NUREG/CR-4832
SAND92-0537
Vol. 3, Part 1

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

Internal Events Accident Sequence Quantification

Main Report

Prepared by
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Sandia National Laboratories
Operated by
Sandia Corporation

Prepared for
U.S. Nuclear Regulatory Commission

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Manuscript Completed: June 1992
Date Published: August 1992

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Office of Nuclear Regulatory Research
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Washington, DC 20555
NRC FIN A1386

ABSTRACT

This volume presents the methodology and results of the internal event accident sequence analysis of the LaSalle Unit II nuclear power plant performed as part of the Level III Probabilistic Risk Assessment being performed by Sandia National Laboratories for the Nuclear Regulatory Commission.

This report describes the new techniques developed to solve the very large and logically complicated fault trees developed in the modeling of the LaSalle systems, for evaluating the large number of cut sets in the accident sequences, for the application of recovery actions to these cut sets, and for the evaluation of the effects of containment failure on the systems and the resolution of core vulnerable accident sequences.

The LOCA, transient, transient-induced LOCAs, and anticipated accidents without scram accidents resulting from internal initiators are evaluated and the final dominant accident sequences are determined. Integrated results are obtained by merging all of the accident sequences' cut sets together and evaluating the resulting expression. Integrated risk reduction, risk increase, and uncertainty importance measures are obtained. Also, an overall ranking of the dominant cut sets is obtained.

The total internal core damage frequency has a mean valve of 4.41E-05/R-yr. with a 5th percentile of 2.05E-6/R-yr., a median value of 1.64E-05/R-yr., and a 95th percentile of 1.39E-04/R-yr. The dominant cut sets all involve loss of the emergency core cooling systems (ECCS) as a result of common mode failure of the diesel generator cooling water pumps which results in delayed failure of the ECCS injection systems and control rod drive and either a complete loss of offsite power resulting in a short or long-term station blackout accident (depending on the status of the reactor core isolation cooling system, RCIC) or a loss of train A AC or DC power resulting in a loss of feedwater control and closure of one set of the main steam isolation valves.

The events most important to risk reduction are: the frequency of loss of offsite power, the non recovery of offsite power within one hour, the diesel cooling water pump common mode failure, and the non-recoverable isolation of RCIC during station blackouts. The events most important to risk increase are: the failure of various AC power circuit breakers resulting in partial loss of onsite AC power, the failure to scram, and the diesel generator cooling water pump random failure rate (determines the magnitude of the common mode contribution). The dominant contributors to uncertainty are: the uncertainty in control circuit failure rates, the uncertainty in relay coil failure to energize, the uncertainty in energized relay coils failing deenergized, the uncertainty in the loss of offsite power frequency, and the uncertainty in diesel generator failure to start.

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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 Estimation Analysis and Human Reliability Screening Analysis.

NUREG/CR-4832 - Volume 6: System Descriptions and Fault Tree 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 organized 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 INTRODUCTION

1.1 Level Of Modeling Detail

As part of the analysis of the core damage frequency from internal initiators, the accident sequences defined for the PRA must be evaluated and their frequencies calculated. This process can range from very easy to extremely difficult depending on the level of detail of the analysis and the analysis tools available. For the LaSalle Probabilistic Risk Assessment (PRA), the inclusion of external initiators on an equal footing with internal initiators required the expansion of the model to include passive failures, diversion paths from spurious operation, additional components not usually modeled, and a greater level of detail in the fault tree modeling to accurately represent the effects of some of the external events.

This additional level of detail required the use of the most powerful tools available and their extension by the development of new techniques to (1) effectively include the additional level of detail in the system fault trees, (2) include some information in the fault trees via transformation equations, and (3) aid in the process of evaluating the accident sequences in an efficient and cost effective manner.

The description of the system modeling effort and of the development of the system fault trees is presented in Volume 6 of this report. The description of the techniques used to include location based information for the external event analyses in the fault tree model and the location data bases and transformation equations are presented in Volumes 8, 9, and 10 of this report on the seismic, fire, and flood analyses respectively. This volume presents the method used to evaluate the very large fault trees developed for the LaSalle PRA and the new solution techniques used in analyzing the accident sequences to obtain the core damage frequency from internal initiators.

2.0 OVERVIEW OF METHODOLOGY

2.1 Description of Steps Used to Determine Core Damage Frequency

The general process used to analyze the accident sequences and obtain the core damage frequency for the internal initiating events can be broken down into a series of steps:

1. Define the initiators to be analyzed. This analysis is described in Volume 4 of this report.

2. Determine the accident sequences that can result from these initiators and the systems necessary to mitigate the accidents. This analysis is described in Volume 4 of this report.

3. Develop fault tree models for the systems appearing in the event trees defining the accident sequences (front-line systems) and their support systems. This analysis is described in Volume 6 of this report.

4. Develop a data base consisting of point estimate values to use in the screening analysis and continue to refine to get values for the final analysis with uncertainty distributions. This analysis is described in Volume 5 of this report.

5. Solve the fault trees of the front-line systems in terms of their basic failures and include their support systems and the interactions between front-line systems, between support systems, and between front-line and support systems. This analysis is described in this volume.

6. Combine these system fault trees into accident sequences using point estimate data to calculate screening estimates of the accident sequences. This analysis is described in this volume.

7. Analyze the sequence cut sets (i.e., combinations of basic failures that can result in the accident sequence) to determine if they make physical sense and evaluate the potential for operator recovery actions mitigating the accident. Define and classify the recovery actions. Add the failures (i.e., non-recovery actions) to the cut sets, develop a method for quantifying the probability of operator failure, and quantify the actions and add to the data base. The definition, classification, adding to the cut sets, and quantifying the non-recovery actions are reported in this volume. The development of the method of evaluating human actions is presented in Reference 1.

8. Develop a method for resolving accident sequences which have uncertain end-states as a result of the inability to quantify the interaction between sequence phenomenology and system performance. Apply this methodology to resolve the core vulnerable accident sequences. This analysis is described in this volume.

9. Using the uncertainty distributions developed for the data, quantify each individual accident sequence and the combined accident sequences (i.e., the integrated results) to obtain the individual sequence and integrated core damage frequencies for internal initiators. The implementation of the data base to quantify the basic events appearing in the fault trees with all of the final uncertainty distributions is presented in Volume 2 of this report in the appendix describing the Latin Hypercube sample input files. The evaluation of the sequence and integrated uncertainty distributions and the importance calculations are reported in this volume.

2.2 References

1. D. W. Whitehead, "Recovery Actions in PRA for the Risk Methods Integration and Evaluation Program (RMIEP) Volume 2: Application of the Data Based Method," NUREG/CR-4834/2 of 2, SAND87-0179, Sandia National Laboratories, Albuquerque, NM, December 1987.

3.0 FAULT TREE SOLUTION METHODS

3.1 Development of Individual System Solutions

Modular fault tree methodology was used to construct the LaSalle system fault trees for LaSalle. The generic modules in Appendix I of NUREG/CR-3268[1] were revised and additional modules were developed for modeling control and actuation circuits based on relay as opposed to solid state logic. An IBM-PC program, MODEDIT, was developed to retrieve these files for modification into plant-specific modules. The program also checks each module for any errors that may have been generated during modification. Another IBM-PC program, INDEX, was developed to identify developed events which do not have a corresponding top gate in another fault tree module. A full description of these programs and the revised generic modules is discussed in "Microcomputer Application of and Modification to the Modular Fault Trees."*

As the fault tree modules for each individual system were completed, they were transferred to the mainframe computer. Using the SETS code[2] procedure Form New Fault Tree, FRMNEWFT; the modules were merged into a system fault tree. The Generate Fault Tree Equation, GENFTEQN, procedure was then used to compute the minimal cut sets of the system. These cut sets were examined by the analyst for validity and any indications of modeling errors. If modeling changes were needed, appropriate changes were made and new cut sets were generated. This process was repeated until the analyst was satisfied with the model of the system. Some systems, such as the electrical actuation system, were developed in parts to help clarify the function being modeled by the analyst. These parts were then merged and checked for errors. Twenty-seven individual fault tree segments were developed during this phase of the analysis. Table 3.1 lists these fault trees.

3.2 Merging Fault Trees

The fault tree segments for support and front-line systems were combined to form a completely merged fault tree for each system that appeared on the event trees (i.e., the front-line systems). The SETS code was used to perform the merging task. Two major problems were of concern in merging the RMIEP fault trees: (1) circular logic and (2) size.

Circular logic often occurs in fault tree models since interdependencies exist among systems. In the LaSalle fault trees, these interdependencies existed between the power distribution system (PDIST) and its support systems (e.g., the heating, ventilation, and air-conditioning system, HVAC,

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* T. L. Zimmerman, N. L. Graves, A. C. Payne Jr., and D. W. Whitehead, "Microcomputer Applications of and Modifications to the Modular Fault Trees," NUREG/CR-4838, SAND88-1887, Sandia National Laboratories, Albuquerque, NM, to be published.

Table 3.1
Fault Tree Segments Developed for the LaSalle Analysis

1.	PDIST	Power Distribution - Includes AC and DC power buses and circuit breakers and the diesel generators.
2.	EPAV1	Electrical Actuation - Part 1 - Includes the actuation circuitry for the AC and DC power circuit breakers and diesel generators.
3.	EPAV2	Electrical Actuation - Part 2
4.	EPAV3	Electrical Actuation - Part 3
5.	HVAC	Heating, Ventilation, and Air-Conditioning Systems - Includes diesel-generator facilities ventilation system and ECCS equipment areas cooling system.
6.	CSCS	Core Standby Cooling System - Includes diesel generator and ECCS room and pump cooling.
7.	LPCS	Low Pressure Core Spray System
8.	LPCI	Low Pressure Coolant Injection System - Mode of RHR.
9.	CSS	Containment Spray System - Mode of RHR.
10.	SCS	Shutdown Cooling System - Mode of RHR.
11.	SPC	Suppression Pool Cooling - Mode of RHR.
12.	ADS	Automatic Depressurization System
13.	RCIC	Reactor Core Isolation Cooling System
14.	HPCS	High Pressure Core Spray System
15.	PCS	Power Conversion System - Includes main steam system and condenser.
16.	MFW	Main Feedwater System
17.	CDS	Condensate System
18.	SWS	Service Water System
19.	TBCCW	Turbine Building Closed Cooling Wat System

Table 3.1
Fault Tree Segments Developed for the LaSalle Analysis

20.	IA	Instrument Air system
21.	DWN	Drywell Pneumatic System (Instrument Nitrogen)
22.	RPT	Recirculation Pump Trip
23.	SBLC	Standby Liquid Control System
24.	VENT	Containment Venting System
25.	RBCCW	Reactor Building Closed Cooling Water System
26.	CRD	Control Rod Drive System (two pumps needed)
27.	CRD1	Control Rod Drive System (one pump needed)

and the core standby cooling system, CSCS, which provide room cooling and pump and seal cooling to the AC power generating equipment and its support equipment) and within the PDIST system itself (e.g., AC/DC power dependencies). Figure 3.1 shows the logical connections that resulted in feedback effects in the LaSalle analysis and the solution used to resolve these dependencies. The solution was implemented in the following fashion.

First, the PDIST, CSCS, and HVAC systems were duplicated with all of the gate names changed to create different but logically equivalent systems (i.e., the primary event names remain the same). This was accomplished with the FRMNEWFT procedure of SETS using the NAME option. Gate names were changed by appending a "1" to the end of each gate name. To insure no gate name exceeded sixteen characters in length (SETS will not accept names longer than 16 characters), the first occurrence of a hyphen was removed from each gate name. In the loop-cutting version of PDIST, PDIST-LC; gates connecting to the front-line systems were removed using the TRIM option with the FRMNEWFT procedure of SETS so that PDIST-LC only fed into the PDIST fault tree and its support systems. The logic loops all involved diesel generators and batteries depending upon themselves through their support systems. The connections back into the support systems were removed from the loop cut versions of the fault trees (i.e., when the loop cut version of the CSCS system, CSCS-LC, was merged with the loop cut version of the PDIST tree, PDIST-LC; the connection back to the CSCS-LC tree via the diesel generator was removed). Appropriate gates in the electrical actuation fault tree EPAVALL were renamed to reflect the appended "1" in the loop cut versions of the other systems so the actuation logic would feed into the PDIST-LC fault tree (no logic loops went through the actuation circuits so a duplicate actuation tree was not required). The three loop-cut systems PDIST-LC, CSCS-LC, and HVAC-LC were merged with PDIST and EPAVALL to form a merged-power fault tree MERGED-PWR with all logic loops removed. This fault tree was later merged with the front-line system fault tree segments and the original support system fault trees to create the completed systems.

Although fault tree size is always a problem of concern, it's seldom of the magnitude encountered in the LaSalle analysis. For this reason, selected fault trees were merged in small groups and these groups were later merged into the one final, large, multi-topped fault tree. This helped in a number of ways: (1) duplicated logic was eliminated early in the merging process, (2) errors were more easily resolved, and (3) SETS runs were of a manageable size in terms of time and output. Front-line system fault trees were eventually all merged into one large group and the fault trees for the supporting power systems into another group. The two groups still contained over 10,000 gates which is more than the largest version of the SETS code could handle without code rewriting. To solve this problem, Form Two of the FRMNEWFT procedure of SETS was used to coalesce and remove single input gates from the fault tree group containing the front-line systems. This reduced the number of gates enough so that it could then be merged with the supporting power system fault tree group to form one very large multi-topped fault tree containing all the front-line systems complete with their supporting systems. The SETS output was carefully reviewed at this point to

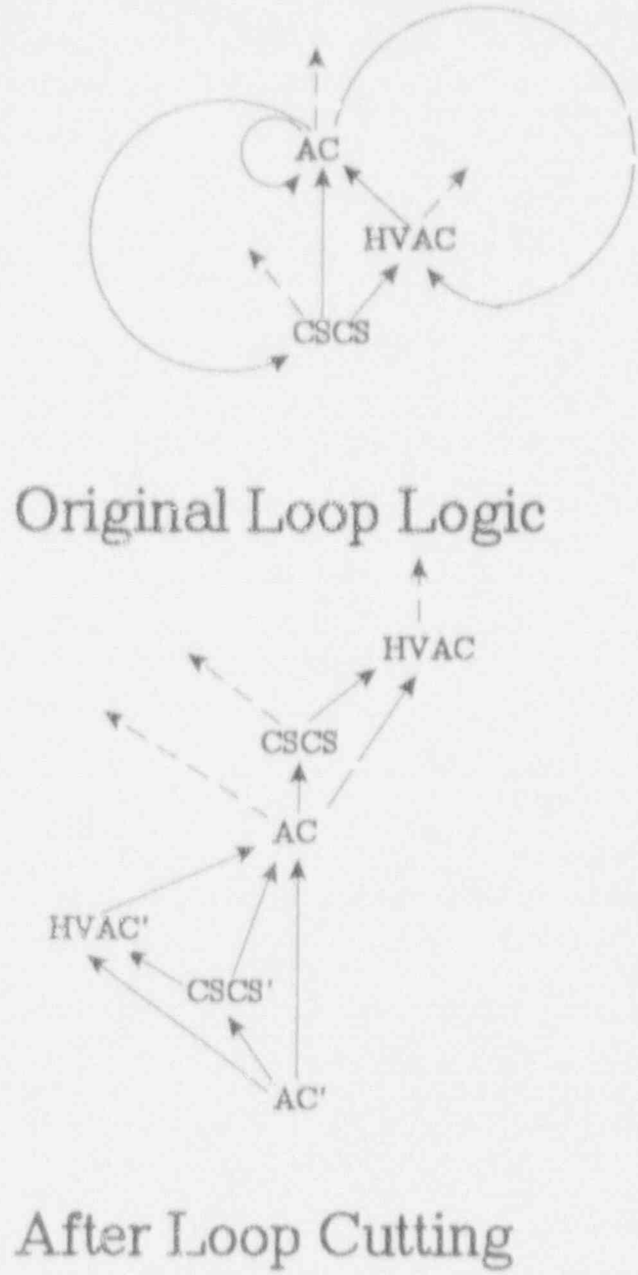


Figure 3.1
Example Fault Tree Logic Loop Resolution

insure (1) the coalescing and single input gate removal did not sever any connections between front-line and support systems (this could occur if the connecting gates were single input gates and were deleted from the tree), (2) no developed events remained in the final merged tree, and (3) each system appearing on an event tree that was represented by a fault tree was represented by a top gate on the merged fault tree.

3.3 Development of Independent Subtrees

The LaSalle system fault trees are large and complex representing the interactions of many support systems and primary events. Even with the use of the SETS computer code on a large mainframe, it is not possible or economical to identify all the minimal cut sets of a system fault tree. One technique that reduces the fault tree size problem is the identification and solution of the largest independent subtrees. Form Three of the FRMNEWFT procedure performs this function by restructuring and then separating a designated fault tree into its stem and a collection of independent subtrees. Independent subtrees can be quantified and evaluated individually and replaced by developed events in the system fault trees (i.e., these portions of the fault trees are treated as single super events). This process was very beneficial when applied to the LaSalle fault trees. The LaSalle trees contain 3451 primary events. This SETS procedure identified 80 existing independent subtrees and created an additional 283 subtrees. These 363 independent subtrees isolated 2928 of the primary events. The use of these subtrees as "super events" resulted in a smaller tree and more efficient solving of the original trees. These events must be resubstituted at the end of the analysis to obtain results in terms of the primary events on the original tree. A thorough and in-depth discussion of the development of independent subtrees is found in NUREG/CR-3547.[8]

3.4 Solving for System Fault Tree Minimal Cut Sets

Even with the use of independent subtrees, the front-line system fault trees for LaSalle were very large. Obtaining an exact listing of minimal cut sets for such large trees is difficult, expensive, and often impossible. For these reasons, it was necessary to probabilistically eliminate cut sets below a selected truncation value. A truncation value of 1E-08 was selected since previous experience has shown that the dominant PRA-estimated core damage sequence frequencies before the application of recovery are in the 1E-04/R-yr. to 1E-05/R-yr. range and that a significant number of cut sets will be retained using this truncation value to give good estimates of the dominant sequence frequencies. Every primary event in the fault tree must have a probability value associated with it in order to eliminate cut sets based on probability. Since independent subtrees are treated as "super events," they too must have a value associated with them. The Generate Fault Tree procedure, GENFTEQN, of SETS was used on the collection of independent subtrees to generate a Boolean equation containing the minimal cut sets for each independent subtree. The Compute Term Value procedure, COMTRMVAL, was used to obtain the sum of the probabilities of the minimal cut sets for each independent subtree. This approximation associated with each "super event" or independent subtree was then added to the list of point value probabilities for each of the primary events for use in computing the minimal cut sets for each of the front-line system equations.

The screening or point estimate values associated with each primary event for computation in this phase of the analysis should be the largest value ever to be associated with the event. Events having smaller values for certain sequences can be reduced later. However, if it becomes necessary to increase the probability value for any primary event after the system cut sets have been obtained, the system cut sets should be resolved. Some cut sets may have been eliminated by the use of the smaller probability value. Having to repeat the process to obtain system cut sets can be very costly and is better avoided by careful review and use of the highest value.

Even with the use of independent subtrees and truncation, obtaining the cut sets for the LaSalle front-line systems was very difficult. The SETS code gives the computer analyst tremendous flexibility in solving fault trees. This flexibility, that allows an analyst to solve very large complex structures, requires the computer analyst to exercise a considerable amount of responsibility in generating and executing the details of a SETS user program. The computer analyst should have a detailed knowledge of the fault tree structure and work very closely with the systems analyst during the front-line system solution effort. Failure to recognize this responsibility can result in excessive computer costs and minimal results.

The size and complexity of the LaSalle system fault trees necessitated careful review of the front-line systems prior to attempting to solve for the system minimal cut sets. Computer output from the Print Block procedure, PRTBLK, was reviewed for each front-line system. This computer output gives the analyst insight into the coalescing and restructuring that occurs during the merging of the individual system fault trees. Simple sketches showing the logic structure can be generated and can be used to determine modifications to the SETS user code to optimize the solution as described below. After reviewing the restructured front-line system fault trees, the Generate Fault Tree Equation procedure was used to generate the SETS user code to solve a given front-line system using the bottom-up method. The PUNCH option was included to prevent the SETS code from attempting to execute the generated code.

A "quasi" bottom-up method was used to solve the stem portion of the front line system fault trees. The bottom-up method generates Boolean equations for selected intermediate events starting from the bottom of the fault tree. Each equation is reduced as it is generated. Progression is made through successively higher levels of the fault tree until the top gate is reached. After reduction, the top-gate equation is a function of only primary events or primary events and independent subtrees (i.e. "super events") if only the stem portion of a fault tree is being solved. A detailed explanation of the bottom-up method of the Generate Fault Tree procedure of SETS is given in

Reference 2. A discussion of its use in accident sequence analysis is found in Reference 3. The approach taken to obtain solutions of the LaSalle fault trees was a "quasi" bottom-up method because the SETS control program produced with the Generate Fault Tree procedure was modified considerably prior to execution of the user code. These modifications included: (1) inclusion of additional stopping points (i.e. selected intermediate events which are solved to obtain their Boolean equations), (2) review and changes to the user code for solution of "AND" gates, (3) use of equations to equate equivalent gates, and (4) removal, insertion, and changes to Delete Block (DLTBLK) and Form Block (FRMBLK) statements.

By review of the SETS user code from Generate Fault Tree Equation and the PRTBLK output, the computer analyst can prepare a list of intermediate events to be used as stop points. Stop points determine a stage in the development of an expression and act as a sort of "temporary super event." Since intermediate events are not normally assigned values, stop points are excluded from computation by use of the Except Noncomplement, EXCEPTNONCMP, option in the Truncate on Term Value statement corresponding to the Substitute Equation having the STOP option.

The stop points are either "AND" gates or intermediate events used multiple times in the fault tree. The use of stop points allows the analyst to solve the cut set equation for an interim gate in piecemeal fashion. Allowing only one or two of the equations for the inputs of a gate to enter the equation for the gate at one time will greatly reduce the number of terms that will be generated by expansion. After simplification, additional inputs can be released.

The use of stop points is particularly effective for multiple input "AND" gates. In this case, the order in which stop points are released can be important. If an "AND" gate, Q, has inputs A, B, C and D, and A and B are known to have events in common, the analyst should release inputs A and B while stopping on C and D. This will result in a smaller number of terms to be combined when the stop points C and D are sequentially released. Another item the computer analyst must monitor is that selected intermediate stop points have not already been solved in a previous computer run. This results in the stop point having no effect.

When individual systems are merged to form front-line system fault trees, the SETS code output contains a list of any occurrences of equivalent gates found in the merged fault tree. Equivalent gates are gates of the same type (i.e. both "AND" or "OR" gates) and having the same inputs. Often this information can be useful in simplifying the solution of an intermediate gate. If an "AND" gate, I, has inputs E, F, G and H, and E and F are equivalent gates, then equations can be used to reduce the number of inputs to gate I from four to three. For example, if E and F are "AND" gates both having inputs K and L then the following equations can be used to set the equivalent gates equal:

```
PROGRAM$ LASALLE-1.
        .
        .
        .
     E = TEMP.
     F = TEMP.
     TEMP = K * L.
     SUBINEQN (TEMP, TEMP).
        .
        .
        .
```

The computer analysis then precedes as if gate I had inputs G, H and TEMP.

The Delete Block and Form Block procedure statements were often removed from the user code at the lower gate levels to speed up computer run time. The name given to a block being formed was changed from the previously used block name to prevent a loss of information if a computer time-limit was encountered during a Delete Block or Form Block procedure. Additional Form Block statements were added to the code at the upper gate levels. This saved the computed information more often in case a computer restart was needed. Since the SETS block file is a sequential file, it is more economical to keep only essential information on the file so previous interim blocks were deleted once a new one was successfully formed. Admittedly, for small problems the computer cost involved in these procedures would probably be nominal but a significant savings is realized when dealing with extremely large system fault trees. Proper use of the Form Block procedure may save the computer analyst from "losing" a 30 minute computer run.

The computer code for solving a front-line system was generally broken into parts for submission to the computer. This allowed the computer analyst to review the output and make appropriate changes to the next section of code to be submitted if needed. This helped to control the computer cost and often prevented the submitting of a costly run that could not be successfully completed.

An equation was used to set the event HIGH-DWPRESSURE to OMEGA (i.e, the event is assumed to always occur) while obtaining the system minimal cut sets for all of the front-line systems except PCS. The event HIGH-DWPRESSURE was set to /OMEGA (i.e., PHI, the event was assumed never to occur) during the computation of the minimal cut sets for PCS. The event HIGH-DWPRESSURE is a flag that indicates the presence or absence of high pressure in the drywell. For all accident sequences where PCS was not available or successful, high drywell pressure (i.e., drywell pressure greater than 1.69 psig setpoint used in the emergency system's actuation logic) was assumed to occur.

The fifteen front-line systems and their number of minimal cut sets are shown in Table 3.2. The number of cut sets shown is prior to substitution for independent subtrees. Substitution for independent subtrees was not made until after the formation of sequences.

Table 3.2 Size of Merged Front-Line System Solutions

Abbreviation	Front Line System	*Number of Cut Sets
RPT	Recirculation Pump Trip	200
SBLC	Standby Liquid Control System	75
RCIC	Reactor Core Isolation Cooling System	317
HPCS	High Pressure Core Spray System	128
LPCS	Low Pressure Core Spray System	157
CDS	Condensate System	459
CSS	Containment Spray System	7920
SPC	Suppression Pool Cooling System	8014
SCS	Shutdown Cooling System	4361
LPCI	Low Pressure Coolant Injection System	5696
ADS	Automatic Depressurization System	2280
PCS	Power Conversion System	509
MFW	Main Feedwater System	610
CRD	Control Rod Drive System	186
VENT	Containment Venting System	289

* Number of cut sets prior to substitution for independent subtrees.

Once system solutions are obtained for all the front-line systems appearing on the accident sequence event trees, the accident sequences can be evaluated. The evaluation of the LaSalle accident sequences is described in chapter 4 of this report.

3.5 References

1. G. B. Varnado, W. H. Horton, and P. R. Lobner, "Modular Fault Tree Analysis Procedures Guide," NUREG/CR-3268, SAND83-0963, Sandia National Laboratories, Albuquerque, NM, August 1983.

2. R. B. Worrell, "SETS Reference Manual," NUREG/CR-4213, SAND83-2675, Sandia National Laboratories, Albuquerque, NM, May 1985.

3. D. W. Stack, "A SETS User's Manual for Accident Sequence Analysis," NUREG/CR-3547, SAND83-2238, Sandia National Laboratories, Albuquerque, NM, January 1984.

4.0 COMPUTATION OF SEQUENCES

Most of the LaSalle sequences were evaluated with a truncation value of 1E-08. This led to some difficulty in generating the sequence solutions as combining failure states often generated millions of intermediate cut sets. Obtaining a cross product of two or more system failures is expensive and sometimes impossible without employing techniques other than simply "ANDING" (i.e. the Boolean operation of conjunction) the systems together. A methodology for performing the accident sequence analysis portion of a PRA using SETS is discussed in detail in NUREG/CR-3547.[1] Additional techniques employed to obtain the LaSalle sequences included: (1) common term removal, (2) separation into parts based on number of literals, (3) separation into parts based on probability, (4) grouping, and (5) intermediate removal of success states.

4.1 Common Term Removal

If sufficient information is known about two systems to indicate that they have a number of cut sets in common, the combined system failure can be calculated without generating all of their intermediate cross products. For example, several of the LaSalle sequences required combining the system failure states of containment spray system (CSS) and suppression pool cooling (SPC) system. As shown in Table 3.2, even in terms of independent subtrees (ISTs) or "super events" these systems contained 7920 and 8014 cut sets, respectively. Since these two systems were known to contain a sizable number of cut sets in common, their cross product was obtained as follows. The Delete Term, DLTRM, procedure was used to obtain the terms of CSS not in common with SPC. Using DLTRM again, this result was then used to separate out the terms of CSS that were also in SPC. A third application of DLTRM was made to obtain the terms of SPC not in common with CSS. The resulting three terms represented: (1) terms in CSS but not in SPC, (2) terms in both CSS and SPC, and (3) terms in SPC but not in CSS. The terms of CSS not in common with SPC were then "ANDED" with the terms of SPC not in common with CSS. This result was then "ORED" (i.e. the Boolean operation of disjunction) with terms common to both CSS and SPC to obtain their complete cross product. Sample SETS code to execute this process is shown below:

```
PROGRAM$ LS1-SEQ.
COMMENT$ COMBINE SYSTEM CUTS SETS FOR CSS AND SPC $
          .
          .
          .

DLTRM (CSS, SPC, X1).
DLTRM (SPC, CSS, X2).
DLTRM (CSS, X1, Y).
CSS-SPC = X1 * X2 + Y.
SUBINEQN (CSS-SPC, CSS-SPC).
```

Some computer costs are heavily weighted to I/O operations. Since DLTRM makes heavy use of I/O, in some cases it may be more efficient to remove the second DLTRM statement from the above code and change the equation to CSS-SPC = X1 * SPC + Y. This would require that the Substitute in Equation procedure, SUBINEQN, be followed by either the Reduce Equation procedure, REDUCEQN, or Truncate on Term Value procedure, TRNTRMVAL, since the result of the SUBINEQN would not be minimal. If the product of two or more failures is common to more than one sequence, it is important to save this result using the Form Block, FRMBLK, procedure so that it is not necessary to compute the product more than once.

4.2 Separation Into Parts Based on Number of Literals

Some LaSalle sequences were extremely difficult and expensive to obtain. These sequences were developed in stages. The cut sets for two or more system failures in a sequence would be combined and the computer output examined before combining this segment of the sequence with other system failures to continue computation of the sequence. If the computer output indicated a segment could not be combined with another system without generating too many intermediate terms for the capacity of the computer code, the sequence segment was sometimes broken into parts. This was accomplished by using the option in the REDUCEQN procedure to truncate the sequence segment on number of literals, j. Using the sequence segment and the j-truncated sequence segment as arguments for the DLTRM procedure, the sequence segment containing greater than j literals was obtained. These two parts, the less than or equal to j literals part of the sequence segment and the greater than j literals part were each "ANDED" with the next system failure state to be included in the sequence and then the results are "ORED" to obtain the next stage. If necessary the process can be applied more than once, but since the computations to obtain the parts can be fairly expensive they should be kept to a minimum. The computer output from each stage in the development of a sequence is used to determine the value of j and whether or not this process is applicable.

4.3 Separation Into Parts Based on Truncation by Probability

Sometimes a large sequence segment would not lend itself to separation into parts based on number of literals (i.e. too many terms containing the same number of literals). In these cases, computer output was reviewed for the possibility of separation into parts based on truncation at some probability level. This process is similar to separation into parts based on number of literals except the TRNTRMVAL procedure is used to obtain a part of the sequence segment truncated at a higher probability value, k, than the value being used for the analysis. To determine the k probability value to be used as the break point requires the analyst to have some knowledge of the magnitude of the cut sets being generated and/or computer output from a COMTRMVAL procedure for the sequence segment or systems composing the sequence segment. The DLTRM procedure is applied to obtain the portion of the sequence segment having probability less than k. As above, the two parts of the sequence segment are combined with the next system or systems of the sequence and then the results are "ORED" to obtain the next stage.

4.4 Grouping

Two types of grouping were used in computing the LaSalle sequences. The first type involved combining and saving combinations of systems that were used in more than one sequence. Combining many of the systems generated a large number of intermediate cut sets which resulted in high computer charges. Because of these computer costs, combinations of systems found in two or more sequences were often formed and the results saved using the Form Block procedure, FRMBLK. These system combinations could then be recalled as needed during a sequence computation.

The second type of grouping used in computing the LaSalle sequences selected systems to be combined based on known commonalities. When combining several systems that create a large number of interim cut sets, the order in which the systems are combined can become very important. Combining two or more systems known to have many cut sets in common prior to combining these systems with another system which does not have cut sets in common with the previous systems generates fewer intermediate terms which have to be eliminated in the Reduce Equation procedure. Obviously, these "groupings" are very judgemental and require the analyst to have or obtain considerable information about the interactions of the physical systems being modeled.

Occasionally, the two types of "grouping" are in conflict with each other. The first type discussed generally helps in reducing the cost of obtaining a solution while the second type of grouping may control whether or not the solution can even be obtained. Unless costs become a major concern, grouping to reduce the number of terms is generally the major deciding factor in dealing with very large problems.

4.5 Intermediate Inclusion of System Success States

The Delete Term procedure, DLTRM, can be used to include the success states of a system in an accident sequence without determining a complement equation for the system. For example, suppose we have the failure equations for two systems, p and q, in disjunctive normal form (i.e., sum of products (cut sets) as opposed, for example, to a factored form). The sequence we wish to evaluate is given by the equation s = p*/q where system p has failed and system q has succeeded. Instead of determining explicitly the complement of q, /q, (which can have a very large number of success cut sets and is usually not done); we delete terms in the equation for p that subsume terms in the equation for q from the equation for p to form a new equation, r. This means that cut sets in the failed system that are physically incompatible with the fact that the other system succeeded are removed from the failed system's equation. The sequence can then be approximated as s = r. In general, the probability of success is usually close to 1.0 probability and the improvement in the estimation of the sequence frequency by elimination of cut sets physically incompatible with the sequence definition more then compensates for the error introduced by not including the probability of success. If the probability of success is small, then a more explicit representation may be needed. The number of terms in equation r will be smaller than the number of terms in equation p unless the systems involved are independent of each other. A more precise discussion of this process is found in NUREG/CR-4213.[2]

Because the inclusion of a system success in this manner has the effect of reducing the number of cut sets in a sequence segment, it is often beneficial to include system successes at intermediate stages of a sequence computation. It results in fewer cut sets in a sequence segment that must still be combined with other system failures. It is important to remember that when a success state for a system is included in a sequence segment prior to combining the last failure system to the sequence segment, it will be necessary to combine the success state again. This is to insure that terms of the success state system have been removed from all of the failure systems occurring in the sequence. Analysts' judgement and familiarity with the modeled systems must be used to determine when including success states at intermediate stages will be useful. Also, when using a particular combination in several different sequences, one must be careful to use only success states that appear in both sequences or evaluate the combination twice, once for each sequence.

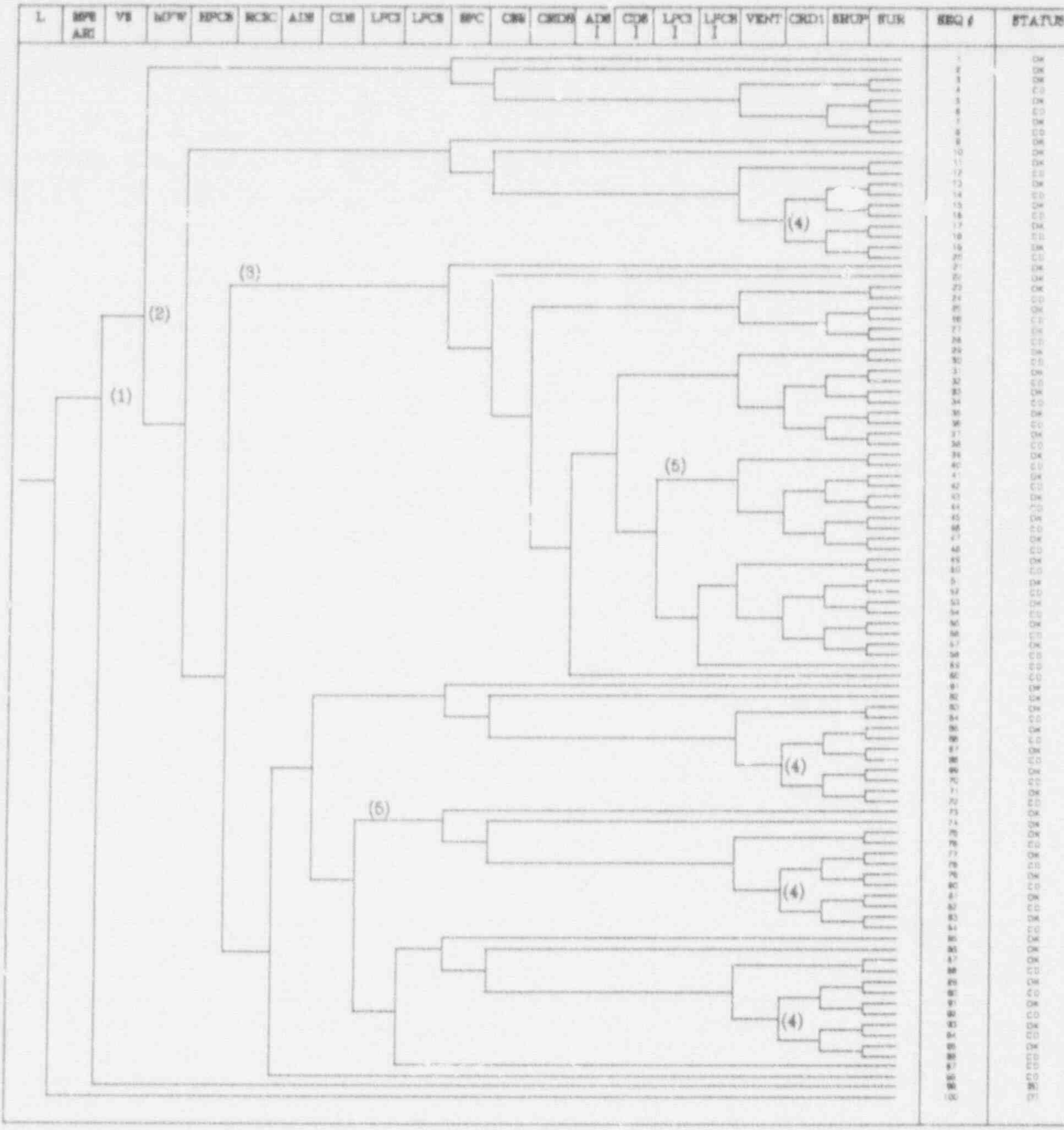
4.6 Solution of Sequences

The evaluation of the LaSalle sequences required the use of all of the techniques discussed above. Some types of sequences, such as the transient and transient-LOCA sequences, were extremely difficult to compute. In some cases, it was not feasible to obtain all of these sequences at the probability truncation value of 1E-08.

4.6.1 LOCA Sequences

The event trees, discussed in Chapter 2 of Volume 4 of this report determined the sequences to be evaluated for Loss of Coolant Accidents, LOCAs. The LOCA event tree is reproduced here as Figure 4.1. Because the severe environment and containment failure expert elicitations had not been performed by the time the screening was to be done, the events SRUP and SUR were not evaluated at this time (see Chapter 6 for a discussion of these events).

Since the LOCA tree was evaluated simultaneously for all LOCA sizes (small, medium, and large), any system specific effects due to the different LOCA



(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 4.1
LaSalle LOCA Event Tree

sizes had to be included directly in the system fault trees. For example, the reactor core isolation cooling (RCIC) system fails due to its inability to supply enough water to make up for the coolant being lost and due to the reactor vessel depressurization that occurs after a medium or large LOCA. Two events representing a medium and large LOCA are placed in the RCIC system fault tree such that, if a medium or large LOCA occurs, the RCIC system fai.s. For other events such as electrical bus failures only partial system failure may result.

Each sequence was multiplied by an initiating event equation to insure every cut set included an appropriate LOCA initiator as indicated by the event trees. After the systems are solved and combined together to form the selected accident sequence, two types of cut sets will be present: 1) cut sets with no initiators coming from the fault trees (i.e., cut sets composed only of random failures of equipment from the failed systems) or 2) cut sets with one or more initiators and possibly some random failures. In order to complete the sequence definition, each cut set must have an initiating event. Those cut sets which already have an initiating event coming from the fault tree solution are complete. Cut sets with multiple initiators are not physically realizable since by definition only one initiating event occurs at a time. The fault trees already contain random events representing the occurrence of an initiator as a random failure given the occurrence of some other initiator. The method used to eliminate these double initiator cut sets will be discussed later. Cut sets with no initiators are independent of the specific initiator type and need to be combined with each initiator to create new cut sets, one for each initiator (i.e., given a cut set X*Y and the three initiators LLOCA, MLOCA, and SLOCA; three cut sets can be created LLOCA*X*Y, MLOCA*X*Y, and SLOCA*X*Y by "ANDING" the cut set with the equation LLOCA + MLOCA + SLOCA).

For sequences one through sixty, the initiating event equation included a small, medium, and large LOCA initiator. Sequences sixty-one through ninety-eight each contained two parts; the first part received a small and medium initiator while the second part received only a large LOCA initiator. This was because, for a large LOCA, the automatic depressurization system (ADS) is not necessary to depressurize the reactor vessel in time for the low pressure injection systems to prevent core damage. Since the initiator does not fail the ADS system but merely renders it unnecessary, the sequences were first evaluated without including ADS success or failure. These cut sets were "ANDED" with the large LOCA initiator to form the large LOCA cut sets. The original cut sets were then combined with ADS success or failure, as appropriate, and the resulting cut sets were "ANDED" with the small and medium LOCA initiators. The two parts of each sequence were then "ORED" together to form the complete sequence. Equations were used to set the transient initiators to /OMEGA (i.e. OMEGA means the event always occurs, PHI = /OMEGA means that the event never occurs) for the LOCA sequence evaluation. This was necessary to remove transient initiators appearing in the cut sets as a result of their inclusion in the fault trees. For some events, the probability of occurrence is different for different sequences. During screening, a single value, the maximum value that can occur in any sequence, is used so that one value can be used and no cut set will be truncated unnecessarily. In the final evaluation of specific sequences, the data used to quantify the events is assigned it's appropriate value. Equation and value block changes used for the LOCA sequence computations are listed in Tables 4.1 and Table 4.2, respectively.

Complement events were not used in the construction of the LaSalle fault trees. This occasionally led to the same primary event being modeled in a different state in various systems and being given a different event name. For example, a valve might be modeled as failed open in one system while the same valve is modeled as failed closed in another system. Combining the system cut sets for these two systems in a sequence could result in a cut set that would not be logically valid since the same valve can not fail both open and closed in the same sequence. Events modeled in more than one failure condition were "flagged" during modeling. An equation containing the products of these "flagged" events was used with the DLTRM procedure to remove the logically invalid or "double-flagged" cut sets from the sequence.

Cut sets containing double initiating events were also considered unnecessary to the analysis. These cut sets were removed in the same manner as the "double-flag" cut sets.

After the "double-flags" and "double-initiators" were removed, substitution was made for the ISTs to obtain LOCA sequence cut sets containing only primary events. Only sequences L4, L6, L8, L12, L14, L16, L18, L20, L24, L26, L28, and L97 had cut sets remaining after this substitution and truncation at 1E-8.

4.6.2 Transient-Induced LOCA Sequences

The event trees, discussed in Chapter 2 of Volume 4 of this report, determined the sequences to be evaluated for transient-induced Loss of Coolant Accidents. The LOCA ev t tree, Figure 4.1, was used to evaluate the transient-induced LOCA sequences. These sequences start out on the transient event tree shown in Figure 4.2 with successful scram and safety relief valve (SRV) opening. The SRVs do not reclose and, depending upon the number of SRVs which fail open, are equivalent to a small, medium, or large LOCA in their effects on system operation and RPV inventory. Because the severe environment and containment failure expert elicitations had not been performed by the time the screening was to be done, the events SRUP and SUR were not evaluated at this time (see Chapter 6 for a discussion of these events).

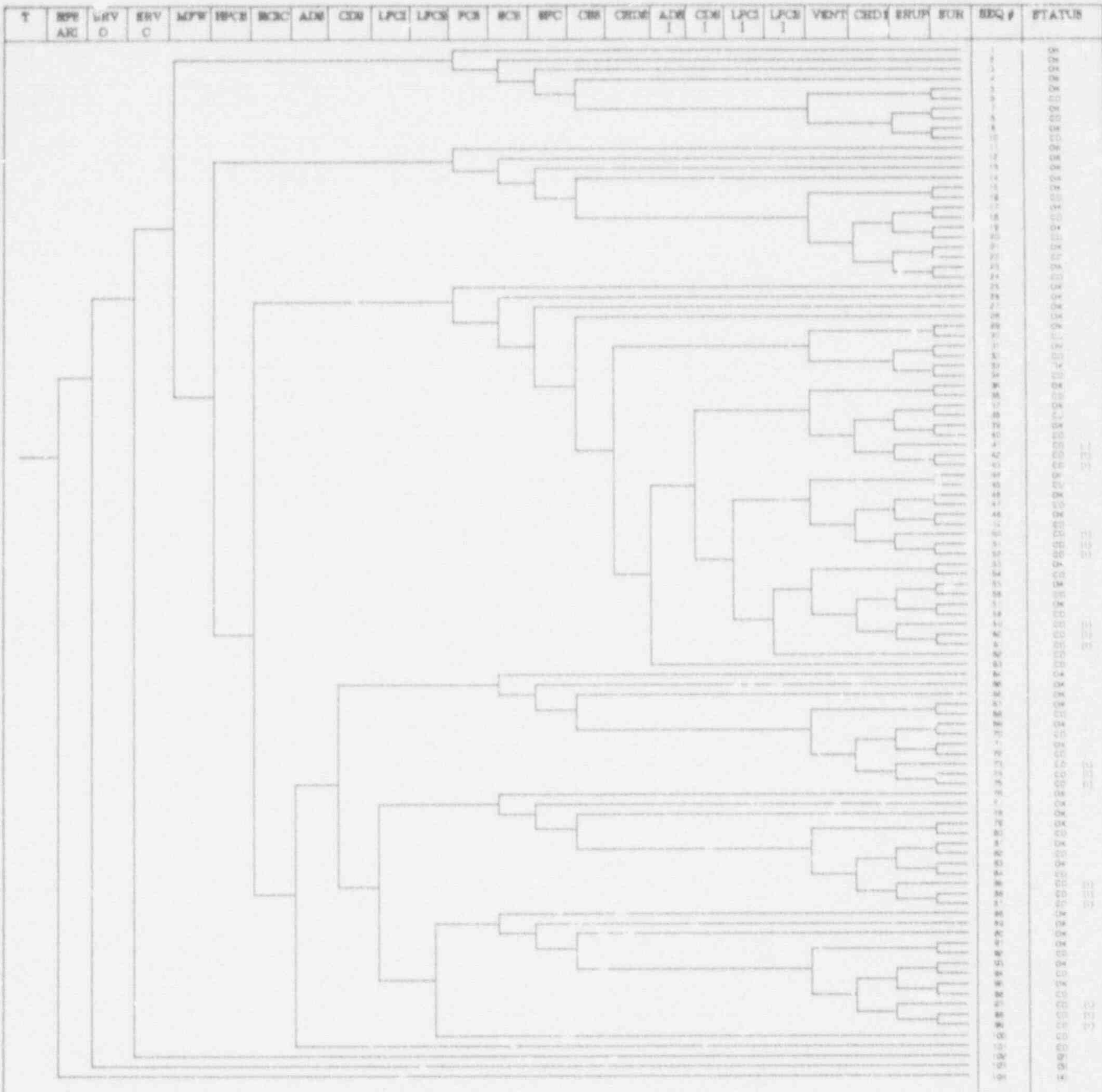
In a similar fashion as for the LOCA sequences, each sequence was multiplied by a transient initiating event equation to insure an initiator was included in each cut set. However, the event SRV C on the transient event tree which represents the transient-induced LOCA was not developed into a full fault tree. A Boolean equation SRV C = $Q_1 + Q_2 + Q_3$ was used to represent this event where Q_1 represents the probability of one of the SRVs demanded open failing to reclose, Q_2 represents the probability of

Table 4.1
Transformation Equations for Initiators and Flags in LOCA Sequence Evaluation

```
BLOCK$ LOCA-PHI-OMEGA.
HIGH-DWPRESSURE = OMEGA.
NOHIGH-DWPRESS = /OMEGA.
T1-IE = /OMEGA.
T2-IE = /OMEGA.
T3-IE = /OMEGA.
T4-IE = /OMEGA.
T5-IE = /OMEGA.
T6-IE = /OMEGA.
T7-IE = /OMEGA.
LOSP-IE = /OMEGA.
T9A-IE = /OMEGA.
T9B-IE = /OMEGA.
T101-IE = /OMEGA.
T102-IE = /OMEGA.
T11-IE = /OMEGA.
T12-IE = /OMEGA.
T13-IE = /OMEGA.
T14-IE = /OMEGA.
T15A-IE = /OMEGA.
T15B-IE = /OMEGA.
```

Table 4.2
Value Block Changes for LOCA Sequence Evaluation

```
COMMENT$ CHANGES FOR ALL VALUE BLOCKS $
 3.4E-3 $ RHR301AX-STR $
 3.4E-3 $ RHR301BX-STR $
 3.4E-3 $ RHR301CX-STR $
 3.4E-3 $ LCSD302X-STR $
 1.2E-3 $ RCID001X-STR $
 1.2E-3 $ HCSD001X-STR $
COMMENT$ CHANGES FOR LOCAS AND TRANS-LOCAS $
 .1     $ TDRFP-T-OE $
 .1     $ MFS-RESET-OE $
 .1     $ ADSMINIT-QOO-OE $
 .01    $ OPERR-INITCSS $
 .1     $ OPFAILS-REOPEN $
 0.0    $ OPTURNSOFF-TURB $
 0.0    $ TRN-A-SCSMODE $
 0.0    $ TRN-B-SCSMODE $
 0.0    $ TRN-AORB-SCSMODE $
```



(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 4.2
LaSalle Transient Event Tree

exactly two of the SRVs demanded open failing to reclose, and Q_3 represents the probability of three or more of the SRVs demanded open failing to reclose. These Qs are equivalent to a small, medium, and large LOCAs respectively. The LOCA initiators appearing in the fault tree were changed to the appropriate Q using transformation equations to represent the effects of the stuck open SRVs on the responding systems. These equations and their associated probability values for each event are shown in Table 4.3. Other events having changes for probability values for transient-induced LOCA sequence evaluation were the same as those shown in Table 4.2 for the LOCA sequence evaluation.

As described in Section 4.5.1, sequence cut sets containing "double-flags" were removed. Cut sets containing two transient initiators were also eliminated from the sequence cut sets. However, cut sets containing the transient initiator T7 which represents a stuck open SRV as an initiating event had to be treated differently since T7 and Q_1 are equivalent. Cut sets with $T7*Q_1$ were transformed to cut sets with only T7 while cut sets with $T7*(Q_2 + Q_3)$ were deleted. The sequences one through sixty and sixty-one through ninety-eight were then evaluated in the same fashion as for the LOCA sequences.

The transient-induced LOCA sequences were evaluated using a probability truncation value of 1E-08. After the "double-flags" and "double-initiators" were removed, substitution was made for the ISTs to obtain the transient-induced LOCA sequence cut sets containing only primary events. Only sequences TL4, TL6, TL8, TL12, TL14, TL16, TL18, TL20, TL24, TL26, TL28, TL30, TL32, TL34, TL36, TL38, TL59, and TL97 had cut sets remaining after this substitution and truncation at 1E-08. Although not all sequences had a large number of cut sets in their solution, most of the transient-induced LOCA sequences were difficult to compute and required considerable use of the techniques described in Sections 4.1 to 4.4.

4.6.3 Transient Sequences

The event trees, discussed in Chapter 2 of Volume 4 of this report, determined the sequences to be evaluated for transients. The transient event tree, Figure 4.2, was used to evaluate the transient sequences. Because the severe environment and containment failure expert elicitations had not been performed by the time the screening was to be done, the events SRUP and SUR were not evaluated at this time (see Chapter 6 for a discussion of these events).

Like the LOCA and transient-induced LOCA sequences the transient sequences were multiplied by a transient initiator equation. Values for the LOCA initiators were set to zero. Probability value changes were made for the events listed in Table 4.4. Cut sets containing "double-flags" and "double-initiators" were eliminated in the same manner as for the LOCA and transient-induced LOCA sequences.

Computation of the transient sequences was extremely difficult. Even with the use of all the techniques described in Sections 4.1 to 4.4, the

Table 4.3
Transformation Equations for Transient-Induced LOCA Sequences

SLOCA-IE = Q1
MLOCA-IE = Q2
LLOCA-IE = Q3

Q = Q1 + Q2 + Q3

PROBABILITY VALUE CHANGES FOR TRANSIENT LOCA

Q1 = .1
Q2 = 4.5E-3
Q3 = 1.2E-4

(ALSO ALL EVENTS LISTED IN TABLE 4.2)

Table 4.4
Value Block Changes for Transient Sequences

```
COMMENT$ CHANGES FOR ALL VALUE BLOCKS $
 3.4E-3 $ RHR301AX-STR $
 3.4E-3 $ RHR301BX-STR $
 3.4E-3 $ RHR301CX-STR $
 3.4E-3 $ LCSD302X-STR $
 1.2E-3 $ RCID001X-STR $
 1.2E-3 $ HCSD001X-STR $
COMMENT$ CHANGES FOR TRANSIENT SEQUENCES $
 .01 $ TDRFP-T-OE       $
 .01 $ MFS-RESET-OE     $
 .01 $ ADSMINIT-QOO-OE $
 .01 $ OPERR-INITCSS    $
 .1  $ OPFAILS-REOPEN  $
 0.0 $ OPTURNSOGC-TURB $
 1.0 $ TRN-A-SCSMODE    $
 1.0 $ TRN-B-SCSMODE    $
 1.0 $ TRN-AORB-SCSMODE$
 .01 $ FCSS2-Q-OE-O     $
 .01 $ C34R601A-Q-OE    $
 .01 $ C34R601B-Q-OE    ^
 .01 $ 1EG0EX-QCO-OE
 .01 $ 2HSFW032-Q-OE-O $
COMMENT$ SET VALUES FOR Q1,Q2, AND Q3 $
 0.0     $ SLOCA-IE, Q1 $
 0.0     $ MLOCA-IE, Q2 $
 0.0     $ LLOCA-IE, Q3 $
```

truncation value had to be relaxed in order to obtain the cut sets for the transient sequences. Sequences twenty-five through one hundred and one were truncated at 5E-08. Sequences one through twenty-four were truncated at 5E-07. All core damage sequences survived the truncation process before the inclusion of the severe environment failures and the application of recovery.

4.6.4 Anticipated Transients Without Scram

The event trees, discussed in Chapter 2 of Volume 4 of this report, determined the sequences to be evaluated for anticipated transients without scram (ATWS) events. The ATWS event tree, Figure 4.3, was used to evaluate the ATWS sequences. Because the severe environment and containment failure expert elicitations had not been performed by the time the screening was to be done, the events SRUP and SUR were not evaluated at this time (see Chapter 6 for a discussion of these events).

The event 1EDC2DEP-FROP-4 which represents DC battery depletion was set to /OMEGA (i.e. PHI) to eliminate the effect of battery depletion for the two systems RPT and SBLC. These systems must perform their functions within the first few minutes of the accident and battery depletion will not occur for several hours; therefore, battery failure can not be a failure mechanism for these systems. Events for which probability changes were made are listed in Table 4.5. Two point estimates were used in the computation of the ATWS sequences. They included: (1) FWL which is represented by the event OPFAILSMFW-8M and is failure of the operator to control feedwater level in an ATWS scenario to a level consistent with condenser makeup limitations within eight minutes, and (2) RPS/ARI, reactor protection and alternate rod insertion systems fail. The screening values used for these point estimates are also listed in Table 4.5.

The ATWS sequences were multiplied by an initiator equation to insure every cut set contained an initiating event. Cut sets containing "double-flags" and "double-initiators" were eliminated in the same manner as for the LOCA, transient-induced LOCA, and transient sequences.

Because of the magnitude (1.0E-05) of the point estimate for RPS in the ATWS sequences, system cut sets were truncated at 1.0E-04 prior to forming the ATWS sequences. The truncation of system cut sets at 1.0E-04 made the sequences easier to compute since fewer terms were generated while combining systems. The overall truncation level was equivalent to 1.0E-09 except for initiators with frequencies greater than 1.0/R-yr. However, the largest of those was 4.5/R-yr. so in all cases the truncation level was at least 1.0E-08/R-yr.

After substitution for the ISTs, the following sequences survived the truncation process: A14, A15, A17, A18, A22, A48, A49, A51, A52, A54, A55, A57, A58, A60, A61, A76, A77, A93, A119, A120, A122, A123, A125, A126, A128, A129, A131, A132, A147, and A148.

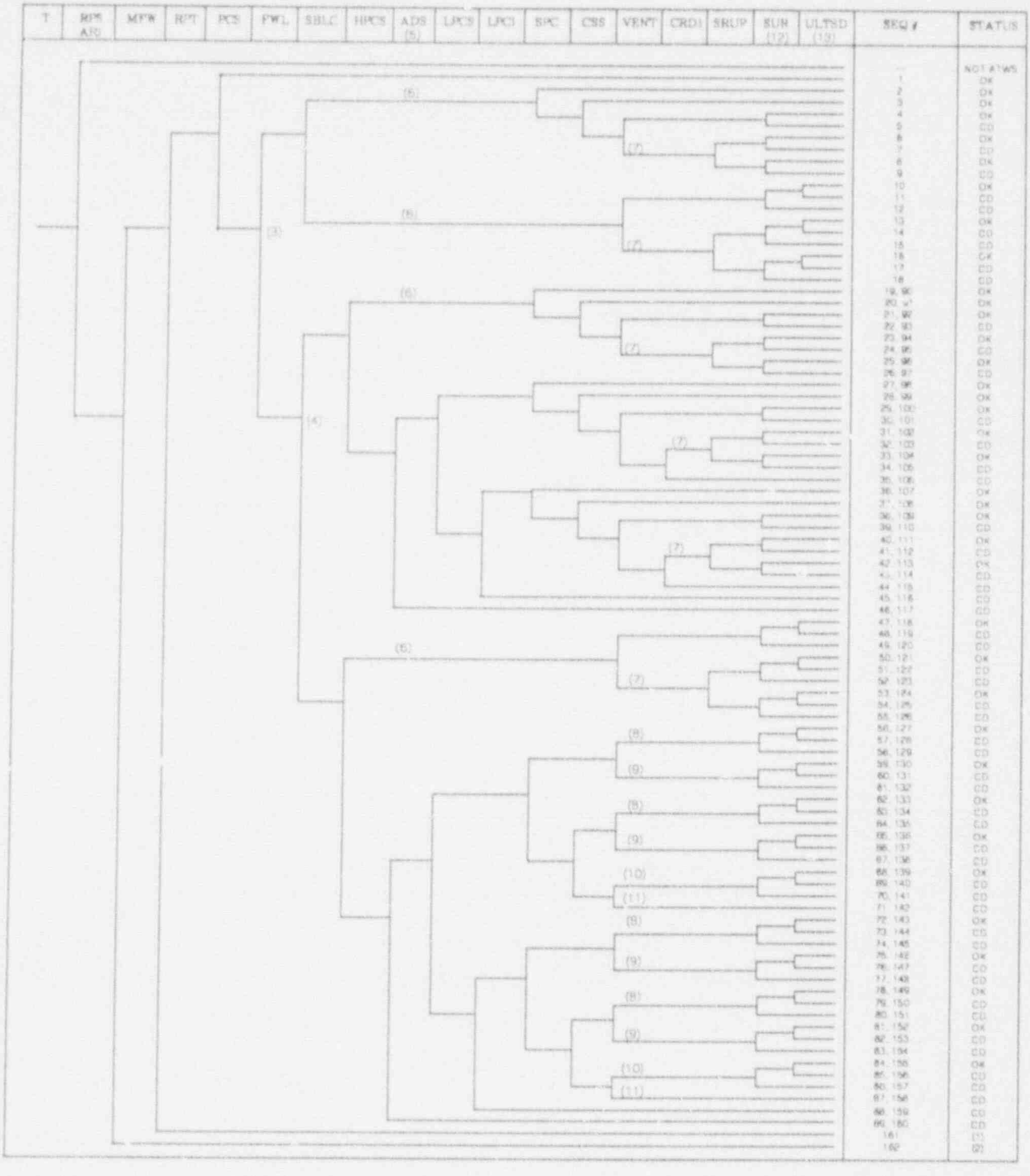


Figure 4.3
LaSalle ATWS Event Tree

Figure 4.3 LaSalle ATWS Event Tree (Continued)

1. If MFW (main feedwater) succeeds, RPT (recirculation pump trip) 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 (recirculation pumps). If RPT does fail, either PCS (power conversion system) 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 (reactor pressure vessel) level can not 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.

4. Transfer sequences from (2).

5. Operators are instructed by EOPs (emergency operating procedures) not to use inhibit switch for ADS (automatic depressurization system) but to reset timer.

6. For cases where no choice is given, ADS success or failure will not affect sequence timing or end result significantly. If the operator opens the SRVs (safety relief valves) to bring pressure down or auto ADS occurs due to low level, power will increase from about 12% to about 18%. LTAS code calculations, described in Volume 4 of this report, show that ADS and subsequent HPCS (high pressure core spray), LPCS (low pressure core spray), or LPCI (low pressure coolant injection) 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.

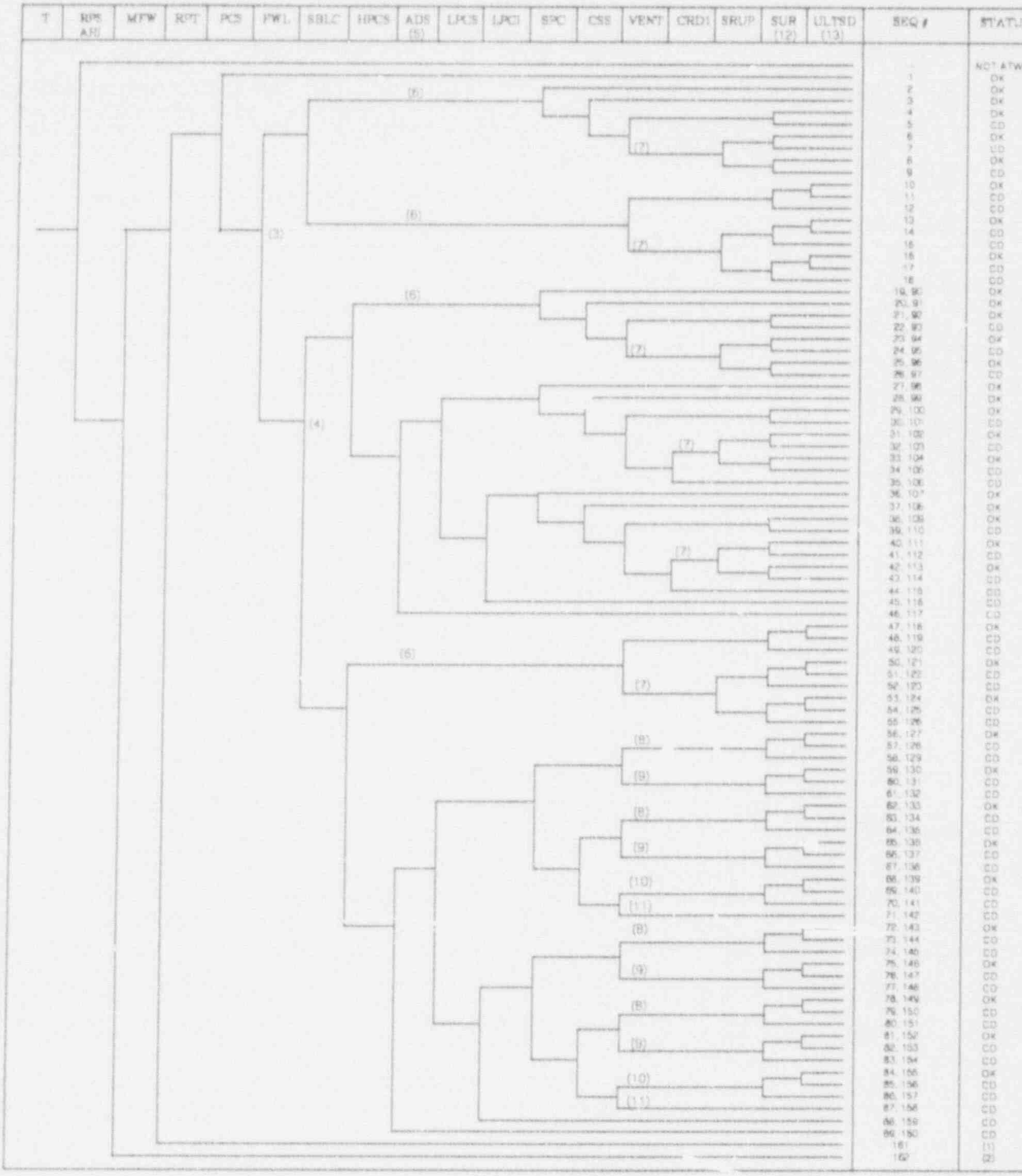


Figure 4.3
LaSalle ATWS Event Tree

Figure 4.3 LaSalle ATWS Event Tree (Continued)

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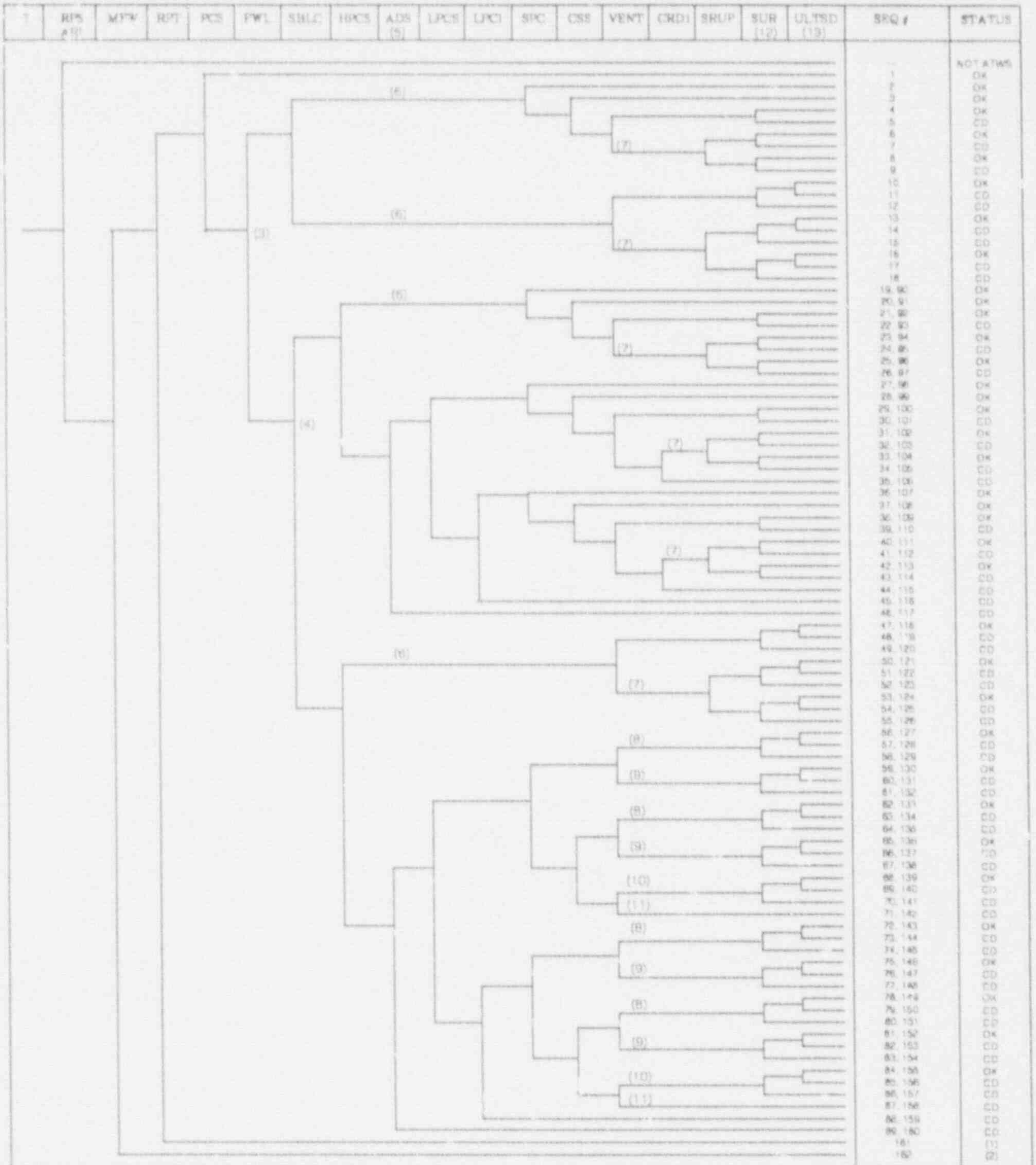

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LaSalle ATWS Event Tree

Figure 4.3 LaSalle ATWS Event Tree (Continued)

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3. MFW can not continue to run for more than about 8 minutes without depleting the main condenser unless the operator controls level.

4. Transfer sequences from (2).

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7. Containment pressure increases until containment failure occurs.

Figure 4.3 LaSalle ATWS Event Tree (Concluded)

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8. RHR (residual heat removal) 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 then oscillatory behavior results (100 psia, 321 F) from RPV pressure exceeding low pressure system shutoff heads, injection valves cycle (11 times/hr.).

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.

Table 4.5
Value Block Changes for ATWS Sequences

```
COMMENT$ 3-24-87 $
COMMENT$ THIS VALUE BLOCK HAS CHANGES FOR ATWS SEQUENCES INCOPPORATED$
COMMENT$ FOR FOLLOWING DATA ---$
  0.5     $ OPFAILSCDS-OE    $
  0.1     $ SLC0000X-Q00-OE  $
  .01     $ OPERR-INITSPC    $
  0.1     $ OPERR-INITCSS    $
  3.0E-02 $ SLOCA-IE         $
  3.0E-03 $ MLOCA-IE         $
  3.0E-04 $ LLOCA-IE         $
COMMENT$  OPFAILSCDS-OE WAS 5.0E-01     $
COMMENT$  SLC0000X-Q00-OE WAS 5.0E-01   $
COMMENT$  OPERR-INITSPC WAS 1.0E-02     $
COMMENT$  OPERR-INITCSS WAS 1.0E-01     $
COMMENT$  SLOCA-IE WAS 1.0E-01          $
COMMENT$  MLOCA-IE WAS 3.0E-03          $
COMMENT$  LLOCA-IE WAS 3.0E-04          $
COMMENT$ END OF ATWS 3-24-87 CHANGES $
```

SCREENING VALUES FOR POINT ESTIMATES FOR ATWS SEQUENCES

Point Estimate	Screening Value
FWL = OPFAILSMFW-8M	0.5
RPS	1.0E-05/yr.

4.7 References

1. D. W. Stack, "A SETS User's Manual for Accident Sequence Analysis," NUREG/CR-3547, SAND83-2238, Sandia National Laboratories, Albuquerque, NM, January 1984.

2. R. B. Worrell, "SETS Reference Manual," NUREG/CR-4213, SAND83-2675, Sandia National Laboratories, Albuquerque, NM, May 1985.

5.0 OPERATOR RECOVERY ACTIONS

At this point in the LaSalle PRA, we have identified a set of potential core damage accident sequences. These accident sequences consist of equipment failures (e.g., pump fails to start and run, valve fails closed, etc.) and human errors (e.g., maintenance, test, etc.) and their estimated probabilities of occurrence. If we stopped at this point, the PRA would not accurately reflect the possibility of potential core damage due to an accident sequence. To accurately reflect this possibility, we must include events in the cut sets which represent the ability of the plant operators and other support personnel to prevent or mitigate core damage during the accident situation. These events are called recovery actions.

A methodology for including recovery actions in the LaSalle PRA was developed and reported in Volume 1 of NUREG/CR-4834[1] and is explained in detail in Volume 2 of NUREG/CR-4834.[2] A summary of the methodology and its development follows.

In the methodology, a recovery action is defined as an action which must be accomplished by the operators (or others) to prevent or mitigate core damage during an accident. It consists of two distinct phases:

1. a diagnosis phase - recognizing that a problem exists with one of the critical parameters and deciding what to do about it, and

2. an action phase - physically accomplishing the action(s) decided upon in the diagnosis phase.

A new data-based model for estimating the contribution from the diagnosis phase for certain type recovery actions was developed after (1) examination of existing models indicated a heavy reliance upon judgement data and (2) results from statistical testing of observed operator behavior indicated a lack of correlation to the corresponding judgement data. This new data-based model for the diagnosis phase was developed using information obtained from simulator drills. These simulator drills were based on preliminary results from the LaSalle PRA. These preliminary results were used to define realistic plant-specific accident scenarios which could potentially lead to core damage. The drills were used to obtain time data on the operator team's ability to respond to the accident scenario. This time data, along with the grouping of operator actions based upon the underlying operational similarity of the actions, provides the basis for the model of the diagnosis phase of the recovery action. It was concluded that existing models for the action phase of the recovery action could be used.

The recovery methodology can be summarized as follows:

1. Appropriate recovery actions are identified. This includes both recovery actions which are to be placed directly on the event

trees or fau't trees and recovery actions which result from examination of the information contained in the cut sets.

2. For the recovery actions which are not included in the event trees or fault trees, a unique event representing the recovery action or set of recovery actions is defined and then added to the appropriate cut sets.

3. The recovery actions are modeled as consisting of a diagnosis phase and an action phase.

4. Estimates of the failure probabilities for each phase are provided using separate models (i.e., the diagnosis phase uses the data-based models developed from the simulator data and the action phase uses existing models).

5. Estimates for each phase are combined to produce a single non-recovery probability.

6. The effect is that the original cut sets' failure probabilities are multiplied by the non-recovery probability of the recovery action(s) to give new cut set failure probabilities. The new cut set failure probabilities now reflect the operators' contribution in reducing or mitigating core damage.

5.1 Application of the Recovery Methodology

As stated above, the recovery methodology used in the LaSalle PRA was developed in NUREG/CR-4834.[1] Figure 5.1, Figure 2.1-1 of Reference 2, provides a flow chart for the application of the recovery methodology. The following sections describe how the recovery methodology was implemented for the Lasalle accident sequences.

Before the sequence can be analyzed to determine whether the operator can intervene to restore failed equipment, the assumptions regarding types of operator recovery actions must be defined. We have included the following recovery considerations in the LaSalle Unit 2 analysis.

1. Failure Mechanism: The fault trees were developed to a level of detail that allows us to identify recoverable and non-recoverable faults. For example, "local faults" of a valve generally included a mechanical failure of the valve that precluded any operator recovery, either remote or local. "Control circuit faults", however, have recovery potential by the operator actions of identifying the problem and possible manual opening or closing of the valve. In general, extraordinary actions were not considered unless they were clearly indicated as being needed and sufficient time was available to perform them.

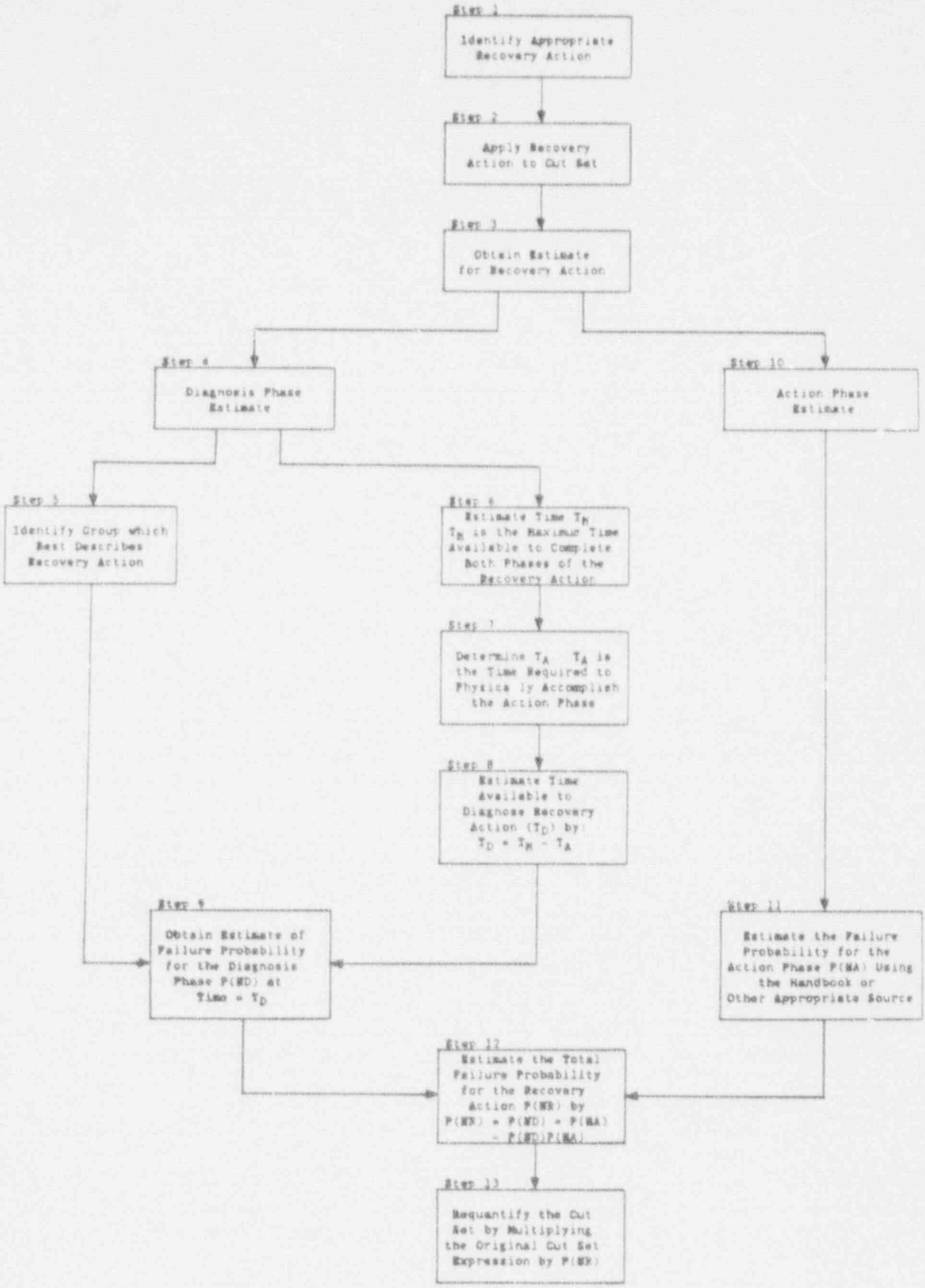

Figure 5.1
Recovery Methodology Flow Chart

2. Failure timing: This can be subdivided into two categories:

   a. The time of the failure with respect to the accident scenario (i.e., the time to the onset of core damage) determined, in part, the state of the operator and his ability to cope with the failure. To pick two extreme examples, much less credit would be given to a recovery action that had to occur within the first two minutes of an accident sequence than to the same type of action that must occur within the first eight hours of an accident sequence.

   b. The time to the "Point of No Return" for equipment damage is also a factor. Some failures are not immediately catastrophic. Many support system failures will cause a front-line system failure only after a period of hours has gone by. Thus, if the operator receives warning of a problem developing, he may have sufficient time to diagnose and correct the situation.

3. Failed Equipment Location: For operations outside of the control room, the operator must have definite indications of a problem with the system of interest and sufficient time to take corrective action. For most locations at LaSalle, an additional ten minutes over the control room time is sufficient for the operator to reach the location.

4. Number of Recovery Actions: Credit was not given for multiple recovery actions unless the actions were performed by a different set of individuals or were distinct enough or separated by a large enough time interval to be regarded as independent. An example of the first case is the recovery of offsite power which was considered as being performed independently of other onsite recovery actions. An example of the second case is recovery of injection in the initial phase of the accident and then recovery of containment heat removal in the many hours available until containment failure.

5.1.1 Identification of Possible Recovery Actions

It is recognized that some recovery actions were included in the event trees and the fault trees. The recovery actions included in the event trees were operator actions that were necessary to model certain accident sequences. The recovery actions included in the fault trees were generally high-level procedural actions. The recovery actions included in the event trees or fault trees are listed in Table 5.1. The remainder of this section deals with the recovery actions which were "ANDED" to the sequence cut sets resulting from various SETS runs.

Table 5.1
Recovery Actions from Event Trees and Fault Trees

1EDC2DEP-FROP-4
ADSMINIT-Q00-OE
CRD-REALIGN-OE
CRD1-REALIGN-OE
MFS-RESET-OE
MODESWTCH-C-OE-O
OPERFAIL-VENT-OE
OPFAILSCDS-OE
OPFAILS-REOPEN
TDRFP-T-OE-O
ADS-INHIBIT-OE
SLC0000X-Q00-OE
SLCC001B-Q00-OE
OPERR-INITCSS
OPERR-INITSPC
OPFAILSMFW-OE

One of the first tasks which must be accomplished to take credit for recovery actions is to identify the potential recovery actions. For LaSalle this was done by:

1. Identifying the basic event failures which were recoverable and

2. Examining the cut sets to determine if any other potential recovery actions existed.

Using the Variable Occurrence Table (VOT) from the SETS run for each sequence, the basic events were examined to determine if they were potentially recoverable. If a basic event was found to be potentially recoverable, it was identified as recoverable and was initially grouped depending upon what type of action was necessary to accomplish the recovery action. This included identifying whether the action could take place in the control room, locally within the plant, or some place else. Table 5.2 is a sample of a VOT from a SETS run with the basic events identified as either recoverable or non-recoverable, and if recoverable then recoverable from the control room, locally, or some place else.

The basic events which were identified as recoverable were grouped into categories depending on whether they were recoverable from the control room (RA-1 type actions), locally (RA-2 type actions), or some place else (e.g., RA-8 type actions). Tables 5.3 through 5.7 list the basic events which were grouped into a specific recovery action type (e.g., RA-1 type actions are listed in Table 5.3). In addition to the recovery actions resulting from examination of the VOT, other recovery actions were identified after examining the cut sets. Table 5.8 lists these recovery actions.

5.1.2 Application of Recovery Actions to Cut Sets

After the basic events were identified as recoverable or non-recoverable, they were prioritized to facilitate the application of the recovery actions to the cut sets. The order of priority was roughly in order of their probability with the easiest actions being taken credit for first (i.e., the lowest non-recovery probability action, that is, the highest recovery probability action was taken credit for first). If two actions were possible and they could be considered independent of each other, then credit would be given for both. It should be noted that if restoring offsite power (i.e., RA-8) was a potential recovery action, then taking credit for at least one more recovery action in the same cut set was always possible (see section 5.1).

After prioritizing the basic events, a global search through the computerized list of cut sets for each accident sequence was conducted to identify each occurrence of a basic event. If the cut set containing the basic event did not have a recovery action, then the recovery action associated with that basic event was "ANDED" to the cut set. If the cut set already contained a recovery action that had been identified by this

Table 5.2
Sample VOT with Recoverable Basic
Events Identified

EVENT NAME	IDENTIFIER	LOCATION
LCSC002A-P-UUM	NR	
LAK10RCA-ROO-LFO	RA-1	CR
LOSP-IE	RA-8	OTHER
T101-IE	NR	
CODG01F-PLG-LF	NR	
CODG01P-PMS-LF	NR	
CODG01P-PMS-CC	NR	
CODG01F-S-UUM	NR	
CODG01P-P-UUM	NR	
EE-CODG01F-PLG	NR	
CCBODG1P-BCO-LF	NR	
RLOSP	RA-8	OTHER
1EB235XA-BCO-LF	NR	
1EB235A-BCO-LF	NR	
AP037X3-ROO-LFO	RA-1	CR
DOVB101X-BCO-LF	NR	
HACTK3-ROO-LFO	RA-1	CR
CSCD300-PLG-LF	NR	
AP040X3-ROO-LFO	RA-1	CR
LAK9ARCA-ROO-LFO	RA-1	CR
LAK18ARA-ROO-LFO	RA-1	CR
LAK9BRCB-ROO-LFO	RA-1	CR
LAK18BRB-ROO-LFO	RA-1	CR
LAK3BRCB-RCO-LFO	RA-1	CR
HCSF004C-VCC-LF	NR	
HCSF004C-VCC-CS	RA-2	LOCAL
HF004CB-BCO-LF	RA-2	LOCAL
HACTCPF1-FUS-LF	RA-1	CR
HACTCPF2-FUS-LF	RA-1	CR
HACTK9-ROO-LFO	RA-1	CR
DGO-GEN-LF	RA-9	LOCAL
DGO-GEN-CC	RA-9	LOCAL
DG2A-GEN-LF	RA-9	LOCAL
DG2A-GEN-CC	RA-9	LOCAL

Table 5.3
Basic Events Which Were Categorized
as RA-1 Type Actions

1E4362S1-ROO-LFO	LAK10RCA-ROO-LFO
1E4362S2-ROO-LFO	LAK10ACP9-LFOO
1E4327SX-ROO-LFO	LAK18ARC-ROO-LFO
1E4327SY-ROO-LFO	LAK18ARA-ROO-LFO
1EB1Y1X-BCO	LAK18BRB-ROO-LFO
1EB2Y4X-BCO	LAK18ACP1-LFOO
213CB16-BCO	LAK18BCP1-LFOO
AP040X3-ROO-LFO	LAK21BRB-ROO-LFO
AP037X3-ROO-LFO	LAK23ARC-ROO-LFO
AP037X3CP7-LFOO	LAK23BPC-ROO-LFO
AP040X3CP7-LFOO	LAK25CBR-ROO-LFO
AP037X4CP1-LFCO	LAK28ACXCP4-LFCO
AP040X4CP1-LFCO	LAK28BCXCP4-LFCO
ADARK9B-ROO-LFO	LAK70ARA-ROO-LFO
ADARK10B-ROO-LFO	LAK70BRB-ROO-LFO
ADARK12B-ROO-LFO	LAK105Cb-ROO-LFO
ADARK35B-ROO-LFO	LAS44AX-QCO-LFO
ADARK38B-ROO-LFO	LAS44BX-QCO-LFO
ADARK39B-ROO-LFO	LCSK1ARC-RCO-LFO
ADSACT-RUM-16	LCSK12AR-ROO-LFO
ADSACT-RUM-18	LF5K14AR-ROO-LFO
ADSACT-IB-TE001	HACTK3-ROO-LFO
ADSACT-IB-TE002	HACTK9-ROO-LFO
ADSACT-RE001-B	HACTK3CP9-LFOO
ADSACT-RE002-B	HACTK9CP7-LFOO
LAK2ARCA-RCO-LFO	HACTCPF1-FUS-LF
LAK2BRCB-RCO-LFO	HACTCPF2-FUS-LF
LAK3ARCA-RCO-LFO	HF001CCB-BCO
LAK3BRCB-RCO-LFO	HC01K14-ROO-LFO
LAK3ARCXCP1-LFCO	RACTK3-ROO-LFO
LAK3BRCXCP1-LFCO	RACTK5-ROO-LFO
LAK9ARCA-ROO-LFO	RACTK17-ROO-LFO
LAK9BRCB-ROO-LFO	
LAK9ACP3-LFOO	
LAK9BCP3-LFOO	

Table 5.4
Basic Events Which Were Categorized
as RA-2 Type Actions

CSCF068A-VCC-CC	RHRF47AA-RUM-1
CSCF068B-VCC-CC	RHRF47BB-RUM-1
CCBF068A-BCO-LF	RHRF48AA-VOO-CC
CSCB068X-BCO-LF	RHRF48BB-VOO-CC
HACTK13CP5-LFCO	RHRH01AX-RUM-1
HACTK13CP3-LFCO	RHRH01BX-RUM-1
HCSF004C-VCC-CS	RHRF55AX-RUM-1
HCSF015C-VCC-CS	RHRF55BX-RUM-1
HCSF023C-VCO-CS	RHRF51AA-RUM-1
HF004CB-BCO-LF	RHRF51BB-RUM-1
HG015CB-BCO-LF	RHRF60AA-RUM-1
HF04CSC-QOC-LF	RHRF60BB-RUM-1
LAK14ARC-RCO-LFO	RHRF65AA-RUM-1
LAK14BRC-RCO-LFO	RHRF65BB-RUM-1
LAK93ARC-ROO-LFO	RHRF64AA-RUM-1
LAK93BRC-ROO-LFO	RHRF74AA-RUM-1
LAK93ACP3-LFCO	RHRF74BB-RUM-1
LAK93BCP3-LFCO	RHRF87AA-RUM-1
LAK105AA-ROO-LFO	RHRF87BB-RUM-1
LAK105BB-ROO-LFO	RHRF88AX-RUM-1
LF5K8AR-ROO-LFO	SCSF06AA-RUM-1
LAK10BB-ROO-LFO	SCSF06BB-RUM-1
LCSC002A-RUM-1	RHRB03AX-BCO-LF
RHRC003B-RUM-1	RHRB03BX-BCO-LF
RHRF64BB-RUM-1	

Table 5.5
Basic Events Which Were Categorized
as RA-8 Type Actions

LOSP-IE
RLOSP

Table 5.6
Basic Events Which Were Categorized
as RA-9 Type Actions

DG0-GEN-LF
DG0-GEN-CC
DG2A-GEN-LF
DG2A-GEN-CC
DG2B-GEN-LF
DG2B-GEN-CC

Table 5.7
Basic Events Which Were Categorized
as RA-15 Type Actions

DG-CM

Table 5.8
Recovery Actions Identified After Examining the Cut Sets

Action*	Description
RA-1	Manual operation of a system or component from the control room.
RA-2	Local operation of components.
RA-3	Open RCIC isolation valve(s) after RCIC room isolation.
RA-4	Isolate recirculation pump seal LOCA AND restore PCS.
RA-5V	Vent through alternate vent path.
RA-6	If one electric power train has failed, one-half of the time the recirculation pump LOCA will occur on the recirculation pump which can be isolated. Isolate recirculation pump seal LOCA AND restore PCS.
RA-7	Open a manual valve that is closed due to unscheduled maintenance.
RA-8	Recover off-site power.
RA-9	Recover DG after loss of off-site power and failure of DG.
RA-10	Replace a fuse in the control room.
RA-11	Manually close SBLC valves after the occurrence of an ATWS, given failure to close the valves following a previous test on the SBLC system.
RA-12	Locally close RWCU valve after the occurrence of an ATWS.
RA-15	Repair of DG common mode failure.
RA-16	Manual start of a DG from the control room and then manual start of an SBLC pump after the occurrence of an ATWS.
RA-CDS	Use condensate system.
RA-DDFP	Use diesel driven firewater pump.

* RA-13 and RA-14 not used.

process, then no additional recovery action was added unless the actions could be considered independent. This process was continued until all the recoverable basic events were examined.

5.1.3 Obtain Estimate for Recovery Action

After the recovery action identifier was applied to the cut sets, we obtained estimates for the failure probabilities of the recovery actions. Since the recovery actions were modeled as consisting of two phases (i.e., diagnosis phase and action phase), each phase was estimated using appropriate models. The following sections discuss how estimates for each phase were obtained.

5.1.3.1 Diagnosis Phase Estimate

Two tasks had to be accomplished before the diagnosis phase failure probability of a recovery action could be estimated. First, where possible, we identified the group which best described the recovery action of interest by searching Table 5.9 (Table 2.1.5-1 in Reference 2). Second, we estimated the time available for the operators to diagnose the recovery action. These tasks are described below.

5.1.3.1.1 Identification of Group Which Best Describes Recovery Action

To identify the group that best described the recovery action, we examined the actions in each group in Table 5.9 and chose the group that contained actions that were most similar to the recovery action of interest or for which the group description was judged to be the best match. If the recovery action could not be described by one of the groups in Table 5.9 or if specific data existed for the recovery action, then other models were used to provide estimates for the recovery action.

For example, basic event AP040X3-ROO-LFO from Table 5.2 represents a failure in the automatic operation of diesel generator "2A". Searching Table 5.9 for the actions which are most similar or the description which best describes the recovery action, we found that the action was best described by Group 3: Manual operation of systems or components which failed to automatically actuate (operate). This process was repeated for each basic event in Tables 5.3 and 5.4. In addition, the recovery actions listed in Tables 5.1 and 5.8 were examined to determine if they could be described by the groups in Table 5.9. Table 5.10 list the recovery actions identified by this process.

Table 5.9
Summary of Ten Groups of Crew Recovery Actions*

Group**	Description	of Recovery Actions
1	Manual operation of system or component to control a critical parameter prior to the automatic actuation (if it has automatic actuation) of the system or component.	1. Drill 1 -- Initiate RHR after ATWS. 2. Drill 2 & 2B -- Initiate SP cooling after RX Trip. 3. Drill 3 -- Initiate RCIC after station blackout. 4. Drill 4 -- Initiate SP cooling after DG1A loads. 5. Drill 6 -- Close MSIVs after Level 7 alarm. 6. Drill 6 -- Close FW valve 1A after Level 7 alarm. 7. Drill 6 -- Initiate SP cooling after RX trip. 8. Drill 6 -- Initiate FP cooling after RX trip.
2	Use of low pressure systems when high pressure systems are unavailable.	1. Drill 8 -- Depressurize after RCIC failure. 2. Drill 8 -- Inject LP after RCIC failure.
3	Manual operation of systems or components which failed to automatically actuate (operate).	1. Drill 3 -- Send B-man to open F013 after F013 failure. 2. Drill 4 -- Reset RCIC isolation after DG 1A loads. 3. Drill 8 -- Request RCIC investigation after RCIC failure.
4	Restoration of safety-related in-house electrical buses or supply equipment.	1. Drill 3 -- Request DG 0 repair after station blackout. 2. Drill 3 -- Request DG 1B repair after station blackout. 3. Drill 3 -- Request DG 1A repair after station blackout. 4. Drill 4 -- Request DG 1B repair after SAT failure. 5. Drill 4 -- Recover DG 1A after DG 1A trouble. 6. Drill 6 -- Request DG A investigation after DC A failure.
5	Restoration of off-site-supplied nonsafety-related electrical buses or supply equipment.	1. Drill 3 -- Request X-tie after station blackout. 2. Drill 3 -- Request SAT repair after station blackout. 3. Drill 4 -- Request SAT repair after SAT failure. 4. Drill 4 -- Request X-tie after SAT failure. 5. Drill 6 -- Restore Bus 151 locally after RX trip.
6	Manual backup of an automatic shutdown function.	1. All Drills -- Mode switch after RX trip. 2. All Drills -- Manual scram after RX trip.
8	Manual override of a system that automatically functions when automatic operation of the system would challenge a critical parameter.	1. Drill 1 -- Jumper VP after drywell isolation. 2. Drill 4 -- Restore VP after drywell isolation. 3. Drill 6 -- Restore VP after DC A failure. 4. Drill 8 -- Restore VP after drywell isolation.
10	Request to use last line of (GARBAGE)*** systems for level control.	1. Drill 4 -- Depressurization after station blackout. 2. Drill 4 -- Request diesel fire pump after station blackout.
11	Local operation of manually controlled components normally operated from the control room when control-room operation fails.	1. Drill 2 & 2B -- Send B-man to close SDV valves after scram reset attempt. 2. Drill 6 -- Request air restoration after service air pressure low alarm.
12	Manual override of a false control signal when no direct indication exists that the control signal is false or erroneous.	1. Drill 4 -- Request bypass of RCIC isolation after RCI(isolation because of room overheating.

*The items listed in this table refer to the correct diagnosis of the required action.
**See corresponding table (Tables 2.1.9-1 through 2.1.9-10) for information to be used in estimating.
***GARBAGE systems are those systems which are used only as a last resort to prevent core damage. These systems inject "dirty" (nonreactor grade) water into the vessel and are used only if no other means of injecting water into the vessel are available.

Table 5.10
Description of Recovery Actions Based Upon
Examination of Group Descriptions
in Table 5.9

Action*	Group	Description	Identifier
RA-1	1	Manual operation of a system or component from the control room that has no automatic actuation or prior to its automatic operation if it has automatic actuation.	RA-1-1
RA-1	3	Manual operation of a system or component from the control room which failed to automatically actuate.	RA-1-3
RA-2	11	Local operation of manually controlled components normally operated from the control room when control-room operation fails.	RA-2-11
RA-2	3	Local operation of a system or component which failed to automatically actuate.	RA-2-3
RA-3	12	Open RCIC isolation valve(s) given occurrence of RCIC room isolation.	RA-3-12
RA-5V	1	Vent through alternate vent path.	RA-5V-1
RA-7	1	Locally open a manual valve closed due to unscheduled maintenance of a pump. Restores heat removal.	RA-7-1
RA-7	3	Locally open a manual valve closed due to unscheduled maintenance of a pump. Restores injection.	RA-7-3
RA-10	1	Replace a fuse in the control room in a system or component that has no automatic operation or prior to its automatic operation if it has automatic operation.	RA-10-1
RA-ATWS-11	11	Local operation of manually controlled SBLC valves normally operated from the control room when control-room operation fails.	RA-ATWS-11-11
RA-ATWS-16	3,1	Manual start of a DG from the control room and then manual start of SBLC pump.	RA-ATWS-16-31

Table 5.10 (Concluded)
Description of Recovery Actions Based Upon
Examination of Group Descriptions
in Table 5.9

Action*	Group	Description	Identifier
RA-ATWS-1	3	Manual operation of a system or component from the control room which failed to automatically actuate after the occurrence of an ATWS.	RA-ATWS-1-3
RA-ATWS-2	3	Local operation of a system or component which failed to automatically actuate after the occurrence of an ATWS.	RA-ATWS-2-3
RA-ATWS-12	3	Locally close the RWCU valve after the occurrence of an ATWS.	RA-ATWS-12-3
RA-CDS	2	Injection of water into the vessel via the condensate system.	RA-CDS
RA-DDFW	10	Injection of water into the vessel via the diesel driven firewater pump.	RA-DDFW

* RA-4, RA-6, RA-8, RA-9, and RA-15 are data based and have no group association.

5.1.3.1.2 Estimating Time T_M

In order for us to be able to estimate the amount of time available for the diagnosis phase of the recovery action, the maximum time available to the operators must be estimated. This maximum time, T_M, is the time during which both phases of the recovery action (i.e., diagnosis phase and action phase) must be completed to ensure the prevention or mitigation of the undesirable outcome. T_M was estimated using thermal-hydraulic computer codes to provide information on core or containment parameters (e.g., pressure, temperature, water level, etc.). Table 5.11 list the estimates of T_M that resulted from the thermal-hydraulic calculations. In Volume 4 of this report, the results of the calculations are discussed in more detail.

5.1.3.1.3 Determination of T_A

After estimates of T_M were obtained, the amount of time required to physically accomplish the action(s) decided upon during the diagnosis phase was determined. This time, T_A, was estimated as the maximum amount of time required by the operator(s) to reach the area where the action takes place plus the time required to accomplish the action(s). The time required to accomplish different classes of actions is presented in Table 5.12.

5.1.3.1.4 Estimate Time Available to Diagnose the Recovery Action, T_D

The following expression was used to estimate the time available to diagnose the recovery action:

$T_D = T_M - T_A$, where

T_M is the maximum time in which both phases of the recovery action must be completed to prevent or mitigate an undesirable outcome during the accident, and

T_A is the time required to physically accomplish the action(s) decided upon in the diagnosis phase.

Table 5.13 list the possible diagnosis times for the LaSalle sequences.

5.1.3.1.5 Estimate Failure Probability for Diagnosis Phase P(ND) at T_D

Given that the group which best describes a recovery action has been identified (Section 5.1.3.1.1) and the amount of time available to diagnose the recovery action has been estimated (Section 5.1.3.1.4), the failure probability for the diagnosis phase of the recovery action was determined by:

Table 5.11
Estimates for T_M Resulting From
Thermal-Hydraulic Calculations

Sequence	T_M	Notes
T12	27 hours	1, 6
	2 hours	2, 6
T24	27 hours	1, 6
	6 hours	2, 6
T30	27 hours	1, 6
	6 hours	2, 6
T40	27 hours	1, 6
	6 hours	2, 6
T50	23 hours	1, 6
	2 hours	2, 6
T59	8 hours	3, 7, 8
	10 hours	4, 7, 8
T64	27 hours	1, 6
	6 hours	2, 6
T76	27 hours	1, 6
	6 hours	2, 6
T88	23 hours	1, 6
	2 hours	2, 6
T97	1 hour	5, 9
T98	80 minutes	5, 9
TL16	27 hours	1, 6
	2 hours	2, 6
TL30	27 hours	1, 6
	6 hours	2, 6
TL36	27 hours	1, 6
	6 hours	2, 6
TL45	27 hours	1, 6
	6 hours	2, 6
TL54	23 hours	1, 6
	2 hours	2, 6
TL62	8 hours	3, 7, 8
	10 hours	4, 7, 8
TL68	27 hours	1, 6
	6 hours	2, 6
TL80	27 hours	1, 6
	6 hours	2, 6
TL92	23 hours	1, 6
	6 hours	2, 6
TL100	48 minutes	5, 9, 10
TL101	54 minutes	5, 9, 10
L16	15 hours	1, 11
	4 hours	2, 11

Table 5.11 (Concluded)
Estimates for T_M Resulting From
Thermal-Hydraulic Calculations

Sequence	T_M	Notes
L30	15 hours	1, 11
	4 hours	2, 11

NOTES:
1 - Amount of time to restore containment heat removal or begin injection of water into the vessel.
2 - Amount of time to begin venting the containment.
3 - Amount of time to begin injection of water into the vessel when no AC power is available.
4 - Amount of time to begin injection of water into the vessel when AC power is initially available.
5 - Amount of time to restore injection of water into the vessel.
6 - LTAS calculation (long-term loss of CHR, high pressure injection available)
7 - LTAS calculation (long-term loss of CHR, RCIC only)
8 - LTAS calculation (long-term loss of CHR, low pressure injection only)
9 - RELAP calculation.
10 - LTAS calculation (small break, steam)
11 - LTAS calculation (small break, liquid)

Table 5.12
T_A for Various Classes of Actions

T_A	Description of Action
2 minutes	Start or stop a system or component from the control room.
2 minutes	Change the state of an operated valve from the control room.
15 minutes*	Locally (i.e., away from the control room) start or stop a system or component.
15 minutes*	Locally change the state of an operated valve given that control room operation of the valve is impossible.
15 minutes*	Locally change the state of a manual valve.
15 minutes	Use the condensate system.
1 hour	Use the diesel driven firewater pump.

* The 15 minutes includes: (1) 10 minutes of travel time and (2) 5 minutes to physically accomplish whatever action is required. The 10 minute travel time is based on a plant walk through by people who were not familiar with the plant layout and as such is considered to be a conservative estimate of the amount of time the operators need to travel from point to point within LaSalle.

Table 5.13
Potential Diagnosis Times

T_M	T_A	T_D
27 hrs	2 min	26 hrs 58 min
	15 min	26 hrs 45 min
23 hrs	2 min	22 hrs 58 min
	15 min	22 hrs 45 min
15 hrs	2 min	14 hrs 58 min
	15 min	14 hrs 45 min
10 hrs	2 min	9 hrs 58 min
	15 min	9 hrs 45 min
8 hrs	2 min	7 hrs 58 min
	15 min	7 hrs 45 min
6 hrs	2 min	5 hrs 58 min
	15 min	5 hrs 45 min
4 hrs	2 min	3 hrs 58 min
	15 min	3 hrs 58 min
2 hrs	2 min	1 hr 58 min
	15 min	1 hr 45 min
	1 hr	1 hr
80 min	2 min	78 min
	15 min	65 min
1 hr	2 min	58 min
	15 min	45 min
54 min	2 min	52 min
	15 min	39 min
48 min	2 min	46 min
	15 min	33 min

(1) identifying the table from Tables 2.1.9-1 through 2.1.9-10 of Reference 2 that corresponds to the group identified in Section 5.1.3.1.1, and

(2) by following the procedures recommended by Reference 2 for using the information contained within the table to obtain an estimate of the diagnosis failure probability for a particular recovery action.

The procedures from Reference 2 are summarized as follows:

(1) In the probability of failure column of the table identified above, select the median value of the failure probability ($P(ND)_{median}$) that corresponds to the amount of time available to diagnose the recovery action. If the amount of time available to diagnose the recovery action is greater than the last time specified in the table, then use the probability of failure value that corresponds to the last time in the table.

(2) Calculate the error factor (EF) associated with the probability of failure value identified in step (1). This is accomplished by dividing the corresponding value of the upper 95% confidence limit by the probability of failure value. If this calculated error factor is greater than 10.0, a value of 10.0 is assumed for the error factor.

(3) Calculate the mean value for the diagnosis failure probability ($P(ND)_{mean}$) at time T_D using the EF from step (2) and the median value from (1) by the following formula which assumes that the distribution at a certain time is log-normal:

$$P(ND)_{mean} = (P(ND)_{median})(\exp([\ln EF/1.645]^2/2))$$

5.1.3.2 Estimate the Failure Probability for the Action Phase, P(NA)

Estimates for the failure probability for the action phase, P(NA), can be computed from any number of different sources. For application to RMIEP, the models and information summarized in Chapters 5 and 20 of Reference 3 (also referred to as the Handbook) were used.

5.1.3.3 Estimate the Total Failure Probability for a Recovery Action, P(NR)

After estimates for the diagnosis phase failure probability (P(ND)) and the action phase failure probability (P(NA)) were obtained, we calculated the total failure probability for the recovery action, P(NR). The failure probability for the recovery action is calculated as the probability of either failing to diagnose the appropriate action or failing to perform the recovery action. P(NR) is calculated using the following expression:

$$P(NR) = P(ND) + P(NA) - P(ND)P(NA)$$

where P(NR) is the failure probability for the recovery action, P(ND) is the failure probability for diagnosing the required action within time T_D, and

P(NA) is the failure probability for physically accomplishing the action within the time T_A.

5.2 Sample Calculation

As an example of how an estimate of the failure probability for a recovery action was made, consider the basic event LAK93ARC-ROO-LFO. The event represents the failure of a normally open motor operated valve to close and to remain closed given that it is demanded closed. The valve in question is normally controlled manually from the control room. The failure probability is estimated as follows:

1. Table 5.9 is searched for the group which best describes the recovery action, in this case group 11.

2. From thermal-hydraulic calculations, it has been determined, for the sequence of interest, that the maximum amount of time available to the operators is 27 hours i.e., $T_M = 27$ hours.

3. From considering the physical actions required to accomplish the recovery action, it is estimated that 15 minutes will be required to accomplish the action i.e., $T_A = 15$ minutes. This 15 minutes includes 10 minutes of travel time and 5 minutes of time to physically close the valve.

4. Given the information in (2) and (3), $T_D = 26$ hours and 45 minutes.

5. Table 2.1.9-9 from Reference 2 (reproduced here as Table 5.14) corresponds to group 11 as identified in (1).

6. Since T_D is larger than the last occurring value of time in Table 5.14, the last value in the probability of failure column is used. Thus, $P(ND)_{median} = 0.00060$.

7. The EF associated with this value of $P(ND)_{median}$ is 10.0 since dividing the corresponding upper 95% confidence limit by the median failure probability results in a value greater than 10.0.

Table 5.14
Group 11, Parameter Estimates from Fit of Lognormal Function
(N = 15, Mean = .85, Standard Deviation = .50)

Time (min.)	Standard Deviation of Point	Probability of Failure	Upper 95% Confidence Limit	Lower 95% Confidence Limit
1	.039	.96	.99	.78
2	.072	.87	.96	.66
3	.088	.77	.90	.56
4	.096	.69	.85	.48
5	.10	.62	.79	.41
6	.10	.56	.74	.36
7	.10	.51	.70	.31
8	.11	.46	.66	.27
9	.11	.42	.63	.24
10	.10	.39	.60	.21
11	.10	.35	.57	.18
12	.10	.33	.55	.16
13	.10	.30	.53	.14
14	.10	.28	.51	.13
15	.098	.26	.49	.11
16	.096	.24	.47	.10
17	.094	.23	.46	.092
18	.092	.21	.44	.083
19	.090	.20	.43	.075
20	.088	.19	.42	.068
21	.086	.18	.41	.062
22	.084	.16	.40	.056
23	.082	.16	.39	.051
24	.080	.15	.38	.047
25	.079	.14	.37	.043
26	.077	.13	.36	.039
27	.075	.12	.35	.036
28	.073	.12	.35	.033
29*	.071	.11	.34	.030
30	.069	.11	.33	.028
31	.068	.10	.33	.026
32	.066	.097	.32	.024
33	.064	.092	.31	.022
34	.063	.088	.31	.020
35	.061	.084	.30	.019
36	.060	.081	.30	.018
37	.058	.077	.29	.016
38	.057	.074	.29	.015
39	.056	.071	.29	.014
40	.054	.068	.28	.013
41	.053	.065	.28	.012

*Extrapolated beyond time = 28.9 min.

Table 5.14 (Concluded)
Group 11, Parameter Estimates from Fit of Lognormal Function
(N = 15, Mean = .85, Standard Deviation =.50)

Time (min.)	Standard Deviation of Point	Probability of Failure	Upper 95% Confidence Limit	Lower 95% Con[illegible]lence Limit
42	.052	.062	.27	.012
43	.051	.060	.27	.011
44	.049	.058	.27	.010
45	.048	.055	.26	.0096
46	.047	.053	.26	.0090
47	.046	.051	.26	.0084
48	.045	.049	.25	.0079
49	.044	.048	.25	.0074
50	.043	.046	.25	.0070
51	.042	.044	.24	.0066
52	.041	.043	.24	.0062
53	.040	.041	.24	.0059
54	.039	.040	.24	.0055
55	.038	.038	.23	.0052
56	.038	.037	.23	.0049
57	.037	.036	.23	.0047
58	.036	.035	.23	.0044
59	.035	.034	.22	.0042
60	.034	.033	.22	.0040
61	.034	.032	.22	.0038
62	.033	.031	.22	.0036
63	.032	.030	.22	.0034
64	.032	.029	.21	.0032
65	.031	.028	.21	.0030
66	.030	.027	.21	.0029
67	.030	.026	.21	.0028
68	.029	.025	.21	.0026
69	.028	.025	.20	.0025
70	.028	.024	.20	.0024
80	.023	.018	.19	.0015
90	.019	.014	.17	.00096
100	.016	.011	.16	.00064
110	.014	.0089	.15	.00044
120	.012	.0072	.15	.00031
180	.0051	.0026	.11	.00005
240	.0026	.0012	.093	.00001
300*	.0015	.00060	.079	.00000

*For times greater than 300 min., use last line of table.

(8) The mean value for the diagnosis failure probability is then calculated. This results in:

$$P(ND)_{mean} = (P(ND)_{median})(\exp((\ln EF/1.645)^2/2))$$

$$= (0.00060)\exp((\ln 10.0/1.645)^2/2))$$

$$= 1.6E\text{-}3$$

9) The action phase of the recovery action is a series of physical actions carried out by the personnel at the plant. For this example, the control room operators would direct someone (e.g., a B-man) to manually close the valve and would monitor control room instrumentation for indications as to the success of the requested action. To estimate the action phase failure probability, a HRA event tree is constructed (see Chapter 5 of Reference 3). The HRA event tree constructed for this example is shown in Figure 5.2. This HRA event tree, in conjunction with the human error probabilities (HEPs) given in Chapter 20 of Reference 3, provide a means of estimating the action phase of the recovery action.

From the HRA event tree, the probability of failing to accomplish the action phase is found by:

$$P(NA) = F_1 + F_2 + F_3 + F_4$$

$$F_1 = 0$$

$$F_2 = (0.001)(1.25)^*(0.003)(1.25)^* = 4.69E\text{-}6$$

$$F_3 = (0.001)(1.25)^*(0.003)(1.25)^* = 4.69E\text{-}6$$

$$F_4 = (0.001)(1.25)^*(0.003)(1.25)^* = 4.69E\text{-}6$$

(NOTE: *1.25 is the multiplier used to convert a median value with EF=3 to a mean value assuming a log-normal distribution.)

$$P(NA) = 0 + 4.69E\text{-}6 + 4.69E\text{-}6 + 4.69E\text{-}6$$

$$= 1.4E\text{-}5$$

(10) With both P(ND) and P(NA) having been determined, the total failure probability for the recovery action is found by:

$$P(NR) = P(ND) + P(NA) - P(ND)P(NA)$$

$$= (1.6E\text{-}3) + (1.4E\text{-}5) - (1.6E\text{-}3)(1.4E\text{-}5)$$

$$= (1.614E\text{-}3) - (2.24E\text{-}8)$$

EP or HEP	Event	Value for EP or HEP	Source*
Δ	Mechanical or physical failure prohibits operator from getting message to B-man	ε	--
A	Error in message from operator	.001 (EF = 3)	Table 20-8 Item (1a)
B	Operator fails to monitor feedback (recovery action)	.003 (EF = 3)	Page 20-13
C	B-man misunderstands message	.001 (EF = 3)	Table 20-8 Item (1a)
D	Operator fails to monitor feedback (recovery action)	.003 (EF = 3)	Page 20-13
E	B-man selects incorrect valve	.001 (EF =3)	Table 20-13 Item (5)
G	Operator fails to monitor feedback (recovery action)	.003 (EF = 3)	Page 20-13

*All values are from the Handbook, except the value for Δ. The value for Δ is based on engineering judgment.

Figure 5.2
HRA Event Tree for Example Application

$= 1.614E\text{-}3$

$= 1.6E\text{-}3$

5.3 Recovery Actions for LaSalle

Following the procedures described above, failure probabilities for the recovery actions in Tables 5.1 and 5.10 were calculated. To determine the final values for use in the PRA, one last factor needed to be considered. In determining the final value for recovery actions, one needs to consider the random failure probability of the equipment to be used in the recovery process. While control circuit failure on a valve can be bypassed by locally manually opening the valve, there is some probability that the valve itself may be locally failed. This fact puts a lower limit on the effectiveness of the operator. The non-recovery failure probability used in the PRA to quantify the cut sets can not have a failure probability less than the corresponding failure probability of the equipment. For purposes of this PRA, the non-recovery probability was not allowed to be below 1.0E-03 which is roughly the failure probability of the types of equipment modeled in the PRA and used for the recovery actions. It was assessed that a more exact model which evaluated a separate random failure for each type of equipment and recovery action was unwarranted. The results of these calculations are presented in Table 5.15.

5.4 References

1. L. M. Weston, D. W. Whitehead, and N. L. Graves, "Recovery Actions in PRA for the Risk Methods Integration and Evaluation Program (RMIEP), Volume 1: Data Based Method," NUREG/CR-4834/1 of 2, SAND87-0179, Sandia National Laboratories, Albuquerque, NM, June 1987.

2. D. W. Whitehead, "Recovery Actions in PRA for the Risk Methods Integration and Evaluation Program (RMIEP), Volume 2: Application of the Data Based Method," NUREG/CR-4834/2 of 2, SAND87-0179, Sandia National Laboratories, Albuquerque, NM, December 1987.

3. A. D. Swain and H. E. Guttmann, "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications, Final Report," NUREG/CR-1278, SAND80-0200, Sandia National Laboratories, Albuquerque, NM, August 1983.

Table 5.15
Recovery Actions in LaSalle PRA

Event Name	Definition	Value	Source
1EDC2DEP-FROP-4	Failure to restore offsite power in 4 hours.	1.0	1
1EDC2DEP-FRP-15H	Failure to restore offsite power in 15 hours.	2.0E-2	1
1EDC2DEP-FRP-27H	Failure to restore offsite power in 27 hours.	2.0E-2	1
ADS-INHIBIT-12M	Operators inhibit ADS in 12 minutes.	2.0E-1	2
ADS-SEI-OE-54M	During a seismic induced accident operators fail to ADS in 54 minutes.	2.2E-3	2
ADS-SEI-OE-80M	During a seismic induced accident operators fail to ADS in 80 minutes.	2.2E-3	2
CRD-REALIGN-OE	Operators fail to realign the CRD system (two pumps available) in 8 hours.	2.1E-3	2
CRD1-REALIGN-OE	Operators fail to realign the CRD system (one pump available) in 12 hours.	2.1E-3	2
MFS-RESET-25M	Operators fail to reset main feed-water trip in 25 minutes.	4.4E-3	2
MFS-RESET-69M	Operators fail to reset main feed-water trip in 69 minutes.	2.1E-3	2
MFS-RESET-95M	Operators fail to reset main feed-water trip in 95 minutes.	2.1E-3	2
MFS-RESET-OE-27H	Operators fail to reset main feed-water trip in 27 hours.	2.1E-3	2
MODESWTCH-OE-69M	Operators fail to change mode switch from run to shutdown in 69 minutes.	1.2E-3	2
MODESWTCH-OE-95M	Operators fail to change mode switch from run to shutdown in 95 minutes.	1.2E-3	2
OP-F-INITCSS-25M	Operators fail to initiate containment spray system in 25 minutes.	4.4E-3	2
OP-F-INITCSS-30M	Operators fail to initiate containment spray system in 30 minutes.	2.7E-3	2
OP-F-INITCSS-56M	Operators fail to initiate containment spray system in 56 minutes.	2.1E-3	2
OP-F-INITCSS-59M	Operators fail to initiate containment spray system in 59 minutes.	2.1E-3	2
OP-F-INITCSS-85M	Operators fail to initiate containment spray system in 85 minutes.	2.1E-3	2
OP-F-INITSPC-85M	Operators fail to initiate suppression pool cooling in 85 minutes.	2.1E-3	2
OP-F-REOPN-FTR	Operators fail to reopen RCIC F063 valve.	4.3E-1	2

Table 5.15 (Continued)
Recovery Actions in LaSalle PRA

Event Name	Definition	Value	Source
OPFAIL-ADS-54M	Operators fail to use automatic depressurization system in 54 minutes.	2.2E-3	2
OPFAIL-ADS-80M	Operators fail to use automatic depressurization system in 80 minutes.	2.2E-3	2
OPFAIL-REOPN-10H	Operators fail to reopen RCIC F063 valve in 10 hours.	2.5E-3	2
OPFAIL-REOPN-1H	Operators fail to reopen RCIC F063 valve in 1 hour.	2.5E-3	2
OPFAIL-REOPN-20M	Operators fail to reopen RCIC F063 valve in 20 minutes.	3.5E-1	2
OPFAIL-SLC0X-30M	Operators fail to start Standby Liquid Control System in 30 minutes.	2.7E-3	2
OPFAIL-SLC0X-56M	Operators fail to start Standby Liquid Control System in 56 minutes.	2.1E-3	2
OPFAIL-SLC0X-59M	Operators fail to start Standby Liquid Control System in 59 minutes.	2.1E-3	2
OPFAIL-SLC0X-85M	Operators fail to start Standby Liquid Control System in 85 minutes.	2.1E-3	2
OPFAIL-SLC1B-30M	Operators fail to start second standby liquid control pump in 30 minutes given that the first pump failed to start.	3.0E-3	2
OPFAIL-SLC1B-56M	Operators fail to start second standby liquid control pump in 56 minutes given that the first pump failed to start.	2.1E-3	2
OPFAIL-SLC1B-59M	Operators fail to start second standby liquid control pump in 59 minutes given that the first pump failed to start.	2.1E-3	2
OPFAIL-SLC1B-85M	Operators fail to start second standby liquid control pump in 85 minutes given that the first pump failed to start.	2.1E-3	2
OPFAIL-VENT-0H	Operators fail to vent in zero hours.	1.0	2
OPFAIL-VENT-20M	Operators fail to vent in 20 minutes.	1.0	2
OPFAIL-VENT-2H	Operators fail to vent in 2 hours.	2.1E-3	2
OPFAIL-VENT-4H	Operators fail to vent in 4 hours.	2.1E-3	2
OPFAIL-VENT-6H	Operators fail to vent in 6 hours.	2.1E-3	2
OPFAILS-REOPEN	Operators fail to reopen RCIC F063 valve.	1.0	2
OPFAILSCDS-OE-8M	Operators fail to control condensate system in 8 minutes.	3.4E-1	2

Table 5.15 (Continued)
Recovery Actions in LaSalle PRA

Event Name	Definition	Value	Source
OPFAILSMFW-8M	Operators fail to control main feed-water system in 8 minutes.	5.0E-1	2
OPS-F-VENT-6H	Operators fail to vent in 6 hours	1.0	2
RA-1-1-10H	Manual operation within 10 hours of a system or component from the control room that has no automatic actuation or prior to its automatic operation if it has automatic actuation.	2.1E-3	2
RA-1-1-15H	Manual operation within 15 hours of a system or component from the control room that has no automatic actuation or prior to its automatic operation if it has automatic actuation.	2.1E-3	2
RA-1-1-23H	Manual operation within 23 hours of a system or component from the control room that has no automatic actuation or prior to its automatic operation if it has automatic actuation.	2.1E-3	2
RA-1-1-27H	Manual operation within 27 hours of a system or component from the control room that has no automatic actuation or prior to its automatic operation if it has automatic actuation.	2.1E-3	2
RA-1-1-8H	Manual operation within 8 hours of a system or component from the control room that has no automatic actuation or prior to its automatic operation if it has automatic actuation.	2.1E-3	2
RA-1-3-10H	Manual operation within 10 hours of a system or component from the control room which failed to automatically actuate.	2.6E-3	2
RA-1-3-13H	Manual operation within 13 hours of a system or component from the control room which failed to automatically actuate.	2.6E-3	2
RA-1-3-1H	Manual operation within 1 hour of a system or component from the control room which failed to automatically actuate.	3.2E-3	2

Table 5.15 (Continued)
Recovery Actions in LaSalle PRA

Event Name	Definition	Value	Source
RA-1-3-15H	Manual operation within 15 hours of a system or component from the control room which failed to automatically actuate.	2.6E-3	2
RA-1-3-23H	Manual operation within 23 hours of a system or component from the control room which failed to automatically actuate.	2.6E-3	2
RA-1-3-27H	Manual operation within 27 hours of a system or component from the control room which failed to automatically actuate.	2.6E-3	2
RA-1-3-48M	Manual operation within 48 minutes of a system or compon nt from the control ro m which failed to automatically actuate.	6.4E-3	2
RA-1-3-54M	Manual operation within 54 minutes of a system or component from the control room which failed to automatically actuate.	4.5E-3	2
RA-1-3-80M	Manual operation within 80 minutes of a system or component from the control room which failed to automatically actuate.	2.6E-3	2
RA-1-3-8H	Manual operation within 8 hours of a system or component from the control room which failed to automatically actuate.	2.6E-3	2
RA-15-10H	Repair of DG common mode failure within 10 hours.	3.8E-1	3
RA-15-1H	Repair of DG common mode failure within 1 hour.	9.1E-1	3
RA-15-23H	Repair of DG common mode failure within 23 hours.	1.2E-1	3
RA-15-27H	Repair of DG common mode failure within 27 hours.	1.0E-1	3
RA-15-48M	Repair of DG common mode failure within 48 minutes.	9.5E-1	3
RA-15-8H	Repair of DG common mode failure within 8 hours.	4.5E-1	3
RA-2-11-15H	Local operation within 15 hours of manually controlled components normally operated from the control room when control-room operation fails.	1.6E-3	2

Table 5.15 (Continued)
Recovery Actions in LaSalle PRA

Event Name	Definition	Value	Source
RA-2-3-10H	Local operation within 10 hours of a system or component which failed to automatically actuate.	2.6E-3	2
RA-2-11-23H	Local operation within 23 hours of manually controlled components normally operated from the control room when control-room operation fails.	1.6E-3	2
RA-2-11-27H	Local operation within 27 hours of manually controlled components normally operated from the control room when control-room operation fails.	1.6E-3	2
RA-2-3-1H	Local operation within 1 hour of a system or component which failed to automatically actuate.	6.9E-3	2
RA-2-3-27H	Local operation within 27 hours of a system or component which failed to automatically actuate.	2.6E-3	2
RA-2-3-48M	Local operation within 48 minutes of a system or component which failed to automatically actuate.	1.6E-2	2
RA-2-3-54M	Local operation within 54 minutes of a system or component which failed to automatically actuate.	1.0E-2	2
RA-2-3-8H	Local operation within 8 hours of a system or component which failed to automatically actuate.	2.6E-3	2
RA-2-3-10H	Local operation within 10 hours of a system or component which failed to automatically actuate.	2.6E-3	2
RA-3-12-2H	Open RCIC isolation valve(s) within 2 hours given RCIC room isolation.	2.4E-3	2
RA-3-12-68M	Open RCIC isolation valve(s) within 68 minutes given RCIC room isolation.	1.8E-2	2
RA-3-12-80M	Open RCIC isolation valve(s) within 80 minutes given RCIC room isolation.	3.5E-3	2
RA-3-12-10H	Open RCIC isolation valve(s) within 10 hours given RCIC room isolation.	2.4E-3	2
RA-4-4H	Isolate recirculation pump seal LOCA <u>AND</u> restore PCS.	1.0E-3	4

Table 5.15 (Continued)
Recovery Actions in LaSalle PRA

Event Name	Definition	Value	Source
RA-5V-1-2H	Operators vent within 2 hours through alternate vent path.	2.1E-3	2
RA-5V-1-6H	Operators vent within 6 hours through alternate vent path.	2.1E-3	2
RA-6-4H	If one electric power train has failed, one-half of the time the recirculation pump LOCA will occur on the recirculation pump which can be isolated. Operators isolate recirculation pump seal LOCA and restore PCS.	5.0E-1	5
RA-7-1-15H	Locally open within 15 hours a manual valve clored due to unscheduled maintenance on RHR pump C003B. Res ores heat removal.	2.1E-3	2
RA-7-1-27H	Locally open within 27 hours a manual valve closed d[illegible] to unscheduled maintenar[illegible] n RHR pump C003B. Restore [illegible] removal.	2.1E-3	2
RA-7-3-8H	Locally open within 8 [illegible]rs a manual valve closed d[illegible] to unscheduled maintenance on RHR pump C003B. Restores injection.	2.1E-3	2
RA-7-3-10H	Locally open within 10 hours a manual valve closed due to unscheduled maintenance on RHR pump C003B. Rest res injection.	2.1E-3	2
RA-8-10H	Restoration within 10 hours of offsite power.	1.7E-2	1
RA-8-15H	Restoration within 15 hours of offsite power.	6.9E-3	1
RA-8-1H	Restoration within 1 hour of offsite power.	1.7E-1	1
RA-8-23H	Restoration within 23 hours of offsite power.	2.5E-3	1
RA-8-27H	Restoration within 27 hours of offsite power.	1.9E-3	1
RA-8-48M	Restoration within 48 minutes of offsite power.	2.2E-1	1
RA-8-80M	Restoration within 80 minutes of offsite power.	1.1E-1	1
RA-8-8H	Restoration within 8 hours of offsite power.	2.0E-2	1

Table .15 (Continued)
Recovery Actions in LaSalle PRA

Event Name	Definition	Value	Source
RA-8-SEI-L1-1H	Restoration within 1 hour of offsite power given that a level L1 seismic event has occurred.	1.0	6
RA-8-SEI-L1-48M	Restoration within 48 minutes of offsite power given that a level L1 seismic event has occurred.	1.0	6
RA-8-SEI-L1-8H	Restoration within 8 hours of offsite power given that a level L1 seismic event has occurred.	1.0	6
RA-8-SEI-L2-1H	Restoration within 1 hour of offsite power given that a level L2 seismic event has occurred.	1.0	6
RA-8-SEI-L2-48M	Restoration within 48 minutes of offsite power given that a level L2 seismic event has occurred.	1.0	6
RA-8-SEI-L2-8H	Restoration within 8 hours of offsite power given that a level L2 seismic event has occurred.	1.0	6
RA-8-SEI-L3-1H	Restoration within 1 hour of offsite power given that a level L3 seismic event has occurred.	1.0	6
RA-8-SEI-L3-48M	Restoration within 48 minutes of offsite power given that a level L3 seismic event has occurred.	1.0	6
RA-8-SEI-L3-8H	Restoration within 8 hours of offsite power given that a level L3 seismic event has occurred	1.0	6
RA-8-SEI-L4-1H	Restoration within 1 hour of offsite power given that a level L4 seismic event has occurred.	1.0	6
RA-8-SEI-L4-48M	Restoration within 48 minutes of offsite power given that a level L4 seismic event has occurred.	1.0	6
RA-8-SEI-L4-8H	Restoration within 8 hours of offsite power given that a level L4 seismic event has occurred.	1.0	6
RA-8-SEI-L5-1H	Restoration within 1 hour of offsite power given that a level L5 seismic event has occurred.	1.0	6
RA-8-SEI-L5-48M	Restoration within 48 minutes of offsite power given that a level L5 seismic event has occurred.	1.0	6
RA-8-SEI-L5-8H	Restoration within 8 hours of offsite power given that a level L5 seismic event has occurred.	1.0	6

Table 5.15 (Continued)
Recovery Actions in LaSalle PRA

Event Name	Definition	Value	Source
RA-8-SEI-L6-1H	Restoration within 1 hour of offsite power given that a level L6 seismic event has occurred.	1.0	6
RA-8-SEI-L6-48M	Restoration within 48 minutes of offsite power given that a level L6 seismic event has occurred.	1.0	6
RA-8-SEI-L6-8H	Restoration within 8 hours of offsite power given that a level L6 seismic event has occurred.	1.0	6
RA-8-SEI-LL1-1H	Restoration within 1 hour of offsite power given that a level LL1 seismic event has occurred.	1.0	6
RA-8-SEI-LL1-48M	Restoration within 48 minutes of offsite power given that a level LL1 seismic event has occurred.	1.0	6
RA-8-SEI-LL1-8H	Restoration within 8 hours of offsite power given that a level LL1 seismic event has occurred.	1.0	6
RA-8-SEI-LL2-1H	Restoration within 1 hour of offsite power given that a level LL2 seismic event has occurred.	1.0	6
RA-8-SEI-LL2-48M	Restoration within 48 minutes of offsite power given that a level LL2 seismic event has occurred.	1.0	6
RA-8-SEI-LL2-8H	Restoration within 8 hours of offsite power given that a level LL2 seismic event has occurred.	1.0	6
RA-9-2H	Repair of DG failure within 2 hours.	8.7E-1	3
RA-9-10H	Repair of DG failure within 10 hours.	5.5E-1	3
RA-9-15H	Repair of DG failure within 15 hours.	4.7E-1	3
RA-9-1H	Repair of DG failure within 1 hour.	9.3E-1	3
RA-9-[illegible]	Repair of DG failure within 23 hours.	4.1E-1	3
RA-9-[illegible]	Repair of DG failure within 27 hours.	4.0E-1	3
RA-9-48	Repair of DG failure within 48 minutes.	9.6E-1	3
RA-9-8H	Repair of DG failure within 8 hours.	6.0E-1	3
RA-9-SEI-1H	Repair of DG failure within 1 hour given that a seismic event has occurred.	1.0	7
RA-9-SEI-48M	Repair of DG failure within 48 minutes given that a seismic event has occurred.	1.0	7
RA-9-SEI-8H	Repair of DG failure within 8 hours given that a seismic event has occurred.	6.4E-1	7

Table 5.15 (Continued)
Recovery Actions in LaSalle PRA

Event Name	Definition	Value	Source
RA-10-1-27H	Replace a fuse within 27 hours in a system or component that has no automatic operation or prior to its automatic operation if it has automatic actuation.	2.1E-3	2
RA-ATW-11-11-30M	Close SBLC F016 or F017 valve within 30 minutes after the occurrence of an ATWS, given the failure to close the valves following a previous test on the SBLC system.	1.0	2
RA-ATW-16-31-30M	Manual start of a DG from the control room AND then manual start of the appropriate SBLC pump within 30 minutes after the occurrence of an ATWS.	1.0	2
RA-ATW-16-31-59M	Manual start of a DG from the control room AND then manual start of the appropriate SBLC pump within 59 minutes after the occurrence of an ATWS.	1.0	2
RA-ATWS-1-3-25M	Manual operation within 25 minutes of a system or component from the control room which failed to automatically actuate after the occurrence of an ATWS.	3.0E-2	2
RA-ATWS-1-3-59M	Manual operation within 59 minutes of a system or component from the control room which failed to automatically actuate after the occurrence of an ATWS.	3.2E-3	2
RA-ATWS-12-3-10M	Locally close RWCU valve F004 within 10 minutes after the occurrence of an ATWS.	1.0	2
RA-ATWS-2-3-25M	Local operation within 25 minutes of a system or component which failed to automatically actuate after the occurrence of an ATWS.	1.1E-1	2
RA-ATWS-2-3-59M	Local operation within 59 minutes of a system or component which failed to automatically actuate after the occurrence of an ATWS.	7.4E-3	2
RA-ATWS-8-25M	Restoration within 25 minutes of offsite power after an ATWS has occurred.	4.0E-1	1

Table 5.15 (Continued)
Recovery Actions in LaSalle PRA

Event Name	Definition	Value	Source
RA-ATWS-8-59M	Restoration within 59 minutes of offsite power after an ATWS has occurred.	1.7E-1	1
RA-ATWS-8-85M	Restoration within 85 minutes of offsite power after an ATWS has occurred.	1.1E-1	1
RA-ATWS-9-59M	Repair of DG failure within 59 minutes after the occurrence of an ATWS.	9.3E-1	3
RA-ATWS-9-85M	Repair of DG failure within 85 minutes after the occurrence of an ATWS.	9.0E-1	3
RA-CDS-2H	Operators use condensate system within 2 hours.	2.2E-3	2
RA-DDFP-2H	Operators use diesel driven fire-water pump within 2 hours.	1.0E-1	2
RA-DELETE	Used to delete invalid cut sets.	0.0	
RA-NONE	No recovery action identified.	1.0	
PCICRMCOOL-DELET	Used to delete not applicable cut sets	0.0	
TDRFP-T-OE-15H	Operators fail to trip turbine driven reactor feedwater pumps within 15 hours. Prohibits motor driven feedwater pump from auto starting.	2.6E-3	2
TDRFP-T-OE-25M	Operators fail to trip turbine driven reactor feedwater pumps within 25 minutes. Prohibits motor driven feedwater pump from auto starting.	3.0E-2	2
TDRFP-T-OE-27H	Operators fail to trip turbine driven reactor feedwater pumps within 27 hours. Prohibits motor driven feedwater pump from auto starting.	2.6E-3	2
TDRFP-T-OE-48M	Operators fail to trip turbine driven reactor feedwater pumps within 48 minutes. Prohibits motor driven feedwater pump from auto starting.	6.4E-3	2
TDRFP-T-OE-69M	Operators fail to trip turbine driven reactor feedwater pumps within 69 minutes. Prohibits motor driven feedwater pump from auto starting.	2.6E-3	2

Table 5.15 (Concluded)
Recovery Actions in LaSalle PRA

Event Name	Definition	Value	Source
TDRFP-T-OE-95M	Operators fail to trip turbine driven reactor feedwater pumps within 95 minutes. Prohibits motor driven feedwater pump from auto starting.	2.6E-3	2

NOTES: 1 - Modeling Time to Recovery of Loss of Off-Site Power Plants, NUREG/CR-5032.
2 - Recover Actions in PRA for the Risk Methods Integration and Evaluation Program (RMIEP) Volume 2: Application of the Data Based Method, NUREG/CR-4834/2 of 2
3 - Station Blackout Accident Analyses (Part of NRC Task Action Plan A-44), NUREG/CR-3226 and Analysis of Core Damage Frequency From Internal Events: Peach Bottom, Unit 2, NUREG/CR-4550/Volume 4.
4 - "RA-4"
5 - "RA-6"
6 - seismic LOSP
7 - 3 plus time requirement

6.0 RESOLUTION OF CORE VULNERABLE ACCIDENT SEQUENCES

6.1 Introduction

In this section, we are concerned with the resolution of an issue that appears at the interface between the Level I and Level II/III analyses and again in the accident progression analysis in the Level II analysis. In the Level I analysis, certain of the end-states of the accident sequences may be initially undefined (e.g., whether or not core damage occurs is unknown). This uncertainty involves the interaction between the containment and the systems that must respond to the accident as described below and this interaction must be evaluated in order to resolve the sequence status. In the Level II analysis, the status of systems after containment failure may not be known. This issue also involves the interaction between the containment response and the systems and must be evaluated in order to evaluate the characteristics of the radioactive release. The Level II aspect of this issue is described in the Level II/III report.*

For the Level I analysis, in the past, engineering judgement with little or no supporting calculations was used to resolve the end-state. (Usually to simply say that it was core damage since almost no information was available and it was conservative to assume so.) For the LaSalle PRA, it was decided to use a more realistic approach in which thermal-hydraulic analyses were to be coupled with expert judgement to determine the survivability of the systems.

The accident sequence end-states which were initially undefined in the LaSalle analysis involved sequences in which core cooling was initially available, but containment heat removal was not sufficient to prevent containment pressurization. For these cases, containment failure or venting is guaranteed, and the resultant steam blowdown to the reactor building could fail critical components of the cooling systems, leading to core damage. The following procedure was used to resolve these end-states:

1. Identify those accident progressions with uncertain end-states.

2. Evaluate the containment failure location, size, and failure pressure.

3. Perform appropriate thermal-hydraulic analyses to evaluate possible reactor building environments for the cases identified in step 2.

4. Evaluate equipment survivability in these environments.

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* T. D. Brown, A. C. Payne Jr., L. A. Miller, J. D. Johnson, D. I. Chanin, A. W. Shiver, S. J. Higgins, and T. T. Sype, "Integrated Risk Assessment for the LaSalle Unit 2 Nuclear Power Plant: Phenomenology and Risk Uncertainty and Evaluation Program (PRUEP), Volume 1 Main Report," NUREG/CR-5305, SAND90-2765, Sandia National Laboratories, Albuquerque, NM, to be published.

5. Develop system models in order to quantify system failure probabilities for use in quantification.

6. Use this information to resolve the question of whether or not the core cooling systems would fail after containment venting or failure.

These steps will be discussed in detail in the remainder of this section.

6.2 Description of Steps in Core Vulnerable Sequence Resolution

6.2.1 Step 1: Define Core Vulnerable Sequences

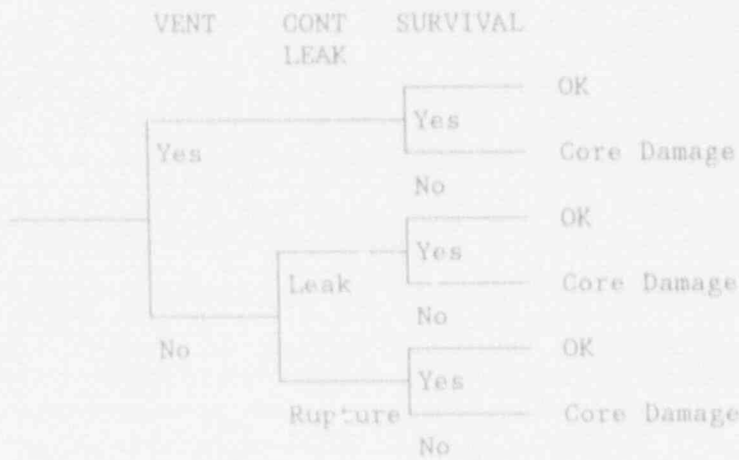
For the LaSalle PRA certain sequences in the Level 1 analysis were not initially resolved (i.e., whether or not the sequence proceeded to core damage was not known). These sequences are the so called 'core vulnerable' sequences in which the core is initially coolable but in which core damage may occur later in the sequence if cooling systems fail in the severe environment created by the accident. In the LaSalle analysis, these sequences arise either from accident sequences in which core cooling is available and containment heat removal has failed (TW) or in anticipated transients without scram (ATWS) where the heat load is beyond the capability of the containment heat removal systems. In either case, the containment heats up and pressurizes. The reactor core isolation cooling (RCIC) system will fail due to back pressure at around .277 MPa (40 psia), and the low pressure systems will fail their function when the automatic depressurization system (ADS) valves reclose at about .689 MPa (100 psia). If other high pressure systems are working, they will continue to operate unless they also fail due to the severe environments after containment venting or failure.

For these types of sequences (TW and ATWS), the emergency procedures direct the operators to vent the containment through 5.1 cm (2") lines in the wetwell and drywell if and when the containment pressure exceeds 0.517 MPa (60 psig). These two 5.1 cm lines can not remove sufficient energy to prevent further pressurization and the operator will be directed to vent using the 0.66 m (26") wetwell and/or drywell lines. The two 0.66 m lines connect via a common 0.46 m (18") line to the standby gas treatment system (SGTS) which limits the relief size. The 0.46 m line connects to the SGTS supply fans which have a short section of ductwork and a rubber boot, both of which are virtually certain to fail if a 0.66 m line is opened. This will release the vented steam into the reactor building instead of to the environment.

If venting did not or cannot occur, then the containment is assessed by the experts to most likely fail when the pressure reaches the 1.41 MPa (190 psig) range. Depending on the containment failure mode and location, steam may be released into the reactor building or to the refueling floor. If the release is to the refueling floor, no severe environments will be generated in the reactor building because the refueling floor walls will fail, thus directing the steam to the outside. If the steam is released directly into the reactor building, steam will fill the reactor building, creating environments of varying severity depending upon the building design and steam blowdown rate.

In order to incorporate these considerations directly into the analysis, additional events were added to the accident sequence event trees as described in Volume 4 of this report. First, a venting question was added, and if venting succeeded, then a system survival question was asked. If venting failed, then a containment failure mode question was added followed by a system survival question (see Figure 6.1).

Figure 6.1
System Failure Resolution



6.2.2 Step 2: Determine Containment Failure Modes

The Structural Expert Elicitation Panel for the NUREG-1150 expert elicitation[1] was asked to evaluate the structural design information and construct a probability distribution for the containment failure pressure. The experts each received structural design information and previous calculations on the LaSalle and similar containments. They received the results of experiments on containments and equipment hatchs, and performed some simplified calculations of their own. Using this information, they were asked to evaluate, at each pressure, the probability of containment failure and then the conditional probability of the containment failing in one of the following eight modes:

1) Wetwell Leak above the water line (WWLaW)
2) Wetwell Leak below the water line (WWLbW)
3) Wetwell Rupture above the water line (WWRaW)
4) Wetwell Rupture below the water line (WWRbW)
5) Drywell Leak (DWL)

6) Drywell Rupture (DWR)
7) Drywell Head Leak (DWHL)
8) Drywell Head Rupture (DWHR)

These modes were selected, firstly, because we needed to differentiate between leaks and ruptures in order to know if the containment pressure would drop to the point were low pressure systems could be used before core damage occurred. Secondly, we had to differentiate based upon the location in order to know if the failure would create a severe environment in the reactor building or if the failure would be to the refueling floor which would bypass the reactor building and not affect the systems environments. Finally, we had to differentiate effects on the source term of suppression pool and secondary containment decontamination.

The overall issue and results of this process for the NUREG-1150 plants are described in Reference 1. For the LaSalle analysis, the results are presented in Appendix C. The mean failure pressure was 191 psig. The marginal failure probabilities for the individual modes (these are the weighted average over all pressures, i.e., the sum of the conditional probabilities at each pressure interval times the probability density of failure in that interval), calculated from the results in Appendix C, are:

Table 6.1
Marginal Failure Probabilities

WWLaW = 0.1094	WWRaW = 0.1111
WWLbW = 0.0156	WWRbW = 0.0105
DWL = 0.0746	DWR = 0.0858
DWHL = 0.5487	DWHR = 0.0442

We use the marginals for the point estimate since the pressure will continue to rise until containment fails for these sequences. As a result of grouping the failure modes, we can calculate various conditional probabilities:

1. The conditional probability of a leak is 0.7483 and a rupture is 0.2516.

2. Given a leak, the conditional probability that it is to the refueling floor is 0.7333 and to the reactor building is 0.2667.

3. Given a rupture, the conditional probability that it is to the refueling floor is 0.1757 and to the reactor building is 0.8243.

6.2.3 Step 3: Evaluate the Reactor Building Environments

The MELCOR[2] code was used to perform the thermal-hydraulic analysis of the effects of containment failure and blowdown from high pressure into the reactor building.[3] A detailed MELCOR model was constructed for the reactor building using information from the plant drawings, the Final Safety Analysis Report, and two volumetric and heat transfer models developed by the architect/engineer for LaSalle (Sargent and Lundy) to perform steam line break calculations. The reactor building was divided into 27 volumes as shown in Figure 6.2. Since the main concern is equipment survival in the lower levels of the reactor building, more detailed noding was used in these regions. Single volumes were used to model the steam tunnels, refueling floor, and the unit 1 reactor building.

MELCOR was chosen to perform most of the thermal-hydraulic analyses for the PRA because (1) it can be used to perform an integrated analysis that considers reactor vessel, primary containment, and reactor building response simultaneously; (2) it is fast running; (3) it has flexible control function capability for modeling flow paths; and, (4) it includes the capability to address uncertainties in modeling parameters and correlations. This detailed deck will also be used for special analyses of reactor building response to hydrogen and carbon monoxide burns and for fission product transport in the Level II/III analysis. The deck has also been simplified and incorporated into another deck being used for integrated calculations for the Level II/III analysis.

6.2.3.1 Reactor Building Model Description

A detailed MELCOR model was constructed for the reactor building using information from the plant drawings, the Final Safety Analysis Report (FSAR),[4] and two computer models developed by the architect/engineer (AE) for LaSalle, Sargent and Lundy, for use in design calculations. One of the Sargent and Lundy models was used to calculate gas flow between rooms and had detailed calculations of flow path areas and resistances. The other model was used for room environment calculations after high energy line breaks and had detailed calculations of room volumes and surface areas. Neither model had estimates of equipment masses or surface areas, so these were estimated based on the Level I location analysis that had identified all the equipment in each room of the reactor building.

It is important to have sufficient nodalization to model the building characteristics that determine the flow patterns for areas where important equipment is located. Also, adequate representation of doors and blowout panels is necessary because the flow patterns can be greatly affected if normally closed flow paths are opened during the severe transients. Slight differences in opening pressure differentials can determine the exact configuration of flow paths for the various scenarios analyzed.

The reactor building was therefore divided into 27 volumes as shown in Figure 6.2. Since the main concern is equipment survival in the lower levels (floors) of the reactor building, more detailed nodalization is used in these regions. The annulus (outside the primary containment on the lower two levels), high pressure core spray (HPCS) and low pressure core

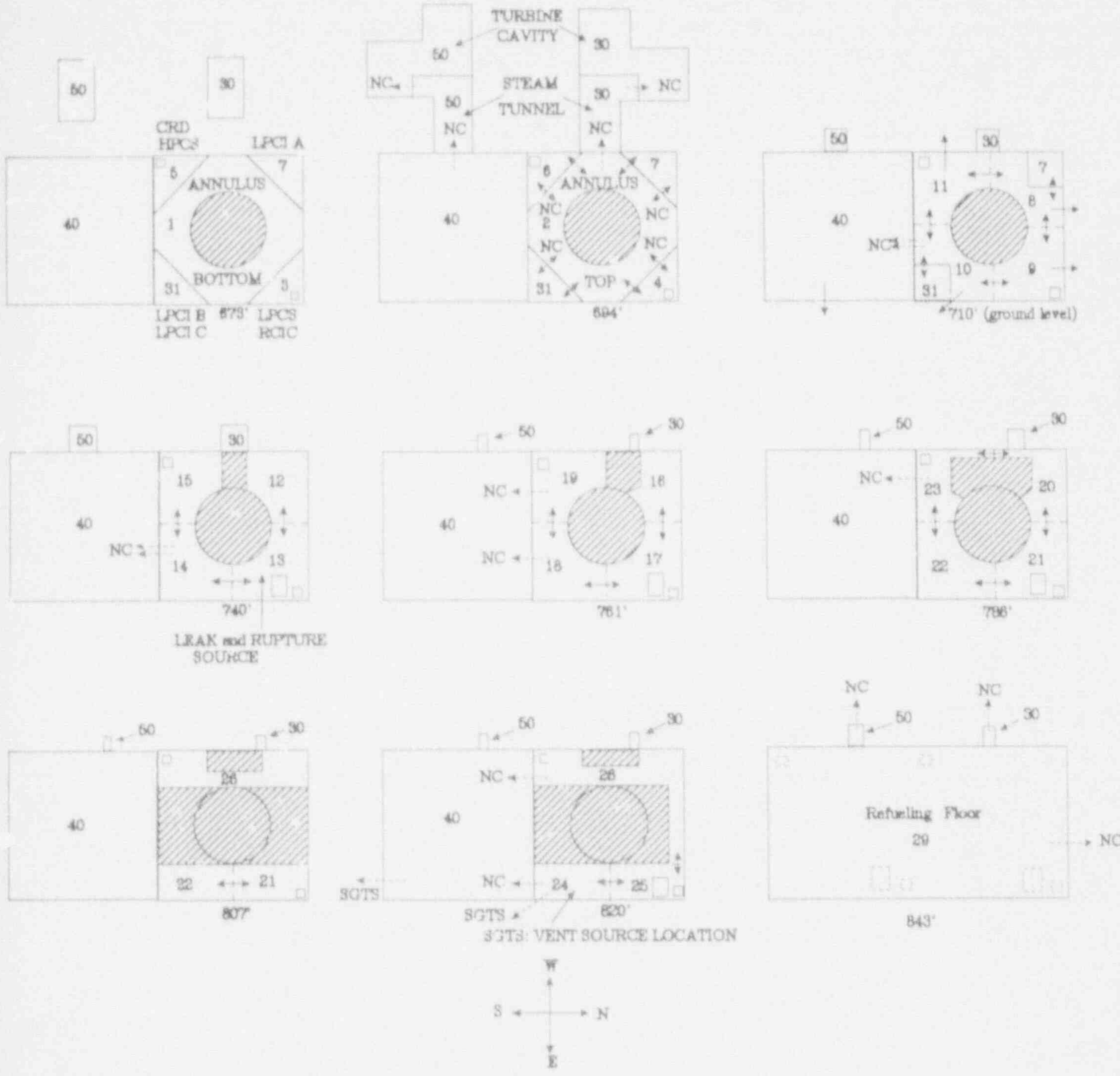


Figure 6.2
MELCOR Nodalization for Reactor Building Model

spray (LPCS) rooms are each divided into two volumes to represent the upper and lower levels. The low pressure coolant injection (LPCI) rooms are modeled with single volumes because the heating, ventilation, and air conditioning system (HVAC) circulates between the upper and lower levels, resulting in well-mixed regions. Levels 710', 740', 761', and 786.5' are each divided into four quadrants to allow the main circulation paths to be calculated. The East portions of levels 807' and 820' are each divided into two volumes and the more dead-ended regions at the West end of the two levels are lumped into a single volume. Single volumes are used to model the steam tunnel (including turbine cavity), refueling floor, and the unit 1 reactor building.

The flow paths in the model are also shown in Figure 6.2. Normally, the corner rooms in the basement of the reactor building are fairly isolated from the other regions, but circulation is increased if doors are blown open during a severe transient. Unlike the basement where the levels are subdivided into rooms that restrict flow, at levels 710' and above, the floors are essentially wide open. Also, there are reasonably large flow areas between the upper levels through stairways and an equipment hatch. Initially, the reactor building is isolated from the refueling floor, but paths can be opened if a door is blown open or concrete slabs are lifted from over the equipment hatch. The walls of the refueling floor level are assumed to fail at 14 kPa (2 psig), opening a 7 m (23 ft) diameter hole to the environment. The reactor building can also vent to the unit 1 reactor building if pressure increases sufficiently to blow open the doors between the two units. In addition, the reactor building can vent from the upper level of the annulus into the steam tunnel and into the turbine cavity if a very small pressure differential is exceeded. A blowout panel in the reactor building return air riser at the top of the steam tunnel is included in the model. All leakage/infiltration paths between the reactor building and environment are lumped into flow paths at the 710 level.

Heat structures are included in all reactor building volumes to model heat transfer to walls, ceilings, floors, and equipment. Heat removal by the room coolers in the basement corner rooms is also modeled. Flow of gases through the standby gas treatment system was included in all runs; failure because of the severe environment was not considered.

A simplified nodalization for the primary containment and reactor pressure vessel (RPV) is used to provide blowdown sources to this detailed reactor building model. The RPV is modeled by a single volume, and 3 volumes are used for primary containment. The containment gases are exhausted to the reactor building at level 820' (volume 324) for cases examining venting, and to level 740' (volume 313) for cases examining containment failure.

6.2.3.2 Results of Analysis

Calculations were performed for venting the primary containment through an 18" diameter (.46m) line from the wetwell to the top of the reactor building and for 2 sizes of drywell breaks: 4" diameter (.10m) and 36" diameter (.91m). To examine modeling sensitivities, 4 variations of the venting calculation were run:

1) Five times the equipment mass
2) Twice the rated heat removal rate for the room coolers
3) Vent area reduced in half (6.4" diameter)
4) Blowout panel on the refueling floor to the outside environment

The reactor building pressure for the 4" drywell break is shown in Figure 6.3. The early pressurization opened one of the doors to unit 1 and the door to the refueling floor, but the blowdown was not large enough to open paths to the environment by either failing the walls of the refueling floor or opening the blowout panel at the top of the steam tunnel. The pressurization was relieved through leakage paths, the SGTS, and condensation on structures. Since the flow was not being forced through the steam tunnel, little steam was drawn down into the emergency core cooling systems (ECCS) rooms in the basement. The reactor building heatup was relatively gradual as shown by the temperatures plotted in Figure 6.4 and listed in Table 6.2.

The pressurization was higher for the 36" diameter drywell break (equivalent to 7 sq ft), as shown in Figure 6.5. All doors and blowout panels were forced open except for three of the doors between the annulus and corner rooms in the basement. With the refueling floor walls failed, most of the blowdown was carried upward through the reactor building rather than being pushed down through the basement and out through the steam tunnel. However, there was sufficient flow down into the basement rooms to cause considerable heatup (i.e., final temperatures > 400K) as shown in Figure 6.6 and Table 6.2.

For the 18" wetwell vent case, the steam entered near the top of the reactor building rather than near the bottom. The pressurization from the blowdown opened 3 of the upper doors to unit 1, the door to the refueling floor, and the steam tunnel upper blowout panel, but the walls of the refueling floor were not predicted to fail. Thus, for this case, the majority of the steam was drawn down through the basement, then into the steam tunnel and turbine cavity before exhausting to the environment. As a result, relatively high temperatures (i.e. ~370 -400K) were predicted in the basement rooms as shown in Figure 6.7 and Table 6.2.

The variation of the 18" vent case with increased steel area was virtually identical to the base case. Pressures and temperatures were only reduced slightly. Using twice the rated heat removal for the room coolers also had negligible effect on the pressures and on the temperatures in all rooms except those directly connected to the room coolers. As seen in Table 6.3, the peak and average temperatures in those rooms were reduced on the order of 5 - 10 K. For the case using half the blowdown rate, the peak pressure was reduced by about 5 kPa (3/4 psig) at the top of the reactor building and decreased back to atmospheric pressure at about twice the rate of the

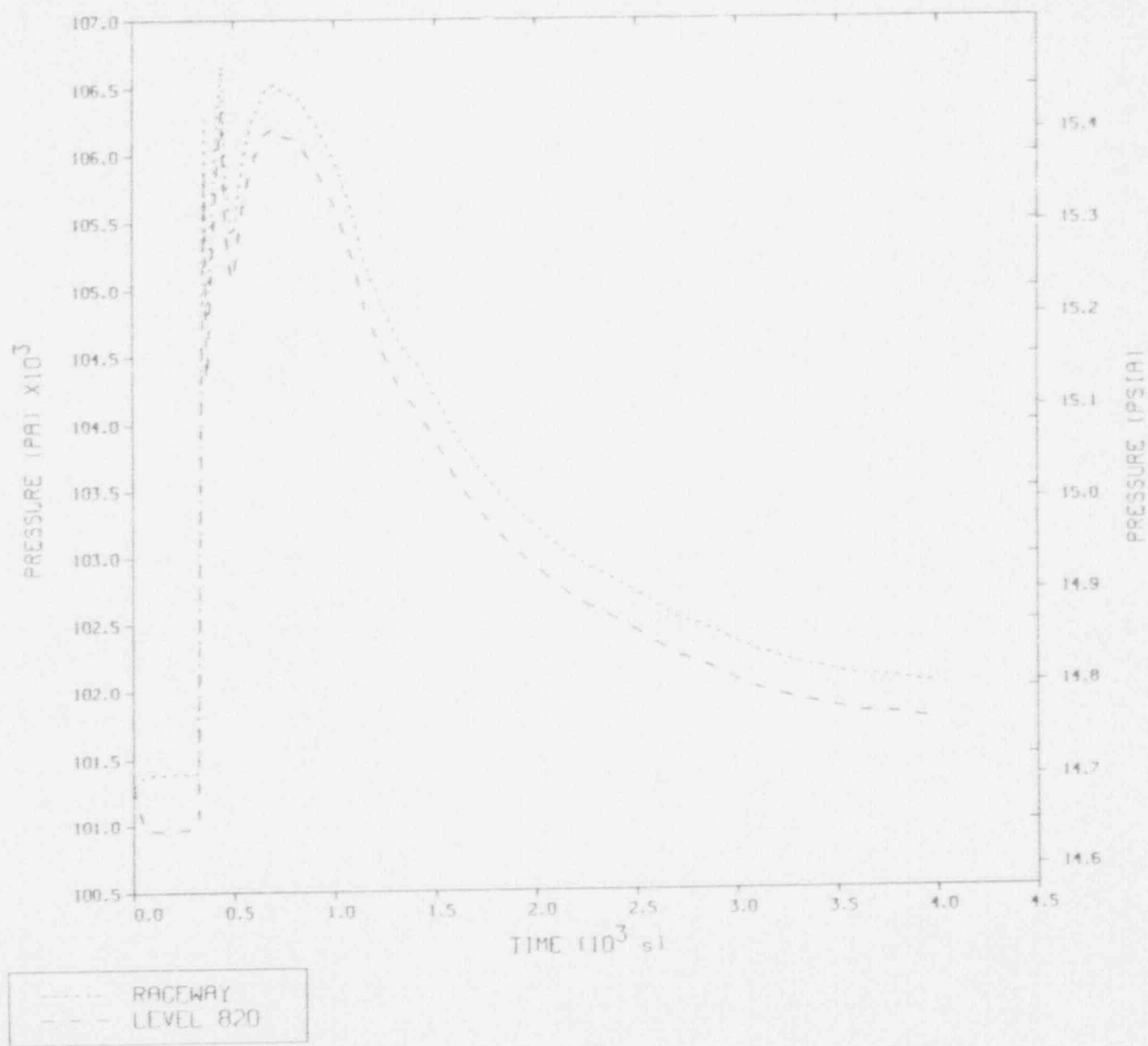


Figure 6.3
Reactor Building Pressures for 4" Drywell Break

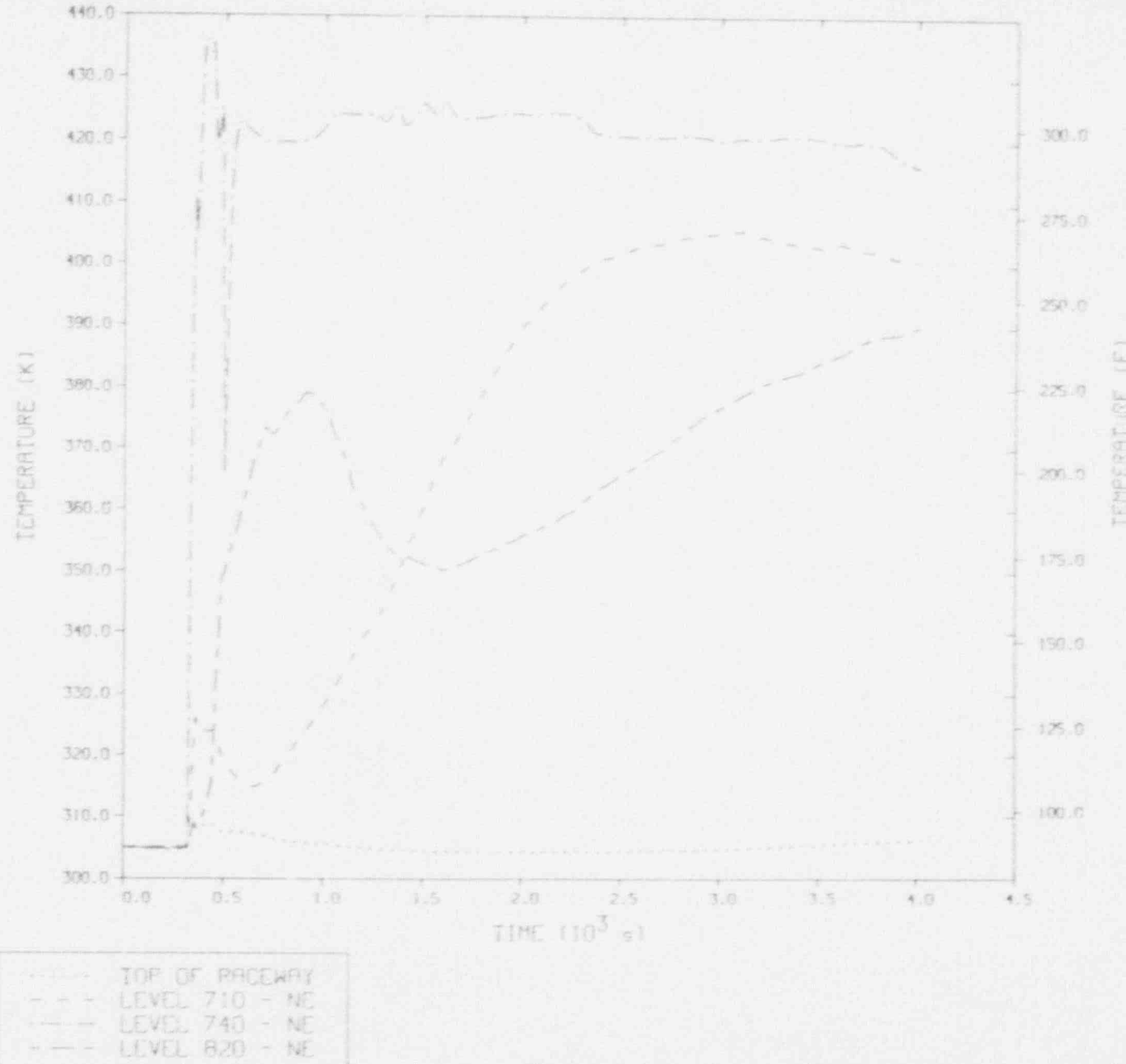


Figure 6.4
Reactor Building Temperatures for 4" Drywell Break

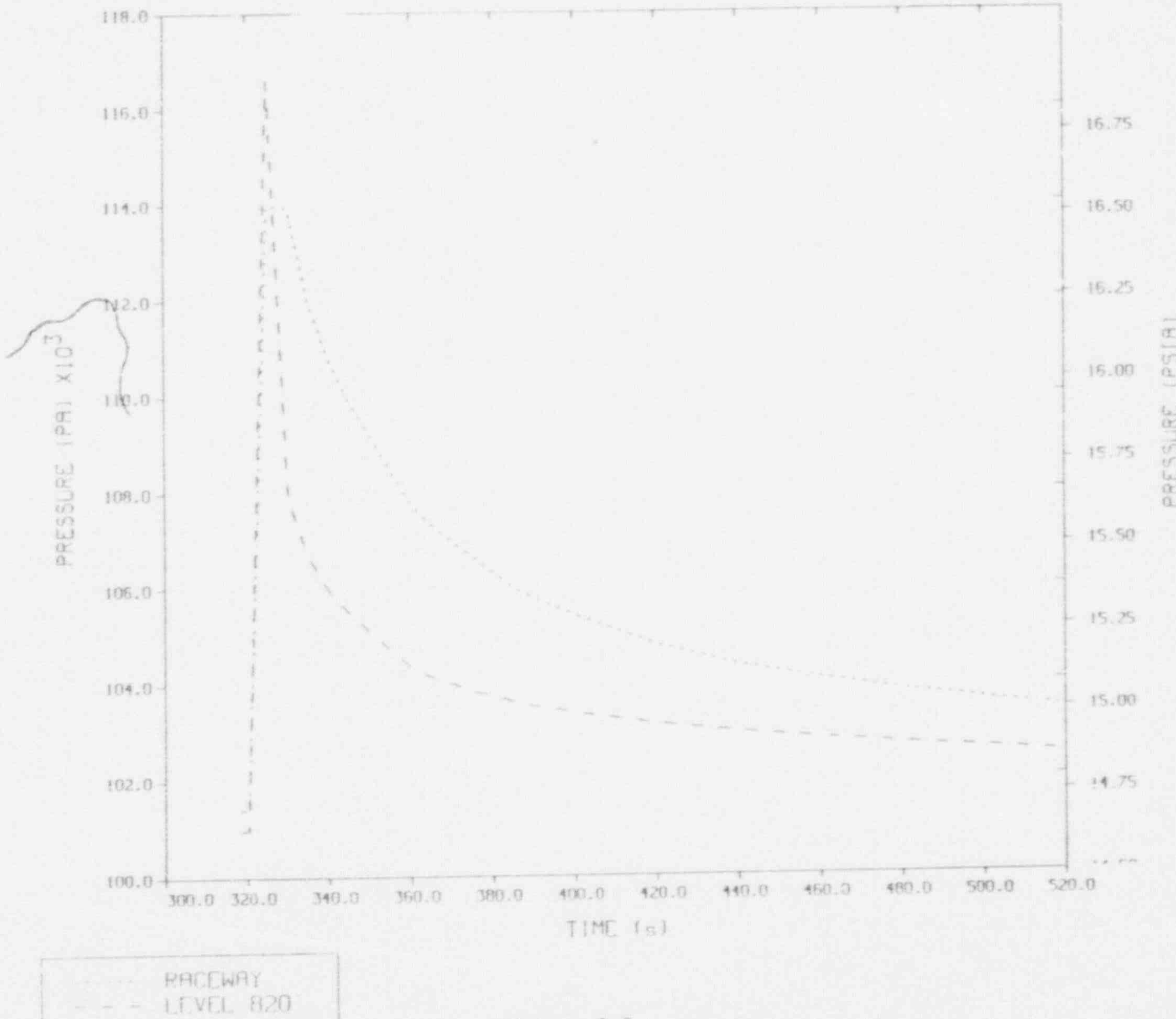


Figure 6.5
Reactor Building Pressures for 36" Drywell Break

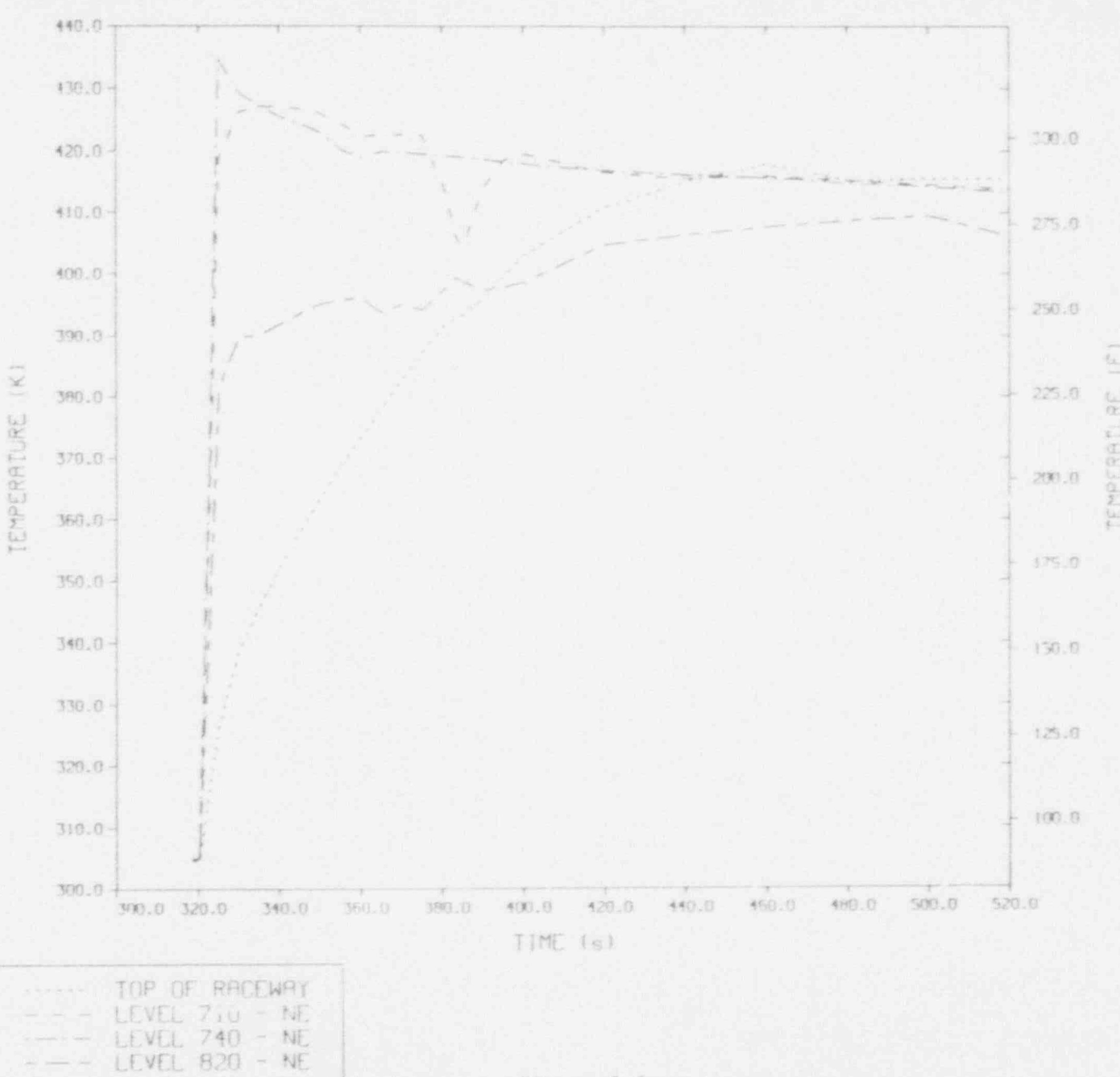


Figure 6.6
Reactor Building Temperatures for 36" Drywell Break

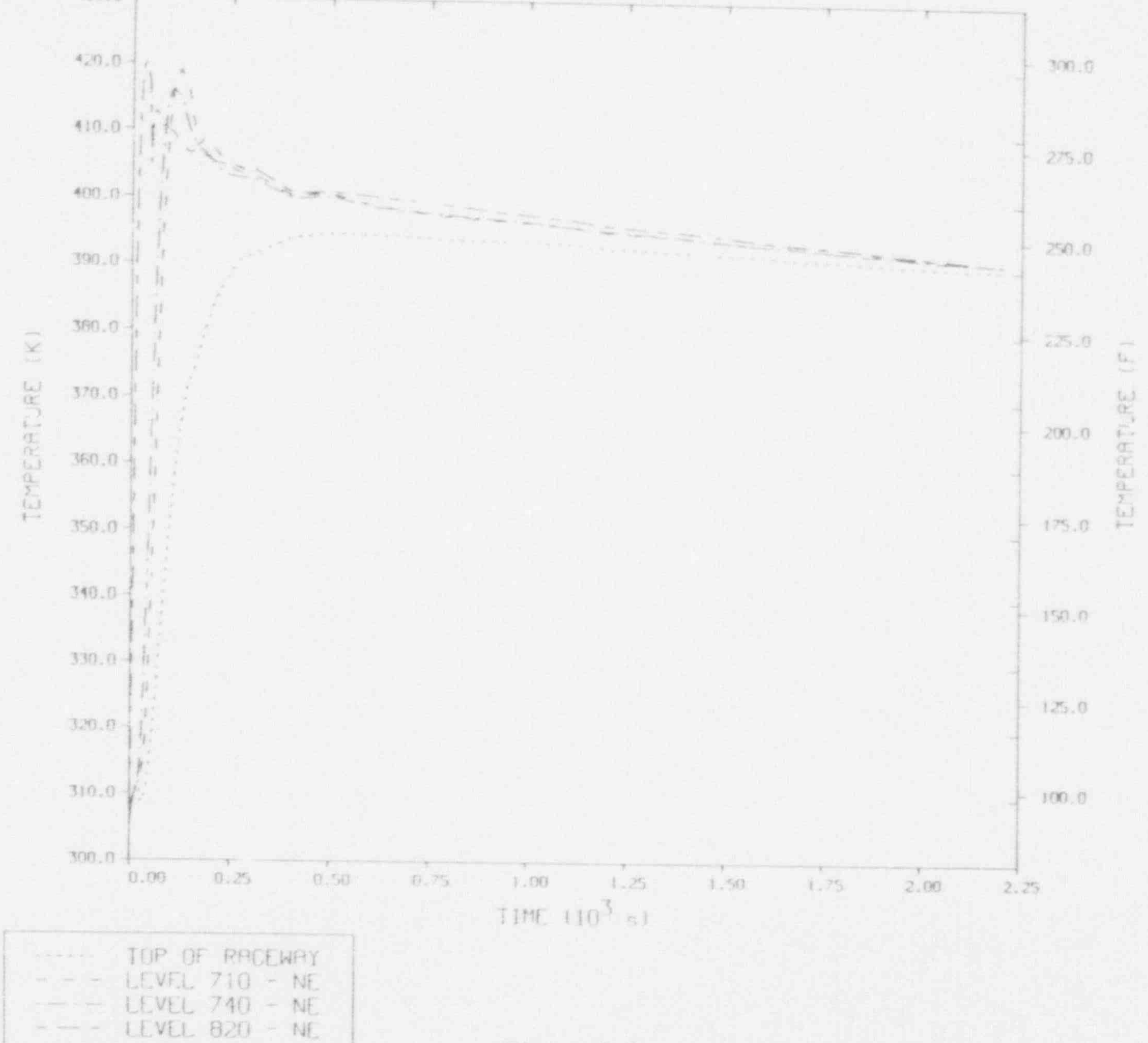


Figure 6.7
Reactor Building Temperatures for Wetwell Venting

Table 6.2
Base Cases' Temperatures (K)

	4" Leak		18" Vent		36" Rupture	
Volume	Peak	Average	Peak	Average	Peak	Average
301	309	309	310	305	320	305
302	309	305	390	390	415	415
303	320	315	375	380	355	345
304	330	325	395	390	380	375
305	309	309	315	308	325	308
306	313	310	390	390	420	420
307	305	297	400	390	373	373
308	365	365	415	390	430	415
309	405	400	420	390	430	415
310	365	365	395	390	430	415
311	395	390	390	390	430	415
313	435	420	410	395	435	410
317	420	415	410	395	435	410
321	400	390	410	395	410	410
324	345	340	415	395	410	410
325	390	390	420	395	400	400
331	305	299	390	390	420	420

Table 6.3
Sensitivity Cases' Temperatures (K)

	5 * Steel Mass		2 * Rated Fan Cooler Q		.5 * Vent Area		Refuel Floor Blowout	
me	Peak	Avg	Peak	Avg	Peak	Avg	Peak	Avg
301	310	305	310	305	310	305	310	305
302	390	390	390	390	385	385	385	385
303	375	375	370	370	355	355	370	370
304	395	395	380	380	375	375	390	390
305	310	310	310	300	310	310	310	310
306	385	385	380	380	380	380	360	360
307	400	395	390	385	310	300	310	300
308	420	390	415	395	350	350	395	395
309	420	390	420	395	390	390	410	395
310	390	390	395	395	380	380	375	375
311	385	385	385	385	380	380	365	365
313	415	395	415	400	400	400	405	400
317	410	395	410	400	415	400	400	400
321	410	395	410	400	415	400	400	400
324	420	395	415	400	415	400	415	400
325	425	395	420	400	425	400	420	400
331	385	385	385	385	310	300	310	300

base case. The smaller blowdown caused a much slower heatup of most of the reactor building, but by the end of the run, the temperatures were approaching the same level as in the base case. The LPCI room response varied more from the base case than the other rooms had because the doors did not blow open, giving a more restricted path into the room, and therefore, the temperatures remained nominal. In the final sensitivity case, the assumed blowout panel from the refueling floor to the environment opened almost immediately. This additional opening relieved the pressure more quickly than in the base case, resulting in about a 5 kPa (3/4 psig) reduction in peak pressure and a more rapid return to atmospheric pressure. About 2/3 of the steam went out through the refueling floor level, reducing the amount of steam being drawn down to lower levels and out the steam tunnel. Therefore, the response in the lower portions of the building resembled the response for the case with reduced vent flow area. However, the venting of steam through the refueling floor opening resulted in a change in the flow patterns such that flow was mainly directed down through the hatch with less circulation around each level. This can be observed by examining the room temperatures in Table 6.3.

6.2.3.3 Model Limitations

Since this is one of many analyses being performed as part of the PRA and due to limited resources, complete sensitivity calculations covering all possible variations in physical parameter estimates, code thermal-hydraulic models, initial conditions, and reactor building models cannot be evaluated explicitly. For the PRA, the impact of these uncertainties must be estimated so the uncertainty can be represented in the final result. Some of the dominant modeling uncertainties are discussed below.

1. We did not model the leakage path from the steam tunnel to the turbine building via the turbine cavity underneath the main turbine. Initially, this volume was believed to be isolated, but later information showed that there were various paths by which steam could reach the turbine building. All of these paths have fairly large flow resistances and we judge that the total flow will be small if any other path is open. However, for leaks, some portion of the flow would be drawn down into the annulus and out the steam tunnel. The information to model this is not available and would be very difficult to either calculate or estimate. Sensitivity runs or engineering judgement could be used to assess the impact on uncertainties. The steam tunnel volume was doubled to account for the cavity volume but this was later found to be too low. The turbine cavity is actually about 35,000 m^3. For leaks, this will draw hot steam down into the lower regions of the reactor building but should not result in significant additional heatup of the corner rooms. For venting and ruptures, the dominant flow paths will not change and, therefore, the environments in the reactor building should not be substantially affected.

2. At the time these calculations were performed, the drywell was predicted to fail at 12.0 bar (160 psig). More recent analyses by the NUREG-1150 expert review group, as described in Appendix C, predicted primary containment failure to occur in the wetwell and at a pressure of 1.41 MPa (190 psig). This difference will not significantly affect expected flow patterns in the reactor building and the resultant threat to equipment.

3. If the doors between the Unit 2 and Unit 1 reactor buildings blow out in more than 1 location, a flow path can form where steam flows from one location in Unit 2, through Unit 1, and back into a second location in Unit 2. Unit 1 is only modeled as a single volume, so any flow entering it is instantaneously mixed with the entire Unit 1 volume, rather than just mixing in a local region. This simplification affects the results, but it is probably less influential than other uncertainties in the problem.

4. The results are probably most sensitive to the setpoints of blowout panels and doors. As was discussed in the Results section, the status of these paths greatly affects temperatures within the various regions of the reactor building. The actual load the doors could withstand is unknown; we estimated values that seemed reasonable.

6.2.3.4 Conclusions

Because of the level of detail of the model, we were able to examine details of reactor building flow patterns that have not previously been examined. This level of detail reduced the uncertainty in a number of variables included in the model that could affect the results of the calculations (e.g. volumes, surface areas, flow path characteristics, and the effects of room cooling) and, therefore, the assessment of equipment survivability. The reduction in the number of uncertain parameters and the experience gained by varying some of them enables us to better use our engineering judgement to estimate the effects of the remaining parameters.

For all of the cases examined, the upper regions of the reactor building were relatively well mixed. For the 4" drywell leak case, the blowout panel in the steam tunnel did not open, so the basement rooms were buffered from the blowdown and remained relatively cool. For the 18" vent case, the steam tunnel blowout panel opened, but the walls of the refueling floor did not fail. As a result, steam was drawn down into the basement rooms, giving higher temperatures. For the 36" rupture case, the steam tunnel blowout panel was opened and the walls of the refueling floor failed. Although this allowed some of the steam to flow up through the reactor building, a substantial amount was still drawn down into the basement rooms, resulting in relatively high temperatures. Sensitivity calculations for the 18" vent case showed that heat transfer uncertainties were much less significant than uncertainties regarding possible flow path configurations.

6.2.4 Step 4: Evaluate Equipment Failure Probabilities.

The Expert Elicitation Panel for the NUREG-1150 Level I issues was supplied with the results of the above severe environment calculations for LaSalle and with a list of the types of equipment that appeared in the reactor building and their qualification characteristics. The experts were asked to assess the failure probability of the different categories of equipment in the various environments. The experts based their evaluation upon their knowledge of test and qualification procedures and results. The results of their analysis are reported in Reference 5. The actual distributions used in the Latin Hypercube[6] sample are reported in the LHS input file in Appendix D of Volume 2 of this report. However, we note here that the conditional failure probabilities were in the 0.1 to 1.0 range with wide distributions.

As an example, we use the control rod drive (CRD) system and its support systems in the case of a leak from the containment to the reactor building. From the expert elicitation, the values for the failure probabilities in severe environments for the CRD and reactor building closed cooling water (RBCCW) pumps and control circuits, the heating, ventilation, and air-conditioning (HVAC) system fan and its control circuit for the CRD room, spurious operation of a motor operated valve in the service water system (SW), and a 480 VAC motor control center are given in Table 6.4.

Table 6.5 contains the list of equipment evaluated, rough estimates of environmental qualification, their locations, the expert case used, and the median probability of failure to give a sample estimate for containment leaks, ruptures, and venting. Table 6.6 contains a summary of all the different cases evaluated for each component examined in Table 6.5 and Table 6.7 gives a summary list of the environments examined. In all the tables the following abbreviations occur: SO = spurious operation, SH = short to ground, FTR = fail to run, QT = qualification temperature, 1 and 10 = 1 or 10 hour exposure.

6.2.5 Step 5: Construct Simplified System Models.

For each system, the original fault tree models were examined and all equipment in the reactor building was identified. For each train, the components which had the highest failure probabilities in the environments to which they were subject were selected to represent train failure. The full system models could have been quantified since sufficient information was available; but, insufficient resources were available for full quantification, the probabilities were high and exact probability calculations would need to be done to get accurate answers, and the current level quality of the environmental, thermal-hydraulic, and expert judgement on containment failure and environmental failure analyses does not really justify that level of effort. Therefore, simple Boolean models were then constructed for the systems. Failure probabilities for the components were selected from the expert judgement results, and the system failure

Table 6.4
Sample Severe Environment Evaluation for CRD System

Event	Value (median)	Location (VOL)	Description
CRDP1FTR-SUR-E?	= 0	305	CRD pump
CRDP1CC-SUR-E?	= 0	305	CRD pump control cir.
2WRP1FTR-SUR-E?	= 0.2510	321	RBCCW pump
2WRP1CC-SUR-E?	= 0.7976	321	RBCCW pump control cir.
SWVY02CC-SUR-E?	= 0.5517	305,306	HVAC fan control cir.
SWVY02LF-SUR-E	= 0	305	HVAC fan
PSW175CC-SUR-E?	= 0.7442*(1-0.4400)	309,313	SW MOV spurious op.
	= 0.4168		
2WRMCC1-SUR-E?	= 0.7015	321	RBCCW MCC

The conditional probability of a leak to the reactor building given containment failure is:

LEAKTRB = 0.2667 (median)

Table 6.5a
Quantification for Leaks:

Types of components and environments:

1) CRD pumps FTR 10 hr, qual - 310 F?, envir - nominal temp (3I2)

2) RBCCW pumps FTR 10 hr, qual - 310 F?, envir - 250 F (3D)

3) Fan motors FTR 10 hr, qual - 355 F 100% hum, envir - nominal temp (3I2)

4 MOV motors FTR 10 hr, qual - 310 F?, envir - 260 F (3G-1) => prob 300 F if leak in wetwell, nominal temp (3H1 or 3I1), 280 F (3E).

5) CRD pump CC FTO 10 hr, qual - 185 F?, envir - nominal temp (3I2)

6) RBCCW pump CC FTO 10 hr, qual - 185 F?, envir - 250 F (3D)

7) Fan CC FTO 10 hr, qual - 185 F 95% hum, envir - (3I2,3H2) nominal temp for leaks on 3F-1 from drywell but would be more severe in 3H2 for leaks from the wetwell only one floor above (200-300 F depending on location of leak, use 240 F). Average the two with a .58/.42 split from expert mode probabilities.

8) Valve CC SO 10 hr, qual - 185 F?, envir - 300 F (3G-1 and 3F-1), nominal temp (3I1, 3H1), 280 F (3E), see #7 above use 240 F * .42 + 100 F * .58 (3H2).

9) MCC SH 10 hr, qual - 340 F 1hr then 320 F 1hr then 160 F 100 days 95% hum, envir - 250 F (3D), 300 F (3G-1).

10) ADS valves FTO 10 hr, qual - 350 F, envi: - 60 psig, 308 F (3J).

11) ADS valves FTO 10 hr, qual - 350 F, envir - 195 psig, 386 F (3J).

12) HPCS pump FTR 10 hr, qual - 310 F, nominal temp (3I2).

Table 6.5a (Concluded)
Quantification for Leaks:

From the expert elicitation, median values are:

Event	Value	Location	Quan Ref
CRDP1FTR-SUR-E?	= 0	3I2	FTR10, nominal
CRDP1CC-SUR-E?	= 0	3I2	FTO10, nominal
PSW175CC-SUR-E?	= .7442*(1-.4400)=.4168	3F-1,3G-1	SO10, QT+115 * /FTR10 QT15
SWVY02CC-SUR-E?	= .5517	3I2,3H2	FTO10, .58 * QT-75 + .42 * QT+50 = .58 * .3736 + .42 * .7976
SWVY02LF-SUR-E	= 0	3I2	FTR10, nominal
2WRP1FTR-SUR-E?	= .2510	3D	FTR10, QT-60
2WRP1CC-SUR-E?	= .7976	3D	FTO10, QT+80
2WRMCC1-SUR-E?	= .7015	3D	SH10, QT+100
HPCSPFTR-SUR-E?	=0	3I2	FTR10, nominal
HPCS01SP-SUR-E?	=0	3I1	SO10, nominal
HPCS23SP-SUR-E?	=0	3H1	SO10, nominal
HPCS04SP-SUR-E?	=.6950*(1-.3035)=.4841	3E	SO10, QT+100 * /FTR10 QT-35
HPCS15SP-SUR-E?	=.2260	3H1,3H2,3I1	SO10, (.58 * QT-75 + .42 * QT+50) * / FTR10QT-210 = (0 * .58 + .42 * .5381) * 1 = .2260
MC1E35Y2A-SUR-E?	=.7015	3G-1	SH10, QT+150
ADS18VAL-SUR-E?	=.0171	3J	FTO10, QT+50**18
LEAKTRB	= .2667		

Therefore:

CRD1 = 0 + 0 + .4168 + .5517 + 0 + .2510 + .7976 + .7016
= .9882
HPCS = .5517 + 0 + 0 + 0 + 0 + .4841 + .2260
= .8210
ADS = .0171
LPCI = .7015
LPCS = .7015
LPCI * LPCS = .7015
HPCS * CRD1 = .5517 + 0 + (0 + 0 + 0 + .4841 + .2260) * (0 + 0 + .4168 + .2510 + .7978 + .7015)
= .8139

where sums were combined using P(A + B) = P(A) + P(B) - P(AB).

Table 6.5b
Quantification for Ruptures:

Types of components and environments:

1) CRD pumps FTR 10 hr, qual - 310 F?, envir - nominal temp (3I2)

2) RBCCW pumps FTR 10 hr, qual - 310 F?, envir - 280 F (3D)

3) Fan motors FTR 10 hr, qual - 355 F 100% hum, envir - nominal temp (3I2)

4 MOV motors FTR 10 hr, qual - 310 F?, envir - 230 F (3G-1) => prob 280 F if rupture in wetwell, 290 F (3H1), nominal (3I1), 280 F (3E).

5) CRD pump CC FTO 10 hr, qual - 185 F?, envir - nominal temp (3I2)

6) RBCCW pump CC FTO 10 hr, qual - 185 F?, envir - 280 F (3D)

7) Fan CC FTO 10 hr, qual - 185 F 95% hum, envir - nominal (3I2), 300 F (3H2) for rupture on 3F-1 from drywell would be the same in 3H2 for ruptures from the wetwell only one floor above.

8) Valve CC SO 10 hr, qual - 185 F?, envir - 285 F (3G-1 and 3F-1), nominal temp (3I1), 290 F (3H1), 280 F (3E), 300 F (3H2).

9) MCC SH 10 hr, qual - 340 F 1hr then 320 F 1hr then 160 F 100 days 95% hum, envir - 280 F (3D), 280 F (3G-1).

10) ADS valves FTO 10 hr, qual - 350 F, envir - 60 psig, 308 F (3J).

11) ADS valves FTO 10 hr, qual - 50 F, envir - 195 psig, 386 F (3J).

12) HPCS pump FTR 10 hr, qual - 310 F, nominal temp (3I2).

Table 6.5b (Concluded)
Quantification for Ruptures:

From the expert elicitation, median values are:

Event	Value	Location	Quan Ref
CRDP1FTR-SUR-E?	= 0	3I2	FTR1, nominal
CRDP1CC-SUR-E?	= 0	3I2	FTO1, nominal
PSW175CC-SUR-E?	= .5659*(1-.2619)=.4177	3F-1,3G-1	SO1, QT+100 * /FTR1QT+35
SWVY02CC-SUR-E?	= .6898	3I2,3H2	FTO1, QT+115
SWVY02LF-SUR-E	= 0	3I2	FTR1, nominal
2WRP1FTR-SUR-E?	= .2619	3D	FTR1, QT-35
2WRP1CC-SUR-E?	= .6898	3D	FTO1, QT+100
2WRMCC1-SUR-E?	= .6220	3D	SH1, QT+125
HPCSPFTR-SUR-E?	=0	3I2	FTR1, nominal
HPCS01SP-SUR-E?	=0	3I1	SO1, nominal
HPCS23SP-SUR-E?	=.5659*(1-.3001)=.3961	3H1	SO1, QT+100 * /FTR1QT-10
HPCS04SP-SUR-E?	=.5659*(1-.3001)=.3961	3E	SO1, QT+100 * /FTR1QT-10
HPCS15SP-SUR-E?	=.5659*(1-.3001)=.3961	3H1,3H2,3I1	SO1, QT+100 * /FTR1QT-10
MC1E35Y2A-SUR-E?	=.6220	3G-1	SH1, QT+150
ADS18VAL-SUR-E?	=.00125	3J	FTO1, QT+50**18
RUPTURETRB	= .8243		

Therefore:

CRD1 = 0 + 0 + .4177 + .6898 + 0 + .2619 + .6898 + .6220
= .9844

HPCS = .6898 + 0 + 0 + 0 + .3961 + .3961 + .3961
= .9317

HPCS * CRD1 = .6898 + 0 + (0 + 0 + .4177 + .2619 + .6898 + .6220) * (0 + 0 + .3961 + .3961 + .3961)
= .8774

LPCI = .6220

LPCS = .6220

LPCI * LPCS = .6220

ADS = .00125

where sums were combined using P(A + B) = P(A) + P(B) - P(AB).

Table 6.5c
Quantification for Venting:

Types of components and environments:

1) CRD pumps FTR 10 hr, qual - 310 F?, envir - nominal temp (3I2)

2) RBCCW pumps FTR 10 hr, qual - 310 F?, envir - 250 F (3D)

3) Fan motors FTR 10 hr, qual - 355 F 100% hum, envir - nominal temp (3I2)

4 MOV motors FTR 10 hr, qual - 31 F?, envir - 240 F (3G-1), nominal temp (3I1), 240 F (3I1), 250 F (3E).

5) CRD pump CC FTO 10 hr, qual - 185 F?, envir - nominal temp (3I2)

6) RBCCW pump CC FTO 10 hr, qual - 185 F?, envir - 250 F (3D)

7) Fan CC FTO 10 hr, qual - 185 F 95% hum, envir - nominal (3I2), 240 F (3H2).

8) Valve CC SO 10 hr, qual - 185 F?, envir - 250 F (3G-1 and 3F-1), nominal temp (3I1), 240 F (3H1), 250 F (3E), 240 F (3H2).

9) MCC SH 10 hr, qual - 340 F 1hr then 320 F 1hr then 160 F 100 days 95% hum, envir - 250 F (3D), 240 F (3G-1).

10) ADS valves FTO 10 hr, qual - 350 F, envir - 60 psig, 308 F (3J).

11) ADS valves FTO 10 hr, qual - 350 F, envir - 195 psig, 386 F (3J).

12) HPCS pump FTR 10 hr, qual - 310 F, nominal temp (3I2).

Table 6.5c (Concluded)
Quantification for Venting:

From the expert elicitation, median values are:

Event	Value	Location	Quantification
CRDP1FTR-SUR-E?	= 0	3I2	FTR10, nominal
CRDP1CC-SUR-E?	= 0	3I2	FTO10, nominal
PSW175CC-SUR-E?	= .6168*(1-.2510)=.4620	3F-1,3G-1	SO10, QT+75 * /FTR10QT-60
SWVY02CC-SUR-E?	= .7976	3I2,3H2	FTO10, QT+50
SWVY02LF-SUR-E	= 0	3I2	FTR10, nominal
2WRP1FTR-SUR-E?	= .2510	3D	FTR10, QT-60
2WRP1CC-SUR-E?	= .7976	3D	FTO10, QT+80
[illegible]RMCC1-SUR-E?	= .7015	3D	SH10, QT+100
HPCSPFTR-SUR-E?	=0	3I2	FTR10, nominal
HPCS01SP-SUR-E?	=0	3I1	SO10, nominal
[illegible]PCS23SP-SUR-E?	=.5381*(1-.2510)=.4030	3H1	SO10, QT+50 * /FTR10QT-60
HPCS04SP-SUR-E?	=.6168*(1-.2510)=.4620	3E	SO10, QT+75 * /FTR10QT-60
HPCS15SP-SUR-E?	=.5381	3H1,3H2,3I1	SO10, QT+50 * 1.0
MC1E35Y2A-SUR-E?	=.6168	3G-1	SH10, QT+75
ADS18VAL-SUR-E?	=0	3J	FTO10, QT-50**18

Therefore:

CRD1 = 0 + 0 + .4620 + .7976 + 0 + .2510 + .7976 + .7015
= .9951
HPCS = .7976 + 0 + 0 + 0 + .4030 +.4620 + .5381
= .9700
CRD1 + HPCS = .7976 + 0 + (0 + 0 + .4620 + .2510 + .7976 + .7015) * (0 + 0 + .403) + .4620)
= .9316
LPCI = .6168
LPCS = .6163
LPCS + LPCI = .6168
ADS = 0

where sums were combined using P(A + B) = P(A) + P(B) - P(AB).

Table 6.6
Summary of Severe Environments:

Component	Severe Environments	Calculational Formula
PSW175CC-SUR-E? =	SO10, QT+115 * /FTR10, QT+15	= E6* (1-E18)
SWVY02CC-SUR-E? =	FTO10, .58*QT-75+.42*QT+50	= .58*E22 + .42*E23
2WRP1FTR-SUR-E? =	FTR10, QT-60	= E16
2WRP1CC-SUR-E? =	FTO10, QT+80	= E24
2WRMCC1-SUR-E? =	SH10, QT+100	= E10
HPCS04SP-SUR-E? =	SO10, QT+100 * /FTR10, QT-35	= E5*(1-E17)
HPCS15SP-SUR-E? =	SO10, .58*QT-7[illegible].42*QT+50)* /FTR10,QT-210	= (.58*E2+.42*E3)*(1- E1[illegible]
MC1[illegible]5Y?A-SUR-E?=	SH10, QT+150	= E11
ADS18VAL-SUR-E? =	FTO10, QT+50**18	= E23**18

Quantification for Ruptures:

PSW175CC-SUR-E? =	SO1, QT+100 * /FTR1,QT+35	= E1*(1-E14)
SWVY02CC-SUR-E? =	FTO1, QT+115	= E21
2WRP1FTR-SUR-E? =	FTR1, QT-35	= E12
2WRP1CC-SUR-E? =	FTO1, QT+100	= E20
2WRMCC1-SUR-E? =	SH1, QT+125	= E7
HPCS23SP-SUR-E? =	SO1, QT+100 * /FTR1,QT-10	= E1*(1-E13)
HPCS04SP-SUR-E? =	SO1, QT+100 * /FTR1,QT-10	= E1*(1-E13)
HPCS15SP-SUR-E? =	SO1, QT+100 * /FTR1,QT-10	= E1*(1-E13)
MC1E35Y2A-SUR-E?=	SH1, QT+150	= E8
ADS18VAL-SUR-E? =	FTO1, QT+50**18	= E19**18

Quantification for Venting:

PSW175CC-SUR-E? =	SO10, QT+75 * /FTR10,QT-60	= E4*(1-E16)
SWVY02CC-SUR-E? =	FTO10, QT+50	= E23
2WRP1FTR-SUR-E? =	FTR10, QT-60	= E16
2WRP1CC-SUR-E? =	FTO10, QT+80	= E24
2WRMCC1-SUR-E? =	SH10, QT+100	= E10
HPCS23SP-SUR-E? =	SO10, QT+50 * /FTR10,QT-60	= E3*(1-E16)
HPCS04SP-SUR-E? =	SO10, QT+75 * /FTR10,QT-60	= E4*(1-E16)
HPCS15SP-SUR-E? =	SO10, QT+50	= E3
MC1E35Y2A-SUR-E?=	SH10, QT+75	= E9

Table 6.7
Final Collapsed List of Severe Environments:

Name	Description	Where Found
E1	= SO1, QT+100	NEW STUFF SO1 #5
E2	= SO10, QT-75	NA IDENTICALLY ZERO
E3	= SO10, QT+50	NEW STUFF SO10 #3
E4	= SO10, QT+75	NEW STUFF SO10 #4
E5	= SO10, QT+100	NEW STUFF SO10 #5
E6	= SO10, QT+125	NEW STUFF SO10 #6 (+125)
E7	= SH1, QT+125	NEW STUFF SH1 #6
E8	= SH1, QT+150	NEW STUFF SH1 #7
E9	= SH10, QT+75	NEW STUFF SH10 #4
E10	= SH10, QT+100	NEW STUFF SH10 #5
E11	= SH10, QT+150	NEW STUFF SH10 #7
E12	= FTR1, QT-35	NEW STUFF FTR1 #4
E13	= FTR1, QT-10	NEW STUFF FTR1 #5
E14	= FTR1, QT+35	NEW STUFF FTR1 #7 (+40)
E15	= FTR10, QT-210	NA IDENTICALLY ZERO
E16	= FTR10, QT-60	NEW STUFF FTR10 #3
E17	= FTR10, QT-35	NEW STUFF FTR10 #4
E18	= FTR10, QT+15	NEW STUFF FTR10 #6
E19	= FTO1, QT+50	NEW STUFF FTO1 #9
E20	= FTO1, QT+100	NEW STUFF FTO1 #9 (+50)
E21	= FTO1, QT+115	NEW STUFF FTO1 #9 (+50)
E22	= FTO10, QT-75	NEW STUFF FTO10 #4
E23	= FTO10, QT+50	NEW STUFF FTO10 #9
E24	= FTO10, QT+80	NEW STUFF FTO10 #9 (+50)

probability was calculated. Since the failure probabilities were mostly above 0.1, the small value approximation could not be used and exact results were calculated. For example, for the CRD system with one train operating (CRD1) and containment leak, the Boolean equation is:

CRD1 = (CRDP1FTR-SUR-E? + CRDP1CC-SUR-E? + PSW175CC-SUR-E?
+ SWVY02CC-SUR-E? + SWVY02LF-SUR-E + 2WRP1FTR-SUR-E?
+ 2WRP1CC-SUR-E? + 2WRMCC1-SUR-E?).

Substituting in the above equation the failure probabili- .es for each event from step 4:

CRD1 = 0.0 + 0.0 + 0.4168 + 0.5517 + 0.0 +
0.2510 +0.7976 + 0.7016
= .9882

where sums were combined using P(A + B) = P(A) + P(B) - P(AB). For this particular system and case, the severe environment failure probability is almost 1.0 and not much was gained; however, for other systems and cases, the values ranged from no additional failure probability to 1.0 depending upon the type and location of the equipment, the environment, and the system design. Table 6.8 contains a description of the simplified equation used for each system.

Because the systems that are available to respond to the survival question are potentially different for each cut set (i.e., combination of component failures that can result in a particular accident sequence), a Boolean equation was constructed for each possible combination based upon an examination of the cut sets of the sequences. Each unique combination is defined to be a different survival event. If we consider a cut set for which only CRD can operate then:

SUR-001-L = CRD1 * LEAKTRB = .9882 * .2667 = .2636

where SUR-001-L is the probability of not being able to use CRD if containment failed in a leak mode and the resulting severe environment failed CRD, since one needs a leak to the reactor building in order to get any severe environment failure. We see that, for this example, the conditional probability of a leak to the reactor building instead of the refueling floor is very important in determining the final amount of recovery credit that can be given.

The definition of each survival event used in the analysis is given in Table 6.9 in terms of system successes. In Table 6.10, the failure equation and an estimate of the probability using the median values for the severe environment failures and the containment failure is given. The system equations were substituted into the survival event definitions and the Boolean expressions were simplified and then converted to probability equations. Because some of the probabilities are large (i.e., greater than 0.1), exact expressions were developed to calculate the probabilities (the

Table 6.8
System Models

ADS System

The ADS system has only its main valves and their solenoid valves in the primary containment. All other components which are necessary for manual operation are in the auxiliary building and not subject to harsh environments. There is a nitrogen bottle station also in the auxiliary building so indefinite operation can be sustained.

The following simplified equation is therefore adequate to represent ADS failure in harsh environments:

ADS-SUR = ADS18VAL-SUR-E?.

That is; all 18 SRVs must fail in order to fail ADS. This must be evaluated for venting at 60 psig or containment failure at about 195 psig.

Table 6.8 (Continued)
System Models

CL System

The components in the condensate system are located throughout the turbine building. Even if the steam gets into the turbine building, for this system to be operating the building HVAC will likely be working since all power will be available. The expected environments for the lower levels into which outside air is being forced, should be mild. Therefore, only random failure is expected. However, since PCS has failed, makeup from the CST is needed for continued operation. The limiting random failure is failure of IA supply to the makeup valves. By examining the system cut sets this ran[illegible]ilure is 6.9E-2. Failure of any compressor will result in sufficient [illegible]on to close the valve. Because of the long time before [illegible], the spare can be easily started. This means that th[illegible]ility is dominated by operator failure to maintain contin[illegible]. There are several single random failures in the low 1E-3 range then more in the 5E-5 range. Therefore, CDS failure can be treated as random and we will use operator failure to be the dominant failure:

CDS-FAIL = OPERFCDSTW = 2.1E-03, Log-normal, EF=10, Group 2 action.

Table 6.8 (Continued)
System Models

CRD System

If the CRD system is already running then the following equation represents the dominant system failures:

CRD2R-SUR = (CRDP1FTR-SUR-E? + CRDP1CC-SUR-E?) * (
CRDP2FTS-SUR-E? + CRDP2FTR-SUR-E? + CRDP2CC-SUR-E?)
+ PSW175CC-SUR-E? * /PSW175LF-SUR-E? +
SWVY02CC-SUR-E? + SWVY02LF-SUR-E? + (2WRP1FTR-SUR-E?
+ 2WRP1CC-SUR-E? + 2WRP1CB-SUR-E? + 1E34XBMCC-SUR-E?
+ 1ET34XBTR-SUR-E? + 1EB234BBK-SUR-E?) * (
1E233TXTR-SUR-E? + 1EB233ABK-SUR-E? + 2WRP2FTS-SUR-E?
+ 2WRP2FTR-SUR-E? + 2WRP2CC-SUR-E?).

The equation we used to approximate the system probability for all cut sets where at least one train of CRD is working was:

For Leaks:

CRD1R-SUR = (CRDP1FTR-SUR-E? + CRDP1CC-SUR-E? + PSW175CC-SUR-E?
+ SWVY02CC-SUR-E? + SWVY02LF-SUR-E + 2WRP1FTR-SUR-E?
+ 2WRP1CC-SUR-E? + 2WRMCC1-SUR-E?) * LEAKTRB.

For Ruptures:

CRD1R-SUR = (CRDP1FTR-SUR-E? + CRDP1CC-SUR-E? + PSW175CC-SUR-E?
+ SWVY02CC-SUR-E? + SWVY02LF-SUR-E + 2WRP1FTR-SUR-E?
+ 2WRP1CC-SUR-E? + 2WRMCC1-SUR-E?) * RUPTURETRB.

For Venting:

CRD1R-SUR = CRDP1FTR-SUR-E? + CRDP1CC-SUR-E? + PSW175CC-SUR-E?
+ SWVY02CC-SUR-E? + SWVY02LF-SUR-E + 2WRP1FTR-SUR-E?
+ 2WRP1CC-SUR-E? + 2WRMCC1-SUR-E?.

The only components in the reactor building needed to operate in this mode (one pump minimum flow at > x hrs) are the pumps and their control circuits, the service water supply MOV which could spuriously close, the room cooling fan control circuit, the RBCCW pumps and their control circuits, and the electrical power support through an MCC, transformer, and circuit breaker (the transformer and circuit breaker are in the MCC). All other components are: (1) in the auxiliary or DC building and not subject to harsh environments, (2) in the main turbine building area where some mild environmental changes are expected, or (3) in the HPCS room (5D2) which is isolated from any harsh environment. The CRD pumps are in 3I2, the CRD control circuits are in 3I2, the fan control circuits are in 3H1 and 3H2, the service water valve is in 3G-1 with CC in 3G-1 and 3F-1, the RBCCW pumps and MCC electrical support are in 3D.

Table 6.8 (Continued)
System Models

Firewater System

This system is in the turbine building and will be dominated by operator failure to start. Success is unlikely unless started before containment failure since some steam will be in the turbine building after containment failure. There are no active valves or other support systems needed; so failure is dominated by operator failure to align. The following equation can be used to quantify DDFW:

DDFW-FAILS = OPERFDDFW = 0.12, log-normal, EF=7.8, Group 10 action.

Table 6.8 (Continued)
System Models

HPCS System

Since the HPCS system is a single train system, it can be represented by the following equation:

For Leaks:

HPCS-SUR = (SWVY02LF-SUR-E? + SWVY02CC-SUR-E? + HPCSPFTR-SUR-E?
+ HPCS01SP-SUR-E? + HPCS23SP-SUR-E? + HPCS04SP-SUR-E?
+ HPCS15SP-SUR-E?) * LEAKTRB.

For Ruptures:

HPCS-SUR = (SWVY02LF-SUR-E? + SWVY02CC-SUR-E? + HPCSPFTR-SUR-E?
+ HPCS01SP-SUR-E? + HPCS23SP-SUR-E? + HPCS04SP-SUR-E?
+ HPCS15SP-SUR-E?) * RUPTURETRB.

For Venting:

HPCS-SUR = SWVY02LF-SUR-E? + SWVY02CC-SUR-E? + HPCSPFTR-SUR-E?
+ HPCS01SP-SUR-E? + HPCS23SP-SUR-E? + HPCS04SP-SUR-E?
+ HPCS15SP-SUR-E?.

The fan is in its own housing and sees the environment in the HPCS room 312 and 3H2 (This is the same fan as for CRD), valve 23 and control circuit are in 3H1, valve 1 and control circuit are in 311, the pump is in 312, valve 4 and control circuit are in 3E, and valve 15 is in 311 with its control circuit in 311, 3H1, and 3H2.

Table 6.8 (Continued)
System Models

LPCI System

The LPCI system is a three train system in which two trains are in the same room. Since the rooms are normally isolated except in the rupture case, the most susceptible components are the room HVAC power supplies on the 710' level of the reactor building (even in the case of a rupture, these are still the most limiting). The system failure can therefore be represented by the following equation:

LPCI-SUR = MCC1E35Y2A-SUR-E? * MCC1E36Y1B-SUR-E?.

The two MCCs are identical and can be said to be completely correlated. The environment they see will also be the same. As a result, we approximated the system failure by using only one of the MCCs and said that the second failed if the first failed. For most of the dominant cases this was also reasonable since only one of the trains was operating due to partial loss of AC power or random failure of the other train. The MCC used is the same MCC that powers LPCS.

For Leaks:

LPCI-SUR = MCC1E35Y2A-SUR-E? * LEAKTRB.

For Ruptures:

LPCI-SUR = MCC1E35Y2A-SUR-E? * RUPTURETRB.

For Venting:

LPCI-SUR = MCC1E35Y2A-SUR-E?.

For failure due to valve cycling only:

LPCI-SUR = LPCIC

Table 6.8 (Continued)
System Models

LPCS System

Since the LPCS system is a single train system, it can be represented by the following equation:

LPCS-SUR = NEHVACF-SUR-E? + NEHVACFCC-SUR-E? + NEHVACBR-SUR-E? +
LPCS01SP-SUR-E? + LPCS12SP-SUR-E? + LPCSPFTR-SUR-E? +
LPCSPCC-SUR-E? + LPCS05FTO-SUR-E? + CSCS35SP-SUR-E? +
MCC1E35Y2A-SUR-E? + CSCSBR-SUR-E? + LPCSN413-SUR-E?.

However, the LPCI and LPCS systems limiting components are the MCCs on the 710' level in the reactor building and these systems can be approximated by only one term (we used the common MCC for LPCS and train A of LPCI, see discussion under LPCI):

For Leaks:

LPCS-SUR = MCC1E35Y2A-SUR-E? * LEAKTRB.

For Ruptures:

LPCS-SUR = MCC1E35Y2A-SUR-E? * RUPTURETRB.

For Venting:

LPCS-SUR = MCC1E35Y2A-SUR-E?.

Table 6.8 (Concluded)
System Models

MFW System

As with the CDS system, the main feedwater system does not have important components in the reactor building. The main components are in the turbine building where the environments are expected to be fairly mild and not result in a significant increase over the random failure rates. For this analysis, main feedwater failure was conservatively estimated as 1E-02 and was only used in ATWS sequences since, for non-ATWS transients, if feedwater is working no core damage results.

Table 6.9a
Basic Event Name of Survival Question
for Containment Leak Sequences

Basic Event Name	Equipment Available
SUR-001-L	CRD1
SUR-002-L	HPCS
SUR-003-L	HPCS + CRD1
SUR-004-L	CDS
SUR-005-L	HPCS + CRD1 + CDS
SUR-006-L	HPCS + CDS

Table 6.9b
Basic Event Name of Survival Question
for Containment Venting Sequences

Basic Event Name	Equipment Available
SUR-001-V	CRD1 + ADS * (DDFW + CDS)
SUR-002-V	CRD1 + ADS * (DDFW + CDS + LPCI)
SUR-003-V	CRD1 + ADS * (DDFW + CDS + LPCS)
SUR-004-V	CRD1 + ADS * (DDFW + CDS + LPCI + LPCS)
SUR-005-V	HPCS + ADS * (DDFW + CDS)
SUR-006-V	HPCS + ADS * DDFW
SUR-007-V	HPCS + DDFW
SUR-008-V	HPCS + DDFW + CDS

Table 6.9c
Basic Event Name of Survival Question
for Containment Rupture Sequences

Basic Event Name	Equipment Available
SUR-001-R	CRD1 + LPCS + DDFW
SUR-002-R	LPCS + DDFW
SUR-003-R	CRD1 + LPCI + DDFW
SUR-004-R	CRD1 + LPCI + LPCS + DDFW
SUR-005-R	LPCI + DDFW
SUR-006-R	LPCI + LPCS + DDFW
SUR-007-R	CRD1 + CDS + LPCI + DDFW
SUR-008-R	CRD1 + CDS + LPCS + DDFW
SUR-009-R	CRD1 + CDS + LPCS + LPCS + DDFW
SUR-010-R	CDS + DDFW
SUR-011-R	CDS + LPCI + DDFW
SUR-012-R	CDS + LPCS + DDFW
SUR-013-R	CDS + LPCI + LPCS + DDFW
SUR-014-R	CRD1 + ADS * (DDFW + CDS)
SUR-015-R	CRD1 + ADS * (DDFW + CDS + LPCI)
SUR-016-R	CRD1 + ADS * (DDFW + CDS + LPCS)
SUR-017-R	CRD1 + ADS * (DDFW + CDS + LPCI + LPCS)
SUR-018-R	CRD1 + ADS * (DDFW + LPCI + LPCS)
SUR-019-R	CRD1 + ADS * (DDFW + LPCI)
SUR-020-R	CRD1 + ADS * (DDFW + LPCS)
SUR-021-R	HPCS + CRD1 + ADS * (DDFW + CDS)
SUR-022-R	HPCS + CRD1 + ADS * DDFW
SUR-023-R	HPCS + ADS * (DDFW + CDS)
SUR-024-R	HPCS + ADS * DDFW
SUR-025-R	HPCS + CRD1 + DDFW
SUR-026-R	HPCS + CRD1 + CDS + DDFW
SUR-027-R	HPCS + DDFW
SUR-028-R	HPCS + DDFW + CDS
SUR-029-R	CRD1 + CDS + DDFW

Table 6.9d
Basic Event Name cf Survival Question
for Containment Failure in ATWS Sequences

Basic Event Name	Equipment Available
SUR-001-A-L	MFW + HPCS
SUR-002-A-L	HPCS
SUR-001-A-R	HPCS + ADS * (LPCS * LPCI)
SUR-001-A-V	HPCS + ADS * (LPCS * LPCI)
SUR-002-A-V	ADS * (LPCS + LPCI)
SUR-001-A-C	LPCIC

Table 6.10a
Failure Equations for Survival Events for Leaks

```
SUR-001-L   = CRD1 = .9882 * .2667 = .2636
SUR-002-L   = HPCS = .8210 * .2667 = .2190
SUR-003-L   = HPCS * CRD1 = .8139 * .2667 = .2171
SUR-004-L   = CDS  = 2.1E-3 * .2667 = 5.6E-4
SUR-005-L   = HPCS * CRD1 * CDS = .2171 * 2.1E-3 = 4.559E-4
SUR-006-L   = HPCS * CDS = .2190 * 2.1E-3 = 4.599E-4
```

Table 6.10b
Failure Equations for Survival Events for Ruptures

SUR-001-R	= CRD1 * LPCS * DDFW = .9844 * .8243 *.6220 * .12 = .0651
SUR-003-R	= CRD1 * LPCI * DDFW = .0651
SUR-004-R	= CRD1 * LPCI * LPCS * DDFW = .0651
SUR-007-R	= CRD1 * CDS * LPCI * DDFW = .9844 * .6220 * .8243 * 2.1E-3 = 1.054E-3
SUR-008-R	= CRD1 * CDS * LPCI * DDFW = 1.054E-3
SUR-009-R	= CRD1 * CDS * LPCI * LPCS * DDFW = 1.054E-3
SUR-010-R	= CDS * DDFW = 2.1E-3 * .8243 = 1.718E-3
SUR-014-R	= CRD1 * (ADS + CDS * DDFW) = .9844 * .8243 * (.00125 + 2.1E-3) = 2.733E-3
SUR-015-R	= CRD1 * (ADS + CDS * DDFW * LPCI) = .9844 *.8243 * (.00125 + 2.1E-3 * .6220) = 2.147E-3
SUR-016-R	= CRD1 * (ADS + CDS * DDFW * LPCS) = 2.147E-3
SUR-017-R	= CRD1 * (ADS + CDS * DDFW * LPCS * LPCI) = 2.147E-3
SUR-018-R	= CRD1 * (ADS + DDFW * LPCI * LPCS) = .9844 *.8243 * (.00125 + .12 * .6220) = 6.246E-2
SUR-019-R	= CRD1 * (ADS + DDFW * LPCI) = 6.246E-2
SUR-020-R	= CRD1 * (ADS + DDFW * LPCS) = 6.246E-2
SUR-021-R	= HPCS * CRD1 * (ADS + DDFW * CDS) = .8774 *.8243 * (.00125 + 2.1E-3) = 2.342E-3
SUR-022-R	= HPCS * CRD1 * (ADS + DDFW) = .8774 * .8243 * (.00125 + .12) = 8.784E-2
SUR-023-R	= HPCS * (ADS + DDFW * CDS) = .9317 * .8243 * (.00125 + 2.1E-3) = 5.856E-3
SUR-024-R	= HPCS * (ADS + DDFW) = .9317 * .8243 * (.00125 + .12) = 9.369E-2
SUR-025-R	= HPCS * CRD1 * DDFW = .9317 * .9844 * .8243 * .12 = 8.979E-2
SUR-026-R	= HPCS * CRD1 * DDFW * CDS = .9317 * .9844 *.8243 * 2.1E-3 = 1.581E-4
SUR-027-R	= HPCS * DDFW = .9317 * .8243 * .12 = 9.174E-2
SUR-028-R	= HPCS * DDFW * CDS = .9317 * .8243 * 2.1E-3 = 1.620E-4
SUR-029-R	= CRD1 * CDS * DDFW = .9844 * .8243 * 2.1E-3 = 1.698E-4

Table 6.10c
Failure Equations for Survival Events for Venting

SUR-001-V	= CRD1 * (ADS + DDFW * CDS) = .9951 * (0 + 2.1E-3) = 2.1E-3
SUR-002-V	= CRD1 * (ADS + DDFW * CDS * LPCI) = .9951 * (0 + 2.1E-3 * .6168) = 1.3E-3
SUR-003-V	= CRD1 * (ADS + DDFW * CDS * LPCI) = 1.3E-3
SUR-004-V	= CRD1 * (ADS + DDFW * CDS * LPCI * LPCS) = 1.3E-3
SUR-005-V	= HPCS * (ADS + DDFW * CDS) = .9700 * (0 + 2.1E-3) = 2.0E-3
SUR-006-V	= HPCS * (ADS + DDFW) = .9700 * (0 + .12) = 1.2E-1
SUR-007-V	= HPCS * DDFW = .9700 * .12 = 1.2E-1
SUR-008-V	= HPCS * DDFW * CDS = .9700 * 2.1E-3 = 2.0E-3

Table 6.10d
Failure Equations for Survival Events for ATWS

SUR-001-A-L = MFW * HPCS = .01 * .8210 * .2667 = 2.2E-03
SUR-002-A-L = HPCS = .8210 * .2667 = .2190
SUR-001-A-R = HPCS * (ADS + LPCS * LPCI) = .9317 * (1.25E-03 + .6220)
* .8243 = .4787
SUR-001-A-V = HPCS * (ADS + LPCS * LPCI) = .9700 * (0 + .6168) = .5983
SUR-002-A-V = ADS + LPCS * LPCI = 0 + .6168 = .6168
SUR-001-A-C = LPCIC = .[illegible]

equations used to calculate the probabilities can be found in Appendix D of Volume 2 of this report in the LHS extender code listing).

6.2.6 St . 6: Resolve Core Vulnerable Sequences.

The venting system fault tree was evaluated and its success and failure were "ande l" to each relevant sequence. This resulted in two sequences, one in which venting was a success and the other in which venting had failed. For the sequence in which venting had failed, two events representing containment failure by leak and rupture ere added and two sequences were created. In one, the containment failed by leak and, in the other, the containment failed by rupture. For each of the three sequences thus created (as defined in the step 1 discussion above), the individual cut sets were examined and the appropriate survival event from Table 6.9, representing the systems which were available to respond, was chosen and added to the cut set. The sequence was then quantified.

The use of the above method gives a point estimate of the Level I core damage frequency. To get an evaluation of the uncertainty, a Latin Hypercube sample (i.e., a stratified Monte Carlo sample) must be formed using the distributions not only for the component failure data commonly used but also for the containment failure locations and modes, the equipment failure probabilities in severe environments, and, if there was a large uncertainty in the environments, the ranges of the environments. The sequences with the added venting, containment failure, and survival events were then evaluated multiple times using this Latin Hypercube sample and the TEMAC[7] code. The result is an uncertainty distribution for each sequence and/or the total core damage frequency that incorporates the uncertainty not only of the random failure distributions for the basic component failures, but also of the uncertainty in phenomena and equipment response to these phenomena, see Chapter 7 of this report and the integrated results presented in Volume 2 of this report.

6.3 Conclusions

The use of expert judgement based upon various supporting calculations allowed us to resolve core vulnerable sequences in a much more realistic and less conservative way than by simply assuming failure.

By application of a systematic procedure, the underlying parameters or processes contributing to the uncertainty in the issue were delineated much more explicitly than has been done in the past. The underlying assumptions and expert judgement that were used to quantify the issue for the PRA are delineated such that people wanting to review the PRA or use it can clearly understand the limitations and areas of applicability.

The uncertainty in the current state-of-knowledge in both PRA modeling and thermal-hydraulic analyses was explicitly incorporated into the PRA, and its importance to the final results calculated.

A simplified version of this methodology was used in the NUREG-1150 analysis of the Peach Bottom plant.

6.4 Interface With Level II/III Analysis

Because the Level II/III analysis evaluates the possibility of containment failure and uses the location, size, and time of failure to calculate the radioactive release to the environment; the same containment failure modes must be used in order to calculate the containment and system failure probabilities in the Level I analysis and pass these values in a consistent fashion to the Level II/III analysis. In order to maintain consistency, for each Level I sample, we will sample the containment failure pressure (even though this is certain for the Level I sequences of interest, the actual failure pressure will impact the mode of containment failure and will also be used in the Level II analysis for sequences which had core damage in the Level I analysis but do not have containment failure until later). The same user function (FORTRAN subroutine used in the accident progression event tree, APET, in the Level II analysis to calculate the containment failure mode) used in the APET was used in the Level I analysis. For the Level I analysis only slow pressurization cases are important (for the Level II analysis, explosive or very rapid pressure increases are also possible). The result is that, for each sample, the same containment failure pressure, location, and size and the same environmental failure probability for the system components and survival events is used in both the Level I and Level II analyses.

6.5 References

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7.0 RESULTS OF THE INTERNAL EVENTS ANALYSIS

7.1 Dominant Sequences

Fifty-four sequences survived the initial screening process described in Chapters 3 and 4. For each of these sequences, the cut sets were individually examined and the appropriate recovery and survival events were determined and added to the cut sets as described in Chapters 5 and 6. The basic event data was reviewed and modified as described in Volume 5 of this report and the sequences were requantified. No sequences or cut sets that survived the initial screening process were truncated for this final quantification. If some cut sets were determined to be unphysical, an event RA-DELETE was added to them so that the cut set frequency would be zero. The cut set remained in the cut set file so that the final disposition of all cut sets could be traced.

Table 7.1 lists all of the sequences that survived the screening process. The sequences are ordered from most dominant to least dominant as determined by the mean value from the TEMAC[1] calculation. Also shown are the 5th percentile, median, 95th percentile, the point estimate, the fractional contribution to the total internal core damage frequency, and the cumulative contribution to the total internal core damage frequency. The last two rows show the algebraic sum of each column and the results of the integrated evaluation.

The mean core damage frequency for internal events is 4.41E-05/R-yr. for the LaSalle plant. The lower 5th percentile = 2.05E-06/R-yr., the median = 1.64E-05/R-yr., and the 95th percentile = 1.39E-04/R-yr. The mean core damage frequency is low considering that this is the first time a PRA has been performed on the plant. Typical core damage frequencies obtained in the past for first time PRAs have been in the low 1.0E-4/R-yr. range. This is usually due to the identification of some design and construction errors that result in a loss of redundancy and some core damage sequences with high frequencies of occurrence. The LaSalle plant, being a modern BWR design, has highly redundant and independent systems which tends to ameliorate these types of problems. While some design faults were found in the analysis, none were of sufficient severity to result in sequences with high core damage frequencies.

The dominant sequence is T100 which contributes 64.1% of the core damage frequency from internal events. In this sequence, we have a transient initiator followed by successful scram and SRV operation. All high and low pressure injection systems fail and core damage ensues. The cut sets fall into two groups: (1) an early core damage scenario where all AC is lost initially and reactor core isolation cooling (RCIC) fails and (2) a late core damage scenario where AC works for a while and then fails. For the late scenario we have about 10 hours for recovery actions to be completed. For the early scenario we have about 80 minutes.

Table 7.1
LaSalle Final Sequence Core Damage Statistics: Internal Events

Sequence	5%	Median	Mean	95%	Point Est	Frac of Int	Cum Int
T100	1.1000E-06	9.0700E-06	2.8700E-05	9.7400E-05	2.1400E-05	6.4144E-01	6.4144E-01
T62	3.1800E-07	2.3900E-06	6.5300E-06	2.4100E-05	4.7700E-06	1.4594E-01	7.8739E-01
T18	0.0000E+00	0.0000E+00	4.9900E-06	2.0900E-05	2.7100E-06	1.1153E-01	8.9891E-01
T20	0.0000E+00	0.0000E+00	1.2800E-06	8.0300E-06	4.8100E-07	2.8608E-02	9.2752E-01
T22	0.0000E+00	0.0000E+00	1.1400E-06	3.9900E-06	5.5700E-07	2.5479E-02	9.5300E-01
T16	1.5500E-08	1.4200E-07	4.3600E-07	1.7500E-06	4.8100E-07	9.7[illegible]45E-03	9.6274E-01
T101	4.1300E-09	6.4500E-08	2.4800E-07	1.0100E-06	2.7600E-07	5.5428E-03	9.6829E-01
T24	0.0000E+00	0.0000E+G	2.2600E-07	1.2100E-06	1.0400E-07	5.0511E-03	9.7334E-01
TL12	2.1500E-09	3.8100E-08	2.1000E-07	7.6400E-07	2.1100E-07	4.6935E-03	9.7803E-01
TL97	2.7600E-09	4.0100E-08	1.9400E-07	5.6200E-07	1.3200E-07	4.3359E-03	9.8237E-01
T38	0.0000E+00	0.0000E+00	1.3500E-07	8.4100E-08	2.3300E-08	3.0172E-03	9.8538E-01
A49	3.9800E-10	1.1100E-08	8.9400E-08	2.4800E-07	5.5200E-08	1.9981E-03	9.8738E-01
A174	5.7600E-10	1.0900E-08	7.9800E-08	1.7600E-07	4.7400E-08	1.7835E-03	9.8916E-01
T50	1.2200E-10	4.2400E-09	7.4200E-08	1.7900E-07	2.8100E-08	1.6584E-03	9.9082E-01
TL59	7.8400E-10	9.2400E-09	5.9000E-08	1.7200E-07	3.1900E-08	1.3186E-03	9.9214E-01
[illegible]	0.0000E+00	0.0000E+00	5.7400E-08	3.5200E-08	1.1300E-08	1.2829E-03	9.9342E-01
T85	9.6300E-12	1.2600E-09	5.4800E-08	9.1000E-08	1.4200E-08	1.2248E-03	9.9465E-01
T82	0.0000E+00	0.0000E+00	4.9500E-08	1.8300E-08	5.8000E-09	1.1063E-03	9.9576E-01
T47	0.0000E+00	0.0000E+00	4.9500E-08	1.7000E-08	6.0500E-09	1.1063E-03	9.9686E-01
A123	0.0000E+00	0.0000E+00	3.2000E-08	1.1500E-07	1.5000E-08	7.1520E-04	9.9758E-01
L14	0.0000E+00	0.0000E+00	1.7200E-08	7.3500E-08	1.0600E-08	3.8442E-04	9.9796E-01
T56	0.0000E+00	0.0000E+00	1.5900E-08	2.2000E-09	8.3600E-10	3.5536E-04	9.9832E-01
T59	1.4500E-11	1.2600E-09	1.5800E-08	5.7900E-08	1.3500E-08	3.5313E-04	9.9867E-01
T41	1.3500E-10	3.3400E-09	1.1500E-08	4.7400E-08	1.5700E-08	2.5702E-04	9.9893E-01
L16	0.0000E+00	0.0000E+00	8.9800E-09	2.2400E-08	1.8800E-09	2.0070E-04	9.9913E-01
A126	0.0000E+00	0.0000E+00	8.8800E-09	5.9900E-08	3.9800E-09	1.9847E-04	9.9933E-01
T73	1.1100E-10	2.7300E-09	8.6000E-09	3.2300E-08	1.2100E-09	1.9221E-04	9.9952E-01
TL18	0.0000E+00	0.0000E+00	5.0700E-09	2.9200E-09	1.0700E-09	1.1331E-04	9.9963E-01
TL14	0.0000E+00	0.0000E+00	2.9900E-09	1.8200E-08	3.1100E-09	6.6826E-05	9.9970E-01
A22	9.5100E-13	8.3100E-11	2.6100E-09	8.1400E-09	1.0300E-09	5.8333E-05	9.9976E-01
A52	0.0000E+00	0.0000E+00	2.2200E-09	9.5100E-09	2.1500E-09	4.9617E-05	9.9981E-01
A93	7.3300E-13	7.9500E-11	1.7300E-09	4.5900E-09	8.4200E-10	3.8665E-05	9.9985E-01
A18	0.0000E+00	0.0000E+00	1.4100E-09	6.6700E-09	6.7300E-10	3.1513E-05	9.9988E-01
T49	0.0000E+00	0.0000E+00	1.0600E-09	7.3100E-10	1.4900E-10	2.3691E-05	9.9990E-01
T32	0.0000E+00	0.0000E+00	1.0400E-09	2.6200E-09	8.5500E-10	2.3244E-05	9.9992E-01
T84	0.0000E+00	0.0000E+00	1.0100E-09	7.1100E-10	1.4200E-10	2.2573E-05	9.9995E-01
TL16	0.0000E+00	0.0000E+00	9.0400E-10	4.7300E-09	3.6200E-10	2.0204E-05	9.9997E-01
T34	0.0000E+00	0.0000E+00	6.7700E-10	1.8200E-10	1.2800E-10	1.5131E-05	9.9998E-01
L12	6.8200E-13	2.1100E-11	1.6000E-10	6.7400E-10	1.8200E-10	3.5760E-06	9.9998E-01
A129	4.6000E-13	1.8500E-11	1.5500E-10	6.5900E-10	1.6500E-10	3.4642E-06	9.9999E-01
T58	0.0000E+00	0.0000E+00	1.5000E-10	1.0300E-10	2.0500E-11	3.3525E-06	9.9999E-01
TL20	0.0000E+00	0.0000E+00	9.4600E-11	2.6900E-10	4.2400E-11	2.1143E-06	9.9999E-01

Table 7.1 (Concluded)
LaSalle Final Sequence Core Damage Statistics: Internal Events

Sequence	5%	Median	Mean	95%	Point Est	Frac of Int	Cum Int
A55	0.0000E+00	0.0000E+00	8.0500E-11	2.8100E-10	5.6600E-11	1.7992E-06	1.0000E+00
A58	1.9700E-12	7.8100E-12	6.2100E-11	2.3200E-10	6.8800E-11	1.3879E-06	1.0000E+00
T40	0.0000E+00	0.0000E+00	3.7600E-11	3.2800E-11	1.1900E-11	8.4035E-07	1.0000E+00
T30	2.0500E-15	3.2000E-13	2.9900E-11	3.7900E-11	6.6200E-12	6.6826E-07	1.0000E+00
A15	0.0000E+00	0.0000E+00	2.7400E-11	1.1700E-10	2.0200E-11	6.1239E-07	1.0000E+00
T72	0.0000E+00	0.0000E+00	1.6300E-11	1.5000E-11	5.7600E-12	3.6430E-07	1.0000E+00
TL36	0.0000E+00	5.9000E-13	1.2500E-11	4.6100E-11	1.8600E-11	2.7937E-07	1.0000E+00
TL38	0.0000E+00	0.0000E+00	2.7100E-12	1.2600E-11	6.2000E-12	6.0568E-08	1.0000E+00
L18	0.0000E+00	0.0000E+00	1.7000E-12	1.0300E-12	4.4400E-13	3.7995E-08	1.0000E+00
L20	0.0000E+00	0.0000E+00	7.1800E-13	5.0900E-13	8.6000E-14	1.6047E-08	1.0000E+00
A148	5.6100E-16	4.2400E-14	6.5500E-13	3.0600E-12	1.4800E-12	1.4639E-08	1.0000E+00
A132	7.0400E-20	2.2500E-17	8.0800E-15	1.0100E-14	2.9400E-15	1.8059E-10	1.0000E+00
SUM	1.4447E-06	1.1789E-05	4.4743E-05	1.6121E-04	3.1430E-05	NA	NA
INTEGRATED	2.0500E-06	1.6400E-05	4.4100E-05	1.3900E-04	3.1400E-05	1.0000E+00	NA

The second most dominant sequence is T62 which contributes 14.6% of the core damage frequency from internal events. In this sequence, we have a transient initiator followed by successful scram and safety relief valve (SRV) operation. All high pressure injection except RCIC fails and containment and primary system heat removal fail. The automatic depressurization (ADS) system works but the low pressure systems are failed. The overall time available to the operators to perform their recovery actions is approximately 2 hours. In some cases (e.g., restoring offsite power when a diesel generator (DG) has run for some period of time) more time is available. The amount of time available depends on the failures that constitute the cut set and what recovery action is being considered.

The third most dominant sequence is T18 which contributes 11.1% of the core damage frequency from internal events. In this sequence, we have a transient initiator followed by successful scram and SRV operation. The main feedwater (MFW) system fails but high pressure core spray (HPCS) and one train of the control rod drive (CRD) system work providing high pressure injection. The normal containment and primary heat removal systems fail, and venting fails. Containment pressure increases until a leak develops. Depending upon its location, this leak will produce an environment which could cause injection systems that are operating or that may be able to operate to fail. The overall time available to the operators to perform their recovery actions is approximately 27 hours. In some cases (e.g., venting) less time is available. The amount of time available depends on the failures that constitute the cut set and what recovery action is being considered.

The fourth most dominant sequence is T20 which contributes 2.9% of the core damage frequency from internal events. In this sequence, we have a transient initiator followed by successful scram and SRV operation. The main feedwater system fails but HPCS and one train of the CRD system work providing high pressure injection. The normal containment and primary heat removal systems fail, and venting fails. Containment pressure increases until rupture occurs. Depending upon its location, this rupture will produce an environment which could cause injection systems that are operating or that may be able to operate to fail. The overall time available to the operators to perform their recovery actions is approximately 27 hours. In some cases (e.g., venting) less time is available. The amount of time available depends on the failures that constitute the cut set and what recovery action is being considered.

The fifth most dominant sequence is T22 which contributes 2.5% of the core damage frequency from internal events. In this sequence, we have a transient initiator followed by successful scram and SRV operation. The main feedwater system and the CRD system fail but the HPCS system works providing high pressure injection. The normal containment and primary heat removal systems fail, and venting fails. Containment pressure increases until a leak develops. Depending upon its location, this leak will produce an environment which could cause injection systems that are operating or that may be able to operate to fail. The overall time available to the operators to perform their recovery actions is approximately 27 hours. In some cases (e.g., venting) less time is available. The amount of time available depends on the failures that constitute the cut set and what recovery action is being considered.

The sixth most dominant sequence is T16 which contributes 0.97% of the core damage frequency from internal events. In this sequence, we have a transient initiator followed by successful scram and SRV operation. The main feedwater system fails but HPCS works providing high pressure injection. The normal containment and primary heat removal systems fail, but the operators are able to vent. Successful venting produces an environment which may cause injection systems that are operating or that may be able to operate to fail. The overall time available to the operators to perform their recovery actions is approximately 27 hours. In some cases (e.g., performing venting) less time is available. The amount of time available depends on the failures that constitute the cut set and what recovery action is being considered.

The seventh most dominant sequence is T101 which contributes 0.55% of the core damage frequency from internal events. In this sequence, we have a transient initiator followed by successful scram and SRV operation. All high pressure injection fails and ADS fails so low pressure systems are not available. Core damage begins in about 80 minutes.

The eighth most dominant sequence is T24 which contributes 0.50% of the core damage frequency from internal events. In this sequence, we have a transient initiator followed by successful scram and SRV operation. The main feedwater system and the CRD system fail but the HPCS system works providing high pressure injection. The normal containment and primary heat removal systems fail, and venting fails. Containment pressure increases until a rupture occurs. Depending upon its location, this rupture will produce an environment which could cause injection systems that are operating or that may be able to operate to fail. The overall time available to the operators to perform their recovery actions is approximately 27 hours. In some cases (e.g., venting) less time is available. The amount of time available depends on the failures that constitute the cut set and what recovery action is being considered.

The ninth most dominant sequence is TL12 which contributes 0.47% of the core damage frequency from internal events. In this sequence, we have a transient initiator followed by a successful scram. The SRVs open but one or more fail to reclose when required (i.e., they stick open), resulting in a transient-induced LOCA. The main feedwater system fails but HPCS works providing high pressure injection. The normal containment and primary heat removal systems fail, but the operators are able to vent. Successful venting produces an environment which may cause injection systems that are operating or that may be able to operate to fail. The overall time available to the operators to perform their recovery actions is approximately 27 hours. In some cases (e.g., performing venting) less time is available. The amount of time available depends on the failures that constitute the cut set and what recovery action is being considered.

The tenth most dominant sequence is TL97 which contributes 0.30% of the core damage frequency from internal events. In this sequence, we have a transient initiator followed by a successful scram. The SRVs open but one or more fail to reclose when required (i.e., they stick open), resulting in a transient-induced LOCA. All high and low pressure injection systems fail and core damage ensues. The cut sets fall into two groups. 1) an early core damage scenario where all AC is lost initially and RCIC fails and 2) a late core damage scenario where AC works for a while and then fails. For the late scenario we have about 10 hours for recovery actions to completed. For the early scenario we have about 48 minutes.

The highest anticipated transient without scram (ATWS) sequence is A49 at 8.94E-08/R-yr. and is the twelfth most dominant sequence contributing only 0.2% of the core damage frequency from internal events. In this sequence, we have a transient initiator followed by initially successful main feedwater. The power conversion (PCS) system fails which leads to the failure of the feedwater turbine-driven pumps from loss of steam or inadaquate level in the condenser. The operator then fails to control the motor-driven feedwater pump injection rate to less than the condensate storage tank (CST) makeup rate of 1800 gpm (the corresponding reactor pressure vessel. The level is 2/3 top of active fuel, TAF) resulting in pump trip and loss of all feedwater. The HPCS system works; but the standby liquid control (SBLC) system fails and the reactor continues to operate at about 9% power. The containment heats up until pressure reaches 60 psig when the operator vents the containment. The resulting severe environments in the reactor building fail HPCS and any other available injection systems and core damage results with a vented containment.

The highest LOCA sequence is L14 at 1.72E-08/R-yr. and is the twenty-first most dominant sequence contributing only 0.04% of the core damage frequency from internal events. In this sequence, we have a LOCA initiator followed by successful scram and vapor suppression operation. The main feedwater system fails but HPCS and one train of the CRD system work providing high pressure injection. The normal containment and primary heat removal systems fail, and venting fails. Containment pressure increases until a leak develops. Depending upon its location, this leak will produce an environment which could cause injection systems that are operating or that may be able to operate to fail. The overall time available to the operators to perform their recovery actions is approximately 15 hours. In some cases (e.g., venting) less time is available. The amount of time available depends on the failures that constitute the cut set and what recovery action is being considered.

The TEMAC code output listing the core damage frequency statistics, risk reduction results, risk increase results, uncertainty importance results, and cut sets for all the sequences is presented in Appendix A.

7.2 Dominant Cut Sets for Integrated Evaluation

In order to obtain an integrated result for internal events, all of the cut sets from all of the sequences were merged together to form one large expression representing the total internal core damage possibilities. A point estimate TEMAC run was made and the cut sets were truncated at 99% for the uncertainty calculations. Originally there were 11,452 cut sets and after truncation 3589 cut sets remained. The full results of the uncertainty calculation are included in Appendix , only selected results are described in this and following sections.

Table 7.2 lists the top fourty-six cut sets from the integrated calculation. These cut sets account for 72.5% of the total core damage frequency from internal initiators.

The two dominant cut sets are short-term station blackouts resulting from a loss of offsite power followed by a common mode failure of the core standby cooling (CSCS) system cooling water pumps which fails the diesel generators and emergency core cooling systems' (ECCS) room cooling. In the dominant cut set, responsible for 21.2% of the core damage frequency, the RCIC inboard isolation valve closes due to a sneak circuit that occurs when offsite power is lost and the emergency DGs are started. The operator fails to reopen the valve in the short time between the DGs starting and then failing soon after due to the loss of cooling and, since the isolation valve is AC powered, it can not be reopened. Offsite power is not restored within 1 hour and core damage results after primary coolant boiloff in about 80 minutes.

In the second cut set, also responsible for 21.3% of the core damage frequency, the valve isolation occurs because RCIC room cooling has failed and the room heats up to the isolation temperature. In an event where all AC power has failed immediately, this high temperature isolation is bypassed and RCIC would continue to work. However, in this case, AC power works for some period of time until the DGs fail on loss of cooling. RCIC is on train A and, if the train A diesel fails before the train B diesel, then the RCIC room temperature will rise on loss of room cooling and RCIC will isolate since train B AC power is available. When train B AC power is then lost, the valve can not be reopened. This event was conservatively modeled as always resulting in isolation. This clearly is not the case, since: (1) some of the time the train B DG will fail before the train A DG, (2) the operator may reopen the valve before the train B DG fails, (3) the time interval between the train A and train B DG failures may not be sufficient for the room to reach the isolation temperature, or (4) the RCIC system could be isolated from the sneak circuit described above.

The third cut set, responsible for 2.3% of the core damage frequency, is similar to the first two except that RCIC continues to work. RCIC fails at about 6 hours when either the battery depletes or the containment pressure results in isolation of the steam discharge line. Core damage occurs about 2 hours after the loss of all injection at about 8 hours. The top three

Table 7.2
INTERNAL EVENTS TOTAL PLANT RUN: CUT SETS

CUT SET NUMBERS, CUT SET ORDERS, CUT SET FREQUENCIES,
CUMULATIVE NORMALIZED CUT SET FREQUENCIES AND CUT SETS
FOR TOP EVENT TOTAL WITH TOP EVENT FREQUENCY 3.11E-05

(THE FIRST COLUMN OF NUMBERS IS THE LINE NUMBERS FOR THE FILE TEMACSETS.DNF)

```
 2   2  5  6.60E-06  0.21235   DGCOOL-BETA       * DGCOOL-PMS-CM     * IE-LOSP           * OPFAILS-REOPEN    *
 3                             RA-8-1H           +
 4   1  5  6.60E-06  0.42470   DGCOOL-BETA       * DGCOOL-PMS-CM     * IE-LOSP           * RA-8-1H           *
 5                             RCICRMCOOL-FLAG   +
 6   3  4  7.13E-07  0.44763   DGCOOL-BETA       * DGCOOL-PMS-CM     * IE-LOSP           * RA-8-8H           +
 7   4  5  4.68E-07  0.46267   DGCOOL-BETA       * DGCOOL-PMS-CM     * IE-T101           * OPFAILSCDS-GE-8M  *
 8                             RA-NONE           +
 9   9  5  4.54E-07  0.47727   DGCOOL-BETA       * DGCOOL-PMS-CM     * EE-MDP-PSW-BC-R   * IE-T9A            *
10                             RA-NONE           +
11   5  5  4.54E-07  0.49187   DGCOOL-BETA       * DGCOOL-PMS-CM     * EE-MDC-IAS-CB-R   * IE-T101           *
12                             RA-NONE           +
13  10  5  4.54E-07  0.50647   DGCOOL-BETA       * DGCOOL-PMS-CM     * EE-MDC-IAS-AB-R   * IE-T9A            *
14                             RA-NONE           +
15   6  5  4.54E-07  0.52107   DGCOOL-BETA       * DGCOOL-PMS-CM     * EE-MDP-PSW-BC-R   * IE-T101           *
16                             RA-NONE           +
17   8  5  4.54E-07  0.53566   DGCOOL-BETA       * DGCOOL-PMS-CM     * EE-MDC-IAS-AB-R   * IE-T101           *
18                             RA-NONE           +
19   7  5  4.54E-07  0.55026   DGCOOL-BETA       * DGCOOL-PMS-CM     * EE-MDC-IAS-CB-R   * IE-T9A            *
20                             RA-NONE           +
21  11  5  3.50E-07  0.56152   DG-FTS-BETA       * DGK-GEN-CM-FTS    * IE-LOSP           * RA-15-8H          *
22                             RA-8-8H           +
23  12  6  2.89E-07  0.57081   DGCOOL-BETA       * DGCOOL-PMS-CM     * IE-T5             * RA-3-12-80M       *
24                             RCICRMCOOL-FLAG   * T5CDS50PERCENT    +
25  15  7  2.79E-07  0.57979   AP040X2-ROO-LFO   * DG0-GEN-LF-F      * DG2B-GEN-LF-FTS   * IE-LOSP           *
26                             OPFAILS-REOPEN    * RA-8-1H           * RA-9-1H           +
27  13  7  2.79E-07  0.58877   AP040X2-ROO-LFO   * DG0-GEN-LF   S    * DG2B-GEN-LF-FTS   * IE-LOSP           *
28                             RA-8-1H           * RA-9-1H           * RCICRMCOOL-FLAG   +
29  14  7  2.79E-07  0.59774   AP039X2-ROO-LFO   * DG0-GEN-LF-FTS    * DG2B-GEN-LF-FTS   * IE-LOSP           *
30                             OPFAILS-REOPEN    * RA-8-1H           * RA-9-1H           +
31  16  4  2.76E-07  0.60604   BATT-BETA         * BATT-FTDP         * IE-LOSP           * RA-8-80M          +
32  17  6  2.41E-07  0.61439   1EB236B-BCO-LF    * CONT-LEAK         * IE-T1             * RA-NONE           *
33                             RHRH01AX-HTX-LFB  * SUR-003-L         +
34  23  7  2.23E-07  0.62157   1E4327NX-ROO-LFO  * AP039X2-ROO-LFO   * DG0-GEN-LF-FTS    * IE-LOSP           *
35                             OPFAILS-REOPEN    * RA-8-1H           * RA-9-1H           +
36  18  7  2.23E-07  0.62876   1E4327NX-ROO-LFO  * AP040X2-ROO-LFO   * DG0-GEN-LF-FTS    * IE-LOSP           *
37                             OPFAILS-REOPEN    * RA-8-1H           * RA-9-1H           +
38  21  7  2.23E-07  0.63594   1E4327NY-ROO-LFO  * AP039X2-ROO-LFO   * DG0-GEN-LF-FTS    * IE-LOSP           *
39                             OPFAILS-REOPEN    * RA-8-1H           * RA-9-1H           +
40  19  7  2.23E-07  0.64312   1E4327NY-ROO-LFO  * AP040X2-ROO-LFO   * DG0-GEN-LF-FTS    * IE-LOSP           *
41                             RA-8-1H           * RA-9-1H           * RCICRMCOOL-FLAG   +
```

Table 7.2 (Concluded)
INTERNAL EVENTS TOTAL PLANT RUN: CUT SETS

22	7	2.23E-07	0.65030	1E4327NY-ROO-LFO *	AP040X2-ROO-LFO *	DG0-GEN-LF-FTS *	IE-LOSP *
				OPFAILS-REOPEN *	RA-8-1H *	RA-9-1H +	
20	7	2.23E-07	0.65748	1E4327NX-ROO-LFO *	AP040X2-ROO-LFO *	DG0-GEN-LF-FTS *	IE-LOSP *
				RA-8-1H *	RA-9-1H *	SCICRMCOOL-FLAG +	
24	6	1.87E-07	0.66380	DG-FTS-BETA *	DGX-GEN-CM-FTS *	IE-LOSP *	RA-15-1H *
				RA-8-1H *	RCIC001X-TDP-LF +		
25	5	1.73E-07	0.66938	CRD-REALIGN-OE *	DGCOOL-BETA *	DGCOOL-PMS-CM *	IE-T5 *
				T5CDS50PERCENT +			
26	6	1.17E-07	0.67316	IE-T5 *	LPCI-BETA *	LPCI-PMS-CM *	/OPFAIL-VENT-2H *
				Q1 *	SUR-007-V +		
27	7	1.17E-07	0.67691	1EB236B-BCO-LF *	CONT-LEAK *	IE-T1 *	RA-NONE *
				RHRC02AA-P-UUM *	SCSF06AA-DENER *	SUR-003-L +	
28	6	1.17E-07	0.68066	1EB236B-BCO-LF *	CONT-LEAK *	IE-T1 *	LCSC002A-P-UUM *
				RA-NONE *	SUR-003-L +		
29	6	1.17E-07	0.68442	1EB236B-BCO-LF *	CONT-LEAK *	IE-T1 *	RA-NONE *
				RHRC02AA-P-NDUUM *	SUR-003-L +		
30	6	1.17E-07	0.68817	1EB236B-BCO-LF *	C0DG01P-P-UUM *	CONT-LEAK *	IE-T1 *
				RA-NONE *	SUR-003-L +		
31	6	9.72E-08	0.69130	1EB236B-BCO-LF *	CONT-LEAK *	IE-T1 *	RA-NONE *
				RHRB01AA-BOO-CC *	SUR-003-L +		
33	6	9.72E-08	0.69442	1EB236B-BCO-LF *	CONT-LEAK *	IE-T1 *	NWVY01CA-FMS-CC *
				RA-NONE *	SUR-003-L +		
32	6	9.72E-08	0.69755	1EB236B-BCO-LF *	C0DG01P-PMS-CC *	CONT-LEAK *	IE-T1 *
				RA-NONE *	SUR-003-L +		
34	7	6.70E-08	0.69971	AP040X2-ROO-LFO *	DG0-GEN-LF-FTS *	DG2B-G-UUM *	IE-LOSP *
				RA-8-1H *	RA-9-1H *	RCICRMCOOL-FLAG +	
35	7	6.70E-08	0.70186	AP040X2-ROO-LFO *	DG0-GEN-LF-FTS *	DG2B-G-UUM *	IE-LOSP *
				OPFAILS-REOPEN *	RA-8-1H *	RA-9-1H +	
36	7	6.70E-08	0.70401	AP039X2-ROO-LFO *	DG0-GEN-LF-FTS *	DG2B-G-UUM *	IE-LOSP *
				OPFAILS-REOPEN *	RA-8-1H *	RA-9-1H +	
37	7	6.70E-08	0.70617	AP040X2-ROO-LFO *	DG0-G-UUM *	DG2B-GEN-LF-FTS *	IE-LOSP *
				RA-8-1H *	RA-9-1H *	RCICRMCOOL-FLAG +	
39	7	6.70E-08	0.70832	AP039X2-ROO-LFO *	DG0-G-UUM *	DG2B-GEN-LF-FTS *	IE-LOSP *
				OPFAILS-REOPEN *	RA-8-1H *	RA-9-1H +	
38	7	6.70E-08	0.71048	AP040X2-ROO-LFO *	DG0-G-UUM *	DG2B-GEN-LF-FTS *	IE-LOSP *
				OPFAILS-REOPEN *	RA-8-1H *	RA-9-1H +	
40	4	6.50E-08	0.71257	IE-T9B *	/OPFAIL-VENT-2H *	RHRH01AX-HTX-LFB *	SUR-005-V +
41	4	6.50E-08	0.71466	IE-T9A *	/OPFAIL-VENT-2H *	RHRH01BX-HTX-LFB *	SUR-005-V +
42	6	6.49E-08	0.71675	DGCOOL-BETA *	DGCOOL-PMS-CM *	IE-LOSP *	Q1 *
				RA-8-48M *	RCICRMCOOL-FLAG +		
44	6	6.48E-08	0.71883	1E42YB-V-UUM *	CONT-LEAK *	CSCD300A-PLG-LF *	IE-T1 *
				RA-NONE *	SUR-003-L +		
43	6	6.48E-08	0.72092	1EB236B-B-UUM *	CONT-LEAK *	CSCD300A-PLG-LF *	IE-T1 *
				RA-NONE *	SUR-003-L +		
45	6	5.76E-08	0.72277	1E4327NX-ROO-LFO *	AP040X2-ROO-LFO *	DG0-G-UUM *	IE-LOSP *
				RA-8-1H *	RCICRMCOOL-FLAG +		
46	6	5.76E-08	0.72462	1E4327NY-ROO-LFO *	AP040X2-ROO-LFO *	DG0-G-UUM *	IE-LOSP *
				RA-8-1H *	RCICRMCOOL-FLAG +		

cut sets, while correct in themselves, double count some of the frequency contribution because they not not completely independent. Due to the complexity of the interactions between the sneak circuit and the system isolation on room temperature for various AC power states, it was not possible to easily model this process exactly in the fault trees. The sneak circuit will always occur if the appropriate DG restarts after the loss of offsite power; but, only if the operator reopens the valve can the room temperature isolation come in to play. If the operator reopens the valve in both cases, then RCIC can continue to work.

The next set of seven cut sets, responsible for 10.3% of the core damage frequency, consists of train A AC or DC power failure and common mode failure of the CSCS cooling water pumps. The cooling water failure results in the failure of all ECCS systems including RCIC (since train B AC is working, RCIC will isolate on high room temperature), the train A DG (train B may start and fail but train B AC is still available from offsite), and the CRD system whose pumps are in the HPCS room. Main feedwater fails when the main steam isolation valves (MSIVs) drift closed on loss of instrument air and the motor-driven pump injection valve fails closed or a turbine pump locks up on loss of DC power resulting in high RPV level, MSIV isolation, and main feedwater high level trip.

7.3 Importance Analysis Results

7.3.1 Risk Reduction

The risk reduction measure calculates the decrease in the core damage frequency when a single basic event's probability is set to zero. The implication is that the component or event represented by this basic event can not fail or occur. This measure tells you how much risk reduction you could gain by making a component perfect versus leaving it at its current reliability.

Risk reduction measures are calculated both for basic event and for initiating events. Risk reductions for each individual sequence and the integrated result are presented in the TEMAC outputs shown in Appendix A. In this section, we will discuss only the integrated results which are shown in Table 7.3.

One important item to note is that since some complement events appear in the LaSalle fault trees and, therefore, in the accident sequence cut sets; some events can have negative risk reductions. That is, decreasing a certain events failure probability can actually result in an increase in risk not a decrease. These events appear at the bottom of the risk reduction list, so you must not look just at the top events in the list.

The importance of this is much more obvious if one looks at individual sequences then for the integrated results. In some sequences only an event or its complement shows up, for example, sequences T18 and T22. Sequence T18 has the event CONT-LEAK while sequence T22 has the event /CONT-LEAK. Reducing the probability of containment failure by leakage increases the

Table 7.3
INTERNAL EVENTS TOTAL PLANT RUN:
RISK REDUCTION BY BASE EVENT (WITH ASSOCIATED UNCERTAINTY INTERVALS)

BASE EVENT	OCCUR	PROB	(RANK)	RISK REDUCTION	(RANK)	LOWER 5%	UPPER 5%
RA-8-1H	608	2.50E-01	(42.0)	1.89E-05	(1.0)	9.15E-07	8.96E-05
DGCOOL-BETA	27	1.10E-01	(49.5)	1.77E-05	(2.5)	2.58E-07	7.48E-05
DGCOOL-PMS-CM	27	2.50E-03	(161.5)	1.77E-05	(2.5)	2.58E-07	7.48E-05
RCICRMCOOL-FLAG	1023	1.00E+00	(9.5)	1.09E-05	(4.0)		
OPFAILS-REOPEN	129	1.00E+00	(9.5)	8.87E-06	(5.0)		
RA-NONE	306	1.00E+00	(9.5)	6.19E-06	(6.0)		
RA-9-1H	468	9.30E-01	(21.0)	5.16E-06	(7.0)	8.70E-08	2.95E-05
DG0-GEN-LF-FTS	807	2.50E-02	(67.0)	5.14E-06	(8.0)	1.13E-07	3.53E-05
AP040X2-ROO-LFO	604	2.00E-02	(84.0)	3.75E-06	(9.0)	2.39E-08	2.43E-05
DG2B-GEN-LF-FTS	561	2.50E-02	(67.0)	2.74E-06	(10.0)	7.93E-08	1.69E-05
SUR-003-L	326	1.60E-01	(46.0)	2.70E-06	(11.0)	0.00E+00	2.09E-05
1EB236B-BCO-LF	176	7.20E-05	(363.5)	2.08E-06	(12.0)	1.58E-11	1.73E-05
RA-8-10H	1179	2.00E-02	(84.0)	1.74E-06	(13.0)	9.65E-09	8.44E-06
RA-9-2H	1162	8.70E-01	(23.0)	1.71E-06	(14.0)	9.51E-09	8.17E-06
RA-8-8H	468	2.70E-02	(64.0)	1.71E-06	(15.0)	7.05E-08	7.83E-06
1E4327NY-ROO-LFO	360	2.00E-02	(84.0)	1.66E-06	(16.0)	8.22E-09	1.21E-05
1E4327NX-ROO-LFO	358	2.00E-02	(84.0)	1.63E-06	(17.0)	8.10E-09	1.21E-05
CONT-LEAK	932	7.50E-01	(24.0)	1.62E-06	(18.0)	0.00E+00	2.73E-05
AP039X2-ROO-LFO	342	2.00E-02	(84.0)	1.56E-06	(19.0)	8.34E-09	1.21E-05
EE-MDP-PSW-BC-R	78	3.30E-01	(39.0)	1.01E-06	(20.0)		
EE-MDC-IAS-CB-R	26	3.30E-01	(39.0)	9.24E-07	(21.5)		
EE-MDC-IAS-AB-R	26	3.30E-01	(39.0)	9.24E-07	(21.5)		
DG0-GEN-LF-FTR	546	1.90E-02	(100.0)	9.10E-07	(23.0)	2.51E-09	5.81E-06
DG2A-GEN-LF-FTR	445	1.90E-02	(100.0)	6.51E-07	(24.0)	2.66E-09	3.74E-06
RA-9-8H	394	6.00E-01	(26.0)	5.76E-07	(25.0)	6.34E-09	4.07E-06
DG-FTS-BETA	14	1.20E-02	(103.0)	5.75E-07	(26)	1.71E-08	1.99E-06
DGX-GEN-CM-FTS	14	2.50E-02	(67.0)	5.75E-07	(26.5)	1.71E-08	1.99E-06
RHRH01AX-HTX-LFB	59	6.20E-03	(105.5)	5.64E-07	(28.0)	1.82E-09	4.34E-06
DG2B-GEN-LF-FTR	361	1.90E-02	(100.0)	5.40E-07	(29.0)	1.75E-09	3.05E-06
DG0-G-UUM	114	6.00E-03	(108.5)	5.15E-07	(30.0)	1.12E-08	2.82E-06
DG2A-GEN-LF-FTS	307	2.50E-02	(67.0)	4.84E-07	(31.0)	9.82E-09	3.61E-06
SUR-002-L	315	1.60E-01	(46.0)	4.84E-07	(32.0)	0.00E+00	3.46E-06
OPFAILSCDS-OE-8M	11	3.40E-01	(37.0)	4.75E-07	(33.0)	6.15E-09	1.88E-06
T5CDS50PERCENT	3	5.00E-01	(30.0)	4.58E-07	(34.0)		
SUR-021-R	154	8.50E-02	(55.5)	4.49E-07	(3[illegible].0)	0.00E+00	7.76E-06
SUR-005-V	68	2.10E-03	(186.0)	3.91E-07	(36.0)	4.18E-09	1.59E-06
RA-15-8H	7	4.50E-01	(33.0)	3.58E-07	(37.0)	8.63E-09	1.42E-06
Q1	111	8.20E-03	(104.0)	3.47E-07	(38.0)	6.34E-09	1.24E-06
DG0-GEN-CC-FTS	114	3.70E-03	(114.0)	3.27E-07	(39.0)	1.43E-09	1.79E-06
RA-1-1-27H	201	2.10E-03	(186.0)	3.14E-07	(40.0)	6.58E-11	1.90E-06
DG2B-G-UUM	59	6.00E-03	(108.5)	3.10E-07	(41.0)	8.84E-09	1.26E-06
OPFAIL-VENT-2H	118	2.10E-03	(186.0)	4.80E-08	(130.0)	-2.59E-09	3.18E-07
RA-5V-1-2H	48	2.10E-03	(186.0)	1.28E-09	(299.0)	-5.34E-10	5.11E-09

Table 7.3 (Concluded)
INTERNAL EVENTS TOTAL PLANT RUN:
RISK REDUCTION BY INITIATING EVENT (WITH ASSOCIATED UNCERTAINTY INTERVALS)

INIT EVENT	OCCUR	FREQ (RANK)	RISK REDUCTION (RANK)	LOWER 5%	UPPER 5%
IE-LOSP	2686	9.60E-02 (7.0)	2.31E-05 (1.0)	1.17E-06	1.12E-04
IE-T101	252	5.00E-03 (10.5)	2.53E-06 (2.0)	4.15E-08	1.07E-05
IE-T1	160	4.50E+00 (1.0)	1.81E-06 (3.0)	9.66E-10	1.48E-05
IE-T9A	59	5.00E-03 (10.5)	1.63E-06 (4.0)	6.48E-08	6.26E-06
IE-T5	104	6.00E-01 (3.0)	9.01E-07 (5.0)	2.19E-08	3.60E-06
IE-T102	118	5.00E-03 (10.5)	4.00E-07 (6.0)	3.43E-11	3.20E-06
IE-T9B	40	5.00E-03 (10.5)	2.70E-07 (7.0)	1.99E-09	1.31E-06
IE-T3	74	6.10E-01 (2.0)	2.44E-07 (8.0)	7.03E-10	2.09E-06
IE-T2	43	5.20E-01 (4.0)	1.15E-07 (9.0)	3.55E-10	8.60E-07
IE-T4	31	4.10E-01 (5.0)	8.84E-08 (10.0)	2.58E-10	6.16E-07
IE-SLOCA	18	3.00E-02 (8.0)	7.40E-09 (11.0)	0.00E+00	5.24E-08
IE-T7	4	1.40E-01 (6.0)	3.26E-09 (12.0)	1.40E-11	1.70E-08

containment failure probability by rupture. In the integrated result these effects are balanced out somewhat. However, one can see by looking at Table 7.3 that two events even in the integrated analysis have negative risk reduction measures. These two events, OPFAIL-VENT-2H and RA-5V-1-2H, represent successful operator venting of the containment. Venting using the current procedures creates severe environments in the reactor building that can fail injection systems leading to core damage sequences. If venting fails and then the containment fails by overpressure, the failure is often to the refueling floor which bypasses the reactor building and no severe environments are created. For the dominant long-term containment heat removal failure sequences which appear in this analysis, HPCS is the system supplying injection. Since HPCS is a high pressure system and does not fail from high containment pressures, the conditional probability of core damage is actually higher if venting occurs than if containment failure occurs. This is because venting always results in severe environments while containment failure only results in severe environments if the failure is in the reactor building.

The most important event for risk reduction is the loss of offsite power initiating event with a risk reduction measure of 2.31E-05/R-yr. The second most important event is the non-recovery of offsite power within one hour with a risk reduction measure of 1.89E-05/R-yr. The third and fourth most important events are concerned with the CSCS cooling water pump common mode failure and are the pump random failure probability and the common mode beta factor which links the pumps together, each with a risk reduction of 1.77E-05/R-yr. The fifth and sixth most important events are related to the RCIC isolation problem either the isolation on room high temperature or the sneak circuit with risk reductions of 1.09E-5/R-yr. and 8.87E-06/R-yr. respectively.

7.3.2 Risk Increase

The risk increase measure calculates the increase in the core damage frequency obtained by setting each basic events failure probability to one. The implication is that the component or event represented by this basic event always fails or occurs. This measures tells you how much increase in risk you would obtain if a component was allowed to degrade to the point of failure versus maintaining it at its current reliability level.

Risk increase measures are calculated only for basic events. Since initiating events are frequencies and can have values greater than 1.0, this calculation is not applicable to them. Risk increases for each individual sequence and the integrated result are presented in the TEMAC outputs shown in Appendix A. In this section, we will discuss only the integrated results which are shown in Table 7.4.

As with the risk decrease measure, certain events can have negative risk increase implying that the risk decreases as their probability is increased. In fact, the same two events that have negative risk decreases have negative risk increases. For example, as the probability of the operator failing to vent increases the core damage frequency goes down

Table 7.4
INTERNAL EVENTS TOTAL PLANT RUN:
RISK INCREASE BY BASE EVENT (WITH ASSOCIATED UNCERTAINTY INTERVALS)

BASE EVENT	OCCUR	PROB (RANK)	RISK INCREASE (RANK)	LOWER 5%	UPPER 5%
1EB236B-BCO-LF	176	7.20E-05 (363.5)	2.89E-02 (1.0)	9.53E-07	2.41E-01
RPS	56	1.00E-05 (378.0)	1.19E-02 (2.0)	1.06E-03	5.76E-02
DGCOOL-PMS-CM	27	2.50E-03 (161.5)	7.05E-03 (3.0)	1.28E-03	2.00E-02
1EB36YB-BCO-LF	54	7.20E-05 (363.5)	2.17E-03 (4.0)	1.40E-04	9.45E-03
1E42YB-V-UUM	27	1.00E-04 (339.5)	1.46E-03 (5.0)	0.00E+00	1.44E-02
1EB235A-BCO-LF	38	7.20E-05 (363.5)	1.13E-03 (6.0)	5.9?E-07	9.41E-03
1EB236B-B-UUM	23	1.00E-04 (339.5)	9.97E-04 (7.0)	0.00E+00	7.96E-03
1EB211YX-BCO-LF	9	7.20E-05 (363.5)	8.11E-04 (8.0)	3.46E-06	6.54E-03
1EB36Y1X-BCO-LF	7	7.20E-05 (363.5)	7.88E-04 (10.5)	2.64E-06	6.53E-03
1EB212YX-BCO-LF	7	7.20E-05 (363.5)	7.88E-04 (10.5)	2.66E-06	6.53E-03
1EB36XB-BCO-LF	7	7.20E-05 (363.5)	7.88E-04 (10.5)	2.64E-06	6.53E-03
1EB2Y22X-BCO	7	7.20E-05 (363.5)	7.88E-04 (10.5)	2.66E-06	6.53E-03
1EB1Y22X-BCO	6	7.20E-05 (363.5)	7.74E-04 (13.0)	5.96E-07	6.52E-03
BATT-FTDP	1	3.60E-04 (269.0)	7.68E-04 (14.0)	9.08E-05	2.26E-03
1EB16212-BCO	5	8.40E-05 (346.5)	7.42E-04 (15.0)	2.64E-06	5.62E-03
1EB16211-BCO	4	8.40E-05 (346.5)	7.29E-04 (16.0)	5.96E-07	5.61E-03
CSCD300A-PLG-LF	42	1.20E-03 (195.5)	2.40E-04 (17.0)	2.14E-06	1.76E-03
DG0-GEN-LF-FTS	807	2.50E-02 (67.0)	2.00E-04 (18.0)	8.68E-06	1.11E-03
1EB235XA-BCO-LF	8	7.20E-05 (363.5)	1.88E-04 (19.0)	4.74E-05	1.15E-03
SUR-005-V	68	2.10E-03 (186.0)	1.86E-04 (20.0)	2.47E-05	5.37E-04
AP040X2-ROO-LFO	604	2.00E-02 (84.0)	1.84E-04 (21.0)	1.86E-05	7.80E-04
C0DG01F-PLG-LF	51	5.80E-04 (224.0)	1.[illegible]3E-04 (22.0)	6.19E-06	1.22E-03
RA-1-1-27H	201	2.10E-03 (186.0)	1.49E-04 (23.0)	1.25E-07	1.02E-03
DGCOOL-BETA	27	1.10E-01 (49.5)	1.43E-04 (24.0)	4.01E-06	6.63E-04
C0DG01P-PMS-LF	80	1.10E-03 (200.0)	1.40E-04 (25.0)	5.83E-06	1.28E-03
EE-C2DG01F-PLG	67	7.20E-04 (210.0)	1.25E-04 (26.0)	1.89E-05	4.78E-04
1E41YA-V-UUM	19	1.00E-04 (339.5)	1.25E-04 (27.0)	0.00E+00	8.36E-04
1EB235YA-BCO-LF	7	7.20E-05 (363.5)	1.24E-04 (28.0)	2.04E-06	9.24E-04
1EB35Y2X-BCO-LF	6	7.20E-05 (363.5)	1.14E-04 (29.0)	5.68E-07	9.04E-04
DG2B-GEN-LF-FTS	561	2.50E-02 (67.0)	1.07E-04 (30.0)	6.59E-06	3.93E-04
LCSC002A-P-UUM	67	3.00E-03 (125.5)	1.02E-04 (31.0)	3.29E-06	7.28E-04
C2DG01F-PLG-LF	41	5.80E-04 (224.0)	9.81E-05 (32.0)	1.25E-05	3.47E-04
1EB235A-B-UUM	14	1.00E-04 (339.5)	9.58E-05 (33.0)	0.00E+00	6.20E-04
RA-8-27H	339	3.20E-03 (119.5)	9.38E-05 (34.0)	2.34E-07	7.83E-04
RA-2-11-27H	66	1.60E-03 (193.0)	9.23E-05 (35.0)	4.44E-08	5.78E-04
RHRH01AX-HTX-LFB	59	6.20E-03 (105.5)	9.04E-05 (36.0)	1.55E-06	7.13E-04
RHRC02AA-P-UUM	43	3.00E-03 (125.5)	9.04E-05 (37.0)	1.57E-06	7.02E-04
EE-C0[illegible]G01F-PLG	53	7.20E-04 (210.0)	8.86E-05 (38.0)	2.61E-06	6.02E-04
C2DG01P-PMS-LF	59	1.10E-03 (200.0)	8.84E-05 (39.0)	1.23E-05	3.51E-04
RHRC02AA-PMS-LF	23	1.10E-03 (200.0)	8.82E-05 (40.0)	1.08E-06	6.87E-04
DG0-GEN-CC-FTS	114	3.70E-03 (114.0)	8.81E-05 (41.0)	2.64E-05	5.58E-04
RHRB01AA-BOO-LF	18	5.00E-04 (251.0)	8.70E-05 (43.0)	1.08E-06	5.68E-04
CSCF068A-VCC-LF	8	5.00E-04 (251.0)	8.70E-05 (43.0)	1.08E-06	5.68E-04

because, for the dominant sequences, there is less probability of severe environments if the containment fails than if its vented as described above.

The most important event for risk increase is the failure of the circuit breaker from 4160 V AC emergency bus 242Y (train B) to 480 V AC buses 236X and 236Y with a risk increase of 2.89E-02/R-yr. This fails all of train B emergency AC power. The second most important event is reactor scram failure with a risk increase of 1.19E-02/R-yr. Even though ATWS sequences at LaSalle are very low and do not dominant the core damage frequency, if the failure to scram probability increased, they would become very important. The third most important event is the CSCS cooling water pump random failure probability which determines the level of the cooling water common mode event. This event has a risk increase of 7.05E-03/R-yr. The next ten events are electric power circuit breaker failures or unavailability due to maintenance which result in degraded AC and DC power states.

7.3.3 Uncertainty Importance

The uncertainty importance is calculated for groups of basic events all of which have the same underlying distribution (i.e., all basic events represented by the same LHS[2] variable). In the Latin Hypercube (LHS) sample, a certain distribution might have been selected for motor-operated valve failure to open. Every basic event appearing in the model that represents a motor-operated valve failing to open is correlated, is represented by the same LHS variable, and has the same value for a particular LHS sample member. The uncertainty importance calculation is performed by performing a polynomial regression on the expected value of the log of the top event conditional on the sampled values of the selected LHS variable. The uncertainty importance is calculated as: (the unconditional variance in the log of the top event - the expectation of the variance of the log of the top event conditional on the selected LHS variable)/(the unconditional variance of the log of the top event). This calculation is performed both for basic event and initiating events.

For the LaSalle analysis, the result of this calculation for each accident sequence and for the integrated result are presented in Appendix A. Only the integrated results will be discussed in this section. The integrated results are presented in Table 7.5.

The dominant class of events, responsible for a 28.4% reduction in the uncertainty of the log risk, is uncertainty in the probability of control circuit failure. This class includes valve, circuit breaker, pump, and fan control circuit failures. The second and third most dominant classes are deenergized relays failure to energize, responsible for a 16.5% and 16.3% reduction (two class were modeled with different exposure times which decoupled the LHS distributions in the LHS sample; they were correlated, however). The fourth and fifth most dominant classes are failure of

Table 7.5
INTERNAL EVENTS TOTAL PLANT RUN:
UNCERTAINTY IMPORTANCE BY BASE EVENT

BASE EVENT	OCCUR	PROB	(RANK)	% REDUCTION IN THE UNCERTAINTY OF LOG RISK	(RANK)	Y.05/TE.05*	Y.95/TE.95*
LPCI-MOV-CM2	4	2.50E-03	(161.5)	28.4	(17.5)	2.81	0.94
RHRB01BB-POO-CC	8	2.50E-03	(161.5)	28.4	(17.5)	2.81	0.94
1EB422B-BCC-CC	11	2.50E-03	(161.5)	28.4	(17.5)	2.81	0.94
C2DG01P-PMS-CC	33	2.50E-03	(161.5)	28.4	(17.5)	2.81	0.94
DAV-MOD-COM-CC	19	2.50E-03	(161.5)	28.4	(17.5)	2.81	0.94
C0DG01P-PMS-CC	36	2.50E-03	(161.5)	28.4	(17.5)	2.81	0.94
1EB425B-BCC-CC	30	2.50E-03	(161.5)	28.4	(17.5)	2.81	0.94
RWCF004X-VOO-CC	6	2.50E-03	(161.5)	28.4	(17.5)	2.81	0.94
DG2V03CB-FMS-CC	30	2.50E-03	(161.5)	28.4	(17.5)	2.81	0.94
CSCC002-PMS-CC	21	2.50E-03	(161.5)	28.4	(17.5)	2.81	0.94
DGCOOL-PMS-CM	27	2.50E-03	(161.5)	28.4	(17.5)	2.81	0.94
SEVY03CB-FMS-CC	35	2.50E-03	(161.5)	28.4	(17.5)	2.81	0.94
1ED423B-BOO-CC	11	2.50E-03	(161.5)	28.4	(17.5)	2.81	0.94
SWVY02CC-FMS-CC	26	2.50E-03	(161.5)	28.4	(17.5)	2.81	0.94
RHRF48BB-VOO-CC	3	2.50E-03	(161.5)	28.4	(17.5)	2.81	0.94
RHRF48AA-VOO-CC	2	2.50E-03	(161.5)	28.4	(17.5)	2.81	0.94
RHRB01AA-BOO-CC	22	2.50E-03	(161.5)	28.4	(17.5)	2.81	0.94
DGHV01CC-FMS-CC	16	2.50E-03	(161.5)	28.4	(17.5)	2.81	0.94
1EB432C-BCC-CC	16	2.50E-03	(161.5)	28.4	(17.5)	2.81	0.94
HCSC001C-PMS-CS	16	2.50E-03	(161.5)	28.4	(17.5)	2.81	0.94
NWVY01CA-FMS-CC	22	2.50E-03	(161.5)	28.4	(17.5)	2.81	0.94
1EB412A-BCC-CC	28	2.50E-03	(161.5)	28.4	(17.5)	2.81	0.94
1EB413A-BOO-CC	28	2.50E-03	(161.5)	28.4	(17.5)	2.81	0.94
DG0V01CA-FMS-CC	27	2.50E-03	(161.5)	28.4	(17.5)	2.81	0.94
1EB433C-BOO-CC	16	2.50E-03	(161.5)	28.4	(17.5)	2.81	0.94
LPCI-PMS-CM	8	2.50E-03	(161.5)	28.4	(17.5)	2.81	0.94
CSCF068A-VCC-CC	2	2.50E-03	(161.5)	28.4	(17.5)	2.81	0.94
SY-REGP-RCIC001X	6	2.50E-03	(161.5)	28.4	(17.5)	2.81	0.94
DHV-MOD-COM-CC	16	2.50E-03	(161.5)	28.4	(17.5)	2.81	0.94
D0V-MOD-COM-CC	27	2.50E-03	(161.5)	28.4	(17.5)	2.81	0.94
1EB234B-BCC-CC	30	2.50E-03	(161.5)	28.4	(17.5)	2.81	0.94
LPCI-MOV-CM1	4	2.50E-03	(161.5)	28.4	(17.5)	2.81	0.94
CSCF068B-VCC-CC	3	2.50E-03	(161.5)	28.4	(17.5)	2.81	0.94
1EB233A-BCC-AS	31	2.50E-03	(161.5)	28.4	(17.5)	2.81	0.94
2DG1PK18-ROO-LFO	17	5.00E-04	(251.0)	16.5	(35.5)	1.00	1.00
CSC02K18-ROO-LFO	12	5.00E-04	(251.0)	16.5	(35.5)	1.00	1.00
1E4327NX-ROO-LFO	358	2.00E-02	(84.0)	16.3	(48.5)	1.64	0.86
RACTK5-ROO-LFO	1	2.00E-02	(84.0)	16.3	(48.5)	1.64	0.86
RACTK3-ROO-LFO	1	2.00E-02	(84.0)	16.3	(48.5)	1.64	0.86
HACTK9-ROO-LFO	67	2.00E-02	(84.0)	16.3	(48.5)	1.64	0.86
LAK14BPC-RCO-LFO	26	2.00E-02	(84.0)	16.3	(48.5)	1.64	0.86
LAK18BRB-ROO-LFO	40	2.00E-02	(84.0)	16.3	(48.5)	1.64	0.86

Table 7.3 (Continued)
INTERNAL EVENTS TOTAL PLANT RUN
UNCERTAINTY IMPORTANCE BY BASE EVENT

BASE EVENT	OCCUR	PROB	(RANK)	% REDUCTION IN THE UNCERTAINTY OF LOG RISK	(RANK)	Y.05/TE.05*	Y.95/TE.95*
LAK10RCA-ROO-LFO	38	2.00E-02	(84.0)	16.3	(48.5)	1.64	0.86
1E4327NY-ROO-LFO	360	2.00E-02	(84.0)	16.3	(48.5)	1.64	0.86
RACTK17-ROO-LFO	1	2.00E-02	(84.0)	16.3	(48.5)	1.64	0.86
LAK2BRCB-RCO-LFO	1	2.00E-02	(84.0)	16.3	(48.5)	1.64	0.86
LAK70ARA-ROO-LFO	1	2.00E-02	(84.0)	16.3	(48.5)	1.64	0.86
LAK23BRC-ROO-LFO	1	2.00E-02	(84.0)	16.3	(48.5)	1.64	0.86
AP039X2-ROO-LFO	342	2.00E-02	(84.0)	16.3	(48.5)	1.64	0.86
LAK70BRB-ROO-LFO	1	2.00E-02	(84.0)	16.3	(48.5)	1.64	0.86
LAK2ARCA-RCO-LFO	1	2.00E-02	(84.0)	16.3	(48.5)	1.64	0.86
HACTK3-ROO-LFO	59	2.00E-02	(84.0)	16.3	(48.5)	1.64	0.86
AP037X3-ROO-LFO	16	2.00E-02	(84.0)	16.3	(48.5)	1.64	0.86
AP040X2-ROO-LFO	604	2.00E-02	(84.0)	16.3	(48.5)	1.64	0.86
LAK9BRCB-ROO-LFO	40	2.00E-02	(84.0)	16.3	(48.5)	1.64	0.86
LAK18ARA-ROO-LFO	27	2.00E-02	(84.0)	16.3	(48.5)	1.64	0.86
LAK14ARC-RCO-LFO	9	2.00E-02	(84.0)	16.3	(48.5)	1.64	.86
LAK93ARC-ROO-LFO	22	2.00E-02	(84.0)	16.3	(48.5)	1.64	0.86
LAK9ARCA-ROO-LFO	25	2.00E-02	(84.0)	16.3	(48.5)	1.64	0.86
LAK93BRC-ROO-LFO	46	2.00E-02	(84.0)	16.3	(48.5)	1.64	0.86
LAK3BRCB-RCO-LFO	23	3.40E-03	(117.5)	16.1	(61.5)	1.00	1.00
LAK3ARCA-RCO-LFO	3	3.40E-03	(117.5)	16.1	(61.5)	1.00	1.00
1EK18ARC-ROO-LFO	16	5.80E-04	(224.0)	15.8	(63.0)	1.00	1.00
SAK6BRCX-RCO-LFO	6	3.90E-02	(59.5)	15.8	(65.5)	1.06	1.00
SAK2ARCX-RCO-LFO	6	3.90E-02	(59.5)	15.8	(65.5)	1.06	1.00
SAK6ARCX-RCO-LFO	6	3.90E-02	(59.5)	15.8	(65.5)	1.06	1.00
SAK2BRCX-RCO-LFO	6	3.90E-02	(59.5)	15.8	(65.5)	1.06	1.00
DG2B-GEN-LF-FTS	561	2.50E-02	(67.0)	6.8	(69.5)	0.84	0.97
DGX-GEN-CM-FTS	14	2.50E-02	(67.0)	6.8	(69.5)	0.84	0.97
DG2A-GEN-LF-FTS	307	2.50E-02	(67.0)	6.8	(69.5)	0.84	0.97
DG0-GEN-LF-FTS	807	2.50E-02	(67.0)	6.8	(69.5)	0.84	0.97
SUR-001-L	54	1.90E-01	(44.0)	6.5	(72.0)	1.00	1.00
SUR-002-A-L	10	1.60E-01	(46.0)	5.4	(73.5)	1.38	0.97
SUR-002-L	315	1.60E-01	(46.0)	5.4	(73.5)	1.38	0.97
SUR-003-L	326	1.60E-01	(46.0)	5.3	(75.0)	1.45	1.00
1EB16211-BCO	4	8.40E-05	(346.5)	5.1	(78.5)	1.00	1.00
CCB0DG1P-BCO-LF	3	8.40E-05	(346.5)	5.1	(78.5)	1.00	1.00
1EB16212-BCO	5	8.40E-05	(346.5)	5.1	(78.5)	1.00	1.00
BC001CB-BCO-LF	1	8.40E-05	(346.5)	5.1	(78.5)	1.00	1.00
CCB2DG1P-BCO-LF	3	8.40E-05	(346.5)	5.1	(78.5)	1.00	1.00
CCBC002-BCO-LF	1	8.40E-05	(346.5)	5.1	(78.5)	1.00	1.00

Table 7.5 (Concluded)
INTERNAL EVENTS TOTAL PLANT RUN:
UNCERTAINTY IMPORTANCE BY INITIATING EVENT

INIT EVENT	OCCUR	FREQ	(RANK)	% REDUCTION IN THE UNCERTAINTY OF LOG RISK	(RANK)	Y.05/TE.05*	Y.95/TE.95*
IE-LOSP	2686	9.60E-02	(7.0)	12.5	(1.0)	1.39	0.98
IE-T3	74	6.10E-01	(2.0)	2.2	(2.0)	1.00	1.00
IE-T4	31	4.10E-01	(5.0)	1.5	(3.0)	1.00	1.00
IE-SLOCA	18	3.00E-02	(8.0)	1.5	(4.0)	1.00	1.00
IE-T5	104	6.00E-01	(3.0)	1.3	(5.0)	1.00	1.01
IE-T'	43	5.20E-01	(4.0)	0.0	(9.0)		
IE-T1	160	4.50E+00	(1.0)	0.0	(9.0)		
IE-T7	4	1.40E-01	(6.0)	0.0	(9.0)		
IE-T102	118	5.00E-03	(10.5)	0.0	(9.0)		
IE-T9A	59	5.00E-03	(10.5)	0.0	(9.0)		
IE-T9B	40	5.00E-03	(10.5)	0.0	(9.0)		
IE-T101	252	5.00E-03	(10.5)	0.0	(9.0)		

energized relays to remain energized, responsible for a 16.1% and 15.8% reduction (these were also divided into two groups). The sixth most dominant class the loss of offsite power initiator which is responsible for a 12.5% reduction. The seventh most dominant class is diesel generator failure to start which is responsible for a 6.8% reduction. The eighth to tenth most dominant classes are the severe environment failure probabilities of various types of equipment, responsible for 6.5%, 5.4%, and 5.3% reductions.

7.4 Insights and Conclusions

Overall, the mean core damage frequency of 4.41E-05/R-yr. for the internal events analysis is very good considering that this is the first time a PRA has been performed on LaSalle and no design or construction deficiencies were found that resulted in excessive core damage potential.

Several changes could be made to systems and procedures that would result in a significant reduction in the current core damage frequency and not be too costly. The first is to eliminate the sneak circuit in the RCIC isolation logic that results in the RCIC steam line inboard isolation valve closing when offsite AC power is lost and the appropriate diesel generator starts. This is is clearly an unwanted result that defeats the purpose of having a DC powered system to mitigate station blackout type accidents. This is particularly true here since the dominant core damage sequence involves a loss of offsite power followed by a delayed loss of the diesel generators as a result of the loss of diesel generator cooling water. This results in a delayed station blackout sequence in which the operator must reopen the isolation valve before the diesel generators fail. Commonwealth Edison Company (CECo) immediately recognized that this was a design deficiency when it was initially found in the PRA analysis. A design modification was devised but implementation was delayed until the PRA was completed so that its relative importance could be assessed. The design change should go in at the next refueling outage.

The second change would be to change the RCIC room temperature isolation logic so that, in cases where train A AC power has fail but train B AC is available, RCIC does not isolate if no other ECCS system is working. The current logic assumes that if either AC power train is working then sufficient other systems are available to cool the core and that RCIC is not needed. For the type of sequences showing up here, a modification as described above would reduce the probability of RCIC isolation in these sequences significantly while introducing a very low probability failure event (i.e., a spurious inhibition signal).

The third major change would be to change the venting procedure so that venting does not result in severe environments in the reactor building. At LaSalle, this can be done solely by changing the procedures since a hardened vent line already exists. The current procedures require that the operator vent the containment through the standby gas treatment system. This system has an open suction line from the reactor building and, even if this is isolated, has some duct work and a rubber boot connecting the vent pipe to the standby gas treatment filter. This duct work and/or boot will certainly fail if the main vent lines are opened. The resulting severe environment in the reactor building has a very high probability of failing the ECCS and CRD systems all of which have components in the reactor building. A simple change in procedure to close the reactor building suction line, isolate the standby gas treatment system, and vent to the steam tunnel should be able to mitigate this problem. The vent and purge system can not be used because it has a similar boot. Venting to the steam tunnel can produce some changes in the turbine building environment as a result of leakage from the turbine cavity into the main building but the blowout panel on the roof should open directing most of the steam out that path. A more detailed study of possible turbine building environments would need to be made before this change could be made. In addition, Level II/III considerations as to the effects on possible radioactive source terms from accidents which progressed to core damage anyway would need to be assessed. Section 4.6.4 of Volume 1 of this report contains a more detailed discussion of this problem.

7.5 References

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NRC FORM 335 (2-89) NRCM 1102, 3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER (Assigned by NRC. Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG/CR-4832
SAND92-0537
Vol. 3 Part 1

2. TITLE AND SUBTITLE

Analysis of the LaSalle Unit 2 Nuclear Power Plant: Risk Methods Integration and Evaluation Program (RMIEP) Internal Events Accident Sequence Quantification Main Report

3. DATE REPORT PUBLISHED

MONTH	YEAR
August	1992

4. FIN OR GRANT NUMBER

A1386

5. AUTHOR(S)

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6. TYPE OF REPORT

Technical

7. PERIOD COVERED (Inclusive Dates)

8. PERFORMING ORGANIZATION — NAME AND ADDRESS (If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)

Sandia National Laboratories
Albuquerque, NM 87185

9. SPONSORING ORGANIZATION — NAME AND ADDRESS (If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)

Division of Safety Issue Resolution
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

This volume presents the methodology and results of the internal event accident sequence analysis of the LaSalle Unit II nuclear power plant performed as part of the Level III Probabilistic Risk Assessment being performed by Sandia National Laboratories for the Nuclear Regulatory Commission. The total internal core damage frequency has a mean valve of 4.41E-05/R-yr. with a 5th percentile of 2.05E-6/R-yr., a median value of 1.64E-05/R-yr., and a 95th percentile of 1.39E-04/R-yr. The dominant sequences involve a loss of off-site power (LOSP), immediate or delayed failure of on-site AC power resulting in station-blackout, and failure of the reactor core isolation cooling system (RCIC). The events most important to risk reduction are: frequency of LOSP, non-recovery of offsite power within one hour, diesel generator (DG) cooling water pump common mode failure, and non-recoverable isolation of RCIC during station blackouts. The events most important to risk increase are: failure of various AC power circuit breakers resulting in partial loss of onsite AC power, failure to scram, and DG cooling water common mode failure. The dominant contributors to uncertainty are: control circuit failure rates, relay coil failure to energize, energized relay coils failing deenergized, frequency of LOSP, and DG failure to start.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

PRA
RMIEP
LaSalle
Level I

13. AVAILABILITY STATEMENT

Unlimited

14. SECURITY CLASSIFICATION

(This Page)
Unclassified

(This Report)
Unclassified

15. NUMBER OF PAGES

16. PRICE

NRC FORM 335 (2-89)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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PENALTY FOR PRIVATE USE, $300

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