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

Summary

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Prepared for
U.S. Nuclear Regulatory Commission

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ABSTRACT

This volume presents an overview of the methodology and results of the integrated accident sequence analysis (Level I) of the LaSalle Unit II nuclear power plant performed as part of the Level III PRA conducted by Sandia National Laboratories for the Nuclear Regulatory Commission. The Level II/III results are presented in associated reports described in the Foreword. This volume contains a summary description of the LaSalle plant, describes the contents of the other nine volumes of this report, their relationships to each other, and the relationship of the LaSalle program to other programs. A step-by-step summary description of the methodology and new techniques used to perform the analysis is presented and discussed. The final results of the Level I analyses for each subanalysis (e.g., internal, fire, flood, and seismic analyses) are discussed individually and the final integrated result obtained by merging all subanalyses and performing an integrated calculation is also discussed. General insights and conclusions from the analysis are discussed.

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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-4332 - 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 XXSOR 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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EXECUTIVE SUMMARY

S.1 OBJECTIVE AND SCOPE

The objectives of the Level I portion of this probabilistic risk assessment (PRA) were:

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 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 up to the point where core damage begins;

3) To evaluate PRA technology developments and formulate improved PRA procedures;

4) 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; and finally

5) To conduct a PRA on a BWR 5, Mark II nuclear power plant, ascertain the plants' dominant accident sequences, evaluate the core and containment response to accidents, and calculate the consequences of the accidents up to the point where core damage begins.

In this study, the term integrated risk assessment means the combination of the various constituent analysis (i.e., internal, seismic, fire, and flood) to form an expression for core damage which includes contributions from all initiators. In subsequent portions of this analysis (i.e., Level II/III), the term integrated risk assessment takes on an expanded meaning. That is, integrated risk assessment includes both the frequency of the accident as well as the resulting consequences for the various constituent analyses (i.e., internal, seismic, fire, and flood).

The scope of the Level I analysis includes:

1) Analysis of full power operation of the LaSalle Unit 2 nuclear power plant,

2) Analysis of core damage accidents that result from both internal and external events,

3) Estimation of the integrated frequency of core damage accidents from internal and external events, and

4) Estimation of the effects of the uncertainties of various mechanical failures and phenomenological occurrences on the combined uncertainty of the final integrated core damage frequency.

S.2 METHODOLOGY

It was recognized at the beginning of this project that current methods in use at the time would not be adequate to satisfy the objectives of the analysis. A parallel project was started to identify the limitations in the methods current in PRA analysis at the start of the project and to develop techniques to extend these methods or develop new methods where needed.

Because of the objective to integrate external events into a common framework with internal events, it was decided to extend the level of detail of the fault tree models to include components that would be affected by and failure modes that might be induced by external events. The result was the inclusion of components such as piping and cables which would not normally be included in an internal event PRA because their passive failure probability would be very low, the modeling of control and actuation circuitry in detail in order to represent accurately the effect of the external events on the systems, and the inclusion of failure modes such as spurious actuation. These additions to the fault trees resulted in very large and logically complicated fault trees which were difficult to solve. New techniques were developed using the SETS code to overcome these difficulties. These techniques were used in NUREG-1150 and were essential in allowing efficient solution of the fault trees in that analysis.

Since many external events involve failures of all components within a common location, methods were developed for mapping the location of all components modeled in the fault trees (including the tracing of all cables and pipes). This allowed the evaluation of location based failures induced by events such as fire and flooding and simultaneous inclusion of multiple random failures consistent with the overall probabilistic truncation probability. This means that standard cut set representations of a fire-induced failure in a certain location combined with random equipment failure in other locations could be generated. These could be combined with the regular internal event sequences to form an integrated representation of core damage from all initiating events. A simplified version of this was used in the NUREG-1150 fire analysis for the Peach Bottom and Surry plants.

In addition to extending the fault tree analysis in level of detail, the number of systems analyzed was also increased to include balance-of-plant systems that could respond to an accident (e.g., main feedwater and condensate) and their support systems (e.g., normal service water and non-safety electric power). These systems were modeled in the same level of detail as the standard safety systems. This was done in order to more accurately model the plant response and to include the effect of interactions between balance-of-plant systems and safety systems. The primary motivation for this was the observation that many past PRAs had identified subtle interactions between safety and balance-of-plant systems and these were felt to be even more important when external events, which can affect many systems at once, were to be analyzed.

The accident sequence event trees were extended to include the interaction with the containment and reactor building response in order to evaluate the interaction between sequence phenomenology and system performance. A method was developed to quantify these interactions and a simplified version was used in the NUREG-1150 BWR analysis to resolve the core-vulnerable sequences. The method involved detailed thermal-hydraulic code modeling of the containment and reactor building in order to evaluate the severe environments that could be generated by the various accident sequences, identification of the components modeled in the fault tree that would be subject to these environments, expert elicitation on the failure probabilities of components in these environments, and quantification of the system failure probabilities in the form of cut sets which could be integrated with the standard random failures. In order to correctly model the accident sequence evolution and identify realistic success criteria, 49 thermal-hydraulic calculations were performed (6 using RELAP5, 4 using MELCOR, and 39 using LTAS). The end result of this process is the most detailed and comprehensive PRA plant model to date.

In order to quantify this model and satisfy the objective of identifying, evaluating, and displaying the uncertainties; extensive data analysis was performed and a new code, TEMAC, was written to evaluate the uncertainties in the Level I results and to perform various importance calculations using Latin Hypercube sampling (stratified Monte Carlo). The random failure data for all components was reevaluated, a new model for quantifying human interactions based on the results of simulator studies was developed (this formed the basis of EPRI's simulator studies), an extensive fire data base was developed, a new method for calculating loss of offsite power and fire initiating event frequency was developed, and a new method for calculating loss of offsite power recovery was developed. All of this data was used to various extent in the NUREG-1150 analysis. Expert elicitation was used to quantify issues such as severe environment failure of components. The LaSalle issues were included in the NUREG-1150 Level I expert elicitation process. The data from the internal events analysis and external event analysis was all put into a similar form and used to quantify the model including uncertainties.

S.3 RESULTS AND CONCLUSIONS

Integrated results were obtained by merging all of the accident sequences' cut sets from the LOCA, transient, transient-induced LOCAs, and anticipated accidents without scram accident sequences resulting from internal initiators with the cut sets from the fire, flood, and seismic analyses accident sequences. The final dominant accident sequences were determined and the integrated risk reduction, risk increase, and uncertainty importance measures were obtained. Also, an overall ranking of the dominant cut sets was obtained.

The total core damage frequency at LaSalle from all events has a mean value of 1.01E-04/yr. with a 5th percentile of 5.34E-6/yr., a median value of 2.92E-05/yr., and a 95th percentile of 2.93E-04/yr. This result is considered to be low given that all initiators (both internal and external) are included in this number and that this is the first time that a detailed PRA has been performed on this plant. Usually, the first time a PRA is performed certain design faults are found that lead to accidents that have significantly higher frequencies of occurrence than they would have without the design faults. At LaSalle, because of the generally good design and high redundancy of BWR type nuclear power plants, while some design deficiencies were found, none compromised redundancy to the point where they created accident sequences which were significantly higher in frequency than those from other sources.

Figure S-1 shows cumulative distribution functions (CDFs) for the fire, flood, seismic, and internal core damage frequencies and the integrated core damage frequency for comparison purposes. Figure S-2 has pie charts showing the relative contributions of accident sequences from various categories of initiators to the mean integrated core damage frequency. These categories are: seismic, fire, flood, and internal with internal broken into LOCAs, ATWS, transients, and transient-induced LOCAs. Figure S-3 has a pie chart showing a finer breakdown of the contribution of internal events initiators to the total mean internal core damage frequency. The internal initiators are broken into: 1) LOSP, 2) AC Bus Failure (T101, T102), 3) DC Bus Failure (T9A, T9B), 4) Turbine Trip (T1, T2), 5) Loss of Feedwater (T3, T4, T5), and 6) All Others.

By examining the above plots and figures, one can see that seismic sequences do not contribute significantly to the integrated core damage frequency at LaSalle. Flood sequences are moderate contributors at all quantiles of the distribution. Since the integrated core damage frequency distribution is very similar to the internal events core damage frequency distribution in all but the 90 to 100th quantile range, the integrated core damage frequency distribution comes mostly from internal events. However, at the very top of the distribution, one can see that the fire sequences contribution actually becomes greater than that for the internal sequences. This occurs at about the 95th percentile. The dominant fire sequence is initiated by a control room fire and the sparse fire data for calculating control room fire initiating event frequencies results in a distribution with very wide uncertainty bounds. The mean value of the fire core damage frequency is dominated by a few of the 400 Latin Hypercube observations and, in these cases, the fire contribution can be substantial.

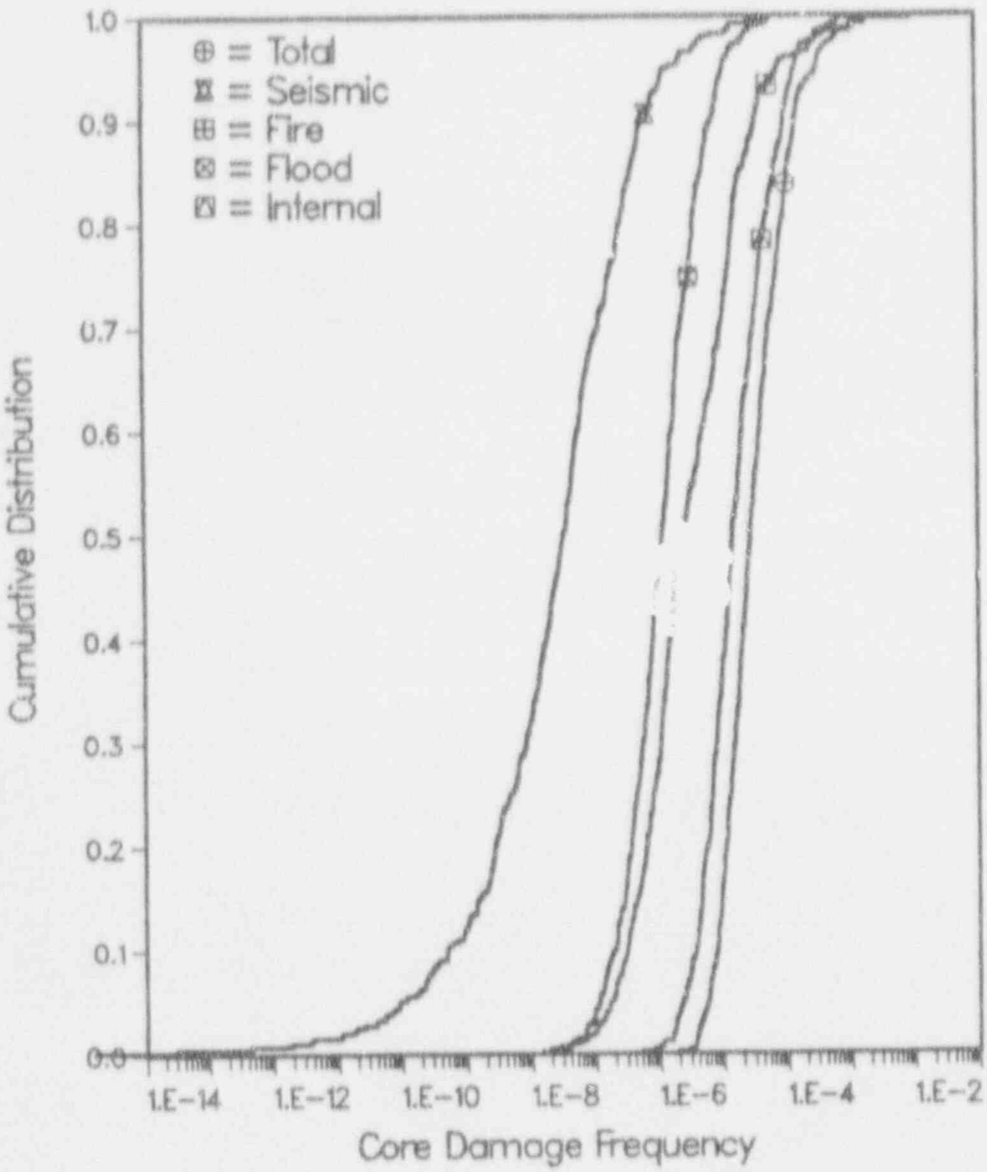
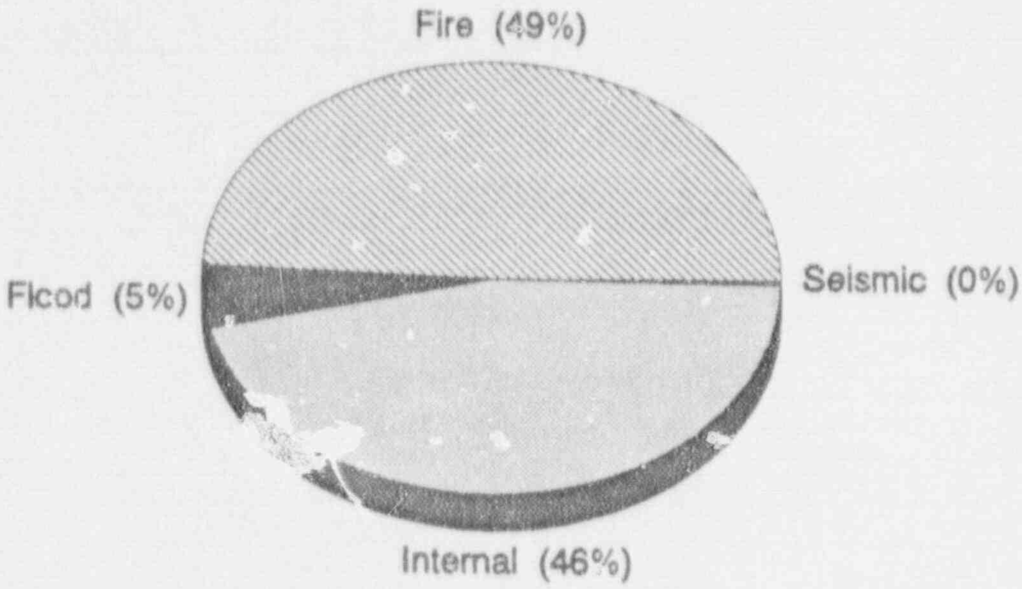
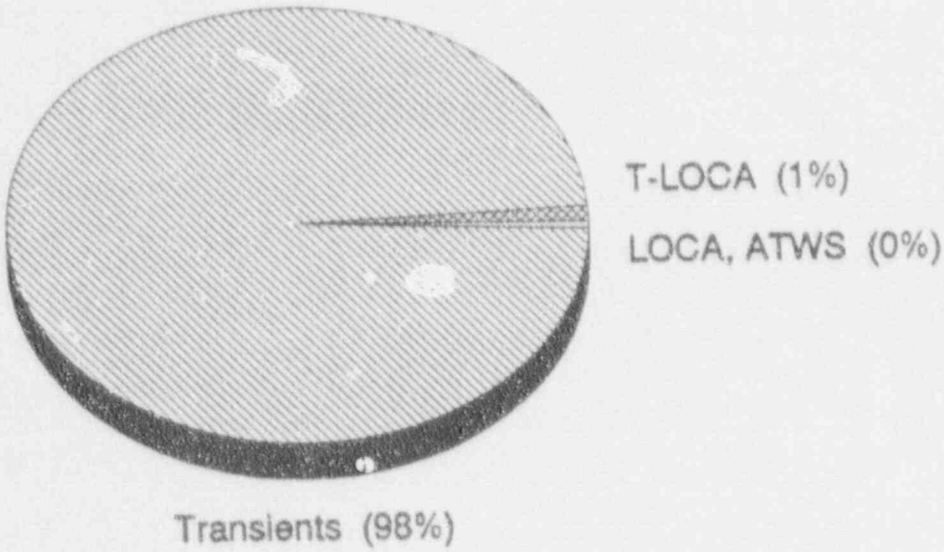


Figure S-1 Fire, Flood, Seismic, Internal, and Integrated Core Damage CDFs.



a. Percentage of Total Core Damage Frequency



b. Percentage of Internal Core Damage Frequency

Figure S-2 Contribution to Integrated Core Damage Frequency.

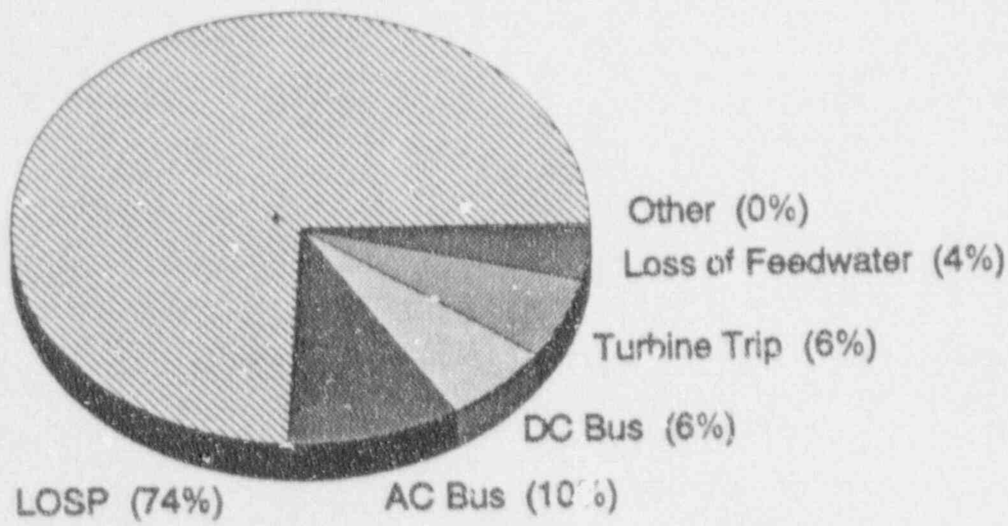


Figure S-3 Initiator Contribution to Internal Core Damage Frequency.

The dominant accident, 35.4% of the mean core damage frequency, involves a loss of all injection as a result of failures occurring after a loss of offsite power. The dominant cut sets of this sequence represent a short-term station blackout type accident. The second most likely sequence, 17.2% of the mean core damage frequency, is the result of a control room fire which is not suppressed and becomes large enough to require evacuation of the control room. Auto actuation of the systems fails as a result of the fire and the operators do not operate the remote shutdown panel correctly due to the high stress. Loss of all injection occurs and short term core damage results.

The events most important to risk reduction are: the frequency of loss of offsite power, the frequency of control room fires, the percentage of control room fires that are not extinguished before smoke forces abandonment of the control room, the probability that the operators will [illegible] successfully recover the plant from the remote shutdown panel, the [illegible] overy of offsite power within one hour, the diesel cooling water [illegible] mmon mode failure, and the non-recoverable isolation of RCIC [illegible] ation blackouts.

[illegible] ts most important to risk increase are: the internal flood pipe [illegible] frequency, the failure of various AC power circuit breakers [illegible] ng in partial loss of onsite AC power, the failure to scram, and [illegible] 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 the control room fire initiating frequency, the uncertainty in relay coil failure to energize, the uncertainty in energized relay coils failing deenergized, and the uncertainty in the response of systems to severe environments in the reactor building.

1.0 INTRODUCTION

1.1 Objective and Scope

The objectives of the Level I portion of this probabilistic risk assessment (PRA) were:

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2) 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 up to the point where core damage begins;

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4) 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; and finally

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The scope of the Level I analysis includes:

1) Analysis of full power operation of the LaSalle Unit 2 nuclear power plant,

2) Analysis of core damage accidents that result from both internal and external events,

3) Estimation of the integrated frequency of core damage accidents from internal and external events, and

4) Estimation of the effects of the uncertainties of various mechanical failures and phenomenological occurrences on the combined uncertainty of the final integrated core damage frequency.

1.2 General Description of the Plant

The LaSalle Unit 2 nuclear power plant is located in the area of Brookfield Township, LaSalle County, Illinois which is 55 miles southwest of Chicago. The LaSalle plant utilizes a Mark II type containment to house a General Electric BWR-5 reactor with a rating of 3293 MWt. The reactor is owned and operated by Commonwealth Edison Company.

There are various injection systems that can be used to cool the core and prevent core damage at LaSalle. Four high pressure and four low pressure injection systems are considered in this analysis. Detailed descriptions and system drawings can be found in Volume 6 of this report, System Descriptions and Fault Tree Definition.

The high pressure injection systems include the high pressure core spray system (HPCS), the reactor core isolation cooling system (RCIC), the main feedwater system (MFW), and the control rod drive system (CRD). The HPCS system has a motor-driven pump with its own dedicated diesel (train C of emergency power). This system draws water from either the condensate storage tank (CST) or the suppression pool and sprays coolant onto the core via a ring sparger located above the core. The RCIC system utilizes a turbine-driven pump. Steam from the reactor pressure vessel (RPV) is used to drive the turbine which pumps water from either the CST or the suppression pool back to the vessel via an injection nozzle in the reactor vessel dome. Because RCIC takes steam from the RPV, operation of the system can not be assured once vessel pressure falls below 57 psig. Also, RCIC isolates when containment pressure reaches about 15 psig. Train A DC power is also required to control this system. The MFW system draws water from the condenser hotwell using two turbine-driven pumps and one motor-driven pump and injects it into the vessel through the main feedwater lines. These pumps require offsite power (i.e., not emergency power). The CRD system can be used to inject water into the vessel via the control rod drives into the lower plenum. The CRD system can only inject several hundred gallons per minute and is therefore only useful once the decay energy has been significantly reduced (i.e., during a long term accident) or in conjunction with another injection system. The high pressure injection systems can be used to provide coolant makeup when the RPV is at either high or low pressure. The only caveat to this statement is that the emergency operating procedures require the RPV pressure to be above 57 psig if RCIC is to be used.

The low pressure injection systems include the low pressure core spray system (LPCS), the low pressure coolant injection system (LPCI), the condensate system (CDS), and the diesel-driven firewater system (DFWS). The LPCS system is a single train system that draws water from the suppression pool using a motor-driven pump. This system is powered by train A of the emergency power system. LPCS sprays coolant into the vessel through a ring sparger located above the core. The LPCI system is a three train system that also draws water from the suppression pool using motor-driven pumps and injects into the core bypass region of the vessel. Train A of LPCI is powered by train A of the emergency power system (EPS) and trains B and C are powered by train B of the EPS. The condensate system draws water from the condenser hotwell and pumps it through the feedwater line into the RPV using four motor-driven condensate pumps. Both MFW and CDS can take water from the CST (limited to a maximum of 1000 gpm) and must be throttled to maintain net positive suction head (NPSH) in this mode. This system requires offsite power. The last resort injection system that is used when all other systems have failed is the diesel-driven firewater system. This system can be manually connected to the MFW injection line to provide injection. The DFWS uses diesel-driven pumps to draw water from the ultimate heat sink. Because this system has its own dedicated diesel-driven pumps, it can operate during a station blackout event. For all of these low pressure injection systems to provide coolant to the core, the RPV must be depressurized.

The Automatic Depressurization System (ADS) is designed to depressurize the reactor vessel to a pressure at which the low pressure injection systems can inject coolant into the reactor vessel. The ADS consists of seven of the eighteen relief valves. Each valve is capable of being manually opened. For the system to be automatically initiated, a low pressure emergency core cooling (ECCS) pump must be running. Thus, the ADS will not be automatically initiated during a station blackout. The operator can also manually initiate the ADS, or he may depressurize the reactor vessel using the eleven Safety Relief Valves (SRVs) that are not connected to the ADS logic. Each valve discharges into the suppression pool. The ADS valves are located in the drywell, and drywell pressures of approximately 85 psig will prevent opening the valves or result in reclosure if they are already open. The ADS also requires at least one train of DC power. Therefore, the RPV can not be depressurized in sequences that involve failure of all DC power or in accidents in which the containment pressure exceeds 85 psig.

Heat can be removed from the containment by the residual heat removal (RHR) system which uses trains A and B of the LPCI system. Suppression pool cooling (SPC) and the containment spray system (CSS) are two modes of the RHR system. The RHR system is a two train system with motor-operated valves and pumps. Each train has a heat exchanger downstream of the pump. In either the SPC or the CSS modes of operation, the RHR system can remove heat from the suppression pool by passing water from the pool through the heat exchangers (with service water on the shell side). In the SPC mode, the water is injected directly back into the suppression pool and, in the CSS mode, the water is sprayed into the drywell atmosphere and drains back into the suppression pool via the drywell downcomers. For accidents that are not LOCAs, the shutdown cooling (SDC) mode of RHR can also be used to remove decay heat from the core. In this mode of operation, water is drawn from the recirculation loops, passed through the RHR heat exchangers, and then returned to the vessel via the recirculation loops. All three modes of RHR (i.e., SPC, CSS, and SDC) require at least one train of emergency AC power and are, therefore, unavailable during a station blackout. By operating the appropriate heat exchanger, train A or B of the LPCI mode can also provide the function of containment heat removal by drawing water from the suppression pool, passing it through the heat exchanger, and injecting the water directly into the vessel. The water then must flow back to the suppression pool either via a break in the primary system piping or via the SRV discharge lines.

The interaction between the injection systems and the primary and secondary containment environments is accounted for in this Level I analysis. Severe environments can be created in the reactor building from containment failure modes that result in steam release to the reactor building (i.e., wetwell or drywell failures) or containment venting (which results in a release of steam into the upper floors of the reactor building). The subsequent effect on injection system components in the reactor building is accounted for in the accident sequence definition. Containment failure via the drywell head goes to the refueling floor and bypasses the reactor building with no concomitant severe environments in the reactor building. The effects of primary containment pressure on system operability are also considered (e.g., RCIC and ADS as mentioned above).

The primary containment is a post-tensioned reinforced concrete structure with a steel liner. The containment, shown in Figure 1.3-1, consists of a lower cylindrical portion founded on the base mat and an upper portion that is in the form of a frustum of a cone. The containment is topped by an elliptical steel dome called the drywell head. The lower portion is called the suppression chamber (or wetwell) and it contains the suppression pool; the upper portion is called the drywell and it houses the reactor pressure vessel (RPV). The primary containment is enclosed by a reinforced concrete reactor building which forms the secondary containment. The primary containment is inerted with nitrogen which eliminates the possibility of hydrogen combustion events during the course of the accident. However, combustion of hydrogen in the reactor building following containment failure is still possible. The internal design pressure of the primary containment is 45 psig. The ultimate containment failure pressure was assessed by a panel of structural experts (see appendix B.7 of the Level II/III analysis in NUREG/CR-5305, Volume 2). The assessed mean failure pressure is 191 psig; the minimum and maximum failure pressures are 140 psig and 275 psig, respectively. The containment failure locations identified by the expert panel included the drywell head, the drywell wall, the wetwell wall above the suppression pool, and the wetwell wall below the suppression pool surface.

The pressure suppression system is the over-and-under configuration. The drywell is located in the upper portion of the containment directly above the suppression chamber which forms the lower portion of the containment. The drywell and the suppression chamber are separated by a reinforced concrete slab which forms the drywell floor. The drywell houses the reactor pressure vessel (RPV) and much of the primary system. The

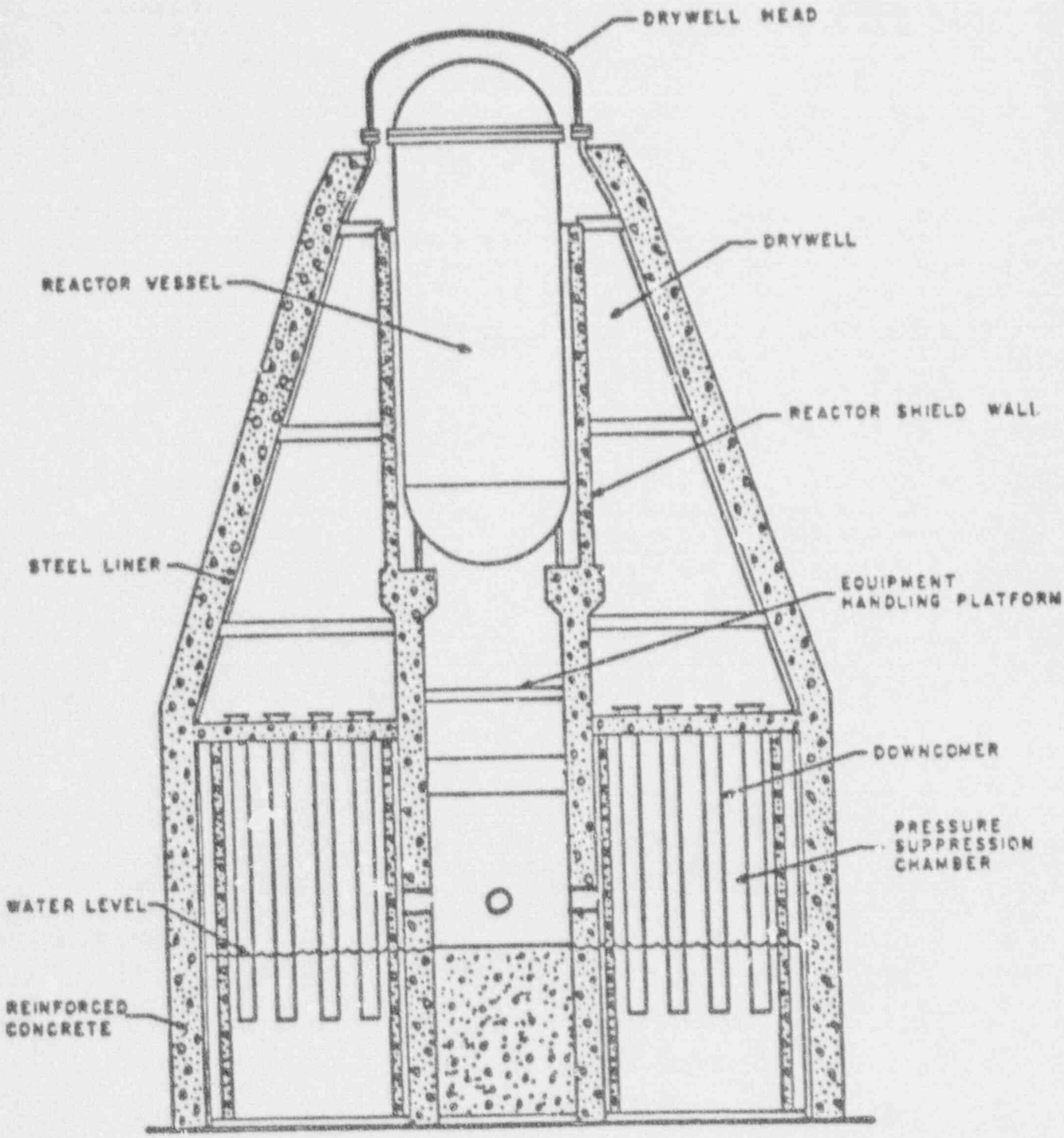

Figure 1.3-1. LaSalle Containment Schematic

suppression chamber contains the suppression pool. The drywell and the suppression chamber communicate through passive vertical vents called downcomers. One end of each downcomer is in the drywell and the other end is submerged in the suppression pool. Gases released in the drywell are vented through the downcomers into the suppression pool where the steam is condensed and the noncondensibles are cooled. In the event that the suppression chamber pressure exceeds the drywell pressure, the noncondensibles that have accumulated in the suppression chamber air space are vented back into the drywell through the drywell vacuum breakers and thereby equilibrate the pressure between the two volumes. The suppression pool is also used to condense the steam and cool the noncondensible gases that are released through the safety relief valve (SRV) tailpipes when the RPV is depressurized. The SRV tailpipes direct the steam from the RPV to suppression pool when the ADS or SRV valves are opened. The tailpipes release the steam and gases through T-quenchers located at the end of the tailpipes near the bottom of the suppression pool. The nominal free volumes of the drywell and the suppression chamber are 219,800 ft^3 and 165,100 ft^3, respectively. The nominal volume of the suppression pool is 128,800 ft^3.

Directly below the reactor pressure vessel is the reactor pedestal cavity. The cavity is large enough to contain all of the core debris should core damage occur and the vessel fail. In addition to holding the core debris, the cavity can also accumulate a large volume of water during the accident. When the cavity is completely flooded, a water depth of 11 feet can be established. The potential for large amounts of water to be in the cavity has implications for the Level II analysis of the accident progression after core damage and vessel breach have occurred.

The LaSalle containment can be vented in the event that the pressure cannot be controlled. For long term containment heat removal accidents and ATWS scenarios, the containment pressure will steadily increase due to the steam released from the saturated suppression pool. The pressure in the containment can be relieved through the containment vent and purge system. The containment can be vented from either the drywell or the suppression chamber using either a 2 inch valve or a 26 inch valve. The vent pipe ties into the standby gas treatment system (SGTS) which releases the gases to the stack. The vent pipe is attached via an 18 inch pipe to the SGTS with a rubber boot. It was assumed that this rubber boot will fail when high pressure steam is released through the vent. Therefore, the steam will be released into the reactor building rather than being directed to the stack when the containment is vented. Inundation of high temperature steam in the reactor building creates a severe environment for motor control cabinets and other equipment located in the reactor building. Failure of equipment due to this steam can result in the loss of vital emergency equipment (e.g., coolant injection systems and containment heat removal systems).

The operators are instructed to vent the containment when the containment pressure exceeds 60 psig regardless of whether or not adequate core cooling is available. Venting requires both divisions of AC power.

1.3 Structure of the Report

This report consists of ten volumes. Each volume presents the results of a specific portion of the analysis.

In this volume, we present an overview of the methodology used to perform this analysis, a summary of the individual analysis results, a summary of the results of the integrated analysis, and insights and conclusions gained as a result of the analysis.

In Volume 2, we present the details of the construction of the Latin Hypercube Sample (LHS), summarize the inputs into the integrated analysis from the internal, fire, and flood analyses, present the final quantification of the seismic sequences, present the details of how the integrated analysis was performed, and then present the final results.

In Volume 3, we present the details of the quantification of the internal event accident sequences. We discuss the procedure for calculating the sequence cut sets, the initial screening quantification of the accident sequences, the application of recovery actions, the quantification of the effects of severe environments created in the primary containment and reactor building on systems responding to the accidents, the reevaluation of the data, and the final quantification of the accident sequences.

In Volume 4, we discuss the internal analysis initiating event identification and quantification and the construction of the functional and systemic accident sequence event trees which were used for both the internal and external accident sequence analyses.

In Volume 5, we discuss the preliminary selection of the data for point estimates of the probability of occurrence of the hardware failures appearing in the system fault trees, the final selection of data distributions for the important failure mechanisms remaining in the final analysis, the initial human factors analysis, and the common cause analysis.

In Volume 6, we present the system descriptions. These descriptions form the bases for the construction of the system fault trees and contain information on system layout and operation.

In Volume 7, we present the results of the external event scoping study. In this study we evaluated a range of external initiators to determine if detailed analyses would be performed as part of this study. No external events other than seismic, internal fires, and internal floods were found to be important enough to warrant detailed analysis for this study.

In Volume 8, we present the results of the initial seismic analysis. The construction of the hazard curve, the response analysis, the structural analysis, the fragility analysis, and the use of the internal event logic models (both event trees and fault trees) to define and evaluate the seismic accident sequences are described.

In Volume 9, we present the results of the fire analysis. The location analysis in which the locations of all equipment and cabling used in the fault tree models is described, the method of incorporating this information directly into the system fault trees and evaluating the accident sequences using the same accident sequence and system models used in the internal events analysis is described. The quantification of the fire initiating event frequencies is discussed and the definition of the fire scenarios and final quantification of the accident sequences is presented.

In Volume 10, we present the results of the flood analysis. The location analysis in which the locations of all equipment and piping used both in the fault tree models and in the balance of plant systems, not included in the accident analysis model, is described. The method of incorporating this information directly into the system fault trees and evaluating the accident sequences using the same accident sequence and system models used in the internal events analysis is also described. The quantification of the flood initiating event frequencies is discussed and the definition of the flood scenarios and final quantification of the accident sequences is presented.

1.4 Relationship to Other Programs

Because of the size and scope of the LaSalle analysis, an extensive planning activity was undertaken to identify all of the tasks in the analysis, which programs would be responsible for the performance of each task, and what the input requirements and output products of each task would be. The Risk Methods Integration and Evaluation Program (RMIEP) at Sandia National Laboratories (SNL) was given overall responsibility for the LaSalle PRA. In addition to coordinating all the various programs contributing to the PRA, the RMIEP program also was responsible for quality assurance, interfacing with the utility to obtain and supply needed plant information to all the other programs, the performance of the internal events Level I analysis, the performance of the internal flooding Level I analysis, the integrated Level I analysis, and the location analysis done in support of the fire and flood analyses. Separate NRC programs working in conjunction with RMIEP performed certain portions of the analysis.

The PRA Methods Development Program at SNL was responsible for developing procedures to consistently model internal and external system faults. This included the incorporation of passive failures such as piping, spurious actuation failures, and cabling into the fault trees. Methods for incorporating location based failures into the internal event system fault trees for the consistent and integrated evaluation of the seismic, fire, and flood accident sequences were developed. The program also identified and evaluated the various human reliability analysis techniques for their use in RMIEP. In conjunction with the PRUEP program performing the Level II/III analysis, methods for performing uncertainty analysis were investigated.[1]

The Integrated Dependent Failure Analysis Program at SNL was responsible for the development of methods for identifying, and modeling common cause failures due to internal event common causes such as grit, vibration etc. and certain external event common cause such as fire and flood.

The Seismic Safety Margins Research Program (SSMRP) at Lawrence Livermore National Laboratories (LLNL) was responsible for the seismic Level I screening analysis. This involved determining the seismic hazard curve for LaSalle, performing the response analysis, the hydrodynamic load analysis, and the structural and equipment fragility analyses. Using the detailed system fault trees enhanced to include piping failures and the accident sequence fault trees from the RMIEP internal event analysis, they solved for the seismic accident sequence cut sets and quantified the results. These results along with uncertainty distributions for the seismically-induced failures and the seismic hazard curve were given back to RMIEP for application of recovery, final quantification, and inclusion in the integrated core damage analysis. See Volumes 2 and 8 of this report for a more complete description of this process.

The Division of Risk Assessment (DRA) Fire Program at SNL was responsible for the identification and quantification of the fire initiating events, the definition of the fire locations and zones, the definition of the fire scenarios, the calculation of the fire propagation and effects, and the quantification of the fire teams emergency response. A fire data base was developed for use in quantifying the fire initiating event frequencies. The location and zone definitions and the zone propagation table were supplied to the RMIEP program which then identified all the cabling important to the modeled system, mapped all the components and cables in the system fault trees into location space, and solved for the accident sequence cut sets containing both location and random failures. The locations were transformed to fire zones and cut sets which were physically unrealizable were eliminated. Recovery actions were applied to the random failures taking into account the possible effects of the specific fire in the cut set. After determination of the dominant cut sets, the cabling in each important location was mapped. The fire program then determined fire scenarios to be evaluated for each location, COMPBRN models were constructed for each scenario, the COMPBRN code[2] was improved, and calculations were performed on all the important scenarios. An evaluation of the fire response was performed and all this information was fed back to the RMIEP program to perform the final sequence quantification and incorporate the results into the integrated analysis. See Volume 9 of this report for a more complete description of this process.

The DRA Data Program at Oak Ridge National Laboratories (ORNL) which developed the In-Plant Reliability Data System (IPRDS) was responsible for providing supporting data for the quantification of component faults appearing in the system fault trees. The data program was redirected to collect data from several BWRs deemed similar to the LaSalle plant in some respects. This data was then analyzed at Los Alamos National Laboratory (LANL) using the FRAC[3] code to determine component failure rates and possible factors that might contribute to different failure rates for the same component. This data was supplied to RMIEP as additional information for the determination of the final data distributions. See Volume 5 of this report for a complete description of the data analysis methods and results.

The Phenomenology and Risk Uncertainty Evaluation Program (PRUEP) at SNL was responsible for the performance of the Level II/III analysis. This consisted of the accident progression analysis, the source term analysis, the consequence analysis, the final integrated risk evaluation, and MELCOR[4] code calculations to evaluate the evolution of several dominant severe accident sequences. The LaSalle analysis was the first application of MELCOR to integrated accident sequence evaluations. See Volume 3 of the Level II/III analysis reported in NUREG/CR-5305 for a complete description of the MELCOR code calculations performed for this analysis.

The MELCOR program at SNL helped in the review of the MELCOR calculations performed by PRUEP and provided support to fix code problems encountered in the course of performing the calculations.

A Brookhaven National Laboratories (BNL) human error study[5] took preliminary accident sequence cut sets after the application recovery but before final quantification for the dominant internal event accident sequences at LaSalle and performed various sensitivity studies involving the human error rates.

A Future Resources Associates relay chatter study[6] for seismic events took system fault tree information and evaluated the potential effects of the chattering of relay contacts as the result of a strong earthquake.

A PRA based inspection of the LaSalle plant was performed by the NRC in 1986.

Finally, the LaSalle simulator studies[7] were the precursors to the EPRI simulator studies.

1.5 Contributions to NUREG-1150

The LaSalle PRA was started before the NUREG-1150 analysis and was performed concurrently with NUREG-1150 when that program started. The LaSalle program was significantly impacted by NUREG-1150. Resources from the LaSalle analysis were diverted to NUREG-1150 which was much higher in priority and the LaSalle schedule was stretched out considerably. However, work on the LaSalle analysis was never halted completely and there were many interactions between the two programs.

The LaSalle analysis is a much more detailed analysis than that performed for NUREG-1150. However, in many cases, methods developed for LaSalle were simplified for use in or used directly in NUREG-1150. The following is a partial list of the contributions of the LaSalle Level I analysis to the

NUREG-1150 program. The contributions to the Level II/III analysis are described in the PRUEP reports.

1. As a result of the level of detail of the LaSalle models, some interesting interactions between systems and within systems were found. Interactions between safety and non-safety systems and interactions between isolation logic and components were identified. The NUREG-1150 plants were examined for interactions with similar characteristics.

2. A detailed method for defining plant damage states for the Level I/II interface was developed for the LaSalle analysis and used for the Peach Bottom NUREG-1150 analysis. The other NUREG-1150 plants used a simplified version of this method. See Volume 1 of the Level II/III reports in NUREG/CR-5305 for a complete description of this method.

3. A method for resolving core vulnerable sequences appearing on the accident sequence event trees was developed for the LaSalle PRA. The method used in the Peach Bottom analysis is very similar to that used for LaSalle. See Volume 3 of this report for a complete description of this method. The following is a brief description:

   a) A detailed MELCOR model of the LaSalle reactor building was constructed and calculations of severe environments for different containment failure modes were performed.

   b) The Level I NUREG-1150 expert elicitation panel was given this environmental information and list of equipment used in BWR systems that might appear in the reactor building. They determined equipment failure probabilities for various ranges of severe environments.

   c) Simplified system models were constructed and quantified using this information and the feedback to the mitigating systems was evaluated in the event trees.

   d) For the LaSalle analysis, the Level I and Level II LHS samples were consistent (i.e., they used the same samples values for the Level I and II analysis and both sampled the containment failure pressure and modes in the same manner); but, for Peach Bottom, this level of integration was not achieved (i.e., the Level I analysis used mean values for the probability of containment failure while the Level II analysis sampled the containment failure similar to the LaSalle Level II analysis).

4. The TEMAC[8] code was developed for the LaSalle PRA to evaluate uncertainty and importance measures for the accident sequence frequency represented by the sequence cut sets. This code was used in NUREG-1150.

5. In order to evaluate the loss of off-site power initiating event frequency distribution and probability distributions for the recovery of off-site power by a certain time, a new computer code was written.[9] This code was used to evaluate the loss of off-site power frequencies and recovery probabilities for all the NUREG-1150 plants. In addition, this code was used to evaluate the fire initiating event frequencies for Surry, Peach bottom, and LaSalle.

6. The location based methodology used to perform the Surry and Peach Bottom fire analysis was developed in the LaSalle PRA and a simplified version was used in NUREG-1150. See Volume 9 of this report for a complete description of this method.

7. Advanced methods were developed in the LaSalle analysis for using the SETS code to solve for the accident sequence cut sets. These methods were used in the Peach Bottom analysis. See Volume 3 of this report for a complete description of these techniques.

8. The generic data base used in NUREG-1150 was influenced by the LaSalle generic data base. See Volume 5 of this report for a complete description of the generic data base.

9. Significant upgrades were made to the COMPBRN code used to perform the fire propagation analysis for various fire scenarios. See Volume 9 of this report for a description of these upgrades. This version of the code was used for the NUREG-1150 fire analyses.

10. The fire data base[10] developed for the LaSalle analysis was used to calculate the fire initiating event frequencies. See Volume 9 of this report for a description of the results.

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2.0 DESCRIPTION OF THE METHODOLOGY USED TO PERFORM THE INTEGRATED ANALYSIS

2.1 Outline of the General Steps Used in the Analysis

One of the primary purposes of the LaSalle PRA was to develop methods for incorporating external event analysis into the PRA on an equal footing with the internal events analysis. A second primary purpose was to represent the uncertainty in the various analyses in a uniform way and to propagate this uncertainty through the analysis to obtain a final integrated result that represented the contribution to core damage from all types of initiators and displayed the uncertainty in the overall result. The relative importance of the contributors from the different initiators could then be evaluated in a consistent manner.

In order to realize these goals, improvements were made in the state-of-the-art of PRA techniques. Improvements were made in various aspects of the internal and external event modeling of the plant. New computer codes and techniques were developed in order to solve the large and detailed models. Data was analyzed and represented in a uniform manner for both internal and external events. New techniques and codes were developed to perform the integrated uncertainty and importance analysis.

The general process used to analyze the accident sequences and obtain the core damage frequency can be broken down into the following series of steps:

1. Define the initiators to be analyzed. This involves a screening analysis of external events as described in Volume 7 of this report and the detailed internal event initiator search 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 was done 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 was done in Volume 6 of this report. Include in the models any additional detail/components necessary for the external event analyses. The specific location based information needed for the external event analyses is included in the appropriate external event analysis volume of this report (see Volumes 8, 9, and 10).

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 was done for random mechanical failures and screening human errors in Volume 5 of this report. External event specific failures were

developed in the appropriate volume dealing with that external event (see Volumes 8, 9, and 10). Severe environment equipment failure and final human error probabilities are described in Volume 3 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. The basic method used to analyze the internally initiated accidents is reported in Volume 3 of this report. The specifics of the location based method used to analyze the seismic, fire and flood sequences is reported in Volume 8 for the seismic analysis. in Volume 9 for the fire analysis, and in Volume 10 for the flood analysis.

6. Combine these system fault trees into accident sequences using point estimate data to calculate screening estimates of the accident sequences. This analysis is reported in Volume 3 of this report for internal events, in Volume 8 for the seismic analysis, in Volume 9 for the fire analysis, and in Volume 10 for the flood analysis.

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 events representing the failure to mitigate the accident (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, procedure for adding recovery actions to the cut sets, and quantification of the non-recovery probabilities for the internal initiators are reported in Volume 3 of this report. The development of the method of evaluating human actions from simulator data is presented in Reference 1. A review and comparison was conducted of various HRA methods and is reported in Reference 2. The recovery actions specific to the individual external event analyses are reported in Volume 8 and in this volume for the seismic analysis, in Volume 9 for the fire analysis, and in Volume 10 for the flood analysis.

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 (i.e., core-vulnerable sequences, sequences which may still proceed to core damage as a result of the interaction between containment phenomenology and the responding systems). Apply this methodology to resolve the core vulnerable accident sequences. This is reported in Volume 3 of this report.

9. Using the uncertainty distributions developed for the data, quantify each individual accident sequence, the combined sequences for each analysis (internal, fire, flood, and seismic), and the combined accident sequences (i.e., the integrated results) to obtain the individual sequence, individual analysis and integrated core damage frequency distributions. 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. The evaluation of the sequence and integrated result uncertainty distributions and the importance calculations are reported in Volume 3 of this report for internal events, in Volume 2 for the seismic analysis, in Volume 9 for the fire analysis, in Volume 10 for the flood analysis, and in Volume 2 of this report for the final integrated analysis.

2.2 Description of the Methods Used in Each Step

2.2.1 Initiating Event Identification

A detailed review of internal event initiators was conducted and plant specific Failure Mode and Effect Analyses (FMEAs) were conducted to identify plant specific initiating events.

For external events, a screening methodology was developed to identify those general categories of initiators for which detailed analyses would need to be done. This methodology is described in Reference 3 and its specific application to the LaSalle plant is presented in Volume 7 of this report. For each of the external events selected for detailed analyses (seismic, internal fire, and internal flood), plant specific initiating events were defined. This analysis-specific initiating event identification is described in the volumes describing each external initiating event analysis.

The seismic analysis screening results are described in Volume 8 of this report. Using the SSMRP methodology developed at LLNL for the NRC, a plant specific hazard curve with uncertainty bounds was calculated.

The internal fire analysis results are described in Volume 9 of this report. Using a fire data base[4] and a new method for calculating loss of offsite power initiating event frequencies[5] developed for the LaSalle analysis, fire initiating event frequencies were defined for every location used in the analysis. The plant was subdivided into a large number of locations for use in both the fire and flood analysis. Fire frequencies were calculated for each location.

The internal flood analysis results are described in Volume 10 of this report. All piping in each location was traced and specific piping failures were identified as initiators depending on their impact on the systems being used to mitigate the accident.

2.2.2 Accident Sequence Delineation

There are two general methods that can be used to define accident sequences. The first, which has been used in a majority of previous PRAs, involves the construction of a separate accident sequence event tree for each initiator. The effect of the specific initiator on each system is included directly in the accident sequence definition. The second method, which is used in this PRA, is to construct a general accident sequence event tree for a class of initiators and then to model the specific effects of the each initiator in the system fault trees. Only three accident sequence event trees are used in the LaSalle analysis: Transient, LOCA, and ATWS. The external event analyses for seismic, fire, and flood events use the transient event tree.

The development of the accident sequence event trees for the internal events analysis is described in Volume 4 of this report. The specific application to each external event is described in the appropriate volume (see Volumes 8, 9, and 10).

2.2.3 Construction of the System Fault Trees

In addition to reviewing thermal-hydraulic calculations done for the LaSalle FSAR accident analysis[6] and the GE generic BWR accident analyses,[7] thermal-hydraulic calculations were performed using the RELAP5 and LTAS codes to determine realistic system success criteria for specific initiating events. These success criteria were used to define the fault tree top events. The results of these calculations are reported in Volume 4 of this report.

2.2.3.1 Level of Modeling Detail

For the LaSalle 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) to include some information in the fault trees via transformation equations, and (3) to aid in the process of evaluating the accident sequences in an efficient and cost effective manner.

2.2.3.2 System Fault Trees

For each system identified as being able to mitigate an accident, a detailed fault tree was developed. This fault tree included a detailed representation of system pipe failures to represent the direct effects of specific pipe failures on the systems for the seismic and flood analyses and detailed modeling of both control and actuation circuitry to accurately reflect the effects of fire induced failures. As part of this effort, exact locations were obtained for all components represented in the fault tree. The effect of cable failure for fire initiators was represented by identifying all the cables in the modeled circuits, all electrical power cabling, and creating a mapping which attached each cable to the appropriate components in the fault trees so that the effect of failure of the cable would be accurately propagated through the fault tree models. The locations through which the cable passed were identified. Additional mappings were set up to include the pipe and cable locations in the fault tree model. The external event analysis volumes of this report (see Volumes 8, 9, and 10) describe, in detail, the location analysis effort and in appendices present the location transformations.

2.2.4 Data Base

The random failure data base for internal events evolved in a series of steps. First, a complete list of all the types of equipment and the failure modes to be modeled was generated. Second, a generic data base was created in dBase that contained screening values for all of the failure rates. Third, another data base was created that contained the specific component failures appearing in the fault trees and information about their generic failure type and test and operating intervals. A dBase program was then run to calculate the unavailabilities of all the fault tree events using the generic data base and the component specific information. This created the screening data base and is reported in Volume 5 of this report. This screening data base was used in the initial fault tree solution and initial accident sequence evaluation for internal events. A new methodology was developed for determining human error probabilities for the screening analysis and is also described in Volume 5 of this report.

After the screening analysis had been performed, the data for all of the remaining event types were reviewed and probability distributions were generated for all of the remaining failure modes. The generation of these probability distributions is also described in Volume 5 of this report. As part of this reevaluation, a new method[5] was developed for calculating the loss of offsite power initiating event frequency and the probability of non-recovery of offsite power within time t. Uncertainty distributions for both are also created. The IPRDS program at ORNL was re-directed to evaluate data from several BWRs similar to LaSalle and the data was analyzed by LANL using their FRAC code. The final distributions were incorporated into a Latin Hypercube sampling scheme for use in the final accident sequence quantification as described in Volume 2 of this report.

For the fire analysis, a new fire initiating event data base was constructed.[4] This fire data base was analyzed using the same method used for the loss of offsite power analysis to obtain fire initiating event frequencies. Also, separate analyses were conducted to determine probability distributions for: the probability of failure of fire barriers, the percentage of small vs large fires, the probability of suppression of fires in various locations, and the fraction of fires from various causes. This data is presented in Volume 9 of this report.

For the flood analysis, pipe and valve initiating event frequencies were generated as reported in Volume 10 of this report.

The seismic analysis was performed by LLNL, the seismic hazard curve and the plant response data was generated using their SSMRP methodology and used a plant specific structural response analysis as reported in Volume 8 of this report.

2.2.5 Fault Tree Solution

For internal events, the front-line system fault trees were merged with their support systems and solved using the screening data base. Because of the very large size of the LaSalle fault trees, new techniques were developed to solve the trees in an efficient manner. These techniques are described in detail in Volume 3 of this report. The screening cut sets were truncated, based on probability, at 1.0E-08.

For the seismic, fire, and flood analysis, the system fault trees were resolved incorporating the location information through transformation equations. This resulted in system cut sets containing both location based failures and multiple random failures. The random failures were probabilistically truncated in a manner consistent with the 1.0E-08 cutoff used for the internal events analysis. The details of the transformations and system solution methods are described in the respective volumes of this report (see Volumes 8, 9, and 10).

2.2.6 Initial Accident Sequence Evaluation

For both internal and external events analysis, the front-line system fault tree solutions were then combined to create the accident sequence cut sets. As with the fault tree solutions, new techniques had to be developed to obtain the complete sequence solutions because of the very large size of the fault trees and number of intermediate cut sets generated during the solution process. These methods are described in detail in Volume 3 of this report and the specific screening analyses are described in the appropriate sub-analyses volume (Volume 3 for internal, Volume 8 for seismic, Volume 9 for fire, and Volume 10 for flood).

2.2.7 Final Accident Sequence Evaluation

For all of the analyses, the data for the random events was re-evaluated as described in Volume 5 of this report. Recovery actions appropriate for the particular analysis, component and cut set were identified and are described in the appropriate volume of this report (Volume 3 for internal, Volume 9 for fire, and Volume 10 for flood) except for the seismic analysis where the final quantification is described in Volume 2 of this report.

Thermal-hydraulic calculations were performed using the RELAP, LTAS, and MELCOR codes to determine the timing of events for various accident sequences. A new method[1] using the simulator was developed to quantify human error rates where appropriate. The recovery actions were then added to the cut sets.

2.2.8 Resolution of Core Vulnerable Sequences

For accident sequences involving loss of containment heat removal but continued success of primary injection, core damage could occur as a result of the interaction between containment response and phenomena and the injection systems operability. Examples of this are: (1) high containment pressure (i.e., >85 psig) can cause the ADS valves to reclose resulting in the loss of low pressure injection, (2) high containment pressure can result in isolation of the RCIC system, and (3) venting of the containment at 60 psig or structural failure of the containment can result in loss of NPSH for pumps taking suction from the suppression pool (not considered likely at LaSalle due to pump design) or equipment failure for equipment located in the reactor building due to the severe environments created in the building from the blowdown of the primary containment.

The first example was taken into account in the event trees by allowing sequences with only low pressure injection to go to core damage after failure of containment heat removal and venting as a result of repressurization of the vessel upon reclosure of the ADS valves. The second example was accounted for by allowing RCIC to fail when containment pressure reached 30 psig and then requiring some other high or low pressure system for success. The third example was accounted for by adding events to the event trees to determine whether the containment was vented or structurally failed by leakage (containment takes greater than 2 hours to depressurize) or rupture (containment takes less than 2 hours to depressurize). Given various locations and sizes of containment failure, the severe environments created in the reactor building were determined by developing a detailed reactor building model and using the MELCOR code to calculate environmental conditions in the reactor building due to the containment blowdown. Separate expert judgement elicitations were used to evaluate equipment failure probabilities in the severe environments and to determine probability distributions for containment failure size and location under different loads. (This was done in conjunction with the NUREG-1150 expert elicitation process.) Simplified Boolean expressions were used to evaluate the system failure probabilities and to define cut set specific survival events which were added to each accident sequence cut set for the core vulnerable accident sequences. This process is described in detail in Volume 3 of this report.

2.2.9 Individual and Integrated Uncertainty Analysis

A review of methods[8] for representing and propagating uncertainty in PRAs was conducted and the Latin Hypercube approach was adopted. A new code, TEMAC,[9] was developed to perform the final quantification of the accident sequences using the Latin Hypercube sample generated by the LHS code.[10] The TEMAC code also calculates various risk importance measures and ranks the basic events by their contribution to mean core damage frequency. Individual accident sequence cut sets were evaluated, and uncertainty distributions and importance measures were calculated. For each analysis (seismic, fire, flood, and internal), the cut sets from all of the surviving sequences were combined and evaluated to obtain the total core damage frequency and importance measures for that specific set of initiators. The individual accident sequence results and the integrated results for each subanalysis are presented in the appropriate volume of this report (Volume 3 for internal, Volume 2 and 8 for seismic, Volume 9 for fire, and Volume 10 for flood). Finally, all of the cut sets from all of the analyses were combined into one global expression and an integrated calculation was performed to obtain an overall core damage frequency and uncertainty distribution and global importance measures. These final global results are presented in Volume 2 of this report.

2.3 References

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3.0 ACCIDENT SEQUENCE DELINEATION

3.1 Introduction

One of the major purposes of the RMIEP program was to develop methods for the integrated evaluation of all Level I initiating events. So while only one set of event trees will be presented in this section, these trees were used in four different analyses: internal events, seismic, fire, and flood. The fault trees to be used with these event trees were expanded from the usual level of detail used in the internal events analysis to include information necessary to perform an integrated evaluation of the internal and external events. This development is presented here so that the reader can understand the accident sequence results presented in Section 4 of this report without having to refer to other volumes.

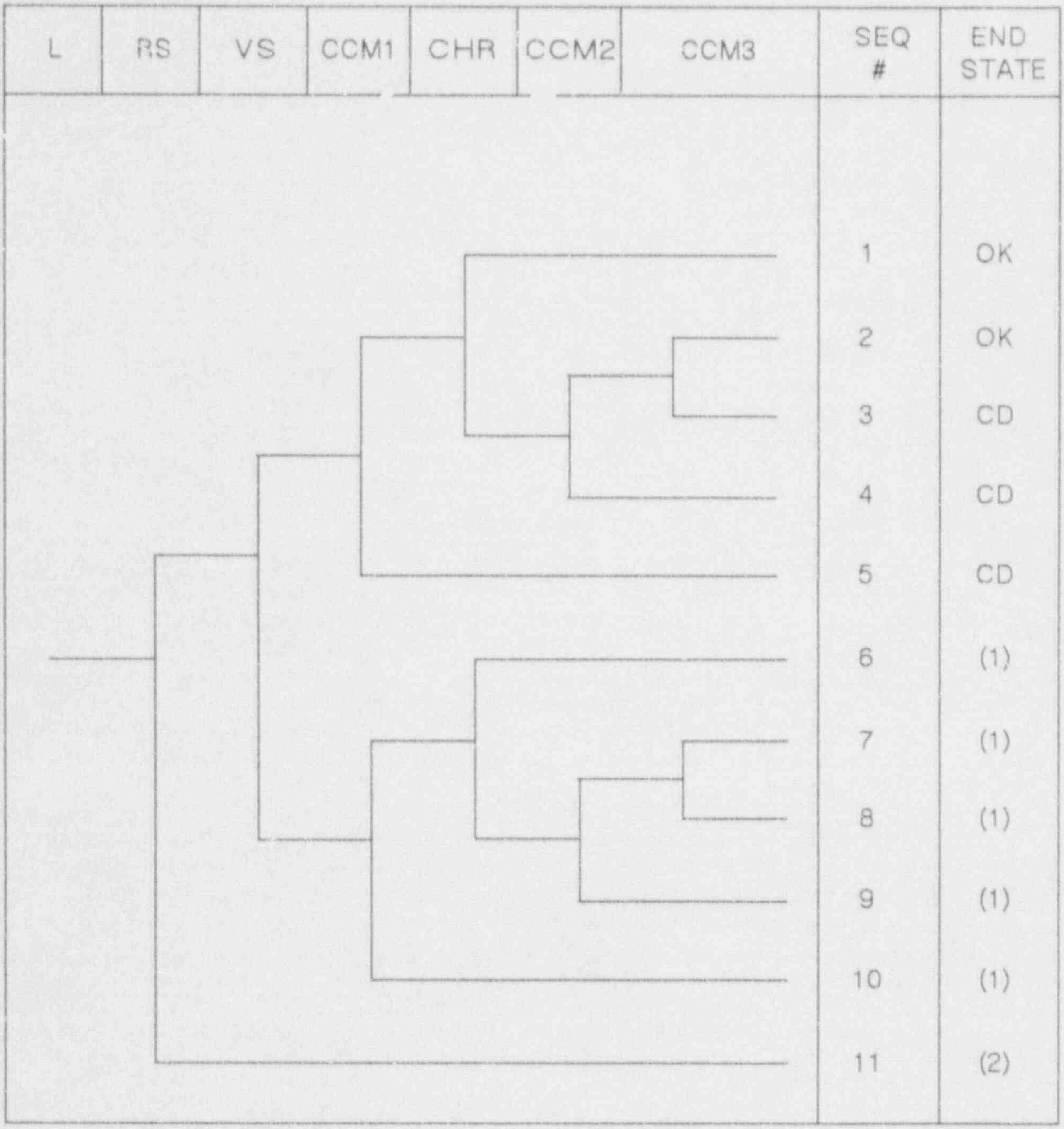
In this section, the functional and systemic event trees used for this analysis will be presented and described. The accident sequences are followed until the end state is resolved into no core damage or core damage. No core damage, or success states, are those in which sufficient systems work in order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

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

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

3.2 Core Damage Functional Event Trees

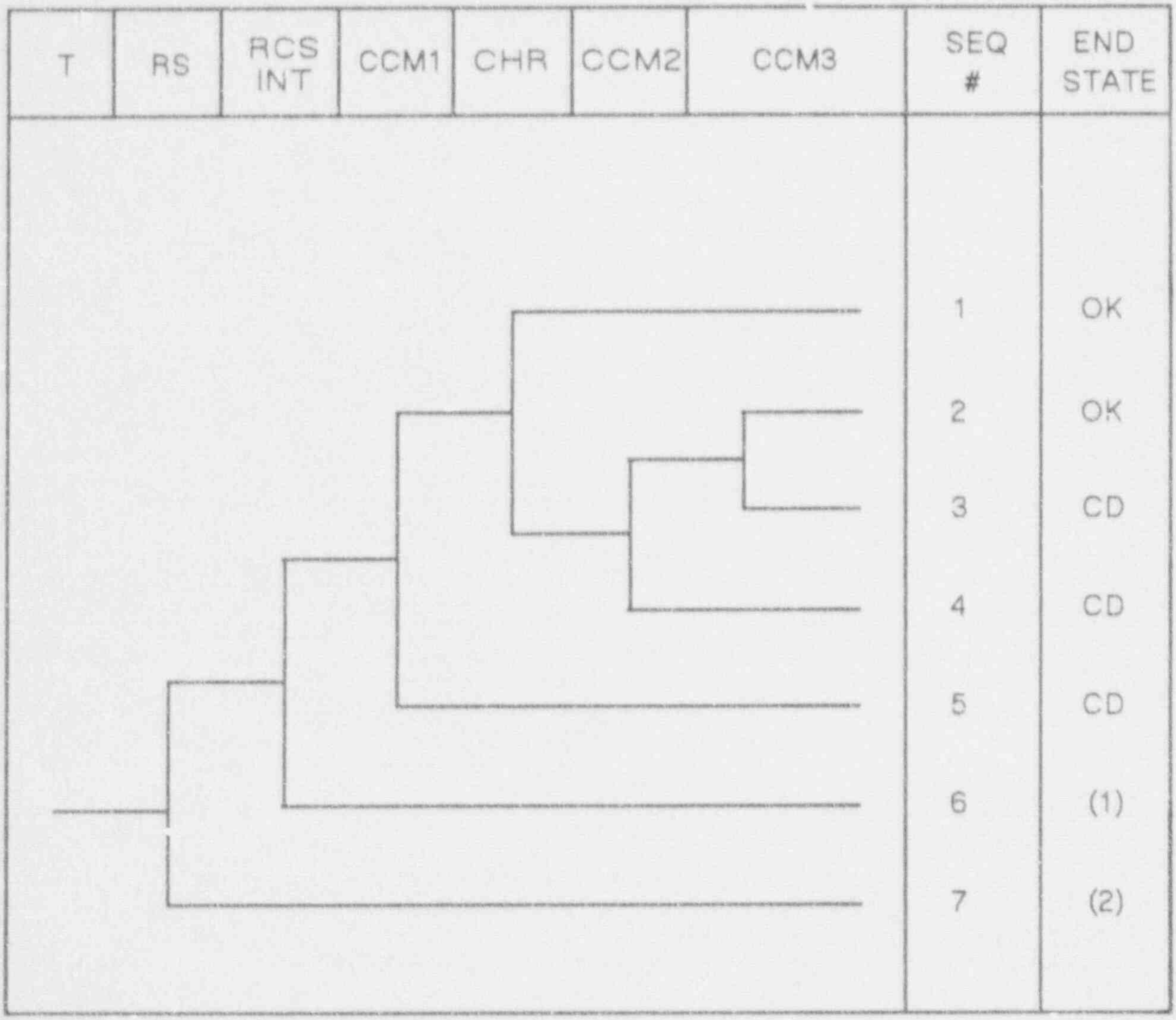
The functional core damage event trees are presented in Figures 3.2-1, 3.2-2, and 3.2-3. The functional event trees delineate the general plant response to loss of coolant (LOCA) accidents, anticipated transients (TRANS), and anticipated transients without scram (ATWS). The delineation is presented in terms of the success or failure of safety functions required to mitigate the transient or loss of coolant initiator. These safety functions (i.e., the top events in Figures 3.2-1, 3.2-2, and 3.2-3) are described in this section.



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

(2) Transfer to ATWS Tree, Figure 2.3.

Figure 3.2-1 LOCA Functional Event Tree



(1) Transfer to LOCA tree

(2) Transfer to ATWS Tree

Figure 3.2-2 Transient Functional Event Tree

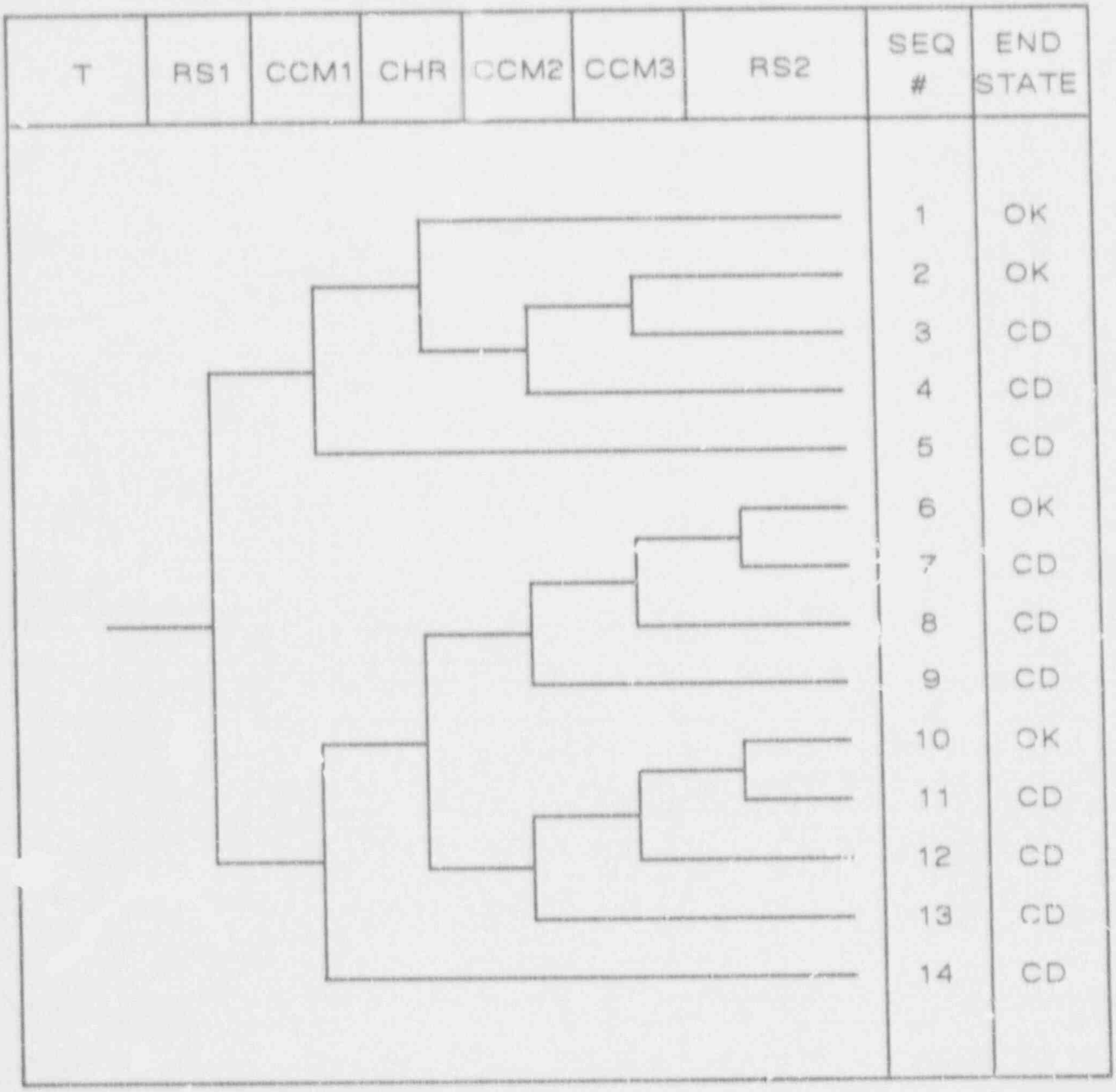


Figure 3.2-3 ATWS Functional Event Tree

3.2.1 Safety Functions

Reactor Subcriticality (RS or RS2)

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

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

Containment and core damage are possible if the operator fails to reduce core power. Sequences involving failure of the reactor subcriticality function are transfered to the ATWS event tree and evaluated there.

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

The event RS represents failure to shutdown the reactor early in the accident. The event RS2 represents the ultimate shutdown of the reactor after it has been stabilized.

Reactor Coolant System Integrity (RCSINT)

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

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

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

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

For ATWS sequences, the LOCA aspects of these sequences do not affect the event tree because the systems used to mitigate an ATWS event can mitigate LOCAs of any size and, for sequences without reactor subcriticality, the ADS valves will be open anyway to transfer the energy to the suppression pool. For the above reasons, this event does not appear explicitly on the ATWS functional event tree.

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

Early Containment Overpressure Protection (Vapor Suppression, VS)

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

If the steam released during a LOCA is not condensed by the vapor suppression pool, pressure will quickly build up in the primary containment and the containment will need to be vented or it will mechanically fail (for large LOCAs, the time to mechanical failure could be as short as 30 sec).

For transients, given failure of the RCS integrity function, heat from the vessel is transported to the suppression pool either directly through a stuck open SRV or via a large LOCA to the drywell and then through the downcomers. Failure of the suppression pool to condense this steam will result in overpressurization and failure of the containment within a very short time (30 sec to 15 min). Containment venting or failure may result in failure of the coolant injection systems and containment heat removal equipment due to the severe environments produced in the reactor building, where most of the systems modeled in this analysis have components, leading to core damage and a radioactive release. Failure of this function is assumed to result in the core being vulnerable to damage. Sequences with failure of the RCS integrity function are transfered to the LOCA tree and evaluated there. This function does not, therefore, explicitly appear on the transient tree.

For ATWS sequences, failure of the vapor suppression function is probabilistically negligible and it is, therefore, not developed on the ATWS functional event tree.

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

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

Core Coolant Makeup (CCM1, CCM2, CCM3)

A LOCA by definition results in loss of coolant from the reactor core that must be replaced in order to prevent core damage. For transients, boil-off of the coolant through the SRVs to the suppression pool will also result in a loss of coolant that must be replaced. The emergency core cooling (ECC) systems are designed to provide cooling water to the core from an external source or from the suppression pool. This cooling water passes through the core, removing heat and transferring it to the vapor suppression pool. If the original source of water was external, the ECC systems would be realigned to take suction from the suppression pool to form a continuous circulation loop for cooling the core upon high level in the suppression pool or low level of the source. Eventually, the stored heat in the suppression pool must be transferred to the ultimate heat sink.

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

Failure of the coolant makeup function will result in loss of core cooling and core damage. Success of this function must be followed by removal of heat stored in the suppression pool.

CCM1 represents successful initial core coolant makeup early in the accident to prevent immediate core damage. CCM2 represents core coolant makeup continuing to be successful after failure of containment heat removal but before containment failure or venting. CCM3 represents continued successful core coolant makeup after containment failure or venting. CCM2 and CCM3 are necessary because of the feedback effects of the containment and reactor building environments on the injection systems performing the coolant makeup function that are described below under the containment heat removal function.

Containment Heat Removal (CHR)

In the later stages of a LOCA or transient initiated accident, the heat buildup in the suppression pool can reach the pool's storage capacity. If this storage capacity is exceeded, the suppression pool will boil and the evolved steam can cause overpressurization and failure of the containment. Containment failure can potentially result in core damage.

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

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

For ATWS sequences, even if the normal heat removal path is available for removal of energy being generated after failure of the reactor subcriticality function, the reactor power level will be in the range of 9-

17%, depending on the systems operating. This is much higher than the capability of the RHR system (about 3%). The energy being generated in the vessel will be deposited in the suppression pool via the SRV discharge lines or directly to the drywell if a LOCA exists. The excess energy, over and above the RHR systems heat removal capacity, will result in rapid containment pressurization.

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

For ATWS sequences, if only low pressure injection systems are working and RHR works, LTAS calculations, described in Volume 4 of this report, show that the containment pressure will equilibrate near the ADS reclosure pressure. The low pressure injection systems stop injecting as the containment pressure rises due to the energy generated when the core is reflooded, the ADS valves reclose, and the RPV repressurizes. As the reactor goes subcritical, with injection stopped, the RHR system can then reduce containment pressure below the reclosure pressure, and the ADS valves reopen. The low pressure injection systems can re-inject water into the core and the process starts over again. If venting occurs, or both RHR and venting are successful, the containment pressure will equilibrate above the vent pressure but below the ADS reclosure pressure. Low pressure injection will go on and off as the RPV pressure goes below and above the low pressure injection pumps shutoff head. These scenarios assume that injection does not fail from the severe environments produced in the reactor building after venting or from the valve cycling in the injection lines as the RPV pressure varies (this is accounted for elsewhere in the model).

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

3.3 Systems Available to Perform Required Functions

The front-line systems available at LaSalle for mitigating LOCAs and transients are presented in Tables 3.3-1 and 3.3-2 respectively. Detailed descriptions of the systems listed are given in the corresponding fault tree analyses sections presented in Volume 6 of this report.

A dependency matrix showing the system interdependences is given in Table 3.3-3 for all of the systems for which fault tree models were developed in this analysis. Detailed descriptions of all of the systems can be found in Volume 6 of this report. The primary systems are listed across the top and the support systems they depend on are listed down the side.

3.4 Systemic Event Trees

The logic and supporting calculations used to develop the systemic accident sequence event trees are described in detail in Volume 4 of this report. For this summary, we simply present the results of that analysis.

3.4.1 LOCAs

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

A: Large LOCA

A large LOCA is any break in the reactor coolant system piping which could lead to the loss of a sufficient amount of coolant to result in a rapid depressurization of the reactor system.

S_1: Medium LOCA

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

S_2: Small LOCA

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

The LTAS code, developed at Oak Ridge National Laboratory (ORNL), was modified to represent the LaSalle plant. The code was base-lined to a

Table 3.3-1
LOCA FUNCTION/SYSTEM RELATIONSHIP

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

Table 3.3-2
TRANSIENT FUNCTION/SYSTEM RELATIONSHIP

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

Table 3.3-3
System Dependency Matrix

Front-Line Systems

Support System	RPS	MFW	HPCS	RCIC	CRD	ADS	CDS	LPCI	LPCS	PCS	SCS	SPC	CSS	VENT	RPT	SBLC
AC	X	X	X	X	X		X	X	X	X	X	X	X	X	X	X
DC	X	X	X	X	X	X		X	X	X	X	X	X		X	
CSCS			X					X	X		X	X	X			
IA		X			X		X			X				X		
IN		X				X				X						
RBCCW					X											
TBCCW		X					X									
SWS																
HVAC ECCS			X	X	X			X	X		X	X	X			
HVAC DG																

Support Systems

Support System	AC	DC	CSCS	IA	IN	RBCCW	TBCCW	SWS	ECCS HVAC	DG HVAC
AC		X	X	X	X	X	X	X	X	X
DC	X		X							
CSCS	X								X	X
IA						X	X	X		
IN										
RBCCW					X					
TBCCW				X						
SWS						X	X			
ECCS HVAC										
DG HVAC	X									

Table 3.3-3 (Concluded)
System Dependency Matrix

Acronym	Description
AC	AC Power System
ADS	Automatic Depressurization System
CDS	Condensate System
CRD	Control Rod Drive System
CSCS	Core Standby Cooling System
CSS	Containment Spray System
DC	DC Power System
HPCS	High Pressure Core Spray System
HVAC DG	Diesel Generator Room Cooling System
HVAC ECCS	ECCS (HPCS, LPCS, LPCI) Room Cooling System
IA	Instrument Air/Service Air Systems
IN	Instrument Nitrogen/Drywell Pneumatic System
LPCI	Low Pressure Coolant Injection System
LPCS	Low Pressure Core Spray System
MFW	Main Feedwater system
PCS	Power Conversion/Main Steam System
RBCCW	Reactor Building Closed Cooling Water System
RCIC	Reactor Core Isolation Cooling System
RPS	Reactor Protection System
RPT	Recirculation Pump Trip System
SBLC	Standby Liquid Control System
SCS	Shutdown Cooling System
SPC	Suppression Pool Cooling System
SWS	Service Water System
TBCCW	Turbine Building Closed Cooling Water System
VENT	Containment Venting System

RELAP5 model used to evaluate transient response. One RELAP5 calculation and twelve small break and three medium break LTAS calculations were performed and are described in Volume 4 of this report.

The systemic event tree for a LOCA initiator is shown in Figure 3.4-1.

3.4.2 Transients With Scram

The eight transient initiating event categories and ten special transient initiating event categories identified in this study are delineated in a single transient event tree. The success criteria for the systems required to mitigate each transient can vary. This variation in the success criteria is accounted for by including the specific effects of the initiator on the responding systems in the system fault trees in a manner that appropriately models the initiators impact on the system response.

The eight transient initiators are:

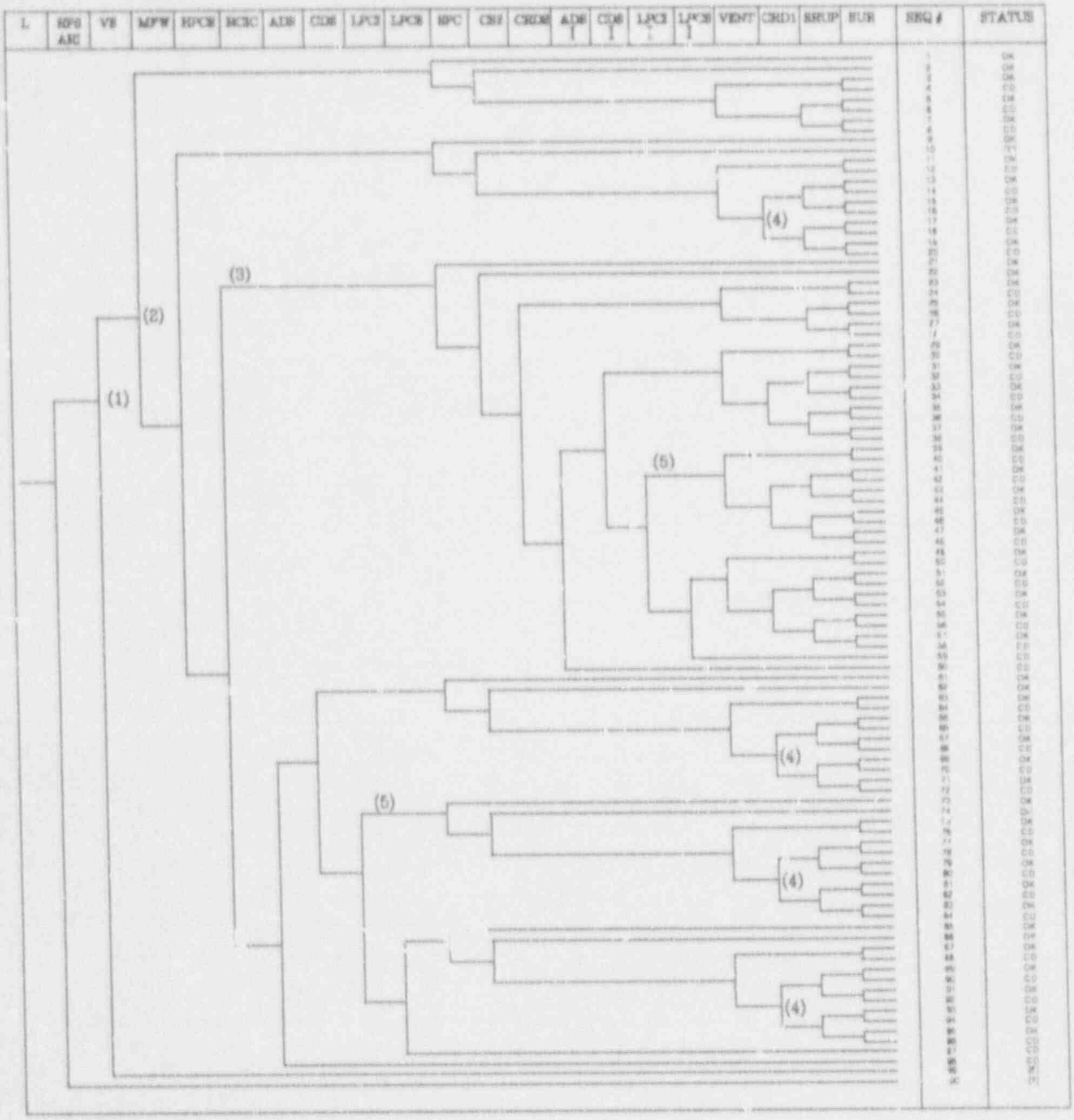
T1: Turbine Trip with Turbine Bypass Available
T2: Turbine Trip With Turbine Bypass Unavailable
T3: Total Main Steam Isolation Valve Closure
T4: Loss of Normal Condenser Vacuum
T5: Total Loss of Feedwater
T6: Partial Loss of Feedwater
T7: Inadvertent Opening of a Safety/Relief Valve
T8: Loss of Offsite Power

The ten special initiators are:

1. DC bus 2A
2. DC bus 2B
3. AC bus 241Y
4. AC bus 242Y
5. Loss of instrument air
6. Loss of drywell pneumatic
7. Loss of 100# drywell pneumatic
8. Total loss of reactor vessel narrow range level instrumentation
9. Loss of train A and C of reactor vessel narrow range level instrumentation
10. Loss of train B and D of reactor vessel narrow range level instrumentation

A detailed discussion of the identification of the initiating events is presented in Volume 4 of this report.

The event tree for a transient initiator is shown in Figure 3.4-2. This event tree was developed by referring to the accident analyses reported in Chapter 15 of the FSAR,[1] LaSalle operating procedures, generic BWR operating procedures,[2] and generic transient thermal-hydraulic calculations.[3]



(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.

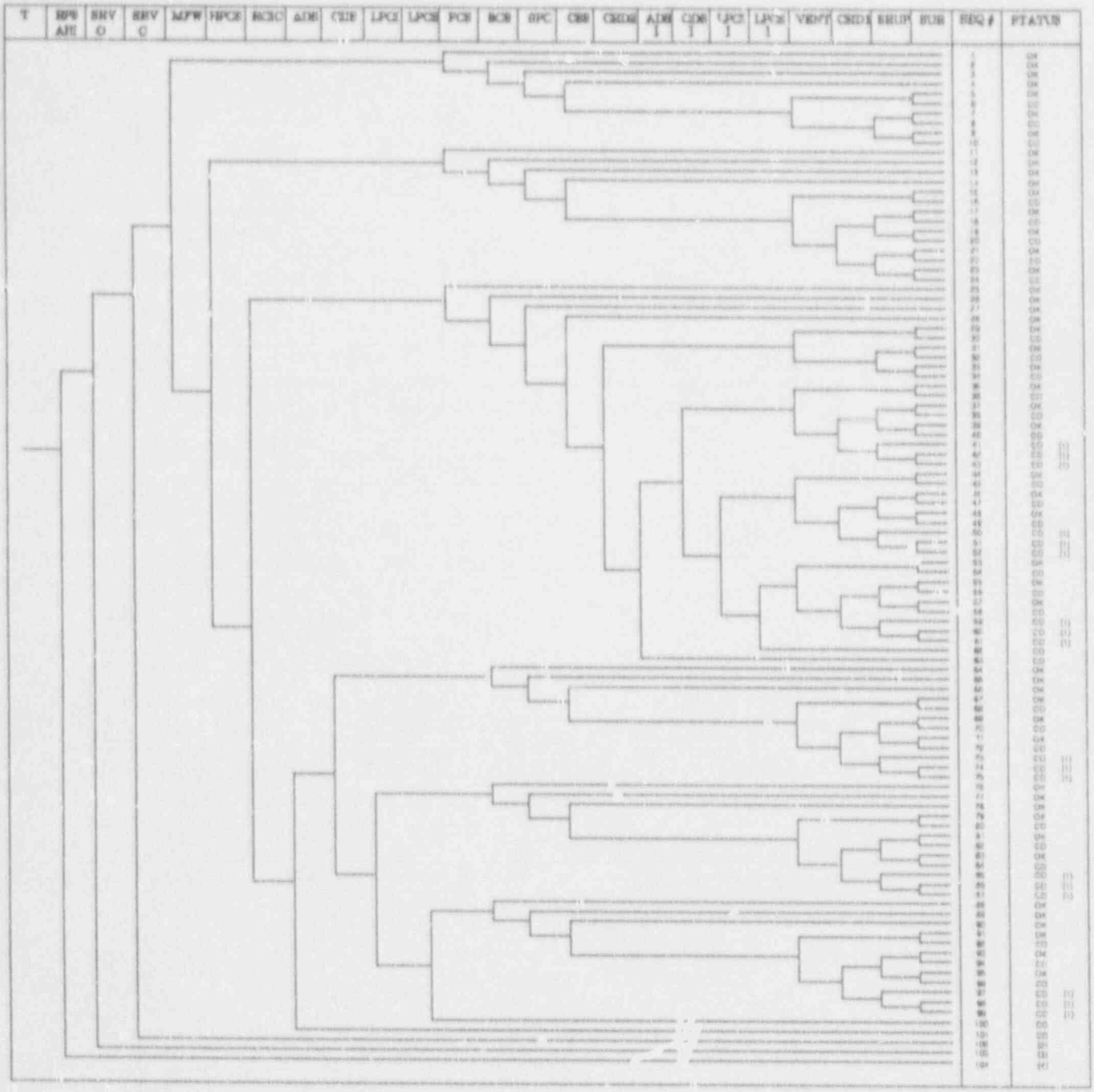
Figure 3.4-1 LaSalle LOCA Systemic Event Tree

Figure 3.4-1 (Continued)
LaSalle LOCA Systemic Event Tree

Event	Descriptor	Description
L	LOCA Initiator	Any loss of coolant initiator
RPS/ARI	Reactor Subcritical: Reactor Protection System (RPS) or Alternate Rod Insertion (ARI)	Use of the RPS or ARI systems to render the reactor subcritical by inserting the control rods
VS	Vapor Suppression	Successful operation of the downcomers and vacuum breakers to mitigate the effects of the vessel blowdown on the containment.
MFW	Feedwater Available	Use of the motor-driven feedwater pump for initial coolant injection.
HPCS	HPCS Available	Use of HPCS system for initial coolant injection.
RCIC	RCIC Available	Use of turbine-driven RCIC pump for initial injection, small LOCA only.
ADS	Reactor Vessel Depressurization	Use of ADS system to depressurize RPV for medium and small LOCAs.
CDS	Condensate Available	Use of CDS system for initial coolant injection.
LPCI	LPCI Available	Use of LPCI system for initial coolant injection.
LPCS	LPCS Available	Use of LPCS system for initial coolant injection.
SPC	Suppression Pool Cooling	Use of SPC mode of RHR for containment heat removal.
CSS	Containment Spray	Use of CSS mode of RHR for containment heat removal.
CRD2	Intermediate Control Rod Drive	Use of two CRD pumps late in accident for small LOCA only.
ADS I	Intermediate Reactor Vessel Depressurization	Use of ADS to depressurizes and use low pressure injection system after RHR and RCIC failure.

Figure 3.4-1 (Concluded)
LaSalle LOCA Systemic Event Tree

Event	Descriptor	Description
CDS I	Intermediate Condensate Available	Use of CDS, after RHR and RCIC failure, to cool the core.
LPCI I	Intermediate LPCI Available	Use of LPCI, after RHR and RCIC failure, to cool the core.
LPCS I	Intermediate LPCS Available	Use of LPCS after RHR and RCIC failure to cool the core.
VENT	Containment Venting	Use of containment venting to reduce containment pressure after failure of RHR.
CRD1	Late Control Rod Drive	Use of one CRD pump in very long-term accidents with loss of containment heat removal and failure of other injection to cool the core, small LOCA only.
SRUP	Containment Failure Mode	Structural failure of containment. Leak (upper branch) or rupture (lower branch).
SUR	Injection System Survival	Survival of any available injection systems in severe reactor building environments after containment failure or venting.



(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 CREA[illegible] LOCA, PROB. OF 18 SRV FTO NEGLIABLE).

(4) TRANSFER TO ATWS TREE.

Figure 3.4-2 LaSalle Transient Systemic Event Tree

Figure 3.4-2 (Continued)
LaSalle Transient Systemic Event Tree

Event	Descriptor	Description
T	Transient Initiators	Any transient or special transient initiator.
RPS/ARI	Reactor Subcritical Reactor Protection System (RPS) or Alternate Rod Insertion (ARI)	Use of the RPS or ARI systems to render the reactor subcritical by inserting the control rods
SRV O	Safety Relief Valves Open	The SRVs open to relieve RPV pressure.
SRV C	Safety Relief Valves Reclose	The SRVs reclose preventing a transient-induced LOCA.
MFW	Feedwater Available	Use of the feedwater system for initial coolant injection.
HPCS	HPCS Available	Use of the HPCS system for initial coolant injection.
RCIC	RCIC Available	Use of the RCIC system for initial coolant injection.
ADS	Reactor Vessel Depressurization	Use of ADS system to depressurize the RPV to use low pressure injection.
CDS	Condensate Available	Use of the CDS system for initial coolant injection.
LPCI	LPCI Available	Use of the LPCI system for initial coolant injection.
LPCS	LPCS Available	Use of the LPCS system for initial coolant injection.
PCS	PCS Available	Use of PCS for containment heat removal.
SCS	Shutdown Cooling	Use of SDC mode of RHR for containment heat removal.

Figure 3.4-2 (Concluded)
LaSalle Transient Systemic Event Tree

Event	Descriptor	Description
SPC	Suppression Pool Cooling	Use of SPC mode of RHR for containment heat removal.
CSS	Containment Spray	Use of CSS mode of RHR for containment heat removal.
CRD2	Intermediate Control Rod Drive	Use of two CRD pumps late in accident for injection.
ADS I	Intermediate Reactor Vessel Depressurization	Use of ADS to depressurize and use low pressure injection system after RHR and RCIC failure.
CDS I	Intermediate Condensate Available	Use of CDS, after RHR and RCIC failure, to cool the core.
LPCI I	Intermediate LPCI Available	Use of LPCI, after RHR and RCIC failure, to cool the core.
LPCS I	Intermediate LPCS Available	Use of LPCS, after RHR and RCIC failure, to cool the core.
VENT	Containment Venting	Use of containment venting to reduce containment pressure after failure of RHR.
CRD1	Late Control Rod Drive	Use of one CRD pump in very long-term accidents with loss of containment heat removal and failure of other injection to cool the core.
SRUP	Containment Failure Mode	Structural failure of containment Leak (upper branch) or rupture (lower branch).
SUR	Injection System Survival	Survival of any available injection systems in severe reactor building environments after containment failure or venting.

In addition, LaSalle specific calculations were performed with RELAP5, LTAS, and MELCOR for various accident sequences. As mentioned in section 3.4.1, the LTAS code was modified to represent the LaSalle plant and five transient calculations were performed. In addition, an integrated model was constructed for use with the MELCOR code. MELCOR calculations beginning at reactor trip and progressing through core damage, melt, vessel breach, containment heatup and failure, and release of radionuclides to the environment were performed. The results of these calculations are described in Volume 3 of the Level II/III report and were mainly used for the Level II/III analysis.

3.4.3 ATWS Event Tree

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

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

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

A study performed at Oak Ridge as part of the SASA program of ATWS sequences for Browns Ferry Unit One[4] indicates that the "instructions provided by the EPGs, if properly interpreted and implemented by the operators, would provide a satisfactory reactor shutdown and accident termination" of the MSIV-closure ATWS analyzed in the study. However, the Oak Ridge study also indicated some potential problem areas. The most important of these is that the operator can be directed to manually reduce reactor pressure during an ATWS. (This is to ensure that the thermal energy released during a LOCA can be condensed in a suppression pool. As the suppression pool temperature increases above 165 °F, the operator is to depressurize the vessel according to a supplied graph.) The calculations performed indicate that manual depressurization during an ATWS is very tricky and, depending on the situation, can result in reactor power and vessel pressure fluctuations. The recommendations from this study were to eliminate such a manual depressurization during an ATWS.

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

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

For this study, the RELAP5 model used for the transient analysis was modified to perform two ATWS calculations. In order to perform more efficient calculations and to evaluate more sequences, the LTAS code was modified, as described in section 3.4.1, to represent the LaSalle plant and basis-lined to the RELAP5 model. A REMONA 3-D calculation was used for the power vs level correlation.[5] Nineteen different ATWS calculations were performed using the LTAS code to investigate different possible system success criteria and to evaluate the accident sequence timing.

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

The general ATWS event tree is shown in Figure 3.4-3.

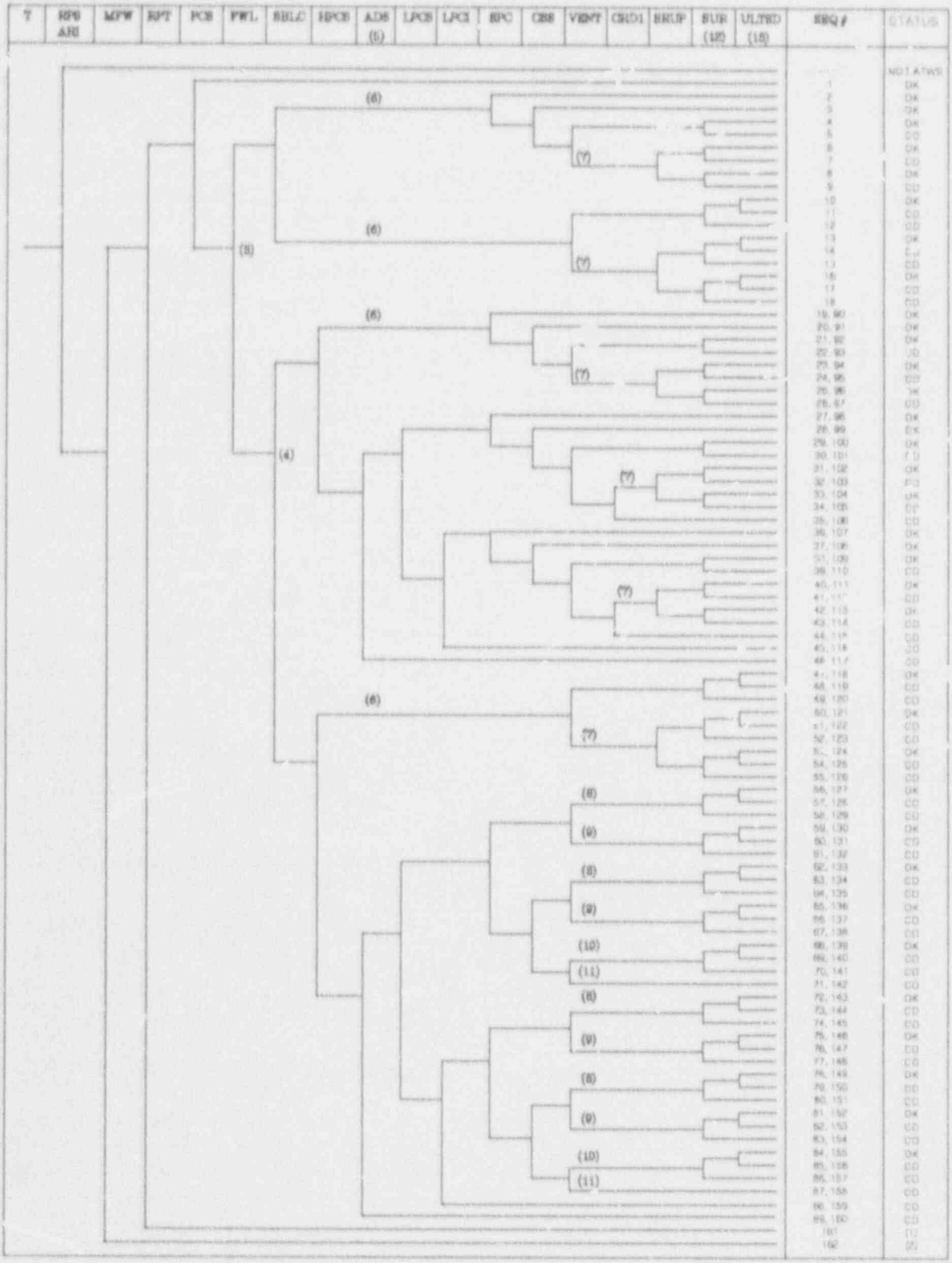


Figure 3.4-3 LaSalle ATWS Systemic Event Tree

Figure 3.4-3 (Continued)
LaSalle ATWS Systemic Event Tree

Event	Descriptor	Description
T	Transient Initiators	Any transient, special transient, or LOCA initiator.
RPS/ARI	Reactor Subcriticality Reactor Protection System (RPS) or Alternate Rod Insertion (ARI)	Use of the RPS or ARI systems to render the reactor subcritical by inserting the control rods
MFW	Feedwater Available	Use of main feedwater for initial coolant injection.
RPT	Recirculation Pump Trip	Use of RPT to reduce reactor power after failure to scram.
PCS	Power Conversion System	Use of PCS in conjunction with MFW to remove power being produced by failure to scram.
FWL	Feedwater Level Control	Operator controls feedwater injection to CST makeup rate after failure of PCS.
SBLC	Standby Liquid Control System	Use of SBLC to inject Boron into the vessel to render the reactor subcritical after RPS/ARI failure.
HPCS	HPCS Available	Use of HPCS for initial coolant injection.
ADS	Automatic Depressurization System	Use of ADS to depressurize the vessel and use low pressure injection systems after HPCS and MFW failure.
LPCS	LPCS Available	Use of LPCS for initial coolant injection.
LPCI	LPCI Available	Use of LPCI for initial coolant injection.
SPC	Suppression Pool Cooling	Use of SPC mode of RHR to remove decay heat if reactor is shutdown or partially remove energy if reactor is not shutdown.

Figure 3.4-3 (Continued)
LaSalle ATWS Systemic Event Tree

Event	Descriptor	Description
CSS	Containment Spray	Use of CSS mode of RHR to remove decay heat if reactor is shutdown or partially remove energy if reactor is not shutdown.
VENT	Containment Venting	Use of containment venting, with or without SPC or CSS to maintain containment pressure below structural failure limit.
CRD1	Late Control Rod Drive	Use of one CRD pump late in the accident to maintain coolant injection.
SRUP	Containment Failure Mode	Structural failure of containment Leak (upper branch) or rupture (lower branch).
SUR	Injection System Survival	Survival of any available injection systems in severe reactor building environments after containment failure or venting.
ULTSD	Ultimate Shutdown	Use of any injection path to put Boron into the core or repair of control rod mechanisms to render reactor subcritical.

Figure 3.4-3
LaSalle ATWS Systemic Event Tree (Continued)

Notes

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

2) If MFW fails, RPT is not relevant since RPV level 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. The injection rate must be controlled to ≤ 1800 gpm, the makeup rate from the CST. This means that RPV level will be below TAF.

4) Transfer sequences from (2).

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

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

7) Containment pressure increases until containment failure occurs.

8) RHR and Venting success - Containment pressure (90 psia, 321 F) remains below ADS reclosure pressure. 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.).

Figure 3.4-3
LaSalle ATWS Systemic Event Tree (Concluded)

Notes

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10) RHR fails and Venting OK - Containment pressure (90 psia, 321 F) remains below ADS reclosure pressure. 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.

3.5 References

1. "LaSalle County Station Final Safety Analysis Report," through Admendment 63, Commonwealth Edison Company, Chicago, Il, July 1983.

2. "BWR Emergency Procedure Guidelines," Revision 3, from Commonwealth Edison Company, Chicago, Il, December 1982.

3. "Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors," NEDO-24708A, Volumes 1 & 2, Class 1, Revision 1, Nuclear Fuel and Services Division, General Electric Company, San Jose, Ca., 95125, December 1980.

4. R. M. Hurrington and S. A. Hodge, "ATWS at Browns Ferry Unit One - Accident Sequence Analysis," NUREG/CR-3470, ORNL/TM-8902, Oak Ridge National Laboratory, Oak Ridge, Tennessee, July 1984.

5. R. M. Hurrington and L. C. Fuller, "BWR-LTAS: A Boiling Water Reactor Long-Term Accident Analysis Simulation Code," NUREG/CR-3764, ORNL/TM-9163, Oak Ridge National Laboratory, Oak Ridge, Tennessee, February 1985.

4.0 Discussion of Core Damage Results

This section summarizes the results of the individual internal, fire, flood, and seismic analyses and the final integrated Level I analysis. Section 4.1 discusses the general results of the integrated analysis; section 4.2, the results of the internal analysis; section 4.3, the results of the fire analysis; section 4.4, the results of the flood analysis; and section 4.5, the results of the seismic analysis. Each section discusses: the dominant accident sequences, the dominant cut sets, the events most significant to risk reduction, the events most significant to risk increase, and the events most important to uncertainty.

4.1 Results of the Integrated Analysis

4.1.1 Introduction

Table 4.1-1 shows the TEMAC code results for the final quantification of all the sequences that remained in the analysis after the initial screening quantification. Some of the results are very low due to the application of recovery actions, the impact of the severe environment analysis, and the data revision performed for the final quantification. Since the general truncation criteria for this analysis was 1.0E-08/yr. in the screening phase, accident sequences remaining in Table 4.1-1 with final frequencies below 1.0E-08/yr. can not be said to be ranked correctly in terms of their absolute contribution to the total frequency of core damage. Other sequences, which did not survive the initial screening, are not in their appropriate places on the table. These sequences, which were dropped from the analysis, may or may not have significant recovery, severe environment, or data effects to reduce their frequencies roughly proportionally to that of the sequences retained in the analysis.

For the LaSalle internal events analysis, the initiating events were included in the fault trees. The result was that there were not as many sequences to solve as in other PRAs. There were a total of 50 transient sequences, 45 LOCA sequences, and 95 ATWS sequences that lead to core damage. Of these 190 sequences, 54 remained to be evaluated in the final quantification and these all appear in Table 4.1-1. The other 136 sequences that did not survive the screening process are in most aspects very similar to the sequences that did survive. The effect of the application of recovery, the severe environment effects, and the data revision upon the frequency of the sequences that did survive was reviewed. Then similarities of the components appearing in the cut sets between those sequences which survived and those which did not were examined. We conclude that the sequences which did not survive screening would have their frequencies reduced roughly the same as similar sequences which did survive the screening. Since sequences with mean frequencies greater than or equal to 1.0E-08/yr. comprise 99.9% of the total core damage frequency of those sequences analyzed, we conclude that the sequences which did not survive the screening process would have a negligible impact on the final result.

Table 4.1-1
LASALLE FINAL SEQUENCE CORE DAMAGE STATISTICS

SEQUENCE	5%	MEDIAN	MEAN	95%	PE	FRAC OF TOT MEAN CDF	CUM FRAC
T100	1.1000E-06	9.0700E-06	2.8700E-05	9.7400E-05	2.1400E-05	3.5413E-01	3.5413E-01
FIRE-CR	6.1800E-14	1.2500E-12	1.3900E-05	3.6700E-06	2.1800E-05	1.7151E-01	5.2565E-01
FIRE-W2	0.0000E+00	0.0000E+00	6.7100E-06	1.3500E-05	3.7700E-06	8.2796E-02	6.0844E-01
T62	3.1800E-07	2.3900E-06	6.5300E-06	2.4100E-05	4.7700E-06	8.0575E-02	6.8902E-01
T18	0.0000E+00	0.0000E+00	4.9900E-06	2.0900E-05	2.7100E-06	6.1572E-02	7.5059E-01
FIRE-Y2	0.0000E+00	0.0000E+00	3.3900E-06	7.9000E-06	2.7400E-06	4.1830E-02	7.9242E-01
FS2	9.5100E-08	1.0900E-06	3.1800E-06	1.0500E-05	3.0400E-06	3.9238E-02	8.3166E-01
FIRE-T	4.4800E-08	6.7700E-07	2.2700E-06	7.7800E-06	2.2500E-06	2.8010E-02	8.5967E-01
FIRE-W1	0.0000E+00	0.0000E+00	1.8000E-06	1.8700E-06	6.9800E-07	2.2210E-02	8.8188E-01
FIRE-Y1	0.0000E+00	0.0000E+00	1.7600E-05	1.9100E-06	7.0100E-07	2.1717E-02	9.0360E-01
T20	0.0000E+00	0.0000E+00	1.2800E-06	8.0300E-06	4.8100E-07	1.5794E-02	9.1939E-01
T22	0.0000E+00	0.0000E+00	1.1400E-06	3.9900E-06	5.5700E-07	1.4067E-02	9.3346E-01
FIRE-P	0.0000E+00	0.0000E+00	5.7300E-07	2.2800E-06	4.3400E-07	7.0703E-03	9.4053E-01
FIRE-AC	0.0000E+00	0.0000E+00	5.4200E-07	2.7900E-06	4.5100E-07	6.6878E-03	9.4721E-01
FIRE-E-S3	6.9400E-09	1.0900E-07	5.0600E-07	2.0900E-06	5.1900E-07	6.2436E-03	9.5346E-01
T16	1.5500E-08	1.4200E-07	4.3600E-07	1.7500E-06	4.8100E-07	5.3799E-03	9.5884E-01
FIRE-S-W	0.0000E+00	0.0000E+00	3.5200E-07	1.5700E-06	1.9300E-07	4.3434E-03	9.6318E-01
TLOSP-01.LL1	8.6700E-12	9.2000E-09	3.1400E-07	5.4200E-07	1.3200E-07	3.8745E-03	9.6706E-01
T101	4.1300E-09	6.4500E-08	2.4800E-07	1.0100E-06	2.7600E-07	3.0601E-03	9.7012E-01
T24	0.0000E+00	0.0000E+00	2.2600E-07	1.2100E-06	1.0400E-07	2.7886E-03	9.7290E-01
FS1	7.9200E-11	3.3900E-09	2.1300E-07	5.5400E-07	1.8800E-07	2.6282E-03	9.7553E-01
TL12	2.1500E-09	3.8100E-08	2.1000E-07	7.6400E-07	2.1100E-07	2.5912E-03	9.7812E-01
TL97	2.7600E-09	4.0100E-08	1.9400E-07	5.6200E-07	1.3200E-07	2.3938E-03	9.8052E-01
FIRE-N	0.0000E+00	0.0000E+00	1.6300E-07	9.2500E-08	8.6100E-09	2.0113E-03	9.8253E-01
T38	0.0000E+00	0.0000E+00	1.3500E-07	8.4100E-08	2.3300E-08	1.6658E-03	9.8420E-01
TLOSP_01.L1	5.4800E-12	3.3800E-09	1.3400E-07	2.2000E-07	5.5600E-08	1.6534E-03	9.8585E-01
FIRE-E-S2	1.5500E-09	3.0800E-08	1.1400E-07	5.6900E-07	1.2200E-07	1.4067E-03	9.8726E-01
TLOSP_01.L2	6.6600E-12	2.3000E-09	9.2300E-08	1.3600E-07	3.6200E-08	1.1389E-03	9.8839E-01
A49	3.9800E-10	1.1100E-08	8.9400E-08	2.4800E-07	5.5200E-08	1.1031E-03	9.8950E-01
A120	5.7600E-10	1.0900E-08	7.9800E-08	1.7600E-07	4.7400E-08	9.8466E-04	9.9048E-01

Table 4.1-1 (Continued)
LASALLE FINAL SEQUENCE CORE DAMAGE STATISTICS

SEQUENCE	5%	MEDIAN	MEAN	95%	PE	FRAC OF TOT MEAN CDF	CUM FRAC
T50	1.2200E-10	4.2400E-09	7.4200E-08	1.7900E-07	2.8100E-08	9.1556E-04	9.9140E-01
TLOSP-03.LL1	1.6900E-13	1.8000E-10	7.0100E-08	6.3900E-08	1.2200E-08	8.6497E-04	9.9226E-01
TL59	7.8400E-10	9.2400E-09	5.9000E-08	1.7200E-07	3.1900E-08	7.2801E-04	9.9299E-01
T70	0.0000E+00	0.0000E+00	5.7400E-08	2.5200E-08	1.1300E-08	7.0827E-04	9.9370E-01
T85	9.6300E-12	1.2600E-09	5.4800E-08	9.1000E-08	1.4200E-08	6.7619E-04	9.9437E-01
T82	0.0000E+00	0.0000E+00	4.9500E-08	1.8300E-08	5.8000E-09	6.1079E-04	9.9499E-01
T47	0.0000E+00	0.0000E+00	4.9500E-08	1.7000E-08	6.0500E-09	6.1079E-04	9.9560E-01
TLOSP_01.L3	4.8300E-12	1.4000E-09	4.2400E-08	7.7800E-08	2.3200E-08	5.2318E-04	9.9612E-01
FIRE-Z	0.0000E+00	0.0000E+00	3.5800E-08	2.0300E-07	3.1900E-08	4.4174E-04	9.9656E-01
A123	0.0000E+00	0.0000E+00	3.2000E-08	1.1500E-07	1.5000E-08	3.9485E-04	9.9696E-01
TLOSP_01.L4	[illegible].5500E-12	6.6700E-10	2.4300E-08	3.4900E-08	1.4500E-08	2.9984E-04	9.9726E-01
TLOSP_03.L1	7.7400E-14	7.2200E-11	2.1500E-08	3.2600E-08	5.1500E-09	2.6529E-04	9.9752E-01
L14	0.0000E+00	0.0000E+00	1.7200E-08	7.3500E-08	1.0600E-08	2.1223E-04	9.9773E-01
TLOSP_01.L5	8.6000E-13	2.7900E-10	1.6900E-08	1.7800E-08	8.7200E-09	2.0853E-04	9.9794E-01
TLOSP_01.L6	2.7000E-13	1.0700E-10	1.6200E-08	1.2300E-08	6.4500E-09	1.99[illegible]E-04	9.9814E-01
T56	0.0000E+00	0.0000E+00	1.5900E-08	2.2000E-09	8.3600E-10	1.9619E-04	9.9834E-01
T59	1.4500E-11	1.2600E-09	1.5800E-08	5.7900E-08	1.3500E-08	1.9496E-04	9.9853E-01
TLOSP_03.L2	7.5100E-14	5.0500E-11	1.4400E-08	2.1400E-08	3.3500E-09	1.7766E-04	9.9871E-01
T41	1.3500E-10	3.3400E-09	1.1500E-08	4.7400E-08	1.5700E-08	1.4190E-04	9.9885E-01
TLOSP_03.L3	6.3600E-14	2.8800E-11	1.1400E-08	1.3700E-08	2.1500E-09	1.4067E-04	9.9899E-01
L16	0.0000E+00	0.0000E+00	8.9800E-09	2.2400E-08	1.8800E-09	1.1081E-04	9.9910E-01
A126	0.0000E+00	0.0000E+00	8.8800E-09	5.9900E-08	3.9800E-09	1.0957E-04	9.9921E-01
T73	1.1100E-10	2.7300E-09	8.6000E-09	3.2300E-08	1.2100E-08	1.0612E-04	9.9932E-01
TLOSP_03.L4	2.7900E-14	1.4400E-11	8.3000E-09	6.1300E-09	1.3400E-09	1.0241E-04	9.9942E-01
FIRE-AA	0.0000E+00	0.0000E+00	7.3100E-09	3.5900E-06	6.7700E-09	9.0199E-05	9.9951E-01
FIRE S-AA	0.0000E+00	0.0000E+00	5.9400E-09	2.7200E-08	3.3400E-09	7.3295E-05	9.9959E-01
TL18	0.0000E+00	0.0000E+00	5.0700E-09	2.9200E-09	1.0700E-09	6.2559E-05	9.9965E-01
TL14	0.0000E+00	0.0000E+00	2.9900E-09	1.8200E-08	3.1100E-09	3.6894E-05	9.9969E-01
TLOSP_03.L5	1.0200E-14	6.0700E-12	2.8700E-09	2.3600E-09	8.0800E-10	3.5413E-05	9.9972E-01
A22	9.5100E-13	8.3100E-11	2.6100E-09	8.1400E-09	1.0300E-09	3.2205E-05	9.9975E-01

Table 4.1-1 (Continued)
LASALLE FINAL SEQUENCE CORE DAMAGE STATISTICS

SEQUENCE	5%	MEDIAN	MEAN	95%	PE	FRAC OF TOT MEAN CDF	CUM FRAC
TL1_01.LL1	4.4100E-14	4.8900E-11	2.4500E-09	5.7400E-09	1.0800E-09	3.0231E-05	9.9978E-01
A52	0.0000E+00	0.0000E+00	2.2200E-09	9.5100E-09	2.1500E-09	2.7393E-05	9.9981E-01
TLOSP_03.L6	2.9700E-15	2.2700E-12	1.8600E-09	1.4000E-09	5.9800E-10	2.2951E-05	9.9983E-01
A93	7.3300E-13	7.9500E-11	1.7300E-09	4.5900E-09	8.4200E-10	2.1347E-05	9.9985E-01
A18	0.0000E+00	0.0000E+00	1.4100E-09	6.6700E-09	6.7300E-10	1.7398E-05	9.9987E-01
TL1_03.LL1	7.3700E-16	1.0200E-12	1.3500E-09	5.1700E-10	1.0000E-10	1.6658E-05	9.9989E-01
T49	0.0000E+00	0.0000E+00	1.0600E-09	7.3100E-10	1.4900E-10	1.3079E-05	9.9990E-01
T32	0.0000E+00	0.0000E+00	1.0400E-09	2.6200E-09	8.55[illegible]0E-10	1.2833E-05	9.9991E-01
T84	0.0000E+00	0.0000E+00	1.[illegible]E-09	[illegible].1100E-10	1.4[illegible]0E-10	1.2463E-05	9.9993E-01
TL16	0.0000E+00	0.0000E+00	9.[illegible]400E-10	4.7300E-09	3.6[illegible]0E-10	1.1155E-05	9.9994E-01
TL1_01.L1	2.3700E-14	2.0600E-11	[illegible].6900E-10	1.9300E-09	4.5600E-10	1.0723E-05	9.9995E-01
T34	0.0000E+00	0.0000E+00	6.7700E-10	1.8200E-10	1.2[illegible]00E-10	8.3536E-06	9.9996E-01
TL1_01.L2	3.3700E-14	1.3600E-11	5.6500E-10	1.1400E-09	2.9700E-10	6.9716E-06	9.9996E-01
TL1_03.L1	3.9800E-16	3.5400E-13	4.0000E-10	2.4700E-10	4.22[illegible]	4.9357E-06	9.9997E-01
TL1_01.L3	2.5300E-14	7.3000E-12	3.7900E-10	6.5400E-10	1.90[illegible]	4.6765E-06	9.9997E-01
TL1_01.L4	1.2800E-14	3.2600E-12	2.5800E-10	3.6100E-10	1.190[illegible]	3.1835E-06	9.9998E-01
TL1_03.L2	2.6400E-16	2.6200E-13	2.5500E-10	1.4100E-10	2.7500E[illegible]	3.1465E-06	9.9998E-01
TL1_03.L3	2.5000E-16	1.5300E-13	2.2900E-10	6.3100E-11	1.7600E-11	2.8257E-06	9.9998E-01
TL1_03.L4	1.1600E-16	8.3100E-14	1.7200E-10	3.8400E-11	1.1000E-11	2.1223E-06	9.9999E-01
L12	6.8200E-13	2.1100E-11	1.6000E-10	6.7400E-10	1.8200E-10	1.9743E-06	9.9999E-01
A129	4.6000E-13	1.8500E-11	1.5500E-10	6.5900E-10	1.6500E-10	1.9126E-06	9.9999E-01
T58	0.0000E+00	0.0000E+00	1.5000E-10	1.0300E-10	2.0500E-11	1.8509E-06	9.9999E-01
TL1_01.L5	5.7500E-15	1.3500E-12	1.0600E-10	1.7600E-10	7.1500E-11	1.3079E-06	9.9999E-01
TL20	0.0000E+00	0.0000E+00	9.4600E-11	2.6900E-10	4.[illegible]400E-11	1.1673E-06	9.9999E-01
TL1_01.L6	1.7000[illegible]	5.6700E-13	8.2400E-11	1.1700E-10	5.2900E-11	1.0167E-06	9.9999E-01
A55	0.0000E+00	[illegible]000E+00	8.0500E-11	2.8100E-10	5.6600E-11	9.9330E-07	1.0000E+00
A58	1.9700E-13	7.8100E-12	6.2100E-11	2.3200E-10	6.8800E-11	7.6626E-07	1.0000E+00
TL1_03.L5	5.7800E-17	3.1300E-14	4.9900E-11	1.5800E-11	6.6300E-12	6.1572E-07	1.0000E+00
T40	0.0000E+00	0.0000E+00	3.7600E-11	3.2800E-11	1.1900E-11	4.6395E-07	1.0000E+00
TL1_03.L6	1.9100E-17	1.2000E-14	3.4800E-11	8.5200E-12	4.9000E-12	4.2940E-07	1.0000E+00
TL2_01.LL1	2.8900E-16	2.5600E-13	3.0500E-11	4.2400E-11	1.2600E-11	3.7634E-07	1.0000E+00

Table 4.1-1 (Concluded)
LASALLE FINAL SEQUENCE CORE DAMAGE STATISTICS

SEQUENCE	5%	MEDIAN	MEAN	95%	PE	FRAC OF TOT MEAN CDF	CUM FRAC
T30	2.9500E-15	3.2000E-13	2.9900E-11	3.7900E-11	6.6200E-12	3.6894E-07	1.0000E+00
A15	0.0000E+00	0.0000E+00	2.7400E-11	1.1700E-10	2.0200E-11	3.3809E-07	1.0000E+00
T72	0.0000E+00	0.0000E+00	1.6300E-11	1.5000E-11	5.7600E-12	2.0113E-07	1.0000E+00
TL36	0.0000E+00	5.9600E-13	1.2500E-11	4.6100E-11	1.8600E-11	1.5424E-07	1.0000E+00
TL3_01.LL1	1.3000E-16	1.2800E-13	1.1000E-11	2.1700E-11	6.2800E-12	1.3573E-07	1.0000E+00
TL2_01.L1	1.3400E-16	1.1000E-13	1.0200E-11	1.7000E-11	5.3100E-12	1.2586E-07	1.0000E+00
TL2_01.L2	1.7300E-16	6.3100E-14	6.4200E-12	1.0900E-11	3.4500E-12	7.9217E-08	1.0000E+00
TL2_01.L3	1.6300E-16	3.5000E-14	4.9800E-12	6.9300E-12	2.2100E-12	6.1449E-08	1.0000E+00
TL3_01.L1	7.0200E-17	5.5100E-14	3.9300E-12	8.8100E-12	2.6500E-12	4.8493E-08	1.0000E+00
TL2_01.L4	8.9200E-17	1.8100E-14	3.5100E-12	4.5200E-12	1.3800E-12	4.3310E-08	1.0000E+00
TL38	0.0000E+00	0.0000E+00	2.7100E-12	1.2600E-11	6.2000E-12	3.3439E-08	1.0000E+00
TL3_01.L2	8.8800E-17	3.3600E-14	2.4200E-12	5.7800E-12	1.7300E-12	2.9861E-08	1.0000E+00
TL3_01.L3	7.5000E-17	1.8000E-14	1.8000E-12	3.2600E-12	1.1100E-12	2.2210E-08	1.0000E+00
L18	0.0000E+00	0.0000E+00	1.7000E-12	1.0300E-12	4.4400E-13	2.0977E-08	1.0000E+00
TL3_01.L4	4.0000E-17	9.6600E-15	1.2200E-12	2.2900E-12	6.9100E-13	1.5054E-08	1.0000E+00
TL2_01.L5	2.5800E-17	7.3700E-15	1.2000E-12	1.8400E-12	8.3300E-13	1.4807E-08	1.0000E+00
TL2_01.L6	8.6600E-18	3.1800E-15	8.7800E-13	1.1100E-12	6.1600E-13	1.0834E-08	1.0000E+00
L20	0.0000E+00	0.0000E+00	7.1800E-13	5.0900E-13	8.6000E-14	8.8595E-09	1.0000E+00
A148	5.6100E-16	4.2400E-14	6.5500E-13	3.0600E-12	1.4800E-12	8.0821E-09	1.0000E+00
TL3_01.L5	1.4200E-17	3.6500E-15	4.4700E-13	1.0100E-12	4.1600E-13	5.5156E-09	1.0000E+00
TL3_01.L6	4.8600E-18	1.5200E-15	3.1900E-13	5.6200E-13	3.0800E-13	3.9362E-09	1.0000E+00
A132	7.0400E-20	2.2500E-17	8.0800E-15	1.0100E-14	2.9400E-15	9.9700E-11	1.0000E+00
TOTAL	1.5932E-[illegible]	1.3717E-05	8.1043E-05	2.2330E-04	6.8769E-05	(Algebraic sum of above)	
INTEGRATED	5.3400E-06	2.9200E-05	1.0100E-04	2.9300E-04	6.7700E-05	(Combined reevaluation)	

The sequences are labeled as follows:

1) transient sequences are Tx from Figure 3.4-2,

2) transient-Induced LOCAs and LOCAs are TLx and Lx, respectively, from Figure 3.4-1,

3) ATWS sequences are Ax from Figure 3.4-3,

4) fire and flood sequences are labeled by the scenario they originate from (FIRE-x or FSx), and

5) seismic sequences are labeled by seven different hazard levels used in the analysis which is shown by the .LXX extension on the sequence name. TLOSP-01 corresponds to internal event sequence T63. TLOSP-03 corresponds to internal event sequence T100. The TLX-0Y sequences are sequences which have the additional failure of 1, 2, or 3 of the SRVs to reclose and, therefore, are transient-induced LOCAs similar to those in the internal events analysis except that they occur with a simultaneous LOSP. The TL1-01 sequence corresponds to internal sequence TL60 which is similar to T63. The TL1-03, TL2-01, and TL3-01 sequences correspond to internal sequence TL97 which is similar to T100.

The total core damage frequency from all events has a mean value of 1.01E-04/yr. with a 5th %-ile of 5.34E-06/yr., a median value of 2.92E-05/yr., and a 95th %-ile of 2.93E-04/yr. This result is considered to be low given that all initiators (both internal and external) are included in this number and that this is the first time that a detailed PRA has been performed on this plant. Usually, the first time a PRA is performed certain design faults are found that lead to accidents that have significantly higher frequencies of occurrence than they would have without the design faults. At LaSalle, because of the generally good design and high redundancy of BWR type nuclear power plants, while some design deficiencies were found, none compromised redundancy to the point where they created accident sequences which were significantly higher in frequency than those from other sources.

The overall integrated core damage cumulative distribution function (CDF) is shown in Figure 4.1-1. A density plot showing the fraction of the Latin Hypercube observations with final core damage frequencies within each frequency interval is overlaid on the integrated core damage CDF plot. Figure 4.1-2 shows CDFs for the total fire, flood, seismic, and internal core damage frequencies and the integrated core damage frequency for comparison purposes. Figure 4.1-3 has pie charts showing the relative contributions of accident sequences from various categories of initiators to the mean integrated core damage frequency. These categories are: seismic, fire, flood, and internal with internal broken into LOCAs, ATWS, transients, and transient-induced LOCAs. Figure 4.1-4 has a pie chart showing a finer breakdown of the contribution of internal events initiators to the total mean internal core damage frequency. The internal initiators

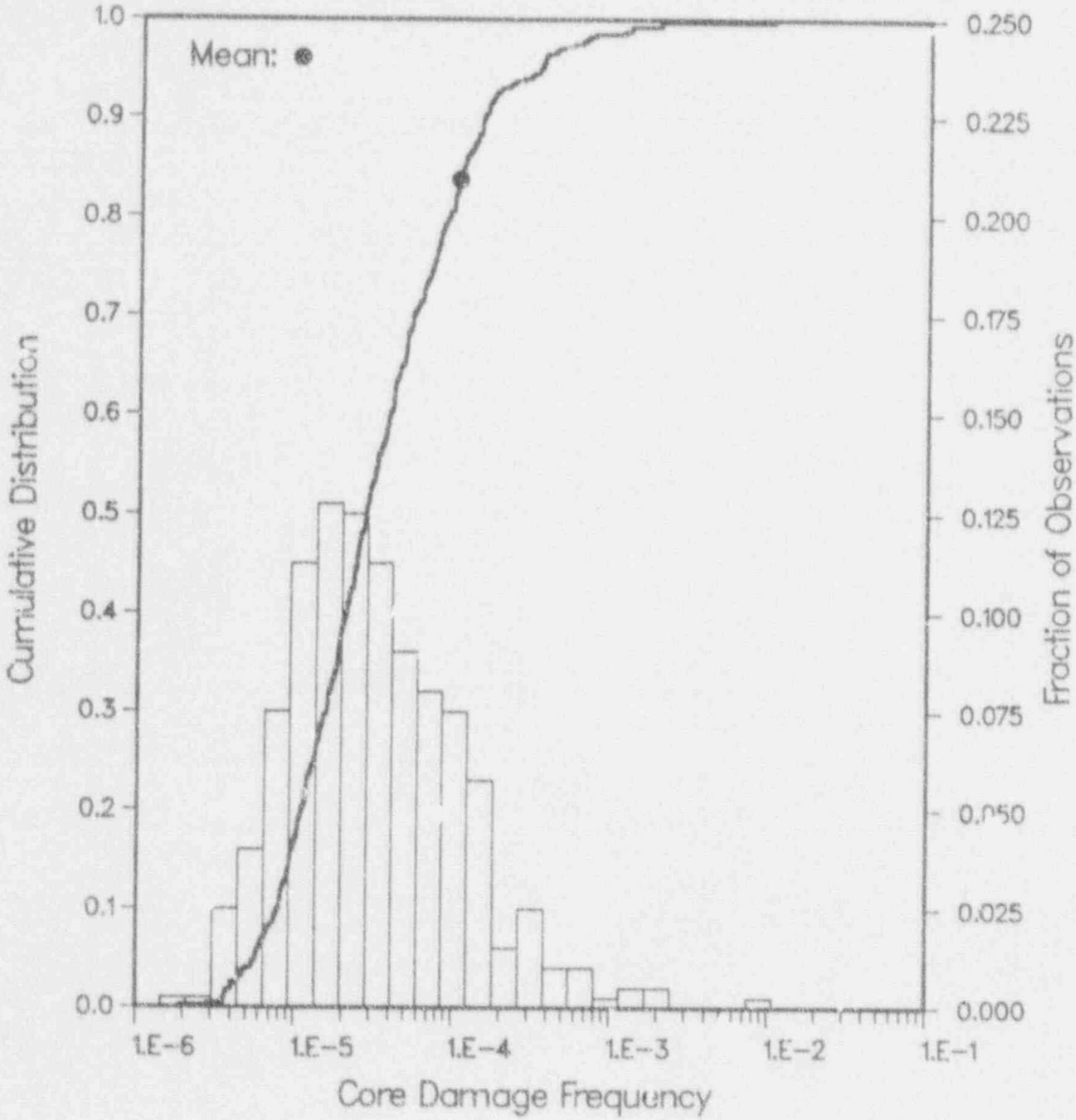


Figure 4.1-1 Integrated Core Damage Frequency Distribution For LaSalle.

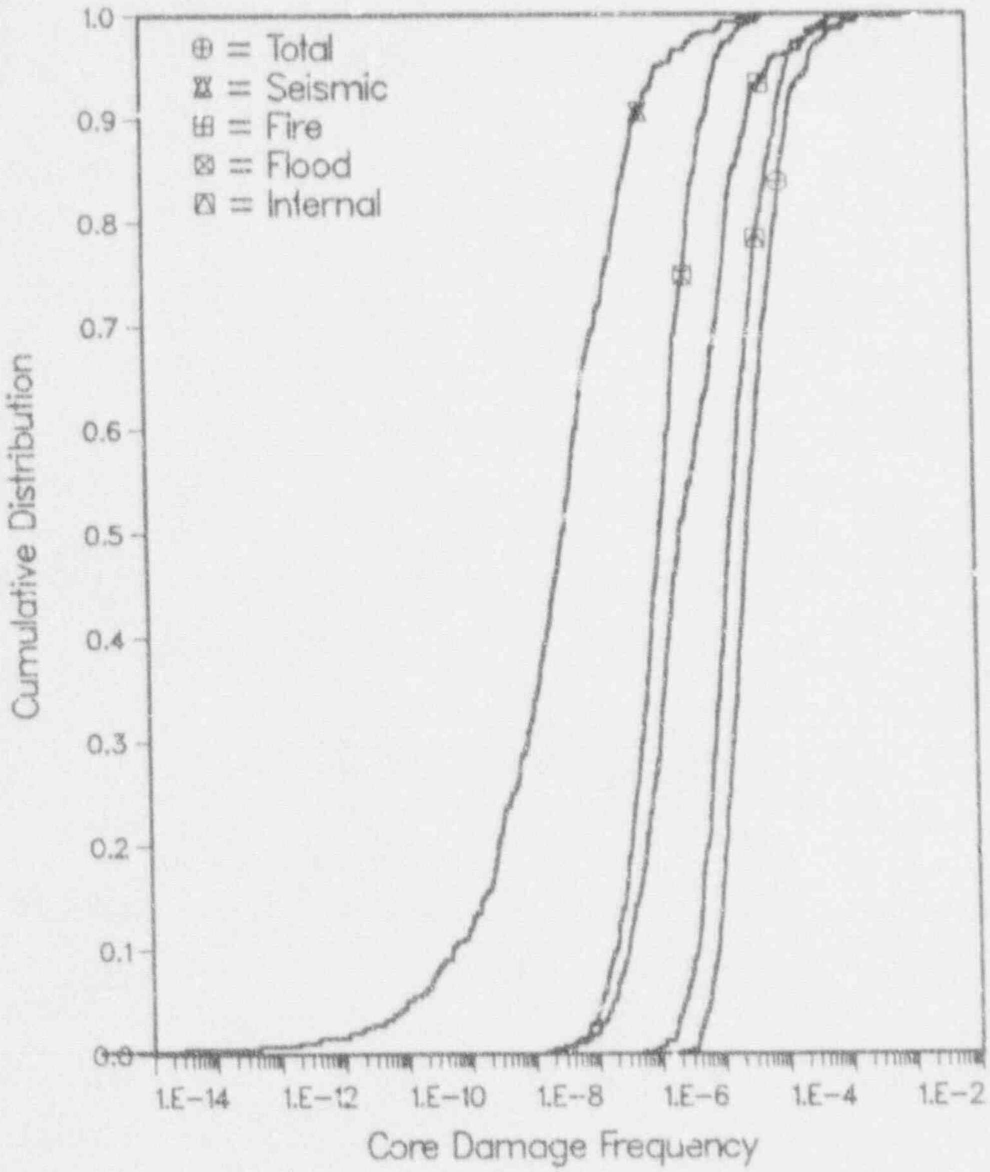
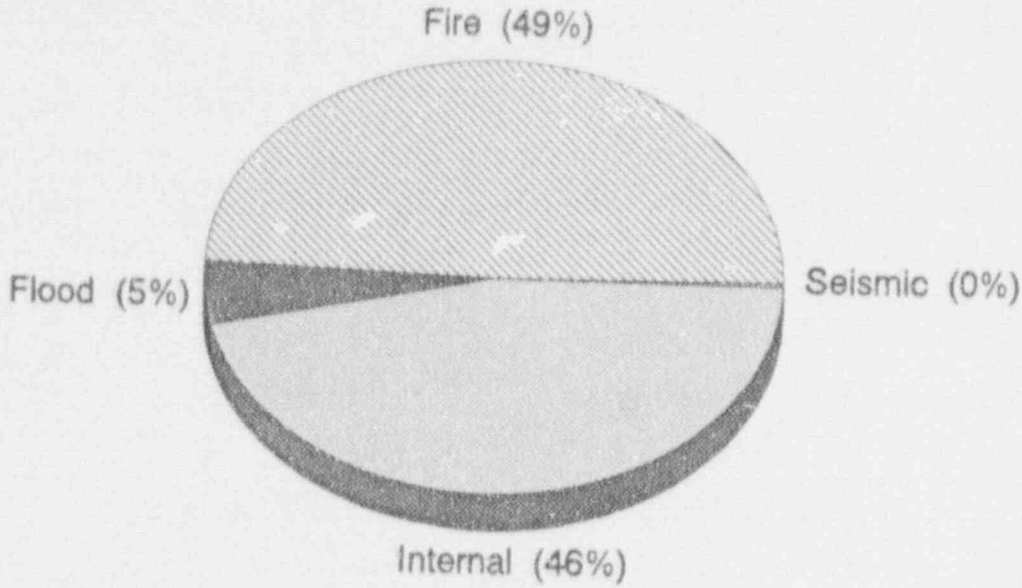
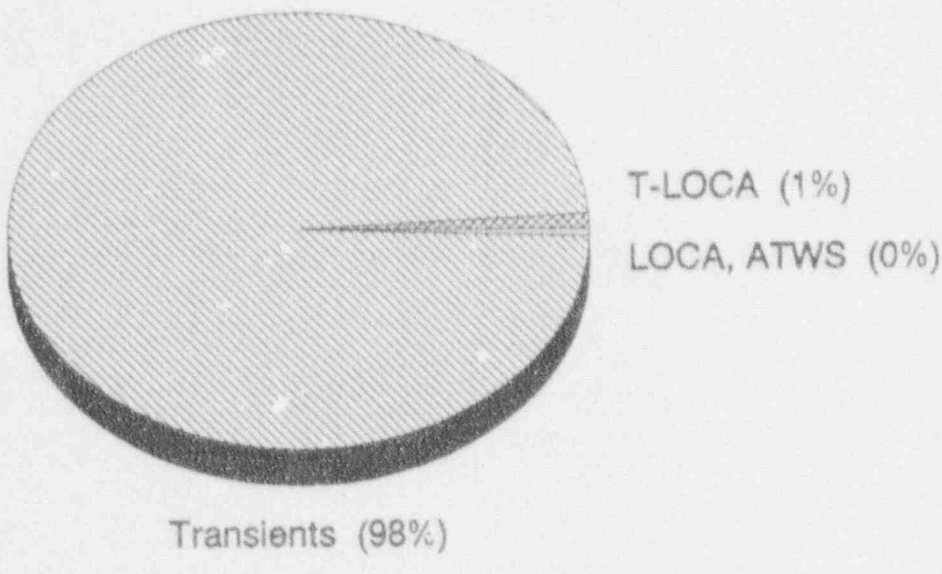


Figure 4.1-2 Fire, Flood, Seismic, Internal, and Integrated Core Damage CDF



a. Percentage of Total Core Damage Frequency



b. Percentage of Internal Core Damage Frequency

Figure 4.1-3 Contribution to Integrated Core Damage Frequency.

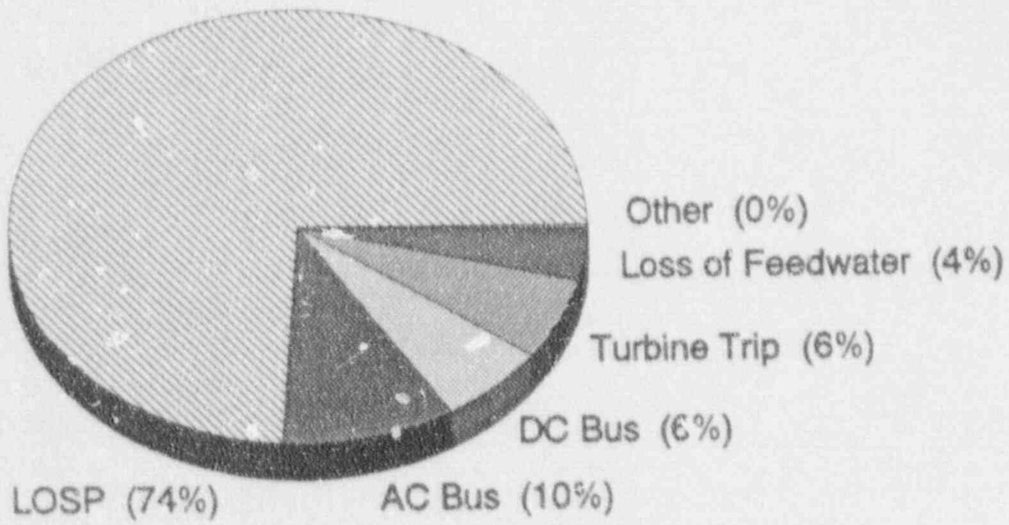


Figure 4.1-4 Initiator Contribution to Internal Core Damage Frequency.

are broken into: 1) LOSP, 2) AC Bus Failure (T101, T102), 3) DC Bus Failure (T9A, T9B), 4) Turbine Trip (T1, T2), 5) Loss of Feedwater (T3, T4, T5), and 6) All Others.

By examining the above plots and figures, one can see that seismic sequences do not contribute significantly to the integrated core damage frequency at LaSalle. Flood sequences are moderate contributors at all quantiles of the distribution. Since the integrated core damage frequency distribution is very similar to the internal events core damage frequency distribution in all but the 90 to 100th quantile range, the integrated core damage frequency distribution comes mostly from internal events. However, at the very top of the distribution, one can see that the fire sequences contribution actually becomes greater than that for the internal sequences. This occurs at about the 95th percentile The dominant fire sequence is initiated by a control room fire and the sparse fire data for calculating control room fire initiating event frequencies results in a distribution with very wide uncertainty bounds. The mean value of the fire core damage frequency is dominated by a few of the 400 Latin Hypercube observations and, in these cases, the fire contribution can be substantial.

The results of the integrated analysis are presented and discussed in detail in Volume 2 of this report.

4.1.2 Dominant Sequences of the Integrated Analysis

In this section we will discuss the characteristics of the dominant sequences which individually contribute greater than 1% of the total core damage frequency These sequences are listed in Table 4.1-1.

The dominant sequence at LaSalle is T100. This sequence contributes 35.4% of the total mean core damage frequency and involves a transient initiator followed by successful scram, successful opening and reclosing of the SRVs, failure of all high pressure injection, successful depressurization of the primary system, and failure of all low pressure injection. The failure of injection can be either immediate or delayed depending on the particular cut set; however, the dominant cut sets have immediate failure. The dominant cut sets have a loss of offsite power initiator followed by loss of onsite power to the safety buses by common mode failure of the diesel generators leading to a station blackout. RCIC fails either immediately or delayed and core damage results before injection can be restored.

The second most dominant sequence at LaSalle is FIRE-CR. This sequence contributes 17.2% of the total mean core damage frequency and involves a control room cabinet fire. The fire grows into a large fire that is not suppressed in time and control room evacuation is necessary. The fire results in failure of the injection systems and control is not successfully reestablished using the remote shutdown panel. Core damage results from the loss of all injection.

The third most dominant sequence at LaSalle is FIRE-W2. This sequence contributes 8.3% of the total mean core damage frequency and involves a fire in the Unit 2 division 2 essential switchgear room. This is a transient combustible fire that grows large enough to damage the train B equipment cabling that passes through the area. The result is failure of train B RHR and any train B injection systems. Random failure of the other RHR train results in a long-term loss of containment heat removal sequence. Injection into the core is successful from either train A injection systems or HPCS depending on the cut set. The containment pressurizes, venting is not possible because of the loss of train B cabling, and the containment fails on overpressure either by leak or rupture. Depending upon its location, this containment failure (e.g., in the reactor building not to the refueling floor) 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. 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 T62 which contributes 8.1% of the total mean core damage frequency. In this sequence, we have a transient initiator followed by successful scram and SRV operation. All high pressure injection except RCIC fails and containment and primary system heat removal fail. The 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 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 fifth most dominant sequence is T18 which contributes 6.2% of the total mean core damage frequency. 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 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 FIRE-Y2 which contributes 4.2% of the total mean core damage frequency. This is started by a transient combustible fire in the Unit 2 division 1 essential switchgear room. The fire is not suppressed in time and results in failure of train A injection system and RHR. Random failure of train B RHR occurs and results in a long-term loss of containment heat removal sequence. Injection into the core is successful from either train B injection systems or HPCS depending on the cut set. The containment pressurizes, venting is not possible because of the loss of train A cabling, and the containment fails on overpressure either by leak or rupture. Depending upon its location, this containment failure (e.g., in the reactor building not to the refueling floor) 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. 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 FS2 which contributes 3.9% of the total mean core damage frequency. This sequence is initiated by a flood resulting from the rupture of a valve in the service water piping in the southeast corner of the ground floor of the unit 2 reactor building and the operator fails to isolate the flood within 7.3 minutes. The flood directly results in the failure of all systems depending on service water. The flood propagates to the corner rooms resulting in failure of all injection and core damage results.

The eighth most dominant sequence is FIRE-T which contributes 2.8% of the total mean core damage frequency. This sequence involves a transient combustible fire in the Unit 2 auxiliary equipment room. The fire results in the failure of train A cabling and the loss of train A injection and RHR. However, the cabling does not result in the loss of power to the venting system and venting is possible. Two sets of cut sets occur, with and without venting. Random failure of train B RHR results in a long-term loss of containment heat removal sequence similar to FIRE-W2.

The ninth most dominant sequence is FIRE-W1 which contributes 2.2% of the total mean core damage frequency. This sequence is the same as FIRE-W2 except that the fire is in a switchgear cabinet.

The tenth most dominant sequence is FIRE-Y1 which contributes 2.2% of the total mean core damage frequency. This sequence is the same as FIRE-Y2 except that the fire is in a switchgear cabinet.

The eleventh most dominant sequence is T20 which contributes 1.5% of the total mean core damage frequency. 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 twelfth most dominant sequence is T22 which contributes 1.4% of the total mean core damage frequency. 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.

All other sequences contribute less than 1% each to the total core damage frequency and in sum contribute less than 7% of the total core damage frequency.

4.1.3 Dominant Cut Sets of the Integrated Analysis

In this section we will discuss the dominant cut sets appearing in the integrated cut set expression for core damage. The percent contributions are based on the point estimate calculation and are not from the means calculated from the distribution. As a result, the relative importance of the cut sets is not evaluated on the same basis as the sequences. Only cut sets that contribute greater than 1% to the total core damage frequency are discussed. A more complete list of the cut sets can be found in Volume 2 of this report.

The dominant cut set, responsible for 32.2% of the total mean core damage frequency, is the cut set that represents the fire in the control room sequence FIRE-CR. From Table 4.1-1, we see that the mean value of this sequence contributes 17.2% of the total mean core damage frequency while its point estimate is 32%. This cut set represents a fire initiated in the control room, not being suppressed before it grows large enough to require evacuation of the control room, and failure of the operators to recover control of the plant from the remote shutdown panel.

The second and third most dominant cut sets, each responsible for 9.8% of the total mean core damage frequency, are from the T100 sequence. These cut sets represent a loss of offsite power followed by delayed failure of the three diesel generators as a result of the common cause failure of CSCS cooling water. This results in station blackout. RCIC fails due to closure of the the inboard isolation valve due to either high room temperature (while onsite AC power is working) or a RCIC isolation sneak circuit on loss of offsite power. The onsite AC power fails before the operator can restore the isolation valve to its open position and all injection is lost. Offsite AC power is not restored in time and core damage occurs in a minimum of 80 minutes.

The fourth most dominant cut set, responsible for 3.9% of the total mean core damage frequency, is from the internal flood sequence, FS2. This cut set represents an internal flood initiated by a service water valve rupture in the southeast corner of the ground floor. The loss of service water directly fails main feedwater and condensate. The operator fails to identify and isolate the flood within 7.3 minutes and the flood fails the RHR B MCC and floods the HPCS, CRD and LPCS, RCIC corner rooms. Overflow also reaches the RHR A room. All injection systems have, therefore, failed and core damage results.

The fifth most dominant cut set, responsible for 1.1% of the total mean core damage frequency, is from the T62 sequence. This cut set involves a loss of offsite power followed by failure of all three diesel generators from common cause failure of the CSCS cooling water pumps. This results in a station blackout. Unlike the T100 sequence RCIC is successful and runs for about 6 hours when it fails on either battery depletion or high pressure in the primary containment resulting in system trip. Offsite power is not restored within 8 hours and delayed core damage results.

These five cut sets contribute 56.6% of the total mean core damage frequency. All other cut sets contribute less than 1.0% each to the total core damage frequency. There are, however, a lot of them and they make up the other 44%.

4.1.4 Risk Reduction Measures for the Integrated Analysis

The risk reduction measure describes the effect on the core damage frequency of decreasing the failure probability or frequency of a specific failure to zero. The component failure or event is assumed not to occur. The total core damage frequency is then reevaluated with this event at zero and the change in total core damage frequency is the risk reduction measure. A complete list of the risk reduction measures for all events contributing to the integrated core damage frequency is given in Volume 2 of this report. Only those events with a risk reduction greater than about 1.0E-05/yr. are discussed here. This measure identifies those events where, if one could reduce the failure probability or modify the design to eliminate the dependency on this event, significant reduction in core damage frequency could be obtained.

The dominant event for risk reduction is the loss of offsite power initiating event frequency with a risk reduction of 2.?6E-05/yr. This event directly affects the frequency of the dominant sequence T100.

Three events are of second, third, and fourth importance for risk reduction. They are all associated with the control room fire sequence: the control room fire initiating event frequency, the failure to suppress the fire, and the failure to recover using the remote shutdown panel. Reduction of any one of these events reduces the second most dominant sequence. Each has a risk reduction of 2.18E-05/yr.

The fifth most dominant event is the failure to recover offsite power within 1 hour with a risk reduction of 1.89E-05/yr. This event is in most of the dominant cut sets for the T100 sequence.

The sixth and seventh most dominant events are the diesel generator cooling water pump common cause beta factor and the pump random failure event associated with this failure. Each has a risk reduction of 1.77E-05/yr.

The eighth most dominant event is high RCIC room temperature resulting in closure of the RCIC inboard isolation valve in station blackout sequences where one of the train A or B diesels runs for a while before failing. The risk reduction is 1.08E-05/yr.

The ninth most dominant event, with a risk reduction of 9.56E-06/yr., is the probability of containment failure by leakage. This event represents the long-term structural failure of the containment from overpressure in loss of containment heat removal sequences. Depending on the location and size of the failure, severe environments can be created in the reactor building which can fail injection systems supplying water to the core whose components are in the reactor building.

The tenth most dominant event, with a risk reduction of 9.26E-06/yr., is FS2 which represents the severity ratio for large fires. This event appears in many of the fire sequences and is the percentage of fires which can be classified as large. In many cases only large fires can result in sufficient damage to create the sequences.

All other events reduce risk less than 1.0E-5/yr. Their descriptions can be found in Appendix B of Volume 2 of this report.

There are two events which have negative risk reduction. For these events, the risk can increase as their probability of occurrence decreases. These two events represent the probability that the operator will vent the primary containment within two hours of reaching the venting setpoint. Since venting the containment will result in severe environments being created in the reactor building, there is some high probability that the systems which are maintaining core cooling and have equipment in the reactor building will fail and core damage will result. If the operator does not vent, the containment will pressurize until it fails on overpressure; however, the most likely containment failure mode is by leakage through the drywell head to the refueling floor. This failure mode will not produce severe environments in the reactor building so systems which are currently working would not fail. The conclusion is that the conditional probability of core damage is less for containment structural failure to the reactor building and subsequent system failure (since most is to the refueling floor) than for venting followed by system failure.

4.1.5 Risk Increase Measures for the Integrated Analysis

The risk increase measure describes the effect on the core damage frequency of increasing the failure probability of a specific event to 1.0. Since initiating event frequencies can be greater than 1.0/yr., they are not, in general, included in this risk measure (i.e., all events beginning with IE- are considered initiators and not evaluated for this calculation, all events which can have values larger than 1.0 must be identified in this manner). For this measure, the component failure or event is assumed to occur all the time. The total core damage frequency is then reevaluated with this event at 1.0 and the change in total core damage frequency is the risk increase measure. This measure identifies those events where an increase in the failure probability from the current level of reliability can result in a significant increase in core damage frequency. It is important, therefore, to insure that these events remain at or below their current failure probabilities. A complete list appears in Appendix A of Volume 2 of this report. Only events with a risk increase greater than 1.0E-03 yr. are discussed.

The dominant event for risk increase with a risk increase of 1.58E-01/yr., is the service water pipe failure frequency for internal floods. This event represents the frequency of pipe failure and is multiplied by the number of feet of pipe of a particular type to get the initiating frequency for a specific flood initiator. Since the number of feet of pipe is greater than 1.0, the event representing the length of pipe was defined with the IE- described above and the actual pipe failure frequency being significantly less than 1.0/yr. was defined as a basic event. Because this particular flood fails all of the responding systems in itself, the core damage frequency of this accident is directly proportional to this event. Since this failure rate is very small, an increase to 1.0 results in a large increase in core damage frequency.

The second most dominant event, with a risk increase of 2.89E-02/yr., is failure of the emergency AC power breaker from 4160 VAC ECCS safety bus 242Y to 480 VAC MCC 236. This event results in failure of much of the train B safety and non-safety equipment and contributes to the dominant core damage sequences.

The third most dominant event, with a risk increase of 1.16E-02/yr., is the reactor protection system failure to scram probability. ATWS accident sequences at LaSalle are not significant contributors to the total core damage frequency; but, their frequency is directly proportional to the failure to scram probability which is 1.0E-05/D. If this event is increased to 1.0 these sequences frequencies increase dramatically.

The fourth most dominant event, with a risk increase of 7.05E-03/yr., is the random failure of the CSCS cooling water pumps used in the calculation of the CSCS cooling water common cause failure probability. The CSCS pumps supply cooling water to the diesel generators and to all of the ECCS equipment rooms and some of the ECCS pumps. Since loss of offsite power followed by common cause failure of the diesel generators resulting in station blackout is the dominant accident sequence at LaSalle, this event will clearly be important.

The next three events, fifth to seventh most dominant, all represent the failure or unavailabilities of parts of the electric power system and have risk increases of 2.24E-03/yr., 1.41E-03/yr., and 1.12E-03/yr., respectively. They are failure of the circuit breaker from 480 VAC MCC 236 to 480 VAC MCC 236Y, the maintenance unavailability of 4160 VAC bus 242Y, and the failure of the circuit breaker to from 4160 VAC bus 241Y to 480 VAC MCC 235. These events all contribute to a partial loss of AC power.

All other events increase risk by less than 1.0E-03/yr. and their descriptions can be found in Appendix B of Volume 2 of this report.

There are three events that can have negative risk increase measures. This implies that increasing these failure probabilities decreases risk. The two events representing the failure of the operator to vent the containment that lead to negative risk decrease measures also appear here. The interpretation is that, since the most likely structural failure is to the refueling floor which does not create severe environments in the reactor building, converting venting into structural failure reduces risk. That is, some percentage of the sequences which would have gone to core damage if venting occurred do not if venting fails and structural failure occurs instead. The CONT-LEAK event represents the probability of structural failure by leakage. Since the containment must fail by some mode in these sequences, as the probability of leakage increases the complement event, containment failure by rupture, decreases. If rupture occurs, then low pressure injection systems can be used to cool the core when containment pressure drops below the ADS reclosure pressure. Some of these low pressure systems such as condensate and diesel-driven firewater are not directly affected by severe environments in the reactor building. This would imply that increasing leakage would increase core damage probability; however, a substantial portion of the leakage probability is leakage to the refueling floor which does not create severe environments in the reactor building. There is a trade-off between these two effects and the reduction in core damage probability from redirecting leakage outside the reactor building is larger than the increase in core damage probability from not being able to use some low pressure systems. Therefore, the core damage probability is reduced by converting ruptures to leaks (Note: this depends critically on the dominant sequence characteristics and should not be generalized to all plants).

4.1.6 Uncertainty Importance Measures for the Integrated Analysis

The uncertainty importance calculation is done differently than the risk reduction and risk increase calculations. The other two importance calculations are done on the point estimate for each individual basic event or initiating event appearing in the cut sets. 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 variable). In the Latin Hypercube 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 variab , and has the same value for a particular LHS sample member. The uncertainty importance calculation is done 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 events and initiating events.

A complete list of events contributing to the uncertainty in the integrated risk appears in Appendix A of Volume 2 of this report. Only events which effect a greater than 5% reduction in the variance of the log of the top event will be discussed.

The dominant class of events is that class representing control circuit failure. This class contributes a 20.8% reduction in the variance of the log of the top event. Valve, pump, fan, and circuit breaker control circuits are included in this class.

The second most dominant class is the variable representing the fire in the control room initiating event frequency. This class is respons bl for a 15.3% reduction in the variance of the log of the top event. to the sparse data base for control room fires, the distribution for this vent is highly skewed and very wide.

The third most dominant class is relay failure to close. This class is responsible for a 12.8% reduction in the variance of the log of the top event.

The fourth most dominant class is relay failure to remain closed. This class is responsible for a 12.7% reduction in the variance of the log of the top event.

The fifth and sixth most dominant classes are again relay failure to close and failure to remain closed. These classes are each sponsible for a 12.5% reduction in the variance of the log of the top event. The difference between these two classes and the previous classes is that a separate but Latin Hypercube variables were used to represent these two classes. The two sets of relays have significantly different test intervals from the previous two that result in very different unavailabilities. This difference in test interval was assumed to break some of the correlation between the failure probabilities.

The seventh and eighth most dominant classes are failure of some SBLC relays which also have a unique test intervals. The two classes again represent failure to close and failure to remain closed. Each class is responsible for a 12.4% reduction in the variance of the log of the top eve t.

The ninth to eleventh most dominant classes are events which represent failure of equipment in the severe environments produced in the reactor building after containment failure by leakage. These events are a combination of the conditional probability of containment failure to the reactor building and the failure of various systems equipment due to the severe environments. The classes represent 9.8, 8.8, and 8.7% reductions in the variance of the log of the top event, respectively.

The twelfth and thirteenth most dominant classes are composed of events which represent circuit breaker failure to remain closed. These classes represent 5.? and 5.1% reductions in the variance of the log of the top event, respectively. The difference is again in the different test intervals for the two sets of circuit breakers.

All other classes represent less than 5% reductions in the variance of the log of the top event. The events that compose them can be understood by looking up the event descriptions in Appendix B of Volume 2 of this report.

4.2 Summary of the Results of the Internal Events Analysis

4.2.1 Dominant Internal Event Sequences

The results of the internal events analysis are presented and discussed in detail in Volume 3 of this report. Table 4.1-1 includes all fifty-four of the sequences that survived the internal events screening process. The sequences are ordered from most dominant to least dominant as determined by the mean value from the TEMAC calculation.

The mean core damage frequency for internal events is 4.41E-05/yr. for the LaSalle plant. The lower 5th %-ile = 2.05E-06/yr., the median = 1.64E-05/yr., and the 95th %-ile = 1.39E-04/yr. A CDF of the core damage frequency resulting from internal event initiated sequences is given in Figure 4.1-2.

The mean core damage frequency is low enough that the NRC's tentative safety goals can be met and is low considering that this is the first time a detailed PRA has been performed on the plant. Typical internal event core damage frequencies obtained in the past for first time PRAs have been in the low 1.0E-4/yr. range. This is usually due to the identification of some design and construction errors that resulted 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 internal sequence is T100 which contributes 64.1% of the mean 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 RCIC fails and (2) a late core damage scenario where AC works for a while and then fails. For the late scenario, there is about 10 hours for recovery actions to be completed. For the early scenario, there is about 80 minutes.

The second most dominant internal sequence is T62 which contributes 14.6% of the mean 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 except RCIC fails and containment and primary system heat removal fail. The 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 DC 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 internal sequence is T18 which contributes 11.1% of the mean 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 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 internal sequence is T20 which contributes 2.9% of the mean 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 internal sequence is T22 which contributes 2.5% of the mean 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.

All other internal sequences contribute less than 5% total to the mean core damage frequency from internal events.

The highest internal ATWS sequence is A49 at 8.94E-08/yr. and is the twelfth most dominant sequence contributing only 0.2% of the mean core damage frequency from internal events. In this sequence, we have a transient initiator followed by initially successful main feedwater. The PCS system fails which leads to the failure of the feedwater turbine-driven pumps from loss of steam or inadequate level in the condenser. The operator then fails to control the motor-driven feedwater pump injection rate to less than the CST makeup rate of 1800 gpm (the corresponding RPV level is 2/3 TAF) resulting in pump trip and loss of all feedwater. The HPCS system works; but the 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 failed containment.

The highest internal LOCA sequence is L14 at 1.72E-08/yr. and is the twenty-first most dominant sequence contributing only 0.04% of the mean 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.

4.2.2 Dominant Cut Sets for the Internal Events analysis

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 frequency. A point estimate TEMAC run was made and the cut sets were truncated at 99% of the point estimate for retention in the uncertainty calculations. Originally there were 11,452 cut sets and after truncation 3589 cut sets remained. TEMAC was then used to perform a full uncertainty calculation on the remaining cut sets. A complete list of the internal initiator cut sets after truncation can be found in Appendix A of Volume 3 of this report.

The two dominant cut sets represent short-term station blackouts resulting from a loss of offsite power followed by a common mode failure of the CSCS cooling water pumps which fails the diesel generators and ECCS room cooling. In the first dominant cut set, responsible for 21.2% of the mean core damage frequency from internal events, 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 21.2% of the mean core damage frequency from internal events, 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 mean core damage frequency from internal events, 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 cut sets, while correct in themselves, double count some of the frequency contribution because they are 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 mean core damage frequency from internal events, consist 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 ECC 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 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.

4.2.3 Risk Reduction Measures for Internal Events

Risk reductions for each individual sequence and the integrated internal event results are presented in the TEMAC outputs shown in Appendix A of Volume 3 of this report. In this section, we will discuss only the integrated internal event results.

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 rather than 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 containment failure probability by rupture. In the integrated result these effects are balanced out somewhat. However, two events even in the integrated analysis have negative risk reduction measure. These two events, OPFAIL-VENT-2H and RA-5V-1-2H, represent success of operator venting of the containment. Venting, using the current procedures, creates severe environments in the reactor building that can result in failure of injection systems and lead 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/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/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/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/yr. and 8.87E-06/yr., respectively.

4.2.4 Risk Increase Measures for Internal Events

Risk increases for each individual sequence and the integrated internal event results are presented in the TEMAC outputs shown in Appendix A of Volume 3 of this report. In this section, we will discuss only the integrated internal event results.

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. In addition, the CONT-LEAK event also has a negative risk increase. For example, as the probability of the operator failing to vent increases the core damage frequency goes down 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 VAC emergency bus 242Y (train B) to 480 VAC buses 236X and 236Y with a risk increase of 2.89E-02/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/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/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.

4.2.5 Uncertainty Importance Measures for Internal Events

For the LaSalle analysis, the result of this calculation for each accident sequence and for the integrated internal event results are presented in Appendix A of Volume 3 of this report. Only the integrated internal event results will be discussed in this section.

The dominant class of events, responsible for a 28.6% reduction in the variance of the log of the top event, 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 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 is 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, respectively.

4.3 Summary of the Results of the Internal Fire Analysis

4.3.1 Dominant Fire Sequences

The results of the internal fire analysis are presented and discussed in detail in Volume 9 of this report. Table 4.1-1 lists all fifteen of the sequences that survived the screening process. The sequences are ordered from most dominant to least dominant as determined by the mean valve from the TEMAC calculation.

The mean core damage frequency for fire events is 3.21E-05/yr. for the LaSalle plant. The lower 5th %-ile = 1.32E-07/yr , the median = 1.99E-06/yr., and the 95th %-ile = 5.94E-05/yr. A CDF of the core damage frequency resulting from internal fire initiated sequences is given in Figure 4.1-2.

The most dominant fire sequence at LaSalle is FIRE-CR. This sequence contributes 43.3% of the total mean core damage frequency from fires and involves a control room cabinet fire. The fire grows into a large fire that is not suppressed in time and control room evacuation is necessary. The fire results in failure of the injection systems and control is not successfully reestablished using the remote shutdown panel. Core damage results from the loss of all injection.

The second most dominant fire sequence at LaSalle is FIRE-W2. This sequence contributes 20.9% of the total mean core damage frequency from fires and involves a fire in the Unit 2 division 2 essential switchgear room. This is a transient combustible fire that grows large enough to damage the train B equipment cabling that passes through the area. The result is failure of train B RHR and any train B injection systems. Random failure of the other RHR train results in a long-term loss of containment heat removal sequence. Injection into the core is successful from either train A injection systems or HPCS depending on the cut set. The containment pressurizes, venting is not possible because of the loss of train B cabling, and the containment fails on overpressure either by leak or rupture. Depending upon its location, this containment failure (e.g., in the reactor building not to the refueling floor) 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. 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 fire sequence is FIRE-Y2 which contributes 10.6% of the total mean core damage frequency from fires. This is started by a transient combustable fire in the Unit 2 division 1 essential switchgear room. The fire is not suppressed in time and results in failure of train A injection system and RHR. Random failure of train B RHR occurs and results in a long-term loss of containment heat removal sequence. Injection into the core is successful from either train B injection systems or HPCS depending on the cut set. The containment pressurizes, venting is not possible because of the loss of train A cabling, and the containment fails on overpressure either by leak or rupture. Depending upon its location, this containment failure (e.g., in the reactor building not to the refueling floor) 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. 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 fire sequence is FIRE-T which contributes 7.1% of the total mean core damage frequency from fires. This sequence involves a transient combustible fire in the Unit 2 auxiliary equipment room. The fire results in the failure of train A cabling and the loss of train A injection and RHR. However, the cabling does not result in the loss of power to the venting system and venting is possible. Two sets of cut sets occur, with and without venting. Random failure of train B RHR results in a long-term loss of containment heat removal sequence similar to FIRE-W2.

The fifth most dominant fire sequence is FIRE-W1 which contributes 5.6% of the total mean core damage frequency from fires. This sequence is the same as FIRE-W2 except that the fire is in a switchgear cabinet.

The sixth most dominant fire sequence is FIRE-Y1 which contributes 5.5% of the total mean core damage frequency from fires. This sequence is the same as FIRE-Y2 except that the fire is in a switchgear cabinet.

All other sequences contribute less than 5% total to the core damage frequency from fires.

4.3.2 Dominant Cut Sets for the Fire analysis

In order to obtain an integrated result for internal fire events, all of the cut sets from all of the sequences were merged together to form one large expression representing the total fire core damage possibilities. TEMAC was then used to perform an uncertainty analysis and all of the cut sets were included. A complete list of the cut sets for the individual and integrated calculations can be found in Appendix F of Volume 9 of this report.

The dominant cut set, responsible for 64.5% of the mean core damage frequency from fires, is the cut set that represents the fire in the control room sequence FIRE-CR. This cut set represents a fire initiating in the control room, not being suppressed before it grows large enough to require evacuation of the control room, and failure of the operators to recover control of the plant from the remote shutdown panel.

The second most dominant cut set, responsible for 1.6% of the mean core damage frequency from fires, is the dominant cut set in the FIRE-W2 sequence. This cut set represents a large transient combustible fire starting in a switchgear room, and failing train B RHR and any train B injection systems. Random failure of the other RHR train by blockage of the RHR heat exchanger results in a long-term loss of containment heat removal sequence. Injection into the core is successful from HPCS. The containment pressurizes, venting is not possible because of the loss of train B cabling, and the containment fails on overpressure by leakage. The containment failure is to the reactor building, not to the refueling floor, and produces a severe environment which causes the HPCS system to fail resulting in core damage.

The third most dominant cut set, responsible for 1.5% of the fire mean core damage frequency, is the dominant cut set in the FIRE-E-S3 sequence. This cut set represents a transient combustible fire in the corridor adjacent to the Unit 2, Division 1, essential switchgear room. The fire is not suppressed in time and failure of offsite power and train A and B power to ECCS systems occurs. AC power is still available for venting and venting is successful. After venting, severe environments are produced in the reactor building and fail the HPCS and diesel-driven fire water systems resulting in core damage

The fourth most dominant cut set, responsible for 1.2% of the fire mean core damage frequency, is the dominant cut set in the FIRE-T sequence. This cut set involves a transient combustible fire in the Unit 2 auxiliary equipment room. The fire results in the failure of train A cabling and the loss of train A injection and RHR. Random failure of the train B RHR heat exchanger by blockage results in a long-term loss of containment heat removal sequence similar to FIRE-W2. However, the fire did not fail cabling to the venting system and venting is successful. After venting, severe environments are produced in the reactor building and fail the HPCS and diesel-driven fire water systems resulting in core damage.

The fifth most dominant cut set, responsible for 1.1% of the fire mean core damage frequency, is the dominant cut set in the FIRE-Y2 sequence. This cut set represents a transient combustible fire in the Unit 2 division 1 essential switchgear room combustible fire. The fire is not suppressed in time and results in failure of train A injection system and RHR. Failure of the train B RHR heat exchanger by blockage occurs and results in a long-term loss of containment heat removal sequence. Injection into the core by HPCS is successful. The containment pressurizes, venting is not possible because of the loss of train A cabling, and the containment fails on overpressure by leakage. Depending upon its location, this containment failure (e.g., in the reactor building not to the refueling floor) will produce an environment which could cause HPCS to fail resulting in core damage.

4.3.3 Risk Reduction Measures for Fire Initiators

For the internal fire analysis, the fire initiating event frequencies were not labeled with IE- and so they appear on the same table with the basic events. The calculation is correct so no change was made. For the integrated calculation described in section 4.1 of this report, the event names were modified to include the IE- and the fire initiating event frequencies appear with the other initiators. Risk reductions for each individual sequence and the integrated fire results are presented in the TEMAC outputs shown in Appendix F of Volume 9 of this report. In this section, we will discuss only the integrated fire results.

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 the fire sequences, unlike the internal event sequences, both the event and its complement can appear in the same sequence but in different cut sets. For the fire analysis, one event has a negative risk reduction measure. This event, OPFAIL-VENT-2H, represents successful venting of the containment. Venting using the current procedures creates severe environments in the reactor building that can fail injection systems and thus the sequence proceeds to core damage. 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 three most important events for risk reduction for fire initiated sequences all occur in the dominant fire sequence and are related to control room fires: the probability that the operators will unsuccessfully control the plant from the remote shutdown panel, the control room fire initiating event frequency, and the fraction of control room fires that are not suppressed before smoke forces abandonment of the control room. Each with a risk reduction measure of 2.18E-05/yr. The fourth most dominant event is the fraction of fires that are large fires with a risk reduction measure of 9.32E-06/yr. The fifth most dominant event is late failure of HPCS from severe environments in the reactor building after containment leakage with a risk reduction measure of 8.80E-06/yr.

4.3.4 Risk Increase Measures for Fire Initiators

Risk increase measures are in general calculated only for basic events. Since initiating events are frequencies and can have values greater than 1.0, this calculations is not usually applicable to them. For the fire analysis, since that fire initiating events were not labeled with IE-, the fire frequencies, which are all less than 1.0/yr., were included in the risk increase calculation. For these events, the risk increase calculation shows the impact of increasing their frequencies to 1.0/yr. This labeling error is corrected in the integrated results presented in Section 4.` of this report. Risk increases for each individual sequence and the integrated fire results are presented in the TEMAC outputs shown in Appendix F of Volume 9 of this report. In this section, we will discuss only the integrated fire results.

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, two events that have negative risk increases. For example, as the probability of the operator failing to vent increases the core damage frequency goes down because, for the dominant sequences, there is less probability of severe environments if the containment fails than if its vented as described in the previous section.

The dominant event from a risk increase standpoint is the frequency of control room fires with a risk increase measure of 3.40E-03/yr. The second most dominant event is the frequency of switchgear room fires with a risk increase measure of 5.56E-04/yr. The third most dominant event is the failure of the operator to successfully control the plant from the remote shutdown panel with a risk increase measure of 3.18E-04/yr. The fourth most dominant event is the area ratio of fire area AC to the area of the auxiliary building. The fifth most dominant event is the fraction of control room fires that are not suppressed before smoke forces abandonment of the control room.

4.3.5 Uncertainty Importance for Fire Initiators

As described in section 4.1.6, 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 variable). For the LaSalle fire analysis, the results of this calculation for each accident sequence and for the integrated fire results are presented in Appendix F of Volume 9 of this report. Only the integrated fire results will be discussed in this section.

The most dominant class in the event representing the failure of the HPCS system from the severe environments created in the reactor building after containment failure by leakage. It is responsible for a 33.8% decrease in the variance of the log of the top event. The second most dominant class is the variable representing the fire in the control room initiating event frequency. This class is responsible for a 17.1% reduction in the variance of the log of the top event. Due to the sparse data base for control room fires, the distribution for this event is highly skewed and very wide. The third most dominant class is the event representing the late failure of HPCS and diesel-driven fire water from severe environments in the reactor building after containment venting. This class is responsible for a 12.5% reduction in the variance of the log of the top event. The fourth most dominant class is the event representing the frequency of fires initiated in the auxiliary building. This class is responsible for a 7.2% reduction in the variance of the log of the top event. The fifth most dominant class is the event representing the area ratio of fire area T to the area of the auxiliary building. This class is responsible for a 5.1% reduction in the variance of the log of the top event.

All other classes contribute less than a 5% reduction in the variance of the log of the top event. Many of the random failures begin contributing just below the 5% level. If one looks at the complete set of importance uncertainty results in Appendix F of Volume 5 of this report, it can be seen that both random failure and fire events contribute significantly at all levels of importance and uncertainty. This result comes about because the LaSalle design requires both random and fire events to occur in most cut sets leading to core damage and, therefore, uncertainties in both groups of events are relatively equal. This result would not occur for plants which had less physical separation of redundant safety systems.

4.4 Summary of the Results of the Internal Flood Analysis

4.4.1 Dominant Flood Sequences

The results of the internal flood analysis are presented and discussed in detail in Volume 10 of this report. Table 4.1-1 lists the two sequences that survived the internal flood screening process. The sequences are ordered from most dominant to least dominant as determined by the mean value from the TEMAC calculation.

The mean core damage frequency for internally initiated flood events is 3.39E-06/yr. for the LaSalle plant. The lower 5th %-ile = 9.62E-08/yr., the median = 1.13E-06/yr., and the 95th %-ile = 3.23E-06/yr. A CDF of the core damage frequency resulting from internal flood initiated sequences is given in Figure 4.1-2.

The most dominant sequence is FS2 which contributes 93.7% of the total mean core damage frequency from internal floods. This sequence is initiated by a flood resulting either from the rupture of the pipe or a valve in the service water piping in the southeast corner of the ground floor of the unit 2 reactor building (location 3G.1) and the operator fails to isolate the flood within 7.3 minutes. The flood directly results in the failure of all systems depending on service water (i.e., MFW and CDS). The flood fails the LPCI train B and C 480 AC MCC on that level failing those trains of RHR. The flood also propagates to the corner rooms 312 (southwest) and 314 (northeast) resulting in failure of HPCS and CRD and LPCS and RCIC, respectively. The operator does not isolate the flood in time and the water level in corner room 314 gets high enough to drain into room 315 (northwest) failing LPCI train A. All injection has failed and core damage results.

The second dominant sequence is FS1 which contributes 6.3% of the total mean core damage frequency from internal floods. This sequence is initiated by a flood resulting from the rupture of service water piping on an upper level (3%) of the reactor building. The flood results in direct failure of MFW and CDS. This flood fails the low pressure injection system pressure permissives on that floor resulting in the failure of LPCI trains A, B, and C. The flood will propagate to the 312 and 314 corner rooms and fail HPCS and CRD and LPCS and RCIC, respectively. It will also drain into room 315 from 314 and fail LPCI A. All injection has failed and core damage results.

4.4.2 Dominant Cut Sets for the Flood analysis

In order to obtain an integrated result for internal flood initiators, all of the cut sets from all of the sequences were merged to form one expression representing the total internal flood core damage frequency. An uncertainty calculation was made using TEMAC that included all of the cut sets. A complete list of the cut sets appears in Volume 10 of this report.

The dominant cut set, responsible for 82.5% of the internal flood mean core damage frequency, represents a flood resulting from failure of the service water system piping on the ground floor of the reactor building, i.e., flood sequence FS2. The flood is initiated by the rupture of a valve in the piping in the southeast corner of the floor. Flood annunciators succeed but the operators are not successful in identifying and isolating the flood within 7.3 minutes. The flood fails all mitigating systems as described above under flood sequence FS2 and core damage results.

The second most dominant cut set, responsible for 10.7% of the internal flood mean core damage frequency, represents the same flood as in the most dominant cut set. The only difference is that the flood is initiated by a pipe break not a valve failure.

The third most dominant cut set, responsible for 5.8% of the internal flood mean core damage frequency, represents a flood resulting from failure of the service water piping on an upper level of the reactor building, i.e., flood sequence FS1. The flood fails the LPCI low pressure permissives on that floor and then fails the other mitigating systems, as described in the discussion of flood sequence FS1 above, resulting in core damage.

All other cut sets contribute less than 1% each to the total internal flood core damage frequency.

4.4.3 Risk Reduction Measures for Flood Initiators

For the internal flood analysis, the event representing the pipe failure frequency was not labeled as an initiating event. In TEMAC, variables whose values can be greater than 1.0 must be labelled as initiators. Since the flood initiating event frequency is really the product of the length of pipe available for the flood under consideration times the frequency of pipe failure per unit length of pipe, it was decided to define the length of pipe as the initiating event and leave the pipe failure frequency per unit length of pipe as a basic event. Risk reductions for each individual sequence and the integrated flood results are presented in the TEMAC outputs shown in Appendix F of Volume 10 of this report. In this section, we will discuss only the integrated flood results.

The event AVAIL-FAC has the highest risk reduction value, 3.23E-06/yr. This makes sense because if the plant was never available the accident could not occur. However, this does not help us much because we want the plant to operate as much as possible and would not want to reduce this event's probability. The next highest risk reduction, 3.20E-06/yr., is for the event OP-FTISOL-FLOOD. This event represents failure of the operator to identify and isolate the flood within 7.3 minutes. Normally, reactor building flooding indication is not too specific and to identify and isolate a particular flood before sufficient water is released to damage critical equipment will not be straight forward. The third highest risk reduction, 2.69E-06/yr., is for the event IE-VALVE-RUP. This event represents the frequency of external valve rupture with no leak before break.

4.4.4 Risk Increase Measures for Flood Initiators

Risk increases for each individual sequence and the integrated flood results are presented in the TEMAC outputs shown in Appendix F of Volume 10 of this report. In this section, we will discuss only the integrated flood results.

The event with the highest risk increase, 1.58E-01/yr., is PIPE-FREQ. This event represents the pipe rupture before leak frequency. TEMAC does not calculate risk increase measures for initiating events identified with the IE- prefix as normally initiators can have values greater than 1.0/yr. However, in this case, we labeled the pipe length as the initiating event so the PIPE-FREQ event was set to 1.0 and a risk increase was calculated.

A similar number could have been calculated for the IE-VALVE-RUP event since it's frequency is much less than 1.0/yr. If this had been done, the valve rupture event would also have had a large risk increase. Obviously, keeping the frequency of pipe and valve ruptures low is important to maintaining a low core damage frequency since these floods in and of themselves can fail sufficient equipment to cause core damage. The next highest events are ANNUC-FAILURE and OP-FTISOL-FLOOD. Each of which has a risk increase of 3.23E-05/yr. Identifying and isolating the flood are of equal value in risk increase. It is interesting to note that the AVAIL-FAC event is the lowest in risk increase while the highest in risk decrease. This implies that improving the availability will not significantly increase the frequency of core damage while decreasing the availability will significantly decrease the core damage frequency.

4.4.5 Uncertainty Importance Measures for Flood Initiators

As described in section 4.1.6, 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 variable). For the LaSalle internal flood analysis, the results of this calculation for each accident sequence and for the integrated flood results are presented in Appendix F of Volume 10 of this report. Only the integrated flood results will be discussed in this section.

The dominant class of events, responsible for a 74.8% reduction in the variance of the log of the top event, represents the valve rupture frequency. The second most dominant class, responsible for a 10.5% reduction in the variance of the log of the top event, represents the pipe failure frequency. The third most dominant class, responsible for a 7.0% reduction in the variance of the log of the top event, represents the failure of the operator to identify and isolate the floods within 7.3 minutes.

4.5 Summary of the Results of the Seismic Analysis

4.5.1 Dominant Seismic Sequences

The results of the seismic analysis are presented and discussed in detail in Volumes 2 and 8 of this report. Table 4.1-1 presents the results of the TEMAC calculations for each individual accident sequence which survived the initial screening analysis. Each sequence is identified by initiator type and seismic level.

The mean core damage frequency was 7.58E-07/yr. with a 5-%ile of 4.07E-11/yr., a median of 1.74E-08/yr., and a 95%-ile of 1.21E-06/yr. A CDF of the core damage frequency resulting from seismically initiated sequences is given in Figure 4.1-2.

The primary characteristic of the dominant sequences at LaSalle is that the only explicitly seismic events appearing in the final cut sets are the seismic initiating event frequencies for each level and the seismically-induced loss of offsite power conditional probabilities at each level. No other seismic failures or seismic related events survived the initial and final quantification. This is very different than the results for many other plants. The LaSalle plant is very well designed from a seismic viewpoint. The detailed structural analysis performed in Volume 8 did not find any structural failures where walls might fall and damage critical equipment, the cabinets and panels were bolted down correctly, and the piping penetrations were designed appropriately to handle any shifting as a result of the seismic event. The accident sequences look, therefore, like seismically-induced transients.

If LOSP was not likely to occur as a result of the seismic event, there would be no dominant seismic sequences at LaSalle. No other seismically-induced initiator has a significant conditional probability and compromises redundancy enough to result in accident sequences with a substantial frequency. The dominant sequences at LaSalle are, therefore, all seismically induced losses of offsite power and look exactly like the equivalent internally initiated sequences except that no credit is given for recovering offsite power after the seismic failure.

Another characteristic of the seismic sequences that follows from that in the previous paragraph is that, for a particular accident sequence, the cut sets for the sequence will be the same for all of the seven seismic hazard levels being analyzed except for the hazard frequency itself and the conditional probability of the LOSP at that level. When one looks at Table 4.1-1, you see, for example, that seismic sequence TLOSP-01 is replicated for each of the seven levels. If one examines the TEMAC output in Appendix C of Volume 2 of this report, you will see that the cut sets for each level are identical except for the level indicator on the initiating event frequency and the conditional LOSP probability. The frequency of the sequence is directly proportional to the hazard frequency at that level and, since the conditional probability of ceramic insulator failure increases at a slower rate than the rate at which the hazard frequency decreases, the accident sequence frequency drops with increasing hazard level.

There were six sequences that survived for final quantification: TLOSP-01, TLOSP-03, TL1-01, TL1-03, TL2-01, and TL3-01. Each of these sequences was evaluated for each of the seven different hazard levels used in the analysis which is shown by the .LXX extension on the sequence name. TLOSP-01 corresponds to internal event sequence T63 and is an intermediate term station blackout where RCIC works for about six hours and then fails on high containment pressure or battery depletion. HPCS may also be working but the HPCS diesel fails at eight hours and in either case core damage ensues. TLOSP-03 corresponds to internal event sequence T100 where RCIC fails initially due to the closure of the inboard isolation valve on high room temperature before all onsite AC power is lost and can not be unisolated later because of the secondary failure of AC power. HPCS works initially; but, again, the HPCS diesel fails at about eight hours and core damage ensues. The TLX-0Y sequences are sequences which have the additional failure of 1, 2, or 3 of the SRVs to reclose and, therefore, are transient-induced LOCAs similar to those in the internal events analysis except that they occur with a simultaneous LOSP. The TL1-01 sequence corresponds to internal sequence TL60 which is similar to T63 described above. The TL1-03, TL2-01, and TL3-01 sequences correspond to internal sequence TL97 which is similar to T100 described above.

4.5.2 Dominant Cut Sets for the Seismic Analysis

In order to obtain an integrated result for seismic events, all of the cut sets from all of the sequences were merged to form one large expression representing the total seismic core damage possibilities. TEMAC was then used to perform an uncertainty analysis and all of the cut sets were included. A complete list of the cut sets for the individual and integrated seismic calculations can be found in Appendix C of Volume 2 of this report.

The top seven cut sets all come from the TLOSP-01 sequence and correspond to the dominant cut set in that sequence evaluated at each of the seven seismic levels. The cut sets represent a seismically-induced LOSP followed by a common mode failure of the diesel generator CSCS cooling water pumps. RCIC works for six hours when it fails on battery depletion or high containment pressure. Core damage ensues several hours later after boil-off. These cut sets are responsible for 73.7% of the mean core damage frequency f seismic events.

.5.3 Risk Reduction Measures for Seismic Initiators

Risk reductions for each individual sequence and the integrated seismic results are presented in the TEMAC outputs in Appendix C of Volume 2 of this report. In this section, we will discuss only the integrate' seismic results.

The dominant events at Lasalle for the seismic analysis are the two events associated with the common mode failure of the CSCS cooling water pumps. The events are the random pump failure probability and the common mode beta factor that multiplies it to create the common mode failure probability. Each event has a risk reduction value of 2.20E-07/yr.

The third and fourth dominant events are the L1 level hazard frequency and seismically-induced LOSP probabilities. Each of these has a 1.44E-07/yr. risk reduction. The initiating events for each level have a monotonically decreasing risk reduction as the hazard level increases and their corresponding contribution to the core damage frequency decreases.

The fifth most dominant event is the non-recovery probat lity of offsite power within eight hours. Since no credit is given for recovery of offsite power in this analysis, clearly an improved estimate of the possibility of recovering offsite power after seismic events would be worthwhile. This event has a risk reduction of 1.14E-07/yr.

4.5.4 Risk Increase Measures for Seismic Initiators

Risk increases for each individual sequence and the integrated seismic results are presented in the TEMAC outputs shown in Appendix C of Volume 2 of this report. In this section, we will discuss only the integrated seismic results.

The dominant event for risk increase is the CSCS cooling water pump common mode beta factor with a risk increase of 1.81E-05/yr. The second most dominant event is the random failure of the CSCS cooling water pumps responsible for a 8.59E-06/yr. risk increase. The next three events (the third, fourth and fifth) represent failure of one of the diesels to run. Since the system configuration is not quite symmetrical the "2A" DG has a 2.50E-06/yr. risk increase, the "2B" DG has a 1.83E-06/yr. increase, and the "0" DG has a 1.80E-06/yr. increase. Similarly for the sixth, seventh, and eighth dominant events represent failure of the diesels to start. The "2B" DG has a 1.69E-06/yr. risk increase, the "0" DG has a 1.66E-06/yr. risk increase, and the "2A" DG has a 1.18E-06/yr. risk increase.

4.5.5 Uncertainty Importance Measures for Seismic Initiators

For the LaSalle analysis, the result of this calculation for each accident sequence and for the integrated seismic results are presented in Appendix C of Volume 2 of this report. Only the integrated seismic results will be discussed in this section.

The dominant class is the conditional probability of ceramic insulator failure resulting in loss of offsite power. This class is responsible for a 52.5% reduction in the variance of the log of the top event.

The second through eighth most dominant classes are the seismic initiating frequencies at the various levels. These events have different distributions but are correlated and are responsible for 33.5-32.0% reductions in the variance of the log of the top event.

The ninth most dominant class, responsible for a 7.3% reduction in the variance of the log of the top event, is the diesel generator failure to run.

The tenth most dominant class, responsible for a 6.3% reduction in the variance of the log of the top event, is the diesel generator failure to start.

All other events are responsible for less than a 5% reduction in the variance of the log of the top event.

4.6 Important Issues and Insights

4.6.1 Seismic Hazard Curve

The seismic hazard curves used in this PRA were based on an early method where the hazard curve was developed for a hypothetical rock outcrop and a site specific soil correction was applied. Since that time, the LLNL method has changed. In the new method, the sites are put into one of eight categories and a generic correction for a given soil category is applied. The soil correction models are significantly different. In addition, there is the curve produced by the Electric Power Research Institute (EPRI) methodology.

Figure 4.6-1 shows the three seismic hazard curves. Because of the different fashion in which the curves are produced and the different factors included in the curves, it is not possible to simply scale the results to perform a sensitivity analysis. In order to determine the impact of the different curves on the final core damage frequency, the analysis would need to be redone.

4.6.2 Relay Chatter

In a separate report,[1] the possible effects of seismically induced relay chatter on the Lasalle core damage frequency were analyzed. System design information, fault trees, and the seismic hazard curve, fragilities, and structural analysis results were all supplied to the analysis team.

In this analysis, relay chatter was found to be potentially important in the electric power system, the automatic depressurization system (ADS), and in the reactor core isolation cooling (RCIC) system. The analysis in this report takes no credit for operator recovery, so the result only indicates the potential importance.

For the LaSalle Level I PRA, these failures were considered; but, they were determined to be probabilistically unimportant to the seismic core damage frequency. This was because, in all cases, the seal-ins can be recovered by the operator and, in the most pessimistic sequence, the operator has about 80 minutes before core damage begins. The mean operator failure to recover probability would be of the order of 2.0E-03 and these cut sets would be significantly less likely than the current dominant cut sets which involve a station blackout with minimal operator recovery potential.

4.6.3 Loss of Off-Site Power Frequency

As described in Reference 2, a new method for calculating the loss of off-site power initiating event frequency distribution and distributions for the probability of recovering offsite power by time t was developed for the LaSalle PRA. The mean frequency of LOSP at LaSalle calculated using this method was 0.096/yr, the 5th %-ile is 0.024/yr., the median is 0.085/yr., and the 95th %-ile is 0.20/yr. The method used assumes each plant has an individual incidence rate for LOSP occurrence which belongs to a superpopulation of incidence rates. A distribution for the superpopulation incidence rate is calculated from the historical data and then a Baysian-based procedure using plant specific data for plant-centered, grid, and weather-induced LOSPs is used to determine a plant specific incidence rate and distribution. At the time the calculation was done, no LOSP had occurred in four operating years (if one includes both units 1 and 2).

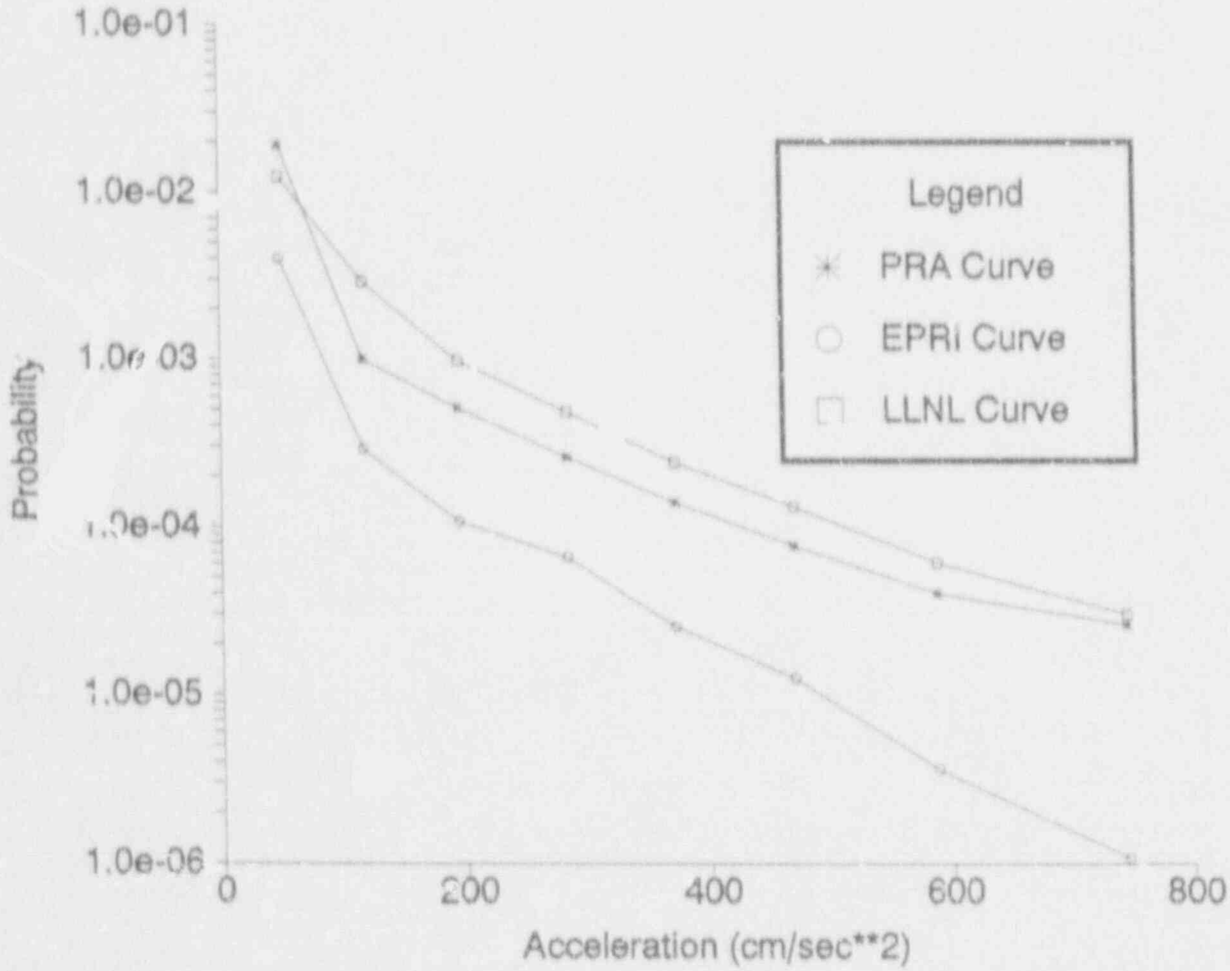


Figure 4.6-1 Comparison of LaSalle Seismic Hazard Curves

Commonwealth Edison Company (CECo), the owner of the LaSalle plant, had contracted with General Electric to perform a PRA in parallel with the NRC sponsored effort described in this report. The results of that analysis are reported in Reference 3. In GE's report, a value of 2.4E-03/yr. is used for the LOSP initiating event frequency based on a CECo analysis[4,5] of the unavailability of safety buses at the LaSalle Station.

SNL analysts felt that this value was much too low based on the historical data. However, CECo felt that they had corrected or designed away the faults that had resulted in prior LOSPs on other grids. In 1989, a LOSP occurred at the LaSalle plant. Assuming one event in 8 operating years, this implies a rate of 0.13/yr. However, since the PRA is trying to represent an accident occurring at any time in the lifetime of the plant, this particular value would not necessarily be the correct one to use. As time goes on and no additional LOSP occurs, the average value per year will decrease. It is judged that the uncertainty distribution used in this analysis will adequately represent any year to year variation in the LOSP initiating event frequency.

This particular difference of opinion is symptomatic, however, of a much larger problem. In the estimation of failure rates for various events, many analysts try to argue away all the failure data by saying that they either fixed the problem or designed a new device that does not have the problem. The result is, in our opinion, a large underestimation of the failure probability. While it is true that one hopes that design improvements and fixes will improve system reliability, we believe that it is clear from experience that, in general, the reliability of new systems improves much slower than would be estimated. This stems from two considerations. First, people do not make a large effort to estimate all the new failure modes they have introduced and all the old ones which have not occurred yet. Second, these failure rates are very small and what is being estimated is the occurrence rate of any unlikely event that could result in system failure. Just because one fixes a particular fault does not mean that one has significantly affected the underlying failure rate.

There are clearly different ways of incorporating the switchyard failures into the analysis. The way done in this analysis is to use generic statistical data on loss of offsite power then modify that data by taking into account switchyard type, grid, and weather and, finally, to update this information with the plant specific data. An alternate method would involve constructing a detailed fault tree model of the switchyard, interfacing this with the in-plant models and the external events such as weather and grid effects, and gathering generic and plant-specific data to quantify the model. This second method does have the appeal of being consistent with current modeling methods and we would recommend going this direction in the future. It is not clear; however, that significantly different final results would be obtained since the underlying data should be consistent and the final answers should be approximately the same.

4.6.4 Containment Venting

As described in Section 4.1, the current procedure for venting the containment contains some positive and negative aspects. Currently, the operator is directed to vent first from the wetwell through the standby gas treatment system. The piping from the containment is connected to the SBGTs by a rubber boot. When the 24" valves are opened, the boot is almost certain to fail. Also, the SBGTS filter box, which is not designed to withstand such pressures, will likely fail. This will occur in both reactor buildings, since the Unit I and Unit II SBGTS are connected. The resulting severe environments have a substantial probability of causing equipment belonging to possible mitigating systems which are located in the reactor building to fail. The result would be the loss of injection and subsequent core damage.

Let us examine the possible cases that appear in the dominant sequences:

1. Loss of RHR and high pressure injection works. The containment pressurizes until the vent threshold is reached at which time the operator vents the containment. In the dominant sequences, HPCS and CRD are the two possible high pressure injection systems that could be working. Both of these system are located in the southwest corner cubicle which is open at the ground floor to the rest of the reactor building. The results of the expert elicitation on equipment failure indicated a mean probability of 0.97 and 0.99, respectively for failure of the systems in the severe environments created in the reactor building if venting occurs at this time.

   What if the operator did nothing and the containment continued to pressurize until structural failure occurred? The most likely containment failure mode (about 0.67) is by the drywell head lifting and the release will be to the refueling floor. In this case, no severe environments will be produced in the reactor building and the high pressure injection systems should continue to work. The sequence is a success. In the case of failure to the reactor building, the probability of system failure is similar to the venting case. Therefore, if the operator vents, core damage is almost certain to occur; while, if the operator does nothing, one-third of the time core damage is likely.

2. Loss of RHR and only low pressure injection working (LPCI, LPCS, CDS, DFW). The containment pressurizes until the venting threshold is reached at which time the operator vents the containment. If CDS or DFW is working, then core damage is not likely since these systems have no components in the reactor building whose failure would result in system failure. However, in the dominant sequences either LPCI or LPCS is working. These systems have many components located in the reactor building. The failure probability from severe environments is about 0.66 because

most components are located in closed corner cubicles (northwest and southeast).

If the operator does nothing, then the containment will pressurize to the SRV reclosure pressure. When the SRVs reclose, the vessel will repressurize and low pressure injection will be lost (for LOCA or transient-induced LOCAs, the vessel will remain depressurized; but, these are not dominant sequences). Core damage will result from loss of injection before the containment pressure reaches the structural failure point.

The result of the probabilistic analysis of the two scenarios can be summarized as: It is not benifical to vent if high pressure injection is available and it is benifical to vent if only low pressure injection is available. In either case, however, venting will result in some possibility of core damage.

A possible change in the venting procedure would eliminate this possibility. A hard pipe vent path exists which goes directly to the steam tunnel. Releasing the steam into the steam tunnel will result in the blowout panel on the roof opening and directing most of the steam to the outside. Some steam would go under the main turbine and could leak up into the turbine building. However, the flow resistance is high and not much steam is expected to go by that path and not many critical components of the safety systems are located where this would affect them. In this case, whether one has high or low pressure injection, venting will not result in system failure.

This type of venting would be similar to that to which the Peach Bottom plant changed as a result of the NUREG-1150 analysis.[6.7] Peach Bottom has a 6-inch pipe through which they can vent directly to the outside, thereby, not creating severe environments in the reactor building.

One note of caution, however, if core damage has already occurred then venting into the reactor building, which would tend to retain more fission products, could lead to smaller radioactive releases. This would be important in the Level II/III analysis.

4.6.5 RCIC Isolation

As part of the PRA, simulator tests were run for various postulated accident sequence scenarios on the LaSalle plant simulator. These tests were run to determine the human error probabilities. During these tests, it was found that the RCIC system was isolating every time a loss of off-site power occurred. This was traced to a contact timing problem in the isolation circuitry.

On LOSP, part of the isolation circuitry is deenergized. This results in a contact in the DC powered portion of the steam leak detection circuit closing. This would simulate a high room temperature signal resulting in closure of the AC powered in-board steam isolation valve except that the valve power is also lost and it can not close. In addition, the loss of power relays in the AC powered portion of the isolation circuit deenergize resulting in their contacts in the DC powered portion of the isolation circuit opening. When on-site AC power is started, AC power to the valve is restored and the loss of power contacts in the DC portion close before the relay in the AC powered portion, which controls the remaining contact in the DC portion, closes. The result is that the DC portion of the circuit is momentarily closed. This directs the valve to close due to high room temperature. The contact locking in the in-board AC powered MOV close circuit is energized before the loss of power contact in the DC portion reopens and the in-board steam isolation valve closes.

If onsite AC power continued to work and no other AC powered systems were available, the operator would have about 80 minutes before core damage would begin and there would be plenty of time to reopen the valve. However, in the case of a station blackout resulting from the failure of diesel generator cooling water, the DGs would start and load; but, they would then fail due to the loss of cooling. This delayed failure would result in the valve closing before the diesels would be likely to fail. After the diesels failed, the valve would not be able to be reopened. All core cooling would be lost and core damage would result unless AC power could be restored in time.

A similar effect happens when one train of AC power fails immediately and the other has a delayed failure. The RCIC steam line has two AC powered isolation valves powered from different AC trains. One valve is in-board and the other out-board. In this particular case, RCIC either will not isolate on starting of the AC train or AC will continue to be available to reopen whichever valve closes. However, if even one train of AC power is available, RCIC will isolate in about 20 minutes on high room temperature. As long as that train continues to work, the isolation can be overridden; but, if some other system is available for injection, it may not be. If the train subsequently fails and the operator has not reopened the isolation valve, he may not be able to depending on which valve closed (i.e., if the out-board valve was the one that had power, then it can be manually opened locally).

Two events were introduced into the fault trees to account for these effects. They are: OPFAILS-REOPEN and RCICRMCOOL-FLAG, respectively. Fault trees are not very good at modeling time dependent effects and, in order to model these effects correctly, the curve for the probability of failure of the operating DG with time would have to be convolved with the curve for the probability of the operator recovering the valve within a certain time. In addition, each cut set could have slightly different timing depending on what was causing the DG failure. As a result, it was decided to conservatively model these events as not being recoverable.

The utility has instituted a design change to correct this problem.

4.6.6 RPS Failure Probability

Because of the complexity of the reactor protection system, a separate detailed fault tree was not constructed for this system in this analysis. An FMEA was performed on the RPS system and all interfaces to other systems were identified. For any other system that relied on RPS logic, that portion of the logic was incorporated into the system's fault tree in detail. The RPS system failure was treated as an undeveloped event and a single number was used to represent failure to scram in the quantification of the ATWS sequences. This number was 3E-05 of which 2E-05 as considered to be due to electrical faults and recoverable.

Since LaSalle has an alternate rod insertion system which operates on a diverse principle and since most electrical faults can be bypassed by deenergizing the RPS electrical buses, electrical faults were considered to be negligible compared to the mechanical faults and not included in the quantification. The final mean value used was, therefore, 1E-05/D for the conditional probability of failure on demand and was non-recoverable.

This approach is similar to that used in NUREG-1150 to quantify the RPS failure probability (see the Peach Bottom analysis as an example[6]).

4.6.7 Use of Qualitative Fire Information In Plant Operations

The location based information that can be obtained from the type of analysis done for this project can be used in many different ways. An example of the use of this information would be to prioritize the areas for increased inspection for the fire watch to reduce the risk from fires.

When trains of various systems are taken out for maintenance, repair, or testing; the results of the fire analysis can be transformed to allow the identification of the areas where because of the reduced redundancy, fires can have the most potential for leading to core damage. Table 4.6-1 shows for the dominant fire areas resulting from the LaSalle analysis, those systems which must fail randomly in addition to those failed by the fire before core damage can occur. By inverting this table, see Table 4.6-2, one can construct a list of the fire areas one should be concerned about if certain systems are made unavailable for various reasons. By increasing the fire inspections for these areas or by making other adjustments to the treatment of these locations which could reduce the possibilities of fires in those areas, the risk from fires can be reduced.

4.6.8 Quality Assurance

The RMIEP program had a very extensive QA plan consisting of both in-house review of the results of each task by someone who had not performed the work, and by an external QA team consisting of many of the leading practitioners in the PRA community, the NRC project manager, and the utility and their architect engineer, Sargent and Lundy.

Table 4.6-1
Dominant Fire Areas and Associated Random Failures

Fire Area*	Random Failures
G	None, Control Room
E(S2)	HPCS
E(S3)	Venting
N	RHR-A
P	RHR-B
T	Venting and RHR-B
S(AA)	DG-A and RHR-A
S(W)	RHR-A
W1	RHR-A
W2	RHR-A
Y1	RHR-B
Y2	RHR-B
Z	RHR-B
AA	DG-A and RHR-A
AC	RHR-B

* See Volume 9 of this report for a detailed description of the fire areas and their significance.

Table 4.6-2
Important Fire Areas Given Unavailability of System

System	Important Fire Areas
Venting	E-S3, T
RHR-A	N, S-W, S-AA, W1, W2, AA
RHR-B	P, T, S-AA, Y1, Y2, Z, AC
DG-A	S-AA, AA
HPCS	E-S2

This plan worked very well until the inception of NUREG-1150. The NUREG-1150 project had a substantial impact on the resources available for the LaSalle PRA and resulted in a substantial lengthening of the schedule. In fact, the LaSalle Level I PRA was not fully completed until after the Level II/III analysis of the Peach bottom NUREG 1150 analysis was completed, since many of the people working on the Peach Bottom analysis were the ones responsible for completing the LaSalle analysis.

The result was that the initial phases of the analysis were reviewed by the whole team; but, the final results of each analysis, i.e., the final seismic, fire, flood, and internal event accident sequence cut sets, were reviewed only by the in-house independent review, the NRC, and the utility.

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11. ABSTRACT (200 words or less)

This volume presents an overview of the methodology and results of the integrated accident sequence analysis (Level I) of the LaSalle Unit 2. nuclear power plant performed as part of the Level III PRA performed by Sandia National Laboratories for the Nuclear Regulatory Commission. The Level II/III results are presented in associated reports described in the Foreword. This volume contains a summary description of the LaSalle plant, describes the contents of the other nine volumes of this report and their relationships to each other, the relationship of the LaSalle program to other programs, a step-by-step summary description of the methodology and new techniques used to perform the analysis, and presents the integrated results obtained by merging all of the accident sequence cut sets from the LOCA, transient, transient-induced LOCAs, and anticipated accidents without scram accident sequences resulting from internal initiators with the cut-sets from the fire, flood, and seismic analyses acci. nt sequences.

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