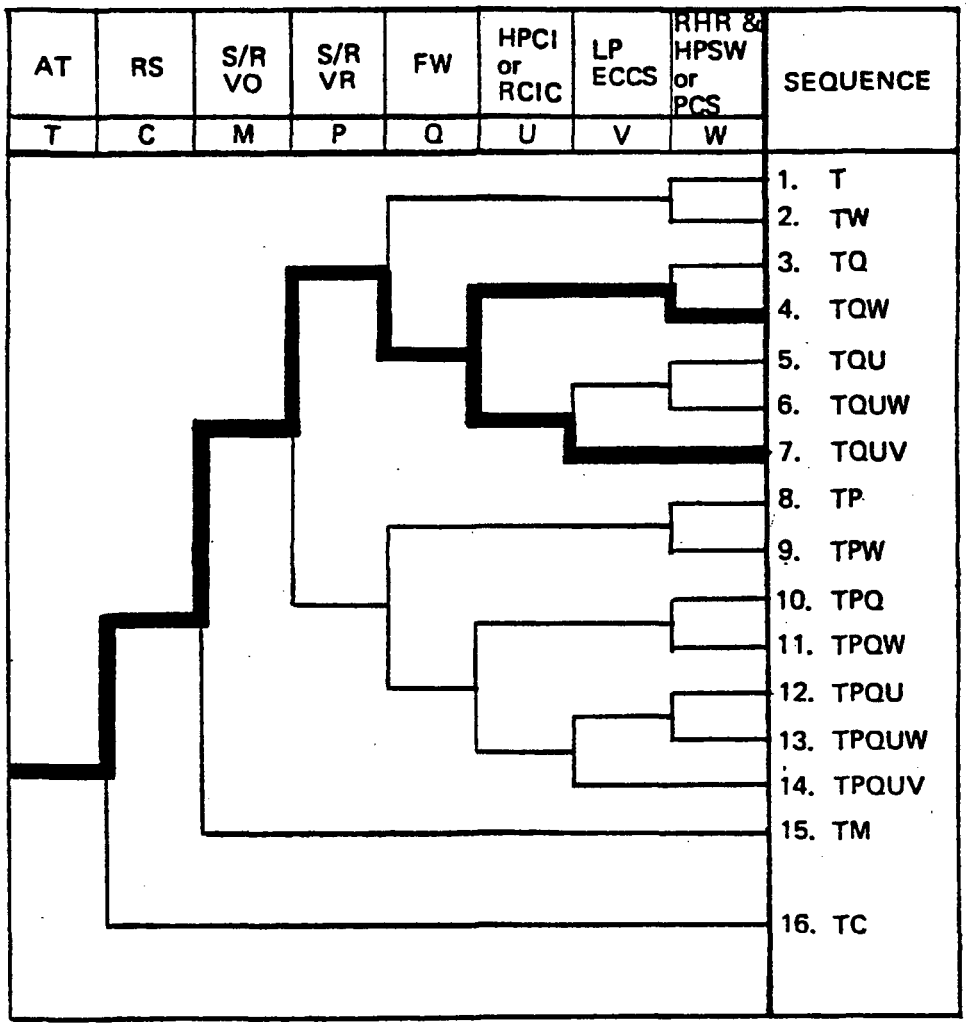

Figure 4.5 Reactor Safety Study Event Tree for BWR Transient Events (Event T): TQUV and TQW Sequences Highlighted.

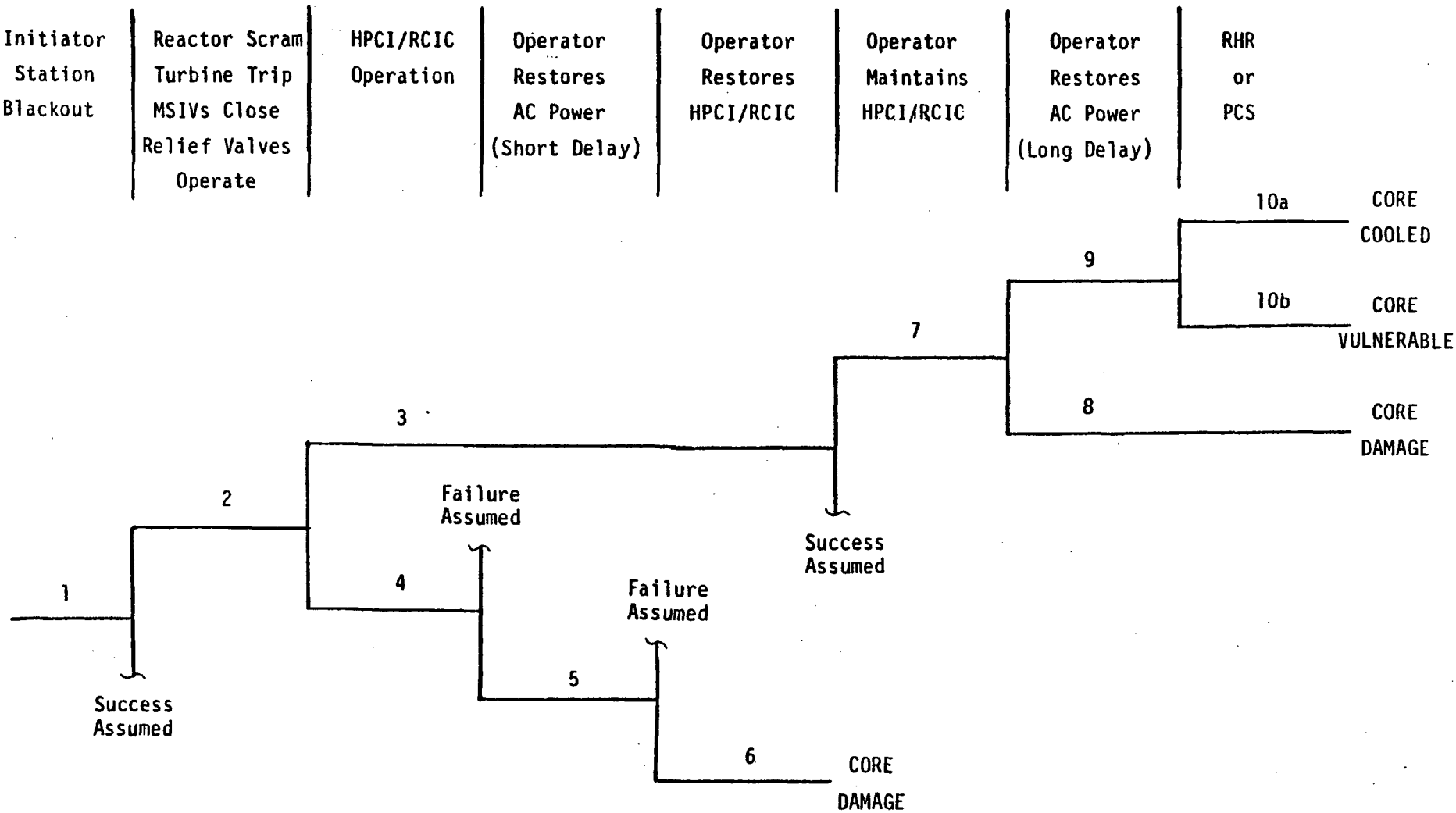

Figure 4.6 Station Blackout - Operator Action Event Tree

Table 4.8

Sequence: SBO

BWR OAET Symptom Matrix Sheet 1

Symptoms 1-19

(See Section 4.2.1.3 for explanation of Symbols)

Symptoms	RPV Level				RPV Press.									RPV Temp		Re-actor Pwr.		Rod Pos.	Bo-ron
OAET State	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19
SBO-1,2	0	1	0	0	1	0	1	0	0	1	0	1	1	0	0	1	0	1	0
SBO-3	0	0	0	0	0	1	0	1	1	1	1	1	1	0	0	1	0	1	0
SBO-4	0	1	1	0	1	0	1	0	0	1	0	1	1	0	0	1	0	1	0
SBO-5	0	1	1	0	1	0	1	0	0	1	0	1	1	0	0	1	0	1	0
SBO-6	0	1	1	1	1	0	1	0	0	1	0	1	1	0	0	1	0	1	0
SBO-7	0	0	0	0	0	1	0	1	1	1	1	1	1	0	0	1	0	1	0
SBO-8	0	1	1	1	1	0	1	0	0	0	1	1	1	0	0	1	0	1	0
SBO-9	0	0	0	0	0	1	0	1	1	1	1	1	1	0	0	1	0	1	0
SBO-10a	0	0	0	0	0	1	0	1	1	0	1	1	1	0	0	1	0	1	0
SBO-10b	0	1	1	1	1	0	1	0	0	0	1	1	1	0	0	1	0	1	0

Table 4.8 (Continued)

Sequence: SBO

BWR OAET Symptom Matrix Sheet 2

Symptoms 20-44

Symptoms	S.P. Level										S.C. Press					S.P. Temp					DW Prs	Drywell Temp.			
OAET State	20	21	22	23	24	25	26	27	28	29	30	31	32	33	34	35	36	37	38	39	40	41	42	43	44
SBO-1,2	0	0	1	0	1	0	0	0	1	0	0	0	0	0	1	1	0	1	1	1	0	0	0	1	1
SBO-3	1	0	1	0	1	1	0	0	1	0	0	0	1	1	1	1	1	1	1	1	0	0	0	1	1
SBO-4	0	0	1	0	1	1	0	0	1	0	0	0	0	0	1	1	1	1	1	1	0	0	0	1	1
SBO-5	0	0	1	0	1	1	0	0	1	0	0	0	0	0	1	1	1	1	1	1	0	0	0	1	1
SBO-6	0	0	1	0	1	1	0	0	1	0	1	1	1	1	0	1	1	1	1	1	0	0	0	1	1
SBO-7	1	0	1	0	1	1	0	0	1	0	0	0	1	1	1	1	1	1	1	1	1	0	0	1	1
SBO-8	1	1	1	1	1	1	0	0	1	0	1	1	1	1	0	1	1	1	1	1	1	0	1	0	1
SBO-9	0	0	1	0	1	1	0	0	1	0	0	0	0	0	1	1	1	1	1	1	1	0	0	1	1
SBO-10a	0	0	1	0	1	1	0	0	1	0	0	0	0	0	1	1	1	1	1	1	1	0	0	1	1
SBO-10b	0	0	1	0	1	1	0	0	1	0	1	1	1	1	0	1	1	1	1	1	1	0	1	0	1

Table 4.8 (Continued)

Sequence: SBO

BWR OAET Symptom Matrix Sheet 3

Symptoms 45-65

Symptoms	PC Level	PC Temp	HPCI		RCIC		LPCI		CS		DG		RHR		ADS			MC		CRD	
OAET State	45	46	47	48	49	50	51	52	53	54	55	56	57	58	59	60	61	62	63	64	65
SBO-1,2	0	0	1	0	1	0	0	1	0	1	0	1	0	1	0	0	1	0	1	0	1
SBO-3	0	0	1	0	1	0	0	1	0	1	0	1	0	1	0	0	1	0	1	0	1
SBO-4	0	0	0	1	0	1	0	1	0	1	0	1	0	1	0	0	1	0	1	0	1
SBO-5	0	0	0	1	0	1	0	1	0	1	0	1	0	1	0	0	1	0	1	0	1
SBO-6	0	0	0	1	0	1	0	1	0	1	0	1	0	1	0	0	1	0	1	0	1
SBO-7	0	0	1	0	1	0	0	1	0	1	0	1	0	1	0	0	1	0	1	0	1
SBO-8	1	0	1	1	1	1	0	1	0	1	0	1	0	1	0	0	1	0	1	0	1
SBO-9	0	0	1	0	1	0	1	0	1	0	1	1	1	1	0	0	1	1	1	1	1
SBO-10a	0	0	1	0	1	0	1	0	1	0	1	1	1	1	0	0	1	1	1	1	1
SBO-10b	1	0	1	1	1	1	0	1	0	1	1	1	0	1	0	0	1	0	1	1	1

Table 4.8 (Continued)

Sequence: SBO

BWR OAET Symptom Matrix Sheet 4

Symptoms 66-79

Symptoms	SLC		RWCU		Alternate Inj.		C/F		Recirc. Pumps		MSIV		RHR Interlock	
OAET State	66	67	68	69	70	71	72	73	74	75	76	77	78	79
SBO-1,2	0	1	0	1	0	1	0	1	0	1	0	1	0	1
SBO-3	0	1	0	1	0	1	0	1	0	1	0	1	0	1
SBO-4	0	1	0	1	0	1	0	1	0	1	0	1	0	1
SBO-5	0	1	0	1	0	1	0	1	0	1	0	1	0	1
SBO-6	0	1	0	1	0	1	0	1	0	1	0	1	0	1
SBO-7	0	1	0	1	0	1	0	1	0	1	0	1	0	1
SBO-8	0	1	0	1	0	1	0	1	0	1	0	1	0	1
SBO-9	1	1	1	1	1	1	1	1	1	0	0	1	0	1
SBO-10a	1	1	1	1	1	1	1	1	1	0	1	1	1	0
SBO-10b	1	1	1	1	1	1	0	1	1	0	0	1	1	0

4.2.4 Anticipated Transients with a Failure to Scram (TC)

This section addresses the physical plant response and operator actions associated with an MSIV closure followed by a failure to insert control rods. Section 4.2.4.1 summarizes the sequence of events. Section 4.2.4.2 contains the Operator Action Event Tree for this sequence along with a description of the operator's role throughout the sequence. Section 4.2.4.3 details the physical plant response to each state in the OAET in terms of the twenty-nine key BWR parameters discussed in Section 3. Since some of the events in the MSIV Closure ATWS sequence are similar to those in the previously discussed TQUV and TW sequences, these discussions (Sections 4.2.1 and 4.2.2, respectively) should be referenced for additional details.

4.2.4.1 Sequence Description

It is anticipated that during 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 insert the control rods following such anticipated transients.

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 MSIV closure initiating event has been selected. This particular event was chosen primarily because 1) the frequency is relatively high, 2) most of the operator actions required in response to this event would be identical to other transient events, 3) the amount and quality of information available concerning the plant response to this event is greater relative to most other events, and 4) it represents a significant challenge to systems and operator response because of the relatively short reaction time.

An MSIV Closure could occur because of loss of condenser vacuum, operator errors, or out of specification reactor system variables such as a low reactor vessel water level trip signal. Following MSIV closure from high power the reactor pressure will rise rapidly. As the pressure rises the SRVs will open to relieve pressure to the suppression pool. However, the pressure rise

will be sufficient to initiate recirculation pump trip (RPT) which rapidly inserts negative reactivity through increased void formation. In addition, the alternate rod injection (ARI) system will be actuated on high reactor dome pressure. The two ATWS mitigating systems have been or will be included in most BWRs to provide diversity and additional redundancy in ATWS mitigation. Subsequent to this initial sharp pressure rise which is terminated by RPT, 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. The water level will continue to drop to a low-low level at which point the recirculation pumps receive a diverse trip signal on low level and the High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) systems are initiated.

Thus far the discussion of the sequence of events has focused on the automatic actions which would occur immediately following an MSIV Closure ATWS. In addition to these automatic actions backup operator responses and the remainder of the potential ATWS mitigation features can be manually initiated. The Operator Action Event Trees represent a useful format for the analysis of these manual operator actions.

The principal operator actions involved with the successful injection of the boron to insert sufficient negative reactivity to bring the reactor subcritical are the following:

1. SLC initiation
2. Maintain the minimum coolant injection flow possible to adequately cool the core while waiting for SLC injection to be complete.
3. Initiate RHR to provide as much suppression pool cooling as possible.

Table 4.9 summarizes a possible sequence of events for a postulated MSIV closure initiated ATWS sequence from a high initial power level.

The MSIV Closure initiated transient coupled with a failure to insert control rods is a severe challenge to the BWR systems since all the steam produced in the reactor is directed to the suppression pool, i.e., the condenser

is isolated from the reactor and not available as an alternate heat sink. This sequence may result in raising the suppression pool temperature to the heat capacity temperature limit.

Table 4.9
Sequence of Events Following An MSIV Closure
Initiated ATWS from High Initial Power

SEQUENCE OF EVENTS	TIME
Nominal 4-Second MSIV Closure	0
Pressure Rise Begins	0
Safety Relief Valves Lift	4 seconds
Some Fuel Experiences Transition Boiling	5 seconds
Recirculation Pumps Trip on High Pressure; ARI is initiated	5 seconds
Vessel Pressure Peaks	9 seconds
ARI Fails	30 seconds
HPCI and RCIC Flow Starts After Level 2 Initiation	1 minute
SLCS Starts*	2 - 15 minutes
Liquid Control Flow Reaches Core	3 - 16 minutes
RHR Flow Begins (Pool Cooling)	11 minutes
Hot Shutdown Achieved**	32 - 45 minutes

*Range of times assumed available to the operator following this initiator to begin SLC injection for successful ATWS mitigation. The time is based upon maintaining an "acceptable" suppression pool temperature.

**Hot shutdown is defined as generated power remaining below 1% NBR.

In the sections below the key operator actions associated with this sequence are delineated. The report focuses on the failure to scram event and the associated operator actions.

4.2.4.2 Operator Response Following a MSIV Closure ATWS

This section focuses on the role of the operator throughout the accident sequence progression following an MSIV closure initiated ATWS sequence.

Figure 4.7 is the system event tree from WASH-1400 used to display the accident sequence involving an anticipated transient coupled with a failure to insert control rods and adequate negative reactivity via the SLC system. The sequence is referred to as TC in WASH-1400 terminology. Figure 4.8 displays in a logic diagram format the relevant operator action events. This operator action event tree (OAET) can be viewed as a version of the system transient event tree (Figure 4.7) which focuses on the key operator actions necessary to recover from the postulated failure events and bring the reactor to a safe shutdown condition. Selected important states to which the plant can evolve as the accident sequence progresses are enumerated on the figure.

Plant state TC-1 sets the initial conditions of the sequence. The postulated initiator involves a closure of all the MSIVs and therefore isolation of the primary system from the principal heat sink, the main condenser. For the generic analysis performed here, the feedwater pumps are assumed to be unavailable due to the loss of steam to the turbine driven FW pumps. During the initiating event a number of plant parameters are changing rapidly leading to initiation of important system functions. The principal automatic functions are those associated with the high system pressure following MSIV closure. Both the recirculation pump trip (RPT) and backup scram initiation signals (ARI) are generated. The benefits of the RPT are twofold. The immediate benefit involves the termination of the pressure excursion within acceptable primary system limits by the rapid insertion of a large void negative reactivity. The second benefit from RPT involves the reduction in steam flow produced and therefore limits the steam flow rate which is transported to the suppression pool.

States TC-2, 3, 4, and 5 address questions regarding negative reactivity insertion and potential operator response.

State TC-2 involves the insertion of the control rods into the core to provide sufficient negative reactivity to bring the reactor subcritical. The failure of the control rods to insert may be the result of either a mechanical or electrical common mode failure. The mechanical redundancy of the control rod drive mechanisms makes the common-mode failure of multiple adjacent control rods unlikely. The electrical diversity in sensors, logic, and scram solenoids help to reduce the potential for common-mode failures leading to failure of multiple rods to insert due to an electrical common-mode failure. In addition, a diverse backup (ARI) is also available to provide added protection against RPS failure. Also the operator is directed to manually backup the scram signal. A failure of the control rods to insert is indicated to the operator by:

1. control rod position indication;
2. control rod bottom light indication; and,
3. reactor power level.

Failure of the operator to obtain these indications will initiate specified operator response to provide manual actions to insert the control rods. These manual actions include:

1. Manual operation of scram valves.
2. Removal of fuses for scram valve solenoids.
3. Reset reactor scram, if scram signal has cleared.
4. Manual insertion of control rods.
5. Individually open the scram test switches for control rods.
6. Manipulation of control rods until inserted beyond 06 notch.

In addition, it is important that the operator monitor the parameters associated with SLC initiation, since steps necessary to begin SLC system initiation should take priority. The indications are a high suppression pool temperature (above 110°F) and a high reactor power (above 3% ARPM).

State TC-3 involves recirculation pump trip (RPT). In a BWR, the recirculation pumps provide a method of changing core reactivity without changing the control rod position. Positive reactivity can be inserted by increasing the recirculation pumps flow. The recirculation pumps are used to control power over a range of approximately 35% power for recirculation flow changes between 30% to 100%. RPT automatically trips the recirculation pumps on either high reactor pressure or low reactor water level. The RPT is effective in rapidly inserting sufficient negative reactivity into the core through void formation to limit both the power and pressure rise to within acceptable limits following an ATWS. The reactor power will drop to approximately 40-50% following a RPT from a 100% turbine trip ATWS. Manual operator action is necessary to verify RPT, or trip the recirculation pumps for MSIV closure cases.

Since feedwater terminates upon MSIV closure, the core power will drop even further following successful RPT to approximately 25-30% flow, determined to a large extent by HPCI/RCIC flow. If water level in the core is reduced to near the top of the active fuel, the core power can be reduced to approximately 8% because of the higher void fraction.

Failure of the RPT (state TC-3a) is postulated to lead to high primary system pressures (above 1500 psi). A large breach of the primary system may then occur, low pressure injection would begin causing recriticality, excess steam would be directed to the suppression pool, and the containment pressure would rise leading directly to a containment challenge of high internal pressure.

Operator action to initiate a manual backup scram also activates ARI. ARI is a plant modification incorporated into many BWRs to provide additional diversity to the electrical portion of the scram system. It incorporates a number of changes including additional sensors, additional logic, and additional solenoid valves to provide added assurance that the postulated common-mode

electrical failures will not prevent control rod insertion. At state TC-4 the automatic backup scram system is postulated to have failed. Operator action is immediately necessary to initiate operation of the standby liquid control system (SLCS). This system is used to inject boron into the core, thus providing additional negative reactivity and reducing the reactor power. As will be seen, the operator actions required for successful SLC are more than merely initiating SLC. The operator must also minimize coolant injection flow while maintaining an adequate reactor water level. In addition to these positive operator actions he may also be expected to inhibit depressurization with the expected ADS.

The symptoms available to the operator which would lead to initiation of the SLC system are the following:

1. The reactor is not shutdown†; and
2. The suppression pool temperature reaches 110°F.

Several functions must occur to provide successful reactivity control. For situations involving an MSIV closure, successful RPT, and a failure to scram, the following manual actions are required:

1. Manual initiation of the SLC system.
2. Lowering the reactor water level to slightly above the TAF.
3. Re-establishing water level and boron mixing when the SLC tank is empty.

All three requirements are dependent upon operator action and are therefore strongly time dependent.

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†Reactor shutdown is a state where power is less than 3% and the control rods are inserted beyond the "maximum subcritical failed withdrawal position."

At state TC-5 the operator has been successful in attempts to inject boron into the vessel with the SLC system. At state TC-5a his attempts have failed, and the reactor power and suppression pool temperature remain elevated. Unacceptable conditions exist and fuel damage may occur at high pressure.

State TC-5 is a very important state in the operator action event tree. It represents a pivotal point in the operator's attempt to deal with an ATWS. The decision at State TC-5 revolves around whether to continue SLC injection and inventory control at high pressure or to require depressurization in order to avoid the possibility of a blowdown sometime later from high reactor pressure while the suppression pool is at elevated temperature. The depressurization however would be carried out with eventually all coolant injection terminated. Also during this portion of the sequence the drywell coolers may be isolated due to high drywell pressure or low reactor water level. If this occurs the drywell temperature may increase, leading to operator action to initiate drywell sprays. It is postulated in this sequence that the operator attempts to continue SLC injection and inventory control at high pressure, rather than attempting depressurization at this time.

At state TC-6 the operator has successfully initiated high pressure coolant injection, and is attempting to maintain coolant inventory in this way. He must also attempt to initiate or continue heat removal from the suppression pool, and maintain its temperature below the heat capacity temperature limit. It is postulated here that the operator cannot maintain the temperature of the suppression pool below this limit, as depicted by state TC-7 on Figure 4.8. The operator must therefore depressurize immediately before the pool temperature exceeds the point at which condensation of steam would be ineffective. If depressurization is attempted subsequent to this time, excessive pressure loads may be placed on the containment, and structural failure may occur.

At state TC-8 the operator has successfully depressurized the reactor. He must now maintain coolant inventory with the high volume, low pressure systems, while preserving the reactivity control by maintaining the vessel level below level 5 or level 6 to avoid washing the boron from the core. The operator must exercise caution at this time when monitoring vessel water level since the level instrumentation may be affected by a rapid blowdown.

At state TC-9 the operator has successfully controlled the vessel water level, thus preserving reactivity control. He must now establish or continue a form of decay heat removal from the containment. This is normally accomplished with the shutdown cooling mode of the RHR system. The operator may also attempt to reestablish heat removal through the main condenser, although this would necessitate re-opening the MSIVs. See section 4.2.2 (TW sequence) for more details.

At state TC-10 heat removal has been established, and the reactor is approaching cold shutdown conditions. The operator must continue the approach to shutdown while monitoring all of the necessary parameters (water level, power level, containment and suppression pool conditions, etc.)

At state TC-8a the operator has been unable to depressurize the vessel, and the water level in the core is failing. Eventually the core will uncover and fuel damage will commence. The operator's actions at this state center on establishing a form of makeup to the vessel while ensuring that containment leakage paths have been minimized.

At state TC-9a the operator has failed to maintain coolant inventory with the low pressure systems, or has been unable to preserve reactivity control. This state is similar to state TC-8a, although the reactor power level may be somewhat higher. Again, as at state TC-8a, core damage will eventually occur as the core uncovers.

At state TC-10a the coolant inventory is being maintained in the vessel, but containment heat removal has not been established. The structural integrity of the containment may be threatened, which in turn results in core vulnerable conditions due to the effects containment failure may have on coolant inventory maintenance. This state is similar to states described previously in Section 4.2.2 for the TW sequence.

4.2.4.3 Symptom Matrix for Failure to Scram Sequence

The Operator Action Event Tree (OAET) states can be described in terms of key plant symptoms which have been defined in Section 3. Table 4.10 summarizes the symptoms that are exhibited at the selected states of the MSIV closure ATWS OAET.

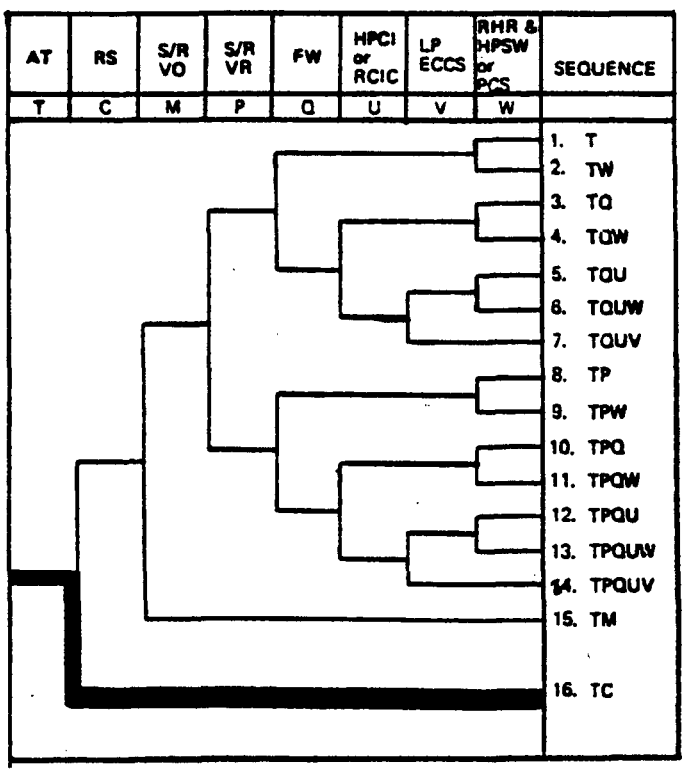

Figure 4.7 Reactor Safety Study Event Tree for BWR Transient Events (Event T): TC (ATWS) Sequence Highlighted

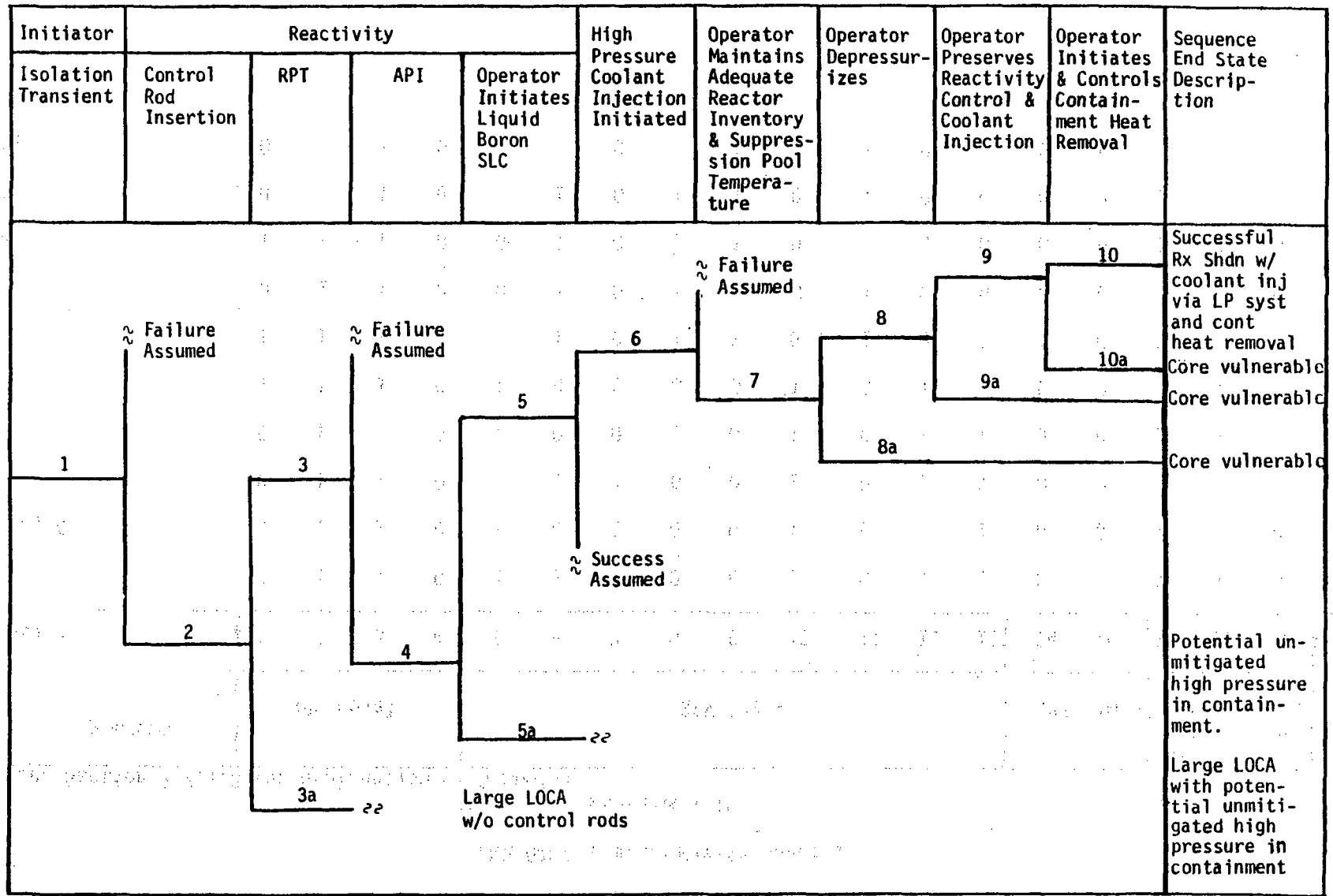

Figure 4.8 Operator Action Event Tree for Sequences Following a Postulated MSIV Closure with a Failure to Scram

Table 4.10

Sequence: TC

BWR OAET Symptom Matrix Sheet 1

Symptoms 1-19

(See Section 4.2.1.3 for explanation of Symbols)

Symptoms	RPV Level				RPV Press.									RPV Temp		Re-actor Pwr.		Rod Pos.	Bo-ron
OAET State	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19
TC-1	0	1	1	0	1	1	1	0	0	1	0	1	1	0	1	1	1	0	0
TC-2, 3, 4	0	1	1	0	1	0	1	0	0	1	0	1	1	0	0	1	1	0	0
TC-5	0	1	1	0	1	0	1	0	0	1	0	1	1	0	0	1	1	0	0
TC-6	0	1	1	0	1	0	0	0	0	1	0	1	1	0	0	1	1	0	0
TC-7	1	1	1	0	1	0	0	0	0	1	0	1	1	0	0	1	1	0	1
TC-8	1	1	1	0	0	1	0	1	1	0	1	0	0	1	0	1	1	0	1
TC-9	0	1	1	0	0	1	0	1	1	0	1	0	0	0	0	1	1	0	1
TC-9a	1	1	1	0	0	1	0	1	1	0	1	0	0	0	0	1	1	0	1
TC-10	0	1	1	0	0	1	0	1	1	0	1	0	0	0	0	1	1	0	1
TC-10a	0	1	1	0	0	1	0	1	1	0	1	0	0	0	0	1	1	0	1

Table 4.10 (Continued)

Sequence: TC

BWR OAET Symptom Matrix Sheet 2

Symptoms 20-44

Symptoms	S.P. Level										S.C. Press					S.P. Temp					DW Prs	Drywell Temp.			
OAET State	20	21	22	23	24	25	26	27	28	29	30	31	32	33	34	35	36	37	38	39	40	41	42	43	44
TC-1	0	0	1	0	1	0	0	1	1	0	0	0	0	0	1	0	0	0	0	1	0	0	0	1	0
TC-2, 3, 4	0	0	1	0	1	0	0	1	1	0	0	0	0	0	1	0	0	0	0	1	0	0	0	1	0
TC-5	0	0	1	0	1	0	0	1	1	0	0	0	0	0	1	0	1	1	0	1	1	0	0	1	0
TC-6	0	0	1	0	1	1	0	1	1	0	0	0	0	0	1	0	1	1	0	1	1	0	0	1	0
TC-7	0	0	1	0	1	1	0	0	1	0	1	1	1	1	1	1	1	1	1	1	1	1	1	0	1
TC-8	0	0	1	0	1	1	0	1	1	0	1	1	1	1	1	1	1	1	1	0	1	1	1	0	1
TC-9	0	0	1	0	1	1	0	1	1	0	1	1	1	1	1	1	1	1	1	0	1	1	1	0	1
TC-9a	0	0	1	0	1	1	0	1	1	0	1	1	1	1	0	1	1	1	1	0	1	1	1	0	1
TC-10	0	0	1	0	1	1	0	1	1	0	1	1	1	1	1	1	1	1	1	0	1	1	1	0	1
TC-10a	0	0	1	0	1	1	0	1	1	0	1	1	1	1	0	1	1	1	1	0	1	1	1	0	1

Table 4.10 (Continued)

Sequence: TC

BWR OAET Symptom Matrix Sheet 3

Symptoms 45-65

Symptoms	PC Level	PC Temp	HPCI		RCIC		LPCI		CS		DG		RHR		ADS			MC		CRD	
OAET State	45	46	47	48	49	50	51	52	53	54	55	56	57	58	59	60	61	62	63	64	65
TC-1	0	0	1	1	1	1	1	1	1	1	1	1	1	1	0	0	1	0	1	1	0
TC-2, 3, 4	0	0	1	1	1	1	1	1	1	1	1	1	1	1	0	0	1	0	1	1	0
TC-5	0	0	1	1	1	1	1	1	1	1	1	1	1	1	0	1	1	0	1	1	0
TC-6	0	0	1	0	1	0	1	1	1	1	1	1	1	1	0	0	1	0	1	1	0
TC-7	0	1	1	1	1	1	1	1	1	1	1	1	1	1	0	1	1	0	1	1	0
TC-8	0	1	0	1	0	1	0	1	0	1	1	1	1	1	1	1	1	0	1	1	0
TC-9	0	1	0	1	0	1	1	0	1	1	1	1	1	0	1	0	1	0	1	1	0
TC-9a	0	1	0	1	0	1	1	1	1	1	1	1	1	1	1	0	1	0	1	1	0
TC-10	0	1	0	1	0	1	1	0	1	1	1	1	1	0	1	0	1	0	1	1	0
TC-10a	0	1	0	1	0	1	1	1	1	1	1	1	1	1	1	0	1	0	1	1	0

Table 4.10 (Continued)

Sequence: TC

BWR OAET Symptom Matrix Sheet 4

Symptoms 66-79

Symptoms	SLC		RWCU		Alternate Inj.		C/F		Recirc. Pumps		MSIV		RHR Interlock	
OAET State	66	67	68	69	70	71	72	73	74	75	76	77	78	79
TC-1	1	1	1	0	1	1	0	1	1	1	0	1	0	1
TC-2, 3, 4	1	1	1	0	1	1	0	1	0	1	0	1	0	1
TC-5	1	0	1	0	1	1	0	1	0	1	0	1	0	1
TC-6	1	0	1	0	1	1	0	1	0	1	0	1	0	1
TC-7	1	0	1	0	1	1	0	1	0	1	0	1	0	1
TC-8	1	0	1	0	1	1	0	1	0	1	0	1	0	1
TC-9	1	0	1	0	1	1	0	1	0	1	0	1	0	1
TC-9a	1	0	1	0	1	1	0	1	0	1	0	1	0	1
TC-10	1	0	1	0	1	1	0	1	0	1	0	1	0	1
TC-10a	1	0	1	0	1	1	0	1	0	1	0	1	0	1

4.3 Loss of Coolant Accidents (LOCAs)

In this section, the physical plant response and possible operator actions following a loss of coolant accident (LOCA) are presented and discussed. Emphasis will be placed upon operator actions to mitigate postulated failures of the automatic safety systems designed to function following a LOCA.

A loss of coolant accident is defined as a breach in the primary reactor coolant system (RCS) which results in a release of fluid (steam or water) either inside or outside of the primary containment. The maintenance of adequate RCS inventory is then threatened. Two general classes of LOCAs can be described: 1) those which occur randomly during normal plant operation and are accident sequence initiating events, and 2) those which occur as a result of some other (transient) event. A well-known example of the second type is a stuck open safety/relief valve (SORV) following a transient initiating event.

Breaks in the first class can be divided further into break-size categories such as "large", "medium", or "small" breaks. These classifications are based upon the combinations of Emergency Core Cooling Systems (ECCS's) necessary to respond to the break. Breaks in the second class can also be divided into these same categories based upon the size of the safety/relief valve (SRV) and the assumed number of stuck-open valves.

For the discussion which follows, the initiating events are assumed to be small LOCAs (see Section 4.1, Selection of Accident Conditions). The WASH-1400 report contains two sub-groupings for small LOCAs, designated S_1 and S_2. For liquid line breaks, a small LOCA in the S_2 group has an effective diameter between about 0.6 and 2.6 inches, while those in the S_1 group are between about 2.5 and 8.5 inches (there is a slight overlap). For breaks in steam lines, the effective diameter is between 1.0 and 4.7 inches (S_2), or 4.7 and 6.0 inches (S_1). The primary differences between the two break size groupings is in the timing of events, such as pressurization of the drywell, and in the requirements for emergency coolant injection. Inventory lost through a break which falls into the S_2 size grouping can be adequately replenished by the Reactor Core Isolation Cooling (RCIC) System alone, whereas breaks in the S_1

group require the High Pressure Coolant Injection (HPCI) System regardless of the status of the RCICS.* Significant differences in plant or operator response, if any, will be discussed in the individual sequence descriptions which follow.

Three LOCA -type sequences are discussed in sections 4.3.1 through 4.3.3: 1) a small-break LOCA combined with a failure to maintain inventory (failures of HPCI and LPCI or depressurization), 2) a small-break LOCA combined with a failure to remove decay heat (failures of the low pressure recirculation systems), and 3) a transient-induced shutdown which results in a stuck-open safety valve.

The physical plant response and operator actions following breaks described in Section 4.3.1 and 4.3.2 are highly dependent upon both the size and the location of the break. Whether the break is inside or outside containment, as well as whether the break can be isolated, are important to subsequent plant behavior and operator response. For the sections which follow, discussion will be limited to breaks inside containment, or breaks outside containment which can be isolated by closing the MSIVs. Unisolable breaks outside containment, such as in the scram discharge volume, will not be discussed, as insufficient information exists at this time to adequately address the operator's response to these breaks, especially in the area of secondary containment control. Additionally, breaks below the elevation of the top of the core have been excluded from consideration.

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*Note: There is of course some conservatism built into this assumption. Realistically, there may be some spectrum of S_1-sized breaks for which the RCIC or CRD pumps could maintain inventory.

4.3.1 Failure to Maintain Inventory after a Small-Break-Loss of Coolant Accident (S_1E)

In this section, the physical plant response and operator actions associated with a small LOCA (S_1) coupled with failures leading to an inability to maintain adequate coolant (event E) are presented and discussed. A summary of the accident sequence is provided below in Section 4.3.1.1. The Operator Action Event Tree for this sequence is provided in Section 4.3.1.2 and used as the framework for a description of the role of the operator throughout the progression of this sequence. In Section 4.3.1.3, the physical plant response at each state in the OAET is presented in terms of the behavior of the twenty-nine key BWR parameters discussed in Section 3.

4.3.1.1 Sequence Description

The initiator for this sequence is a small break in the Reactor Coolant System (RCS) piping. A small LOCA is defined as a breach in the RCS which is too small to depressurize the system so that the low pressure Emergency Core Cooling Systems (ECCS's) can be used, but is sufficiently large so that the control rod hydraulic supply system cannot adequately replenish the fluid lost through the break. Without any supplemental cooling, coolant inventory will eventually become depleted and core damage will occur.

Two categories of small breaks were defined in WASH-1400, designated S_1 and S_2. For liquid line breaks, a small LOCA in the S_2 group has an effective diameter between about 0.6 and 2.6 inches (area = $.002ft^2$ - $0.04ft^2$), while those in the S_1 group are between about 2.5 and 8.5 inches (0.03 - $0.39ft^2$ area). For breaks in steam lines, the effective diameter is between 1.0 and 4.7 inches (0.01 - 0.12 ft^2 area) for S_2, or 4.7 and 6.0 inches (0.12 - $0.20ft^2$ area) for S_1. The RCICS is capable of replacing inventory lost through breaks in the S_2 group, whereas the HPCIS is required for S_1 breaks. For the discussion which follows it is assumed that the initiator is a small break in the

S_1 group, with subsequent trip of the feedwater pumps. Therefore the HPCIS is required. However, by definition of the sequence, all injection fails so that even if the break were an S_2 break it would be assumed that the RCIC and HPCI had failed.

Table 4.11 is a compilation of key events which may occur during a small break LOCA sequence with failure to maintain inventory. The timing for these events is based upon Reference 6. The timing is based on the very conservative assumption that feedwater is unavailable from time 0. If feedwater is available throughout the sequence, inventory should be maintained without the need for ECCS operation. For the discussion which follows it is postulated that feedwater is available for some time, but eventually trips. Figure 4.7 highlights this sequence on the system event tree developed in WASH-1400 for the small break LOCA. In Section 4.3.1.2 a detailed operator action event tree (OAET) is developed to display the operator actions in conjunction with the key plant states.

The postulated break location is a very important parameter in describing the plant response to the S_1 initiator. The specific line in which the break occurs (i.e., recirculation, feedwater, or steam line), and whether it is inside or outside containment will influence the availability of plant systems and the timing of their response. The Reactor Safety Study did not examine these factors in detail for its small break analysis. However, they are very important in assessing operator response and therefore are addressed in this discussion. The following sequence description considers the plant response to a recirculation line break. Subsequently, the effects of break location are addressed and the important differences highlighted.

The immediate consequence of a break in the recirculation line is a rapid increase in drywell pressure. At 2.0 psig in the drywell a reactor scram is initiated, a containment isolation signal is generated, and an ADS permissive signal (high drywell pressure) is sent. A high drywell pressure scram also generates a signal for the initiation of the HPCIS.

Table 4.11
Key Sequence of Events in "S_1E" Sequence*

Time (sec)				
0.001 ft^2 liquid break	0.1 ft^2 liquid break	0.001 ft^2 steam break	0.1 ft^2 steam break	Event
0	0	0	0	Initiating event : small break in RCS.
30-60	3-45	30-60	3-45	High Drywell Pressure Trip; Reactor Scram; HPCI initiation signal; High pressure ADS signal; Containment Isolation Signal. HPCI fails to start.
15-45	15-45	15-45	15-45	Level 3 (Low vessel level) reached; Reactor Scram signal generated, reactor scrams if not already tripped; RHR shutdown cooling isolation valves signalled to close; ADS level permissive.
30-45	30-45	∿150	150	Level 2 (low-low vessel level) reached; RCIC/HPCI initiation signal generated; Trip Recirc pumps; close MSIV's and containment isolation valves if not already closed. HPCI/RCIC fail to start.
120-150	120-150	120-150	120-150	SRV setpoints reached.
30-325	3-120	30-600	3-500	Operator attempts to start high pressure systems.
275-325	100-120	450-600	450-500	Level 1 reached; Initiate LPCS and RHR; Start Standby diesels; low level ADS signal generated - ADS Timer starts.
250-450	100-150	500-900	450-600	Operator Attempts to Depressurize and start LPCI. Depressurization and/or LPCI fail.
450	150	600-900(est.)	600	TAF reached.

*Assuming no operator action <u>and</u> immediate FW trip with ramp to zero flow in 1 second.

The small break in the recirculation line is defined to be small enough that the feedwater system, if in operation, is adequate to provide makeup for the fluid lost through the break. The break itself should not cause an immediate loss of feedwater. However, it is postulated that for small breaks, feedwater will eventually be lost due to, for example, a low pressure trip (following a failure to place the reactor mode switch in "shutdown"). After feedwater is lost, the HPCI pumps are required to maintain level until the system can be depressurized to below the shutoff head of the low pressure ECC systems. By definition of event E in the S_1E sequence, the HPCI system is assumed to be unavailable or is not activated. Thus, there is insufficient high pressure flow to maintain the inventory in the vessel.

If feedwater or HPCI flow is not restored, the vessel water level will drop as the fission product decay heat and stored heat generate more steam. When the water level reaches the Level 2 setpoint (low-low vessel level), another signal for HPCI/RCIC initiation is generated. At the same time the recirculation pumps are tripped and the main steam isolation valves are closed if they are still open. The power conversion system (feedwater, main condenser, turbine) cannot be used to remove heat unless the operator takes action to re-open the MSIVs. The pressure in the vessel rises rapidly to the relief valve setpoint(s), and would, without operator intervention, begin to cycle about these settings. The operator should take control of the safety valves to minimize cycling.

The HPCI and RCIC systems are again assumed to be unable to maintain coolant inventory. The RCIC System is designed to provide makeup to the core during a reactor shutdown in which feedwater flow is not available. The RCIC consists of one steam driven turbine pump with associated piping valves and controls. Reactor steam drives the turbine and steam exhaust from the turbine discharges to the suppression pool. Suction for RCIC comes initially from the Condensate Storage Tank (CST). Suction can be manually transferred to the suppression pool. The RCIC is capable of delivering approximately 500 gpm to the core via a feedwater line.

The HPCI System is a part of the Emergency Core Cooling System (ECCS) designed to supply water to the core during small break accidents which do not produce a rapid system depressurization. The HPCI system also consists of a single reactor steam driven turbine pump. The system is capable of delivering approximately 5000 gpm to the core. In the normal configuration, water is taken from the CST and discharged into a feedwater line (different line than used by RCIC). If the CST level drops to a low level setpoint or the suppression pool level gets too high, automatic transfer of suction is made to the suppression pool.

In this accident sequence, it is postulated that both the RCIC and HPCI fail to maintain adequate inventory. As discussed previously in Section 4.2.1, reference 15 reports that "a large number of BWR plants are not meeting the performance goals established for HPCI and RCIC reliability. Specifically, there have been four instances [in less than three years] when, upon demand, neither HPCI or RCIC would operate." The simultaneous unavailability of both high pressure makeup systems is therefore not an incredible event.

The failure of HPCI and RCIC following the LOCA initiator results in the absence of any means to replenish and adequately maintain vessel inventory at a high pressure. The Control Rod Drive (CRD) hydraulic supply pumps can help the situation, but the rate of inventory loss will exceed the makeup capability of the CRD pumps for many hours. The feedwater system will also be unavailable to provide coolant unless the operator has taken action to re-open the MSIVs.

With no high pressure systems available, the only option to prevent core damage is to depressurize the system and attempt to utilize the low pressure ECCS. The plant systems are designed to accomplish this function automatically. When the vessel low-low water level set point is reached (level 1), the "timer" for the Automatic Depressurization System (ADS) is activated.*

The Automatic Depressurization System (ADS) is designed to rapidly reduce reactor system pressure. Several of the safety/relief valves are designated as ADS valves. For ADS operation, these valves are actuated by a separate

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* LPCS and RHR pumps are also started at this time, as are the standby diesel generators.

logic which requires the following conditions:

(1) low-low vessel water level (below Level 1) for a period of at least two minutes

(2) high drywell pressure

(3) adequate discharge pressure from any LPCIS pump or two CSIS pumps

(4) permissive level signal from Level 3 (this signal is required to prevent spurious activation involving a false Level 1 indication).

Once all of the criteria are satisfied, the ADS valves automatically open and dump steam into the suppression pool. This causes a rapid drop in the coolant system pressure. The ADS can also be activated manually.

If depressurization is successful, then the low pressure ECCS or the condensate system can be utilized to replenish coolant and remove decay heat. Two low pressure ECC systems are available. The Low Pressure Coolant Injection system (LPCIS) is one of the three operating modes of the Residual Heat Removal system (RHRS). The LPCIS consists of two independent loops, each with two pumps in parallel.* The pumps take suction from the suppression pool and discharge water into the two recirculation loops. The LPCIS pumps are automatically started on low vessel water level (Level 1) and begin to deliver flow to the core when the system pressure drops below $\sim$ 450 psi. Full flow is delivered at around 300 psi.

The other low pressure ECCS is the Core Spray Injection System (CSIS). The CSIS has two trains, each having one or two AC motor driven pumps. Water is drawn from the suppression pool and discharged into the vessel via separate core spray spargers. These two loops are not interconnected.

The condensate system can also be used to deliver water to the core. This system is a low pressure system which provides water to the vessel through

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*At the time of the Reactor Safety Study the two LPCIS trains were cross tied and a "loop selection" logic was used to inject water from both trains into the intact loop for recirculation line breaks.

the feedwater lines. The condensate pumps draw suction from the hotwell. The hotwell inventory can be maintained by the condensate storage tank.

The combination of pumps required for successful maintenance of inventory at low pressure is a function of the break size and processes which have occurred up to this time. In most instances, any one of the LPCI pumps would be capable of maintaining inventory once level is restored to normal.

Failure of depressurization or low pressure injection leaves no mechanism to replenish the water escaping through the break or being boiled off by the decay heat. Core damage will inevitably occur. If depressurization fails, core melt will occur at a relatively high pressure as determined by the break losses and any safety relief valve discharge. The water level will reach the top of the fuel within 15-30 minutes. If the vessel is depressurized, the core will be uncovered during the blowdown. With insufficient low pressure ECCS flow, the core melt will occur at low pressure.

The previous discussion describes the S_1E sequence as considered for a recirculation line break. The response to a break in the feedwater line or steam line inside containment would differ primarily in the amount of time available before the feedwater pumps are lost. However, for a break outside the drywell some of the early events are significantly different. Though the final end state of the S_1E sequence is the same as for breaks inside containment for the "hands off" sequence, these differences are very important in determining what response the operator can take.

For breaks outside containment, the system response differs in that there is no high drywell pressure signal to actuate the HPCI and trip the reactor. When the postulated break is inside of the steam line tunnel, the first indications would be an increase in temperature and radioactivity in the tunnel. The temperature would eventually rise to the trip point and cause a Group I isolation signal to be given. Group I isolation causes a MSIV trip signal to be initiated. As the MSIVs pass through the 90% open position a

reactor scram is initiated. When the MSIVs reach the 100% closed position two events occur. First, for the breaks considered in this discussion,the leak will be isolated; and second, the PCS will be removed from operation.* Pressure will quickly rise to the SRV set points and will be relieved to the suppression pool. Without the feedpumps in operation, RPV level drops to the low-low level setpoint, at which time the initiation signal for HPCI and RCIC is generated. Since it is assumed that the high pressure makeup systems have either failed or are insufficient, the system must be depressurized with the intent of using the low pressure ECCS's. From this point the sequence description parallels that described previously for a TQUV sequence (see Section 4.2). The absence of a high drywell pressure signal for breaks outside containment should be noted. Without this signal the ADS will not activate automatically. Operator action is required to manually open these valves.

If the break is inside of the turbine building, plant response will be significantly different. The absence of leak detection devices within the turbine building will result in a longer period of time between the onset of the break and any subsequent automatic or operator actions such as reactor scram or MSIV closure. Without quick operator intervention the break will continue to release radionuclides outside of containment. Reactor scram will eventually occur due to a steam-feed flow mismatch, low turbine generator output, or a low vessel water level.

If the leak outside containment is on the large end of the small break spectrum, then the initial symptoms will be different from the results for breaks on the small end of the spectrum. When the leak is large, the first indications would be a drop in turbine generator output, a decrease in the condensate storage tank level, an increase in the radioactivity in the turbine building ventilation and an increase in area radiation monitor readings. There would be an increase in the background noise level, building humidity, and the leak may be visible. When the leak is small there may only be a slight increase in turbine building ventilation radioactivity. Background noise level will rise

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*As mentioned previously, unisolable breaks outside of containment are not considered in these sections.

but it may be discernable only from a short distance from the break. The leak may be visible. If the leak is neither large enough nor in a line where the fluid has high enough radioactivity, a high radiation alarm may not be sounded. In this case, the only indication would be frequent sump pump operation, until such time the operator makes a tour of the area. Once the operator recognizes that a small leak has occured he would manually initiate plant shutdown unless the low vessel water level trip setpoint has already resulted in a reactor scram. The operation of the ADS would again require manual initiation. Recognizing that the leak is outside of containment, the operator would isolate by closing the MSIVs. From this point on the description would again follow that of the TQUV sequence except that the operator will be much more hesitant to re-open the MSIVs to restore feedwater.

4.3.1.2 Operator Response

The role of the operator throughout the progression of the S_1E sequence will be examined in this section. As was the case for the physical description of the events, the affects of the size and location of the break upon the operator's response will be discussed.

To logically address the potential operator response as the sequence develops, an operator action event tree was developed for the S_1E sequence. This diagram is presented in Figure 4.10. This figure will provide the framework for the discussion of operator response in this section. This OAET is similar to the system event tree in Figure 4.9 with modifications to reflect operator responses to the key events. For simplicity, several events were combined into a single event tree heading. The different branches of the event tree have been assigned alphanumeric identifiers for referencing the plant status in the following discussion.

Subsequent to the identification of the initiating event, the majority of the operator actions are similar to those which have been described for the TQUV sequence (Section 4.2.1). The principal operator actions for the S_1E category can be categorized as follows:

(1) identify the nature of the initiating event and its location, if possible

(2) verify occurrence of automatic system responses (e.g. scram, containment isolation, etc.), or manually initiate

(3) identify unavailability of HPCI and RCIC

(4) restore a source of high pressure coolant injection

(5) manually depressurize the system

(6) identify failure of low pressure ECCS's

(7) restore LP ECCS's or condensate/feedwater system

(8) establish long term cooling with the RHRS or PCS

The first operator response for this sequence is to identify the existence of the small break. If the break is inside containment, one of the initial indications will be a rise in drywell pressure with the scram setpoint reached at 2.0 psig. The high drywell pressure setting initiates a reactor trip and provides a start signal for the diesel generators, HPCI, and the low pressure ECCS pumps. The operator should immediately verify that these automatic actions have occurred. As illustrated in Figure 4.10, state 1, this sequence assumes a successful reactor trip. The timing of these automatic responses will depend on the break location and its size, but the high drywell pressure setting is expected to be achieved within one minute after a recirculation line break of the S_1 category. Response times for breaks outside of containment may vary considerably.

The operator must also monitor the drywell pressure to ensure that it does not threaten the integrity of the containment structure. As the containment drywell temperature and pressure rise the operator may initiate containment spray operation. These sprays must be manually activated if the drywell pressure approaches the containment design pressure, or the temperature approaches the ADS valve qualification temperature*. Spray actuation is always manual.

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*The drywell coolers in some plants trip on low-low vessel water level. This will contribute to the pressure/temperature rise. Failure of the vacuum breakers to remain closed between the drywell and the wetwell, if it occurs, would result in vapor suppression failure, also contributing to increasing drywell temperature/pressure.

For breaks inside of the steam line tunnel, the operator would not see a high drywell pressure response. Instead, temperature and radioactivity levels in the tunnel would increase to the high tunnel temperature setpoint . At this point a Group I isolation signal would be generated and the MSIVs would close. This would initiate a scram while also isolating the leak and removing the PCS from service. The operator should verify the reason for the MSIV closure (i.e. a leak in the steam tunnel) and ensure that the reactor scram has occurred. Following MSIV closure the pressure in the vessel will rise rapidly to the SRV setpoints. The vessel level will begin to fall and will reach the level 3 setpoint, at which time the response is as described for breaks inside containment. Significant differences between state 1 in this case and the case described for breaks within the containment are the absence of high drywell pressure (necessary for automatic depressurization to occur) which in turn results in an absence of an initiation signal for the HPCI pumps prior to level 2. The isolation of the break with the MSIVs would also result in an increased hesitancy on the part of the operator to restore feedwater, since this action would require re-opening the MSIVs, thus allowing further escape of radionuclides through the break.

If the break occurs inside of the turbine building there will be no immediate automatic actions initiated. However, the plant will eventually trip due to one of a variety of reasons including low vessel level (level 3). Up until that time the operator would be alerted to the leak only by monitoring his other plant performance indicators. These include turbine generator output, which may drop somewhat due to the decrease in steam flow to the turbine. The radiation level in the turbine building would increase due to the activity released through the break. Turbine building humidity would also increase. If the operator notices these changes and establishes that they are due to a leak, his actions would include isolating the leak with the MSIVs and bringing the plant to a shutdown condition. He may choose to insert the control rods manually before closing the MSIVs so as to avoid the rapid insertion caused by a reactor scram. This decision would be based upon the size of the leak and the rate of change of the plant conditions due to this break. Regardless of the order in which the steps are taken, the closure of the MSIVs will again result in a loss of the PCS and a rise in system pressure. In anticipation of this the

operator may choose to attempt a manual start of the HPCI and RCIC pumps before he closes the MSIVs. If the operator isolates the break, his subsequent actions are much like those described for the TQUV sequence in section 4.2.

If the break is inside containment the operator's first indication of the leak will be a rise in drywell pressure. The high drywell pressure should automatically initiate HPCI operation. For the S_1E sequence the HPCI system is assumed to fail or is unavailable. The operator must therefore take immediate action to maintain feedwater operation. This action is critical since RCIC and control rod drive water are insufficient to accommodate for the fluid lost through an S_1 category break. Furthermore, the operator has no way of knowing this with the feedwater operating. The flow controller on the feedwater will compensate for the leakage out the break and the water level may remain relatively stable. If RCIC and control rod drive water are available, the operator could mistakenly assume that they would be capable of maintaining inventory if feedwater were lost.

The small break and subsequent events associated with the initial few minutes of this sequence would not be expected to produce a feedwater trip. However, the pressure in the vessel will decay slowly. The operator should place the reactor mode switch in the "shutdown" position. This will bypass the low pressure closure setpoint for the MSIVs. If the operator fails to do this, the MSIVs will close at 850 psi which automatically trips the feed pumps, unless the feedwater system is taken out of its automatic "run" mode. The time before this pressure is reached varies with the break size and location, but will be at least five minutes for a liquid line break. Switching the feedwater system to level control is a routine procedure following scram and therefore the operator should be relatively familiar with the manual operation of the feedwater system. However, care must be taken to assure that feedwater does not provide too much coolant, as the system can trip off on high water level.

If the break occurs in a main steam line, the system response may be somewhat different. The key operator action of keeping the feedwater system on line may involve a more rapid operator response, as the pressure reduction may occur much more quickly. If the MSIVs close, steam flow to the feed pump turbines is cut off. To restart feedwater would require reopening the MSIVs as well as the normal operations to recover from a feedwater trip. This action may not be possible in the time available to prevent core damage.

It is postulated for this S_1E sequence that the feedwater system successfully maintains inventory in the vessel for sometime, but eventually trips when the pressure decays to 850 psi, or due to some flow instabilities. The MSIVs also close. This state is indicated as state 2 in the OAET. Up until the time of feedwater trip the operation of the HPCI pumps has not been required, although a signal to start should have been generated by the high drywell pressure. At state 2, the closure of the MSIVs will result in some system repressurization, possibly up to the setpoints of the safety/relief valves. The loss of feedwater, coupled with the fluid loss through the break and the SRVs (if open) results in a decrease in vessel level. The elevated system pressure requires that a high pressure inventory makeup system be established. The HPCI system would normally satisfy this requirement, but for the S_1E sequence it is not available (state 3). Restoration of a high pressure system is thus necessary. The use of the feedwater system would require operator action to re-open the MSIVs and restart the feedwater pumps.

The operator may also be able to quickly restore HPCIS operation. For instance, if the automatic actuation logic failed to align and activate the HPCI system, the operator could manually initiate HPCI. If the system has been isolated for maintenance, but the maintenance act or existing system condition at the time of the accident does not preclude the operation of the system, the operator may be able to restore the system to service by utilizing the controls provided in the control room.

If the feedwater system is placed back in service and is successful in maintaining level, or if the HPCI system has been restored (state 4), then the primary system can be depressurized and cooled using the feedwater and condensate systems or the RHR system. This process would be analogous to a normal plant cooldown after scram. The major difference is the continued leakage of water into the drywell.

If the main feedwater system trips and cannot quickly be restored, and the HPCIS also remain unavailable, there is insufficient high pressure makeup. This condition is represented by state 4a in the operator action event tree.

If the operator is unable to restore feedwater and HPCI is unavailable, the vessel water level will continue to decrease. The rate of vessel

inventory depletion may be slow if RCIC or control rod drive hydraulic systems are functioning. However, for breaks in the S_1 size range, these systems are insufficient to match the break flow losses. The system must then be depressurized and the low pressure injection systems used to maintain inventory. The ADS is designed to automatically blowdown the system if this situation occurs. The initiating conditions for ADS operation are:

- o high drywell pressure
- o low-low vessel water level (below level 1) for a two minute period
- o permissive signal form level 3 water level sensor
- o adequate discharge pressure from one LPCI pump or two CSIS pumps.

These conditions may not be satisfied for the S_1E sequence. The failure of the ADS may be due to one of three reasons:

1) no drywell pressure signal (if break is outside of containment)
2) failure of the ADS logic or valves, or
3) inadequate discharge pressure from the low pressure ECCS's.

Failures of the first two types would require operator action to open the ADS valves. However, for the third type of failures the operator would be hesitant to depressurize the vessel without having some means of coolant injection available. Therefore his first actions at that point, given that he recognizes that the low pressure systems are unavailable, are associated with restoring one of these systems. As with the high pressure ECCS, maintenance outages may be a significant contributor to the unavailability of either the core spray or injection systems. It is possible that one of these systems could be temporarily bypassed, but not functionally disabled. In this case, alert operator response could return the system to operation. He then can proceed to depressurize the system by using the ADS or other means. These other means include the other safety/relief valves, steam dump to the condenser, use of the Reactor Water Cleanup System, or even blowdown through the HPCI/RCIC turbines. More detail is provided in Section 4.2.1, in the description of the TQUV sequence.

Successful automatic depressurization with the ADS is indicated by state 5. If the operator has not been successful in using either the ADS or other means to depressurize the reactor (state 6a), the core will uncover and eventually begin to melt. The onset of melting is predicted to occur within approximately one half hour after the beginning of the transient. Operator response would then be to ensure operation of containment systems and monitor the progression of the melt as needed to effectively mitigate the consequences.

If automatic depressurization fails but one of the alternate means of depressurization is successful (including manual depressurization through the ADS or other safety/relief valves), the plant is in state 6. At this point the low pressure injection systems are assumed to be successful in reflooding the core and maintaining vessel inventory. Operator action in this state involves monitoring vessel water level and pressure, and controlling the low pressure systems as necessary to maintain core coverage without overfilling the vessel. The pressure must be kept below the shutoff head of the LPCI and/or CSIS pumps (about 450 psi). To do this, the operator would use the same mechanism which was utilized to depressurize the system initially. The next major action is to bring the plant to a stable, cool shutdown condition by switching to a recirculation mode of operation. These actions are discussed in Sections 4.2.2 and 4.3.2.

From examination of the operator action event tree, it can be seen that if the ADS successfully reduces system pressure (state 5), then the S_1E sequence postulates that low pressure injection subsequently fails. This means that the LPCI and CSIS were available when depressurization occurred, but they subsequently fail or do not supply enough coolant to restore water level in the core. The reactor is now depressurized, the core has been uncovered during the blowdown, but coolant inventory cannot be maintained. Immediate operator action is required to prevent core melting.

If the operator cannot restore the low pressure ECCS's, another potential operator action is to use the condensate system. The electrically driven condensate pumps could be restarted and deliver water from the condenser

hotwell via the feedwater lines.* The main feed pumps would not be required as the condensate pumps are capable of delivering coolant to the vessel after blowdown. In this instance, water from the condensate storage tank could be used to supplement the hotwell inventory as a source of injection water until the bypass system can be activated. The condensate storage tank inventory can be maintained using the fire water system.

If the operator is successful in providing low pressure to the vessel (states 7a or 8), the operator must bring the system to a stable, shutdown condition. Actions necessary to accomplish this are discussed in detail in Section 4.2.2 (TW sequence) and Section 4.3.2 (S_1I sequence). The ability to provide long term cooling may be impaired at state 8 if the failure mode of the low pressure ECI involved failure of all LPCI pumps, as these pumps are required for torus cooling. Response to this type of failure is also discussed in the aforementioned sections.

If the operator fails to provide any low pressure coolant injection the plant will be in a state represented by state 8a of the OAET. Core damage is expected to occur.

Several different situations have been discussed in the previous sections in response to several postulated failures in this sequence. In summary, these potential means of maintaining inventory are:

(1) Control of feedwater or restoration of HPCI (state 4)

(2) Condensate system or restoration of LP ECI (state 8)

(3) Successful operation of LP ECI (state 7a)**

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*The condensate pumps were utilized during the Browns Ferry fire incident.

**This sequence involves automatic ADS failure, but successful depressurization by alternate means.

4.3.1.3 Symptom Matrix

The symptoms exhibited by the plant throughout the progression of the S_1E sequence are tabulated in Table 4.12. The behavior of each of the key BWR parameters (delineated and discussed in Section 3) is described in Table 4.12 for each state appearing in the operator action event tree (Figure 4.10).

As discussed in Section 3, multiple symptoms involving the same parameter may be indicated in Table 4.12. These multiple (and sometimes contradictory) symptoms are due to either:

1) A state where the parameter is going through a transition from one trend to another; in these cases the operator may actually observe either or both symptoms.

2) A state where the physical plant response (e.g. voiding in the core) may affect the symptoms observed by the operator in these; cases indications of these symptoms may be highly unreliable.

3) A state where the behavior of a certain parameter is simply unknown or highly uncertain.

If potential ambiguities are identified involving OAET states with such multiple symptoms, these few cases can be examined individually in more detail.

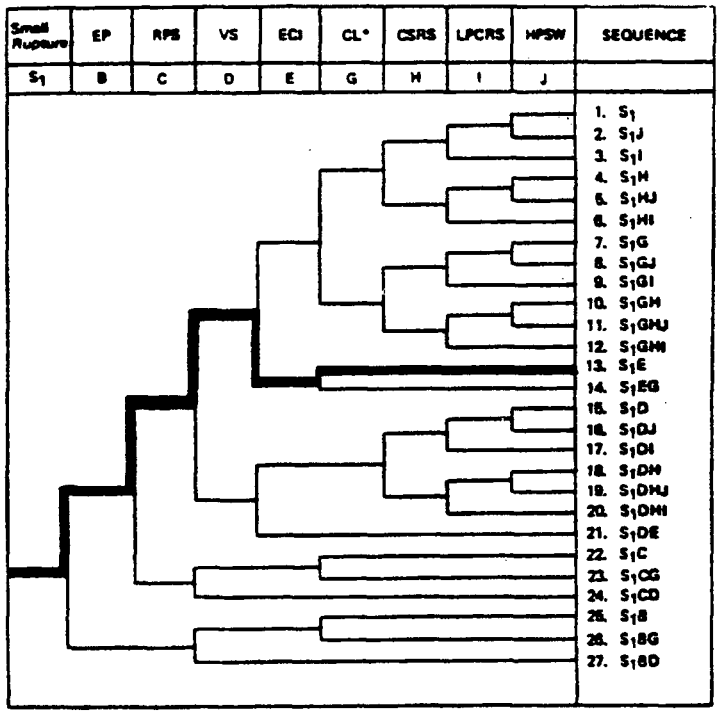

Figure 4.9 Reactor Safety Study Event Tree for BWR Small LOCA (S_1, approximately 2.5-8 inch diameter): S_1E Sequence Highlighted.

Figure 4.10 S_1E Sequence - Operator Action Event Tree

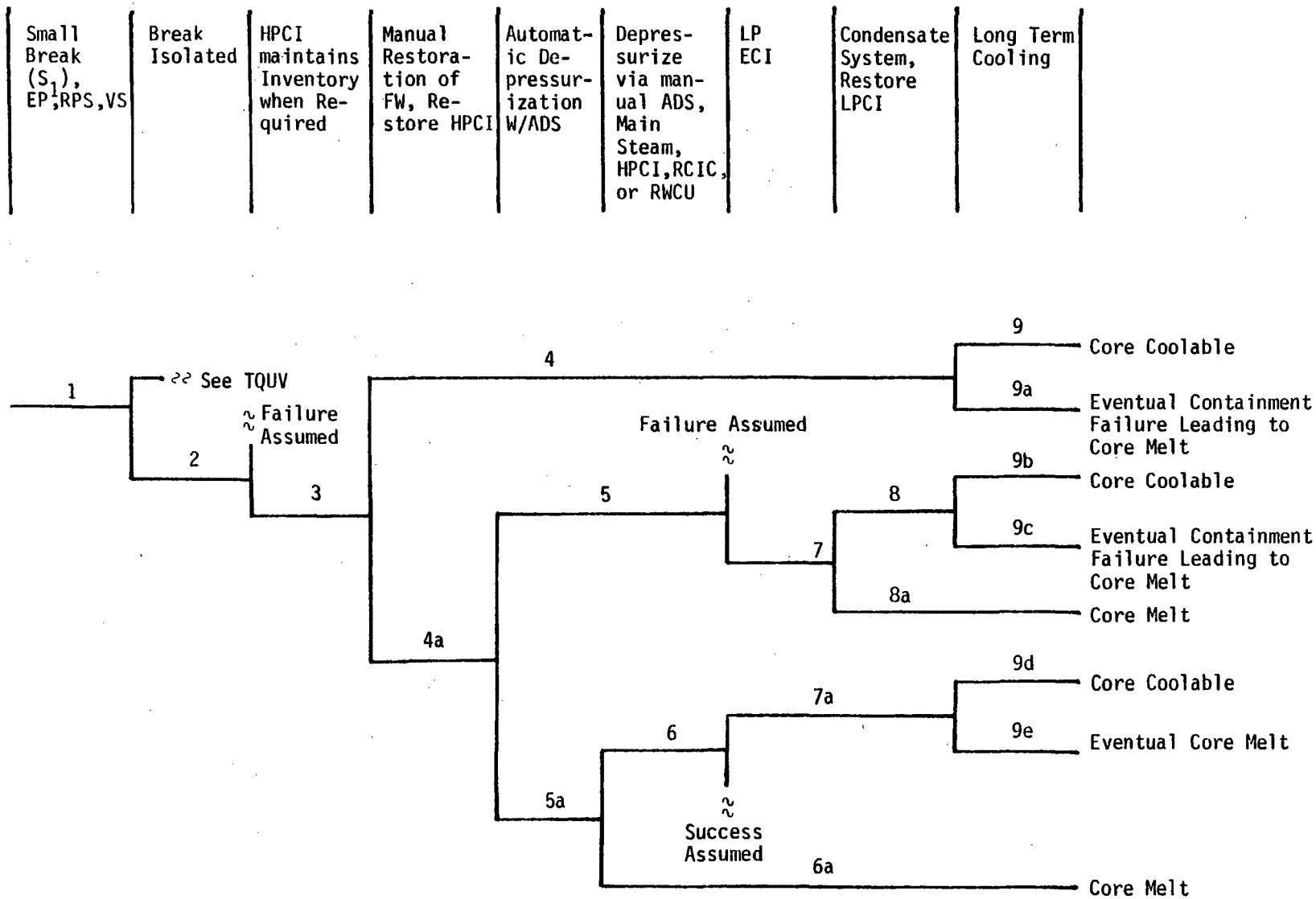

Table 4.12

Sequence: S_1E

BWR OAET Symptom Matrix Sheet 1

Symptoms 1-19

(See Section 4.2.1.3 for explanation of Symbols)

Symptoms	RPV Level				RPV Press.									RPV Temp		Re-actor Pwr.		Rod Pos.	Bo-ron
OAET State	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19
S_1E-1	0	0	0	0	1	1	1	0	0	1	0	1	1	0	0	1	0	1	0
S_1E-2	0	0	0	0	1	1	1	1	0	1	0	1	1	0	0	1	0	1	0
S_1E-3	0	1	0	0	0	1	1	1	1	1	0	1	1	0	0	1	0	1	0
S_1E-4	0	1	0	0	0	1	1	1	1	1	1	1	1	0	0	1	0	1	0
S_1E-4a	0	1	1	0	0	1	1	1	1	1	0	1	1	0	0	1	0	1	0
S_1E-5	1	1	1	1	0	1	0	1	1	0	1	1	1	1	0	1	0	1	0
S_1E-5a	0	1	1	0	0	1	1	1	1	1	0	1	1	0	0	1	0	1	0
S_1E-6	1	1	1	1	0	1	0	1	1	0	1	1	1	1	0	1	0	1	0
S_1E-6a	0	1	1	1	0	1	1	1	1	1	0	1	1	0	0	1	0	1	0
S_1E-7	1	1	1	1	0	1	0	1	1	0	1	1	1	0	0	1	0	1	0
S_1E-7a	1	1	1	0	0	1	0	1	1	0	1	1	1	0	0	1	0	1	0
S_1E-8	1	1	1	0	0	1	0	1	1	0	1	1	1	0	0	1	0	1	0
S_1E-8a	0	1	1	1	0	1	0	1	1	0	1	1	1	0	0	1	0	1	0

Table 4.12 (continued)

Sequence: S_1E

BWR OAET Symptom Matrix Sheet 2

Symptoms 20-44

Symptoms	S.P. Level										S.C. Press					S.P. Temp					DW Prs	Drywell Temp.			
OAET State	20	21	22	23	24	25	26	27	28	29	30	31	32	33	34	35	36	37	38	39	40	41	42	43	44
S_1E-1	0	0	1	0	1	0	0	0	1	0	0	0	0	0	1	1	0	1	0	1	1	0	0	1	1
S_1E-2	0	0	1	0	1	0	0	0	1	0	0	0	0	0	1	0	1	1	0	1	1	0	0	1	1
S_1E-3	0	0	1	0	1	1	0	0	1	0	0	0	0	0	1	0	1	1	0	1	0	0	0	1	1
S_1E-4	0	0	1	0	1	1	0	0	1	0	0	0	0	0	1	0	1	1	0	1	0	0	0	1	1
S_1E-4a	0	0	1	0	1	1	0	0	1	0	0	0	0	0	1	0	1	1	0	1	0	0	0	1	1
S_1E-5	0	0	1	0	1	1	0	0	1	0	0	0	0	0	1	0	1	1	0	1	0	0	0	1	1
S_1E-5a	0	0	1	0	1	1	0	0	1	0	0	0	0	0	1	0	1	1	0	1	0	0	0	1	1
S_1E-6	0	0	1	0	1	1	0	0	1	0	0	0	0	0	1	0	1	1	0	1	0	0	0	1	1
S_1E-6a	0	0	1	0	1	1	0	0	1	0	0	0	0	0	1	0	1	1	0	1	0	0	0	1	1
S_1E-7	0	0	1	0	1	1	0	0	1	0	0	0	0	0	1	0	1	1	0	1	0	0	0	1	1
S_1E-7a	0	0	1	0	1	1	0	0	1	0	0	0	0	0	1	0	1	1	0	1	0	0	0	1	1
S_1E-8	0	0	1	0	1	1	0	0	1	0	0	0	0	0	1	0	1	1	0	1	0	0	0	1	1
S_1E-8a	0	0	1	0	1	1	0	0	1	0	0	0	0	0	1	0	1	1	0	1	0	0	0	1	1

Table 4.12 (continued)

Sequence: S_1E

BWR OAET Symptom Matrix Sheet 3

Symptoms 45-65

Symptoms	PC Level	PC Temp	HPCI		RCIC		LPCI		CS		DG		RHR		ADS			MC		CRD	
OAET State	45	46	47	48	49	50	51	52	53	54	55	56	57	58	59	60	61	62	63	64	65
S_1E-1	0	0	1	1	1	1	1	1	1	1	1	0	1	0	0	0	1	1	1	1	0
S_1E-2	0	0	1	1	1	1	1	1	1	1	1	0	1	0	0	0	1	1	1	1	0
S_1E-3	0	0	0	1	1	1	1	1	1	1	1	0	1	0	0	0	1	1	1	1	0
S_1E-4	0	0	1	1	1	1	1	1	1	1	1	0	1	0	0	0	1	1	1	1	0
S_1E-4a	0	0	0	1	1	1	1	1	1	1	1	0	1	0	1	1	1	1	1	1	0
S_1E-5	0	0	0	1	1	1	1	1	1	1	1	0	1	0	1	0	0	1	1	1	0
S_1E-5a	0	0	0	1	1	1	1	1	1	1	1	0	1	0	1	0	0	1	1	1	0
S_1E-6	0	0	0	1	1	1	1	1	1	1	1	0	1	0	1	0	0	1	1	1	0
S_1E-6a	0	0	0	1	1	1	1	1	1	1	1	0	1	0	1	0	0	1	1	1	0
S_1E-7	0	0	0	1	1	1	0	1	0	1	1	0	1	0	1	0	0	1	1	1	0
S_1E-7a	0	0	0	1	1	1	1	0	1	1	1	0	1	0	1	0	0	1	1	1	0
S_1E-8	0	0	0	1	1	1	1	0	1	1	1	0	1	0	1	0	0	1	1	1	0
S_1E-8a	0	0	0	1	1	1	0	1	0	1	1	0	1	0	1	0	0	1	1	1	0

Table 4.12 (continued)

Sequence: S_1E

BWR OAET Symptom Matrix Sheet 4

Symptoms 66-79

Symptoms	SLC		RWCU		Alternate Inj.		C/F		Recirc. Pumps		MSIV		RHR Interlock	
OAET State	66	67	68	69	70	71	72	73	74	75	76	77	78	79
S_1E-1	1	0	1	0	1	0	1	1	1	0	1	1	0	1
S_1E-2	1	0	1	0	1	0	1	1	1	0	1	1	0	1
S_1E-3	1	0	1	0	1	0	1	1	1	0	1	1	0	1
S_1E-4	1	0	1	0	1	0	1	1	1	0	1	1	0	1
S_1E-4a	1	0	1	0	1	0	0	1	1	0	0	1	0	1
S_1E-5	1	0	1	0	1	0	0	1	1	0	0	1	0	1
S_1E-5a	1	0	1	0	1	0	0	1	1	0	0	1	0	1
S_1E-6	1	0	1	0	1	0	0	1	1	0	1	1	0	1
S_1E-6a	1	0	1	0	1	0	0	1	1	0	0	1	0	1
S_1E-7	1	0	1	0	1	1	0	1	1	0	0	1	0	1
S_1E-7a	1	0	1	0	1	0	1	1	1	0	1	1	0	1
S_1E-8	1	0	1	0	1	0	1	1	1	0	1	1	0	1
S_1E-8a	1	1	1	1	0	1	0	1	1	0	0	1	0	1

4.3.2 Small Break LOCA with Failure to Remove Decay Heat from the Containment (S_1I)

The previous section dealt with an accident initiated by a small break in the RCS piping followed by a failure to maintain inventory in the vessel. This section will discuss a small break LOCA with successful inventory maintenance, but with a failure to remove decay heat in the long term. A summary of the accident sequence is included in Section 4.2.2.1. The Operator Action Event Tree (OAET) for this sequence is provided as part of Section 4.3.2.2, while Section 4.3.2.3 presents in tabular form the physical plant response at each plant state in terms of key parameters discussed in Section 3.

4.3.2.1 Sequence Descriptions

A loss of coolant accident (LOCA) is defined as being a break in the primary system that causes a release of primary fluid to the atmosphere either inside or outside of the primary containment. A LOCA can be the initiating event for an accident sequence or it can be induced by another accident. For this analysis the break is postulated to occur while at full power and without prior warning.

LOCAs directly affect one of the critical safety functions - maintenance of adequate coolant inventory. When coolant inventory is lost and not replaced, eventual core damage will result. To protect a BWR core against this possibility, a number of emergency cooling systems have been provided. They include the High Pressure Coolant Injection System (HPCI), Reactor Core Isolation Cooling System (RCIC), Low Pressure Coolant Injection System (LPCI), and Core Spray System.

This section addresses small LOCAs. As described previously in Section 4.3.1 a small LOCA is defined as a breach in the primary system which is too small to cause a depressurization of the reactor pressure vessel sufficient to allow use of the low pressure injection systems. This requires the use of HPCI and RCIC to maintain reactor pressure vessel (RPV) coolant inventory until the vessel is depressurized. This sequence assumes that a small break occurs

(event S_1), and the HPCI and/or RCIC is successful in supplying the vessel with coolant. The low pressure emergency core coolant injection systems (LPCI, CSIS) begin to operate when the system pressure drops below the shutoff head of the pumps. However, by definition of this sequence, the RHR fails to remove decay heat from the suppression pool (event I). Failure to repair the RHR and restore it to service, coupled with a failure to provide cooling with the Power Conversion System (PCS), results in a rise in containment pressure until its predicted failure pressure is reached. The Reactor Safety Study, which assumed coolant makeup only from the torus, describes the subsequent events as follows:

> "Since the water in the suppression pool will be at the saturation temperature associated with the partial pressure of steam within the containment, the rapid depressurization which occurs upon containment failure will cause the water in the suppression pool to flash and cause cavitation of the low pressure ECCS pumps. These pumps will then have insufficient NPSH to continue operating. Hence, the core will not be maintained in a reflooded condition, the water remaining in the core region will be vaporized, and the core will melt."

Table 4.13 is a compilation of key events which may occur during the S_1I sequence. Figure 4.11 presents the system event tree developed in WASH-1400 for small breaks in BWRs. The S_1I sequence is highlighted. A more detailed event tree is developed in Section 4.3.2.2 which displays the operator actions in conjunction with the key plant states.

As described in Section 4.3.1 for the S_1E sequence, the plant and operator responses may vary depending upon the size and location of the break. This is especially true for breaks which occur outside of the containment. For the discussion which follows, the break is assumed to be a $0.1ft^2$ break in the recirculation line inside of containment. Discussion concerning other locations is included in Section 4.3.1.

The immediate response to this break is a rapid rise in drywell pressure. When the high drywell pressure trip setpoint of 2.0 psig is reached a reactor scram signal is generated. The HPCI pumps are signalled to start, and the containment isolation valves are closed.

Table 4.13

Key Sequence of Events in "S_1I" Sequence

Time* ($.1ft^2$ recirc. line)	Event
0	Break in recirc. line.
3-30 (est.) sec.	High drywell pressure scram; HPCI initiation signal; ADS permissive; containment isolation valves close; drywell coolers trip.
33-60 (est.) "	HPCI flow enters vessel.
∿120 "	RCIC, HPCI, FW Trip on high vessel level (level 8).
∿150 "	Level 3 reached, HPCI restarts. ADS level permissive. Break flow exceeds HPCI flow - vessel level drops.
∿250 "	Level 2 Trip Recirc. Pumps, Close MSIVs.
∿300 "	Vessel depressurized to 350 psia, LPCI begins to inject.
600 "	Failure of pool cooling and shutdown cooling modes of RHRS.
∿1800 "	Vessel refilled.
2 hours	Pool temperature exceeds 49°C (120°F).
5-7	Suppression Pool Temperature Exceeds 212°F.
18	Drywell Coolers Fail due to High Drywell temperature and pressure.
23	Containment pressure ∿90 psia. Suppression Pool Temperature ∿320°F
23-27	RPV repressurizes.
28-35	Drywell Fails when internal pressure exceeds failure limit (132 psia for Mark I)**

* hours unless otherwise noted.

** WASH-1400 used a failure pressure of 177 psia, estimated to be reached in 27 hours.

For the S_1I sequence it is assumed that the automatic systems successfully maintain inventory. The feedwater system may have been in operation, but has tripped, and the MSIVs have closed (see also Section 4.3.1)*. The HPCIS has assumed primary responsibility for inventory maintenance, and the primary system is cooling and depressurizing.

When the vessel pressure drops below the shutoff head of the low pressure ECCS's (around 450 psig), these systems begin to provide coolant to the vessel. As the pressure continues to fall, the high pressure systems eventually trip (at approximately 100 psig). Coolant inventory control is then dependent upon the operation of the LPCI and core spray systems. Decay heat is transferred directly to the suppression pool, and the temperature of this pool rises.

For the S_1I sequence it is postulated that the Low Pressure Coolant Recirculation System (LPCRS) fails, thus resulting in a failure to cool the torus. The LPCRS normally circulates water from the suppression pool through RHR heat exchangers where the pool water is cooled by plant service water and then returned to the pool. Failure to provide this circulation and/or heat removal results in a continuing rise in suppression pool temperature and wetwell air pressure. If torus cooling is not reestablished, the temperature and pressure in the drywell will also eventually increase. Without torus cooling it is postulated that the containment structure will eventually fail.

In addition to threatening the containment integrity, the rising drywell pressure has one other effect. When the drywell pressure increases sufficiently, it is postulated that the safety/relief valves will be forced closed due to backpressure, insufficient differential pressure across the valve, or failure of the air supply. The vessel pressure will begin to rise until it exceeds the shutoff head of the low pressure pumps, and continue upward until reaching the setpoints of the safety valves. Flow from the low pressure ECCS's would halt unless the operator establishes another method of depressurization,

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*The feedwater, HPCI, and RCIC pumps would trip at high level (level 8). The HPCI pumps would reset when water level falls back to the low-low level (level 2), at which time the MSIVs would close. Inventory would be maintained by the HPCIS.

and a high pressure system would be required to replace coolant which is lost through the unisolated break. If coolant inventory is not maintained, the core will eventually uncover and fuel damage will occur. Hence, the core could be damaged even before the containment fails.

4.3.2.2 Operator Response

The discussion in Section 4.3.2.1 focused on a "hands off" description of the S_1I sequence following postulated system failures. This section will present operator actions which can be taken in response to the system failures. *A detailed operator action event tree for the S_1I is included as Figure 4.12.*

As depicted in the OAET and as is discussed below, the principal operator actions can be categorized as follows:

1) Identify occurrence of break and determine its location, if possible

2) Verify occurrence of automatic system responses to small LOCA inside containment (e.g. scram, containment isolation, HPCIS initiation, etc.)

3) Verify inventory control with HPCI, RCIC, LPCI, CSIS

4) Identify failure of torus cooling (LPCRS failure)

5) Lower vessel pressure, maintain or restore vessel coolant inventory

6) Restore torus cooling with LPCRS or establish decay heat removal through PCS.

The immediate operator response is to identify the initiating event and verify that the correct actions in response to this event are implemented. The break inside of containment is characterized by a high drywell pressure and possibly a slight drop in vessel level. The feedwater controller should automatically open to compensate for the loss out of the break, so reactor vessel level should remain relatively constant.

A high drywell pressure scram will be initiated at 2.0 psig. At this time the control rods should insert to reduce reactor power to decay heat levels. The high pressure scram should also result in a containment isolation signal (the MSIVs remain open) and an initiation signal for the HPCIS. The operator should verify that these actions occur, or he should initiate them. For the S_1I sequence it is assumed that these automatic responses all occur (states 1 and 2 of Figure 4.12).

The operation of the HPCI system increases the level in the vessel. It is postulated in this sequence that the water level increases to the level 8 (high level) elevation. When the level reaches this point, the HPCI, RCIC, and feedwater pumps trip. The HPCI pumps automatically restart once the level drops back to the low-low level setpoint (level 2)*. This is also indicated by state 2. The system may thus repressurize, due to the MSIV closure, and the safety valves may open. The HPCIS will be required to maintain the level while the system is at elevated pressure. The operator should attempt to prevent oscillation of the water level by controlling the HPCI pumps.

With the HPCIS maintaining level, the vessel pressure begins to decline. Successful inventory maintenance is indicated by state 3 of the OAET for this sequence. With the RCIC and/or HPCI systems maintaining vessel level, the fission product decay heat transported to and stored in the suppression pool via the safety/relief valves, HPCI/RCIC discharge, and break will result in a gradual increase in the pressure in the primary containment. The containment cooling mode of the RHR system is designed to limit the suppression pool water temperature. This mode of the RHR system, which is manually initiated, takes suction from the suppression pool, pumps the fluid with any of four low pressure pumps through the RHR system heat exchangers, and returns the cooled fluid back to the suppression pool. The Service Water (SW) System is used to transport the heat from the RHR heat exchangers to the environment.

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* The MSIVs also close at this time.

In the S_1I sequence it is postulated that suppression cooling fails and the operator is unable to immediately restore it to operation (state 4). Therefore, the suppression pool temperature continues to rise. An important point to note in considering operator actions at this stage in the accident sequence is that there is a maximum suppression pool temperature above which complete condensation of blowdown steam may not occur. The operator should be aware that once this temperature limit is reached, primary system blowdown could subject the primary containment to a pressure transient which could threaten its structural integrity. In order to prevent this situation the operator should reduce primary system pressure to approximately 100-200 psi when the suppression pool temperature rises above 120°F*. It is assumed for this analysis that the operator is successful in these actions (state 5).

Following reactor depressurization the operator should verify the operation of the inventory control systems (LPCI, RCIC, etc.). If the LPCI pumps are not available (their failure may have resulted in the suppression pool cooling failure), the preferred course of action would be to maintain coolant injection with either:

(a) the RCIC aligned to the CST to maintain adequate lube oil cooling despite potentially high suppression pool levels; or

(b) the CRD system also aligned to the CST**

Once inventory maintenance is assured (state 6), the operator must continue his attempts to restore torus cooling with the RHR or PCS. Once again

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*Because the system has been depressurizing all along due to the successful maintenance of coolant inventory and the continued discharge through the break and SRVs, the pressure drop at state 5 due to depressurization to 100-200 psi may not be severe.

**If the low pressure systems are available they may be used unless the system repressurizes. Note also that while HPCI operation can continue the operation may be somewhat degraded if the suppression pool temperature exceeds the recommended design lube oil cooling temperature of 170°F.

it is postulated that the operator cannot restore suppression pool cooling. If suppression pool cooling has been unsuccessful, one of the operator's first responses would be to attempt cooling with the PCS.

In order to restore the PCS, the operator would be required to reopen the MSIVs that closed on low level, take the required actions to verify or re-establish vacuum in the condenser and perform the control actions to establish steam flow from the vessel through the bypass valves to the condenser. Many of these actions will be similar to those involved in a start-up procedure. Perhaps one of the more difficult problems the operator might encounter in restoring PCS is starting up the main feedwater pumps under the conditions of low steam flow to the turbine. The electrically driven condensate pumps can be used for returning water to the vessel but the primary system pressure must be appreciably reduced for these pumps to deliver adequate flow. Therefore, this option may not be available if the operator cannot maintain depressurization of the system. Opening the MSIVs may provide enough initial depressurization for this method to be successful. Depressurization through the pressure regulators may also be useful. At state 7 it is also postulated that the operator is unsuccessful in completely restoring the PCS to service. The MSIVs remain closed.

Without cooling, the temperature of the suppression pool continues to slowly rise. The pressure of the wetwell air space also rises until it exceeds the pressure of the drywell and the vaccuum breakers open. At that time the drywell containment pressure will then begin to slowly rise. Eventually, the pressure will cause the high back pressure trip of both the RCIC (around 25psig at approximately 14-17 hours, depending upon the drywell cooler status and the effectiveness of the sprays*) and the HPCI (about 65 psig at approximately 20-26 hours). If this occurs, only the control rod drive pumps (a high pressure system) and the low pressure systems would be available for inventory control. By definition, the CRD pumps are not adequate to maintain inventory for S_1-sized breaks at high pressure.

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*The operator should initiate suppression spray operation after noticing an increase in suppression chamber pressure. He should initiate drywell sprays if the drywell temperature approaches 340°F.

Up until state 7 the reactor system has been depressurizing through the break, the safety valves, and the HPCI/RCIC discharge. The suppression pool is continuing to act as a heat sink and the containment temperature and pressure are rising. At this point the ability to maintain depressurization through the safety/relief valves becomes questionable for one of several reasons. The air-operated safety valves may close due to failures of the air supply, failure to maintain sufficient differential pressure across the valves, or failure of the valve components themselves, as described below.

To maintain depressurization, the operator is instructed to hold one SRV in the open position. To do this requires that there be adequate air pressure from the accumulators or air compressors to actuate the pilot operated solenoids which are used for the "relief" portion of the safety/relief valves. If the differential pressure across the valve actuators is less than approximately 30-50 pounds (between the air system and the containment), the valves will close and cannot be re-opened manually. With typical accumulator pressures of 100 psia, containment pressures in excess of about 70 psia are likely to compromise the ADS air supply effectiveness. Since the containment pressure without RHR or PCS recovery will rise to above this "threshold", it is very possible that the air supply will be inadequate to maintain the valves in an open position. (One other possible reason for valve closure is that the high containment temperature and humidity may also quickly exceed the environmental qualification envelope for the air operated SRVs, resulting in failures of components necessary for the valves to remain open (see reference 5).

For the above reasons it is postulated that the safety valves fail to remain open at state 7. The size of the leak is such that it does not adequately maintain reactor depressurization. The operator must either establish an alternate pathway to relieve reactor pressure or restore a high pressure system to operation. The possible alternative depressurization paths are to the main condenser or through the HPCI and RCIC system turbines, while the only high pressure systems capable of maintaining inventory for a break of this size are the HPCIS and the feedwater system.

At state 7 the operator may attempt to open the MSIVs or the drain valves which bypass the MSIVs, even if he cannot restore the main condenser to operation. If the condenser cannot be placed in service before these valves are opened, the seals on the condenser may burst and create a leak path outside of containment. However, if the operator does not depressurize through the MSIVs, coolant inventory may not be maintained, especially since the CRD and RCIC systems are postulated to be incapable of keeping up with leakage through the break. The vessel pressure may rise back to the safety valve setpoints, at which time they will open and increase the apparent size of the leak path.

If the main steam valves cannot be opened, the operator can attempt depressurization through the HPCI or RCIC steam lines. However, this may require some rather complex actions. The HPCI and RCIC lines may be isolated due to the high temperature which exists in the areas containing the pumps. Thus the operator may need to bypass some isolation valve interlocks before he can release steam through these lines. Defeating the interlocks may require entering the high temperature areas, and therefore it may not be possible to bypass them. In addition, depressurization through the HPCI/RCIC lines will be inside containment, and will add energy to an already hot and pressurized enclosure. Again, the operator may be very hesitant to bypass these lines and increase the potential for containment failure and radionuclide release outside of containment.

If the operator has been successful in depressurizing the vessel, as indicated by state 8 of the OAET, he must still maintain coolant inventory. This can be provided by the low pressure ECC systems, if available, or the condensate system. The condensate system draws suction from the hotwell and provides water to the vessel through the feedwater lines. The hotwell inventory can be maintained by the CST, which in turn can be provided with makeup water by the fire water system. It is postulated that given the operator's success in maintaining inventory up to this point, continued inventory makeup will also be successful (state 9). However, the operator must still restore the RHR or PCS to operation to ensure continued core coolability.

It may be possible that depressurization at state 8, if through the MSIVs, was accompanied by restoration of the main condenser. The plant would then be in state 10 as a direct result of the actions taken to reach state 8. If the PCS is in service, the majority of decay heat will bypass the suppression pool and be removed by the main condenser. The rate of increase in suppression pool temperature and containment pressure should decline. However, the break inside containment will continue to discharge and add energy to the containment. Hence, some form of torus cooling may eventually be required, but it is postulated here that if the operator successfully re-establishes heat removal with the PCS, this will be sufficient to prevent over-pressurization of the containment. However, if cooling of the core is being accomplished with the addition of water from an outside source (i.e. no recirculation of suppression pool water is occurring), the suppression pool water level will continue to rise. Eventually it will rise to the point where structural integrity may be threatened. The operator may feel compelled to halt the addition of water from outside sources, even if this is cooling the core. This is obviously a point of concern and of potential conflict to the operator. If recirculation cannot be restored, the containment may leak or fail. If cooling is not continued to the core, fuel damage may occur. One possible solution if this situation occurs is for the operator to continue to cool the core, but at a minimum injection rate, while also continuing to repair a low pressure recirculation system. Recirculation of suppression pool water should be implemented as soon as possible.

In the unlikely event that the RHR is not also restored by the time that the PCS has reduced primary pressure to below 50 psia, the normal switchover to RHR cool down could not be accomplished (see discussion which follows concerning RHR repair). Continued use of the main condenser below this pressure would result in turbine shaft gland seal leakage, loss of condenser vacuum, and eventual blowout of the condenser rupture disks. Cooling could continue in this mode but a pathway would exist between the primary system and the turbine building enclosure, as described previously for state 8.

If the operator has not restored the PCS to service at states 8 or 9, the other principal alternative for restoring containment heat removal is the repair of the RHR system. There are however some risks associated with the RHR

restoration if it occurs very late in the accident sequence. For example, with the suppression pool at elevated temperatures, i.e. in the range of 300°F for times greater than 20 hours, there may be some difficulty in the initiation of effective RHR pool cooling at this point. Specifically, as the suppression pool fluid is pumped by the RHR pumps through the RHR heat exchangers there may be some flashing of the steam (this is possible but unlikely if the RHR system is maintained at sufficiently high pressure). If flashing does occur it may come in contact with cool water present in the RHR piping or in the RHR heat exchangers. This contact could result in violent condensation and, in the extreme, damage to the RHR system. Pressures and temperatures within the RHR system may be extremely important in this operation of RHR restoration under elevated containment temperatures.

Successful restoration of PCS or RHR operation is indicated by state 10 on the OAET. At this state the operator must monitor the approach to cold shutdown conditions. At state 10a the operator has not been able to establish containment cooling. Although reactor coolant inventory is being maintained, the core is vulnerable to damage due to the potential for containment failure. In addition, the containment conditions could affect the ability to maintain inventory, depending upon the method being used to provide coolant to the core. If suction is being taken from the pool, these pumps could cavitate due to a loss of net positive suction head. Other components (instrumentation and control) could fail if their environmental qualification limits are exceeded. If core damage occurs, it will be at low pressure in state 10a.

Failure of the operator to maintain the vessel at low pressure is indicated by state 8a. At this state the operator did not or could not depressurize the vessel through the MSIVs, the drain valves, or the HPCI/RCIC turbines. With the MSIVs closed the reactor vessel has repressurized to above the shutoff head of the low pressure ECCS pumps. (For some spectrum of S_1 breaks, the leak size may be large enough to prevent repressurization. However, this is not postulated for the sequence under discussion.) Continued steaming through the break and perhaps the safety valves causes the vessel water level to drop. The operator must therefore maintain inventory by placing a high pressure system

into service. Due to the size of the break, only the HPCIS or feedwater systems are capable of adequately replenishing the lost fluid.* The HPCIS pumps have previously tripped due to high backpressure or high temperature in the pump areas. The feedwater system has also been tripped off due to one of a variety of reasons. It is unlikely that the operator will be able to restore one of these systems to operation at this time. However, he should continue his attempts to restore these systems, while also supplying whatever coolant possible with the CRD and RCIC pumps. If coolant inventory maintenance is unsuccessful, the core will eventually uncover and fuel damage will occur.

Despite its unlikely nature, it is postulated that the operator does provide sufficient high pressure coolant makeup to keep the core covered (state 9a). Actions taken from this point on are analogous to those taken in state 9. Failure to restore some form of containment cooling may eventually result in containment failure and core damage. Core damage at state 10c, if it occurs, will be at an elevated pressure.

4.3.2.3 Symptom Matrix

The symptoms exhibited by the plant at each of the states described for the S_1I sequence are tabulated in Table 4.14. The behavior of each of the key BWR parameters discussed in Section 3 is delineated in this table.

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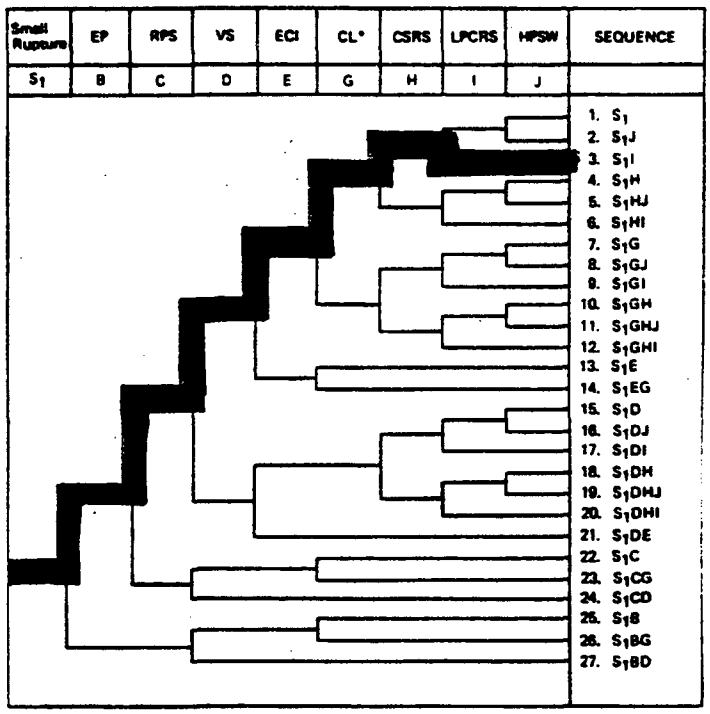
*As mentioned in Section 4.3, some breaks may be small enough for the RCIC or CRD pumps to maintain inventory. However, the large end of the S_1 break spectrum is discussed here.



*Containment Leakage less than 100%/day.

Figure 4.11 Reactor Safety Study Event Tree for BWR Small LOCA (S_1, approximately 2.5-8 inch diameter): S_1I Sequence Highlighted.

Figure 4.12 S_1I Sequence - Operator Action Event Tree

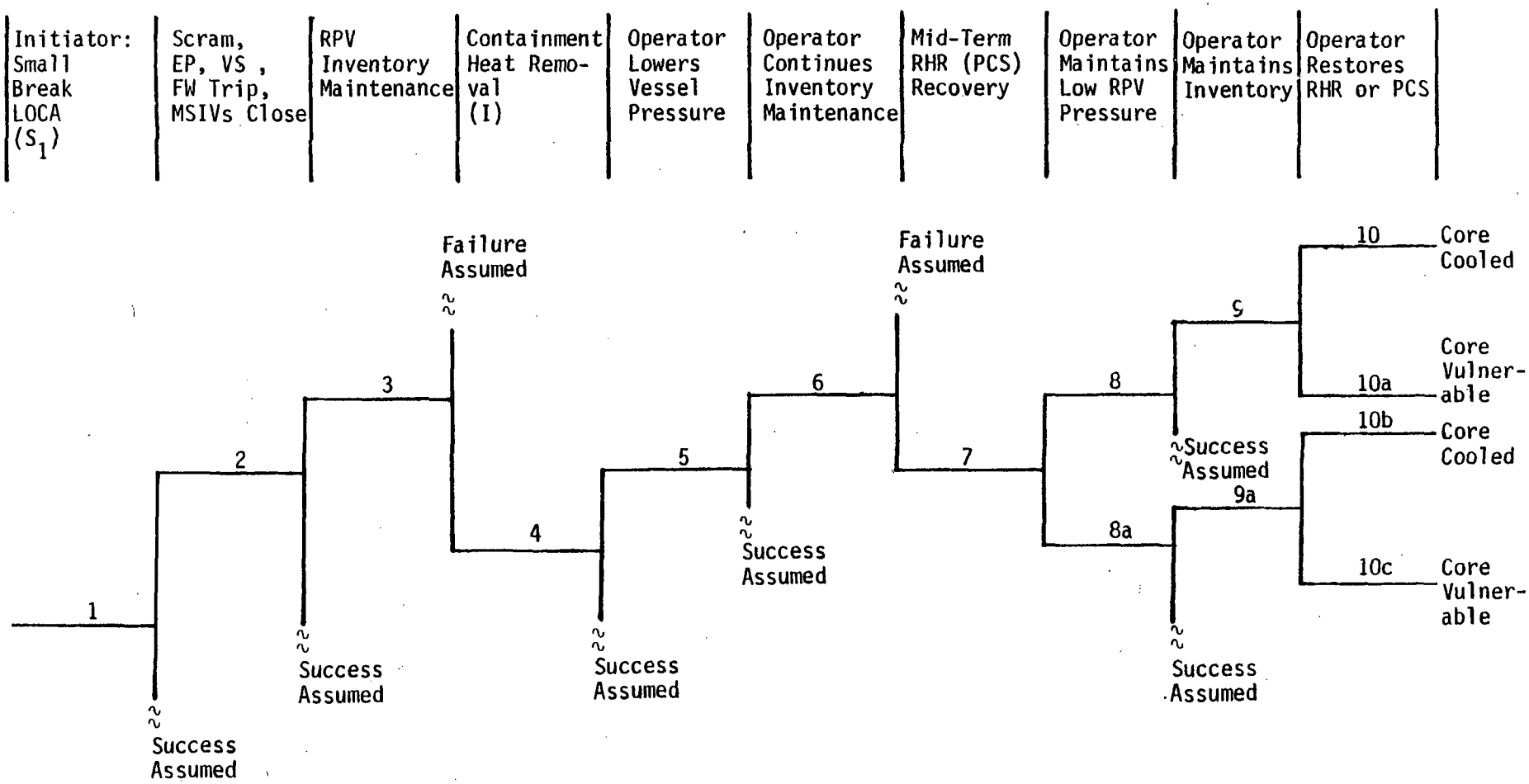


Table 4.14

Sequence: S_1I

BWR OAET Symptom Matrix Sheet 1

Symptoms 1-19

(See Section 4.2.1.3 for explanation of Symbols)

Symptoms	RPV Level				RPV Press.									RPV Temp		Re-actor Pwr.		Rod Pos.	Bo-ron
OAET State	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19
S_1I-1,2	0	0	0	0	0	1	0	1	0	1	0	1	1	0	0	1	0	1	0
S_1I-3	0	0	0	0	1	1	1	0	0	1	0	1	1	0	0	1	0	1	0
S_1I-4	0	0	0	0	1	1	1	0	0	1	0	1	1	0	0	1	0	1	0
S_1I-5	0	0	0	0	0	1	0	1	1	0	1	1	1	1	0	1	0	1	0
S_1I-6	0	0	0	0	0	1	0	1	1	0	1	1	1	0	0	1	0	1	0
S_1I-7	0	0	0	0	0	1	0	1	1	0	1	1	1	0	0	1	0	1	0
S_1I-8	0	0	0	0	0	1	0	1	1	0	1	1	1	0	0	1	0	1	0
S_1I-8a	0	1	0	0	1	1	1	0	0	1	0	1	1	0	0	1	0	1	0
S_1I-9	0	0	0	0	0	1	0	1	1	0	1	1	1	0	0	1	0	1	0
S_1I-9a	0	0	0	0	1	1	1	0	0	1	0	1	1	0	0	1	0	1	0
S_1I-10	0	0	0	0	0	1	0	1	1	0	1	1	1	0	0	1	0	1	0
S_1I-10a	0	1	1	0	0	1	0	1	1	0	1	1	1	0	0	1	0	1	0
S_1I-10b	0	0	0	0	1	1	1	0	0	1	0	1	1	0	0	1	0	1	0
S_1I-10c	0	1	1	0	1	1	1	0	0	1	0	1	1	0	0	1	0	1	0

Table 4.14 (continued)

Sequence: S_1I

BWR OAET Symptom Matrix Sheet 2

Symptoms 20-44

Symptoms	S.P. Level										S.C. Press					S.P. Temp					DW Prs	Drywell Temp.			
OAET State	20	21	22	23	24	25	26	27	28	29	30	31	32	33	34	35	36	37	38	39	40	41	42	43	44
S_1I-1,2	0	0	1	0	1	0	0	0	1	0	0	0	0	0	1	1	0	1	0	1	1	0	0	1	1
S_1I-3	1	0	1	0	1	1	0	0	1	0	0	0	0	0	1	1	1	1	0	1	1	0	0	1	1
S_1I-4	1	0	1	0	1	1	0	0	1	0	0	0	1	1	1	1	1	1	0	1	1	0	0	1	1
S_1I-5	0	0	1	0	1	1	0	0	1	0	0	0	1	1	0	1	1	1	0	1	1	0	0	1	1
S_1I-6	0	0	1	0	1	1	0	0	1	0	0	0	1	1	0	1	1	1	0	1	1	0	0	1	1
S_1I-7	0	0	1	0	1	1	0	0	1	0	0	0	1	1	0	1	1	1	1	1	1	0	1	1	1
S_1I-8	0	0	1	0	1	1	0	0	1	0	0	0	1	1	0	1	1	1	1	1	1	0	1	1	1
S_1I-8a	1	0	1	0	1	1	0	0	1	0	0	0	1	1	0	1	1	1	1	1	1	0	1	1	1
S_1I-9	0	1	1	1	1	1	0	0	1	0	0	0	1	1	0	1	1	1	1	1	1	0	1	1	1
S_1I-9a	1	1	1	1	1	1	0	0	1	0	0	0	1	1	0	1	1	1	1	1	1	0	1	1	1
S_1I-10	1	1	1	1	1	1	0	0	1	0	0	0	1	1	0	1	1	1	1	1	1	0	1	1	1
S_1I-10a	1	1	1	1	1	1	0	0	1	0	1	1	1	1	0	1	1	1	1	1	1	0	1	1	1
S_1I-10b	0	1	1	1	1	1	0	0	1	0	0	0	1	1	0	1	1	1	1	1	1	0	1	1	1
S_1I-10c	0	1	1	1	1	1	0	0	1	0	1	1	1	1	0	1	1	1	1	1	1	0	1	1	1

Table 4.14 (continued)

Sequence: S_1I

BWR OAET Symptom Matrix Sheet 3

Symptoms 45-65

Symptoms	PC Level	PC Temp	HPCI		RCIC		LPCI		CS		DG		RHR		ADS			MC		CRD	
OAET State	45	46	47	48	49	50	51	52	53	54	55	56	57	58	59	60	61	62	63	64	65
S_1I-1,2	0	0	1	0	1	0	1	1	1	1	1	0	0	1	0	0	1	1	1	1	1
S_1I-3	0	0	1	0	1	0	1	1	1	1	1	0	0	1	0	0	1	1	1	1	1
S_1I-4	0	0	1	1	1	1	1	1	1	1	1	0	0	1	0	0	1	1	1	1	1
S_1I-5	0	0	1	1	1	1	1	1	1	1	1	0	0	1	0	0	1	1	1	1	1
S_1I-6	0	0	1	1	1	1	1	1	1	1	1	0	0	1	0	0	1	1	1	1	1
S_1I-7	0	0	1	0	1	1	1	1	1	1	1	0	0	1	0	0	1	1	1	1	1
S_1I-8	0	0	1	1	1	1	1	1	1	1	1	0	0	1	0	0	1	1	1	1	1
S_1I-8a	0	0	1	1	1	1	1	1	1	1	1	0	0	1	0	0	1	1	1	1	1
S_1I-9	1	0	1	1	1	1	1	1	1	1	1	0	0	1	0	0	1	1	1	1	1
S_1I-9a	1	0	1	1	1	1	1	1	1	1	1	0	0	1	0	0	1	1	1	1	1
S_1I-10	0	0	1	1	1	1	1	1	1	1	1	0	1	1	0	0	1	1	1	1	1
S_1I-10a	0	0	1	1	1	1	1	1	1	1	1	0	0	1	0	0	1	0	1	1	1
S_1I-10b	0	0	1	1	1	1	1	1	1	1	1	0	1	1	0	0	1	1	1	1	1
S_1I-10c	0	0	1	1	1	1	1	1	1	1	1	0	0	1	0	0	1	0	1	1	1

Table 4.14 (continued)

Sequence: S_1I

BWR OAET Symptom Matrix Sheet 4

Symptoms 66-79

Symptoms	SLC		RWCU		Alternate Inj.		C/F		Recirc. Pumps		MSIV		RHR Interlock	
OAET State	66	67	68	69	70	71	72	73	74	75	76	77	78	79
S_1I-1,2	1	0	1	0	1	0	1	1	1	0	1	1	0	1
S_1I-3	1	0	1	0	1	0	1	1	1	0	0	1	0	1
S_1I-4	1	0	1	0	1	0	1	1	1	0	0	1	0	1
S_1I-5	1	0	1	0	1	0	1	1	1	0	0	1	0	1
S_1I-6	1	0	1	0	1	0	1	1	1	0	0	1	0	1
S_1I-7	1	0	1	0	1	0	1	1	1	0	0	1	0	1
S_1I-8	1	0	1	0	1	0	1	1	1	0	1	1	0	1
S_1I-8a	1	0	1	0	1	0	1	1	1	0	0	1	0	1
S_1I-9	1	0	1	0	1	0	1	1	1	0	1	1	0	1
S_1I-9a	1	0	1	0	1	0	1	1	1	0	0	1	0	1
S_1I-10	1	0	1	0	1	0	1	1	1	0	1	1	1	1
S_1I-10a	1	0	1	0	1	0	0	1	1	0	1	1	0	1
S_1I-10b	1	0	1	0	1	0	1	1	1	0	1	1	1	1
S_1I-10c	1	0	1	0	1	0	0	1	1	0	1	1	0	1

4.3.3 Transient Induced Stuck Open Safety/Relief Valve

In this section the plant response and operator actions associated with a stuck open safety/relief valve (SORV) following a transient initiator are discussed. The discussion which follows will focus primarily upon the differences between a transient-induced SORV and the S_1 LOCAs discussed in Sections 4.3.1 and 4.3.2. Because of the similarities in sequence progression, the Operator Action Event Trees constructed for the S_1I and S_1E sequences are adequate for this discussion, and no OAET is included specifically for the SORV event. For similar reasons, symptom tabulation is not presented explicitly for this event.

4.3.3.1 Sequence Description and Operator Response

Sections 4.3.1 and 4.3.2 presented sequence descriptions for sequences initiated by small breaks in the primary system. The plant response and operator actions following a transient-induced stuck open safety/relief valve are in many ways quite similar to those discussed in the previously mentioned sections. This section will highlight the differences, if any, as the sequence proceeds.

The stuck-open safety/relief valves discussed in this section are roughly equivalent in plant response to a small steam line break inside containment ($\sim$ 0.1 ft^2 break area). However, because steam discharge through the SRVs is routed directly to the suppression pool, no high drywell pressure scram is generated for SORV events. Therefore, use of the ADS, if necessary, must be manually initiated.

Another possible difference between a transient-induced SORV event and a sequence initiated by a small break of the S_1 category is the availability of feedwater. The S_1E and S_1I sequences described previously postulated the availability of feedwater during the initial stages of the sequence. A transient event may be initiated in such a way that feedwater is unavailable from the beginning of the sequence. In fact, a loss of feedwater could be the initiating event. Loss of feedwater eliminates one of the potential high pressure coolant systems which could be used to maintain level. For purposes of discussion, and because of the demands it places upon the ECCS, the initiator is

assumed here to be a loss of all feedwater flow. Following the loss of feedwater, the water level in the vessel drops until it reaches the low-low level setpoint (Level 2), at which point the MSIVs close. The system pressure then rises to the setpoints for the SRVs, and they open. At this point it is postulated that one of the valves sticks in the open position.

Table 4.15 is a compilation of key events which may occur following a loss of feedwater event which results in a stuck-open safety valve. The timing of the events is based upon Reference 6, and is presented for both the case in which all high pressure safety systems are assumed to be operable (the most likely case), and the case in which the HPCI fails but the RCIC is operating. As is seen in this table, some operator actions have been included in the sequence of events. Analysis performed by GE (see reference 6) indicates that no operator action is necessary to prevent core uncovery if the HPCI and/or RCIC operate following a loss of feedwater flow and a stuck-open SRV. Only in the case where all high pressure systems fail is operator action necessary to prevent core uncovery (see Section 4.2.1, TQUV sequence, and Section 4.3.1, S_1E sequence, for more discussion on operator response to high pressure ECCS failure).

The initial response to a loss of feedwater event is described in detail in Section 4.2.1. To summarize, the loss of feedwater results in a reduction in reactor vessel water level. A low level (level 3) scram signal is generated, and the control rods insert to terminate full power operation. Following the scram there is a sharp drop in water level due to the collapse of voids coupled with the loss of feedwater makeup. The water level therefore drops rapidly to the low-low water level setpoint (level 2).

When the water level reaches the level 2 setpoint, the HPCI and RCIC pumps are signaled to start. At the same time, the MSIVs close and the recirculation pumps trip. Closing the MSIVs results in an immediate increase in reactor system pressure. The pressure increases to the opening setpoints of the SRVs ($\sim$ 1100 psia) at which time they open to relieve primary pressure. When the pressure is reduced, the valves should reseat, re-opening only if the vessel pressure again increases. It is at this point that one SRV is postulated to remain open, thus providing a continuous path for inventory escape from the vessel.

The operator will be aware of the failure to close by monitoring the reactor pressure. A continuous rapid decline in pressure below the point where the valves should reclose will be one indication of the presence of a leak path. An increase in the suppression pool temperature which is not accompanied by an increasing drywell pressure should direct the operator's attention to the safety valves (as opposed to a pipe break). Once the operator becomes aware that a valve is stuck open he should immediately attempt to close it by halting air to the pilot valve. This will cause the air pressure in the solenoid valve to bleed off, and the valve should then reseat due to its own weight. If the valve reseats, the operator could proceed with actions appropriate for establishing cold, shutdown conditions.

If the SRV remains open, a steam/water mixture will continue to escape through it and into the suppression pool. Unless a source of makeup is provided to the vessel, the core will eventually uncover as the decay heat continues to produce steam. The HPCI and RCIC systems should supply water until the reactor depressurizes sufficiently through the open SRV to allow the low pressure ECC systems to inject into the vessel. Coolant inventory can then be maintained with these systems until the shutdown cooling mode of the RHR can be placed in service. Failure to remove decay heat is treated in Sections 4.2.2 and 4.3.2 (TW and S_1I sequences).

If the high pressure systems do not operate, the vessel water level will continue to fall. For the SORVs being considered here, depressurization through the SORV will not be sufficient by itself to decrease reactor pressure below the shutoff head of the low pressure ECCS pumps before the core is uncovered. Therefore a more rapid depressurization, such as through the ADS valves, would be necessary. This depressurization must be accomplished manually, as the high drywell pressure signal necessary for automatic depressurization would not be present. Failure of the high pressure systems is treated in Sections 4.2.1 and 4.3.1 for the TQUV and S_1E sequences.

Following depressurization and inventory maintenance, the operator's actions will be very similar to those described in Sections 4.3.1 and 4.3.2. (However, some plant systems may not be available for other transient initiating events, i.e., loss of off-site power.) The operator's concerns throughout the

sequence are similar to those for the S_1E and S_1I sequences: maintain inventory, establish decay heat removal, and isolate the leak, if possible. Inventory makeup with the feedwater system may be difficult with loss of feedwater as the initiating event. Extensive repairs may be necessary, and therefore this system may not be available at any time in the sequence.

More discussion on operator actions appropriate for this sequence and in response to postulated failures are contained in previous sections.

Table 4.15

Key Sequence of Events in "SORV" Sequence

Time (sec.)		Event
HPCI & RCIC	RCIC only	
0.0	0.0	Trip of all feedwater pumps initiated.
5.0	5.0	Feedwater flow decays to zero.
15.0	15.0	Water level reaches low level scram (Level 3). Reactor scram is initiated. All primary system isolation valves except the main steam line isolation valves (e.g., RHR shutdown isolation valves, RWCU isolation valves, and containment) are initiated to close. Automatic depressurization permissive.
34.0	34.0	Wide range sensed water level reaches low low water level (Level 2). Recirculation pumps are tripped. The main steam isolation valves are closed. RCIC and HPCI systems are initiated.
39.0	39.0	MSIV's fully closed.
64.0	64.0	HPCI and/or RCIC flow starts to enter the vessel.
93.0	88.0	Group 1 safety/relief valves start to open.
97.0	92.0	All Group 1 safety relief valves except one close on closing setpoint. The reactor continues to depressurize through the stuck open relief valve. Reactor level decreases slowly.
380.0	NA	Water level increases and approaches normal water level. The operator takes manual control of HPCI and RCIC to maintain normal water level.
1800.	NA	Operator initiates steam condensing mode if available. Operator can also choose to de-isolate and use the main condenser as a sink. When the reactor pressure reaches 150 psia the operator can switch RHR to shutdown cooling mode.
NA	1800	RCIC flow equals SORV flow. In-shroud level reaches a plateau (4.3 ft. above TAF). Reactor pressure is approximately 340 psia and decreasing slowly.
NA	1900	Operator opens a relief valve to speed up the depressurization.
NA	2480	LPCS flow enters the reactor. In-shroud level bottoms out (2.4 ft. above TAF), reactor is quickly reflooded.

Section 5

EXAMINATION OF BWR EPGs USING THE RESULTS OF THE OPERATOR ACTION EVENT TREE ANALYSIS

Section 4 contains a detailed description of plant and operator responses during a variety of postulated BWR accident sequences. Using the operator action event trees developed for these sequences as guidelines, the plant behavior at each of the various plant states has been tabulated as a function of the key BWR parameters listed in Table 3.1. The results of this tabulation process have been presented in tabular form in each of the subsections of Section 4.

This section presents an examination of these tabulated symptoms when compared to the action sets developed from the EPGs and previously described in Section 3. The symptoms present at each state of an OAET were compared to the symptom set/action set matrix (see Tables 3.2 and 3.3) and the EPG steps which the operator could be instructed to perform at each of the OAET states were identified. The action sets indicated by the EPGs at each of the OAET states were then examined to answer three questions:

1) Are the EPG steps included in the indicated action sets compatible with the actions indicated by the OAET?

2) If more than one set is indicated for a plant state, are the indicated action sets compatible with each other?

3) Are the action sets complete, i.e., are all significant actions in the OAETs addressed by the indicated action sets?

The following sections examine the guidance offered by the EPGs for each of the analyzed plant states. Tables 5.1 through 5.5 present a summary of the action sets which are indicated at each OAET state, and the accompanying text discusses the results of the examination with respect to each of the three questions listed above for the accident sequences selected for analysis.

5.1 Examination of Guidance Following Failure to Maintain Inventory After a Small-Break* Loss of Coolant Accident

In this section, an examination is made of the guidance provided by the BWR EPGs and their potential application to plant-specific emergency procedures. The examination is for accident conditions involving a small break loss of coolant accident followed by a failure to maintain inventory. A detailed discussion of each of the plant states discussed below is presented in Section 4.3.1 and should be referenced for a more complete understanding of the plant response and operator actions at these states.

State S_1E -1,2

At this state a small break has occurred in the primary coolant system (PCS). The feedwater system is maintaining inventory, but a high drywell pressure scram has been initiated. Because the break is inside containment, the break cannot be isolated by closure of the containment isolation valves.

The operator's duties at this state include verification of scram and all associated automatic responses to the high drywell pressure trip, including actuation of the HPCI. The operator should attempt to isolate the break. For example, if the break were outside of containment the operator could close the MSIVs. The operator should also monitor containment conditions and initiate drywell or suppression chamber cooling as needed. Most importantly, the operator should maintain water level in the vessel by controlling the feedwater and HPCI systems (for more detail on operator actions, see Section 4.3.1)

As seen in Table 5.1, several actions sets are indicated by the symptoms which may be present at this state. Sets A1 and A2 deal with initiating or verifying the scram, isolation of containment, and ECCS initiation. The operator is also instructed by these sets to minimize SRV cycling (if it is occurring), and to maximize heat removal through the turbine

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* The definition of an S_1 (small-break) initiator is the same as the Reactor Safety Study,[10] that is, the RCIC system does not have sufficient capacity to maintain adequate inventory.

bypass valves to the main condenser. Set A6 may also be indicated by the symptoms of these states. Its instructions are similar to those of A1 with the addition of steps to line up and start pumps in two or more low pressure injection trains (condensate, core spray, LPCI). This action is correct in that it provides a backup to the automatic initiation of low-pressure systems by the high drywell pressure signal and will also provide a permissive signal for ADS operation. Sets A1 and A6 also instruct the operator to prevent automatic ADS operation by resetting the timer (if it has started), but only if water level can be maintained above the top of the active fuel. It should be noted, however, that the contingency procedures for the EPGs also direct the operator to prevent automatic initiation of the ADS even if the water level cannot be maintained above the top of the active fuel. But for this latter case, the operator is then directed to depressurize the plant manually using the ADS values.

Other sets are possibly indicated by the symptoms at this state. Sets B1, B2, B3, B4, B5, and B10 instruct the operator to check for stuck-open relief valves and close them, if possible, since this may be the source of the primary system leakage. He is also instructed to initiate suppression chamber and drywell sprays, and to monitor and maintain suppression pool parameters (level, temperature) within limits.

The comparison of the action sets with the discussion of the OAET for this sequence identified a few areas where additional clarification in the development of plant-specific procedures and training programs may be necessary to eliminate ambiguity. One such area concerns the operation of the feedwater system during the early stages of a small break accident sequence. Although this sequence postulates HPCI system failure, the HPCI system is designed to initiate operation automatically upon receipt of a high drywell pressure signal. If HPCI does initiate, this additional source of injection into the vessel will cause a rapid increase in the vessel water level, and the high level trip setpoint may be quickly reached. This would result in the simultaneous trip of both the HPCI and the feedwater system. The feedwater system would not restart without operator actions. Rapid operator intervention to control HPCI operation could prevent this. These instructions would be applicable to any sequence in which HPCI operation is initiated and the feedwater pumps are still in operation (such as the S_1I and TW sequences). In the process of producing plant specific procedures and training programs from these guidelines, consideration should be

given to providing more explicit instructions concerning control of feedwater during the initial stages of accident sequences.

An additional area where more detailed guidance may be appropriate is the initiation (and subsequent termination) of the drywell cooling fans. The drywell cooler fans at some plants trip on high drywell pressure and/or low reactor water level, and the operator may need to bypass some interlocks to initiate operation of the fans. Caution statements to this effect should be considered when producing plant-specific procedures.

State S_1E-3

At this state it is postulated that the feedwater system has tripped but the HPCI pump and RCIC have failed to deliver adequate coolant to the vessel to restore and maintain inventory. The vessel water level is continuing to drop as coolant is discharged through the break and possibly the SRVs.

The operator's immediate response to this situation should be to attempt restoration of the feedwater and HPCI/RCIC systems. If the operator can restore one of these systems to service rapidly he will be able to maintain adequate reactor water level and allow the reactor to continue its gradual cooldown and depressurization.

Several of the action sets that may be indicated at this state are those previously indicated at states S_1E-1 and 2: namely A1, A2, B1, B2, B3, B4, B5, and B10. Two additional sets are indicated at state S_1E-3: A5 and B9. Like set A1, A5 instructs the operator to initiate ECCS operation and to restore and maintain vessel level with the condensate/feedwater, HPCI, or other systems. Set B9 deals with maintaining the suppression pool level and pressure within limits. This set also instructs the operator to terminate injection to the RPV from sources outside of the primary containment if adequate core cooling is assured. (Note that for this postulated OAET state adequate core cooling is still not assured. Therefore the conditions necessary for set B9 are not present and the operator should not terminate injection.)

The EPG steps included in the action sets for this state do not provide the operator with timely, explicit, instructions to restart the HPCI/RCIC or feedwater pumps. Although the instructions included in the action sets for this state indicate that the operator should restore and maintain level with the condensate/feedwater, RCIC, HPCI, or other systems (Step RC/L-2), they do not say "restart HPCI and RCIC" (as indicated at Step C1-7). This timely advice may be helpful in restoring level more quickly than might otherwise occur. In the process of developing plant specific procedures and training materials to aid the operator in his response to the situation, consideration should be given to providing more detail concerning potential failure modes and restoration procedures for the HPCI/RCIC system. For example, if and when the HPCI transfers from the CST to the suppression pool, there may be circumstances which would allow the operator to switch back to the CST, e.g., high suppression pool temperature.

At this state, a high drywell pressure exists, the water level is dropping in the vessel, and the HPCI/RCIC and feedwater systems (all high pressure systems) are unavailable. The ADS is designed to depressurize the vessel and allow low pressure injection to occur following such occurrences. However, instead of allowing this to occur, the EPGs instruct the operator to prevent automatic operation of the ADS. Specifically, action set A5 includes steps within Contingency #1, Level Restoration. Contingency #1 cautions the operator while performing steps C1-4 through C1-8 that "if ... RPV water level drops below [ADS initiation setpoint], prevent automatic initiation of ADS". Step C1-7, however, says "if no CRD pump is operating but at least 2 injection subsystems are lined up for injection with pumps running, EMERGENCY RPV DEPRESSURIZATION IS REQUIRED." Step C2-1.1, in Contingency #2 (entitled "Emergency RPV Depressurization"), states "If suppression pool water level is above [elevation of top of SRV discharge device], open all ADS valves." If these manual depressurizations are intended to produce a more controlled depressurization process than automatic ADS, more explicit discussion should be included concerning how to accomplish this controlled depressurization in the plant-specific procedures and be addressed in the associated operator training programs.

State S_1E-4

At this state the operator has successfully restored a high-pressure coolant injection system to operation and the vessel level is increasing or recovered.

The appropriate operator actions at this state are to maintain water level, depressurize and cooldown the reactor, and proceed to cold shutdown. The action sets indicated at this state are the same as indicated for states S_1E-1, 2, and 3, with one addition. Action sets A1, A2, and A5 are indicated, and instruct the operator to maintain RPV pressure and level. Sets B1, B2, B3, B4, B5, B9, and B10 also are indicated, and basically instruct the operator to isolate any SORVs and initiate suppression chamber and drywell spray operation. The one additional action set indicated at this state is set X4. This set instructs the operator to line up and start low pressure injection systems, and to depressurize the reactor through the main condenser, RHR or other means (ADS valves, SRVs, HPCI, etc).

No problems or potential ambiguities appear to be present among the action sets present at this state. However, elaboration may be required on the continued use of the drywell and suppression chamber sprays. At state S_1E-4 the relative leak size has diminished due to the lowered system pressure, and the sprays may not be needed. Continued use of the sprays could create problems for equipment which is located in the spaces being sprayed. By providing guidance as to when the sprays may be turned off, equipment malfunctions might be minimized. Consideration should be given to providing this guidance in the plant specific procedures.

State S_1E-4a

At this state the operator has been unable to restore a source of high pressure coolant makeup, and the vessel level is continuing to drop.

The operator's actions at this time should be to depressurize the reactor and initiate or confirm operation of the low pressure injection systems. Automatic depressurization through the ADS valves may occur when all

necessary permissive signals (high drywell pressure, low vessel level) are generated unless the operator intervenes.

The action sets indicated at this state are the same as at state S_1E-4, with one addition. The additional set, X1, instructs the operator to start low pressure injection pumps, attempt to restart the HPCI/ RCIC pumps, and depressurize the vessel using the ADS valves augmented by other SRVs and other systems.

Most of these indicated actions are consistent with those indicated by the OAET for this sequence. However, as for state S_1E-3, more detailed guidance may be appropriate in the plant-specific procedures regarding manual versus automatic rapid depressurization. Contingency #1 cautions the operator while performing steps C1-4 through C1-8 (action set X1) that "if ...RPV water level drops below [ADS initiation setpoint], prevent automatic initiation of ADS". Step C1-7, however, says "if no CRD pump is operating but at least 2 injection subsystems are lined up for injection with pumps running, EMERGENCY RPV DEPRESSURIZATION IS REQUIRED." Step C2-1.1, in contingency #2 (entitled "Emergency RPV Depressurization", which may also be indicated at state S_1E-4a by set X1), states "If suppression pool water level is above [elevation of top of SRV discharge device], open all ADS valves." The rationale for instructing the operator to prevent automatic ADS while at the same time calling for emergency depressurization by opening all ADS valves should be made clear by providing the procedural guidance and training necessary to obtain the benefits of manual versus automatic depressurization.

State S_1E -5

At this state the vessel has automatically depressurized. Vessel water level is low, and makeup with a low pressure system is required.

The operator should confirm or initiate operation of the low pressure injection systems. These systems may be in operation since adequate discharge pressure from low pressure pumps is a requirement for automatic initiation of the ADS.

With one addition, the EPG action sets indicated at this state are the same as indicated for state S_1E-4a. Discussion for that state should be

referenced for details. The additional set indicated for state S_1E-5 is set X3.This set of actions instructs the operator to line up and inject with low pressure ECCS or alternate injection systems, eventually taking suction from the suppression pool and terminating the injection of water from outside sources.

At this point the operator may not know the vessel level (because of the rapid depressurization). If this is the case, the EPGs instruct him to consult contingency #6, RPV Flooding. Those steps, plus the actions indicated by the EPGs, are consistent with what the operator should be doing under the conditions which exist at this state.

State S_1E-5a

At this state the water level has continued to drop in the vessel, but automatic depressurization has failed or has been prevented by the operator. Depressurization is necessary to begin coolant makeup with the low pressure systems.

The operator at this time should manually depressurize the reactor through the ADS or safety/relief valves, or though any of the other available depressurization paths (main steam line, HPCI/RCIC turbines, etc.). The operator should then ensure that low pressure injection commences.

This state is essentially a continuation of state S_1E-4a. The entire set of actions indicated for that state are also indicated for this one, and there are no additional action sets. The comments generated for state S_1E-4a are also applicable for state S_1E-5a.

State S_1E-6

At this state the operator has successfully depressurized the vessel to below the shutoff head of the low pressure injection systems. Coolant inventory makeup is now required.

The operator must confirm or initiate operation of some source of coolant makeup, and monitor the RPV water level. His actions at this point are zanalogous to those at state S_1E-5.

The action sets indicated for this state are also the same as those present at state S_1E-5. See discussion of that state for more detail.

State S_1E-6a

At this state the operator has been unable to depressurize the vessel and initiate low pressure coolant injection. The high pressure systems are still out of service, and the water level in the vessel is therefore continuing to decline. Without sufficient makeup, the core will uncover.

The operator's actions at this time should be focused on continuing efforts to restore some source of inventory makeup to the vessel. Depressurization attempts should be continued.

Because this state is a continuation of state S_1E-5a, the actions* indicated by the symptoms at this state are the same as at state S_1E-5a. Comments for that state are also applicable here.

State S_1E-7

At this state the reactor vessel has depressurized but the low pressure ECI systems are no longer in operation. The vessel water level is not being maintained.

The operator must quickly restore the LPCI system to service, or initiate operation of another low pressure source such as the condensate system. The operator must then verify recovery of the vessel water level.

The action sets indicated by the symptoms present at this state are the same as those present for state S_1E-5. These sets include instructions to restore level with the low pressure systems (including the condensate system.)

The EPG steps included in the action sets for this state are correct, and include adequate instruction to attempt coolant injection with other low pressure systems besides the LPCI.

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* Actions necessary to respond to core damage have not been addressed either in the OAET discussion or in the EPG action steps.

State S_1E-7a

At this state the vessel is depressurized and low pressure injection systems are in operation. The vessel water level is being restored.

The appropriate operator actions at this state are primarily to continue plant cooldown, initiating the RHR shutdown cooling mode when possible.

The indicated action sets are the same as those indicated at states S_1E-5 and S_1E-6. These action sets deal with maintaining vessel level and decay heat removal, and are appropriate for this state.

Although the actions indicated are correct, they do not provide clear guidance about when (or if) systems should be turned off, especially the drywell or suppression chamber sprays. Consideration sould be given to providing more explicit guidance in the plant-specific procedures. See discussion of sprays in State S_1E-4.

State S_1E-8

The operator has restored some low pressure coolant makeup systems to operation and the vessel water level is recovered. The operator actions applicable for this state are analogous to those discussed for state S_1E-7a. The action sets indicated for this state are also the same as for that state.

Plant-specific procedures or training material should describe the reactor water level instrumentation response under these abnormal conditions so the operator will be aware of potential variations in indicated readings.

State S_1E-8a

At this state the system is at low pressure, but there are no injection systems in operation and the vessel level is continuing to drop. This state is a continuation of state S_1E-7. The discussion of that state should be referenced for more detail.

Table 5.1

Indicated Action Sets for the S_1E Sequence

OAET State	Indicated Action Set(s)*
S_1E-1	A1, A2, A6 B1, B2, B3, B4, B5, B10
S_1E-2	A1, A2 B1, B2, B3, B4, B5, B10
S_1E-3	A1, A2, A5 B1, B2, B3, B4, B5, B9, B10
S_1E-4	A1, A2, A5 B1, B2, B3, B4, B5, B9, B10 X4
S_1E-4a	A1, A2, A5 B1, B2, B3, B4, B5, B9, B10 X1, X4
S_1E-5	A1, A2, A5 B1, B2, B3, B4, B5, B9, B10 X1, X3, X4
S_1E-5a	A1, A2, A5 B1, B2, B3, B4, B5, B9, B10 X1, X4
S_1E-6	A1, A2, A5 B1, B2, B3, B4, B5, B9, B10 X1, X3, X4

*See Tables 3.2 and 3.3 for EPG Steps included in these Sets

Table 5.1 (continued)
Indicated Action Sets for the S_1E Sequence

OAET State	Indicated Action Set(s)*
S_1E-6a	A1, A2, A5 B1, B2, B3, B4, B5, B9, B10 X1, X4
S_1E-7	A1, A2, A5 B1, B2, B3, B4, B5, B9, B10 X1, X3, X4
S_1E-7a	A1, A2, A5 B1, B2, B3, B4, B5, B9, B10 X1, X3, X4
S_1E-8	A1, A2, A5 B1, B2, B3, B4, B5, B9, B10 X1, X3, X4
S_1E-8a	A1, A2, A5 B1, B2, B3, B4, B5, B9, B10 X3, X4

*See Tables 3.2 and 3.3 for EPG Steps included in these Sets

5.2 Examination of Guidance Following a Small-Break LOCA with Failure to Remove Decay Heat from the Containment

Section 4.3.2 contains the OAET and detailed discussion for this sequence. The symptom matrix for the states in the OAET for this sequence are also included there. Table 5.2 contains a summary of the action sets which may be indicated by the symptoms which exist during the progression of this sequence.

States S_1I-1,2

At state S_1I-1 a small break LOCA has occurred in the recirculation line inside containment. The drywell pressure is increasing due to the mass and energy discharged through the break. The feedwater controller is increasing feedwater flow to the vessel, compensating for the loss of coolant through the break. The level in the vessel is therefore relatively stable. At state S_1I-2, however, the feedwater system has tripped due to either low vessel pressure or high vessel water level. The MSIVs are assumed in this analysis to have also closed due to the low reactor pressure setpoint, or due to a low vessel water level after the feedwater system trips.

The operator must ensure that the HPCI system is in operation and must maintain reactor vessel water level with this and other high pressure systems (e.g., RCIC, CRD). Due to the postulated size of the leak, the CRD and RCIC systems cannot maintain level by themselves.

The action sets which may be indicated by the symptoms at this state include A1, A2, A5, A6, B1, B2, B3, B4, B5, and B10. These sets are the same as or very similar to those described previously for states S_1E-1,2. Sets A1, A2, A5, and A6 include steps to scram the reactor, confirm or initiate ECCS operation, isolate the containment, and restore and maintain RPV level with the feedwater or HPCI systems. Gradual RPV depressurization is also included as an appropriate action in these sets. B1, B2, B3, B4, B5, and B10 instruct the operator to close any stuck open safety/relief valves (SORVs), operate drywell cooling and sprays, and initiate suppression chamber spray operation.

The actions included in these steps appear to be the proper response to this initiating event. However, the high reactor water level trip of HPCI and feedwater (level 8) or low pressure MSIV closure setpoint trip (approximately 850 psi) may be avoided if the operator is aware of this possibility and acts quickly to assume manual control. See Section 5.1, state S_1E-1, for more discussion.

State S_1I-3

At this state the operator has been successful in maintaining the reactor vessel water level with the HPCI. The suppression pool level and temperature are increasing due to the continuing discharge of energy into the pool from the break or through the safety valves.

One of the operator's primary concerns at this state is to initiate containment cooling. This entails initiation of the RHR system to circulate water from the suppression pool through the RHR heat exchangers and provide cooling to the secondary side of the RHR heat exchangers.

With the exception of action set A5, all of the sets indicated for state S_1I-1 are also indicated here (A5 contains instructions also included in A6, which is indicated). Set A6 includes steps to initiate RHR operation and to proceed to cold shutdown. In addition, set B9, which directs the operator to maintain the suppression pool level within limits (among other things) is also indicated for this state.

The steps included in the action sets are consistent with those indicated by the OAET.

State S_1I-4

At this state the operator has been unable to establish suppression pool cooling. The suppression chamber pressure is rising, as is the suppression pool temperature.

The operator's immediate response should be to initiate or continue a controlled depressurization of the reactor vessel and to maintain inventory

with the available low pressure systems. Attempts to repair and restore operation of the RHR should also commence immediately. Decay heat removal through the main condenser should also be considered at this point.

All of the action sets indicated for state S_1I-3 are also indicated for this state. Additionally, sets B6 and X6 are indicated for this state. Set B6 includes steps to shut down the recirculation pumps and the drywell cooling fans. The drywell sprays should be turned on if not already in operation. Set X6 instructions include steps to depressurize the RPV with the ADS and other systems.

The EPG steps included in these action sets are correct. However, there are no explicit instructions to "repair" or "restore" the RHR. This information should be considered for inclusion in the plant-specific procedures and associated training programs.

Another area where the plant-specific procedures could beneficially provide more detailed guidance concerns the use of the main condenser as a heat sink if the RHR system cannot be placed in service. Using the main condenser would require that the operator open the MSIVs. Use of the main condenser as a heat sink would satisfy the decay heat removal requirement, while at the same time halting or minimizing the addition of energy to the suppression pool, thus bypassing the need for suppression pool cooling. However, there is only one location within the EPGs which even mentions re-opening the MSIVs to establish the main condenser as a heat sink, and that is referenced only if boron injection is required. At no other time do the EGPs explicitly instruct the operator to use the main condenser, not even in the contingency dealing with alternate shutdown cooling (Contingency #6). Since the balance of plant systems and operation is highly stylized and plant-specific, the individual plant-specific procedures and training programs should provide guidance in these areas, especially for the situations similar to those described for State S_1I-4. However, without successful restoration of the condenser vacuum, seals in the main condenser may fail if the MSIVs are open, thus creating a direct path to the environment. Therefore the operator may not wish to reopen the MSIVs without having full use of the main condenser, and yet this may be his only alternative for depressurization. Thus the operator requires guidance at this state to aid in establishing the source of heat removal.

State S_1I-5

The reactor has been successfully depressurized to below the shutoff head of the low pressure systems. However, the unavailability of suppression pool cooling has not been corrected.

The operator must confirm or initiate operation of the low pressure coolant makeup systems, and then maintain adequate inventory in the vessel. Because the failure of suppression cooling may be due to a failure of the LPCI pumps, the operator may have to use the condensate system, or some other low pressure system, to provide makeup. Attempts to remove decay heat must also be continued.

The action sets which may be indicated by the symptoms of this state are those that are also indicated in states S_1I-3 and S_1I-4. Comments for those states are also applicable here. Additionally, a failure of the LPCI pumps (if this is the cause of RHR failure) could also disable the drywell sprays without operator actions to restore spray operation from outside of the containment. This could have long-term effects on the containment conditions.

The plant-specific procedures and training should also give guidance to the operator for postulated degraded states in which containment sprays are unavailable due to LPCI pump unavailability. Unique plant-specific system configurations may allow containment spray using other systems such as service water and/or fire pumps. Such off-normal configurations should be indicated as feasible under certain degraded conditions. These conditions should also be defined if possible.

State S_1I-6

At this state the operator has been successful in maintaining inventory in the vessel with one or more of the available low pressure systems. The suppression chamber temperature and pressure are continuing to rise. The operator must continue his attempts to recover the ability to remove decay heat from containment. This can be accomplished by restoring either the main condenser or RHR to service as a heat sink.

The indicated actions for this state are the same as those for state S_1I-5. The actions focus on maintaining inventory with a low pressure system, and initiating RHR operation.

With the exception of the comments discussed for states S_1I-4 and S_1I-5, there do not appear to be any discrepancies or contradictions among the various EPG action steps indicated for this state.

State S_1I-7

At this state the operator has not been able to restore the main condenser or RHR to service, and therefore containment decay heat removal is not available. The suppression pool temperature is continuing to rise, as is the suppression chamber pressure.

Due to high containment pressure, temperature, or humidity resulting from the failure to remove decay heat from containment, the ability to maintain the RPV depressurizated via relief valve operation may be impaired. The operator must take action to maintain the RPV at a low pressure. If the operator does not, inventory makeup must be provided by high pressure systems whose availability is questionable at this time due to high backpressure on turbine pump systems or high room temperature trips (see Section 4.3.2 for details). To depressurize, one action the operator may take is to open the MSIVs, or he may release pressure through the HPCI/RCIC turbines directly to the suppression pool.

As mentioned for state S_1I-4, guidance is necessary in the EPGs to aid the operator in depressurization and heat removal. Action set X6, step C2-1, instructs the operator to depressurize the RPV using the ADS or other safety/relief valves or other systems such as the main condenser or HPCI/RCIC. The operator is told to use these systems in the order which will minimize the release to the environment. At state S_1I-7, which may occur at times more than one day following shutdown, the suppression chamber and drywell pressures could be increasing, little decay heat removal would be occurring (there is some conduction through the containment walls), and there is a possibility of containment structural failure due to overpressurization. The operator should establish some form of heat removal and RPV depressurization. The most likely

pathway is through the MSIVs or drain valves and through the main condenser. This will remove heat and depressurize the vessel at the same time. However, without successful restoration of the condenser vacuum, the seals in the main condenser may fail if the MSIVs are open, thus creating a direct path to the environment. Therefore the operator may not wish to use the MSIVs without having full use of the main condenser, and yet this may be his only alternative for depressurization (especially if the HPCI/RCIC lines have been isolated due to high temperature). The plant-specific procedures should incude these comments.

State S_1I-8

At this state the operator has successfully maintained the vessel at low pressure. His actions now are to maintain the vessel water level with the available low pressure systems while continuing his attempts to establish a source of decay heat removal.

The action sets indicated by the possible symptoms of this state include all of those also indicated for states S_1-5, 6, and 7. An additional action set, X4, is also indicated here. Set X4 includes EPG step C2-1, depressurization, which is also included in action set X6. Set X4 also instructs the operator to line up and start pumps in two or more low pressure injection trains.

The action sets indicated at this step appear to be correct with the restrictions imposed by state S_1I-7 and discussed for that state. Many of the action sets indicated for this state may include steps that have already been accomplished and would no longer be necessary.

State S_1I-8a

At this state the operator has been unable to maintain low pressure in the RPV, and the RPV has repressurized to above the shutoff head of the low pressure systems. A high pressure source of coolant makeup is therefore necessary.

The action sets indicated for this state are the same as those indicated for states S_1I-5, 6, and 7. No additional sets are indicated by the symptoms present.

To aid the operator in efficient and timely restoration of a high pressure system, additional guidance may be necessary in the plant-specific procedures. At this state the operator did not or could not depressurize the vessel through the MSIVs, the drain valves or the HPCI/RCIC turbines. With the MSIVs closed the reactor vessel has repressurized to above the shutoff head of the low pressure ECCS pumps. Continued steaming through the break and perhaps the safety valves causes the vessel water level to drop. The operator must therefore maintain inventory by placing a high pressure system into service. Due to the size of the break, only the HPCI or feedwater systems may be capable of adequately replenishing the lost fluid. The HPCI pumps have previously tripped due to high backpressure or high temperature in the pump areas. The feedwater system has also been tripped off due to one of a variety of reasons. Guidance may be necessary for the operator to take action to bypass interlocks and restore a pump to service. He should also continue supplying whatever coolant possible with the CRD and RCIC pumps. This may require switching suction from the suppression pool to the CST. These steps should also be considered in the plant-specific procedures or training programs.

State S_1I-9

At this state the operator is successfully maintaining vessel level with low pressure systems, but decay heat removal has not yet been assured. The operator must still restore the RHR or main condenser to service before the reactor can be brought to a stable shutdown condition.

The symptoms which may exist at this state indicate all of the same action sets as for states S_1I-5, 6, and 7. However, two additional action sets may also be indicated. Action set B11 instructs the operator to continue drywell spray operation, while action set B12 instructs the operator to terminate injection into the RPV from sources outside of the primary containment regardless of adequate core cooling. This last set of actions may be indicated if problems develop with suppression pool instrumentation and the

operator is either unable to determine to pool level or reads a false signal from a faulty instrument.

The possible indication of action set B12 could result in some problems, not the least of which is the onset of fuel damage. If the operator has been unable to restore the RHR to service, it may be due to a failure to initiate the recirculation of suppression pool water with the low pressure pumps. If this is the case, the operator may be maintaining vessel level by injecting water from the CST or some other source outside of the containment. If the operation of this injection is halted without initiating recirculation, the vessel level will drop and the fuel will uncover. Because of the adverse conditions which exist within the suppression chamber and drywell, it may be difficult for the operator to determine whether the instrument readings for the pool level are correct or not. If injection is continued from outside the containment, and the suppression pool level is indeed high, the containment may be challenged due to the excessive amount of water. However by halting injection, the fuel may be damaged regardless of the pool water level. In either case, failure to restore RHR operation may eventually result in containment overpressurization failure. Because of these uncertainties it appears that operator training should thoroughly treat the plant-specific features which may affect the decision to terminate flow to the RPY since this step alone may induce a degraded core state. Based upon the EPGs the operator is also proceeding to drain the suppression pool into the CST, as per step SP/L-1 in action sets B1, B2, and B4, among others.

In addition to the possible problem with the injection of water into the vessel, the operator may have problems with restoration of the RHR. For example, with the suppression pool at elevated temperatures, i.e. in the range of 300°F for a time period greater than 20 hours, there may be some difficulty in the initiation of effective RHR pool cooling at this point. Specifically, as the suppression pool fluid is pumped by the RHR pumps through the RHR heat exchangers there may be some flashing (this is possible but unlikely if the RHR system is maintained at sufficiently high pressure). If flashing does occur the steam may come in contact with cool water present in the RHR piping or in the RHR heat exchangers. This contact could result in violent condensation and water hammer which may damage the RHR system. Pressures and temperatures within

the RHR system may be extremely important in this operation of RHR restoration under elevated suppression pool and containment temperatures. The operator should be made aware of these potential problems in his plant-specific procedures or associated training.

State S_1I-9a

At this stage the operator has been unable to maintain the vessel at a low pressure, but has been successful in restoring a source of high pressure makeup. The vessel water level is being maintained. The operator's actions at this state are similar to those at state S_1I-9. The action sets and comments generated by this state are the same as for state S_1I-9 (see state S_1I-9 for more details).

States S_1I-10,10b

At these states the operator has successfully established a form of decay heat removal through either the PCS or the RHR. The system is approaching stable shutdown conditions. The operator must monitor and maintain this trend.

The action sets indicated by the possible symptoms at this state include the majority of those also indicated for states S_1I-9 and 9a. However, action set B12 (termination of injection regardless of core cooling) is not indicated for states S_1I-10 and 10b, but sets A3 and A4 are. Set A3 instructs the operator to initiate RHR operation (which he may have done already to reach this stage), while set A4 instructs him to initiate suppression pool cooling, establish a flow path through a SRV to the suppression pool, start suction from the suppression pool, and proceed to cold shutdown. It also includes instructions to close the RPV head vents, MSIVs, main steam drain valves and HPCI/RCIC isolation valves.

With the exception of set A4, the indicated action sets appear to present no problems. However, A4 could be incorrect if the operator is

cooling with the PCS and is unable to use the RHR. In this case, closing the MSIVs and other indicated valves would result in placing the reactor back in a state similar to state S_1I-7, in which the vessel is repressurizing and decay heat removal is non- existent (except via conduction through the containment walls). The discussions for states S_1I-4 and S_1I-7 in this section contain additional details. Additionally, the first step in action set A4 (step C5-1) includes the instruction to "initiate suppression pool cooling." This may not be possible since almost all RHR failures also disable suppression pool cooling. Thus, if the operator is using the PCS because he could not use the RHR, he should not pursue action set A4. This obvious caution can be easily included in the operator training program and/or plant-specific procedures.

States S_1I-10a,10c

At these states inventory is being maintained but the RHR and PCS are not in service. The containment conditions are near the point where containment failure could occur. The core may be vulnerable to core damage as a result of events cascading from containment failure. The operator should continue his attempts to establish decay heat removal while also maintaining inventory in the vessel.

The action sets which the operator may be directed to follow include several which have been indicated at previous states. Sets A1, A2, A5, A6, B1, B2, B3, B4, B5, B10, and B11 are sets which are indicated at state S_1I-1 and are also indicated for states S_1I-10a and 10c. Action sets B6, B9, and X6 are first indicated for states S_1I-3 and 4, while set X4 is also indicated at state S_1I-8. Action sets new to states S_1I-10a and 10c are sets X1 and X1. Action set X3 is also indicated for state S_1I-10a, but not for S_1I-10c (sets X1 and X3 are very similar in nature, as described below).

Set X1 directs the operator to check on the status of the SRVs and core spray or motor-driven feedwater pumps. If enough SRVs can be opened for emergency depressurization, or if the core spray or feedwater pumps are available, the operator is instructed to close the MSIVs, main steam line drain valves, and HPCI/RCIC/RHR isolation valves. He then should inject into the RPV with the FW or CS pumps until the SRVs open, emergency depressurization is accomplished, and RPV pressure is stable above the RPV flooding pressure or the vessel water level is increasing. The other two sets, sets X1 and X3, are similar in many ways. They both include instructions to monitor the RPV level and pressure, depressurize through the ADS valves, and attempt low pressure injection. The primary difference between the two sets is that set X1 instructs the operator to restart the HPCI/RCIC pumps, while X3 does not. This instruction is not necessary at state S_1I-10c because at that state the vessel level is being maintained by a high pressure system (which may be the HPCI, RCIC, or feedwater system), so that restart of these pumps is not necessary.

Some of the instructions indicated for states S_1I-10a and 10c appear, for the most part, to be adequate. However, set X1 contains instructions to close the MSIVs if SRVs can be opened or the feedwater or core spray pumps are available. As discussed for states S_1I-10 and 10b, action set A4, closing the MSIVs may not be advisable, especially at state S_1I-10c. Even if the feedwater pumps are on or the SRVs are open, if the RHR system cannot be restored and the MSIVs are closed, the suppression pool temperature will continue to rise and the containment may be threatened. Decay heat removal through the MSIVs and main condenser should be continued for as long as possible until the RHR system can be placed in service. Only then should the MSIVs be closed.

The guidance offered in the EPGs at these states concerning vessel depressurization may be inappropriate. Rapid depressurization into a suppression pool which has an already elevated temperature may create conditions which will challenge the integrity of the containment. More statements in the plant-specific procedures cautioning the operators of this possibility may be required.

Table 5.2

Indicated Action Sets for the S_1I Sequence

OAET State	Indicated Action Set*
S_1I-1,2	A1, A2, A5, A6 B1, B2, B3, B4, B5, B10
S_1I-3	A1, A2, A6 B1, B2, B3, B4, B5, B9, B10
S_1I-4	A1, A2, A6 B1, B2, B3, B4, B5, B6, B9, B10 X6
S_1I-5	A1, A2, A5, A6 B1, B2, B3, B4, B6, B9, B10 X6
S_1I-6	A1, A2, A5, A6 B1, B2, B3, B4, B6, B9, B10 X6
S_1I-7	A1, A2, A5, A6 B1, B2, B3, B4, B5, B6, B9, B10 X6
S_1I-8	A1, A2, A5, A6 B1, B2, B3, B4, B5, B6, B9, B10 X4, X6
S_1I-8a	A1, A2, A5, A6 B1, B2, B3, B4, B6, B9, B10 X6
S_1I-9	A1, A2, A5, A6 B1, B2, B3, B4, B6, B9, B10, B11, B12 X6
S_1I-9a	A1, A2, A6 B1, B2, B3, B4, B6, B9, B10, B11, B12 X6
S_1I-10, 10b	A1, A2, A3, A4, A5, A6 B1, B2, B3, B4, B6, B9, B10, B11 X6
S_1I-10a, 10c	A1, A2, A5, A6 B1, B2, B3, B4, B5, B6, B9, B10, B11 X1, X3, X4, X5, X6

*See Tables 3.2 and 3.3 for EPG Steps included in these Sets

5.3 Examination of Guidance Following a Transient Induced Stuck Open Safety/Relief Valve

This section briefly describes the differences in indicated EPG action sets between the SORV event and the small break LOCA sequences described in Section 5.1 and 5.2.

As described in Section 4.3.3, some spectrum of transient induced SORVs are equivalent in size to small LOCAs, and the plant response to these events is very similar. Therefore, the symptoms which may be present at each state in the progression of the accident sequence will also be nearly the same. Because of this similarity, the expected operator response to a SORV event will be similar to that for small LOCAs.

The major parameter which will differentiate a SORV event from by a small LOCA is drywell pressure. Pressure in the drywell should not increase immediately following a SORV, whereas it would if a small break occurred inside containment. Any differences in operator actions should be expected to be a direct result of the presence or absense of a high drywell pressure. These actions include initiating drywell sprays, taking manual action to close SORVs, and initiating ADS operation. A comparison of the action sets generated for the TQUV and TW sequences (in which drywell pressure is low in the initial stages of the events) with the S_1E and S_1I sequences (with high drywell pressure), reveals that the operator's response is nearly the same. This is because the action sets indicated by a high drywell pressure are also indicated by other symptoms which would be present during a SORV event.

An investigation of the action sets delineated in Section 3 reveals that a high drywell pressure results in the indication of only two action sets, namely RC1 and CC4. RC1 includes actions to scram and to verify isolation of containment and initiation of ECCS operation. Set CC4 instructs the operator to close all SORVs, operate the standby gas treatment system and drywell purge, and maintain the suppression pool level within limits. Although a high drywell pressure is not present immediately following a transient-induced SORV, the

symptoms which are present, such as low level and increasing suppression temperature, also indicate that the EPG steps included in action sets RC1 and CC4 should be carried out. Thus these sets are indicated for all of the investigated sequences, regardless of whether a high drywell pressure exists or not. The presence of high drywell pressure serves only as a diverse mechanism for indicating the same actions.

Another area which should be closely examined when producing plant-specific procedures is concerned with the initiation of the ADS. Automatic actuation of this system requires a high drywell pressure signal. A failure of the HPCI and RCIC systems concurrent with the SORV could result in core uncovery before sufficient depressurization through the SORV occurs to allow for low pressure ECCS operation. The operator would be required to manually open the ADS or other safety/relief valves for depressurization purposes. Although the operator is instructed to depressurize the vessel with the ADS valves or other valves and systems if the level continues to fall past the low level setpoint (level 2, where the HPCI and RCIC pumps should start), he is also instructed to inhibit automatic operation of the ADS valves if he cannot maintain the vessel level above the ADS initiation setpoint (level 1). Additional discussion on the ADS instructions within the EPGs can be found in Sections 5.1 and 5.2.

5.4 Examination of Guidance Following a Failure to Maintain Inventory After Transient Event

This section examines the guidance given to the operators by the EPGs following a transient event with a failure to maintain inventory. A detailed discussion is presented in Section 4.2.1. Table 5.3 lists the indicated action sets for each of the states discussed below.

States TQUV-1,2

At state TQUV-1 a loss of feedwater initiating event has occurred, resulting in a decreasing vessel water level. At state TQUV-2 the water level has fallen past the low level (level 3) scram setpoint, and has reached the low-low level setpoint (level 2). The reactor control rods have inserted, the MSIVs have closed, and the reactor pressure has increased to the point where the SRVs open to relieve vessel pressure.

The immediate operator actions for this state are to verify that the automatic responses to a loss of feedwater event have taken place (i.e., scram, MSIV closure and HPCI/RCIC initiation). The operator must then act to restore and maintain the water level in the vessel.

The symptoms for these states may indicate a variety of EPG action sets. The sets which may be indicated are A1, A2, A5, B1, B2, B3, B4, B5, and B10. Sets A1 and A2 deal with initiating or verifying the scram, isolation of containment, and ECCS initiation. The operator is also instructed by these sets to minimize SRV cycling (if it is occurring), and to maximize heat removal through the turbine bypass valves to the main condenser. Set A5 may also be indicated by the symptoms of these states. Its instructions are similar to those of A1 with the addition of steps to line up and start pumps in two or more low pressure injection trains (condensate, core spray, LPCI). This action will not be effective at maintaining inventory at this state since the vessel is still at high pressure, but it does provide a permissive signal for the ADS.

Sets B1, B2, B3, B4, B5, and B10 instruct the operator to check for stuck-open relief valves and close them, if possible, since this may be the source of leakage. The operator is also instructed to initiate suppression chamber and drywell sprays, and to monitor and maintain suppression pool parameters (level, temperature) within limits.

Areas within the EPG which may require elaboration and additional clarification for this sequence in the plant-specific procedures are similar to those discussed previously in Section 5.1 for states S_1E-1 and 2. These areas deal with control of the feedwater pumps and initiation of drywell cooling. See the discussion of states S_1E-1 and 2 in Section 5.1 for more detail.

State TQUV-3

At this state the water level in the vessel has dropped to the low-low vessel level setpoint (level 2), where an initiation signal for the HPCIS and RCICS should be generated. However, these systems have failed to operate, and the water level continues to fall. The operator must take immediate steps to restore the feedwater and/or HPCIS/RCICS to service. If this can be accomplished rapidly the operator may be able to restore water level and allow for gradual system cooldown and depressurization.

Several of the action sets that may be indicated at this state are those previously indicated at states TQUV-1 and 2: A1, A2, A5, B1, B2, B3, B4, B5, and B10. Additional action sets indicated at state TQUV-3 are sets B9 and LR1. Action set B9 deals with maintaining the suppression pool level and pressure within limits. This set also instructs the operator to terminate injection to the RPV from sources outside of the primary containment if adequate core cooling is assured. Set X1 instructs the operator to attempt restart of the HPCI/RCIC pumps, as well as to depressurize and initiate low pressure injection.

The EPG steps included in the action sets for this state do not contain explicit instructions to restart the HPCI/RCIC or feedwater pumps.

More detailed guidance concerning the use of the ADS valves for depressurization should also be considered in the plant-specific procedures. These concerns are discussed in more detail in Section 5.1 for state S_1E-3.

State TQUV-4

The operator has successfully restored a source of high pressure coolant makeup at this state, and has been able to recover water level. The vessel pressure is slowly decreasing as the system cools down. The operator's duties now include monitoring the cooldown and switching to the RHR system. The action sets indicated at this state are nearly identical to those indicated for states TQUV-1,2 and 3. All sets indicated at state TQUV-3 are indicated for state TQUV-4, except sets B2 and B3. Action sets A1, A2, and A5 are indicated, and instruct the operator to maintain RPV pressure and level. Sets B1, B4, B5, B9, and B10 also are indicated, and instruct the operator to isolate any SORVs and initiate suppression chamber and drywell spray operation. An additional set is indicated here: set X4. This set contains similar instructions to set X1 except it does not contain the instructions to restart the HPCI/RCIC pumps.

The actions indicated appear to be consistent with those indicated by the OAET, but the EPGs may not provide sufficient direction as far as initiating RHR or PCS operation. EPG action step RC/P-4, "Initiate RHR shutdown cooling", is not included in any of the indicated action sets. One other possible concern is with the drywell sprays. There are no instructions to halt the spray operation although the instructions do say to maintain the suppression pool level within limits, which may require that the operator discontinue the use of the sprays. See Section 5.1, state S_1E-4, for more discussion on the drywell and suppression chamber sprays, and for the need for consideration of these areas when producing plant specific procedures.

State TQUV-4a

At this state the operator has been unable to restore a source of high pressure coolant makeup, and the vessel level is continuing to drop.

The operator's actions at this time should be to depressurize the reactor and initiate or confirm operation of the low pressure injection systems. Manual depressurization through the ADS or other valves must be initiated. Automatic depressurization will not be accomplished soon enough to prevent core uncovery.

Because this state is a continuation of state TQUV-3, the action sets indicated at this state are the same as at that state.

The indicated actions are consistent with those indicated by the OAET for this sequence, with the exception of any comments generated for states TQUV-3 and S_1E-3.

State TQUV-5

At this state the vessel has been manually depressurized. Vessel water level is low, and makeup with a low pressure system is required.

The operator should confirm or initiate operation of the low pressure injection systems. Presumably these systems should be in operation since the operator should verify that adequate discharge pressure exists before initiating depressurization.

All of the action sets previously indicated for state TQUV-4a are also indicated for this state. In addition, three other action sets are indicated by the possible symptoms for this state. They are sets A7, X3, and X4. Set A7 includes many of the EPG steps which comprise action set A1. Additional steps instruct the operator to close the MSIVs, drain valves, and steam condensing isolation if enough SRVs for emergency depressurization can be opened or core spray or feedwater pumps are available. He also is instructed by step C6-3 to inject into the RPV with the core spray, feedwater, LPCI, condensate, CRD, and alternate injection pumps until enough SRVs are open for emergency depressurization. Action sets X3 and X4 instruct the operator to

initiate low pressure injection with the available systems, depressurizing the RPV if necessary. The RHRS should be placed in service as soon as possible.

Because the action sets indicated at this state are the same as at state TQUV-4a, the operator is instructed to carry out a variety of steps which he has already taken. The operator will know this, of course, and will therefore perform only those steps unique to this state. Those include initiation of low pressure injection systems, and start-up of the RHR system when possible. These are entirely consistent with the OAET.

State TQUV-5a

At this state the operator has been unable to depressurize the vessel and initiate low pressure coolant injection. The high pressure systems are still out of service, and the water level in the vessel is therefore continuing to decline. Without sufficient makeup, the core will uncover.

The operator's actions at this time should be focused on continuing efforts to restore some source of inventory makeup to the vessel. Depressurization attempts should be continued.

This state is a continuation of state TQUV-4a, and the actions* indicated by the symptoms at this state are the same as at that state. Comments for that state are also applicable here.

*Actions necessary to respond to core damage have not been addressed either in the OAET discussion or in the EPG action steps.

State TQUV-6

At this state the vessel is depressurized and low pressure injection systems are in operation. The vessel water level is being restored.

The appropriate operator actions at this state are to continue plant cooldown, initiating RHR cooling when possible.

The indicated action sets are the same as those indicated at state TQUV-5. These action sets deal with maintaining vessel level and decay heat removal.

Although most of the actions indicated in the EPGs are correct, they do not provide clear guidance about when (or if) systems should be turned off, especially the drywell or suppression chamber sprays. Discussion of the spray systems is included in Section 5.1 for states S_1E-4a and S_1E-5a. Other comments concerning depressurization have been addressed as part of the discussion for state TQUV-5.

State TQUV-6a

At this state the reactor vessel has depressurized but the low pressure ECI systems have subsequently failed. The vessel water level is not being maintained.

The operator must quickly restore the LPCI system to service, or initiate operation of another low pressure source such as the condensate system. He must then verify recovery of the vessel water level.

The action sets indicated by the symptoms present at this state are the same as those for state TQUV-5. These sets include instructions to restore level with the low pressure systems (including the condensate system.)

The EPG steps included in the action sets for this state are correct, and include adequate instruction to attempt coolant injection with other low pressure systems other than the LPCIS.

State TQUV-7

The operator has restored some source of low pressure makeup, and the vessel water level is recovering. The operator actions and indicated action sets for this state are the same as for state TQUV-6, and comments for that state are also applicable here.

State TQUV-7a

This state is a continuation of state TQUV-6a in which the low pressure ECC systems have failed to maintain inventory. The operator must try to restore the feedwater/condensate system to operation, which may require opening the MSIVs.

All of the action sets indicated for state TQUV-6a are also indicated for state TQUV-7a. With the exception of the comments generated for states TQUV-5 and 6a, the action sets appear to be adequate for the conditions existing at this time.

State TQUV-8

With the exception of the source of makeup, this state is the same as states TQUV-6 and 7. The operator must maintain inventory in the hotwell and CST for continued feedwater pump operation, and then switch to RHR cooling as soon as possible. See states TQUV-6 and 7 for additional discussion.

State TQUV-8a

At this state the system is at low pressure, but there are no injection systems in operation and the vessel level is continuing to drop. This state is a continuation of state TQUV-7a. The discussion of that state should be referenced for more detail.

Table 5.3

Indicated Action Sets for the TQUV Sequence

OAET State	Indicated Action Set(s)*
TQUV-1,2	A1, A2, A5 B1, B4, B5, B10
TQUV-3	A1, A2, A5 B1, B2, B3, B4, B5, B9, B10 X1
TQUV-4	A1, A2, A5 B1, B4, B5, B9, B10 X1, X4
TQUV-4a	A1, A2, A5 B1, B2, B3, B4, B5, B9, B10 X1
TQUV-5	A1, A2, A5, A7 B1, B2, B3, B4, B5, B9, B10 X1, X3, X4
TQUV-5a	A1, A2, A5 B1, B2, B3, B4, B5, B9, B10 XI
TQUV-6	A1, A2, A5, A7 B1, B2, B3, B4, B5, B9, B10 X1, X3, X4
TQUV-6a	A1, A2, A5, A7 B1, B2, B3, B4, B5, B9, B10 X1, X3, X4

Table 5.3 (continued)

Indicated Action Sets for the TQUV Sequence

OAET State	Indicated Action Set(s)*
TQUV-7	A1, A2, A5, A7 B1, B2, B3, B4, B5, B9, B10 X1, X3, X4
TQUV-7a	A1, A2, A5, A7 B1, B2, B3, B4, B5, B9, B10 X1, X3, X4
TQUV-8	A1, A2, A5, A7 B1, B2, B3, B4, B5, B9, B10 X1, X3, X4
TQUV-8a	A1, A2, A5, A7 B1, B2, B3, B4, B5, B9, B10 X1, X3, X4

*See Tables 3.2 and 3.3 for EPG steps included in these sets

5.5 Examination of Guidance Following a Failure to Remove Decay Heat after Transient Event

This section briefly describes the differences in indicated action sets between the TW sequence and the S_1I sequence (described in Section 4.3.2 and 5.2).

The TW sequence is very similar in plant response to the S_1I sequence, as evidenced by the discussions in Sections 4.2.2 and 4.3.2. The primary difference in plant response between the two sequences is the absence of a high drywell pressure during the initial stages of the TW sequence. Despite the absence of this symptom, the indicated action sets for each of the TW states are exactly the same as for the corresponding S_1I states.

The only action sets which may be indicated if a high drywell pressure is present are sets A1 and B4. However, these sets may also be indicated by other symptoms which will be present in the TW sequence. Set A1 is indicated as an appropriate set of actions if the water level in the vessel drops to the low level scram setpoint (level 3). This is the case in the TW sequence. Set B4 is indicated not only if a high drywell pressure exists, but also if the suppression pool temperature begins to rise. Following the closure of the MSIVs (which is postulated to occur very soon into this sequence) the safety valves lift and discharge steam into the pool. Thus the temperature rises, and the actions included in set B4 should be indicated.

The major differences in possible actions for the TW and S_1I sequences are those relating to the operation of the ADS. Because automatic depressurization through the ADS valves requires a high drywell pressure, this function would not occur until very late in the TW sequence. If the high pressure makeup systems fail before the vessel is depressurized, manual activation would be required for low pressure injection to begin before the core uncovers. However, by definition of the TW sequence, the operator successfully maintains inventory in the vessel, and the need for rapid depressurization to allow low pressure ECCS operation is unlikely. Therefore no problems or concerns with the EPG action sets other than those discussed for the S_1I states appear to be present for the TW sequence.

Table 5.4

Indicated Action Sets for the TW Sequence

OAET State	Indicated Action Set*
TW-1,2	A1, A2, A5, A6 B1, B2, B3, B4, B5, B10
TW-3	A1, A2, A6 B1, B2, B3, B4, B5, B9, B10
TW-4	A1, A2, A6 B1, B2, B3, B4, B5, B6, B9, B10 X6
TW-5	A1, A2, A5, A6 B1, B2, B3, B4, B6, B9, B10 X6
TW-6	A1, A2, A5, A6 B1, B2, B3, B4, B6, B9, B10 X6
TW-7	A1, A2, A5, A6 B1, B2, B3, B4, B5, B6, B9, B10 X6
TW-8	A1, A2, A5, A6 B1, B2, B3, B4, B5, B6, B9, B10 X4, X6
TW-8a	A1, A2, A5, A6 B1, B2, B3, B4, B6, B9, B10 X6
TW-9	A1, A2, A5, A6 B1, B2, B3, B4, B6, B9, B10, B11, B12 X6
TW-9a	A1, A2, A6 B1, B2, B3, B4, B6, B9, B10, B11, B12 X6
TW-10, 10b	A1, A2, A3, A4, A5, A6 B1, B2, B3, B4, B6, B9, B10, B11 X6
TW-10a, 10c	A1, A2, A5, A6 B1, B2, B3, B4, B5, B6, B9, B10, B11 X1, X3, X4, X5, X6

*See Tables 3.2 and 3.3 for EPG Steps included in these Sets

5.6 Examination of Guidance in Response to a Station Blackout (SBO) Event

This section addresses the EPG evaluation performed for the station blackout event. A detailed discussion of this sequence, as well as the OAET, can be found in Section 4.2.3.

States SBO-1,2

At state SBO-1 a loss of offsite power event has occurred at the plant. The emergency diesel generators, which should start immediately in such an event, have failed to do so, thus disabling the AC powered equipment. However, a successful reactor scram has occurred (state SBO-2) accompanied by a turbine trip, closure of the MSIVs, and operation of the safety/relief valves. The operator should verify operation of the inventory control systems while attempting to restore a source of AC power.

Many of the action sets indicated by the symptoms present at these and other station blackout states are the same as those present for the TQUV and TW sequences described previously (see Section 4.2.3 for more discussion). The action sets which are indicated at state SBO-1,2 are sets A1, A2, A5, A6, B1, B2, B3, B4, B5, B10, and X1. Sets A1, A5, and A6 specifically instruct the operator to confirm or initiate emergency diesel generator operation, along with scram, containment isolation, and ECCS operation. Safety valve cycling should be minimized. Set A2 instructs the operator to depressurize the RPV. Sets B1, 2, 3, 4, 5, and 10 include instructions to close all SORVs and to monitor and maintain suppression pool, suppression chamber, and drywell parameters within limits, using the sprays if necessary.

Although the instructions included in these action sets are not incorrect, they may not provide sufficiently detailed guidance to support efficient response to the station blackout event. There is no guidance for the operator as far as shedding non-vital loads, ensuring long-term DC battery operation, or switching operation to DC or steam-turbine driven systems (if available). Plant specific implementation of these guidelines should ensure that a rapid transition to instructions regarding these actions can be accomplished. Other comments regarding the action sets indicated for states TQUV-1,2 (Section 5.4) or TW-1,2 (Section 5.5) are also applicable here.

State SBO-3

At this state operation of the turbine-driven HPCI/RCIC pumps has been verified, and water level in the vessel is being maintained by these systems. The operators should continue to maintain inventory in this way while restoration of AC power is attempted. Once AC power is restored, available low pressure systems should then be used to continue inventory makeup.

The action sets indicated for this state include all of those indicated for states SBO-1 and 2. In addition, action sets B6, B9, and X6 are also indicated. Set B6 includes instructions to verify or initiate recirculation pump trip, to shut down the drywell cooling fans, and to initiate drywell sprays. Set B9 deals with maintenance of the suppression pool level and RPV pressure. The instructions in set X6 are to depressurize the RPV using the ADS valves or other systems.

Performance of many of these actions may be prevented or impaired by the lack of AC power. As for states SBO-1 and 2, the guidelines do not provide explicit guidance to support efficient response to the blackout event. Action set X6, which includes the instructions to depressurize, should be used with extreme caution since the unavailability of AC power will render the low pressure ECCS inoperable. If the RPV is depressurized to below 100 psig, the HPCI/RCIC pumps will trip, and no source of makeup to the vessel would be available. Although there are caution statements to this effect, they may require further emphasis given that a station blackout has occurred.

State SBO-4

At this state the station blackout has occurred, the reactor has scrammed, and the MSIVs have closed. However, the HPCI/RCIC pumps have failed to provide adequate inventory to the vessel to make-up for that being lost through the SRVs. The operator must immediately attempt to restore these steam turbine driven pumps to operation or provide a source of AC power for other pumps.

With the exception of action set X6, all of the action sets indicated at state SBO-3 are also indicated here. One set which is indicated at state SBO-4 which was not indicated elsewhere is set X2. This set includes instructions to restart the HPCI/RCIC pumps, monitor the RPV pressure and water level, and line up and start pumps in two or more injection trains (condensate, core spray, LPCI) or alternate injection systems (RHR-SW crosstie, fire, Standby liquid control, etc.).

The actions indicated in the action sets appear to be appropriate for the conditions which exist at this state. However, in addition to these action sets it would be valuable to provide extensive training and plant-specific procedures for operator guidance in establishing a source of AC power to the plant or in providing unique plant-specific features such as alignment of diesel-driven fire pumps, etc., to mitigate such events. Given that a source of power is available, the instructions given in the action sets should be carried out as indicated.

State SBO-5

This state is a continuation of state SBO-4 in which the operator has been unable to repair the emergency diesel generators or offsite power. The water level in the vessel is continuing to fall, and only the turbine-driven pumps in the HPCI/RCIC systems can be used to provide makeup. The operator must restore these pumps to operation to prevent core uncovery. The action sets indicated for this state are exactly the same as for state SBO-4. Comments generated for that state are also applicable here.

State SBO-6

At this state there still is no AC power, and no source of coolant makeup to the reactor. The operator has been unable to place the HPCI/RCIC systems into operation. The vessel water level is dropping, and the core will uncover. The operator must continue his attempts to restore AC power and/or inventory makeup.

pressure systems in operation when possible. Suction should be taken from the suppression pool. The RHR should be used when it becomes possible. Suppression chamber, drywell, and suppression pool cooling should also be initiated.

The action sets indicated for this state include all of those indicated for states SBO-3 and 7 except for sets B6 and X6 (which contain instructions which are included in other indicated action sets).

Because AC power is available, depressurization and low pressure coolant injection can be accomplished, and the concerns expressed at earlier states regarding these actions are not as important. The indicated action sets appear to be consistent with the actions identified by the OAET analysis for this state, although use of the PCS as an alternate heat sink is not identified, so that opening the MSIVs is not included as a possible action in response to a RHR shutdown cooling failure.

State SBO-10a

At this state the operator has been successful in establishing a means of long-term decay heat removal, and the reactor is nearing cold shutdown conditions. The operator need only continue his monitoring activities to ensure the eventual cooldown of the system.

The action sets indicated for this state are A1, A3, A4, A5, A6, B1, B2, B3, B4, B5, B9, and B10. These include instructions to initiate RHR cooling and to proceed to cold shutdown. Other instructions included in these action sets are those already carried out to arrive at this state.

State SBO-10b

A source of AC power is available, the vessel inventory is being maintained, but a long-term decay heat sink has not been established. Without this, the suppression pool temperature will continue to rise, causing an increase in suppression chamber and drywell pressure. The containment integrity may be challenged, which may in turn affect the ability to cool the core. The operator must therefore continue his attempts to establish a means of containment heat removal.

The action sets indicated by the symptoms present at this state are A1, A4, A6, B1, B2, B4, B5, B6, B9, B10, B12, X5, X6, X1, X2, X3, X4, and X7. These sets include instructions for emergency RPV depressurization, RPV flooding, alternate shutdown cooling (by the method of RPV flooding) and other steps necessary for coolant injection.

Although there are numerous actions indicated for this state, they do not include instructions concerning the use of the PCS as a heat sink. In fact, the operator is instructed in these action sets (i.e., set A4) to close the MSIVs so that the RPV can be flooded. According to the symptoms present at this state he is also instructed, by the indicated action sets, to open all ADS valves to depressurize the vessel. This may not be advisable with the elevated conditions which exist in the containment and suppression pool. Rapid depressurization may result in undesirable dynamic loading of the containment. Other comments regarding the EPGs as they relate to this state can be found in Section 5.5 for the TW sequence.

Table 5.5

Indicated Action Sets for the SBO Sequence

OAET State	Indicated Action Set*
SBO-1,2	A1, A2, A5, A6 B1, B2, B3, B4, B5, B10
SBO-3	A1, A2, A5, A6 B1, B2, B3, B4, B5, B6, B9, B10 X6
SBO-4	A1, A2, A5, A6 B1, B2, B3, B4, B5, B9, B10 X2
SBO-5	A1, A2, A5, A6 B1, B2, B3, B4, B5, B9, B10 X2
SBO-6	A1, A2, A5, A6 B1, B2, B3, B4, B5, B6, B9, B10 X2, X5, X6, X7
SBO-7	A1, A2, A5, A6 B1, B2, B3, B4, B5, B6, B9, B10 X6
SBO-8	A1, A2, A6 B1, B2, B4, B5, B6, B9, B10, B11, B12 X2, X3, X4, X5, X6, X7
SBO-9	A1, A2, A5, A6 B1, B2, B3, B4, B5, B9, B10
SBO-10a	A1, A3, A4, A5, A6 B1, B2, B3, B4, B5, B9, B10
SBO-10b	A1, A4, A6 B1, B2, B4, B5, B6, B9, B10, B12 X1, X2, X3, X4, X5, X6, X7

*See Tables 3.2 and 3.3 for EPG Steps included in these Sets

Section 5.7 Examination of Guidance Following an ATWS Event

This section addresses the EPG evaluation performed for the anticipated transient event with a failure to scram. A detailed discussion of this sequence, designated TC, can be found in Section 4.2.4.

State TC-1

At this state an event has occurred which requires a rapid plant shutdown. For the TC accident sequence the initiating event is postulated to be a closure of all MSIVs. The operator should verify or initiate a reactor shutdown with the control rods. Verification of all automatic responses to the initiating event should also occur.

As summarized in Table 5.6, several action sets are indicated for state TC-1. These are sets A1, A6, A10, B5, B10, X1, X4, X5, and X9. Sets A1 and A6 contain similar instructions to initiate or verify reactor scram, containment isolation, and ECCS operation. Safety valve cycling should be minimized, and heat removal to the main condenser maximized. Set A6 also includes instructions to line up and start pumps in two or more low pressure injection trains, thus providing a permissive signal for ADS operation. Additionally, both sets A1 and A6 instruct the operator to prevent automatic ADS operation if the water level can be maintained above the top of the active fuel. Set A10 contains instructions to trip the recirculation pumps and attempt manual insertion of the control rods.

Sets B5 and B10 deal with drywell and suppression chamber cooling. Set B5 contains an instruction to initiate suppression chamber sprays, while set B10 is comprised of steps to shutdown the recirculation pumps and drywell cooling fans and to initiate drywell spray operation.

Steps contained in Sets X1 and X4 include actions to line up and start pumps in low pressure injection trains, restart the HPCI and RCIC pumps if not operating, and depressurize the RPV using either the ADS valves augmented by other SRVs, or other systems such as the main condenser, RHR, or main steamline drains. Set X5 contains instructions for emergency RPV flooding with the core

spray, feedwater, or other pumps until the vessel repressurizes and at least 3 SRVs are open for emergency depressurization. Finally, set X9 instructs the operator to terminate or prevent all injection into the RPV except from the boron injection systems and CRD systems until the RPV pressure is below the minimum Alternate RPV Flooding Pressure, at which time injection should be slowly increased to maintain the level above the top of the active fuel.

As can be discerned from the brief description given above for the action sets indicated at this state, the actions suggested by possible symptoms are quite drastic and offer the potential for operator confusion. The operator is instructed to prevent automatic depressurization through the ADS valves while at the same time rapidly (manually) depressurizing through the same valves. He is told by one of the set of actions which may be indicated to initiate emergency RPV flooding, while another set, indicated by slightly different symptoms, instructs him to halt all injection into the RPV except through the boron injection and CRD systems. There is no priority or explicit order of importance attached to any of these instructions. Given the time constraints placed upon the operator by such an ATWS event, the ability of the operator to successfully interpret and carry out the proper instructions in the early stages is strongly dependent upon plant specific training, control room indication, and adaptation of the generic EPGs to the individual plants.

States TC-2,3,4

At state TC-2 the control rods have failed to automatically insert into the core following receipt of a scram signal. At state TC-3 the recirculation pumps have tripped. At state TC-4 the alternate rod injection system has also failed to shutdown the reactor. The operator must successfully initiate boron injection into the core to terminate the production of power.

As seen from Table 5.6, the action sets indicated for these states are the same as indicated previously for state TC-1. The comments generated for that state are also applicable here. The need for SLC system operation is increasingly important at these states, and its importance may require some highlighting within plant-specific procedures. When developing plant-specific procedures for ATWS events, consideration of relative action importance may be necessary.

State TC-5

At this state the operator has successfully initiated boron injection. He must now restore and maintain the reactor vessel water level above the top of the active fuel while also preserving reactivity control.

State TC-5 is a very important state in the operator action event tree. It represents a pivotal point in the operator attempt to deal with ATWS. The decision at state TC-5 revolves around whether to continue SLC injection and inventory control at high pressure or to require depressurization in order to avoid the possibility of a blowdown sometime later while the suppression pool is at elevated temperature.

All of the action sets previously discussed for states TC-1,2,3, and 4 are also indicated for this state. In addition, sets A8, A11, A13, B1, B4, and X8 are also indicated. Set A8 instructs the operator to open the MSIVs to re-establish the main condenser as a heat sink. Sets A11 and A13 reiterate the instructions to inject boron into the RPV with the SLC and/or RWCU systems. Sets B1 and B4 deal with isolating any SORVs which may be present. Finally, set X8 contains instructions to lower the RPV water level by terminating and preventing all injection into the RPV (except from boron injection and CRD systems) until reactor power drops below 3% or the water level drops below the top of the fuel.

The potential for contradictory instructions concerning depressurization and injection is again present here. Because of the possibility of a subsequent blowdown when the suppression pool temperature is high, the operator may feel more strongly compelled to depressurize now than at other states.

The benefits of depressurization are:

1. It is consistent with other EPG and operator instructions to depressurize when in doubt.

2. It provides backup coolant injection paths via the low pressure systems.

3. It may prevent a challenge to the containment due to blowdown of the RPV from high pressure with the suppression pool at elevated temperature.

The potential drawbacks to these procedures are:

1. They require astute operator action to control the injection to the vessel via the high volume low pressure systems by controlling reactor water level below level 5 or 6 to avoid washing the boron from the core.

2. They place a great deal of emphasis on the water level system to remain valid throughout the blowdown and subsequent plant states, i.e., possible high drywell temperature and low reactor pressure.

Therefore, when developing plant-specific procedures from these generic EPGs, consideration should be given to including more specific instructions on depressurization. Adequacy of level instrumentation should also be investigated and noted.

Operator action to open the MSIVs may be very plant-specific and therefore is appropriately deferred to the development of plant-specific procedures and training.

State TC-6

At this state the operator has successfully initiated high pressure injection into the vessel. He must now maintain the reactor vessel water inventory while also establishing suppression pool cooling. If the suppression pool temperature becomes excessive (greater than approximately 140°F) cooling of the high-pressure turbine-driven injection pumps may degrade, increasing their failure probability.

The action sets indicated for this state are the same as for state TC-5, with one addition: set B9. This set includes instructions to maintain the suppression pool level within limits.

The comments previously mentioned for the prior states continue to be important at this state. In addition, since the pool temperature will rise rapidly during an MSIV closure ATWS event, pool cooling is desirable as soon as possible to prevent reaching unacceptable temperatures. Plant-specific procedures developed from these EPGs should highlight this important fact.

State TC-7

At state TC-7 the operator has been unable to maintain the suppression pool water temperature below the heat capacity temperature limit. The operator

must therefore immediately depressurize the vessel to avoid placing excessive loadings on the containment structure due to ineffective steam condensation.

The action sets indicated for state TC-6 are also indicated here. Sets A7, B2, B6, and X6 are also indicated. Set A7 includes instructions for RPV flooding until enough SRVs for emergency depressurization are open. Set B2 instructs the operator to operate available drywell cooling, while set B6 contains actions directed at shutting down the drywell cooling fans and initiating drywell spray operation. Set X6 instructs the operator to depressurize with the ADS valves augmented by other SRVs and other systems. It also contains actions to initiate RHR shutdown cooling.

The actions to depressurize the reactor are appropriate for this state. However, RPV flooding may not be an appropriate step to take, since boron dilution and wash-out may occur if excessive water is injected into the vessel. These steps should be addressed in more detail when developing plant-specific procedures.

State TC-8

At this state the operator has depressurized the vessel. He must now maintain coolant injection while preserving reactivity control.

Many of the action sets indicated for state TC-8 are the same as for state TC-7. Sets A8, B6, B10, and X1 are not present at state TC-8 (whereas they were for state TC-7). Meanwhile, two additional sets not previously indicated for state TC-7 are indicated here: sets A5 and X3. The actions included in set A5 are very similar to those already indicated as part of sets A1 and A6, and deal primarily with maintaining reactor water level while

controlling RPV pressure. Set X3 also contains actions already indicated as parts of other sets. These deal with establishing low pressure flow to the vessel.

Previous discussions concerning maintenance of water level and vessel pressure are appropriate here. Emphasis may be necessary in plant-specific procedures or training regarding the effect of a rapid system blowdown upon the boron inventory within the vessel. If excessive amounts of boron are flushed from the reactor vessel, recriticality may be possible upon reflood. Therefore extreme caution should be exercised during vessel depressurization to ensure that water level remains known and below level 8.

State TC-9

At state TC-9 the operator has been successful at preserving reactivity control and vessel water level. A continuing source of decay heat removal is necessary to bring the plant to a cold shutdown condition. The operator must therefore establish heat removal through the RHRS or PCS.

The action sets indicated for this state are those previously indicated for state TC-8. Action set A8 is also indicated. This set instructs the operator to establish the main condenser as a heat sink by opening the MSIVs.

With the exception of previously noted areas of concern, the action sets indicated for this state appear to be appropriate.

State TC-9

At this state the operator has been unable to maintain reactivity control or vessel inventory. Core damage may eventually occur if adequate coolout inventory makeup sources cannot be aligned to balance the steaming rate from decay heat. The operator must continue his attempts to maintain inventory and reactivity control.

The action sets indicated here consist of the entire list also indicated for state TC-9. These actions appear to be appropriate for this state.

State TC-10

The plant is approaching a cold shutdown conditions. The operator must continue to monitor this approach.

The action sets indicated for state TC-10 are the same as for state TC-9. See that state for any comments.

State TC-10a

A source of decay heat removal has not been established, although the core is covered and reactivity control is being maintained. The core is vulnerable for damage if the containment overpressurizes and fails. The operator must continue in attempts to establish heat removal with the RHRS or the power conversion system (PCS), and consider containment venting if available.

The actions sets for this state are the same as for state TC-9, and are also appropriate here. See state TC-9 for further discussions.

Summary

There are a number of unique features involved in the reactivity control guidelines. These features require prudent implementation on a plant-specific basis. Some of the plant-specific features which have a strong impact on the EPG implementation are:

1. The MSIV closure setpoint
2. The SLC capacity
3. Manual or automatic SLC operation
4. Injection location of SLC
5. SLC concentration in the storage tank

These features may influence the details of the plant-specific procedures and training which result from EPG implementation.

Table 5.6

Indicated Action Sets for the TC (ATWS) Sequence

OAET State	Indicated Action Set(s)*
TC-1	A1, A6, A10 B5, B10 X1, X4, X5, X9
TC-2,3,4	A1, A6, A10 B5, B10 X1, X4, X5, X9
TC-5	A1, A6, A8, A10, A11, A13 B1, B4, B5, B10 X1, X4, X5, X8, X9
TC-6	A1, A6, A8, A10, A11, A13 B1, B4, B5, B9, B10 X1, X4, X5, X8, X9
TC-7	A1, A6, A7, A8, A10, A11, A13 B1, B2, B4, B5, B6, B9, B10 X1, X4, X5, X6, X8, X9
TC-8	A1, A5, A6, A7, A10, A11, A13 B1, B2, B4, B5, B9 X3, X4, X5, X6, X8, X9
TC-9	A1, A5, A6, A7, A8, A10, A11, A13 B1, B2, B4, B5, B9 X3, X4, X5, X6, X8, X9
TC-9a	same as TC-9
TC-10	same as TC-9
TC-10a	same as TC-9

*See Tables 3.2 and 3.3 for EPG steps included in these sets.

SECTION 6

SUMMARY RESULTS ON EXAMINATION OF BWR EMERGENCY PROCEDURE GUIDELINES

In the preceding sections operator action event trees (OAETs) have been used to systematically examine the guidance provided by the BWR Emergency Procedure Guidelines (Revision 2) during selected accident sequences. The selected accident sequences chosen for the EPG examination include sequences for which adequate coolant inventory makeup is unavailable (TQUV)+, adequate containment heat removal is unavailable (TW)+, small break LOCAs, station blackout induced failure of coolant inventory makeup has occurred, and an anticipated transient without scram (ATWS)+ induced challenge of containment integrity. The accidents selected for examination are chosen based upon unique features which challenge the operator or based upon past PRA analyses which have identified these sequences as potential risk contributors.

In accordance with the EPG review process summarized in Section 2, the individual OAET states were systematically examined to answer four basic questions regarding the guidance provided in the EPGs:

1) Is the Emergency Procedure Guidelines' collection of symptom sets complete?

2) Are the instructions in the EPGs always correct?

3) Are the EPG action sets always complete?

4) Are the instructions in the EPGs always unambiguous?

In this section the results of this examination are collected and summarized.

The OAET techniques demonstrate the feasibility of the methodology as applied to the BWR symptom-based procedures and establish the adequacy of the EPGs for all states examined. However, while this evaluation covers a broad

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+ Designations are taken from shorthand notation of the Reactor Safety Study[10].

spectrum of potential BWR risk-producing sequences, the analysis is not intended as a complete evaluation of all possible accident sequences. The end states of the analysis included stable, hot shutdown and various potential core or containment challenges. The examination of the adequacy of the EPGs beyond core melt or containment failure was not performed.

In addition to examining the general adequacy of the EPGs for the states examined, the OAET-based examination of the EPGs has identified the following items as deserving particular attention in the process of producing plant-specific procedures from the EPGs and in the development of plant-specific training programs:

1) Control of Feedwater following a small break or transient event: As noted in Section 5.1 (States S_1E-1,2) and Section 5.2 (S_1I-1,2), the EPGs provide general guidelines and options available to the operator; however, there is little guidance in the EPGs concerning control of feedwater during a situation in which the HPCI/RCIC pumps may be signalled to start manually or due to high drywell pressure. In many of these cases proper control of the feedwater system alone would be effective in maintaining vessel inventory. However, with the rapid addition of flow from the HPCI pump, the vessel level may rise to the high-level trip setpoint, where all pumps (including Feedwater) would trip. The feedwater pumps (which are not safety-related) would not automatically restart at the low-level setpoint, and in addition, the MSIVs may close. Rapid and knowledgeable operator response could prevent the feedwater trip as well as the subsequent events. Therefore, it is judged to be prudent to incorporate in the plant-specific procedures or the training program additional guidance to the operator on response regarding preserving feedwater and throttling HPCI during such challenges.

2. Restart of feedwater, HPCI, and RCIC pumps: Discussion in Section 5.1 (State S_1E-3) and Section 5.2 (State S_1I-8a) indicates that more timely and efficient operator response to pump trips could result if additional information is provided. The interface between feedwater and the other high-pressure turbine-driven systems, particularly in potential degraded plant states[+], is highly plant-specific and therefore is appropriately treated in the plant-specific procedures and training programs. Guidance on the operation of these systems is found by the Operator Action Event Trees to pervade all the accident sequences examined; therefore, the importance of operator training on the plant-specific features of the high-pressure injection systems available is highlighted by this analysis.

[+] For example, sequences occurring in which a high reactor building temperature may occur could lead to the isolation of HPCI, RCIC, and the closure of the MSIVs.

3. Availability of main condenser: The Operator Action Event Tree evaluation in Sections 5.2 (S_1I-small LOCA) and 5.4 (TW) describes the potential use of the main condenser as a heat sink in the event that RHR is unavailable for containment heat removal. There are a number of steps within the EPGs which indicate that the operator should be controlling pressure and heat removal through the turbine bypass valves, i.e. with the MSIVs open. These guidelines make a strong implication that the MSIVs should be kept open, e.g. turn MODE switch from RUN to STARTUP. However, as discussed in the comparative analysis performed in Section 5, the EPGs do not explicitly identify the main condenser as a primary heat sink (with the exception of reactivity control contingency guidelines). In particular, no generic guidance is given on reopening the MSIVs. Since the balance of plant systems (BOP) are highly stylized, it seems appropriate that detailed guidance be properly provided in the plant-specific procedures and training programs. In the conversion of the EPGs to plant-specific procedures it would be beneficial to incorporate plant-unique features which would enhance maintaining the MSIVs open or reopening the MSIVs.

4. Use of drywell or suppression chamber sprays: The use of the sprays to cool the containment air space may be indicated at almost every OAET state discussed in Section 5. However, at no point is there guidance as to when (or if) these sprays can be turned off (except possibly in the case of excessive suppression pool level). Minimal use of the sprays could decrease the chance of equipment damage in other systems due to moisture and flooding. Operator training can easily treat the best approach to termination of containment sprays on a plant-specific basis.

5. Restoration of RHR following extended down-time: The S_1I and TW sequences (Section 4.2.2 and 4.3.2) respectively postulate a failure of the RHR to remove decay heat. A necessary and correct operator response in these situations is to attempt repairs and restoration of this system. However, as discussed in Section 5.2 (State S_1I-9), restoration of this system late in the sequence could result in rapid condensation and flashing which could disable the system. The EPGs contain no cautions on this matter. It may be appropriate to include such a caution in the plant-specific procedures if the unique plant configuration warrants it.

6. Response following a station blackout event: As mentioned in Section 5.5 regarding station blackout, there may be additional information which can usefully be incorporated in the plant-specific procedures to expand the operator's perspective on the appropriate response and its long-term consequences. For accident sequences which have unique and characteristic initiators, such as station blackout, the information developed by each plant in the existing event-based procedures may contain valuable information. While the EPGs are function-oriented, there are certain aids which can assist the operator in optimizing the response if the sequence is amenable to easy identification in

event space, thereby allowing parallel implementation of the existing event-based guidance. Such guidance can be explicitly included in associated plant-specific procedures and/or the training program. For the specific case of station blackout, plant-specific procedures should include such items as:

- o load shedding from the batteries
- o the preferred mode of coolant injection
- o long-term actions

7. Use of ADS: The operator response at several of the OAET states examined includes initiation of reactor system depressurization. This response is necessary following failures of the high-pressure coolant injection systems or containment heat removal systems. The examination of the EPG action steps which relate to system depressurization reveal that the operator is instructed to inhibit automatic operation of the ADS regardless of reactor inventory as determined by water level. Specifically, action set A5, which is indicated at many states, includes such steps within Contingency #1, "Level Restoration." Contingency #1 cautions the operator while performing step C1-4 through C1-8 that "if RPV...water level drops below [ADS initiation setpoint], prevent automatic initiation of ADS." Step C1-7, however, states "if no CRD pump is operating but at least 2 injection subsystems are lined up for injection with pumps running, EMERGENCY RPV DEPRESSURIZATION IS REQUIRED." Step C2-1.1, in Contingency #2 (entitled "Emergency RPV Depressurization"), states "If suppression pool water level is above [elevation of top of SRV discharge device], open all ADS valves." The prevention of automatic ADS at the same time as the instruction to initiate manual Emergency Depressurization appears somewhat confusing based solely on the EPGs. The operator training program should ensure that the operators are aware of the bases for this distinction, ensure that no hesitancy exists to implement depressurization if required, and describe when and how the operator could obtain the benefits of a more controlled manual depressurization.

8. Drywell and suppression chamber cooling: Based upon possible symptoms, the operator may be instructed from the beginning of every sequence investigated to initiate drywell and suppression chamber cooling. However, these instructions are not explicit with regard to when, or if, sprays can be turned off (as mentioned above). Specific cases in which termination of drywell sprays might be advisable should be covered in the plant-specific procedures (17). Additionally, there are some OAET states identified where the instructions regarding cooling operations may require thorough training to avoid confusion (see, for example, Section 5.1, states S_1E-1,2). At these states, the action sets B2 and B3 are indicated as appropriate. Indicated action set B2 includes step DW/T-1, which states: "operate available drywell cooling" (which presumably includes drywell fans). However, action set B3 includes step DW/T-3, "shut down...drywell cooling fans and initiate drywell sprays." These statements could create some confusion for the untrained operator. Notwithstanding the

timing of drywell fan cooling initiation and termination, the drywell cooler fans at some plants may also trip on high drywell pressure and/or low reactor water level. Therefore the operator may be required to bypass interlocks to initiate operation of the fans. There are no caution statements to this effect in the EPGs. A thorough training program could be expected to adequately ensure operator understanding of these actions.

9. Response to ATWS events: Section 5.7 identified conditions which may occur following an ATWS event in which the operator may potentially be faced with contradictory instructions from the EPGs. These conflicting instructions include: (1) prevent automatic depressurization through the ADS valves, and (2) rapidly depressurize by manually opening the ADS valves (see Section 6.2.1). Also indicated are instructions to (3) terminate and prevent injection into the RPV except with the SLC and CRD systems and (4) flood the RPV with core spray, feedwater, or other pumps until enough SRVs for emergency depressurization are open. During the initial stages of an ATWS event the operator has very little time to respond, especially if he is to prevent excessive suppression pool temperature rise. The difficult nature of the instructions make it necessary to thoroughly integrate the ATWS contingency procedures into the operator training program. Other specific comments concerning ATWS are as follows:

   a) The EPGs indicate that for cases in which (1) any control rod is out beyond notch 06, (2) power is indeterminant, and (3) the suppression pool temperature is approaching 110°F the operator should act as if he had an ATWS. This could occur in loss of offsite power cases (with the APRMs powered by normal power) when coupled with delayed RHR operation or the occurrence of an SORV. This reaction may lead to a number of inadvertent SLC initiations, with extended down time for cleanup. It may be erring on the side of safety. However, there may also be ATWS contingency actions taken by the operators unnecessarily, i.e. lowering the water level to the top of the fuel, which may actually reduce the margin of safety.

   b) The generic EPGs provide no guidance to the operator on the best technique for controlling the high-volume flow rate, low-pressure systems if he must use them during an ATWS condition. Possible plant differences in valve arrangements, preferred power supplies to low-pressure pumps, and available flow indication may all lead to varying choices of operator response. In this same vein the preferred course of action for the operator in the unlikely event that the reactor water level indication becomes indeterminant during an ATWS should be identified. The operator guidance is best treated in plant-specific procedures and training programs.

   c) The implementation on a plant-specific basis of the Reactivity Control Guideline of Contingency 7 may be influenced by the plant-specific decision on the MSIV closure setpoint on reactor water level. The impact of the MSIV closure setpoint should be

examined in the conversion of the generic EPGs into the plant-specific procedures. The examination may be only for consistency in training or there may be reason to modify the plant-specific decision on ATWS procedures or MSIV closure setpoints.

d) Other plant-specific decisions which may impact the implementation of the ATWS contingency procedures include questions such as:

   (1) Are the power range monitors on a power supply which will be available following anticipated transients, e.g. loss of offsite power?

   (2) Are the water level instrumentation arrangements adequate for the operator to control water level near the top of the fuel?

   (3) Is the plant turbine bypass capacity greater than 50% such that the ATWS procedures could be dramatically affected?

   (4) Are the drywell coolers adversely affected by the postulated ATWS conditions? And what operator actions may be necessary to keep them available if desired?

e) No explicit guidance is given to the operator in case of a confirmed partial ATWS or a turbine trip ATWS from low power level. In these cases, lowering the water level may be unnecessary and inappropriate because the generated steam can be effectively removed to the condenser without adverse impacts on the plant. Operator training can effectively cover these cases with appropriate cautions included in the plant-specific procedures.

10. Termination of RPV injection

Training and plant-specific implementation of the EPG action to terminate RPV injection based upon containment water level should be given serious consideration within the operator training program. An event parallel to TMI could be postulated to occur in certain rare accident sequences where the operator could be sufficiently concerned with overfilling (e.g. the RPV at TMI or the containment in a BWR) as indicated by potentially suspect non-safety-related instrumentation that the operator terminates in-flow to the RPV. Cautions in the plant-specific procedures coupled with adequate training should circumvent this difficult area.

11. Return to normal or suggested parameters

The operator training program may also be able to treat rare abnormal occurrences in which the operator finds his parameters far in excess of those suggested in the EPGs. Under such conditions, the training program could provide useful guidance to

the operator on the safest approach to normal parameters. One example of this, cited in the OAET review, is the possibility of reaching high suppression pool temperatures due to loss of containment heat removal and the subsequent closure of the SRVs due to high containment pressure. Subsequent recovery of RHR and then SRV capability would allow the operator the capability to rapidly depressurize; however, it may be prudent to avoid a rapid Emergency Depressurization in such cases involving return to normal parameters.

12. Other plant-specific features: There may also arise other plant-specific features from accident sequences which can be effectively treated in the plant-specific procedures, but which are not explicitly addressed in the EPGs. Examples of such items which have derived from the EPG examination are:

    - Hydrogen control following accidents such as large LOCA or severely degraded core conditions.

    - For certain remote-probability accident sequences the control room indication remaining available to the operator may be important. The operator training program should define potential degraded instrumentation states for the operator.

    - The integration of the remote shutdown panel into the emergency procedures.

    - The operation of ADS may be adversely affected during the accident sequences evaluated in this report via high containment temperature, high containment pressure, and loss of instrument air. Special training or cautions in the plant-specific procedures could avoid operator confusion if such conditions ever arose.

    - Suppression pool stratification can be avoided by proper training.

    - While the EPGs recommend RPV flooding if high drywell temperatures exist and the reactor depressurizes, plant-specific features such as small vertical reference by drops may lead individual plants to judiciously exercise this recommendation, taking into consideration their plant-specific features. Avoidance of RPV flooding is generally a desirable goal.

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EG&G Idaho, Inc.
P.O. Box 1625
Idaho Falls, Idaho 83415