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Analysis of the LaSalle Unit 2 Nuclear Power Plant: Risk Methods Integration and Evaluation Program (RMIEP)

Initiating Events and Accident Sequence Delineation

Prepared by
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Sandia National Laboratories
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ABSTRACT

This volume presents the results of the initiating event and accident sequence delineation analyses of the LaSalle Unit 2 nuclear power plant performed as part of the Level III PRA being performed by Sandia National Laboratories for the Nuclear Regulatory Commission.

The initiating event identification included a thorough review of extant data and a detailed plant specific search for special initiators. For the LaSalle analysis, the following initiating events were defined: eight general transients, ten special initiators, four LOCAs inside containment, one LOCA outside containment, and two interfacing LOCAs.

Three accident sequence event trees were constructed: LOCA, transient, and ATWS. These trees were general in nature so that a tree represented all initiators of a particular type (i.e., the LOCA tree was constructed for evaluating small, medium, and large LOCAs simultaneously). The effects of the specific initiators on the systems and the different success criteria were handled by including the initiating events directly in the system fault trees. In this way, if an initiator failed or partially failed a system, then the effects of the initiator would propagate correctly through the fault trees. The initiator would appear in the sequence cut sets just like any other failure event.

The accident sequence event trees were extended to include the evaluation of containment vulnerable sequences. In order to model this, additional events were added to the event trees in order to develop the sequences until resolution of the state of the core was obtained. This process included: an expert elicitation of containment failure pressure, location, and size; the evaluation of reactor building environments as a result of containment failure or venting using the MELCOR code; a determination of the location and type of equipment in the reactor building; the construction of simple Boolean expressions for failure of the systems in these severe environments; and an expert elicitation to evaluate equipment failure probabilities in these environments. This methodology was used in simplified form in the NUREG-1150 analyses.

These internal event accident sequence event trees were also used for the evaluation of the seismic, fire, and flood analyses. The system fault trees were expanded to include cabling, piping, passive failures, multiple spurious actuations, and location information for all components, piping, and cabling. The results of these analyses are presented in Volumes 8, 9, and 10 of this report, respectively.

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FOREWORD

LaSalle Unit 2 Level III Probabilistic Risk Assessment

In recent years, applications of Probabilistic Risk Assessment (PRA) to nuclear power plants have experienced increasing acceptance and use, particularly in addressing regulatory issues. Although progress on the PRA front has been impressive, the usage of PRA methods and insights to address increasingly broader regulatory issues has resulted in the need for continued improvement in and expansion of PRA methods to support the needs of the Nuclear Regulatory Commission (NRC).

Before any new PRA methods can be considered suitable for routine use in the regulatory arena, they need to be integrated into the overall framework of a PRA, appropriate interfaces defined, and the utility of the methods evaluated. The LaSalle Unit 2 Level III PRA, described in this and associated reports, integrates new methods and new applications of previous methods into a PRA framework that provides for this integration and evaluation. It helps lay the bases for both the routine use of the methods and the preparation of procedures that will provide guidance for future PRAs used in addressing regulatory issues. These new methods, once integrated into the framework of a PRA and evaluated, lead to a more complete PRA analysis, a better understanding of the uncertainties in PRA results, and broader insights into the importance of plant design and operational characteristics to public risk.

In order to satisfy the needs described above, the LaSalle Unit 2, Level III PRA addresses the following broad objectives:

1. To develop and apply methods to integrate internal, external, and dependent failure risk methods to achieve greater efficiency, consistency, and completeness in the conduct of risk assessments;

2. To evaluate PRA technology developments and formulate improved PRA procedures;

3. To identify, evaluate, and effectively display the uncertainties in PRA risk predictions that stem from limitations in plant modeling, PRA methods, data, or physical processes that occur during the evolution of a severe accident;

4. To conduct a PRA on a BWR 5, Mark II nuclear power plant, ascertain the plant's dominant accident sequences, evaluate the core and containment response to accidents, calculate the consequences of the accidents, and assess overall risk; and finally

5. To formulate the results in such a manner as to allow the PRA to be easily updated and to allow testing of future improvements in methodology, data, and the treatment of phenomena.

The LaSalle Unit 2 PRA was performed for the NRC by Sandia National Laboratories (SNL) with substantial help from Commonwealth Edison (CECo) and its contractors. Because of the size and scope of the PRA, various related programs were set up to conduct different aspects of the analysis. Additionally, existing programs had tasks added to perform some analyses for the LaSalle PRA. The responsibility for overall direction of the PRA was assigned to the Risk Methods Integration and Evaluation Program (RMIEP). RMIEP was specifically responsible for all aspects of the Level I analysis (i.e., the core damage analysis). The Phenomenology and Risk Uncertainty Evaluation Program (PRUEP) was responsible for the Level II/III analysis (i.e., accident progression, source term, consequence analyses, and risk integration). Other programs provided support in various areas or performed some of the subanalyses. These programs include the Seismic Safety Margins Research Program (SSMRP) at Lawrence Livermore National Laboratory (LLNL), which performed the seismic analysis; the Integrated Dependent Failure Analysis Program, which developed methods and analyzed data for dependent failure modeling; the MELCOR Program, which modified the MELCOR code in response to the PRA's modeling needs; the Fire Research Program, which performed the fire analysis; the PRA Methods Development Program, which developed some of the new methods used in the PRA; and the Data Programs, which provided new and updated data for BWR plants similar to LaSalle. CECo provided plant design and operational information and reviewed many of the analysis results.

The LaSalle PRA was begun before the NUREG-1150 analysis and the LaSalle program has supplied the NUREG-1150 program with simplified location analysis methods for integrated analysis of external events, insights on possible subtle interactions that come from the very detailed system models used in the LaSalle PRA, core vulnerable sequence resolution methods, methods for handling and propagating statistical uncertainties in an integrated way through the entire analysis, and BWR thermal-hydraulic models which were adapted for the Peach Bottom and Grand Gulf analyses.

The Level I results of the LaSalle Unit 2 PRA are presented in: "Analysis of the LaSalle Unit 2 Nuclear Power Plant: Risk Methods Integration and Evaluation Program (RMIEP)," NUREG/CR-4832, SAND92-0537, ten volumes. The reports are organized as follows:

NUREG/CR-4832 - Volume 1: Summary Report.

NUREG/CR-4832 - Volume 2: Integrated Quantification and Uncertainty Analysis.

NUREG/CR-4832 - Volume 3: Internal Events Accident Sequence Quantification.

NUREG/CR-4832 - Volume 4: Initiating Events and Accident Sequence Delineation.

NUREG/CR-4832 - Volume 5: Parameter Est.mation Analysis and Human Reliability Screening Analysis.

NUREG/CR-4832 - Volume 6: System Descriptions and Fault 'ree Definition.

NUREG/CR-4832 - Volume 7: External Event Scoping Quantification.

NUREG/CR-4832 - Volume 8: Seismic Analysis.

NUREG/CR-4832 - Volume 9: Internal Fire Analysis.

NUREG/CR-4832 - Volume 10: Internal Flood Analysis.

The Level II/III results of the LaSalle Unit 2 PRA are presented in: "Integrated Risk Assessment For the LaSalle Unit 2 Nuclear Power Plant: Phenomenology and Risk Uncertainty Evaluation Program (PRUEP)," NUREG/CR-5305, SAND90-2765, 3 volumes. The reports are organiz d as follows:

NUREG/CR-5305 - Volume 1: Main Report

NUREG/CR-5305 - Volume 2: Appendices A-G

NUREG/CR-5305 - Volume 3: MELCOR Code Calculations

Important associated reports have been issued by the RMIEP Methods Development Program in: NUREG/CR-4834, Recovery Actions in PRA for the Risk Methods Integration and Evaluation Program (RMIEP); NUREG/CR-4835, Comparison and Application of Quantitative Human Reliability Analysis Methods for the Risk Methods Integration and Evaluation Program (RMIEP); NUREG/CR-4836, Approaches to Uncertainty Analysis in Probabilistic Risk Assessment; NUREG/CR-4838, Microcomputer Applications and Modifications to the Modular Fault Trees; and NUREG/CR-4840, Procedures for the External Event Core Damage Frequency Analysis for NUREG-1150.

Some of the computer codes, expert judgement elicitations, and other supporting information used in this analysis are documented in associated reports, including: NUREG/CR-4586, User's Guide for a Personal-Computer-Based Nuclear Power Plant Fire Data Base; NUREG/CR-4598, A User's Guide for the Top Event Matrix Analysis Code (TEMAC); NUREG/CR-5032, Modeling Time to Recovery and Initiating Event Frequency for Loss of Off-Site Power Incidents at Nuclear Power Plants; NUREG/CR-5088, Fire Risk Scoping Study: Investigation of Nuclear Power Plant Fire Risk, Including Previously Unaddressed Issues; NUREG/CR-5174, A Reference Manual for the Event Progression Analysis Code (EVNTRE); NUREG/CR-5253, PARTITION: A Program for Defining the Source Term/Consequence Analysis Interface in the NUREG-1150 Probabilistic Risk Assessments, User's Guide; NUREG/CR-5262, PRAMIS: Probabilistic Risk Assessment Model Integration System, User's Guide; NUREG/CR-5331, MELCOR Analysis for Accident Progression Issues; NUREG/CR-5346, Assessment of the XSOR Codes; and NUREG/CR-5380, A

User's Manual for the Postprocessing Program PSTEVNT. In addition the reader is directed to the NUREG-1150 technical support reports in NUREG/CR-4550 and 4551.

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1.0 INTERNAL INITIATING EVENT ANALYSIS

1.1 Introduction

This section describes the methodologies used to identify and quantify the initiating events for the LaSalle Unit 2 Nuclear Power Plant. An initiating event is defined as an anomalous event which requires a plant shutdown and challenges the plant systems that can be used to mitigate the event (either safety or balance of plant systems that are included in the PRA model). For such events, subsequent failures in the modeled systems could result in core damage and radionuclide release from the fuel. Initiating events occurring during startup, shutdown, refueling, or other modes of operation are not considered in this analysis, nor are those events concerned with sources of radioactivity other than the core (e.g., the spent fuel pool, etc.). The objective of the initiating event task is to identify and quantify all probabilistically significant initiating events applicable to the LaSalle Unit 2 nuclear power plant that can occur while the plant is at power.

The remainder of this section contains an overview of the task, a methods review, and the actual analysis and results.

1.2 Overview

Methodologies used in the initiating event analysis task may be grouped into those relating to identification and those relating to quantification. The methodology used for identification is that referred to as a comprehensive engineering evaluation ("PRA Procedures Guide," terminology).[1] Identification of initiating events was performed first on a generic and then on a plant specific basis. The generic sources for initiators included the following:

1. EPRI NP-2230,[2]
2. Nuclear Power Experience,[3]
3. NUREG/CR-3862,[4] and
4. Past PRAs of plants with similar characteristics.

Review of these sources resulted in a list of initiating events potentially applicable to the LaSalle Unit 2 design.

The second step in the identification of initiating events was a detailed study of the LaSalle Unit 2 design and operating experience. This step was used to identify LaSalle-specific initiators not already identified in the generic review, and to evaluate those initiators identified in the first step as to their applicability to LaSalle Unit 2. The two steps described above were applied to both the transient and loss of coolant accidents (LOCA) classes of initiators.

Various quantification methodologies were used, depending on the type of initiator. Because of the limited operating experience for LaSalle Unit

2, generic frequencies obtained from U.S. BWR commercial nuclear power plant experience were generally used. Exceptions to this include the LOCAs and certain special initiators which were evaluated on a plant-specific basis.

The relationship of the initiating event analysis task to other systems analysis tasks is shown in Figure 1.1. The basic input to this task was from the plant design and operation information collection task. The event tree analysis task interacted with the initiating event analysis task in the grouping of initiators into categories. Also the fault tree analysis task interacted with the initiator task in order to accurately determine the effect of an initiator upon systems responding to the accident. Finally, interaction also occurred with the data base and uncertainty analysis task in order to quantify the initiator frequencies.

Output from the initiating event analysis task, a list of initiating event categories and frequencies applicable to LaSalle Unit 2, was input to the event tree analysis task. The initiator list was also input to the fault tree analysis task so that initiators could be modeled in the fault trees and their effects on the systems could be accurately represented, where appropriate.

1.3 Methods

No new methodologies were developed for the initiating event analysis task. The work essentially followed guidelines presented in the "PRA Procedures Guide"[1] and the "IREP Procedures Guide".[5] A survey of BWR operational experience and past PRAs was combined with a comprehensive analysis of the LaSalle Unit 2 design. Comprehensive searches for special initiators and interfacing system LOCAs were conducted. The resulting initiating events were then grouped into categories based on similarity of effect on the balance-of-plant and safety systems. Finally, the initiating event categories were quantified.

1.4 Analysis

1.4.1 Initiating Event Identification

As mentioned in the overview, initiating event identification occurred in two steps. The first step involved a survey of U.S. boiling water reactor (BWR) experience and various PRA studies. EPRI NP-2230,[2] Nuclear Power Experience,[3] and NUREG/CR-3862[4] were used in the BWR operating experience survey. The EPRI NP-2230 source represents events occurring at 16 different BWRs over 102 plant years, spanning the period from 1964 through 1980. NUREG/CR-3862 is an update of the EPRI analysis through 1983 using different statistical techniques. Initiators identified in this survey are listed in Table 1.1.

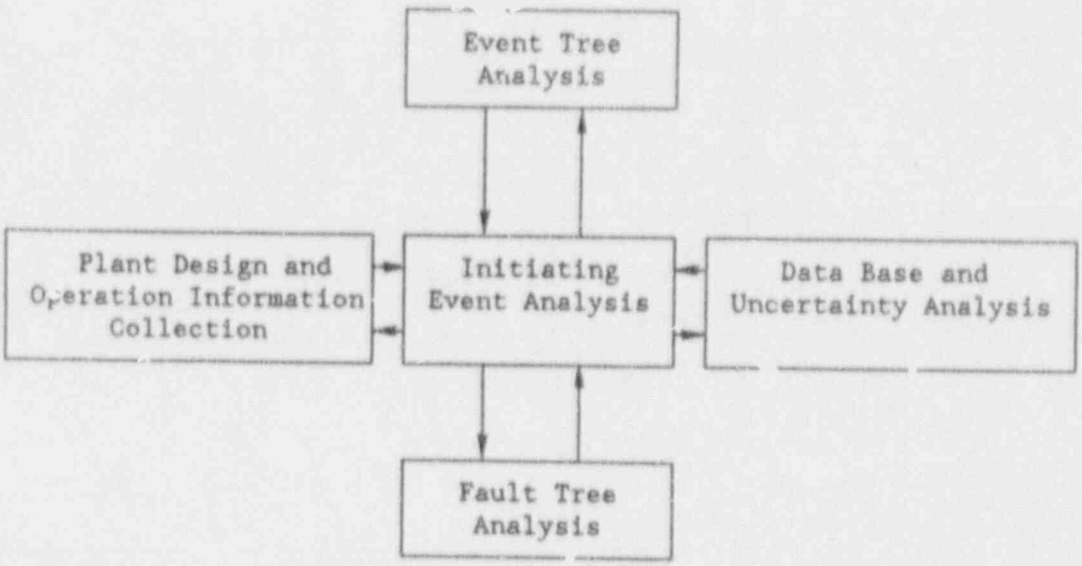

Figure 1.1 Initiating Event Analysis Task Interfaces

Table 1.1 Potential Internal Initiating Events for LaSalle Unit 2

	Potential Initiating Event for LaSalle Unit 2 (EPRI NP-2230 Nomenclature)	Initiator Type	Source	Applicable to LaSalle Unit 2 Study?
1)	Electric load rejection	Transient (general)	EPRI NP-2230	Yes
2)	Electric load rejection with turbine bypass valve failure	Transient (general)	EPRI NP-2230	Yes
3)	Turbine trip	Transient (general)	EPRI NP-2230	Yes
4)	Turbine trip with turbine bypass valve failure	Transient (general)	EPRI NP-2230	Yes
5)	Main steam isolation valve closure	Transient (general)	EPRI NP-2230	Yes
6)	Inadvertent closure of one main steam isolation valve	Transient (general)	EPRI NP-2230	Yes (above 80% power)
7)	Partial main steam isolation valve closure	Transient (general)	EPRI NP-2230	Yes
8)	Loss of normal condenser vacuum	Transient (general)	EPRI NP-2230	Yes
9)	Pressure regulator fails open	Transient (general)	EPRI NP-2230	Yes
10)	Pressure regulator fails closed	Transient (general)	EPRI NP-2230	Yes
11)	Inadvertent opening of a safety/relief valve (stuck)	Transient (general)	EPRI NP-2230	Yes
12)	Turbine bypass fails open	Transient (general)	EPRI NP-2230	No (does not cause a plant trip)

Table 1.1 Potential Internal Initiating Events for LaSalle Unit 2 (Continued)

Potential Initiating Event for LaSalle Unit 2 (EPRI NP-2230 Nomenclature)	Initiator Type	Source	Applicable to LaSalle Unit 2 Study:
13) Turbine bypass or control valves cause increased pressure (closed)	Transient (general)	EPRI NP-2230	Yes
14) Recirculation control failure - increasing flow	Transient (general)	EPRI NP-2230	Yes
15) Recirculation control failure - decreasing flow	Transient (general)	EPRI NP-2230	Yes
16) Trip of one recirculation pump	Transient (general)	EPRI NP-2230	Yes
17) Trip of all recirculation pumps	Transient (general)	EPRI NP-2230	Yes
18) Abnormal startup of idle recirculation pump	Transient (general)	EPRI NP-2230	Yes
19) Recirculation pump seizure	Transient (general)	EPRI NP-2230	Yes
20) Feedwater - increasing flow at power	Transient (general)	EPRI NP-2230	Yes
21) Loss of feedwater heater	Transient (general)	EPRI NP-2230	Yes
22) Loss of all feedwater flow	Transient (general)	EPRI NP-2230	Yes
23) Trip of one feedwater pump (or condensate pump)	Transient (general)	EPRI NP-2230	Yes
24) Feedwater - low flow at power	Transient (general)	EPRI NP-2230	Yes

Table 1.1 Potential Internal Initiating Events for LaSalle Unit 2 (Continued)

	Potential Initiating Event for LaSalle Unit 2 (EPRI NP-2230 Nomenclature)	Initiator Type	Source	Applicable to LaSalle Unit 2 Study:
25)	Low feedwater flow during startup or shutdown	Transient (general)	EPRI NP-2230	No - startup/shutdown events are not considered
26)	High feedwater flow during startup or shutdown	Transient (general)	EPRI NP-2230	No - startup/shutdown events are not considered
27)	Rod withdrawal at power	Transient (general)	EPRI NP-2230	Yes
28)	High flux due to rod withdrawal at startup	Transient (general)	EPRI NP-2230	No - startup/shutdown events are not considered
29)	Inadvertent insertion of rod or rods	Transient (general)	EPRI NP-2230	Yes
30)	Detected fault in reactor protection system	Transient (general)	EPRI NP-2230	Yes
31)	Loss of offsite power	Transient (general)	EPRI NP-2230	Yes
32)	Loss of auxiliary power (loss of auxiliary transformer)	Transient (general)	EPRI NP-2230	Yes
33)	Inadvertent startup of HPCI/HPCS	Transient (general)	EPRI NP-2230	Yes
34)	Scram due to plant occurrences	Transient (general)	EPRI NP-2230	Yes
35)	Spurious trip via instrumentation, reactor protection system fault	Transient (general)	EPRI NP-2230	Yes

Table 1.1 Potential Internal Initiating Events for LaSalle Unit 2 (Continued)

Potential Initiating Event for LaSalle Unit 2 (EPRI NP-2230 Nomenclature)	Initiator Type	Source	Applicable to LaSalle Unit 2 Study?
36) Manual scram - no out of tolerance condition	Transient (general)	EPRI NP-2230	Yes
37) Cause unknown	Transient (general)	EPRI NP-2230	Yes
38) Partial loss of compressed gas system	Transient (special)	Nuclear Power Experience	Yes
39) Complete loss of service water system	Transient (special)	Nuclear Power Experience	Yes
40) Loss of an emergency AC bus	Transient (special)	Nuclear Power Experience	Yes
41) Partial loss of reactor vessel water level measurement system	Transient (special)	Nuclear Power Experience	Yes
42) Loss of an emergency DC bus	Transient (special)	Past PRAs	Yes
43) Small primary system LOCA inside containment	LOCA	Past PRAs	Yes
44) Medium primary system LOCA inside containment	LOCA	Past PRAs	Yes
45) Large primary system LOCA inside containment	LOCA	Past PRAs	Yes
46) Primary system LOCA outside containment	LOCA	Past PRAs	Yes

Table 1.1 Potential Internal Initiating Evencs for LaSalle Unit 2 (Concluded)

Potential Initiating Event for LaSalle Unit 2 (EPRI NP-2230 Nomenclature)	Initiator Type	Source	Applicable to LaSalle Unit 2 Study:
47) Reactor vessel rupture	LOCA	Past PRAs	Yes
48) Interfacing system LOCA	LOCA	Past PRAs	Yes

The operational experience search was supplemented by examining Nuclear Power Experience for events of significance in BWRs similar to LaSalle Unit 2. The plants included were the following:

1. LaSalle Unit 1,
2. Nine Mile Point Unit 2,
3. Brunswick Unit 1,
4. Brunswick Unit 2,
5. Peach Bottom Unit 2, and
6. Peach Bottom Unit 3.

At the time of the search, Nuclear Power Experience was up-to-date through April 1984, which means that plant events through approximately January 1984 were included. Therefore, the search of this source was both a check of the comprehensiveness of the EPRI NP-2230 event categories and an update of recent years experience. The search of Nuclear Power Experience resulted in four additional initiators being identified:

1. Partial loss of compressed gas system,
2. Loss of service water system,
3. Loss of an emergency AC bus, and
4. Partial loss of reactor vessel water level measurement system.

These initiators are also listed in Table 1.1.

Several PRAs of BWRs were also surveyed. The studies were:

1. Shoreham,[6]
2. Limerick,[7] and
3. Peach Bottom.[8]

From these studies, additional initiating events were identified which are also shown in Table 1.1. In general, the new events from these sources fell in the LOCA category.

The initiating events identified in Table 1.1 are historically divided into two groups: Transients and LOCAs. The transient category may be further divided into general and special initiators. Special initiators involve failures in support systems which result in both a plant trip or shutdown and adverse effects in one or more of the systems used to mitigate the accident. The LOCA category may be subdivided into LOCAs within containment, LOCAs outside containment, and interfacing system LOCAs. The divisions are shown in the table. In general, the transients are frequent enough such that they appear in the operational survey of U.S. BWRs. Although very small leakages in primary systems of BWRs have occurred, the sizes covered in the LOCA categories in Table 1.1 have not yet occurred. (An exception is the small LOCA, which has occurred when primary pump seal leakages are counted.)

The second step in the initiating event identification process involved a review of LaSalle Unit 2 information to eliminate any initiators from the first step which are not applicable to LaSalle or to this study and to identify any additional initiators specific to LaSalle. The following initiators from Table 1.1 were found not to be applicable:

Initiator	Reason Not Applicable
12. Turbine bypass fails open	Does not cause a plant trip at LaSalle Unit 2. However will require a controlled shutdown.
25. Low feedwater flow during startup or shutdown	Only events occurring while the plant is at or near full power are considered in this study.
26. High feedwater flow during startup or shutdown	Only events occurring while the plant is at or near full power are considered in this study.
28. High flux due to rod withdrawal at startup	Only events occurring while the plant is at or near full power are considered in this study.

In addition, all support systems for LaSalle Unit 2 were examined for failures which would cause a plant trip and which would adversely affect front-line systems (i.e., those systems that could mitigate the accident and which were included in the PRA model). Such a search is used to identify what are termed special initiators or special transients. The results of this examination are shown in Table 1.2. Twenty-two special initiators are identified in the table. However, six of these can be included as contributors to the loss of all feedwater and loss of condenser vacuum general initiators listed in Table 1.1. Another six can be subsumed by other special initiators as indicated in the table. The remaining ten special initiators are listed below:

1. Loss of 125 VDC bus 2A,
2. Loss of 125 VDC bus 2B,
3. Loss of 4160 VAC bus 241Y,
4. Loss of 4160 VAC bus 242Y,
5. Loss of instrument air (IA),
6. Loss of normal drywell pneumatic (IN, instrument nitrogen),
7. Loss of 100# drywell pneumatic,
8. Total loss of reactor vessel narrow range level instrumentation (false high level indications),
9. Loss of channels A and C of the reactor vessel narrow range level instrumentation (false high level indications), and
10. Loss of channels B and D of the reactor vessel narrow range level instrumentation (false high level indications).

Table 1.2 Summary of LaSalle Unit 2 Support System Special Initiator Search

Support System	Failure Causes a Reactor Scram?	Reason for Reactor Scram	Front-line or Support System Affected Transient	Special Initiator?	Covered by General Transient
<u>Electrical Power</u>					
250 VDC Bus 2	No	-	Reactor Core Isolation Cooling (RCIC)		No
125 VDC Bus 2A	High probability	High pressure in drywell or high reactor water level or FW or CDS trip	Condensate, ADS, RHR, LPCI, LPCS, RPS	Yes	No
Bus 2B	High probability	High pressure in drywell or high reactor water level or FW or CDS trip	FW, ADS, RHR, LPCI, RPS	Yes	No
Bus 2C	No	-	HPCS, DGs	No	-
4160 VAC Bus 241Y	High probability	CRD low pressure or high containment pressure	RHR, LPCI, LPCS, DGs, CRD, RBCCW, IN, RPS	Yes	No
Bus 242Y	High probability	CRD low pressure or high containment pressure	RHR, LPCI, DGs, RBCCW, CRD, RPS	Yes	No
Bus 243Y	No	-	HPCS	No	-
Bus 241X	Yes, if power is not reduced quickly	Loss of condenser vacuum	Service Water	Yes	Loss of condenser vacuum
Bus 242X	Yes, if power is not reduced quickly	Loss of condenser vacuum	Service water	Yes	Loss of condenser vacuum
6190 VAC Bus 251	Yes, if power is not reduced quickly	EHC instabilities	Condensate	Yes	Loss of all FW
Bus 252	Yes, if power is not reduced quickly	EHC instabilities	FW	Yes	Loss of all FW

See Note 1 for abbreviations.

Table 1.2 Summary of LaSalle Unit 2 Support System Special Initiator Search (Continued)

Support System	Failure Causes a Reactor Scram?	Reason for Reactor Scram	Front-line or Support System Affected Transient	Special Initiator?	Covered by General Transient
<u>Cooling Water</u>					
Service Water System (SWS)	Shutdown required	Recirculation pump seal failure	Condensate, FW	Yes	Loss of Condenser vacuum
Turbine Building Closed Cooling Water (TBCCW)	Shutdown required	Generator overheating	Condensate, FW	Yes	Loss of all FW
Reactor Building Closed Cooling Water (RBCCW)	Shutdown required	Recirculation pump seal failure	None	No	-
Core Standby Cooling System (CSCS)	No	-	RHR, LPCI, LPCS	No	-
<u>Compressed Gas</u>					
Service Air (SA)	No	-	None	No	-
Instrument Air (IA)	Shutdown required	CRD discharge volume high or control rods drift up	FW, MSIVs	Yes	No
Normal Drywell Pneumatic (IN or DPS)	Shutdown required	MSIV closure	ADS, MSIVs, SRVs	Yes	No
100# Drywell Pneumatic (IN or DPS)	Shutdown required	MSIV closure	MSIVs, SRVs	Yes	No
<u>Heating, Ventilation, and Air-Conditioning</u>					
Primary Containment	No	-	None	No	-
Auxiliary and Radwaste Area Ventilation	No	-	None	No	-
Turbine Building Area Ventilation	No	-	None	No	-

See note 1 for abbreviations.

Table 1.2 Summary of LaSalle Unit 2 Support System Special Initiator Search (Continued)

Support System	Failure Causes a Reactor Scram?	Reason for Reactor Scram	Front-line or Support System Affected Transient	Special Initiator?	Covered by General Transient
Control Room Area Ventilation					
Control Room	No	-	Operator environment	No	-
Auxiliary Electrical Equipment Room	No	-	Control and instrumentation	No	-
Engineered Safety Features					
Switchgear Heat Removal	No	-	Many	No	-
DG Facilities Ventilation	No (not normally running)	-	DGs	No	-
CSCS Equipment Areas Cooling	No (not normally running)	-	RHR, LPCI, LPCS	No	-
<u>Miscellaneous</u>					
All Reactor Vessel Narrow Range Level Instrumentation	Yes	RPV high level, low level does not degrade systems	FW, HPCS, RCIC, ADS, RHR, LPCS, RPS	Yes (Fail High Only, Reference Legs)	No
Channel A (Reference Leg)	0.5	FW control failure	FW fails, HPCS and RPS degraded	Yes	No, see A*C below*
Channel B (Reference Leg)	0.5	FW control failure	FW failed, ADS, RHR B&C, RCIC, and RPS degraded	Yes	No, see B*D below*
Channel C (Reference Leg)	No	-	HPCS, RPS	No	-
Channel D (Reference Leg)	No	-	ADS, RHR A, LPCS, RCIC, RPS	No	-

See note 1 for abbreviations.

Table 1.2 Summary of LaSalle Unit 2 Support System Special Initiator Search (Concluded)

Support System	Failure Causes a Reactor Scram?	Reason for Reactor Scram	Front-line or Support System Affected Transient	Special Initiator?	Covered by General Transient
Channels A*B or A*D or C*B or C*D (Reference Legs)	Yes	RPS scram on high level	FW failed, HPCS, RCIC, ADS, and either LPCS*RHR A or RHR B*C degraded	Yes	No, see A*C or B*D below*
Channels A*B or C*D (Variable Legs)	Yes	RPS scram on low level	FW, HPCS, RCIC, ADS, RHR, and RPS	No Auto Actuation of systems	-
Channels A*C (Reference Legs)	0.5	FW control system	FW, HPCS failed, RPS degraded	Yes	No[2]
Channels B*D (Reference Legs)	0.5	FW control system	FW, RCIC, ADS, LPCS, RHR failed, RPS degraded	Yes	No[2]

Note 1 ADS = Automatic Depressurization system, CDS = Condensate system, CRD = Control Rod Drive system, CSCS = Core Standby Cooling system, DGs = Diesel Generators, EHC = Electro-Hydraulic Control system, FW = Feedwater system, HPCS = High Pressure Core Spray system, IA = Instrument Air system, IN (DPS) = Instrument Nitrogen (or Drywell Pneumatic) system, LPCI = Low Pressure Coolant Injection system, LPCS = Low Pressure Core Spray system, MSIV = Main Steam Isolation Valves, RBCCW = Reactor Building Closed Cooling Water system, RCIC = Reactor Core Isolation Cooling system, RHR = Residual Heat Removal system, RPS = Reactor Protection system, SA = Service Air system, SRV = Safety Relief Valves, SWS = Service Water system, TBCCW = Turbine Building Closed Cooling Water system. All these systems are described in detail in Volume 6 of this report.

Note 2 Rather than model all of these as separate initiators, A*C and B*D were used used to represent the rest. These two affect the minimum and maximum systems of all the possible combinations that can also result in reactor scram. They represent, therefore, the minimum and maximum impact on the plant of instrument line failures that also fail responding systems. These initiators were not significant for the internal events analysis but were included so that they could evaluated in the seismic analysis where multiple pipe failures were thought to be more likely. For LaSalle, pipe failures were unlikely even in seismic events and so these initiators do not show up as important in the overall analysis.

All of these types of special initiators are covered by the list in Table 1.1. However, the special initiator search resulted in specific events being identified. For example, under the general event concerning loss of an emergency DC bus, the applicable events for LaSalle Unit 2 are losses of buses 2A and 2B, but not bus 2C.

The special initiator search was conducted in the following manner: (1) the initiating event analyst examined the reactor trip system and determined all parameters whose variation could result in a plant trip, (2) the systems analysts, in the process of modeling the front-line and support systems, looked for direct and indirect connections to balance of plant systems and the parameters which could potentially result in plant trip and where failures of systems or components could degrade the modeled systems and simultaneously result in a plant trip, (3) some of the initiators were evaluated using the LaSalle simulator to see if the simulator would predict a plant trip, (4) all potential special initiators were discussed with system engineers at the architect/engineer (A&E) for LaSalle, Sargent & Lundy, and the PRA team's analysis was evaluated to determine if the trip would actually occur for each special initiator, and (5) the systems and initiating event analysts evaluated whether or not the initiator should be represented in the model as a special initiator or included in some previously defined class.

The LOCA search for LaSalle Unit 2 was divided into three categories: LOCAs inside containment, LOCAs outside containment, and interfacing system LOCAs. For LOCAs within containment, the following systems had piping which was a part of the primary system:

1. Recirculation loops,
2. Reactor Water Cleanup (RWCU) within the containment,
3. Residual Heat Removal (RHR) (except for containment spray) within the containment,
4. Reactor Core Isolation Cooling (RCIC) piping within the containment,
5. Standby Liquid Control (SLCS) piping within the containment,
6. Control Rod Drive (CRD) piping within the containment,
7. High Pressure Core Spray (HPCS) piping within the containment,
8. Main steamlines within the containment, and
9. RCIC/RHR steamline within the containment.

Distinctions were made for the Small, Medium, and Large LOCA cases. The Small LOCA case was defined as a liquid or steam pipe break small enough such that RCIC would be sufficient to keep the core covered. NEDO-24708A[9] indicates that, for BWRs with primary system characteristics similar to LaSalle, liquid pipe breaks up to and including 0.005 ft^2 (1-inch diameter) and steam pipe breaks up to and including 0.1 ft^2 (4-inch diameter) can be handled by RCIC. A Large LOCA was defined as a steam or liquid pipe break large enough to cause rapid enough vessel depressurization such that the low pressure injection systems could operate successfully to prevent core damage. Based on information presented in NEDO-24708A,[9] steam and liquid pipe break sizes greater

than or equal to 0.3 ft^2 (8-inch diameter) were considered to be Large LOCAs. Finally, Medium LOCAs covered liquid pipe breaks of the range >0.005 to <0.3 ft^2 and steam pipe breaks of the range >0.1 to <0.3 ft^2. Transient-Induced LOCAs resulting from stuck open safety relief valves (SRVs) which discharge to the suppression pool through the SRV discharge lines and not directly into the drywell were also divided into Small (1 SRV stuck open), Medium (2 SRVs stuck open), and Large (3 or more SRVs stuck open). Transient sequences having transient-induced LOCAs were evaluated separately from standard LOCAs.

LOCAs outside of the drywell were divided into three separate cases: steamline breaks, feedwater or condensate piping breaks, and interfacing system LOCAs. For steamline breaks, closure of the main steam isolation valves (MSIVs) is required in order to isolate the break from the reactor vessel. The size of the break is not an important consideration, assuming that MSIV closure occurs. Therefore, only one initiating event was used to represent all steamline breaks outside of the drywell. Accident sequences consisting of steamline breaks followed by failure of the inboard and outboard MSIVs to close and failure of sufficient systems such that core damage may result are probabilistically negligible. Similarly, a steamline break outside the primary containment but before the outboard MSIV with failure of the inboard MSIV and sufficient systems such that core damage may result is also negligible. The flood analysis, reported in Volume 10 of this report, considered the effects of unmitigated steamline breaks and found them to be negligible.

The second type of LOCA considered outside of the containment was a feedwater or condensate piping break. Because the feedwater lines penetrating the containment each have two check valves in series, isolation of a pipe break from the reactor vessel is automatic. A single initiator was used to represent ruptures in the feedwater or condensate systems. The flood analysis, reported in Volume 10 of this report, considered the effects of pipe breaks in all systems and found them to be negligible except for two breaks in the service water piping.

Interfacing system LOCAs are the third type of LOCA outside of the containment. A comprehensive search was conducted for piping penetrating the containment and connecting to the primary system. Water lines with diameters of one-inch or larger were considered, as were steamlines with diameters of four-inches or larger. The results are shown in Table 1.3. The table indicates the containment isolation valves in each line, and whether or not the piping outside of the containment is rated for high pressure. A line was considered to have potential for an interfacing system LOCA if piping beyond the isolation valves is rated only for low pressure or if there are fewer than two isolation valves. The following lines were considered to have the potential for interfacing system LOCAs:

1. RHR suction from recirculation loop A for shutdown cooling (one),
2. RHR injection for shutdown cooling to recirculation loop A and B (two),

Table 1.3 Interfacing LOCA Piping Survey

Piping Description	Pipe No. and Size	Isolation Valves Inside Drywell	Isolation Valves Outside Drywell	High Pres. Piping Beyond?	Possible Significant Interfacing System LOCA?*
Water Piping (≥1")					
1) RHR suction from recirc. Loop A	2RR18A-20	1 MOV	1 MOV	Yes (treated as No)	Yes
2) RHR S/D cooling return to recirc. Loop A	2RR07AA-12	1 check or 1 MOV (2" line)	1 MOV	Yes (treated as No)	Yes
3) RHR S/D cooling return to recirc. Loop B	2RR07AB-12	1 check	1 MOV	Yes (treated as No)	Yes
4) RWCU suction from recirc. Loops A and B	2RT01B-6	1 check	1 check	Yes	No
5) SLCS injection line	2SC02B-1 1/2	1 check	1 check	Yes	No
6) RCIC injection line	2RI24B-6	1 check	1 check	Yes	No
7) RHR reactor vessel injection line (LPCI) (There are three lines)	2RH40BA-12 (2RH40BB-12) (2RH53B-12)	1 check	1 MOV	Yes (treated as No)	Yes
8) LPCS	2LP02B-12	1 check	1 MOV	Yes (treated as No)	Yes
9) HPCS injection line	2HP02D-12	1 check	1 MOV	Yes	No
10) FW inlet line (There are two lines)	2FW02BA-24 (2FW02BB-24)	1 check	1 check and 1 MOV	Yes	No
11) CRD drive water insert line (185 of them). Note - there are also 185 3/4" withdraw lines with similar configurations	2RD56A-1 etc.	-	1 manual	Yes	Yes

* If there are two or more isolation valves plus high pressure piping beyond the valves (outside the drywell), then the line was not considered to have significant interfacing system LOCA potential. Random failure of high pressure piping plus failure of two or more valves to isolate the break was considered to be negligible compared with cases where low pressure piping would be subjected to high pressure.

Table 1.3 Interfacing LOCA Piping Survey (Concluded)

Piping Description	Pipe No. and Size	Isolation Valves: Inside Drywell	Isolation Valves: Outside Drywell	High Pres. Piping Beyond?	Possible Significant Interfacing System LOCA?*
Steamline Piping (≥4")					
12) Main steamline (four of them)	2MS01EA-26 (2MS01EB-26) (2MS01EC-26) (2MS01ED-26)	1 pneumatic (MSIV)	1 pneumatic (MSIV)	Yes	No
13) RCIC turbine inlet steamline	2RI01B-4	1 MOV	1 MOV	Yes	No
14) RCIC/RHR heat exchanger steamline	2RI01A-10	1 MOV	1 MOV	Yes	No

* If there are two or more isolation valves plus high pressure piping beyond the valves (outside the drywell), then the line was not considered to have significant interfacing system LOCA potential. Random failure of high pressure piping plus failure of two or more valves to isolate the break was considered to be negligible compared with cases where low pressure piping would be subjected to high pressure.

3. Low Pressure Coolant Injection (LPCI) and Low Pressure Core Spray (LPCS) injection lines (four), and
4. CRD drive water insert lines (185).

The RHR, LPCI, and LPCS systems are low pressure systems; but, at LaSalle, the piping in these systems was the same as that used for the high pressure systems. The piping itself, therefore, is high pressure piping. However, the piping was treated as low pressure piping in evaluating the potential for interfacing LOCAs in these systems.

1.4.2 Initiating Event Categorization

In order to minimize the event tree development efforts, the initiating events were grouped into categories. Grouping was accomplished by examining the following effects on the plant of each initiating event:

1. Trip signals expected following the initiator,
2. Plant systems required to respond to the initiator, and
3. Effect of the initiator on the availability of plant systems required to respond.

Initiators that resulted in a similar plant response were combined into a group. The final initiating event categories for LaSalle Unit 2 are shown in Table 1.4. Twenty-two categories were used. The unique characteristics of each category are also listed in the table.

1.4.3 Anticipated Transient Without Scram (ATWS) Events

All of the initiators that lead to a plant trip, with the exception of two, can also occur with subsequent failure to scram. The events that can not have a subsequent failure to scram were initiating events 35 and 36, both of which involve scraming of the reactor when no out of tolerance conditions exist. In such a case, successful reactor scram has occurred by definition of the event and failure to scram is not possible. For this analysis, a separate evaluation of the turbine trip with turbine bypass available category with these two events removed was not performed. While this initiator category will have a frequency for ATWS events that is slightly high, this difference is well within the uncertainty bounds of the initiator frequency and does not significantly affect the final results.

1.4.4 Quantification of Initiating Event Categories

Quantification of the LaSalle Unit 2 initiating event categories listed in Table 1.4 can be divided into four separate cases. The four cases are:

Table 1.4 Initiating Event Categories for LaSalle Unit 2

Initiating Event Category	Initiating Events Included	EPRI NP-2230 Transient Designator	Comments
Transients (General)			
1) Turbine trip with turbine bypass available	1) Electric load rejection	BWR1	Main steam stop valves closure is expected to cause a reactor trip for these initiating events. None of the safety systems are affected.
	3) Turbine trip	BWR3	
	14) Recirculation control failure - increasing flow	BWR14	
	15) Recirculation control failure - decreasing flow	BWR15	
	16) Trip of one recirculation pump	BWR16	
	17) Trip of all recirculation pumps	BWR17	
	18) Abnormal startup of idle recirculation pump	BWR18	
	19) Recirculation pump seizure	BWR19	
	20) Feedwater - increasing flow at power	BWR20	
	21) Loss of Feedwater heater	BWR21	
	27) Rod withdrawal at power	BWR27	
	29) Inadvertent insertion of rod or rods	BWR29	
	30) Detected fault in reactor protection system	BWR30	
	33) Inadvertent startup of HPCI/HPCS	BWR33	
	34) Scram due to plant occurrences	BWR34	
	35) Spurious trip via instrumentation, reactor protection system fault	BWR35	
	36) Manual scram - no out of tolerance condition	BWR36	
	37) Cause unknown	BWR37	

Table 1.4 Initiating Event Categories for LaSalle Unit 2 (Continued)

Initiating Event Category	Initiating Events Included	EPRI NP-2230 Transient Designator	Comments
2) Turbine trip with turbine bypass unavailable	2) Electric load rejection with turbine bypass valve failure	BWR2	Similar to category 1, but with failure of the turbine bypass. It is assumed that initiating events 10 and 13 result in loss of the turbine bypass.
	4) Turbine trip with turbine bypass valve failure	BWR4	
	10) Pressure regulator fails closed	BWR10	
	13) Turbine bypass or control valves cause increased pressure (closed)	BWR13	
3) Total main steam isolation valve closure	5) Main steam isolation valve closure	BWR5	Main steam isolation valve closure is expected to cause a reactor trip. It is assumed that reopening of the valves is not possible. In such a case the turbine-driven feedwater pumps and the turbine bypass are lost. It is assumed that initiating events 6, 7, and 9 lead to closure of all main steam isolation valves.
	6) Inadvertent closure of one main steam isolation valve	BWR6	
	7) Partial main steam isolation valve closure	BWR7	
	9) Pressure regulator fails open	BWR9	
4) Loss of normal condenser	8) Loss of normal condenser vacuum	BWR8	Low condenser vacuum is expected to result in a turbine trip and a feedwater trip. Reactor Trip occurs as a result of main steam stop valve closure. The motor-driven feedwater pump starts automatically upon feedwater trip. The turbine-driven feedwater pumps and turbine bypass are lost.
5) Total loss of feedwater	22) Loss of feedwater flow	BWR22	Loss of all feedwater, including the motor-driven pump, occurs. No recovery of feedwater is assumed to be possible. Reactor trip occurs as a result of [illegible] level in the reactor [illegible]
	24) Feedwater - low flow at power	BWR24	
	48) Feedwater or condensate LOCA outside containment	None	

Table 1.4 Initiating Event Categories for LaSalle Unit 2 (Continued)

Initiating Event Category	Initiating Events Included	EPRI NP-2230 Transient Designator	Comments
6) Trip of one feedwater or condensate pump	23) Trip of one feedwater pump (or condensate pump)	BWR23	Similar to category 5, but the remaining two feedwater trains are assumed to be available.
7) Inadvertent opening of a safety-relief valve (stuck)	11) Inadvertent opening of a safety/relief valve (stuck)	BWR11	No immediate automatic reactor trip occurs. Manual trip may occur; if not, then automatic trip will occur later as a result of high containment pressure.
8) Loss of offsite power	31) Loss of offsite power	BWR31	Total loss of all offsite electric power sources. Reactor trip occurs due to main steam stop valve closure.
	32) Loss of auxiliary power (loss of auxiliary transformer)	BWR32	
9) Loss of 125 VDC bus	42) Loss of 125 VDC Bus 2A	None	Reactor trip from high drywell pressure as a result of drywell chiller isolation, loss of FW or CDS from loss of FW control, and partial losses of ADS, RHR, and LPCI occur. Also, for bus 2A LPCS is lost.
	42) Loss of 125 VDC Bus 2B	None	
10) Loss of 4160 VAC bus	40) Loss of 4160 VAC Bus 241Y	None	Reactor trip from CRD low press or high drywell press. Loss of 1 drywell chiller, 1 RBCCW pump, 1 CRD pump, 1 train RPS, and 1 IN compressor (241Y). Closure of in/outboard isolation valves. Partial losses of RHR, LPCI, and DGs. Also, for bus 241Y LPCS is lost.
	40) Loss of 4160 VAC Bus 242Y	None	
11) Loss of instrument air	38) Loss of instrument air	None	Reactor trip from CRD high discharge volume level or control rod drift. Feedwater is lost, and outboard MSIVs drift closed.

Table 1.4 Initiating Event Categories for LaSalle Unit 2 (Continued)

Initiating Event Category	Initiating Events Included	EPRI NP-2230 Transient Designator	Comments
12) Loss of drywell pneumatic	38) Loss of normal drywell pneumatic	None	Shutdown is required by the technical specifications. Loss of ADS and inboard MSIVs. Partial loss of SRVs.
13) Loss of 100# drywell pneumatic	39) Loss of 100# drywell pneumatic	None	Similar to category 12, but with no effect on ADS.
14) Complete loss of reactor vessel narrow range level instrumentation (false high level indications)	41) Complete loss of reactor vessel narrow range level instrumentation (false high level indications)	None	Reactor trip on either high or low level in the reactor vessel. Loss of HPCS and RCIC.
15) Loss of Channels A and C or C and D of reactor vessel narrow range level instrumentation (false high level indications)	41) Loss of channels A and C of reactor vessel narrow range level instrumentation (false high level indications)	None	Reactor trip on either high or low level in the reactor vessel. Loss of HPCS.
	41) Loss of channels B and D of reactor vessel narrow range level instrumentation (false high level indications)	None	Reactor trip on either high or low level in the reactor vessel. Loss of RCIC, ADS, LPCS, and RHR.
LOCA (Inside Containment)			
16) Small LOCA inside containment (≤ 0.005 ft^2 for liquid, ≤ 0.1 ft^2 for steam)	43) Small LOCA inside containment (≤ 0.005 ft^2 for liquid, ≤ 0.1 ft^2 for steam)	None	Reactor trip as a result of high drywell pressure or low level in the reactor vessel. RCIC is capable of providing adequate coolant makeup.
17) Medium LOCA inside containment (>0.005 to <0.3 ft^2 for liquid, >0.1 to <0.3 ft^2 for steam)	44) Medium LOCA inside containment (0.005 to 0.3 ft^2 for liquid, 0.1 to 0.3 ft^2 for steam)	None	Similar to category 16. However, RCIC is not adequate for coolant makeup.
18) Large LOCA inside containment (≥ 0.3 ft^2)	45) Large LOCA inside containment (≥ 0.3 ft^2)	None	Reactor trip as a result of high containment pressure or low level in the reactor vessel. The primary coolant system depressurizes rapidly and remains depressurized.

Table 1.4 Initiating Event Categories for LaSalle Unit 2 (Concluded)

Initiating Event Category	Initiating Events Included	EPRI NP-2230 Transient Designator	Comments
19) Reactor vessel rupture	47) Reactor vessel rupture	None	Reactor trip as a result of high containment pressure or low level in the reactor vessel. The break size is large enough such that coolant level cannot be maintained in the reactor vessel. Such an event is assumed to lead directly to core damage.
<u>LOCA (Outside Containment)</u>			
20) Steamline LOCA outside containment	48) Steamline LOCA outside containment	None	Reactor trip as a result of main steam isolation valve closure. Turbine-driven feedwater and the turbine bypass are lost.
<u>LOCA (Interfacing System, Outside Containment)</u>			
21) RHR or LPCS LOCA outside containment without isolation	48) RHR suction line for recirculation loop A, LOCA outside containment without isolation	None	Non-isolatable LOCA in large piping (12 to 20 inch diameter). Reactor trip as a result of low level in the reactor vessel. Since isolation is not possible, the coolant level cannot be maintained in the reactor vessel. Such an event is assumed to lead directly to core damage.
	48) RHR shutdown cooling return line (two) to recirculation loop, LOCA outside containment without isolation	None	
	48) LPCI or LPCS injection line (four), LOCA outside containment without isolation	None	
22) CRD LOCA outside drywell without isolation	48) CRD drive water insert line (185), LOCA outside drywell without isolation	None	Non-isolatable LOCA in small pipe (1-inch diameter). Manual trip.

1. General transients,
2. Special transients (or initiators),
3. LOCAs, and
4. Interfacing system LOCAs.

Screening values for the initiating event categories are discussed below. A general discussion of the whole data analysis, screening, and final values for the initiating events is presented in Volume 5 of this report on Parameter Estimation Analysis and Screening Human Reliability Analysis.

Frequencies for the LaSalle Unit 2 general transient categories were obtained from the data presented in Reference 4. This source, which is an update of EPRI NP-2230,[2] is based on the operating experience of 25 U.S. commercial BWR nuclear plants and covers 228 reactor years of experience up through the end of 1983. Events occurring during the first year of commercial operation were included, but events occurring at low power (less than 26%) were excluded. The event grouping in Reference 4 is identical to that used in EPRI NP-2230, so each category frequency was obtained by combining the appropriate data for the contributing EPRI events. The resulting frequencies for the LaSalle Unit 2 general transient categories are shown in Table 1.5.

Special transients (or initiators) were evaluated on generic bases. The frequency for loss of a DC or an AC bus was taken from NUREG-4550.[10] Both types of events were assigned a frequency of 5E-03/yr. The other special initiators were assigned a frequency of 3.0E-03/year. This value was obtained by calculating a 50% Chi-Square estimate based on no events in 275 BWR reactor years of operation.*

LOCAs inside containment were assigned frequencies based on the data base developed for the National Reliability Evaluation Program (NREP).[11] The Small LOCA frequency is 0.03/year, the Medium LOCA frequency is 3.0E-03/year, and the Large LOCA was assigned a frequency of 3.0E-04/year. Finally, the reactor vessel rupture frequency was assumed to be 3.0E-07/year, based on WASH-1400.[8]

The steamline LOCA outside containment was quantified based on no occurrences in 275 BWR reactor years of experience, similar to what was done for several of the special initiators. The 50% Chi-Square estimate is 3.0E-03/year.

Interfacing system LOCAs outside containment were quantified based on the LaSalle Unit 2 design. All seven of the RHR and LPCS lines included in this category include two isolation valves, both of which must fail in

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* At the end of 1983, approximately 275 BWR reactor years of operation had occurred (neglecting atypical plants such as Big Rock Point and Dresden Unit 1).

Table 1.5 Frequencies of LaSalle Unit 2 Initiating Event Categories

Initiating Event Category	Frequency (Yr^{-1}) Screening/Final	Reference for Screening Value
<u>Transients (General)</u>		
1) Turbine trip with turbine bypass available	5.0/4.5	Reference 4
2) Turbine trip with turbine bypass unavailable	0.5/0.52	Reference 4
3) Total main steam isolation valve closure	0.7/0.61	Reference 4
4) Loss of normal condenser vacuum	0.4/0.41	Reference 4
5) Total loss of feedwater	0.6/0.6	Reference 4
6) Trip of one feedwater or condensate pump	0.2/0.2	Reference 4
7) Inadvertent opening of a safety-relief valve (stuck)	0.2/0.14	Reference 4
8) Loss of offsite power	0.1/0.096	Reference 4
<u>Transients (Special)</u>		
9) Loss of 125 VDC bus (T9A-Bus A or T9B-Bus B)	5.0E-03/5.0E-03	Reference 10
10) Loss of 4160 VAC bus (T101-Bus 241Y or T102-242Y)	5.0E-03/5.0E-03	Reference 10
11) Loss of Instrument Air	3.0E-03/3.0E-03	No events in 275 BWR reactor years. 50% Chi Square estimate.

Table 1.5 Frequencies of LaSalle Unit 2 Initiating Event Categories (Continued)

Initiating Event Category	Frequency (Yr^{-1}) Screening/Final	Reference for Screening Value
12) Loss of drywell pneumatic	3.0E-03/3.0E-03	No events in 275 BWR reactor years. 50% Chi Square estimate.
13) Loss of 100# drywell pneumatic	3.0E-03/4.4E-03	No events in 275 BWR reactor years. 50% Chi Square estimate.
14) Complete loss of reactor vessel narrow range level instrumentation (false high level indication	3.0E-03/4.0E-14	No events in 275 BWR reactor years. 50% Chi Square estimate.
15) Loss of two channels of reactor vessel narrow range level instrumentation (false high level indications) (T15A-A and C or T15B-B and D)	3.0E-03/2.0E-07	No events in 275 BWR reactor years. 50% Chi Square estimate.
<u>LOCA (Inside Containment)</u>		
16) Small LOCA inside containment ($\leq$0.005 ft^2 for liquid, $\leq$0.1 ft^2 for steam)	0.03/0.03	Reference 11
17) Medium LOCA inside containment (0.005 to 0.3 ft^2 for liquid, 0.1 to 0.3 ft^2 for steam)	3.0E-04/3.0E-04	Reference 11
18) Large LOCA inside containment ($\geq$0.3 ft^2)	3.0E-04/1.0E-04	Reference 11
19) Reactor vessel rupture	<3.0E-07	Reference 8

Table 1.5 Frequencies of LaSalle Unit 2 Initiating Event Categories (Concluded)

Initiating Event Category	Frequency (Yr^{-1}) Screening/Final	Reference for Screening Value
<u>LOCA (Outside Containment)</u>		
20) Steamline LOCA outside containment	3.0E-03	No events in 275 BWR reactor years. 50% Chi Square estimate.
<u>LOCA (Interfacing System, Outside Containment)</u>		
21) RHR or LPCS LOCA outside containment without isolation	2.0E-04	LaSalle-specific analysis
22) CRD LOCA outside of drywell without isolation	3.0E-07	LaSalle-specific analysis

order for low pressure piping beyond the valves to be exposed to high pressure. Six of the lines include a check valve inside containment and an MOV outside containment, while the seventh contains two MOVs. Each case is analyzed below.

For a check valve and MOV combination, both valves cannot be open or have experienced a catastrophic internal leakage at the time of plant startup. (In such a case the open flowpath to the low pressure piping would be noticed.) Therefore, failure of both valves must occur as a result of either simultaneous failure or failure of one combined with unavailability of the other. Simultaneous failure was assessed to be negligible. Therefore two cases remained: failure of the check valve combined with (previous) unavailability of the MOV, and failure of the MOV combined with (previous) unavailability of the check valve. Internal catastrophic leakage of a check valve occurs with a frequency of 5.0E-07/hr. As an initiator, the frequency is:

(5.0E-07/hr) * (8760 hr/yr) = 4.4E-03/yr.

Unavailability of the MOV results from catastrophic internal leakage or failure of the MOV to have closed prior to startup but with an indication that the valve did close. (Spurious operation of the MOV was neglected because normal leakage past the check valve will result in the MOV being exposed to high pressure on one side. With such a pressure differential, the motor on the MOV is incapable of opening the valve.) The MOV catastrophic internal leakage frequency is 5.0E-07/hour. The MOV is tested every 18 months, so the average unavailability over this period is:

(0.5) * (5.0E-07/hr) * (13140 hr) = 3.3E-03.

Unavailability resulting from failure to close but with a closed indication was quantified by modifying the fail to operate value of 3E-03/demand. Reference 12 indicates that 2.5% of MOV failures to operate were cases in which the valve did not operate but there was an indication that correct operation did occur. Therefore, unavailability resulting from this failure mode is:

(1 demand) * (3E-03/demand) * (0.025) = 7.5E-05.

Combining the check valve initiator and the MOV unavailabilities results in:

(4.4E-03/yr) * (3.3E-03 + 7.5E-05) = 1.5E-05/yr.

The other case considered for the check valve and MOV combination was check valve unavailability combined with MOV failure as the initiator. The initiator frequency is:

(5.0E-07/hr) * (8760 hr/yr) = 4.4E-03/yr.

Check valve unavailability results from catastrophic internal leakage. The average unavailability over the 18-month test period is:

(0.5) * (5.0E-07/hr) * (13140 hr) = 3.3E-03.

Combining the MOV initiator with the check valve unavailability results in:

(4.4E-03/yr) * (3.3E-03) = 1.5E-05/yr.

Combining the above two cases for the check valve and MOV combination results in:

1.5E-05/yr + 1.5E-05/yr = 3.0E-05

for each of the three lines with this valve combination.

The other line included in this interfacing system LOCA category has two MOVs. [illegible] astrophic internal leakage initiator frequency is:

(5.0E-0 [illegible] (8760 hr/yr) = 4.4E-03/yr.

Un [illegible] a MOV is:

[illegible] 05 = 3.4E-03.

Combin [illegible] itiator with the unavailability and accounting for each MOV possibly being the initiator results in:

(2) * (4.4E-03/yr) * (3.4E-03) = 1.5E-05/yr.

The combined frequency for the seven large piping interfacing system LOCAs is then:

(6) * (3.0E-05) + (1.5E-05) = 2.0E-04/yr.

Another interfacing system LOCA category involves the CRD drive water lines. An interfacing system LOCA would occur if the piping ruptured and the manual valve (on each line) were not closed. No CRD drive water line ruptures have occurred while at power, so the frequency of such an event was based on a 50% Chi-Square estimate based on no events in 275 BWR reactor years. The result is 3.0E-03/yr. It was assumed that because of the significant time involved before core uncovery would occur, mechanical failure of the manual valve dominates any human errors. A manual valve failure to operate probability is 1.0E-04/demand. Combining this failure with the initiator frequency results in 3.0E-07/yr.

1.5 References

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2.0 ACCIDENT SEQUENCE DELINEATION

2.1 Introduction

The evaluation of the risk to the public imposed by operation of LaSalle Unit 2 requires calculation of the frequency and magnitude of radioactive releases to the environment due to potential core damage accidents. This calculation is done in a three stage process.

In the first stage, the accident sequences which can lead to core damage are delineated and evaluated. This is called a Level I analysis and includes both internal and external events. Core damage is defined as the release of radioactive fission products from the fuel. The delineation of the accident sequences is usually provided in the form of event trees. These event trees define the different sequences of events, usually plant system successes and failures, that can lead to core damage. The event trees are supplemented by system fault trees. The fault trees delineate the different ways component, human, and phenomenological failures or occurrences can result in system failure. By solving the fault trees and then combining them to form a particular sequence, we can qualitatively and quantitatively evaluate the thousands of combinations of component, human, and phenomenological events that can result in the sequence. Each unique combination of failures that can lead to a sequence is termed a cut set for that sequence.

In the second stage, the progression of each accident is evaluated and the characteristics of the source term (i.e., the amount of fission products released to the environment) are calculated. In order to do this, the accident sequence cut sets are first regrouped into plant damage states (PDSs). Each PDS presents unique initial and boundary conditions to the evaluation of the accident progression from the time of core damage to the termination of the accident. The various possible accident progressions are then evaluated using an accident progression event tree which evaluates the conditional probability of the accident progressions and groups the progressions into accident progression bins, which present unique initial and boundary conditions to the source term analysis. A parametric code is used to evaluate the magnitude, timing, and energy of the release to the environment for each accident progression bin. This parametric code uses input from more detailed accident sequence analysis codes and expert judgement (both from independent experts and from in-house experts, depending on the issue) in order to extrapolate and interpolate estimates of the source terms for all the accident progression bins coming out of the accident progression analysis. This accident progression and source term calculation form a Level II analysis.

In the third stage, the accident progressions are partitioned by similar source term characteristics and consequences are calculated for each partition using a consequence analysis code. The final risk to the public is then calculated by combining the frequencies of the PDSs, with

the conditional probabilities of the accident progression bins, the conditional probabilities of the partition groups, and the magnitudes of the consequences. This consequence and risk calculation form a Level III analysis.

The Level I analysis defining the initial set of accident sequences to be analyzed in this PRA is presented in this chapter and the final results of the analysis are presented in Volume 2 of this report. The Level II/III analyses results are reported in a series of reports issued by the Phenomenology and Risk Uncertainty Evaluation Program (PRUEP) as described in the foreword.[1]

One of the major purposes of the RMIEP program was to develop methods for the integrated evaluation of all Level I initiating events. So while only one set of event trees will be presented in this chapter, these trees will be used in four different analyses: internal events, seismic, fire, and flood. The fault trees to be used with these event trees have been expanded from the usual level of detail used in the internal events analysis to include information necessary to perform an integrated evaluation of the internal and external events. Each of these analyses are discussed in detail in separate volumes of this report (Volumes 3, 8, 9, and 10, respectively).

2.2 Overview of Evaluation Process

In this chapter, functional and systemic event trees will be defined where the accident sequence is followed until the end state is resolved into no core damage or core damage. No core damage, or success states, are those in which sufficient systems work in order to prevent core damage. This may mean only core heat removal is successful or both core and containment heat removal are successful, depending on the particular systems being used.

For some sequences, in which core heat removal is successful but containment heat removal fails, core damage does not result directly from the system failures but from phenomenological events in the containment which can possibly lead to failure of the core heat removal function and result in subsequent core damage. The event trees will include the feedback effects on the core heat removal systems as a result of the containment phenomenology in order to predict if core damage will occur given failure of the containment heat removal systems and the subsequent containment phenomenology.

The end states of the accident sequences are either: (1) success-no core damage but containment may or may not have failed, (2) core damage without direct containment failure or (3) core damage with containment failure (either controlled release - venting or uncontrolled release - structural failure).

2.3 Core Damage Functio.al Event Trees

The functional core damage event trees are presented in this section. The functional event trees delineate the general plant response to LOCAs, anticipated transients, and anticipated transients without scram (ATWS). The delineation is presented in terms of success or failure of safety functions required to mitigate the transient or LOCA.

For each safety function identified in the functional event trees, the systems available to perform the function were identified. The success criteria for each system was also defined. These steps provided the information necessary to delineate the systemic event trees and the results are presented in Sections 2.4 and 2.5, respectively.

Systemic event trees for LOCAs, anticipated transients with :ram, and ATWS are developed in Section 2.6. The interaction between articular initiating event and mitigating system is modeled by i .uding the initiating event in the system fault tree in the appropriate place so that the effect of its occurrence is properly propagated in the tree and fails the appropriate components in the system.

2.3.1 LOCA Functional Event Tree (L)

The LOCA functional event tree represents the general plant response to a loss-of-coolant accident. To mitigate a LOCA (L), it is necessary to shutdown the nuclear reaction (reactor subcriticality, RS), protect the containment from early overpressurization by condensing steam released from the vessel (early containment overpressure protection or vapor suppression, VP), keep the fuel covered with water and remove decay heat (core coolant makeup, CCM), and protect the containment from late overpressurization by transferring heat from the containment to the ultimate heat sink (containment heat removal, CHR). Another function usually considered is removal of radioactive nuclides from the containment before or following core damage. However, in this study, this function will be addressed in the accident progression event tree in the Level II/III analysis.

The four functions discussed above and their interactions following a LOCA are shown in the event tree illustrated in Figure 2.1. Each of the functions is discussed in more depth in the following paragraphs. In constructing this event tree, potential interactions between functions are considered.

Reactor Subcriticality (RS)

Following a LOCA, it is necessary to limit the core heat generation by shutting down the clear reaction. This is normally done by inserting the control rods into the core. Backup systems and procedures are available for reducing core power given a failure to insert the control rods.

2.3 Core Damage Functional Event Trees

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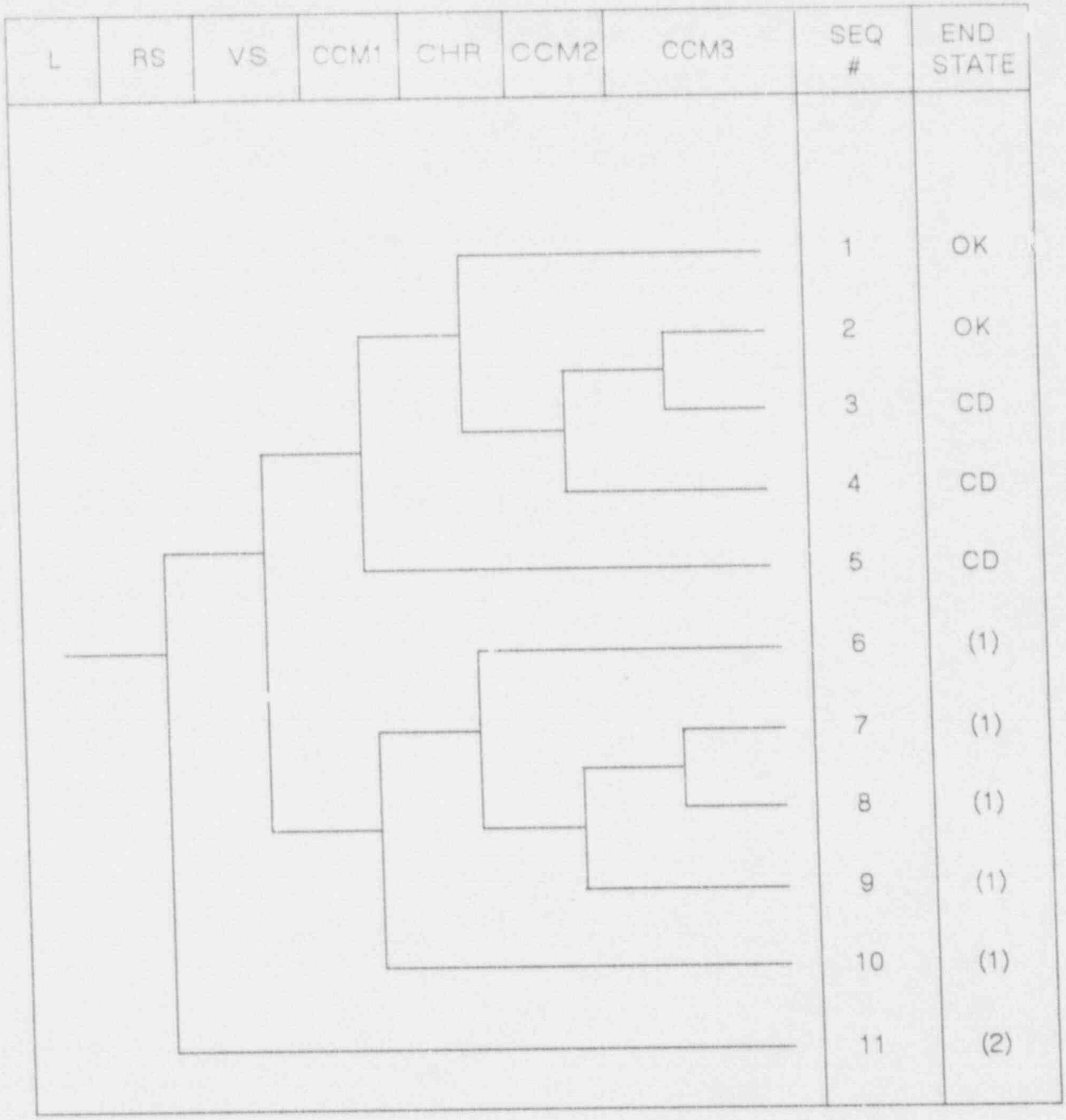


(1) Sequence proceeds similar to VSS success except much faster. CHR success may be unlikely.

(2) Transfer to ATWS Tree, Figure 2.3.

Figure 2.1 LOCA Functional Event Tree

Failure to shutdown the nuclear reaction would cause large amounts of the remaining water in the core to boil off much quicker than with only decay heat available. The loss of coolant would result in a power reduction; however, the coolant makeup systems would reintroduce coolant into the core for decay heat removal. The reactor power would increase again and boil off the coolant. This cycle could continue until the operator took manual control of the plant in order to reduce core power and stabilize the plant or until subsequent phenomenological or mechanical interaction resulted in system failure. Failure to stabilize the plant can result in core damage and/or containment failure. Sequences with failure of the reactor subcriticality function are transferred to the ATWS event tree.

If the reactor subcriticality function is successful, it is still necessary to remove heat from the core and replace lost coolant.

Early Containment Overpressure Protection (Vapor Suppression, VS)

During a LOCA, the normal heat removal path is disrupted by the pipe break and coolant is released to the containment. The steam generated by the hot coolant released during a LOCA is released into the drywell and forced by its own pressure to flow through downcomers into the wetwell. The wetwell contains a pool of water, called the suppression pool, for condensing the steam and thus reducing the temperature and pressure of the drywell. This vapor suppression pool has sufficient heat capacity for storing all the heat released to the containment for several hours after a LOCA before it becomes necessary to transfer heat from the containment to the ultimate heat sink.

If the steam released during a LOCA is not condensed by the vapor suppression pool, pressure will quickly buildup in the primary containment and the containment will need to be vented or it will mechanically fail (for large LOCAs, the time could be as short as 30 sec). Containment venting or failure may result in failure of the coolant injection systems and containment heat removal equipment due to the severe environments produced in the reactor building where most of the systems modeled in this analysis have components.

If the early containment overpressure protection function is available, it is still necessary to replace lost coolant in the core and eventually remove the heat transported to the suppression pool.

The vapor suppression pool also removes radioactivity released during a LOCA. This occurs as radioactive particles released during the LOCA are forced through the suppression pool water where the particles are essentially filtered and retained in the water. Noncondensible gases are not affected and remain in the primary containment atmosphere.

Core Coolant Makeup (CCM1, CCM2, CCM3)

A LOCA by definition results in loss of reactor coolant inventory that must be replaced in order to prevent core damage. The emergency core

cooling (ECC) systems are designed to provide cooling water to the core from an external source or from the suppression pool. This cooling water passes through the core, removing heat and transferring it to the vapor suppression pool. If the original source of water was external, the ECC systems would be realigned to take suction from the suppression pool to form a continuous circulation loop for cooling the core upon high level in the suppression pool or low level of the source. Eventually, the stored heat in the suppression pool must be transferred to the ultimate heat sink.

Non-emergency related systems are also capable of injecting water from external sources into the vessel during a LOCA. However, these systems are not capable of recirculating water from the suppression pool.

CCM1 represents failure of the injection systems before CHR failure, CCM2 represents failure of the injection systems after CHR failure but before containment failure, and CCM3 represents failure after containment failure. Failure of the coolant makeup function will result in loss of core cooling and core damage. Success of this function must be followed by removal of heat stored in the suppression pool.

<u>Containment Heat Removal (CHR)</u>

In the later stages of a LOCA, the heat buildup in the suppression pool can reach the pool's storage capacity. If this storage capacity is exceeded, the suppression pool will boil and evolved steam can cause overpressurization and rupture of the containment.

The containment heat removal (CHR) systems transfer heat to the ultimate heat sink from the suppression pool via heat exchangers. The containment heat removal systems during a LOCA are aligned to take suction from the suppression pool, pass the water through heat exchangers, and inject it into the core (low pressure coolant injection, LPCI, mode), into the drywell (containment spray, CSS, mode), or back into the suppression pool (suppression pool cooling, SPC, mode).

If the containment heat removal and core coolant makeup function are successful, the plant is stabilized and core damage is averted. The LOCA is thus mitigated and no other functions are required.

Failure of the containment heat removal function can have a feedback effect that results in failure of the core coolant makeup function. This failure can come about either before or after containment venting or structural failure of the containment from overpressure created by the failure to remove decay heat. As the containment pressurizes, the containment pressure, temperature, and suppression pool temperature all increase. High containment pressure can result in isolation and failure of the reactor core isolation cooling (RCIC) system. Low pressure injection systems will fail to inject when the automatic depressurization system (ADS) valves reclose and the reactor pressure vessel (RPV) repressurizes (this is not important for LOCAs where the RPV will remain depressurized from the break itself). Very high pressures and

temperatures can result in direct failure of the ADS valves which are not designed for such environments. High suppression pool temperatures can result in failure of systems pumping such high temperature water or from loss of net positive suction head (NPSH) when the pool becomes saturated (e.g., high pressure core spray, HPCS; low pressure core spray, LPCS; and LPCI). After containment venting or failure, high temperature steam may be blown into the reactor building depending upon the location of the failure (failure to the refueling floor will not blow steam into the reactor building). This blowdown will create severe environments in the reactor building well beyond the harsh environments usually evaluated. Most systems have components in the reactor building that would be subject to such environments and failure of the ECC and other systems after containment failure due to these environments would result in core damage with an already failed containment.

Sequence Descriptions

A brief description of each sequence on the LOCA functional event tree is provided below.

Sequence 1:

All of the functions work as required. The core is kept cooled and the containment is intact. No core damage results.

Sequence 2:

All the functions succeed except for containment heat removal. The core will be kept cooled until the heat transferred to the containment exceeds the suppression pool's heat capacity (hours after the initiation of the LOCA). The suppression pool water will boil, causing overpressurization and finally either venting or structural failure of the containment will occur. The core coolant makeup function does not fail before containment venting or failure and does not fail after containment venting or failure. Core damage does not occur; the sequence results in successful shut down of the reactor with a failed containment.

Sequence 3:

The containment heat removal function fails but the core coolant makeup function continues until after containment venting or failure. The injection systems then fail due to the severe environment in the reactor building and core damage results with an already failed containment.

Sequence 4:

The containment heat removal function fails and the core coolant makeup function fails before the containment is vented or fails. The injection systems fail from conditions in the containment and core damage results in an intact containment.

Sequence 5:

The initial core coolant makeup function fails and core damage results from the loss of all injection in an intact containment.

Sequences 6-10:

The reactor is shutdown, but the early containment overpressure protection function fails initially, resulting in overpressurization of the containment. Core damage could result depending on subsequent system and containment behavior. These sequences are similar to sequences 1-5 but with much shorter times (depending on LOCA size and bypass location and size) and different system success criteria.

Sequence 11:

The reactor fails to shutdown following initiation of the LOCA. Systems are available to mitigate LOCA with failure to scram accidents. However, since such sequences have their own characteristics, they are transferred to the ATWS event tree.

2.3.2 Transient Functional Event Tree

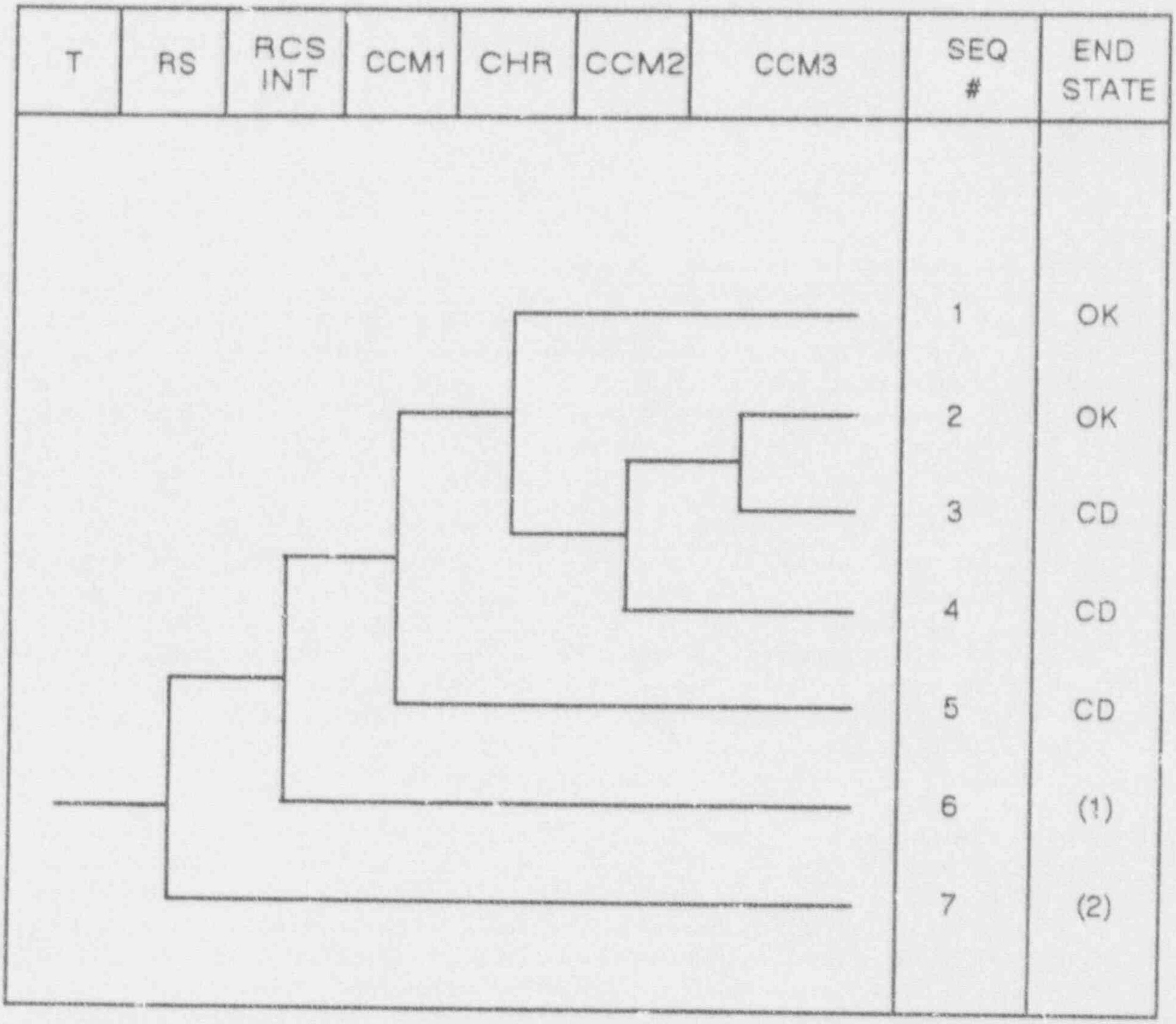
The transient functional event tree represents the general plant response to a transient event. Following a transient (T), it is necessary to shutdown the nuclear reaction (reactor subcriticality, RS), to protect the reactor pressure boundary from overpressurization if the normal heat removal path is unavailable (RCS integrity, RCS INT), to transfer steam to the containment for condensation (early containment overpressure protection if there is a LOCA or transient-induced LOCA, i.e., a safety relief valve, SRV, stuck open), provide makeup water to the vessel (core coolant makeup, CCM), and to remove heat released to the containment during the transient (containment heat removal, CHR). As with a LOCA, the function of removing radioactivity released to the containment will be addressed in the accident progression event tree.

The functions discussed above and their interactions following a transient are shown in the event tree illustrated in Figure 2.2. Each of the functions is discussed in more depth in the following paragraphs.

Reactor Subcriticality (RS)

Following a transient, it is necessary to limit the core heat generation by shutting down the nuclear reaction. This is normally done by inserting the control rods into the core. Backup systems and procedures are available for reducing core power given a failure to insert the control rods.

Failure to reduce core power following a transient can result in boiling off of the core coolant much quicker than if only decay heat is available. Since the turbine bypass capability is only 25 percent of full power, the normal heat removal system (if available) would not be



(1) Transfer to LOCA tree, Figure 2.1, after RS success.

(2) Transfer to ATWS Tree, Figure 2.3.

Figure 2.2 Transient Functional Event Tree

capable of removing all the generated steam. Excess pressure would be relieved to the containment through the safety/relief valves. Containment and core damage are possible if the operator fails to reduce core power. Sequences involving failure of the reactor subcriticality function are transfered to the ATWS event tree and evaluated there.

Successful subcriticality must be followed by removal of heat from the reactor.

RCS Integrity (RCS INT)

Given successful reactor subcriticality, the decay heat in the core will continue to produce steam. The RCS integrity function allows the reactor coolant system pressure to be relieved by the opening of the safety/relief valves and the transferring of the steam to the suppression pool if the normal heat removal path (power conversion system, PCS) has failed. Even if the turbine bypass is available, the transient effects of the reactor shutdown may require the opening of some SRVs. Multiple openings of the relief valves will occur if turbine bypass is not available.

Failure of the relief valves to open will result in overpressurization and possible rupture of the reactor vessel. In this analysis, it is assumed that the vessel rupture will result in the equivalent of a large LOCA and would transfer to the LOCA event tree. The rupture is most likely to occur at the omega seal on the reactor head. Successful operation of the injection systems could mitigate this event. It has been assumed in some previous studies that all of the check valves on the injection lines would freeze shut from the high pressure and would not be able to reopen after pressure decreased from the LOCA. This assumption seems much too severe given the proof testing pressure of the valves and vessel. The pressure rise is not instantaneous but quasi-static and would result in slow pressurization of the RPV from a mechanical standpoint. Also, after pressure decreased, the injection systems would tend to force water back into the vessel. In any case, failure of sufficient SRVs to open is an unlikely event and these sequences are probabilistically negligible and not developed further.

If overpressure protection succeeds, the pressure in the vessel is reduced but coolant is lost from the vessel to the vapor suppression pool. It thus becomes necessary to provide coolant to the vessel to keep the core covered.

Once the pressure in the vessel is relieved, the safety/relief valves should reclose to minimize coolant loss. If one or more of the valves fail to reclose, a continuous flow of steam from the vessel to the suppression pool will occur. Such an occurrence would require that the suppression pool remain intact, that makeup water be supplied to the vessel, and that the heat transferred to the suppression pool be eventually transferred to the environment. These sequences transfer to the LOCA tree because they have an unmitigated loss of primary coolant from the RPV. They are not equivalent to a LOCA because the flow is

directly to the suppression pool instead of to the drywell. They are called transient-induced LOCAs and will be evaluated separately.

Successful reclosure of the safety/relief valves must be followed by decay heat removal from the vessel.

Early Containment Overpressure Protection (VS)

Given failure of the RCS integrity function, heat from the vessel is transported to the suppression pool either directly through a stuck open SRV or via a large LOCA to the drywell and then through the downcomers. Failure of the suppression pool to condense this steam will result in overpressurization and failure of the containment within a very short time (30 sec to 15 min). Overpressurization of the containment can fail the equipment required for emergency core cooling and containment heat removal, thus leading to core damage and a radioactive release. Thus failure of this function is assumed to result in the core being vulnerable to damage. Sequences with failure of the RCS integrity function are transfered to the LOCA tree and evaluated there. This function does not, therefore, explicitly appear on the transient tree.

Successful vapor suppression operation must be followed by makeup of reactor vessel coolant and removal of heat released to the containment.

Core Coolant Makeup (CCM1, CCM2, CCM3)

The emergency core cooling (ECC) systems are designed to provide cooling water to the core from an external source or from the suppression pool. This cooling water passes through the core, removing heat and transferring it to the vapor suppression pool. If the original source of water was external, the ECC systems would be realigned to take suction from the suppression pool to form a continuous circulation loop for cooling the core on high level in the suppression pool or low level of the source. Eventually, the stored heat in the suppression pool must be transferred to the ultimate heat sink.

Non-emergency related systems are also capable of injecting water from external sources into the vessel during a transient. However, these systems are not capable of recirculating water from the suppression pool.

CCM1 represents failure of the injection systems before CHR failure, CCM2 represents failure of the injection systems after CHR failure but before containment failure, and CCM3 represents failure after containment failure. Failure of the coolant makeup function will result in loss of core cooling and a core damage. Success of this function must be followed by removal of heat stored in the suppression pool.

Containment Heat Removal (CHR)

If the normal heat removal path is unavailable for removal of residual heat following reactor scram, residual heat is transferred to the

containment. Eventually this heat must be removed or containment failure will occur. Containment failure can potentially result in core damage.

Failure of the containment heat removal function can have a feedback effect that results in failure of the core coolant makeup function. This failure can come about either before or after containment venting or structural failure of the containment from overpressure created by the failure to remove decay heat. As the containment pressurizes, the containment pressure, temperature, and suppression pool temperature all increase. High containment pressure can result in isolation and failure of the RCIC system. Low pressure injection systems will fail to inject when the ADS valves reclose and the RPV repressurizes. Very high pressures and temperatures can result in direct failure of the ADS valves which are not designed for such environments. High suppression pool temperatures can result in failure of systems pumping such high temperature water or from loss of NPSH when the pool becomes saturated. After containment venting or failure, high temperature steam may be blown into the reactor building depending upon the location of the failure (failure to the refueling floor will not blow steam into the reactor building). This blowdown will create severe environments in the reactor building well beyond the harsh environments usually evaluated. Most systems have components in the reactor building that would be subject to such environments and failure of the ECC and other systems after containment failure due to these environments would result in core damage with an already failed containment.

Successful residual heat removal can result in core stability if core coolant makeup continues to be available.

Sequence Descriptions

A brief description of each sequence in the transient functional event tree is provided below.

Sequence 1:

The reactor scrams, safety/relief valves open to relieve pressure and successfully reclose, core coolant makeup is provided and the containment heat removal systems function to remove residual heat. No core damage and containment is intact.

Sequence 2:

All the functions succeed except for containment heat removal. The core will be kept cooled until the heat transferred to the containment from the SRVs exceeds the suppression pool's capacity (hours after the transient initiation). The suppression pool water will boil, causing overpressurization and finally either venting or structural failure of the containment will occur. The core coolant makeup function does not fail before containment venting or failure (CCM2) and does not fail after containment venting or failure (CCM3). Core damage does not occur; the

sequence results in successful shutdown of the reactor with a failed containment.

Sequence 3:

The containment heat removal function fails but the core coolant makeup function continues until after containment venting or failure. The injection systems then fail due to the severe environment in the reactor building and core damage results with an already failed containment.

Sequence 4:

The containment heat removal function fails and the core coolant makeup function fails before the containment is vented or fails. The injection systems fail from conditions in the containment and core damage results in an intact containment.

Sequence 5:

The initial core coolant makeup function fails and core damage results from the loss of all injection in an intact containment.

Sequence 6:

Following the initiating event, the reactor is successfully scramed, but the RCS integrity function fails. Steam from the vessel is released to the containment. This sequence becomes a large LOCA, if the SRVs have failed to open, and a small, medium, or large LOCA if 1, 2, or ≥ 3 SRVs fail to reclose, and transfers to the LOCA tree for evaluation.

Sequence 7:

The reactor subcriticality function fails following a transient. A separate ATWS event tree is used to depict the sequences of events following failure of the reactor subcriticality function. Transfer to ATWS event tree.

2.3.3 ATWS Functional Event Tree

The ATWS functional event tree represents the general plant response to any transient or LOCA event followed by failure to render the reactor subcritical using the reactor protection and alternate rod insertion systems. Following an ATWS transient, it is necessary to shut down the nuclear reaction using alternate means such as the standby liquid control (SBLC) system (reactor subcriticality, RS), to protect the reactor pressure boundary from overpressurization if the normal heat removal path is unavailable (RCS integrity, RCS INT), to transfer steam to the containment for condensation (early containment overpressure protection if there is a LOCA or transient-induced LOCA, i.e., an SRV stuck open), provide makeup water to the vessel (core coolant makeup, CCM), and to remove heat released to the containment during the transient (containment heat removal, CHR). As with the LOCA and transient trees, the function

of removing radioactivity released to the containment will be addressed in the accident progression event tree.

The functions discussed above and their interactions following a transient are shown in the event tree illustrated in Figure 2.3. Each of the functions is discussed in more depth in the following paragraphs.

Reactor Subcriticality (RS1)

Following an ATWS event, it is still necessary to limit the core heat generation by shutting down the nuclear reaction. This is normally done by inserting the control rods into the core; however, for ATWS scenarios, normal mechanisms for inserting the control rods into the core have already been assessed to have failed. The most likely reason for failure to scram given the existence of the alternate rod insertion system, which makes electrical failure to scram probabilistically small, is mechanical failure of the rods to insert. Backup systems and procedures are available for reducing core power given a mechanical failure to insert the control rods.

Failure to reduce core power following an ATWS transient can result in quickly boiling off the core coolant until the reactor water level has stabilized due to the balance between the amount of water being injected into the core and the amount of water being boiled off. If turbine trip also does not occur and the reactor continues as before, then no accident results. If the turbine trips, since the turbine bypass capability is only 25 percent of full power, the normal heat removal system (if available) would not be capable of removing all the generated steam early in the sequence. The vessel pressure would increase rapidly due to the high energy generation rate which would equilibrate at a rate consistent with the particular injection system being used or at the decay heat level, if no injection was available. Excess pressure would be relieved to the containment through the SRVs.

RCS Integrity (RCS INT)

Whether or not reactor subcriticality is successful, energy will continue to be produced either at some equilibrium power level consistent with the injection rate or at the decay heat level. The RCS integrity function allows the reactor coolant system pressure to be relieved by the opening of a sufficient number of the safety/relief valves and the transferring of the steam to the suppression pool if the normal heat removal path (PCS) has failed. Even if the turbine bypass is available, the transient effects of the reactor shutdown may require the opening of some SRVs. Multiple and/or continuous openings of the relief valves will occur if turbine bypass is not available.

Failure of the relief valves to open will result in overpressurization and possible rupture of the reactor vessel. In this analysis, it is assumed that the vessel rupture will result in the equivalent of a large

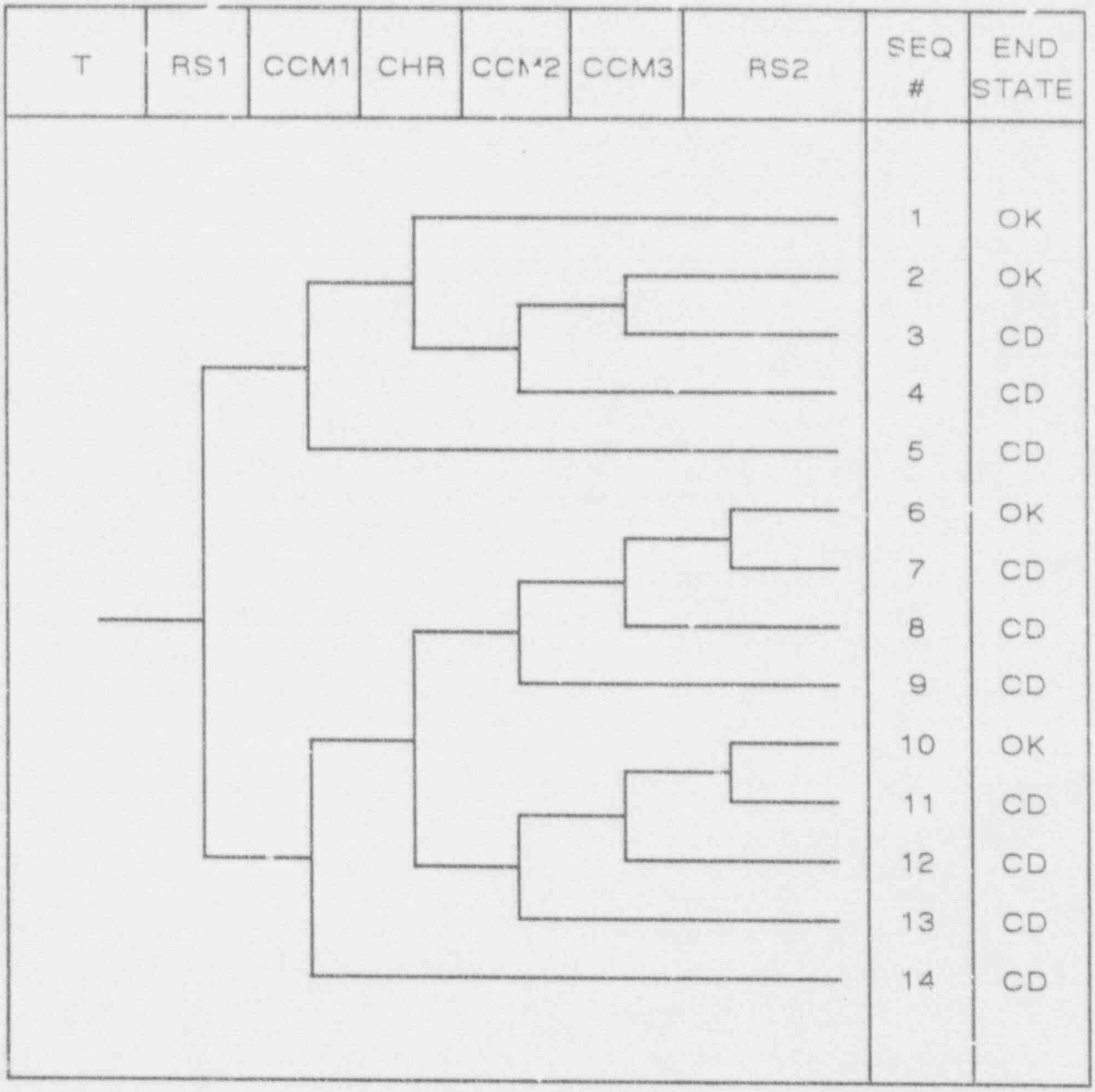


Figure 2.3 ATWS Functional Event Tree

LOCA. The rupture is most likely to occur at the omega seal on the reactor head. Successful operation of the injection systems could mitigate this event. It has been assumed in some previous studies that all of the check valves on the injection lines would freeze shut from the high pressure and would not be able to reopen after pressure decreased from the LOCA. This assumption seems much too severe given the proof testing pressure of the valves and vessel. The pressure rise is not instantaneous but quasi-static and would result in slow pressurization of the RPV from a mechanical standpoint. Also, after pressure decreased, the injection systems would tend to force water back into the vessel. Failure of sufficient SRVs to open is an unlikely event and these sequences are probabilistically negligible and not developed further.

If overpressure protection succeeds, the pressure in the vessel is reduced but coolant is lost from the vessel to the vapor suppression pool. It thus becomes necessary to provide coolant to the vessel to keep the core covered.

If reactor subcriticality succeeds then, once the pressure in the vessel is relieved, the safety/relief valves should reclose to minimize coolant loss. If one or more of the valves fail to reclose, a continuous flow of steam from the vessel to the suppression pool will occur. Such an occurrence would require that the suppression pool remain intact, that makeup water be supplied to the vessel, and that the heat transferred to the suppression pool be eventually transferred to the environment. The LOCA aspects of these sequences do not affect the event tree because the systems used to mitigate an ATWS event can mitigate LOCAs of any size and, for sequences without reactor subcriticality, the ADS valves will be open anyway to transfer the energy to the suppression pool. For the above reasons, this event does not appear explicitly on the ATWS functional event tree.

Successful reclosure of the safety/relief valves must be followed by decay heat removal from the vessel.

Early Containment Overpressure Protection (VS)

Given failure of the RCS integrity function, heat from the vessel is transported to the suppression pool either directly through a stuck open SRV or via a large LOCA to the drywell and then through the downcomers. Failure of the suppression pool to condense this steam will result in overpressurization and failure of the containment within a very short time (30 sec to 15 min). Overpressurization of the containment can fail the equipment required for emergency core cooling and containment heat removal, thus leading to core damage and radioactive release. Thus failure of this function is assumed to result in the core being vulnerable to damage. Sequences with failure of the early containment overpressure function are probabilistically negligible and not developed on the ATWS functional event tree.

Successful vapor suppression operation must be followed by makeup of reactor vessel coolant and removal of heat released to the containment.

Core Coolant Makeup (CCM1, CCM2, CCM3)

The emergency core cooling (ECC) systems are designed to provide cooling water to the core from an external source or from the suppression pool. This cooling water passes through the core, removing heat and transferring it to the vapor suppression pool. If the original source of water was external, the ECC systems would be realigned to take suction from the suppression pool to form a continuous circulation loop for cooling the core upon high level in the suppression pool or low level of the source. Eventually, the stored heat in the suppression pool must be transferred to the ultimate heat sink.

Non-emergency related systems are also capable of injecting water from external sources into the vessel during a transient. However, these systems are not capable of recirculating water from the suppression pool.

CCM1 represents failure of the injection systems before CHR failure, CCM2 represents failure of the injection systems after CHR failure but before containment failure, and CCM3 represents failure after containment failure. Failure of the coolant makeup function will result in loss of core cooling and a core damage. Success of this function must be followed by removal of heat stored in the suppression pool.

Containment Heat Removal (CHR)

Even if the normal heat removal path is available for removal of energy being generated after failure of the reactor subcriticality function, the the reactor power level will be in the range of 9-17% depending on the systems operating, which is much higher than the capability of the RHR system (about 3%). The energy being generated in the vessel will be deposited in the suppression pool via the SRV discharge lines or directly to the drywell if a LOCA exists. The excess energy, over and above the RHR system's heat removal capacity, will result in rapid containment pressurization.

Given failure of the reactor subcriticality function and failure of secondary means to render the reactor subcritical in time to prevent containment venting or structural failure, a feedback effect can occur that results in failure of the core coolant makeup function. This failure can come about either before or after containment venting or structural failure of the containment. As the containment pressurizes, the containment pressure, temperature, and suppression pool temperature all increase. High containment pressure can result in failure of the RCIC system and low pressure injection systems when the ADS valves reclose. Very high pressures and temperatures can result in direct failure of the ADS valves which are not designed for such environments. High suppression pool temperatures can result in failure of systems pumping such high temperature water or from loss of NPSH when the pool becomes saturated. After containment venting or failure, high

temperature steam may be blown into the reactor building depending upon the location of the failure (failure to the refueling floor will not blow steam into the reactor building). This blowdown will create severe environments in the reactor building well beyond the harsh environments usually evaluated. Most systems have components in the reactor building that would be subject to such environments and failure of the ECC and other systems after containment failure due to these environments would result in core damage with an already failed containment.

If only low pressure injection systems are working and RHR works, LTAS[2] calculations (described in Section 2.6.3) show that containment pressure will equilibrate near the ADS reclosure pressure. The low pressure systems inject when the ADS valves open as the reactor goes subcritical. When injection stops, the RHR system reduces containment pressure below the reclosure pressure, and the ADS valves reopen. The low pressure injection systems stop injecting as the containment pressure rises due to the energy generated when the core is reflooded, the ADS valves reclose, and the RPV repressurizes. If venting occurs, or both RHR and venting are successful, the containment pressure will equilibrate above the vent pressure but below the ADS reclosure pressure. Low pressure injection will go on and off as the RPV pressure goes below and above the low pressure injection pumps shutoff head. These scenarios correspond to the cases where injection does not fail from the severe environments produced in the reactor building after venting or from the valve cycling in the injection lines as the RPV pressure varies.

Successful residual heat removal can result in core stability if core coolant makeup continues to be available.

Sequence Descriptions

A brief description of each sequence in the transient functional event tree is provided below.

Sequence 1:

The reactor is successfully shutdown using alternate means, core coolant makeup is provided, and the containment heat removal systems function to remove residual heat. While the containment will pressurize more than in a normal transient or LOCA due to the energy generation rate initially being higher than what can be removed by the containment heat removal system, once the reactor is shutdown the containment heat removal systems will begin to reduce containment pressure. A stable end state is reached, core damage is averted, and the containment is intact.

Sequence 2:

Alternate means of shutting down the reactor succeed but containment heat removal fails. The core will be kept cooled until the heat transferred to the containment from the SRVs exceeds the suppression pool's capacity (from about a half hour after the transient initiation to many hours,

depending on when reactor subcriticality succeeds). The suppression pool water will boil, causing overpressurization, and finally either venting or structural failure of the containment will occur. The core coolant makeup function does not fail either before or after containment venting or failure. Core damage does not occur; the sequence results in successful shutdown of the reactor with a failed containment.

Sequence 3:

The reactor is successfully shutdown using alternate means, injection is initially successful, containment heat removal fails, but the core coolant makeup function continues until after containment venting or failure. The injection systems then fail due to the severe environment in the reactor building and core damage results with an already failed containment.

Sequence 4:

The reactor is successfully shutdown using alternate means, injection is initially successful, containment heat removal fails, and the core coolant makeup function fails before the containment is vented or fails. The injection systems fail from conditions in the containment and core damage results in an intact containment.

Sequence 5:

The reactor is successfully shutdown using alternate means but the initial core coolant makeup function fails and core damage results from the loss of all injection in an intact containment.

Sequence 6:

The reactor is not successfully shutdown using alternate means, core coolant makeup is provided, and the containment heat removal systems function to remove some energy from containment. The containment will continue to pressurize until one of three things occurs depending upon the injection systems available: (1) if low pressure systems are being used and the containment is vented, the pressure equilibrates at a level slightly higher than the vent pressure and injection continues; (2) if low pressure systems are being used and the containment is not vented, the reclosure of the ADS valves on high containment pressure results in oscillation of the containment pressure near the ADS reclosure pressure and in oscillatory opening and closing of the ADS valves and use of the low pressure systems; or (3) if high pressure systems are being used, the containment pressurizes until containment venting or failure occurs. The injection systems continue to work both before and after containment venting or failure and a quasi-steady state is reached. Ultimately the reactor is shutdown and core damage is adverted with a failed containment.

Sequence 7:

This sequence is identical to sequence 6 except that ultimate shutdown is not achieved. For this study, the sequence was assessed to go to core damage with a failed containment. The sequence is probabilistically low and further development was felt to be unwarranted.

Sequence 8:

This sequence is also similar to sequence 6 except that the injection systems fail after containment venting or failure and core damage results with a failed containment.

Sequence 9:

This sequence is similar to sequence 6 except that the injection systems fail before containment failure and core damage occurs with an intact containment; since, once injection fails, the reactor power level drops quickly to within the range of the operating containment heat removal system.

Sequence 10:

The reactor is not successfully shutdown using alternate means, core coolant makeup is provided but the containment heat removal systems fail. The containment will continue to pressurize until one of two things occurs depending upon the injection systems available: (1) if low pressure systems are being used, the containment is vented and pressure equilibrates at a level slightly higher than the vent pressure; or (2) if high pressure systems are being used, the containment pressurizes until containment venting or failure occurs. The injection systems continue to work both before and after containment venting or failure and a quasi-steady state is reached. Ultimately the reactor is shutdown and core damage is averted with a failed containment.

Sequence 11:

This sequence is identical to sequence 10 except that the reactor is not ultimately shutdown. For this study, the sequence was assessed to go to core damage with a failed containment. The sequence is probabilistically low and further development was felt to be unwarranted.

Sequence 12:

This sequence is similar to sequence 10 except that the injection systems fail after containment venting or failure due to the severe environments in the reactor building.

Sequence 13:

The reactor is not successfully shutdown using alternate means, core coolant makeup is provided using low pressure injection systems only, and

the containment heat removal systems fail. The containment continues to pressurize and the low pressure systems fail before containment venting or failure occurs. Core damage occurs in an intact containment.

Sequence 14:

The reactor is not successfully shutdown using alternate means, core coolant makeup fails initially, and core damage occurs in an intact containment.

2.4 Systems Available to Perform Required Functions

The front-line systems available at LaSalle for mitigating LOCAs and transients are presented in Tables 2.1 and 2.2 respectively. Detailed descriptions of the systems listed are given in the corresponding fault tree analyses chapters presented in Volume 6 of this report. Some information on the systems is also presented in the following section on system success criteria. Additional information can be found in the LaSalle FSAR.[3]

2.5 Success Criteria

In order to construct the event trees and front-line system fault trees, it is necessary to define the front-line system success criteria. The front-line system success criteria specify the minimum number of subsystems (trains) or components whose operation is required to successfully perform one of the functions used in the functional event trees. As such, the complement of the system success criteria also defines the system failure criteria which is the top event of the corresponding fault tree.

The system success criteria are different for LOCA, transient, and LOCA or transient with failure to scram (ATWS) initiators. The success criteria for each function required for LOCAs, transients, and ATWS are delineated in the following sections. The success criteria are summarized in Tables 2.3, 2.4, and 2.5.

2.5.1 LOCAs (L)

A large LOCA is defined as a break in the reactor coolant boundary sufficient to cause rapid vessel depressurization. The size of such a LOCA is ≥ 0.3 ft^2 for both a steam and liquid break. This result is based on calculations reported in Reference 4. In particular, a case involving a reactor isolation event with failure of high pressure injection and successful ADS for a BWR6 indicates that the opening of 3 SRVs is sufficient to depressurize the RPV fast enough that low pressure injection systems can mitigate the transient. The flow area of 3 SRVs is approximately 0.3 ft^2. Thus, a steam break of this size or larger is

Table 2.1
LOCA FUNCTION/SYSTEM RELATIONSHIP

Function	Systems
Reactor Subcriticality	Reactor Protection System (RPS) Recirculation Pump Trip (RPT) Alternate Rod Insertion (ARI) Standby Liquid Control System (SBLC)
Early Containment Overpressure Protection	Vapor Suppression System (VSS)
Core Coolant Makeup (High Pressure)	Main Feedwater (MFW) High Pressure Core Spray (HPCS) Reactor Core Isolation Cooling (RCIC) Control Rod Drive (CRD)
(Low Pressure)	Automatic Depressurization System (ADS) Low Pressure Core Spray (LPCS) Low Pressure Coolant Injection (LPCI) Condensate System (CDS)
Containment Heat Removal	Residual Heat Removal System (RHR) Suppression Pool Cooling (SPC) Containment Spray System (CSS) Shutdown Cooling System (SCS)

Table 2.2
TRANSIENT FUNCTION/SYSTEM RELATIONSHIP

Function	Systems
Reactor Subcriticality	Reactor Protection System (RPS)
	Recirculation Pump Trip (RPT)
	Alternate Rod Insertion (ARI)
	Standby Liquid Control System (SBLC)
RCS Integrity	Safety/Relief Valves (SRV) open
	SRV Closure
Early Containment Overpressure Protection	Vapor Suppression System (VSS)
Core Coolant Makeup	
(High Pressure)	Main Feedwater (MFW)
	High Pressure Core Spray (HPCS)
	Reactor Core Isolation Cooling (RCIC)
	Control Rod Drive (CRD)
(Low Pressure)	Automatic Depressurization System (ADS)
	Low Pressure Core Spray (LPCS)
	Low Pressure Coolant Injection (LPCI)
	Condensate System (CDS)
	Diesel Driven Fire Water (DDFW)
Containment Heat Removal	Residual Heat Removal System (RHR)
	Suppression Pool Cooling (SPC)
	Containment Spray System (CSS)
	Shutdown Cooling System (SDC)
	Power Conversion System (PCS)

Table 2.3
LOCA SUCCESS CRITERIA

Accident Initiator	Reactor Subcriticality	Early Containment Overpressure Protection	Core Coolant Makeup	Containment Heat Removal
Large LOCA Liquid Break $\geq$ 0.[illegible] ft^2 Steam Break $\geq 0.3\ ft^2$	<5 adjacent rods fail to insert and <30 rods fail to insert	Steam released from break is directed to vapor suppression pool	1 MD FW pump and 1/4 condensate transfer trains or HPCS or LPCS or 1/3 LPCI or 1/4 condensate trains	1/2 RHR in suppression pool cooling or containment spray modes
Intermediate LOCA Liquid Break > 0.005 to $< 0.3\ ft^2$ Steam Break > 0.1 to $< 0.3\ ft^2$	Same as above	Same as above	1 MD FW pump and 1/4 condensate trains or HPCS or ADS (3/7 SRVs auto or manual) and LPCS or 1/3 LPCI or 1/4 condensate train	1/2 RHR in suppression pool cooling or containment spray modes

Table 2.3 (Concluded)
LOCA SUCCESS CRITERIA

Accident Initiator	Reactor Subcriticality	Early Containment Overpressure Protection	Core Coolant Makeup	Containment Heat Removal
Small LOCA Liquid Break ≤ 0.005 ft^2 Steam Break ≤ 0.1 ft^2	Same as for Large LOCA	Same as for Large LOCA	RCIC or 1 MD FW pump and 1/4 condensate trains or HPCS or 3/7 ADS or SRVS and LPCS or 1/3 LPCI or 1/4 condensate train	1/2 RHR in suppression pool cooling or containment spray modes

Table 2.4
TRANSIENT WITH AUTOMATIC REACTOR SCRAM
SUCCESS CRITERIA

Function	System
Reactor Subcriticality	<5 adjacent rods fail to insert and <30 rods fail to insert
RCS Integrity	12 of 18 SRVs open All open SRVs reclose 1 open - small LOCA 2 open - intermediate LOCA 3 or more open - large LOCA
Early Containment Overpressure	Steam released through SRVS condensed in suppression pool
Core Coolant Makeup	1/3 FW pumps and 1/4 condensate trains or RCIC or HPCS or ADS (3/7 SRVs) and LPCS or 1/3 LPCI or 1/4 condensate trains
Containment Heat Removal	1/2 RHR in any mode or PCS

Table 2.5
TRANSIENT WITHOUT AUTOMATIC SCRAM
SUCCESS CRITERIA

Function	System Success Criteria
Reactivity Control	RPT and ARI or RPT and Manual Scram or RPT and 1/2 SBLC and Manual Level Control and ADS Inhibit or RPT and Manual Level Control and ADS Inhibit*
RCS Integrity	16 of 18 SRVs open as required
Early Containment Overpressure Protection	Vapor Suppression System
Core Coolant Makeup	1/3 FW pumps and 1/4 condensate trains or HPCS or ADS (3/7 SRVs) and LPCS or 1/3 LPCI
Heat Removal	1/2 RHR in any mode or PCS

* When PCS is available

assessed to result in rapid vessel depressurization. A liquid break of this size is assessed to result in a faster depressurization rate.

A medium LOCA is defined as a break of a size such that RCIC is not sufficient to mitigate the accident alone and is not large enough to depressurize the RPV fast enough for the low pressure injection systems to prevent core damage if the high pressure injection systems have failed.

Calculations discussed in Reference 4 indicate that RCIC is sufficient to keep the core covered during a loss of feedwater transient with one stuck open relief valve (SORV) in a BWR5. Steamline breaks greater than 0.1 ft^2 (flow area of one SRV) are thus too large for RCIC. Other calculations performed for a BWR4 and BWR6 indicate that RCIC will maintain the core covered for liquid breaks less than 0.005 ft^2. Similar results are expected for a BWR5. The upper boundaries of the break sizes correspond to the lower limit for large LOCAs. Thus the break sizes for medium LOCAs are:

Steam: $0.1\ ft^2 < A < 0.3\ ft^2$
Liquid: $0.005\ ft^2 < A < 0.3\ ft^2$

A small LOCA is defined as a LOCA where RCIC alone can maintain the core covered. As mentioned previously, calculations performed in Reference 4 indicate that RCIC can successfully mitigate steam breaks up to and including 0.1 ft^2 and liquid breaks up to and including 0.005 ft^2. Plant specific calculations described in Section 2.6.1 confirm this result.

Reactor Subcriticality (KS)

Reactor subcriticality using the reactor protection system (RPS) or the alternate rod insertion system (ARI) would not be accomplished for any size LOCA if more than 5 adjacent control rods fail to insert or more than 30 control rods fail to insert.[5]

Early Containment Failure (VS)

Vapor suppression is successful if the steam/water released to the drywell from the break is transported from the drywell to the suppression pool where it is condensed. The transport is conducted through downcomers connecting the drywell and suppression pool.

Vacuum breakers are provided to allow a return flow path for noncondensibles from the suppression chamber to the drywell. Opening of the vacuum breakers is required to prevent pressurization of the suppression chamber. Subsequent reclosing of the vacuum breakers is also required to prevent backflow of gas from the drywell to the wetwell bypassing the suppression pool.

The steam released to the drywell during a LOCA can bypass the suppression pool if a downcomer ruptures in the wetwell air space or if vacuum breakers fail open or closed. The exact bypass flow area which would result in containment overpressurization and failure is calculated in the LaSalle FSAR[3] as a function of break area. The maximum allowable leakage area is approximately 0.05 ft^2 (a 1.5" diameter hole).

In the Reactor Safety Study,[6] failure of the VSS during a large LOCA was found to lead to containment rupture by overpressurization within 30 seconds. Since a similar short period is assumed for the Mark II containment, no credit can be taken for use of other containment pressure reduction features such as suppression pool sprays or drywell sprays. For smaller LOCAs and transient-induced LOCAs with failure of the SRV discharge line in the wetwell airspace, up to 15 minutes or more could be available depending on the amount of bypass.

<u>Core Coolant Makeup (CCM)</u>

Core coolant makeup following a LOCA can be accomplished with the motor-driven feedwater train (MFW), with water supplied to the pump suction by a single condensate/condensate booster pump train. The motor-driven feedwater pump is capable of operating at 40% of reactor rated steam flow; automatic startup and control on reactor vessel water level occurs if the turbine-driven feedwater pumps trip. On depletion of the condenser hotwell, makeup from the condensate storage tank is limited to 1800 gpm.

The turbine-driven feedwater pumps are not available following a LOCA due to low steam pressure and/or mainsteam isolation valve (MSIV) closure which stops steam flow to the pump turbines. Reopening of the MSIVs is possible but no credit is taken for such an action in this analysis due to violation of containment isolation and potential for radioactive release outside the containment.

HPCS has sufficient capacity for mitigating all sizes of LOCAs. The capacity of RCIC on the other hand is insufficient for coolant makeup during a large or medium LOCA but is sufficient for a small LOCA. In any case, the RCIC turbine would fail from low steam pressure for large and medium LOCAs.

Use of the low pressure systems can be achieved immediately following a large LOCA since the reactor vessel would be depressurized. However, for small and medium LOCAs, reactor vessel depressurization would be required. Reactor vessel depressurization can be accomplished automatically through ADS initiation or manually by opening SRVs. Successful depressurization requires that the vessel remain depressurized once low pressure injection is initiated (i.e., the relief capacity must be sufficient to remove the generated steam in order to prevent repressurization). As previously stated in the discussion on the definition of a large LOCA, calculations reported in Reference 4 indicate that the opening of 3 SRVs will result in such a sustained

depressurization. Success is thus defined as automatic opening of 3 out of 7 ADS valves or manual opening of any 3 SRVs.

The single train of LPCS can also be used to mitigate any size LOCA. Calculations presented in Reference 4 indicated that the single LPCS train is sufficient for flooding the core during a design basis liquid break in a BWR4 and following ADS initiation during an isolation event in a BWR6. These results are said to be applicable to a BWR5 such as LaSalle. Plant-specific calculations perormed for this analysis and reported in Section 2.6.1 indicate the adequacy of the LPCS by itself for mitigating a large LOCA.

A single train of LPCI can also mitigate any size LOCA. A calculation presented in Reference 4 indicates that a single LPCI train is sufficient to mitigate a DBA suction break in a BWR4. Another calculation indicates that for an isolation event in a BWR6, reactor depressurization followed by injection from one LPCI train is sufficient for mitigation. The results are said to also be applicable for a BWR5. For very long-term core cooling when the coolant in the RPV is subcooled, steam cooling will no longer be available to cool any uncovered portions of the core and one LPCI train may not be sufficient to keep the level in the core high enough to prevent the upper portion of the core from reheating and going to core damage. Two LPCI trains would be sufficient in this case. A plant-specific RELAP5[7] calculation, reported in Section 2.6.1, for a recirculation line break with one train of LPCI shows LPCI can successfully mitigate a DBA suction break.

One train of condensate pumps is also sufficient for maintaining core coolant level for any size LOCA. However, the condensate system is not automatically controlled and operator action would be required to control the system. This system is normally operating and should be available, provided that coolant is available in the condenser hotwell. As for feedwater, makeup from the condensate storage tank is limited to 1800 gpm.

<u>Containment Heat Removal (CHR)</u>

Containment heat removal can be accomplished following a LOCA by the RHR in three of its four possible operating modes:

1. Low Pressure Coolant Injection,
2. Suppression pool cooling, and
3. Containment spray.

No credit is taken in this analysis for the steam condensing and shutdown cooling modes of operation since the break dumps heat directly to the containment and these modes of operation will not remove the energy from the containment. According to the LaSalle FSAR,[3] one RHR heat exchanger is sufficient for maintaining the suppression pool temperature below limiting temperatures. This is assumed to be true in any mode of RHR operation. Flow from one of two service water pumps is required to remove the heat from one heat exchanger. The PCS is considered

unavailable since the MSIVs close upon a low-low (Level 1) reactor vessel water level signal. Reopening of the MSIVs is possible and heat removal by the PCS may be feasible; however, the preferred flow path will be out of the break directly to the drywell. The amount of energy that could be removed by the PCS would depend critically on the size of the break and the relative resistance of the two paths. Also, after core damage has begun, radioactivity could be transported directly outside the containment. We did not have the resources to calculate the range of conditions under which PCS would be a viable option for this scenario. At best this would only slow down the containment pressurization and at worst would be ineffectual.

2.5.2 Transients With Automatic Reactor Scram

The system success criteria for most transients with automatic reactor scram is summarized in Table 2.4. As indicated, the success criteria for the reactor subcriticality and core coolant makeup function are the same as those previously listed for a small LOCA. However, the core coolant makeup function may have different success criteria depending on the initiator. For transient initiators resulting in MSIV closure, the turbine-driven feedwater pumps would only be available if the operator reopened an MSIV.

The RCS integrity function is required for most transients. The most limiting transient for this function is an MSIV closure event. With scram, approximately 12 of 18 SRVs must open.[4] Since common cause failures would dominate the failure probability of five or more valves to open, a more precise definition is not required.

All of the SRVs must reclose following relief of the vessel pressure. Failure of one SRV to reclose (0.1 ft^2 flow area) will result in the equivalent of a small LOCA. Failure of two SRVs to reclose will result in the equivalent of a medium LOCA, and failure of three or more to reclose results in the equivalent of a large LOCA.

The early containment overpressure protection function has not in the past been represented on transient event trees. However, during a transient, steam released to the containment is released via the SRVs to the suppression pool (during a LOCA, the steam is released to the drywell). Containment integrity during a transient can be challenged if an SRV fails to reclose and the SRV line breaks in the wetwell airspace[6] causing steam to bypass the suppression pool. Success then requires that the SRV line associated with a stuck open SRV remain intact.

The successful operation of PCS for heat removal during a transient requires that one steam bypass line to the condenser be available and the condenser be capable of condensing steam. For many transient initiating events, portions of the PCS may not be available but may be recoverable. For example, in an MSIV closure transient, the MSIVs may be reopened.

Any one of three modes of RHR can generally be used for most transients. No credit is taken for use of the steam condensing mode due to the complex actions required for its initiation. Also, the shutdown cooling mode cannot be successfully used during an inadvertently open relief valve (IORV) initiator or a transient with an SORV since it does not remove heat from the containment. One of two RHR loops is required for most transients. However some transients can potentially put a greater demand on the RHR system. For example, an SORV will not result in an immediate automatic reactor scram and therefore can result in a large heat load dumped to the suppression pool. The success criteria of the RHR for an SORV transient may require that it be initiated sooner than for other transients or both loops of the RHR may be required, depending on when reactor scram occurs.

2.5.3 Transients Without Automatic Reactor Scram (ATWS)

The success criteria for a general ATWS event tree is presented in Table 2.5. Reactivity control can be accomplished by any of several methods. Recirculation pump trip (RPT, both pumps - manual or automatic) is required in each method to reduce core power to between 30 and 50%. RPT and Alternate Rod Insertion (ARI) will result in successful power reduction if the control rods can be inserted. ARI will not occur until approximately 10 - 30s after a normal scram would occur. RPT would be required in this interim period to reduce power.

RPT and a timely manual scram will also decrease power. The operator is directed by procedures to immediately try to manually scram the reactor by pressing the manual scram buttons or placing the mode switch in shutdown. The time available to initiate manual scram before core damage or containment failure occurs is not known. However, from the LTAS calculations described in Section 2.6.3, we see that it could take a long time to get core damage if the reactor is not shutdown depending on what systems are operating. However, it would seem to be prudent to shutdown the reactor as quickly as possible. Certainly shutting down the reactor before venting or containment failure which would occur in about 40 to 60 minutes would be prudent.

RPT and timely manual initiation of one of two SBLC pumps (43 gpm) will result in reactor shutdown if the operator takes two additional steps: (1) controls all coolant injection systems to maintain water level at the top of the active fuel (TAF) and (2) prevents ADS actuation. The flow rate is adequate to inject sufficient boron to shut down the reactor within approximately 20 minutes. The system is manually initiated. During the interim period, the power must be reduced further by RPT and reduction of the vessel water level to the TAF. LaSalle has implemented the version of the ATWS rule which requires an alternate rod insertion system and doubling the boron concentration.

Manual level control to the TAF in conjunction with RPT can reduce core power to approximately 15% at 1000 psig. This heat load can be handled

by the PCS if available. If the PCS is not available, the heat load would be dumped to the suppression pool. Since the combined RHR heat removal capacity is approximately 3%, containment failure would occur. A further reduction in power is possible by reducing system pressure. However, no credit is currently being taken in this analysis for this action since control of power at low pressure is difficult (power/pressure oscillations can occur if the level is allowed to deviate much over the TAF) and the LTAS calculations show that level can not be maintained above about 2/3 TAF except with feedwater or condensate. We assume in this analysis that ADS will occur in all cases even if high pressure injection by HPCS is working.

The requirement for RCS integrity during an ATWS is more severe than during a transient with scram. Sixteen of 18 SRVs[2] must open to relieve the RCS pressure. The requirement for early containment overpressure protection is the same as for transients with scram.

The success criteria for core coolant makeup requires use of the high pressure injection systems or depressurization and use of low pressure injection systems. The RCIC system cannot provide sufficient flow to keep the core covered. Also, depending on the initiating event, MSIV closure may occur and the two turbine-driven feedwater pumps would be initially unavailable.

The heat removal function success criteria are highly dependent on the initiating event and the actions taken. If the initiating event results in the unavailability of the PCS, the entire heat load is dumped to the suppression pool. The suppression pool heat load can range from 9%-17%, if reactor subcriticality is not successful, to 1% twenty minutes after boron injection begins. Operation of one RHR loop is assumed to be required to remove the integrated heat load after reactor subcriticality is successful. One RHR train will turn around the containment temperature and pressure increases and begin to return containment conditions to normal. Only the suppression pool cooling and containment spray modes are assumed to be successful. If reactor subcriticality is not successful, operation of a least one train of RHR can result in equilibration of containment pressure and temperatures at levels consistent with the the ADS reclosure pressure, if only low pressure injection systems are working, or slightly above vent pressures if high pressure injection and venting work. Venting by itself will also result in equilibration of containment conditions at levels near the vent setpoint after venting occurs.

If the PCS is available, the majority of the heat load will be transferred to the condenser. However, for a period of time, the heat load in excess of the turbine bypass capacity (25%) would be transferred to the suppression pool. This heat load is assumed to be below the heat capacity of the suppression pool and heat removal is not required.

2.6 Systemic Event Trees

2.6.1 LOCAs

As stated previously, the three LOCA initiating events are evaluated on a single LOCA event tree. This is possible since the general plant response is similar for all three sizes of LOCAs. However, the success criteria for safety-related systems varies with the size of the LOCA. The difference in the success criteria is accounted for by inclusion of the initiating events in the system fault trees.

A description of the plant response for each of the three LOCA sizes is presented in the subsequent paragraphs. The LOCA event tree is then presented with descriptions for each event. The initiating event interactions with the systems are discussed at this level.

A: Large LOCA

A large LOCA is any break in the reactor coolant system piping which could lead to the loss of a sufficient amount of coolant to result in a rapid depressurization of the reactor system. A large LOCA demands that the reactor be scramed, coolant released to the containment be condensed, coolant makeup be supplied to the vessel, and heat be removed from the containment.

Following initiation of a large LOCA, the drywell pressure should increase rapidly to the reactor scram setpoint of 1.69 psig. In addition, the reactor vessel water level should decrease to the reactor scram setpoint (level 3, 12.5 inches). The HPCS, LPCS, and LPCI systems, and all diesel generators (DGs) are also signaled to start at the high drywell pressure setpoint (1.69 psig).

The RCIC system should initiate upon a low reactor vessel water level signal (Level 2, -50 inches). However, RCIC capacity is insufficient for a large LOCA. Furthermore, steam to the RCIC turbine is isolated when the reactor vessel pressure decreases to 57 psig. This should occur rapidly following a large LOCA. The HPCS system and the HPCS D/G also receive initiation signals to start upon a Level 2 signal. The recirculation pumps are tripped.

The water level in the vessel continues to decrease to the low-low level setpoint (Level 1) of -129 inches which should initiate MSIV closure. Closure of the MSIVs should result in loss of steam to both turbine-driven feedwater pumps. Automatic startup of the motor-driven pump will not occur since the TDRFPs do not actually trip.

The LPCI and LPCS systems will receive a second signal to start at the Level 1 setpoint (i.e., in addition to the high drywell pressure). In addition, the ADS timer should begin its 105 second run to begin depressurization. (ADS is not required because a large break will depressurize the vessel.) The reactor vessel pressure should decrease to

the LPCS and LPCI injection valve opening setpoint of 500 psig. Within 60 sec, the HPCS, LPCS, and LPCI systems should all be injecting into the vessel.

After the reactor vessel level is stabilized, the reactor operator should initiate containment heat removal using the RHR. (An interlock prevents flow through the RHR heat exchangers for the first ten minutes following LPCI initiation.)

S_1: Medium LOCA

A medium LOCA is of a size such that rapid vessel depressurization does not occur. Therefore a high pressure coolant injection system is required or the vessel must be depressurized. The size of a medium LOCA is dependent upon location. A liquid break between .0005 and 0.3 ft^2 or a steam break in the range 0.1 to 0.3 ft^2 will result in a medium LOCA.

The sequence of events following a medium LOCA would be similar to those delineated for a large LOCA except ADS would be required to depressurize the vessel.

S_2: Small LOCA

A small LOCA is characterized by slow or no vessel depressurization and a gradual inventory loss from the vessel. The high pressure coolant makeup systems including RCIC can be utilized to mitigate a small LOCA. A small LOCA is defined as a liquid break less than or equal to 0.0005 ft^2 or a steam break ≤ 0.1 ft^2.

Immediately after a small break in the reactor pressure boundary, the vessel pressure may slowly decrease while the drywell pressure will increase. The increase in drywell pressure will provide a scram signal to the RPS and will also initiate the ECCS. At this time, RPV pressure may increase, depending on the size of the small LOCA, due to the imbalance caused by the turbine trip and some SRVs may open. For all small breaks, the feedwater capacity should be capable of compensating for any reactor vessel inventory lost out the break. Furthermore, HPCS will begin injecting water after initiation of a high drywell pressure signal. (LPCS and LPCI will also initiate.)

If the reactor vessel water level continues to decrease, a low water level (Level 2) signal will initiate RCIC and send a second initiate signal to HPCS. A further decrease to the low-low level (Level 1) will result in a second initiation signal to LPCI and LPCS and begin the ADS 105 sec rundown. MSIV closure will also occur.

MSIV closure can also occur due to other signals related to the initiating event. These signals include main steamline high temperature, radiation, or low pressure. Because of this potential for MSIV closure, the PCS is conservatively assumed unavailable for a small LOCA.

The sequence of events following a small LOCA are similar to those delineated for a medium LOCA except that RCIC can provide adequate coolant makeup.

The LTAS code developed at Oak Ridge National Laboratory (ORNL) was modified to represent the LaSalle plant. The following modifications were made by ORNL: (1) passive heat sinks for the Mark-II containment were added, (2) the RHR heat exchanger model was improved, and (3) a capacity to model small break LOCAs was added. Additional modifications made by SNL include: (1) additional SRVs and LaSalle control logic were added, (2) all injection systems were added with LaSalle-specific head curves, control logic, and injection location, (3) capability to simulate enhanced CRD flow, (4) drywell venting capability, (5) stuck-open vacuum breaker capability, and (6) LaSalle-specific core characteristics. The code was base-lined to a RELAP5 model used to evaluate transient response (see Section 2.6.2). One RELAP5 calculation and twelve small break and three medium break LTAS calculations were performed. The following cases were run:

1. A small LOCA with successful ADS but with no injection (0.6 and 1 in. diameter breaks),

2. A small LOCA with RCIC success, with and without ADS,

3. A small LOCA, RCIC and 1 CRD pump operating, no ADS or containment venting,

4. A small LOCA, MFW success, level controlled to TAF, no ADS, venting occurs,

5. A small LOCA, HPCS injection fails when suppression pool temperature reaches 300 °F, no ADS or venting,

6. A small LOCA, HPCS injection fails when suppression pool temperature reaches 300 °F, no ADS or venting, no late manual RPV depressurization,

7. A small LOCA, RCIC and 2 CRD pumps working, start second CRD pump at 12,000 sec, no ADS or venting,

8. A small LOCA, ADS and LPCS work,

9. A small LOCA, RCIC and 2 CRD pumps working, enhanced CRD flow when second pump started at 12,000 sec., two flow rates: maximum and 180 gpm,

10. A medium LOCA: 0.3 ft^2, no injection, no ADS,

11. A medium LOCA, HPCS injection,

12. A medium LOCA, MFW injection, level controlled to TAF, and

13. RELAP5, recirculation line break with one LPCI train operating.

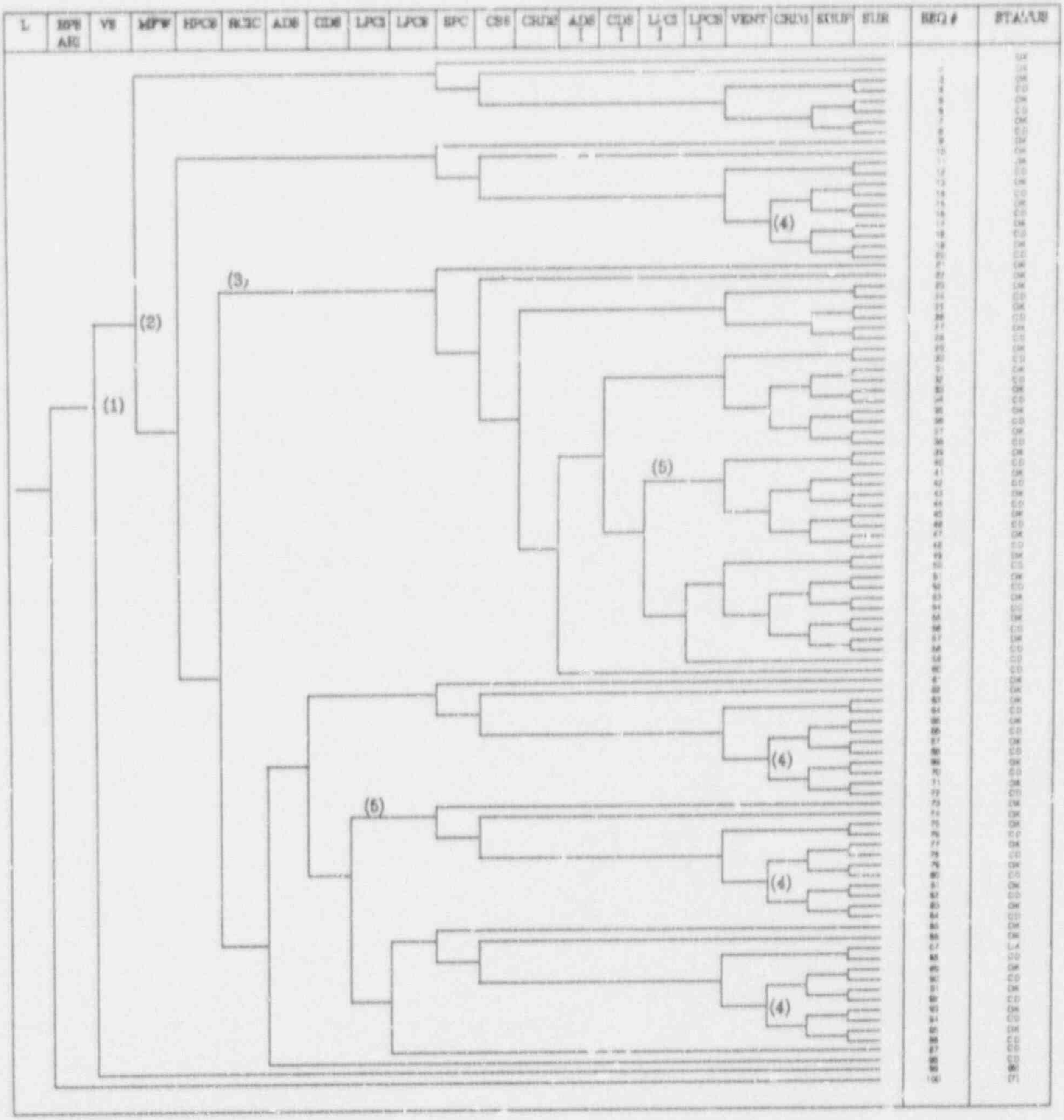
The event tree for a LOCA is shown in Figure 2.4. Each event is described subsequently.

L - LOCA Initiator - This initiator represents all different sizes of possible LOCAs. A large LOCA is defined as a large steam line break or liquid line break sufficient to depressurize the reactor vessel so that low pressure injection systems can prevent core damage. A medium LOCA is a steam or liquid line break which does not depressurize the reactor vessel rapidly enough to prevent core damage using low pressure injection systems without ADS operation. A small LOCA is a break small enough such that reactor vessel depressurization does not occur rapidly and RCIC is capable of supplying sufficient makeup to prevent core uncovery.

RPS/ARI - Reactor Subcritical - A reactor scram signal should be initiated by a high drywell pressure signal and the control rods should insert. A second scram signal should be generated upon low reactor water level. Failure of any 5 adjacent rods to insert to notch position 06 or thirty or more rods to insert to notch position 06 will result in failure to scram. Credit is given for the alternate rod insertion system, electrical failures are assessed to be negligible and only random mechanical failures are used in the quantification of this event. No fault tree was developed for this analysis. The event was quantified as a single event. Failure to automatically scram will transfer to the ATWS event tree.

VS - Vapor Suppression - The steam released into the drywell during a LOCA must flow down into the vapor suppression pool for condensation in order to prevent overpressurization of the containment. As was assumed in the Reactor Safety Study,[6] the definition for failure of this event is steam bypass of the suppression pool (i.e., direct steam inflow into the wetwell airspace). This is assumed to occur if one downcomer ruptures in the wetwell airspace, if two vacuum breakers rupture, or if floor or penetration seals between the drywell and wetwell rupture. For transient-induced LOCAs that result from a stuck-open SRV, failure of the SRV discharge line in the wetwell air space will produce similar effects.[8]

Failure of the vapor suppression system during a large LOCA can result in containment failure in less than one minute.[6] Insufficient time would be available for the operator to initiate drywell or suppression pool sprays, since he would be locked out for 10 minutes by the actuation logic, or vent the containment. Containment failure can potentially result in failure of the emergency core cooling systems and thus may result in core damage. For small and medium LOCAs and transient-induced LOCAs, more time may be available to initiate sprays given steam bypass of the suppression pool. If the bypass level is sufficiently small, greater than 10 minutes may be available and containment sprays might be used to mitigate the pressure rise. The required actions are proceduralized in LGA-03: "Containment Control".[9]



(1) TRANSFER FROM TRANSIENT SEQUENCE # 103.

(2) TRANSFER FROM TRANSIENT SEQUENCE # 102.

(3) RCIC SUCCESS POSSIBLE FOR SMALL LOCA ONLY.

(4) CRD SUCCESS POSSIBLE FOR SMALL LOCA OR STEAM BREAK ONLY.

(5) FOR VERY LONG-TERM SEQUENCES WITH A LARGE LOCA WHERE THE CORE IS AT 2/3 TAF MAY GET SUBCOOLING AND MELT THE TOP OF THE CORE IF ONLY ONE LPCI PUMP IS OPERATING.

(6) TRANSFERS TO (2), DOWNCOMER, VACUUM BREAKER, OR SRV DISCHARGE LINE FAILURE, SAME SYSTEM SUCCESS CRITERIA, SEQUENCE OCCURES IN SHORTER TIME.

(7) TRANSFER TO ATWS TREE.

Figure 2.4 LaSalle LOCA Systemic Event Tree

Because of the low probability of failure combined with the low LOCA initiating event probabilities, these sequences were not developed further. However, they would evolve just like the sequences with VS success but with much shorter containment pressurization times.

MFW - Feedwater Available - The turbine-driven feedwater pumps would coastdown following a LOCA since MSIV closure would occur. Thus the motor-driven feedwater pump must be manually started to supply coolant to the reactor vessel during a LOCA. The capacity of this pump is sufficient for maintaining the reactor vessel water level for any size LOCA. Failure of this system is defined as failure of the feedwater flow path, the feedwater pump, or failure of all four condensate and condensate booster pumps to provide flow to the feedwater pump from the condenser hotwell.

For a feedwater injection line break, the availability of feedwater is not known. Depending upon the size and location of the break, it is possible that a sufficient amount of feedwater may be diverted out the break rendering feedwater insufficient by itself for maintaining the reactor vessel water level. Since there are two injection paths, if the break in one is large enough to divert significant flow, the vessel would be completely depressurized and about half the flow should go to the vessel. Even if less flow went to the vessel, it does not require a large percentage to be successful given the total feedwater flow rate. After the depletion of the condenser hotwell when makeup flow limits the feedwater injection rate to 1800 gpm, the affected path can be isolated (i.e., isolate the injection path with the break, force water into the vessel through the other injection line, and then out the line with the break).

Operation of the feedwater system for providing makeup following a LOCA would be terminated following initiation of the RHR system for containment heat removal and recirculation back into the vessel.

HPCS - HPCS Available - The HPCS system has sufficient capacity for maintaining the reactor vessel water level by itself for any size LOCA. Failure of the HPCS system is defined as failure of the single motor-driven pump to provide sufficient flow to the reactor vessel from the condensate storage tank. Failure can also occur if the HPCS system fails to automatically realign to the suppression pool when the condensate storage tank level is low.

RCIC - RCIC Available - The single steam-driven RCIC train has sufficient capacity for coolant makeup following a small LOCA only. Failure of RCIC is defined as failure of the single pump to provide flow to the reactor vessel from the condensate storage tank. Failure can also occur if the RCIC system fails to automatically realign to the suppression pool when the condensate storage tank level is low.

ADS - Reactor Vessel Depressurization - This event consists of either automatic or manual depressurization of the reactor vessel to allow injection from the low pressure ECCS during a small or medium LOCA.

Automatic depressurization would occur by the ADS if the reactor vessel water level decreased to Level 1 due to failure of the high pressure systems. Sufficient time would also be available for the operator to manually depressurize the vessel using the SRVs if the ADS failed. In both cases, successful depressurization requires the opening of 3 valves. Note that for large LOCAs, reactor vessel depressurization occurs due to the initiator.

<u>CDS - Condensate Available</u> - The condensate system is designed to remain running even though the feedwater system may trip. Water from the condensate pumps will flow through the feedwater control valves to the vessel. One condensate/condensate-booster train can provide sufficient coolant to the vessel during a LOCA. The coolant is provided from the condenser hotwell. Makeup to the hotwell must be provided by the condensate makeup system which is limited to 1800 gpm and supplies water from the condensate storage tank.

<u>LPCI - LPCI Available</u> - Any one of the three LPCI trains can be successfully used to mitigate a large LOCA. The pumps take suction from the suppression pool and initially bypass the RHR heat exchangers.

<u>LPCS - LPCS Available</u> - The single train LPCS system has sufficient capacity for coolant makeup following a large LOCA. Failure of the LPCS system is defined as failure of the single motor-driven pump to provide flow to the reactor vessel from the suppression pool.

<u>SPC - Suppression Pool Cooling</u> - The energy released during a LOCA will be transported to the containment where most of it should be absorbed by the suppression pool. The operator is directed by procedures to initiate suppression pool cooling when the pool temperature reaches 100 °F. If LPCI has been initiated, an interlock will prevent switching from LPCI to suppression pool cooling for ten minutes.

<u>CSS - Containment Spray</u> - In the unlikely event that suppression pool cooling fails, the primary containment pressure and temperature will continue to increase. The operator is directed by procedures to initiate suppression pool sprays when the suppression chamber pressure exceeds 19.5 psig. The operator is also directed to initiate drywell sprays if containment pressure can not be maintained below 60 psig. Suppression pool sprays were not modeled as a separate system in this analysis. Drywell sprays were used to represent both spray modes.

<u>CRD2 - Intermediate Control Rod Drive</u> - For small LOCAs, in those cases where the main feedwater and HPCS systems have failed and the RCIC system is the only high-volume high-pressure system working, if containment heat removal fails, then RCIC will isolate when containment pressure reaches 40 psia or may fail due to loss of DC power (for station blackout type sequences) or high suppression pool temperature (i.e., greater than 250 °F). By this time the decay heat load is low enough that the CRD system can supply sufficient makeup to maintain the core level high enough to prevent core damage if AC power is available and both pumps are working.

ADS I - Intermediate Reactor Vessel Depressurization - For small LOCAs with initial RCIC success, this event consists of either automatic or manual depressurization of the reactor vessel to allow injection from the low pressure ECCS when delayed RCIC failure occurs after about 6 hours. All other high pressure injection systems have failed and so the operator must depressurize and use low pressure injection systems to prevent core damage (vessel pressure may already be fairly low due to the initiator, ADS is needed to maintain low pressure when the high-volume low-pressure systems initiate). Automatic depressurization would occur by the ADS if the reactor vessel level decreased to Level 1 due to failure of the high pressure systems. Sufficient time would also be available for the operator to manually depressurize the vessel using the SRVs if the ADS failed. In both cases, successful depressuri tion requires opening of 3 valves.

CDS I - Intermediate Condensate Available - For the small LOCA case, if all the high pressure injection systems have failed except RCIC and then RCIC fails after about 6 hours, the operator will depressurize the RPV and attempt to use low pressure injection systems. If AC power is available and the system status has not been accounted for before in the analysis of the sequence, then the condensate system may be used to maintain core cooling. Water from the condensate pumps will flow through the feedwater control valves to the vessel. One condensate/condensate-booster train can provide sufficient coolant to the vessel during a LOCA. The coolant is provided from the condenser hotwell. Makeup to the hotwell must be provided by the condensate makeup system which is limited to 1800 gpm and supplies water from the condensate storage tank.

LPCI I - Intermediate LPCI Available - For the small LOCA case, if all the high pressure injection systems have failed except RCIC and then RCIC fails after about 6 hours, then the operator will depressurize the RPV and attempt to use low pressure injection systems. If AC power is available and the system has not been accounted for before, then any one of the three LPCI trains can be successfully used to mitigate the LOCA. The pumps take suction from the suppression pool and initially bypass the RHR heat exchangers.

LPCS I - Intermediate LPCS Available - For the small LOCA case, if all the high pressure injection systems have failed except RCIC and then RCIC fails after about 6 hours, then the operator will depressurize the RPV and attempt to use low pressure injection systems. If AC power is available and the system has not been used before, then the single train LPCS system has sufficient capacity to mitigate the LOCA. Failure of the LPCS system is defined as failure of the single motor-driven pump to provide flow to the reactor vessel from the suppression pool.

VENT - Containment Venting - After containment heat removal failure, the containment pressure will continue to increase due to the decay heat being dumped to the suppression pool. I. he containment heat removal systems can not be recovered, then when the containment pressure reaches 60 psig, the operator is directed to vent the containment through the

Standby Gas Treatment System. Containment venting can occur during any size LOCA if all containment heat removal systems fail.

CRD1 - Late Control Rod Drive - If containment heat removal has failed and the operator does not vent the containment when containment pressure reaches 60 psig, then for the small LOCA case, the CRD system may be used to maintain core cooling with only one pump operating. For cases with only low pressure injection working, the ADS valves will close when containment pressure reaches about 85 psig due to inadequate differential pressure; however, since a LOCA exists, the RPV will not repressurize. The low pressure injection systems (LPCI and LPCS) may fail due to high suppression pool temperatures that result in pump failure. For the case of HPCS supplying high pressure injection, since the HPCS suction will be from the suppression pool by this time, a similar result may occur.

SRUP - Containment Failure Mode - If the containment heat removal function has failed and the containment is not vented, then the containment pressure will continue to rise until structural failure of the containment occurs. This event asks, roughly, how large the break in containment is, either a leak (success branch) or a rupture (failure branch). For the purposes of this analysis, we need to know how fast the containment depressurizes, whether or not we can use any low pressure injection system to maintain core cooling after the containment failure, and how long the containment blowdown will last in order to evaluate the severity of the reactor building environments produced by the blowdown. A leak implies that the containment pressure remains too high to use the ADS system to depressurize the RPV and the low pressure injection systems to prevent core damage; while a rupture implies that the containment depressurizes fast enough that successful operation of the ADS and a low pressure injection system will prevent core damage. For LOCA initiators, the RPV will remain depressurized this late in the accident so low pressure injection can continue to be used through the containment pressurization and failure. An expert judgement elicitation was performed to generate probability distributions for the containment failure pressure, location, and size.

SUR - Injection System Survival - After containment failure, severe environments will be produced in the reactor building. Many of the injection systems have components at various positions in the reactor building and these components will be exposed to these severe environments. Simple Boolean equations were generated to represent each system that had components that might be subject to these environments and an expert judgement elicitation was performed for various component types and environments in order to quantify the failure probabilities.

2.6.2 Transients With Scram

The eight transient initiating event categories and ten special transient initiating event categories identified in this study are delineated in a single transient event tree. The success criteria for the systems

required to mitigate each transient can vary. This variation in the success criteria is accounted for by including the specific effects of the initiator on the responding systems in the system fault trees in a manner that appropriately models the initiator's impact on the system response.

The sequence of events following each transient initiator is described in the subsequent paragraphs. The transient event tree is then described with differences in the system success criteria clearly delineated.

T_1: Turbine Trip with Turbine Bypass Available

A variety of turbine system malfunctions will initiate a turbine trip. Some examples include moisture separator and heater drain tank high levels, large turbine vibrations, and loss of control fluid pressure. Only those turbine trips in which steam bypass to the condenser is still available are included here. Turbine trips without turbine bypass and loss of condenser are treated as separate events.

Turbine trips can occur at any power level. For power levels less than 25%, reactor scram is not necessary if the turbine bypass is available and sufficient makeup is provided to the vessel. For power levels greater than 25% (the turbine bypass rating), automatic reactor scram should occur. Table 2.6 delineates the events following a turbine trip from high power (i.e., 100% power).

The sequence of events for a turbine trip with bypass was determined in computer code simulations reported in Chapter 15 of the FSAR.[3] The sequence of events is shown in Table 2.6. As indicated, a turbine trip is followed by automatic scram, turbine stop valve closure, and recirculation pump trip. The turbine bypass valves will open to regulate the pressure but opening of relief valves is also required to relieve excess pressure. Feedwater will continue to run and the vessel water level will swell to Level 8, at which point an automatic feedwater trip occurs.

The water level in the vessel will begin to drop and the operator is directed by LaSalle Procedure LOA-TG-04[10] to maintain the reactor water level above Level 4 by using the feedwater system if possible. The motor-driven reactor feedwater pump (MDRFP) will not automatically start on tripping of the turbine-driven reactor feedwater pumps (TDRFP) due to the Level 8 signal. If motor-driven feedwater flow is not established within 30 seconds, the reactor water level will decrease to Level 2 where both RCIC and HPCS would be automatically initiated. MSIV closure is also indicated in Table 2.6 as occurring at Level 2. However, the MSIV closure setpoint was subsequently lowered to Level 1. Thus the remaining sequence of events shown in Table 2.6 is not applicable.

If sufficient coolant makeup is not provided in time, the reactor water level will continue to decrease to Level 1 where LPCI and LPCS would automatically initiate and MSIV closure would occur. ADS would occur on

Table 2.6
SEQUENCE OF EVENTS FOLLOWING A TURBINE TRIP WITH BYPASS

Time (sec)	Event
0	Turbine trip initiates closure of main stop valves. Turbine trip initiates bypass operation
0.01	Main turbine stop valves reach 90% open position and initiate reactor scram trip. Main turbine stop valves reach 90% open position and initiate a recirculation pump trip (RPT).
0.10	Turbine stop valves closed. Turbine bypass valves start to open to regulate pressure.
0.19	Recirculation pump motor circuit breakers open causing a decrease in core flow to natural circulation.
1.55, 1.69, 1.84, 2.02, and 2.30	Relief valves actuated sequentially by groups: 1, 2, 3, 4, and 5.
4.53	Feedwater trip on high water lever (L8).
(Est) 5.2, 5.5, 5 9, 6.2 and 7.0	Relief valves close sequentially by groups: 5, 4, 3, 2, and 1.
30.6	Bypass valve begins to close on pressure signal.
32.2 (est)	Turbine bypass closed.
38.67	Low level trip (L2) initiates a main steam line isolation. Low level trip (L2) initiates RCIC and HPCS (not simulated).
39.23	Turbine bypass reopens on pressure increase at turbine inlet.

low-low level and 8 minute confirmatory low level. It would take approximately six minutes for the water level to decrease from Level 2 to Level 1.[4] During this time period, the operator could manually try to initiate HPCS and RCIC and, failing to do so, manually depressurize the vessel and use low pressure injection.

Subsequent to MSIV closure, the reactor can be de-isolated to provide access to the main condenser for containment heat removal. This can be performed if the RHR system fails.

T_2: Turbine Trip With Turbine Bypass Unavailable

A turbine trip without bypass is a turbine bypass initiator coupled with a failure of the turbine bypass valves to open. This combination of events results in the unavailability of the PCS for decay heat removal.

The sequence of events for a turbine trip without bypass was determined in computer code simulations and was reported in the FSAR.[3] The sequence of events is shown in Table 2.7. As indicated, the sequence of events is similar to that of a turbine trip with bypass with the difference being in the repeated opening of safety relief valves. With the PCS unavailable, periodic relief valve operation is required to reduce the pressure.

T_3: Total Main Steam Isolation Valve Closure

Inadvertent MSIV closure can occur due to manual action, spurious signals such as low reactor water level or low condenser vacuum, low steamline pressure, or due to MSIV failure. Closure of one MSIV will not initiate a reactor scram. The main steamlines are sized to carry full rated steam flow with one line closed. Closure of three or more MSIVs will initiate a reactor scram. The following discussion delineates the sequence of events following closure of all MSIVs.

The sequence of events following an MSIV closure event was determined in computer code simulations and reported in Chapter 15 of the LaSalle FSAR.[3] The sequence of events is shown in Table 2.8. As indicated, MSIV closure results in periodic pressure increases requiring relief valve operation. Flow from the turbine-driven feedwater pumps is lost due to the loss of steam flow to their turbines. Actual tripping of the turbine-driven feedwater pumps (and thus starting of the motor-driven feedwater pump) does not occur.

The water level in the vessel will drop to Level 2 and automatically initiate RCIC and HPCS. If RCIC and HPCS fail to automatically initiate, the operator can try to manually initiate these systems or initiate motor-driven feedwater manually. If these actions fail, the water level will continue to decrease and the low pressure ECCS will automatically initiate. ADS will occur on low-low level and confirmatory low level. The operator may manually depressurize the vessel to allow injection from the low pressure systems if ADS fails.

Table 2.7
SEQUENCE OF EVENTS FOLLOWING A TURBINE TRIP WITHOUT BYPASS

Time (sec)	Event
0	Turbine trip initiates closure of main stop valves.
	Turbine bypass valves fail to operate
0.011	Main turbine stop valves reach 90% open position and initiate reactor scram trip and a recirculation pump trip (RPT).
0.1	Turbine stop valves closed
0.19	Recirculation pump motor circuit breakers open causing decrease in core flow to natural circulation.
1.13, 1.24, 1.36, 1.50, and 1.65	Relief valves actuated sequentially by groups: 1, 2, 3, 4, and 5.
5.98	L8 trip initiates a feedwater pump trip.
(Est) 7.3, 7.7, 8.0, 8.3 and 9.2	Relief valves close sequentially by groups: 5, 4, 3, 2, and 1.
Through 25	Re-actuation of relief valves (one or two) to relieve pressure.
26.70	Low level trip (L2) initiates a main steamline isolation.
	Low level trip (L2) initiates RCIC and HPCS.

Table 2.8
SEQUENCE OF EVENTS FOLLOWING AN MSIV CLOSURE

Time (sec)	Event
0	Initiate closure of all main steamline isolation valves (MSIV).
0.3	MSIV's reach 90% open
0.3	MSIV position trip scram initiated.
1.6	Loss of feedwater begins as turbine loses steam supply.
2.42, 2.51, 2.60, 2.70, and 2.80	Relief valves actuated by groups 1, 2, 3, 4, and 5.
3.0	All main steam line isolation valves closed.
4.47	Recirculation runback on low level alarm (L4) and feedwater flow <20%.
Est. 6, 9, 7.3, 7.6, 7.9, and 8.8	Relief valves reclose by groups 5, 4, 3, 2, and 1.
10.35	Group 1 relief valves re-actuate on high pressure.
10.90	Group 2 relief valves re-actuate on high pressure
Est. 15.9, 17.2	Relief valves reclose by groups 2, and 1.
18.7	Vessel water low level trip (L2) initiates recirculation pump trip. Vessel water low level trip (L2) initiates RCIC & HPCS (not simulated).
20.84	Group 1 relief valves cycle open and closed on pressure.

With the MSIVs closed, the PCS is unavailable and decay heat must be removed by the RHR system in one of its operating modes. Reopening of the MSIVs is not considered possible since the reason for the closing may preclude quick recovery.

T_4: Loss of Normal Condenser Vacuum

A loss of condenser vacuum can occur due to some single equipment failures. The vacuum decay rate is a function of the specific cause of failure. Some failures and the estimated decay rates are listed below.

Cause	Estimated Decay Rate
Failure or isolation of steam jet air ejectors	< 1 inch Hg/min.
Loss of sealing steam to shaft gland seals	1 to 2 inches Hg/min.
Opening of vacuum breaker valves	2 to 12 inches Hg/min.
Loss of one or more circulating water pumps	4 to 24 inches Hg/min.

The sequence of events following a loss of condenser vacuum was determined in computer code simulations reported in Chapter 15 of the LaSalle FSAR.[3] The calculated sequence of events for a scenario involving a vacuum decay rate of 2 inches Hg/sec is shown in Table 2.9. As indicated, once the condenser vacuum decreases to 21.6 inches Hg, the main steam stop valves and the TDRFPs trip. Reactor scram is initiated by the main steam stop valve closure. The main steam stop valve closure also initiates turbine bypass valve opening but the continued loss of vacuum (7 inch Hg) results in their closure as well as MSIV closure.

The turbine-driven feedwater pump trip can result in automatic initiation of the motor-driven feedwater pump. The MDRFP should be up to speed within 5 seconds (note that the MDRFP was not modeled in the Chapter 15 analysis) and should maintain the reactor vessel water level above the Level 2 setpoint. If, however, the MDRFP fails to start, RCIC and HPCS should be initiated when the reactor vessel water level reaches Level 2.

If all the high pressure coolant systems fail to operate, the reactor water level will continue to decrease. At a Level 1 setpoint, LPCS and LPCI will automatically initiate and ADS will occur on low-low level and confirmatory low level. If ADS fails, the operator can manually depressurize the vessel. Flow from the condensate system through the feedwater system to the vessel can also occur once the vessel is depressurized.

With the reactor vessel isolated due to MSIV closure, the decay heat in the core following scram will be removed by periodic opening of the

Table 2.9
SEQUENCE OF EVENTS FOLLOWING A LOSS OF CONDENSER VACUUM

Time (sec)	Event
0	Initiate simulated loss of condenser vacuum at 2 inches Hg/sec.
5.00	Low condenser vacuum main turbine trip initiated. Turbine trip initiates feedwater trip. Main turbine trip initiates turbine bypass valve operation.
5.01	Main turbine stop valves reach 90% open position and initiat reactor scram trip and recirculation pump trip. (RPT) Turbine stop valves closed and turbine bypass valves start to open to regulate pressure.
5.14	Recirculation pump motor circuit breakers open causing decrease in core flow to natural circulation.
6.65, 6.81, 6.98, and 7.18	Relief valves automatically actuate by Groups 1, 2, 3, and 4.
10.00	Low condenser vacuum initiates turbine bypass valve closure and main steamline isolation valve closure.
Est. 10.3	Turbine bypass valve(s) close.
Est. 11.0, 11.3, 11.9 and 12.7	Relief valves reclose by Groups 4, 3, 2, and 1.
13.42 and 13.85	Pressure relief valves reopen by Groups 1 and 2.
Est. 20.1	Group 2 relief valves close.
Est. 23.8	Low vessel level (L2) trip initiates RCIC and HPCS (not simulated)
Est. 32.2	Group 1 relief valves close.
40.85	Group 1 relief valves cycle open and closed on pressure.

safety relief valves. The decay heat will be transported to and stored in the suppression pool. The decay heat must be ultimately removed by use of one of the operating modes of the RHR system. No credit for reopening of the MSIVs or use of the PCS is taken since it is dependent on the severity of the transient initiator.

T_5 and T_6: Total Loss of Feedwater and Partial Loss of Feedwater

A complete or partial loss of feedwater can occur from pump failures, condensate system failure, interruption of driving steam flow, feedwater controller failures, operator error, or a spurious high reactor vessel water level signal. Some of these initiators will not only trip the turbine-driven feedwater pumps but will also prevent startup of the motor-driven feedwater pumps.

The sequence of events following failure of one feedwater pump is different from the sequence of events following loss of all feedwater. Both initiators are discussed in this section with the differences in events clearly delineated. A loss of complete or partial feedwater results in a decreased subcooling in the core and a subsequent reduction in core power and pressure. The water level in the core drops to Level 3 initiating an automatic scram. Following scram, the turbine bypass valves would open and begin controlling the reactor pressure. When only one feedwater pump fails, the other two should be available for coolant injection. However, if all the feedwater pumps fail, the reactor water level would drop to Level 2 initiating RCIC and HPCS and tripping the recirculation pumps.

If both RCIC and HPCS fail, the reactor water level would continue to decrease to Level 1. LPCS and LPCI would automatically initiate and MSIV closure would occur. ADS would occur on low-low level and confirmatory low level. If ADS fails, the operator could manually open the SRVs to depressurize the reactor and allow low pressure coolant injection.

Subsequent to MSIV closure, the reactor can be de-isolated to provide access to the main condenser for heat removal.

T_7 - Inadvertent Opening of a Safety/Relief Valve

An inadvertent open relief valve or IORV is different from a transient with an SORV in that an immediate reactor scram does not occur. Thus approximately 6% of the generated steam flow would be released to the suppression pool. The suppression pool temperature would increase at a rate of approximately 2 °F per minute.[3]

An IORV may be detected by any one of numerous indications in the control room. These include SRV open alarms, suppression pool level or temperature indications, or an SRV downcomer piping temperature indication. Once detected, the operator is directed to try to close the valve. If he cannot close the valve within two minutes or the suppression pool temperature is 110 °F, the operator is to scram the reactor.

If the operator fails to manually scram the reactor, the continued blowdown will increase the suppression pool temperature and drywell pressure. Failure to manually scram the reactor before the suppression pool temperature exceeds 110 °F can lead to condensation instability due to localized effects around the quencher. It is expected that the drywell pressure would increase to 1.69 psig and result in an automatic reactor scram signal. It is expected that high containment pressure would occur approximately 20 minutes into the transient and would result in high suppression pool temperatures (135 °F at 20 minutes).

Following reactor scram, the reactor vessel pressure would decrease to the MSIV closure setpoint. It is assumed that the operator is unable to inhibit the low pressure closure in time by turning the reactor mode switch to SHUTDOWN. MSIV closure results in loss of steam to the turbine-driven feedwater pumps and a pump coastdown. Actual tripping of the turbine-driven feedwater pumps will not occur and thus automatic initiation of the motor-driven feedwater pump will also not occur.

The water level in the vessel will drop to Level 2 resulting in an auto start to both RCIC and HPCS. If both these systems fail to provide sufficient flow to the vessel, the operator may try to manually initiate the motor-driven feedwater pump. If he does not, the reactor vessel water level will drop to Level 1 resulting in automatic initiation of LPCI and LPCS. If the drywell pressure is high, automatic depressurization of the vessel will occur. If the high drywell pressure signal is not available, then low-low level with confirmatory low level should result in ADS initiation. If ADS fails, the operator could manually depressurize the vessel to allow low pressure system injection.

Following an IORV initiator with MSIV closure assumed, all the decay heat would be dumped to the suppression pool. Operation of the RHR would be required to remove this heat. No credit is currently taken for the steam condensing mode of RHR for this initiator.

T_E - Loss of Offsite Power

Loss of offsite power is an event which affects the balance of plant systems operation. Initiation of the diesel generators should occur automatically to provide power to required safety systems.

The sequence of events following a loss of offsite power event was determined in computer code simulations and reported in Chapter 15 of the LaSalle FSAR.[3] The sequence of events are shown in Table 2.10. The loss of power results in a load rejection and turbine trip. Main steam stop valve closure initiates a reactor scram. All balance of plant systems requiring AC power trip off. This includes the condensate circulating water system, and the recirculation pumps.

The loss of circulating water results in a subsequent loss of condenser vacuum which initiates MSIV closure. The turbine-driven feedwater pumps stop due to the loss of steam flow and also trip due to a loss of

Table 2.10
SEQUENCE OF EVENTS FOLLOWING A LOSS OF OFFSITE POWER

Time (sec)	Event
Approx - 0.015	Loss of grid causes turbine-generator to detect a loss of electrical load.
0	Turbine trip initiated by loss of generator load. Turbine-generator PLU trip initiates main turbine control valve fast closure. Recirculation system pump motors trip off. Circulating water pump trip. Condensate and condensate booster pump trip. Turbine stop valve closure initiates reactor scram. Electric feedwater pump motor is tripped.
0.01	Turbine control valves closed.
0.10	Turbine steam bypass valves open to regulate pressure.
1.61, 1.76, 1.92, 2.12 and 2.56	Relief valves actuated sequentially by Groups 1, 2, 3, 4, and 5.
Est. 5.1, 5.4, 5.8, 6.0 and 6.9	Relief valves reclose sequenti. by Groups 5, 4, 3, 2, and 1.
30	Loss of condenser vacuum initiates MSIV closure and turbine steam bypass valve(s) closure.
32.4	Reactor vessel low Level 2 trip initiates HPCS and RCIC (not simulated).
50 plus	Group 1 relief valves automatically cycle to regulate pressure.

condenser vacuum trip. The reactor vessel water level decreases to the Level 2 setpoint where RCIC and HPCS are initiated. If both RCIC and HPCS fail, the reactor vessel water level will continue to decrease to Level 1 resulting in automatic initiation of LPCS and LPCI. ADS will occur on low-low level and confirmatory low level. The operator could manually depressurize the RPV should auto-depressurization fail.

Decay heat removal would be achieved by the RHR system in one of its operating modes. The main condenser would be unavailable following a loss of offsite power initiator due to loss of the circulating water system.

In the case of complete loss of AC power (i.e., station blackout, loss of offsite power and failure of the diesel generators), coolant makeup can still be provided by the steam-driven RCIC system. However, RCIC requires DC power for control of the system and room cooling for the pump. DC power is provided by a dedicated battery in case of a station blackout and is designed to provide the necessary [illegible] for a period of about 6 hours. If AC power is not restored during the period or other recovery actions taken, RCIC will fail on DC power failure.

The RCIC pump room cooling system requires AC power for operation; however, the high temperature trip is inoperative if both train A and B of onsite AC power have failed. In this case, the RCIC system can operate without room cooling. However, upon loss of offsite power, the RCIC steam line is isolated if onsite AC power is restored using the DGs. This is due to a sneak circuit which results in an isolation signal on the train to which onsite power is restored. If onsite power continues to work, the operator can reopen the isolation valves and restore RCIC operation. However, if the DG"2A" (train B AC power) fails quickly (e.g., the DG cooling water has failed so the DG runs for several minutes then fails), then the inboard isolation valve, F063, which closed on power restoration can not be reopened until power is restored. If train A or B of onsite power is operating, then RCIC high room temperature trip will trip RCIC in about 30 minutes if the RCIC room HVAC is not working (which is on train A only, so train B AC power only will result in isolation; if AC power then fails, the inboard isolation valve could not be reopened). This isolation can be restored by the operator, if AC power does not fail in the mean time or is recovered in time to prevent core damage.

The RHR system cannot function without AC power. Therefore, the containment pressure and temperature would increase during a station blackout. However, containment integrity would not be challenged for approximately 20 hours (the containment was initially assessed to fail at 100 pisg, this was later revised to 190 psig by the expert panel and the time to failure would be about 50 hrs.) given no containment heat removal. RCIC would fail to inject into the core long before this time resulting in core damage.

Recovery of offsite power is not included in the event tree as a separate event but will be treated as a recovery action and applied to the

appropriate sequence cut sets. Without offsite power, the PCS, the condensate system, and the feedwater system would be unavailable.

The event tree for a transient initiator is shown in Figure 2.5. This event tree was developed by referring to the accident analyses reported in Chapter 15 of the FSAR,[3] LaSalle operating procedures, generic BWR operating procedures,[11] and generic transient thermal-hydraulic calculations.[4] In addition LaSalle-specific calculations were performed with RELAP5,[7] LTAS,[2] and MELCOR[12] for various accident sequences.

The LaSalle RETRAN-3 deck was obtained from the utility and converted to a RELAP5 model. The LaSalle FSAR was reviewed and modifications were made to the model to make it as accurate as possible. Plant specific data was obtained from Commonwealth Edison for steady state operation and two transients (a turbine trip and an MSIV closure). The model was modified to accurately represent this information. Three calculations were performed to examine transient response:

1. MSIV closure with no injection,

2. MSIV closure with RCIC only, and

3. MSIV closure with normal ECCS response.

As mentioned in Section 2.6.1, the LTAS code was modified to represent the LaSalle plant and five transient calculations were performed:

1. Modified Station Blackout with a stuck open SRV, RCIC operation, and one CRD pump,

2. Long-term containment heat removal failure, MSIV failure, HPCS, RCIC, 1 CRD pump, and LPCS available, containment venting at 147 psia and 60 psia,

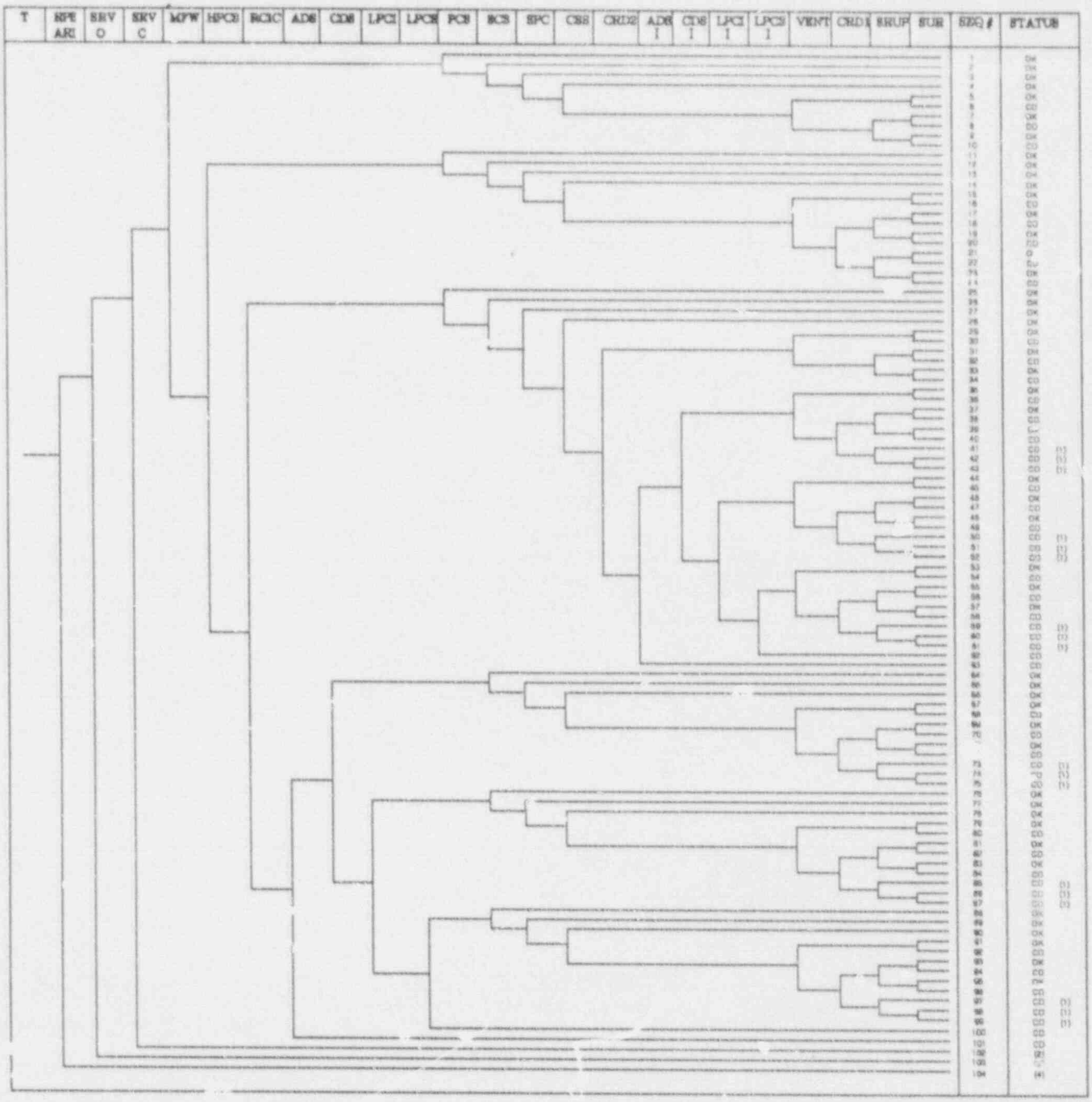
3. Station Blackout with ADS, no injection,

4. Station Blackout without ADS, no injection, and

5. Station Blackout without ADS, no injection, and a stuck open SRV.

In addition, an integrated model was constructed for use with the MELCOR code. MELCOR calculations beginning at reactor trip and progressing through core damage, melt, vessel breach, containment heatup and failure (may be before core damage for long-term loss of containment heat removal failure), and release of radionuclides to the environment were performed. The following calculations were performed mainly for the level II accident progression and source term analysis:

1. Short-term high pressure station blackout, sensitivities for containment failure in the drywell head, drywell, and wetwell, and for leaks, intermediate ruptures, and ruptures, and various hydrogen ignition limits,



(1) USED TO RESOLVE CORE DAMAGE RECOVERY, LOW PRESSURE SYSTEMS FAIL ON ADS CLOSURE AT ABOUT 85 PSIG, BOILOFF AND CORE DAMAGE OCCUR BEFORE CONTAINMENT FAILURE (MEAN VALUE, 195 PSIG).

(2) TRANSFER TO LOCA TREE (1 SRV FTC = SMALL LOCA, 2 SRV FTC = MEDIUM LOCA, AND >= 3 SRV FTC = LARGE LOCA).

(3) TRANSFER TO LOCA TREE (OVERPRESSURE CREATES LOCA. PROB. OF 18 SRV FTO NEGLIABLE).

(4) TRANSFER TO ATWS TREE.

Figure 2.5 LaSalle Transient Systemic Event Tree

2. Short-term low pressure station blackout, sensitivities for pedestal reflooding on core-concrete interactions failing pedestal wall, wetwell leak only,

3. Intermediate-term station blackout, wetwell leak only, and

4. Long-term loss of containment heat removal sequence with containment failure in wetwell, injection failure from severe environments in reactor building after containment failure.

All of these calculations are described in detail in the Level III report.*

Each of the events depicted in the event tree and the interactions with other events is described below.

T - Transient Initiators - A variety of plant malfunctions will initiate a transient. The transients included here are: turbine trip with and without bypass available, total MSIV closure, loss of condenser vacuum, total and partial loss of feedwater, inadvertently opened relief valve, and loss of offsite power. In addition, these special initiators are included: loss of DC bus A or B, loss of ECC 4160 V AC buses 241Y and 242Y, loss of instrument air, loss of drywell pneumatic, loss of 100# drywell pneumatic, complete loss of reactor vessel narrow range level indication, and loss of reactor vessel narrow range indication channels A and C or B and D.

RPS/ARI - Reactor Subcritical - After a transient initiator, an automatic reactor scram should occur for a variety of reasons depending on the specific initiator. For turbine trips (with and without bypass), loss of condenser vacuum, and loss of offsite power transients; main steam stop valve closure should initiate an automatic reactor scram. For MSIV transients, closure of three MSIVs will result in an automatic reactor scram signal. A low reactor water level (Level 3) signal will generate a reactor scram signal during a loss of feedwater transient.

For an IORV initiator, it may be detected early by one of many indication lights or alarms. Included are SRV open lights and alarms, downcomer piping temperature measurements, and suppression pool level and temperature alarms. Following operator awareness of an IORV he is directed by procedure[14] to try to close the IORV. If he fails to close the IORV, he is directed to manually scram the reactor. If the operator

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* C. J. Shaffer, L. A. Miller, and A. C. Payne Jr., "Integrated Risk Assessment For the LaSalle Unit 2 Nuclear Power Plant: Phenomenology and Risk Uncertainty Evaluation Program (PRUEP), Volume 3: MELCOR Code Calculations," NUREG/CR-5305/3 of 3, SAND90-2765, Sandia National Laboratories, Albuquerque, NM, to be published.

is unaware of the IORV or fails to manually scram the reactor, the steam transferred to the suppression pool will increase the suppression pool temperature at a rate of approximately 2 °F/minute. Within approximately 20 minutes, the suppression pool temperature would have increased by 40 °F and local condensation instabilities would be expected to have occurred. Steam would thus be released to the drywell resulting in an increase in pressure. A high drywell pressure reactor scram signal (1.69 psig) would be expected to occur.

Failure to scram the reactor results in an ATWS which is modeled in a separate event tree (see Section 2.6.3). The sequences shown in Figure 2.5 include successful reactor subcriticality.

<u>SRV O - Safety Relief Valves Open</u> - Closure of the main steam stop valves or MSIVs while at power will result in a pressure increase which must be relieved in order to prevent reactor coolant pressure boundary failure. Cycling of the safety relief valves may occur. Failure of a sufficient number of valves to open is assumed to result in a large rupture of the pressure boundary. The sequence involving failure of this event is transferred to the LCCA event tree for further evaluation. These sequences are probabilistically negligible and are not developed any further.

Relief valve openings are expected for all transients except for loss of feedwater and an IORV. The IORV will result in sufficient pressure reduction to preclude subsequent relief valve openings. Following loss of feedwater, the reactor pressure will decrease and the safety relief valves should not be demanded. If however, MSIV closure occurs following loss of RCIC and HPCS, the pressure in the vessel would rise requiring an SRV opening. Failure of all the SRVs to open is assumed to result in a large rupture of the reactor pressure boundary. These sequences are probabilistically negligible and are not developed any further.

<u>SRV C - Safety Relief Valves Reclose</u> - The safety relief valves that open during the transient must reclose in order to prevent an excessive discharge of coolant from the reactor vessel to the suppression pool. The sequence of events following a stuck open relief valve (SORV) is identical to the LOCA events with respect to the coolant makeup systems and containment heat removal systems. (The MSIVs are assumed to close on low pressure rendering the PCS unavailable.) Failure of one SRV to reclose will behave like a small LOCA, two SORVs will behave like a medium LOCA, and three or more SORVs will result in a sequence of events similar to a large LOCA. Transfers to the LOCA event tree are made following failure of SRVs to reclose.

<u>MFW - Feedwater Available</u> - Depending on the transient, the feedwater system can be in one of five states. For turbine trips with or without bypass, a high reactor water level signal (Level 8) would be generated tripping the turbine driven feedwater pumps and preventing startup of the motor-driven pump. Use of the feedwater system for these transients requires that the operator start any one of the three pumps once the Level 8 signal clears and is reset.

An MSIV closure transient results in coastdown of the turbine-driven pumps. The motor-driven pump does not start automatically since actual tripping of both turbine-driven pumps is required for this action. The motor-driven pump must be manually started by either directly initiating the pump or by manually tripping the turbine-driven pumps. No credit is taken for re-establishing steam flow to the turbine-driven pumps. Note that closure of the MSIVs may result in loss of condenser vacuum and a subsequent turbine-driven pump trip which would result in automatic initiation of the motor-driven pump. Also, the operator may eventually close the TDRFP discharge valves in order to prevent an uncontrolled level increase resulting from condensate booster flow through the TDRFPs. Closure of these valves also would result in automatic initiation of the MDRFP. However, both of these are not considered due to the potential delays in their occurrence.

For a loss of condenser vacuum initiator, the turbine-driven pumps will automatically trip and the motor-driven pump will automatically start.

For loss of feedwater and offsite power transients, all feedwater is not available. The loss of feedwater initiator is defined as a loss of all feedwater. A loss of offsite power initiator would result in a loss of condenser vacuum and a subsequent turbine-driven pump trip. Loss of offsite power would also render the motor-driven pump unavailable.

Following an IORV initiator, the reactor vessel pressure will decrease and an MSIV closure will occur if the operator fails to turn the reactor mode switch to SHUTDOWN as directed by procedures. If he turns the mode switch, both turbine-driven feedwater pumps would continue to operate. If he fails to turn the mode switch, the result would be the same as an MSIV closure event (i.e., the motor-driven pump would have to be manually initiated).

HPCS - HPCS Available - Failure of the feedwater pumps to operate will result in a reactor vessel level decrease. A low reactor water level signal (Level 2) will automatically initiate the HPCS system within approximately 30 seconds. If the system fails to initiate automatically, the operator can attempt manual initiation. The system is available for all transients.

RCIC - RCIC Available - The RCIC system is also designed to start automatically upon receipt of a Level 2 reactor water level signal. The operator can also manually initiate the system. The system is available for all transients.

ADS - Reactor Vessel Depressurization - In the event that the high pressure injection systems are all unable to maintain the reactor vessel coolant level, the reactor vessel can be depressurized so that low pressure injection systems can be utilized. The preferred method of depressurization is through the turbine bypass valves to the condenser. However, the reactor vessel is isolated for most transients. No credit for this method of depressurization is thus taken. The next method of

depressurization would be manual opening of three or more SRVs including the ADS valves. Automatic initiation of ADS will occur if the reactor vessel level decreases to Level 1 and either a high drywell pressure signal occurs or if ten minutes elapse.

CDS - Condensate Available - The condensate system is designed to remain running following a transient with feedwater trip. Flow is recirculated back to the condenser while the reactor vessel is at high pressure. When the reactor vessel pressure decreases, water from the condensate system can flow through the feedwater pumps to the vessel or can be bypassed around the MDRFP to the vessel. In order to control the condensate flow, the condensate must pass through the feedwater control valve located at the outlet of the MDRFP. The operator must close the TDRFP outlet valves and initiate the feedwater control valve or single-element flow control. The condensate system is assumed unavailable for condensate system initiators resulting in loss of feedwater and also for a loss of offsite power transient.

LPCI - LPCI Available - The Low Pressure Coolant Injection mode of the RHR system is also designed to automatically initiate if the reactor vessel level reaches the Level 1 setpoint. The operator can also manually initiate LPCI. Water injection will only occur if the reactor vessel pressure is in the LPCI operating range. The system is available for all transients.

LPCS - LPCS Available - The Low Pressure Core Spray system is designed to automatically initiate upon a low reactor water level signal (Level 1). The operator may also manually initiate LPCS. Water injection, however, will not occur unless the vessel pressure is in its operating range. The system is available for all transients.

PCS - PCS Available - Following a turbine trip with bypass, loss of feedwater, or IORV transient; steam should be directed to the main condenser via the turbine bypass valves. This flow path should prevent a large release of steam to the containment through successive openings of the SRVs. The PCS may become unavailable due to subsequent failure of the condenser or due to MSIV closure. MSIV closure will occur if the steamline pressure decreases below 850 psig or the reactor vessel level decreases below the Level 1 setpoint. The operator is instructed by the scram procedure to prevent the former signal from closing the MSIVs during cooldown by placing the reactor mode switch out of the RUN position. The Level 1 signal is assumed to occur following failure of all high pressure injections systems and subsequent vessel depressurization. The PCS is unavailable for turbine trips without bypass, loss of condenser vacuum, MSIV closure, and loss of offsite power transients.

SCS - Shutdown Cooling - Once the RPV pressure has been decreased below the shutdown cooling interlocks, procedures direct the operator to initiate the shutdown cooling mode of the RHR. With the shutdown cooling mode initiated, no further heat would be released to the containment.

With failure of the suppression pool cooling mode, the operator would be directed to depressurize the reactor vessel and initiate shutdown cooling. The mode of RHR operation is applicable for all transients.

SPC - Suppression Pool Cooling - If the safety/relief valves are repeatedly opened to relieve reactor vessel pressure (as in the case where PCS failure occurs) or are opened manually by the operator to depressurize the vessel, steam will be released to the suppression pool. When a sufficient amount of steam is released, the suppression pool water temperature will reach 100 °F. At this temperature, the operator is required to initiate suppression pool cooling. This mode of RHR operation can be used to mitigate all potential transients.

CSS - Containment Spray - In the unlikely event that suppression pool cooling fails, the primary containment pressure and temperature will continue to increase. The operator is directed by procedures to initiate suppression pool sprays when the suppression chamber pressure exceeds 19.5 psig. The operator is also directed to initiate drywell sprays if containment pressure can not be maintained below 60 psig. Suppression pool sprays were not modeled as a separate system in this analysis. Drywell sprays were used to represent both spray modes. This mode of operation can mitigate all potential transients.

CRD2 - Intermediate Control Rod Drive - In those cases where the main feedwater and HPCS systems have failed and the RCIC system is the only high volume high pressure system working; if containment heat removal fails, then RCIC will isolate when containment pressure reaches 40 psia or may fail due to loss of DC power (for station blackout type sequences) or high suppression pool temperature (i.e., greater than 250 °F). By this time the decay heat load is low enough that the CRD system can supply sufficient makeup to maintain the core level high enough to prevent core damage if AC power is available and both pumps are working.

ADS I - Intermediate Reactor Vessel Depressurization - For sequences where RCIC is the only high pressure injection system available, this event consists of either automatic or manual depressurization of the reactor vessel to allow injection from the low pressure ECCS when delayed RCIC failure occurs after about 6 hours. All other high pressure injection systems have failed and so the operator must depressurize and use low pressure injection systems to prevent core damage. Automatic depressurization would occur by the ADS if the reactor vessel lev decreased to Level 1 due to failure of the high pressure systems. Sufficient time would also be available for the operator to manually depressurize the vessel using the SRVs if the ADS failed. In both cases, successful depressurization requires opening of 3 valves.

CDS I - Intermediate Condensate Available - If all the high pressure injection systems have failed except RCIC and then RCIC fails after about 6 hours, the operator will depressurize the RPV and attempt to use low pressure injection systems. If AC power is available and the system has not been accounted for before, then the condensate system may be used to

maintain core cooling. Water from the condensate pumps will flow through the feedwater control valves to the vessel. One condensate/condensate booster train can provide sufficient coolant to the vessel. The coolant is provided from the condenser hotwell. Makeup to the hotwell must be provided by the condensate makeup system which is limited to 1800 gpm and supplies water from the condensate storage tank.

LPCI I - Intermediate LPCI Available - If all the high pressure injection systems have failed except RCIC and then RCIC fails after about 6 hours, the operator will depressurize the RPV and attempt to use low pressure injection systems. If AC power is available and the system has not been accounted for before, then any one of the three LPCI trains can be successfully used to mitigate the transient. The pumps take suction from the suppression pool and initially bypass the RHR heat exchangers.

LPCS I - Intermediate LPCS Available - If all the high pressure injection systems have failed except RCIC and then RCIC fails after about 6 hours, the operator will depressurize the RPV and attempt to use low pressure injection systems. If AC power is available and the system has not been used before, then the single train LPCS system has sufficient capacity to mitigate the transient. Failure of the LPCS system is defined as failure of the single motor-driven pump to provide flow to the reactor vessel from the suppression pool.

VENT - Containment Venting - After containment heat removal failure, the containment pressure will continue to increase due to the decay heat being dumped to the suppression pool. If the containment heat removal systems can not be recovered, then when the containment pressure reaches 60 psig, the operator is directed to vent the containment through the Standby Gas Treatment System. Containment venting can occur during any transient if all containment heat removal systems fail as long as AC power is available.

CRD1 - Late Control Rod Drive - If containment heat removal has failed and the operator does not vent the containment when containment pressure reaches 60 psig, then the CRD system may be used to maintain core cooling with only one pump operating. For cases with only low pressure injection working, the ADS valves will close when containment pressure reaches about 85 psig due to inadequate differential pressure resulting in the inability of all low pressure injection system to inject coolant into the core. The low pressure injection systems (LPCI and LPCS) may fail due to high suppression pool temperatures that result in pump failure. For the case of HPCS supplying high pressure injection, since the HPCS suction will be from the suppression pool by this time, a similar result may occur.

SRUP - Containment Failure Mode - If the containment heat removal function has failed and the containment is not vented, then the containment pressure will continue to rise until structural failure of the containment occurs. This event asks, roughly, how large the break in containment is, either a leak (success branch) or a rupture (failure

branch). For the purposes of this analysis, we need to know how fast the containment depressurizes, whether or not we can use any low pressure injection system to maintain core cooling after the containment failure, and how long the containment blowdown will last in order to evaluate the severity of the reactor building environments produced by the blowdown. A leak implies that the containment pressure remains too high to use the ADS system to depressurize the RPV and the low pressure injection systems to prevent core damage; while a rupture implies that the containment depressurizes fast enough that successful operation of the ADS and a low pressure injection system will prevent core damage. An expert judgement elicitation was performed to generate probability distributions for the containment failure pressure, location, and size.

<u>SUR - Injection System Survival</u> - After containment failure, severe environments will be produced in the reactor building. Many of the injection systems have components at various positions in the reactor building and these components will be exposed to these severe environments. Simple Boolean equations were generated to represent each system that had components that might be subject to these environments and an expert judgement elicitation was performed for various component types and environments in order to quantify the failure probabilities.

2.6.3 <u>ATWS Event Tree</u>

Because of the unique characteristics of the ATWS events, the differences among the various initiating events in their effect on the accident progression are judged to be small. One general systemic ATWS event tree has been constructed and the effects of the various initiators will be inserted into the system fault trees for those systems that are affected. Individual ATWS trees for each initiator were constructed to determine if any differences were significant enough to warrant separate trees. There were none.

The LaSalle 2 ATWS procedure was revised to correspond to the BWR Emergency Procedure Guidelines (EPGs) Revision 3.[11] The EPGs address an ATWS situation in Contingency #7 "Level/Power Control". The EPGs were used in guiding the construction of the ATWS event tree.

The EPG strategy for dealing with an ATWS can be summarized as follows: (1) attempt manual scram, (2) begin manual insertion of control rods and initiate SBLC if manual scram fails, (3) reduce core power by taking manual control of the reactor vessel injection systems and lowering the reactor vessel water level to the top of the core (which increases the core void fraction but also prevents boron mixing), (4) once sufficient sodium pentaborate has been injected, increase the rate of reactor vessel injection so that normal reactor vessel water level is restored to promote natural circulation flow and boron mixing, and (5) bring the reactor to cold shutdown.

A study performed at Oak Ridge as part of the SASA program of ATWS sequences for Browns Ferry Unit One[15] indicates that the "instructions

provided by the EPGs, if properly interpreted and implemented by the operators, would provide a satisfactory reactor shutdown and accident termination" of the MSIV-closure ATWS analyzed in the study. However, the Oak Ridge study also indicated some potential problem areas. The most important of these is that the operator can be directed to manually reduce reactor pressure during an ATWS. (This is to ensure that the thermal energy released during a LOCA can be condensed in a suppression pool. As the suppression pool temperature increases above 165 °F, the operator is to depressurize the vessel according to a supplied graph.) The calculations performed indicate that manual depressurization during an ATWS is very tricky and, depending on the situation, can result in reactor power and vessel pressure fluctuations. The recommendations from this study were to eliminate such a manual depressurization during an ATWS.

According to the EPGs, if the reactor cannot be shut down during a transient, if the suppression pool temperature reaches 110 °F, and if the drywell pressure is above 1.69 psig, then the operator is to lower the RPV water level by terminating and preventing all injection into the RPV except from the SBLC. The operator is to maintain the water level at the top of the active fuel (TAF) with a high pressure injection system until the boron has been injected <u>and</u> the control rods have been manually inserted.

Feedwater would be the first choice of injection systems for some transient initiators since it should be automatically available. The high temperature of the feedwater is also desirable since it results in less reactivity than the relatively cold water contained in the condensate storage tank. RCIC and CRD are assumed insufficient for maintaining the water level at the TAF. The Browns Ferry Study indicated that the two systems could maintain 2/3 of the core covered with the remaining 1/3 cooled by steam flow. This reduced level has the benefit of further reducing core power. However, the LaSalle RCIC system is different than the system at Browns Ferry. The LaSalle system sprays at the top of the vessel while the Browns Ferry system injects into the downcomer. The spray system is assumed not to be as effective as the injection system and thus no credit was taken for its operation.

For this study, the RELAP5 model used for the transient analysis was modified to perform two ATWS calculations. The first was an ATWS with HPCS injection only and no ADS (i.e., the vessel remained at high pressure). The second was an ATWS with LPCS injection and successful depressurization of the vessel using the ADS system. In the first case, the level oscillates between -140 and -160 in. (TAF) and reactor power is approximately 15%. In the second case, level dropped temporarily to about -315 in. then recovered. The calculation was terminated when level began oscillating wildly near the level 1 setpoint.

In order to perform more efficient calculations and to evaluate more sequences, the LTAS code developed and used by ORNL for the Brown's Ferry study was modified, as described in Section 2.6.1, to represent the

LaSalle plant and base-lined to the RELAP5 model. A REMONA-3B[16] calculation was used for the power vs level correlation. Nineteen different ATWS calculations were performed using the LTAS code to investigate different possible system success criteria and to evaluate the accident sequence timing. These calculations evaluated the effects of : (1) main feedwater being initially available or not, (2) HPCS and LPCS success or failure and failure on different high suppression pool temperatures (300 or 350 °F, the pumps use suppression pool water to cool their bearings, LPCI uses external cooling water and would not be affected), (3) LPCI operability and number of pumps, (4) containment heat removal (RHR) operability, (5) ADS operation, and (6) venting success or failure.

Consider the following scenario: HPCS works, the vessel is at high pressure, RHR does not work, and venting does not occur. Then the calculations show that the vessel water level stabilizes at about 2/3 TAF with reactor power at about 9%. The containment pressurizes until HPCS fails on high suppression pool temperature (after this analysis was done, expert elicitations were obtained for the Grand Gulf and LaSalle HPCS pump sensitivity to high suppression pool temperatures as part of the NUREG-1150 analysis,[17] sult was that the experts did not expect the HPCS pumps to fail as a lt of the effects of seal leakage into the pump rooms for the type o pumps at LaSalle and Grand Gulf) or, if HPCS does not fail, until the containment fails at a mean pressure of 191 psig (obtained from expert judgement elicitations also performed as part of the NUREG-1150 expert judgement process). After containment failure, severe environments may be created in the reactor building depending upon the location of the failure and these environments may result in injection system failure and core damage with a failed containment. If HPCS continues to work, the containment pressure will stabilize at some intermediate level depending on the size of the break.

In the actual runs, the HPCS and LPCS systems were assumed to fail at either 300 or 350 °F for the different calculations. If the systems failed on temperature (which they did not do in many of the calculations), these sequences then became sequences with ADS and LPCI working. The following variations on the scenario presented above can be evaluated from these calculations:

1. HPCS works, RHR works, no venting - No significant impact on accident progression due to the fact that RHR can only remove the energy equivalent to 3% reactor power and the power level is 9% Containment pressure increases until failure;

2. HPCS works, No RHR, venting success - containment venting definitely will create severe environments in the reactor building and the sequence timing has containment failure at 60 psig, not 195 psig, so is somewhat faster. Containment pressure equilibrates at 95-100 psia, with reactor power at 9%;

3. HPCS works, venting and RHR success - Similar to (2) but pressure stabilizes between 93-100 psia;

4. HPCS works, ADS success, no venting or RHR - reactor power at about 17% until ADS recloses on high drywell pressure, then decreases back to 9%. Containment pressurizes until failure;

5. HPCS works, ADS and venting success, no RHR - reactor power initially at 17%, decreases to 9% when ADS recloses due to the lower injection rate of coolant. Pressure stabilizes between 95-100 psia, ADS recloses at 100 psia;

6. HPCS works, ADS and RHR success, no venting - Similar to (1), no significant impact on accident progression due to the fact that RHR can only remove the energy equivalent to 3% reactor power and the power level is initially at 17% decreasing to 9% when ADS recloses. Containment pressurizes until failure;

7. HPCS works, ADS and RHR and venting success - reactor power initially at 17%, decreases to 9% after ADS reclosure. Containment pressure oscillates between 95-100 psia;

8. HPCS fails, ADS and LPCI work, no RHR or venting - reactor power at 17% until containment pressure results in ADS closure. Enter oscillatory state with low pressure injection working on and off as containment pressure oscillates about ADS reclosure pressure;

9. HPCS fails, ADS and LPCI work, no RHR, venting success - reactor power at 17% until HPCS fails, decreases to 9% on LPCI startup. Enter oscillatory state with low pressure injection working on and off as pressure oscillates about ADS reclosure pressure. After 20 minute delay from 60 psig, venting occurs, pressure drops to about 80 psia, and power level drops a percent; Some instability in the power level;

10. HPCS fails, ADS and LPCI work, no venting, RHR success - reactor power, level, and pressure oscillate wildly as LPCI injects and then stops on high RPV pressure. The average power level is low, about 2.5% but spikes up to 50% for short periods of time, RPV pressure varies from low (200 psia) to high (SRV setpoint, 1150 psia), containment pressure at about 100 psia;

11 HPCS fails, ADS and LPCI work, venting and RHR success - reactor pressure at about 250 psia, reactor power at 9%, and containment pressure equilibrates at about 90-100 psia.

In all of these calculations, the water level stabilized near 2/3 TAF. From these calculations, it appears that the operator will not have to terminate injection in order to reduce the level to TAF and the equilibrium power level will be of the order of 9-17% depending on the systems operating and the ADS status. Only if main feedwater or condensate is working will the amount of coolant injected be so large that deliberate reduction of level would be necessary.

The Browns Ferry ATWS study also indicated that the effect of one or two SORVs on an ATWS sequence is negligible. This is because several SRVs are open during the early part of an ATWS sequence so that the occurrence of an SORV would not be recognized until the reactor power had decreased to within the capacity of the SORV. This is also expected to be true for the LaSalle plant. For LOCA initiators, these ATWS sequences act like sequences with ADS operation and can be evaluated the same way.

The general ATWS event tree is shown in Figure 2.6. Each event is discussed below.

T - Transient Initiators - The transient may result in MSIV closure or the condenser being available. For cases where the condenser is available, the reactor power can be reduced by lowering the reactor vessel level to the TAF and the heat can be dumped to the condenser. Initiation of SBLC may not be necessary but manual rod insertion would be eventually required. If MSIV closure occurs, a large heat load would be dumped to the containment through the safety/relief valves. Actuation of SBLC or manual rod insertion and RPV level control would both be required to reduce the reactor power level. However, to reduce the power level to within the RHR capacity would require the level be reduced below 2/3 TAF according to our RELAP5 and LTAS calculations. Current procedures do not account for this possibility. Containment pressure will increase until containment venting or structural failure occurs. LOCA or transient-induced LOCA initiators all result in MSIV closure. They evolve similar to the transient initiators because, for cases where the reactor is not rendered subcritical, the SRVs will be open anyway simulating a LOCA. The available systems that are modeled in the ATWS event tree can all mitigate a LOCA of any size and their response would be basically the same.

RPS/ARI - Reactor Subcriticality - An automatic reactor scram signal should be generated for most transients or LOCAs. The RPS sensors, logic, scram solenoids, and control rod drive mechanism function as required. The electrical portion of the scram function is backed up by ARI. ARI consists of an alternate set of sensors, logic and solenoid valves which provide a backup to the electrical portion of the RPS. RPS de-energizes to actuate while ARI energizes to actuate. The addition of an ARI system is assessed to reduce the probability of electrical failures to negligibly low levels. Therefore, this event represents mechanical failure of the rods to insert and is assumed not to be directly recoverable. For this event tree, all sequences have RPS/ARI failure.

MFW - Feedwater Available - The feedwater/condensate system would be the first choice for maintaining the water level at TAF and later between 12.5" and 58.5". Depending on the transient (and whether MSIV closure occurs), all three feedwater pumps may be available. The operator may eventually have to switch to a recirculation system if the suppression pool level increases above the load limit curve (a function of RPV pressure) or the maximum containment water level limit.

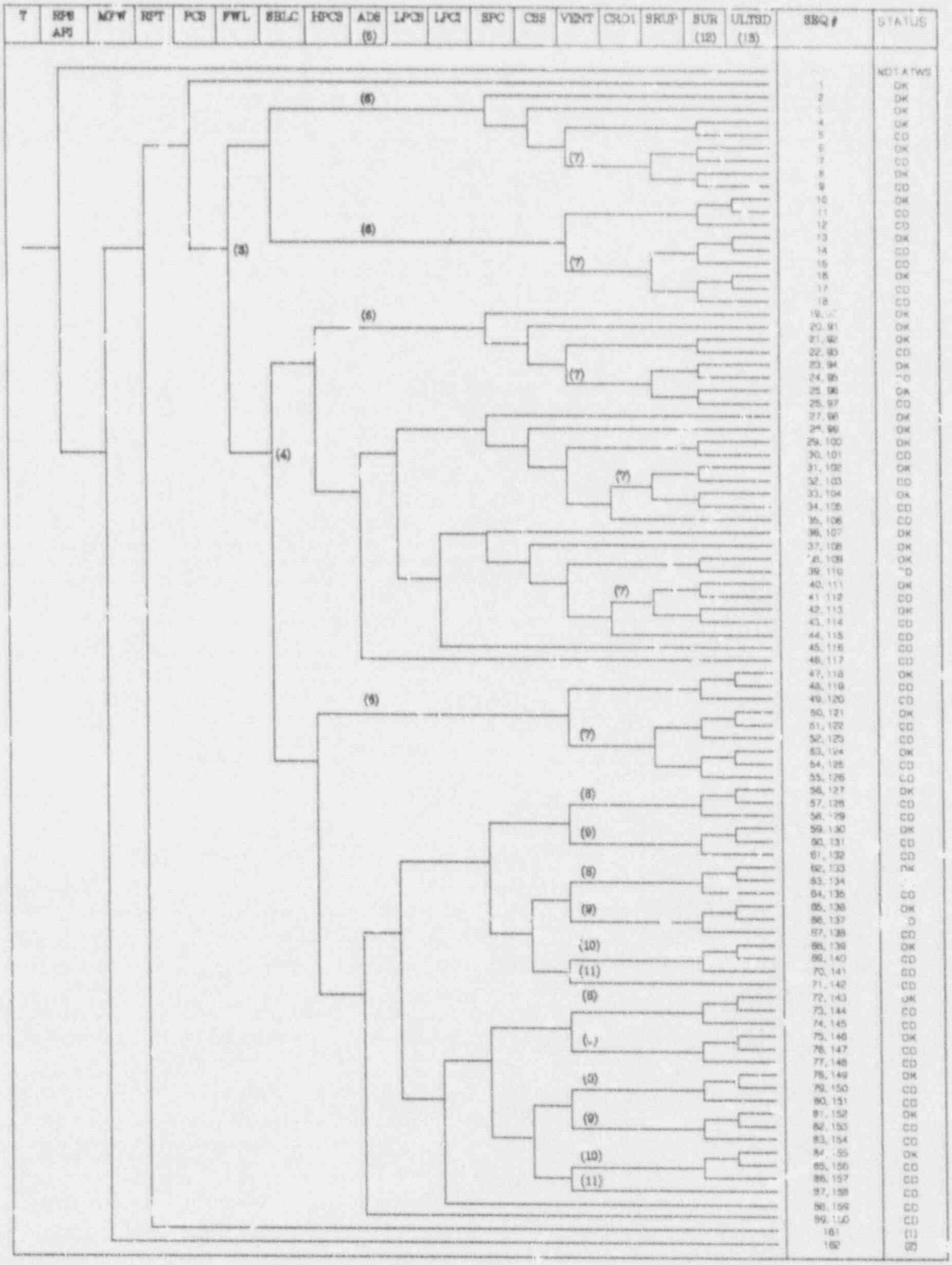

Figure 2.6 LaSalle ATWS Systemic Event Tree

Figure 2.6 LaSalle ATWS Systemic Event Tree (continued)

1) If MFW succeeds, RPT failure will be negligible since it depends upon the same power sources as MFW. If power fails MFW, then it will also fail the RCPs. If RPT does fail, either PCS will have succeeded in which case we have an ok sequence or, if PCS fails, MFW will behave as in note (3) and the RCPs will fail on low suction pressure (the peak pressures will be below level D stress limits).

2) If MFW fails, RPT is not relevant since RPV level cannot be maintained and the resulting low level will result in RCP failure on low suction pressure. Sequences transfer to (4).

3) MFW can not continue to run for more than about 8 minutes without depleting the main condenser unless the operator controls level. The injection rate must be controlled to ≤ 1800 gpm, the makeup rate from the CST. This means that RPV level will be below TAF.

4) Transfer sequences from (2).

5) Operators are instructed by EOPs not to use inhibit switch for ADS but to reset timer.

6) For cases where no choice is given, ADS success or failure will not affect the sequence timing or end result significantly. If the operator opens the SRVs to bring pressure down or auto ADS occurs due to low level, power will increase from about 12% to about 18%. LTAS code calculations show that ADS and subsequent HPCS, LPCS, or LPCI injection will not produce excessive power spikes. Level will remain at about 2/3 TAF, the low pressure injection systems will inject enough to raise pressure above their shutoff heads, and, if HPCS is working, they will remain shutoff since the pressure will not decrease back below their shutoff heads. If HPCS is not working then oscillatory behavior results (mild pressure variations).

7) Containment pressure increases until containment failure occurs.

8) RHR and Venting success - Containment pressure remains below ADS reclosure pressure (90 psia, 321 °F). Oscillatory behavior results from RPV pressure exceeding low pressure system shutoff heads, injection valves cycle (16 times/hr.).

9) RHR OK and Venting failure - Containment pressure increases to ADS reclosure pressure and then oscillatory behavior results (100 psia, 321 °F) from RPV pressure exceeding low pressure system shutoff heads; injection valves cycle (11 times/hr.)

Figure 2.6 LaSalle ATWS Systemic Event Tree (concluded)

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10) RHR fails and Venting OK - Containment pressure remains below ADS reclosure pressure (90 psia, 321 °F). Oscillatory behavior results from RPV pressure exceeding low pressure system shutoff heads; injection valves cycle (16 times/hr.).

11) RHR and Venting fail - ADS valves reclose at about 85 psig, RPV repressurizes above LPCS and LPCI shutoff heads, boiloff and core damage occurs long before containment failure.

12) Upon containment leak or rupture to the reactor building, severe environments may result in equipment failure.

13) Ultimate Shutdown - Requires alternate rod insertion or boron injection by some alternate means.

RPT - Recirculation Pump Trip - Both recirculation pumps are required to trip during an ATWS. RPT is initiated by a turbine stop valve closure, a turbine control valve fast closure, high reactor vessel pressure, or a reactor vessel Level 2 signal. The tripping of the pumps will reduce reactor power to approximately 30 to 50%. Failure to trip the recirculation pumps will result in high reactor power and pressure beyond the safety/relief valve capacity, which is assumed to result in reactor vessel rupture and a large LOCA. If main feedwater succeeds, then RPT failure is unlikely since the power sources (DC) that RPT depends on are the same as those for feedwater. If RPT failure occurs for reasons other than power, either: (1) PCS and MFW will be working with no scram having occurred, in which case the plant can continue operating normally or (2) PCS will be failed and MFW will work. In the second case, MFW can work at most for 8 minutes before the condenser is drained, feedwater fails, and the recirculation pumps trip independently on low suction pressure. If MFW fails then RPT will be irrelevant since the low RPV level will result in recirculation pump trip on low suction pressure.

PCS - Power Conversion System - If the PCS is available following the initiating transient and the plant has not tripped, then it is as if an event has not occurred. The plant continues to run normally. If feedwater is degraded (i.e., only the motor-driven pump is available), then with successful RPT the PCS can remove the energy being generated.

FWL - Feedwater Level Control - If PCS fails, then the RPV water level will need to be reduced to lower reactor power and to stay within the makeup capability of the condensate makeup system which is only about 1800 gpm. At this rate, water level can not be maintained above TAF. This event was assigned a 0.5 probability of success.

SBLC - Standby Liquid Control System - The operator must manually initiate the SBLC if the reactor power is not reduced and the PCS is unavailable. One of the two trains must inject the contents of the boron injection tank into the reactor vessel. The injection would take approximately 20 minutes. During this time, the operator must further control reactor power, pressure, level, and suppression pool cooling. If SBLC is successful, then the sequence proceeds like a normal transient.

HPCS - HPCS Available - The BWR Owners Group Emergency Procedures Guidelines[11] recommends use of core spray systems for reactor vessel level control under ATWS conditions only if the level cannot be maintained by high-pressure injection systems, the condensate and feedwater systems, or by LPCI. This is because of the unknown phenomenology associated with the spraying of large amounts of water onto the top of a partially uncovered core under ATWS conditions. Thus it is assumed HPCS will be used only after feedwater has failed. The HPCS flow cannot be throttled since the injection isolation valve can only go fully open or closed. Thus the operator must control the flow by fully opening or closing the isolation valve. RPV water level will equalize near 2/3 TAF with HPCS working.

ADS - Automatic Depressurization System - If SBLC succeeds but all high pressure injection has failed, then the water level will fall to level 1 and ADS will occur. The low pressure injection systems will then be able to inject. If SBLC does not succeed, then even with HPCS operation, the operator will not be able to maintain the RPV level above the TAF according to the LTAS calculations and ADS will occur.

LPCS - LPCS Available - The Low Pressure Core Spray system is designed to automatically initiate upon a low reactor water level signal (Level 1). The operator may also manually initiate LPCS. Water injection, however, will not occur unless the vessel pressure is in its operating range. The system is available for all ATWS events. The LPCS injection pressure is slightly higher than that for LPCI so it will come on first.

LPCI - LPCI Available - The Low Pressure Coolant Injection mode of the RHR system is also designed to automatically initiate if the reactor vessel level reaches the Level 1 setpoint. The operator can also manually initiate LPCI. Water injection will only occur if the reactor vessel pressure is in the LPCI operating range. The system is available for all ATWS events.

SPC - Suppression Pool Cooling - For events where the PCS is unavailable, the generated heat will be dumped to the suppression pool. Removal of this heat must be accomplished by the suppression pool cooling mode of RHR. For sequences with successful SBLC operation, the RHR system will be able to remove the heat being dumped to the suppression pool and maintain containment temperatures and pressures in the acceptable range. If SBLC does not succeed, then reactor power will be in the 9-17% range and RHR will not be able to remove all the energy. However, if venting occurs or if only low pressure injection systems are working, then successful RHR will result in stabilization of containment pressure and temperatures at levels near the ADS reclosure pressure or the vent pressure and prevent containment pressure from increasing into the range where structural failure of the containment might occur (>150 psig according to the expert elicitation).

CSS - Containment Spray - In the unlikely event that the suppression pool cooling mode of RHR fails, the containment pressure and temperature would increase. The containment spray mode of RHR can be initiated to remove some of the energy. The effects of CSS are similar to SPC.

VENT - Containment Venting - After containment heat removal failure in cases with successful SBLC or in all cases with SBLC failure, the containment pressure will continue to increase due to the decay heat or equilibrium power being dumped to the suppression pool. When the containment pressure reaches 60 psig, the operator is directed to vent the containment through the Standby Gas Treatment System. If the reactor is shutdown, then containment pressure will decrease back to near atmospheric. In all cases with reactor shutdown failure but some injection, successful venting will result in stabilization of containment pressure at some level above the vent pressure.

CRD1 - Late Control Rod Drive - If the reactor is shutdown, containment heat removal failed, and the operator does not vent the containment when containment pressure reaches 60 psig, then the CRD system may be used to maintain core cooling with only one pump operating in the long-term. CRD would not be usable in cases with liquid breaks below the level of the core.

SRUP - Containment Failure Mode - If some high pressure injection system is working and either a) the reactor is successfully shutdown and the containment heat removal function has failed or b) if the reactor is not shutdown, then if the containment is not vented, the containment pressure will continue to rise until structural failure of the containment occurs. This event asks, roughly, how large the break in containment is, either a leak (success branch) or a rupture (failure branch). For the purposes of this analysis, we need to know how fast the containment depressurizes, whether or not we can use any low pressure injection system to maintain core cooling after the containment failure, and how long the containment blowdown will last in order to evaluate the severity of the reactor building environments produced by the blowdown. A leak implies that the containment pressure remains too high to use the ADS system to depressurize the RPV and the low pressure injection systems to prevent core damage; while a rupture implies that the containment depressurizes fast enough that successful operation of the ADS and a low pressure injection system will prevent core damage. An expert judgement elicitation was performed to generate probability distributions for the containment failure pressure, location, and size.

SUR - Injection System Survival - After containment failure, severe environments will be produced in the reactor building. Many of the injection systems have components at various positions in the reactor building and these components will be exposed to these severe environments. Simple Boolean equations were generated to represent each system that had components that might be subject to these environments and an expert judgement elicitation was performed for various component types and environments in order to quantify the failure probabilities.

ULTSD - Ultimate Shutdown - For those cases where reactor shutdown has not occurred but the reactor has been brought to a temporarily stable state, one last chance of shutting down the reactor is allowed. This can be done by several different methods none usually considered in a PRA. A probability distribution for this event was determined by in-house expert elicitation.

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This volume presents the results of the initiating event and accident sequence delineation analyses of the LaSalle Unit II nuclear power plant performed as part of the Level III PRA being performed by Sandia National Laboratories for the Nuclear Regulatory Commission. The initiating event identification included a thorough review of extant data and a detailed plant specific search for special initiators. For the LaSalle analysis, the following initiating events were defined: eight general transients, ten special initiators, four LOCAs inside containment, one LOCA outside containment, and two interfacing LOCAs. Three accident sequence event trees were constructed: LOCA, transient, and ATWS. These trees were general in nature so that a tree represented all initiators of a particular type (i.e., the LOCA tree was constructed for evaluating small, medium, and large LOCAs simultaneously). The effects of the specific initiators on the systems and the different success criteria were handled by including the initiating events directly in the system fault trees. The accident sequence event trees were extended to include the evaluation of containment vulnerable sequences. These internal event accident sequence event trees were also used for the evaluation of the seismic, fire, and flood analyses.

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