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Integrated Risk Assessment for the LaSalle Unit 2 Nuclear Power Plant

Phenomenology and Risk Uncertainty Evaluation Program (PRUEP)

Prepared by
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ABSTRACT

A Level III probabilistic risk assessment (PRA) was performed for the LaSalle Unit 2 nuclear power plant. This plant is located in Brookfield Township, LaSalle County, Illinois which is 55 miles southwest of Chicago. The objective of this study was to provide an estimate of the risk to the offsite population during full power operation of the plant and to include a characterization of the uncertainties in the calculated risk values. Uncertainties were included in the accident frequency analysis, accident progression analysis, and the source term analysis. Only weather uncertainties were included in the consequence analysis. The risk estimates presented in this report include contributions from both internal and external initiators.

The offsite risk to the public due to the operation of LaSalle County Station is relatively low, especially with respect to the NRC safety goals. The mean individual early fatality risk within 1 mile is 1.1E-10/R-yr which is more than three orders of magnitude below the safety goal. Similarly, the mean individual latent cancer fatality risk is 8.5E-09/R-yr which is slightly more than two orders of magnitude below the safety goal. In fact, the entire uncertainty distributions for these two risk measures lie below the safety goals. The mean values for early fatality risk and for latent cancer fatality risk are 1.2E-08/R-yr and 0.25/R-yr, respectively.

The low values for risk can be attributed to the low core damage frequency, the fast evacuation of the public away from the plant, and plant features that reduce the source terms that result from a core damage accident. The risk at LaSalle is dominated by the Fire plant damage state (PDS) group and the Transient PDS group. The LOCA, and Transient-Induced LOCA PDS groups, on the other hand, are very minor contributors to the risk. The Flood, Anticipated Transients Without Scram (ATWS), and Seismic groups are in-between. The risk is dominated by accidents that progress to vessel breach and that involve loss of containment integrity. Given that core damage occurs, it is likely that the accident will proceed to vessel breach and the containment's integrity will be compromised. Although notable, the mean conditional probability of core damage arrest prior to vessel breach is fairly small, approximately 0.15. The probability of core damage arrest is driven by the recovery of AC power for the short term station blackouts, the lack of available or recoverable injection systems for the other accidents, and the probability and effects of in-vessel steam explosions. The mean probability that the containment isolation will fail sometime during the accident is 0.88. It is about equally likely that the containment will be vented during the accident, mean probability of 0.46, or will structurally fail, mean probability of 0.42.

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FOREWORD

LaSalle Unit 2 Level III Probabilistic Risk Assessment

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

NUREG/CR-5305 - Volume 1: Main Report

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

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

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

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

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

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LIST OF ACRONYMS

AC	Alternating Current
ADS	Automatic Depressurization System
APB	Accident Progression Bin
APET	Accident Progression Event Tree
ATWS	Anticipated Transeints Without Scram
Ba	Barium
BWR	Boiling Water Reactor
CCDF	Complementary Cumulative Distribution Function
CCI	Core-Concrete Interactions
CD	Core Damage
CDF	Cumulative Distibution System
CDS	Condensate System
CDT	Core Damge Time
Ce	Cerium
CECo	Commonwealth Edison Company
CF	Containment Failure
CH	Chronic Health Effect Weight
CHR	Containment Heat Removal
Cs	Cesium
CSCS	Core Standby Cooling System
CSS	Containment Spray System
DC	Direc. Current
DCH	Direct Containment Heating
DDFW	Deisel-Driven Firewater System
DF	Decontamination Factor
DFWS	Deisel-Driven Firewater System
EF	Early Fatalities
EH	Early Health Effect Weight
EPRI	Electric Power Institute
EPS	Emergency Power System
EPZ	Emergency Planning Zone
EVSE	Ex-Vessel Steam Explosion
FCI	Fuel-Coolant Interaction
FCMCD	Fractional Contribution to CoreDamge
FCMR	.ractional Contribution To Mean Risk
FDA	Food And Drug Administration
FWA	Frequency Weighted Average
GG	Grand Gulf
GI	Gastro-Intestinal
HPCS	High Pressure Core Spray System
HPME	High Pressure Melt Ejection
I	Iodine
La	Lanthanum
LCF	Latent Cancer Fatalities
LHS	Latin Hypercube Sample
LLNL	Lawerence Livermore National Labortatory
LOCA	Loss of Coolant Accident
LOSP	Loss of Off-Site Power

LIST OF ACRONYMS (Concluded)

LPCI	Low pressure Coolant Injection System
LPCS	Low Pressure Core Spray System
LPI	Low Pressure Injection
MCCI	Molten Core-Concrete Interactions (see CCI)
MFCCD	Mean Fractional Contribution to Core Damage
MFCR	Mean Fractional Contribution To Risk
MFW	Main Feedwater System
NRC	Nuclear Regulatory commission
NUS	
PB	Peach Bottom
PDS	Plant Damage State
PRA	Probabilistic Risk Assessment
PRUEP	Phenomenology and Risk Uncertainty Evaluation Program
RBDF	Reactor Building decontamination Factor
RCIC	Reactor Core Isolation Cooling system
RHR	Residual Heat Removal System
RMIEP	Risk M[illegible]ds Integration and Evaluation Program
ROSP	Random [illegible]ss of Off-Site Power
RPV	Reactor Pressure Vessel
Ru	Ruthenium
SARRP	Severe Accident Risk Reduction Program
SB	Station Blackout
SBLC	Standby Liquid Control System
SBO	Station Blackout
SDC	Shutdown Cooling System
SF	Split Fraction
SGTS	Standby Gas Treatment System
SLC	Standby Liquid Control System
SNL	Sandia National Laboratories
SPC	Suppression Pool Cooling System
Sr	Strontium
SRV	Safety Relief Valve
SSMRP	Seismic Safety Margins Research Program
T1	First Release Time
TC	Fail to Scram Accident Sequences (WASH-1400)
TDELAY	Evacuation Deleay Time
Te	Tellurium
TQUV	Short-Term Loss of All Injection Sequence (WASH-1400)
TW	Warning Time
TW	Long-Term Accident Sequence With Loss of CHR (WASH-1400)
VB	Vessel Breach
ZO	Zero-One

S.0 SUMMARY

S.1 Objective and Scope

The objectives of the Level II/III portion of this probabilistic risk assessment (PRA) are:

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.

In this study, the term integrated risk assessment means the combination of the various constituent analyses (i.e., accident frequency analysis, accident progression analysis, source term analysis, and the consequence analysis) to form an expression for risk which includes contributions from all initiators.

The scope of this study includes:

1. Analysis of full power operation of the LaSalle County Station Unit 2 nuclear power plant,

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

3. Estimation of the risk to the offsite population, and

4. Estimation of the combined uncertainties from the accident frequency analysis, accident progression analysis, source term analysis, and limited consequence uncertainties.

S.2 General Description of Methodology

The general steps used to perform the LaSalle Level III PRA are outlined in this section. The framework of this method was developed in this program but was first applied in the NUREG-1150 study.[1] Some of the details of this method are presented in the methodology volume of NUREG/CR-4551.* Some of the methods that are described in NUREG/CR-4551 have been improved for the LaSalle study. These improvements are described in the appropriate sections of this report. The general steps used in this PRA are:

1. Identify and determine the frequency of the accident sequences that lead to core damage that are initiated from both internal and external events. This process is the Level I portion of the PRA and is described in NUREG/CR-4832.**

2. Group the cut sets that are associated with the accident sequences into plant damage states (PDSs) where each PDS presents unique initial and boundary conditions to the accident progression analysis.

3. Identify and determine the conditional probabilities of the many possible accident progression paths following core damage. A detailed accident progression event tree (APET) is used in this portion of the analysis and is evaluated using the EVNTRE code.[2]

4. Group the thousands of paths that are propagated through the APET into bins that represent unique boundary conditions to the source term analysis.

5. Estimate source terms for all of the accident progression bins that have been passed from the accident progression analysis to the source term analysis. The source terms are estimated using a simple parametric code called LASSOR developed for this analysis.

6. The thousands of source terms that are generated in the source term analysis are then combined into source term groups based on similar

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* E.D. Gorham, et al., "Evaluation of Severe Accident Risks: Methodology for the Accident Progression, Source Term, Consequence, Risk Integration and Uncertainty Analyses," NUREG/CR-4551, Vol. 1, Rev. 1, SAND86-1309, Sandia National Laboratories, Albuquerque, NM, to be published.

** A. C. Payne Jr., T. T. Sype, D. W. Whitehead, and A. W. Shiver, "Analysis of the LaSalle Unit 2 Nuclear Power Plant: Risk Methods Integration and Evaluation Program (RMIEP), Volume 2: Integrated Quantification and Uncertainty Analysis," NUREG/CR-4832/2 of 10, SAND92-0537, Sandia National Laboratories, Albuquerque, NM, to be published.

characteristics of the source terms. A frequency-weighted mean source term is determined from each group. This process is referred to as partitioning and is performed with the PARTITION code.[3]

7. Consequences are calculated for each source term group based on the frequency-weighted mean source term for that group. The MACCS code[4] is used to calculate the consequences.

8. Combine the frequency of the accident progression bins with the consequences associated with the APB to form an expression of risk. This process was performed with the PRAMIS code[5] and its associated post processors.

9. Present the risk results in a format such that contributors to risk can be identified and the uncertainty in risk displayed.

S.3 General Description of the Plant

The LaSalle County Station Unit 2 nuclear power plant is located in Brookfield Township, LaSalle County, Illinois which is 55 miles southwest of Chicago. The plant is owned and operated by the Commonwealth Edison Company (CECo). The LaSalle plant utilizes a Mark II type containment to house a General Electric BWR-5 reactor and is rated at 3293 MWt and 1078 MWe.

There are various injection systems that can be used to cool the core after accidents and arrest the core damage process at LaSalle. Four high pressure and four low pressure injection systems are considered in this analysis. 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). All of these systems can inject up to the safety relief valve setpoints (1146-1205 psig). The low pressure injection systems include the low pressure core spray system (LPCS), the low pressure coolant injection system (LPCI), the condensate system, and the diesel-driven firewater system (DFWS). The DFWS is used as a last resort injection system when all other systems have failed. This system can be manually connected to the MFW injection line to provide injection. For these last four systems to provide coolant to the core, the reactor pressure vessel (RPV) must be depressurized from the normal operating pressure of about 1055 psig to 150-500 psig depending on the system operating.

Vessel depressurization is accomplished with the Automatic Depressurization System (ADS) which is designed to depressurize the reactor vessel to a pressure at which the low pressure injection systems can inject coolant to the reactor vessel.

Heat can be removed from the containment by the residual heat removal (RHR) system which uses trains A and B of LPCI system. Suppression pool cooling

(SPC) and the containment spray system (CSS) are two modes of the RHR system. 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 heat exchangers. In the CSS mode, water is sprayed into the drywell. 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 removed from the vessel via a recirculation loop, passed through the RHR heat exchangers, and then injected back into the vessel. The LPCI system can also be used to remove decay heat from the containment indirectly by taking water from the suppression pool, passing it through the heat exchangers, injecting the water into the vessel, and then having the water return to the suppression pool via a LOCA or the SRV discharge lines. All four modes of RHR (i.e., SPC, CSS, SDC, and LPCI with the heat exchangers) require AC power and are, therefore, unavailable during a station blackout.

The primary containment is a post-tensioned reinforced concrete structure with a steel liner. The containment, shown in Figure S-1, consists of a lower cylindrical portion founded on the base mat and an upper portion 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 of the containment 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 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. The primary containment is inerted with nitrogen which eliminates the possibility of hydrogen combustion events during the course of the accident. The internal design pressure of the primary containment is 45 psig. The ultimate containment failure pressure was assessed by a panel of structural experts. 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 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 operators are instructed to vent the containment when the containment pressure exceeds [illegible] psig regardless of whether or not adequate core cooling is available. Venting requires both divisions of AC power.

Directly below the reactor pressure vessel is the reactor pedestal cavity. This cavity consists of an upper portion directly beneath the reactor vessel and a lower portion separated from the upper by a concrete floor.

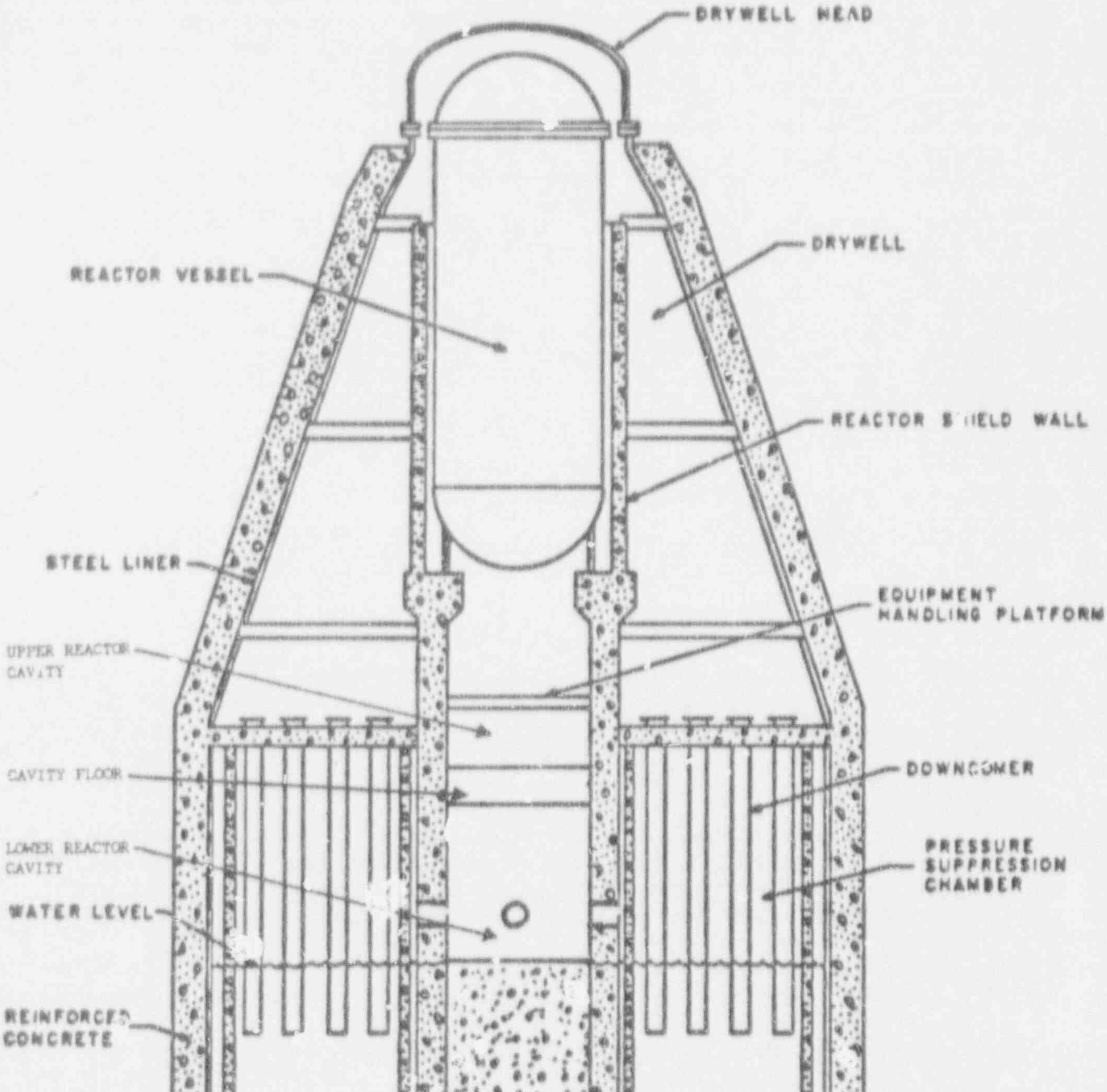

Figure S-1. LaSalle Containment Schematic

Both portions of the cavity are large enough to contain all of the core debris released at the time of vessel breach. In many scenarios, it is possible that the upper cavity floor will fail and the core debris will be released into the lower cavity which connects to the wetwell via some vent holes above the level of core debris. The lower pedestal is partially filed with concrete and contains no water. This design is significantly different from some other Mark II containment designs and the LaSalle results should not be assumed to be valid for other Mark II containments. Because a large amount of the core debris can not enter the drywell due to the recessed cavity design, direct attack of the drywell wall by core debris is not an issue at LaSalle as it is for the Mark I containments and, since the containment is inerted, hydrogen burns are not an issue as in Mark III containments.

The primary containment is enclosed by a reinforced concrete reactor building which forms the secondary containment (Figure S-2). This building houses much of the equipment used by the safety and non-safety systems that can mitigate an accident. Containment failure to the reactor building, which results in severe environments in the reactor building, can result in failure of the mitigating systems. This is a significant contributor to the core damage frequency in the Level I analysis[6].

S.4 Results

S.4.1 Plant Damage State Definition Results

For the LaSalle analysis, 30 plant damage states were defined with some 16 additional sub-plant damage states included in the analysis. These PDSs resulted from a detailed examination of all of the cut sets (combinations of equipment failures resulting in an accident sequence leading to core damage) from the top 50 dominant accident sequences from the Level 1 analysis[6].

The total core damage frequency distribution from the Level I analysis had a mean value of 1.01E-04/R-yr. with a 5th percentile of 5.34E-06/R-yr., a median of 2.92E-05/R-yr., and a 95th percentile of 2.93E-04/R-yr. The Level II/III LHS sample resulted in a mean of 1.04E-04/R-yr., a 5th percentile of 5.74E-06/R-yr., a median of 2.76E-05/R-yr., and a 95th percentile of 3.25E-04/R-yr. The difference between the two distributions is less than 10%. Given that the Level I analysis used 270 primary variables and that the Level II/III analysis used only 103 of the Level I variables for the uncertainty analysis (the other Level I variables were fixed at their mean values), the Level II/III sample is a very good approximation of the Level I sample. Individually, the PDS distributions may show more variation than this; but, the dominant PDSs are very close to the Level I results as a result of the variable selection process described in Chapter 2.

The dominant plant damage states are IT2, FI5, and FL2 with 0.368, 0.107, and 0.105 mean fractional contributions of the total core damage frequency, respectively.

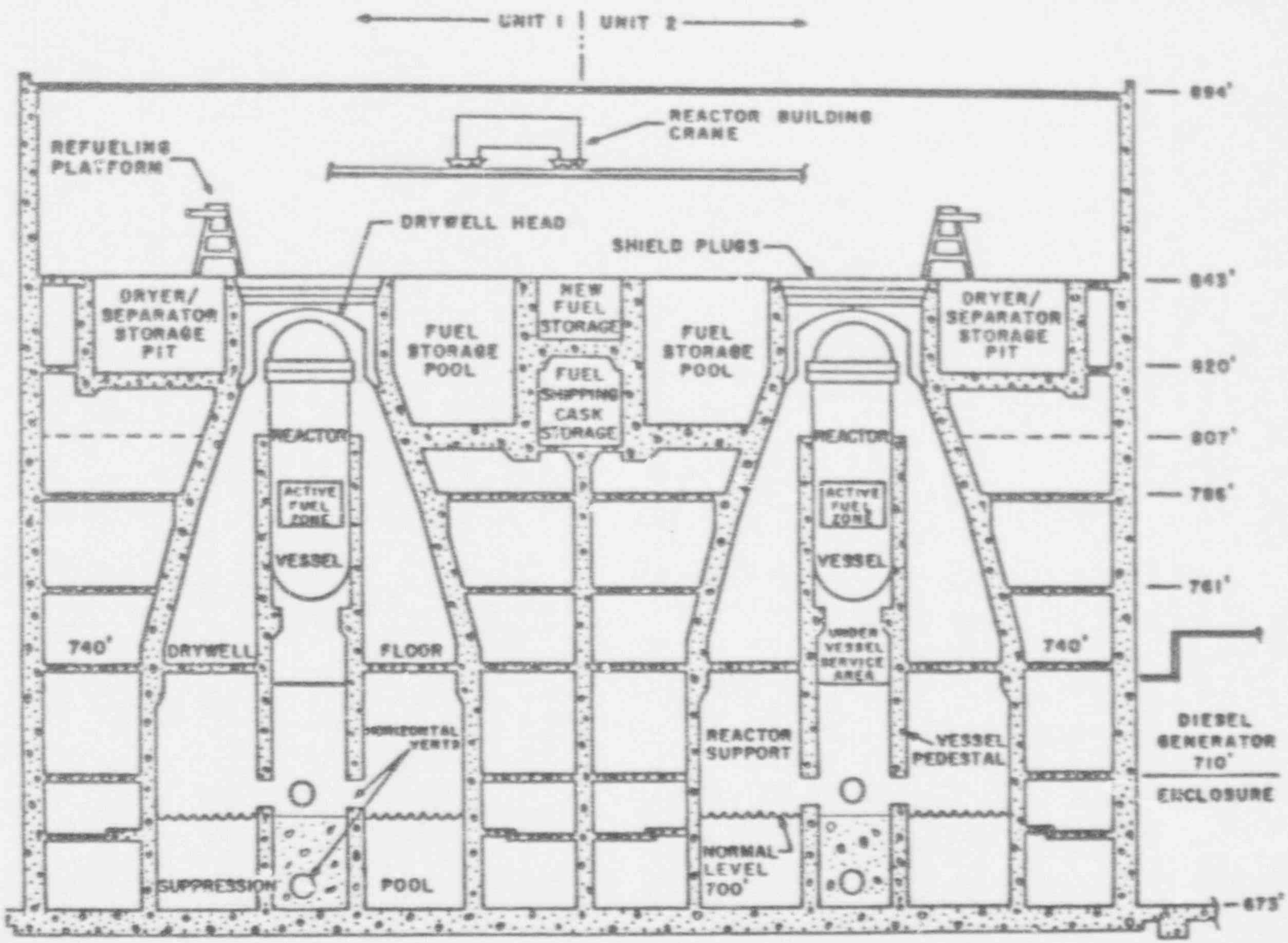

Figure S-2. LaSalle Reactor Building Schematic

IT2 is a transient-initiated short-term station blackout PDS with core damage beginning at about 80 minutes after the accident initiation. Containment failure has not yet occurred and venting is recoverable if AC power is restored. This PDS is composed of three sub-PDSs: 1) ADS and almost all injection systems are recoverable if AC power is restored, 2) ADS is available during the core damage process and almost all injection is recoverable if AC power is restored, and 3) ADS is available during the core damage process but only MFW, CDS, and RCIC are recoverable if AC power is restored.

FI5 is a fire-initiated accident resulting directly in a partial loss of containment heat removal. Random failures complete the loss of containment heat removal, and a long-term loss of containment heat removal (TW type) sequence results. Primary injection into the RPV is available; the HPCS system is mainly used although other systems may be used for some part of the time. Containment pressurizes, RCIC isolates at 30 psig, the ADS valves reclose at 100 psig, and any low pressure injection fails. High pressure injection continues, and the containment pressurizes until structural failure of the containment to the reactor building occurs anywhere from 140 to 275 psig (mean value 191 psig). The severe environment created in the reactor building by the blowdown results in failure of any remaining injection systems, and core damage occurs with a failed containment.

FL2 is an internal flood-initiated by a service water pipe break on the ground floor of the reactor building. The flood fails all injection systems except diesel-driven fire water which is not used in time to prevent core damage. ADS and containment venting are available. This sequence is a short-term loss of all injection (TQUV type) sequence.

The mean fractional contributions to the total core damage frequency of the different accident classes are: seismic 1.5%, fire 23.8%, flood 11%, ATWS 0.5%, LOCAs 0.2%, transients 62.4%, and transients-induced LOCAs 0.8%.

S.4.2 Accident Progression Analysis Results

The majority of the accidents analyzed in this study will proceed to vessel failure. Although notable, the mean conditional probability of core damage arrest is fairly small, approximately 0.15. The probability of core damage arrest is driven by the recovery of AC power for the short-term station blackouts and the lack of available or recoverable injection systems for the other accidents. Given that core damage occurs, it is very likely that the containment's integrity will be compromised during the course of the accident by either containment failure or by containment venting. The mean probability that the containment will remain intact throughout the accident is only 0.12. Furthermore, it is fairly likely that the containment will fail early in the accident--the mean probability of early containment failure is 0.33. It is also likely that the operators will vent the containment during the accident--the mean probability of containment venting is 0.46. Given that core damage occurs, it is likely that the core debris released from the vessel will participate in core-concrete interactions. The mean probability of CCI, conditional on core damage, is 0.77. Thus, the potential exists for a large release late in the accident.

The events that result in containment failure before core damage are slow pressurization events that result from the accumulation of steam and noncondensibles during accidents in which containment heat removal is lost or inadequate (i.e., long-term loss of containment heat removal accidents, long-term station blackout accidents, and ATWS accidents). Events that result in containment failure around the time of vessel breach include fast pressurization of the containment from loads accompanying vessel breach (i.e., DCH, ex-vessel steam explosions, RPV blowdown), Alpha mode events, drywell failure induced by reactor pedestal failure, and cavity drain line isolation failure. Late in the accident, the events that result in containment failure include the slow pressurization of the containment from the steam and noncondensibles generated during CCI and failure of the reactor pedestal caused by concrete erosion during CCI.

S.4.3 Source Term Analysis Results

The source term results showed that the pressure of the RPV does not significantly affect the source term for a particular accident progression if vessel breach occurs. This is due to the differences in the fraction of a species released from the fuel before vessel breach being negated when the vessel breaches and the radionuclides in the vessel revolatilize (i.e., low release before vessel breach results in high release after vessel breach and vice versa).

The release path through which the radionuclides pass was determined to be important. The path that resulted in the highest release to the environment is through the wetwell above the water line. If the cavity floor has failed, radionuclides may leave the containment without being scrubbed by sprays or the suppression pool. The venting pathway is also through the wetwell above the water line. In most cases, containment failures in the drywell or drywell head were accompanied by successful operation of containment sprays which significantly reduced the amount of radionuclides being released. Also, for many of the cases for which containment failure in the drywell head occurred, core damage did not occur because severe environments were not created in the reactor building and the injection systems did not subsequently fail.

The analysis showed that late iodine revolatilization results in significant releases due to much of the iodine being scrubbed by the pools initially and revolatilized later when the removal mechanisms are not as effective. For this reason, the modeling of late iodine revolatilization may need to be investigated further.

The regression analysis shows that much of the uncertainty in the final risk results is due to the uncertainty in the source term parameters. Much of the uncertainty may be due to the fact that the distributions being used were elicited for cases with large uncertainties in the initial and boundary conditions. Some of this uncertainty might be eliminated by a more detailed case structure in the APET and LASSOR codes with new distributions for these more specific cases.

S.4.4 Partitioning Results

The initial partitioning process developed for the LaSalle analysis and used in NUREG-1150 was improved for the final analysis. The improved process uses parameters important in determining the consequences to help define the source term groups. The improved partitioning process resulted in a total of 97 non-zero partitions being defined. There were 20 high-g seismic, 23 low-g seismic, and 54 fire, flood, and internal partitions defined. Out of 75,680 source terms only 1,860, or less than 2.5%, of the source terms were not partitioned directly and had to be distributed to the nearest defined partition (significantly less than in the old process). All of the partitions defined were important either to early fatalities, latent cancers, or frequency (in the old process many partitions were defined that were unnecessary and did not contribute).

The result of the new partitioning process was that most of the partitions were defined using many, if not all, of the parameters selected and that each partition is important to the final result (i.e., only one empty partition was defined, this partition is necessary in case any source terms with zero release exist). The total number of partitions is less than that produced in the old partition process[3] used in NUREG-1150; but, the level of resolution is substantially increased as the source terms are grouped much more homogeneously. The risk distribution is more accurately modeled since the increased homogeneity of the partition groups allows a more accurate representation of the final consequences for each group in MACCS.

S.4.5 Risk Analysis Results

The offsite risk at LaSalle is relatively low, especially with respect to the proposed NRC safety goals. The mean individual early fatality risk, 1.1E-10/R-yr, is more than three orders of magnitude below the safety goal. Similarly, the mean individual latent cancer fatality risk, 8.5E-09/R-yr, is slightly more than two orders of magnitude below the safety goal. In fact, the entire distributions for these two risk measures lie below the safety goals. The mean values for early fatality risk and for latent cancer fatality risk are 1.2E-08/R-yr and 0.25/R-yr, respectively. The risk results are summarized in Table S-1.

For all of the consequence measures, the risk is dominated by the Fire PDS group and the Transient PDS group. These groups are also the dominant contributors to the core damage frequency. The LOCA and Transient-Induced LOCA PDS groups, on the other hand, are very minor contributors to the risk. The Seismic, ATWS, and Flood groups are intermediate contributors.

Table S-1
Distributions for Annual Risk at LaSalle for
Internal and External Initiators
(All Values per Reactor-Year)
(Population Doses in Person-Rem)

Risk Measure	5th %-ile	Median	Mean	95th %-ile
Core Damage	5.7E-6	2.7E-5	1.0E-4	3.2E-4
Early Fatalities	1.9E-13	1.5E-10	1.2E-8	2.5E-8
Latent Cancer Fatalities	7.3E-3	6.5E-2	2.5E-1	8.4E-1
Population Dose Within 50 Miles	2.7E+0	1.9E+1	6.6E+1	2.3E+2
Population Dose Entire Region	4.3E+1	3.9E+2	1.5E+3	5.2E+3
Individual Early Fatality Risk, 0 to 1 Mile	3.6E-15	2.5E-12	1.1E-10	3.0E-10
Individual Latent Cancer Fatality Risk, 0 to 10 Miles	3.8E-10	2.6E-09	8.5E-09	2.6E-08

For early fatality risk and individual risk of early fatality within one mile, the risk is dominated by accidents that progress to vessel breach and that involve loss of containment integrity. Vessel breach is important because it affects the magnitude of the source term. If the core damage process is arrested, there are only in-vessel releases which are likely to be scrubbed by the suppression pool. If, on the other hand, vessel breach occurs, there are both in-vessel releases and ex-vessel releases. Releases from core-concrete interaction, which is ex-vessel, can be a significant contributor to the total release. Loss of containment integrity is important because, if the containment remains intact, the release to the environment through normal leakage is extremely small and is a negligible contributor to risk. For this analysis, the timing of the release did not show up as a significant contributor to risk. This is because, for most of the dominant accident sequences, the population evacuates long before any release occurs and, therefore, variations in containment failure time did not influence the risk results. However, the fact that the containment design is such that most containment failures will occur after evacuation is clearly an important determiner of the final risk results.

Latent cancer fatalities depend primarily on the frequency of containment failure (or venting) and the total amount of radioactivity released. Thus, like early fatality risk, accidents that progress to vessel failure and involve a loss of containment integrity are the dominant contributors to the latent cancer fatality risk. Again, the timing of containment failure is not a particularly important direct contributor.

There are several factors that lead to these low values for risk. First, the core damage frequency for LaSalle, mean = 1.0E-04/R-yr., is relatively low. While this appears high when compared to the NUREG-1150 plants[1], it must be remembered that this is an integrated value and represents not only accidents from internal initiators but also accidents from fire, flood, and seismic initiators. In fact, if one adds up the mean core damage frequencies for the internal, fire, and seismic analyses at the Peach Bottom plant analyzed in NUREG-1150,[7] the mean core damage frequency is 1.4E-04/R-yr. with the LLNL seismic hazard curve and 6.6E-05/R-yr. with the EPRI seismic hazard curve. This is in the same range as the LaSalle core damage frequency.

Second, while the mean conditional probability that the containment's integrity will be compromised during the accident is fairly high, 0.88, the consequences of the dominant accidents are relatively low because: 1) about half of the containment failures are late failures, 2) there is a fast evacuation speed for the population around the plant, 3) many of the sequences which had containment failure at or before vessel breach are long-term sequences with containment failure still many hours into the accident, and 4) of the impact of other plant features that reduce the magnitude of the source term. Two of these other plant features that reduce the source term are the suppression pool and the reactor building. In the majority of the accidents analyzed, the in-vessel releases (i.e., early releases) are scrubbed by the suppression pool. In addition to the suppression pool, the reactor building surrounding the LaSalle containment also traps a portion of the radionuclides that escape the containment.

For all of the non-seismic source term groups that were generated during the partitioning process (see section 4.6) the population in the emergency planning zone, EPZ, began evacuation before the start of the release. Thus, the dose received by the evacuating population was generally small and no early fatalities resulted from this fraction of the population. Because of the rapid evacuation, the timing of containment failure was not important in the LaSalle analysis. Changes in the evacuation assumptions could change this conclusion.

S.5 References

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7. A. C. Payne Jr., R. J. Breeding, H. N. Jow, J. C. Helton, L. N. Smith, and A. W. Shiver, "Evaluation of Severe Accident Risks: Peach Bottom Unit 1," NUREG/CR-4551, Volume 4, Part 1, Revision 1, SAND86-1309, Sandia National Laboratories, Albuquerque, NM, December 1990.

1.0 INTRODUCTION

1.1 Objective and Scope

The objectives of the Level II/III portion of this probabilistic risk assessment (PRA) are:

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.

In this study, the term integrated risk assessment means the combination of the various constituent analyses (i.e., accident frequency analysis, accident progression analysis, source term analysis, and the consequence analysis) to form an expression for risk which includes contributions from all initiators.

The scope of this study 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 risk to the offsite population, and

4. Estimation of the combined uncertainties from the accident frequency analysis, accident progression analysis, source term analysis, and limited consequence uncertainties

1.2 General Description of Methodology

The general steps used to perform the LaSalle Level III PRA are described in this section. The framework of this method was developed in this program but was first applied in the NUREG-1150[1] study. Some of the details of this method are presented in the methodology volume of NUREG-4551.* Some of the methods that are described in NUREG-4551 have been improved for the LaSalle study. These improvements are described in the appropriate sections of this report. The general steps used in this PRA are:

1. Identify and determine the frequency of the accident sequences that lead to core damage and that are initiated from both internal and external events. This process is the Level I portion of the PRA and is described in NUREG/CR-4832.**

2. Group the cut sets that are associated with the accident sequences into plant damage states (PDSs) where each PDS presents unique initial and boundary conditions to the accident progression analysis.

3. Identify and determine the conditional probabilities of the many possible accident progression paths following core damage. A detailed accident progression event tree (APET) was used in this portion of the analysis and was evaluated using the EVNTRE code[2].

4. Group the thousands of paths that are propagated through the APET into bins that represent unique boundary conditions to the source term analysis.

5. Estimate source terms for all of the accident progression bins that have been passed from the accident progression analysis to the source term analysis. The source terms were estimated using a simple parametric code call LASSOR developed for this analysis.

6. Group the thousands of source terms that are generated in the source term analysis into source term groups based on similar

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* E.D. Gorham, et al., "Evaluation of Severe Accident Risks: Methodology for the Accident Progression, Source Term, Consequence, Risk Integration and Uncertainty Analyses," NUREG/CR-4551, Vol. 1, Rev. 1, SAND86-1309, Sandia National Laboratories, Albuquerque, NM, to be published.

** A. C. Payne Jr., T. T. Sype, D. W. Whitehead, and A. W. Shiver, "Analysis of the LaSalle Unit 2 Nuclear Power Plant: Risk Methods Integration and Evaluation Program (RMIEP), Volume 2: Integrated Quantification and Uncertainty Analysis," NUREG/CR-4832/2 of 10, SAND92-0537, Sandia National Laboratories, Albuquerque, NM, to be published.

characteristics of the source terms. A frequency-weighted mean source term was determined from each group. This process is referred to as partitioning and was performed with the PARTITION code.[3]

7. Calculate the consequences for each source term group based on the frequency-weighted mean source term for that group. The MACCS code[4] was used to calculate the consequences.

8. Combine the frequency of the accident progression bins with the consequences associated with the APB to form an expression for risk. This process was performed with the PRAMIS code[5] and its associated post processors.

9. Present the risk results in a format such that contributors to risk can be identified and the uncertainty in risk displayed.

1.3 General Description of the Plant

The LaSalle County Station Unit 2 nuclear power plant is located in Brookfield Township, LaSalle County, Illinois which is 55 miles southwest of Chicago. The plant is owned and operated by the Commonwealth Edison Company (CECo) and the architect/engineer was Sargent & Lundy (S&L). The LaSalle plant utilizes a Mark II type containment to house a General Electric BWR-5 reactor and is rated at 3293 MWt and 1078 MWe.

There are various injection systems that can be used to cool the core in abnormal situations and arrest the core damage process at LaSalle. Four high pressure and four low pressure injection systems are considered in this analysis. Detailed descriptions of the injection systems and their cooling and electrical support systems can be found in the Level I portion of this analysis in Volume 6 of NUREG/CR-4832.*

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). All of these systems can inject water into the primary system up to the relief valve setpoints (1146-1205 psig). The HPCS system has a motor-driven pump with its own dedicated diesel generator. This system draws water from either the condensate storage tank or the suppression pool. 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

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* A. C. Payne Jr., T. L. Zimmerman, N. L. Brisbin, N. L. Graves, J. C. LaChance, S. A. Eide, J. A. Lambright, and J. A. Perez, "Analysis of the LaSalle Unit 2 Nuclear Power Plant: Risk Methods Integration and Evaluation Program (RMIEP), Volume 6: System Descriptions and Fault Tree Definition," NUREG/CR-4832/6 of 10, SAND92-0537, Sandia National Laboratories, Albuquerque, NM, to be published.

condensate storage tank or the suppression pool back to the vessel. Thus, RCIC can not be used once the vessel fails and the steam supply to the turbine is lost. DC power is also required to control this system. For use in accident mitigation, the MFW system draws water from the condenser hotwell using a motor-driven pump. This pump requires offsite power (i.e., not emergency power). The CRD system can be used to inject water into the core through the control rod drives. 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 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 with train A powered by train A of the emergency power system (EPS) and trains B and C powered by train B of the EPS. The condensate system draws water from the condenser hotwell using four motor-driven pumps and pumps it through the feedwater line into the RPV. This system requires offsite power. The last resort injection system, used when all other systems have failed, is the diesel-driven firewater system (DFWS). This system can be manually connected to the MFW injection line to provide injection. The DFWS uses two diesel-driven pumps to draw water from the ultimate heat sink (a 2058 acre seismically qualified lake built for the plant). For these low pressure systems to provide coolant to the core, the RPV must be depressurized from normal operating pressure about 1055 psig to 150-500 psig depending on the system operating.

The Automatic Depressurization System (ADS) is designed to depressurize the reactor vessel to a pressure at which the various low pressure injection systems can inject coolant into the reactor vessel. The ADS consists of seven of the eighteen relief valves. Each ADS valve is capable of being manually opened. For the ADS system to be automatically initiated, a low pressure injection 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 separately discharges into the suppression pool via a tailpipe. The tailpipes have vacuum relief valves to prevent water from being drawn into the lines as the steam condenses. If these valves fail open, a release to the drywell can result. The ADS valves are located in the drywell and drywell pressures above approximately 85 psig will prevent opening the ADS valves as a result of the loss of differential pressure required to hold the valves open. The ADS also requires DC power. Therefore, the RPV can not be depressurized in sequences that involve failure of DC power or in accidents in which the containment presrure exceeds 85 psig.

Heat can be removed from the containment by the residual heat removal (RHR) system which uses trains A and B of LPCI system. The RHR system is a two train system with motor-operated valves and pumps. Both trains have a heat exchanger in parallel with a bypass line downstream of the pump. In either the suppression pool cooling (SPC) mode or the containment spray (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 the core standby cooling system (CSCS) providing cooling water from the ultimate heat sink on the shell side). In the CSS mode, water is sprayed into the drywell. 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 removed from the vessel via a recirculation loop, is passed through the RHR heat exchangers, and is then injected back into the vessel. The LPCI system can also be used to remove decay heat from the containment indirectly by taking water from the suppression pool, passing it through the heat exchangers, injecting the water into the vessel, and then having the water return to the suppression pool via a LOCA or the SRV discharge lines. All four modes of RHR (i.e., SPC, CSS, SDC, and LPCI with the heat exchangers) require AC power and are, therefore, unavailable during a station blackout.

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 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 of the primary containment 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 (see Figure 1.3-2). 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. 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 an over-and-under configuration. The drywell is located in the upper portion of the containment directly above

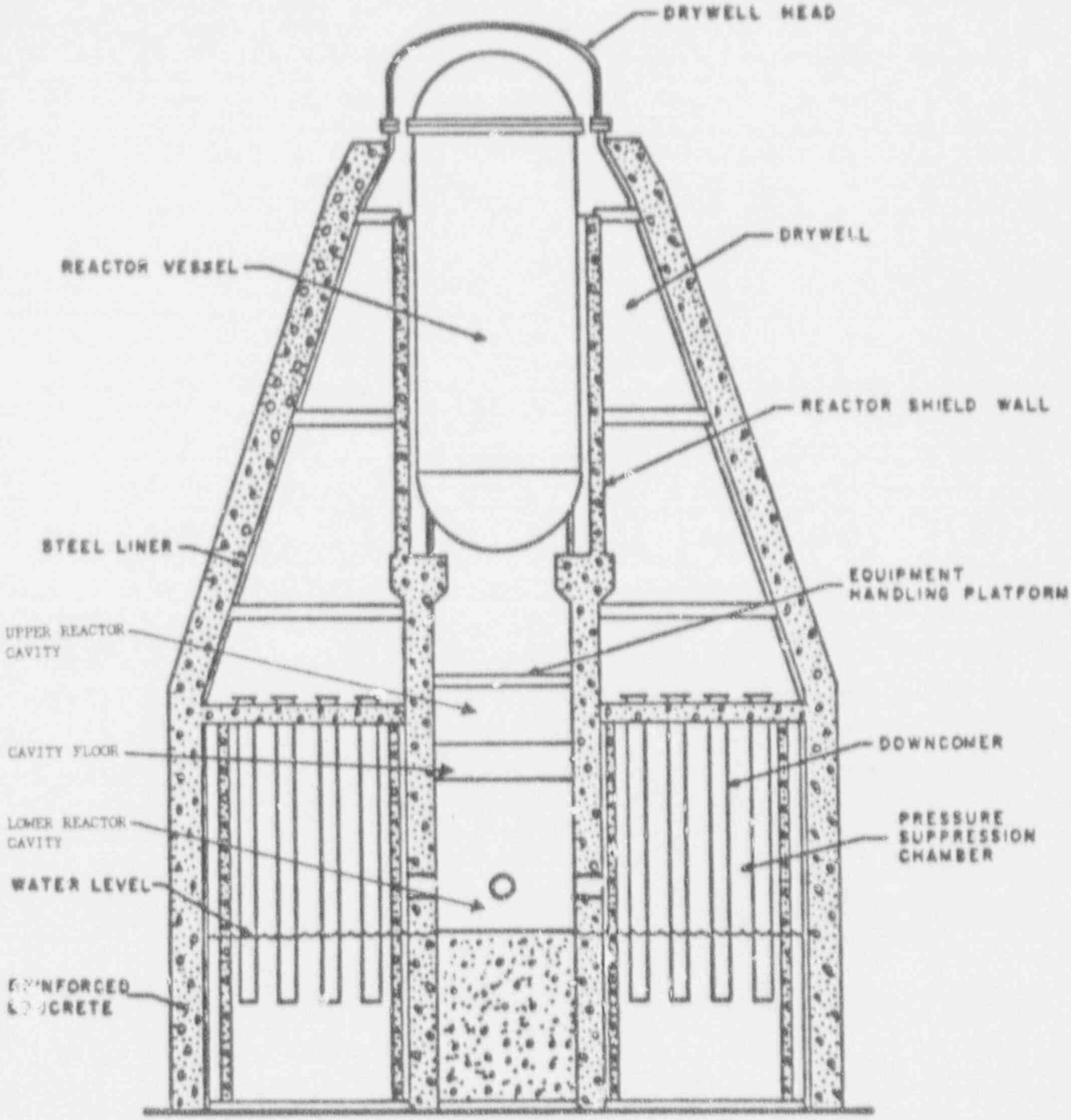

Figure 1.3-1. LaSalle Containment Schematic

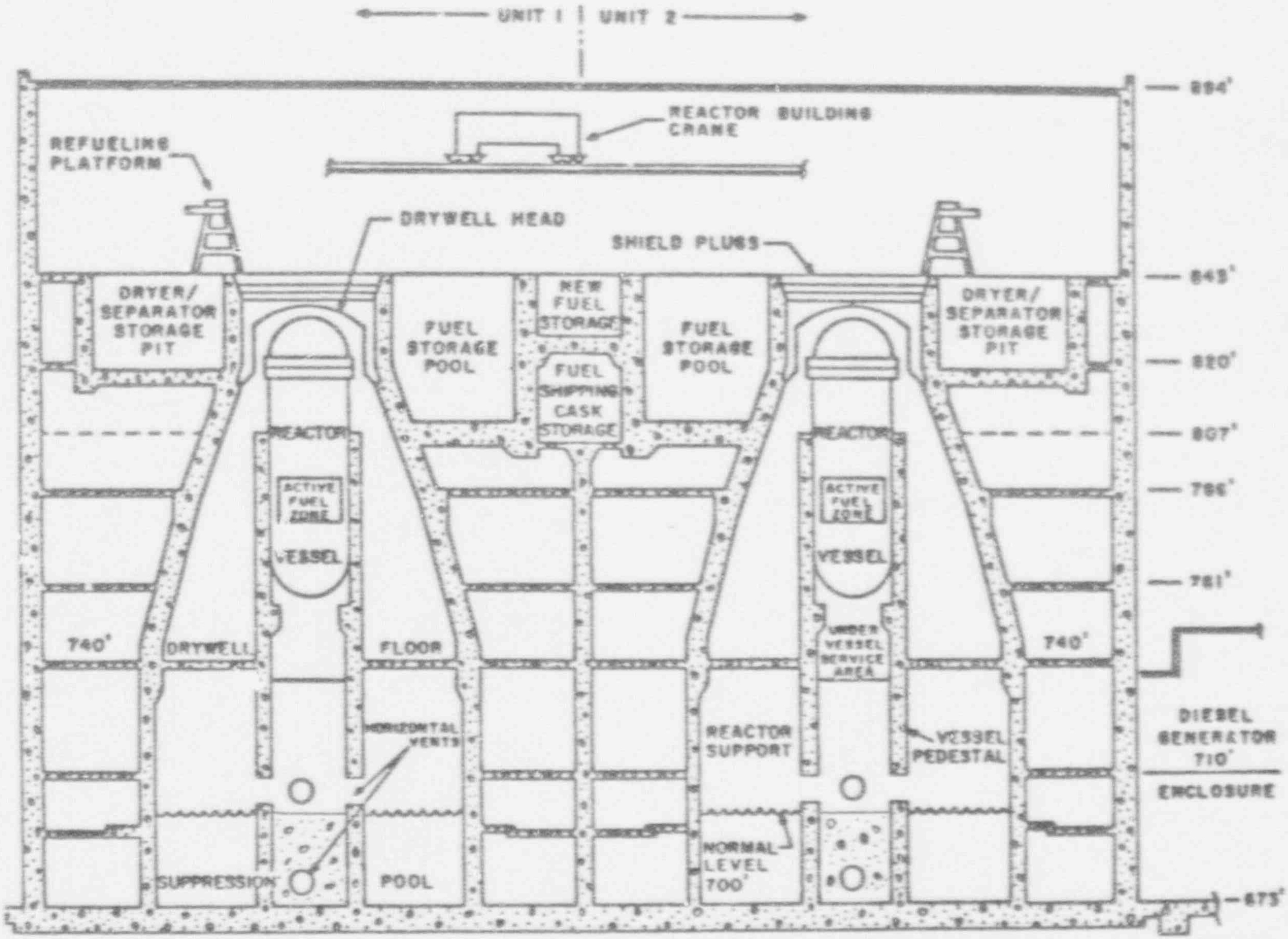

Figure 1.3-2. LaSalle Reactor Building Schematic

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 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 noncondensibles 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 tailpipes have check valves that connect to the drywell and if they fail open can result in release to the drywell instead 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 released at the time of vessel breach. There are scenarios, however, in which a portion the core debris may not remain in the cavity. For example, during a direct containment heating event a portion of the core debris that is ejected at high pressure may be dispersed into the drywell. In other scenarios it is possible that the cavity floor will fail and the core debris will be released into the lower pedestal which connects to the wetwell. However, because a large amount of the core debris can not enter the drywell, direct attack of the drywell wall by core debris is not an issue at LaSalle as it is for the Mark I containments. 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 two major implications. First, when core debris is released from the vessel at the time of vessel breach the potential exist for large fuel-coolant interactions (FCIs) to occur if the cavity is full of water. These FCIs can fail the containment directly from quasi-static pressure loads or can fail the RPV pedestal which can then lead to drywell failure (e.g., penetration failure). An FCI can also fail the pedestal drain pipes. In addition to FCI considerations, a large amount of water in the cavity can cool the core debris that is released from the reactor vessel and thus mitigate the releases associated with core-concrete interactions.

The LaSalle containment can be vented if 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 to the SGTS with a rubber boot. It is 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.

1.4 Structure of the Report

The main report is composed of seven chapters with this introduction being the first chapter. The formation of the plant damage states from the accident sequences developed in the Level I portion of this PRA is discussed in Chapter 2. The models and results of accident progression analysis are presented in Chapter 3. The source term model, the quantification of the source term issues, the source term results, and the partitioning process are presented in Chapter 4. The consequence analysis is described in Chapter 5 and the risk results are presented in Chapter 6. A summary of the results and insights from this study are presented in Chapter 7.

The codes, input files, and data used to perform this analysis are included in the appendices that form Volume 2 of this report. The MELCOR calculations used to support the accident progression and source term analysis are presented in Volume 3.

1.5 References

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2.0 PLANT DAMAGE STATE DEFINITION

2.1 Introduction

The output of the Level I analysis is a set of dominant accident sequences that result in core damage. These accident sequences are represented by Boolean expressions composed of "cut sets". Each cut set is composed of one or more terms which represent the failure of individual mechanical or electrical components in certain specific ways or represent the occurrence of certain physical processes. The term component should be understood loosely to represent any structure or device. A cut set, therefore, represents a specific set of failures that can result in core damage.

Each accident sequence actually represents a class of accidents, all having certain general similarities, which are being treated as one entity for the purpose of the Level I analysis. The Level I accident sequences are defined by an event tree in terms of the success or failure of various systems which can be used to mitigate a particular initiating event, any general operator actions that might impact the general development of the accident, and any phenomenological events produced during the accident that might induce failure of the mitigating systems. Each accident has a spectrum of semi-defined or undefined initial and boundary conditions such as: (1) the exact reactor power level, (2) the time in the refueling cycle, (3) the control rod position and history, (4) the specific distribution of radioactive decay products in the core, (5) the specific water level in the core, (6) the time in the expected forty year life of the plant at which the accident occurs, (7) the extent of partial system failures, and (8) the variation in the physical properties of components and physical structures due to plant age and weather. In most PRAs, all we know is that the accident occurs at power, sometime during the plant operating life, and that the mitigating systems do not successfully operate to prevent core damage. The above uncertainties affect both Level I considerations (e.g., system success criteria and accident sequence timing for determination of recovery actions and core damage times) and Level II and III considerations (e.g., the exact nature and timing of the core melt process, whether or not certain in-vessel and ex-vessel phenomena such as steam explosions or hydrogen burns might occur, the failure mode of the containment, and the effects of the release).

The phenomena, systems, and timing needing to be analyzed for the Level II/III analysis are somewhat different than those needed for the Level I analysis and the accident sequences need to be rearranged to efficiently continue the analysis. In order to perform the Level II/III analysis, the accident sequence cut sets are, therefore, rearranged into plant damage states (PDSs) where each plant damage state presents unique initial and boundary conditions to the accident progression analysis which constitutes the first part of the Level II analysis. It needs to be remembered that these plant damage states still represent classes of accidents and do not necessarily have the specificity in initial and boundary conditions that one might like. For example, all of the eight uncertain initial and boundary conditions described above still exist in the definition of the plant damage states, just as they existed in the definition of the accident sequences. This means that there is still a lot of uncertainty that can exist in the initial conditions. This uncertainty manifests itself in the probabilistic model as stochastic variation of various parameters that may be, but are not necessarily, constrained by the very general definitions of the plant damage states. This implies, even for a specific PDS, that the accident may progress in widely divergent directions for different ranges and combinations of initial conditions. This will influence the phenomena selected, the parameter distributions used, and the models that one uses to perform the accident progression analysis. One specific thermal-hydraulic calculation will not, in general, be sufficient to understand the accident progression. The probabilistic model that analyzes the evolution of the accidents from the point of core damage to the release of radioactive fission products to the environment is the accident progression event tree (APET). This model is described in Chapter 3 of this report.

For a given Level I accident sequence, each cut set in that sequence may imply certain additional information about the state of various systems not important to the Level I analysis but very important or important in a different way to the Level II analysis (e.g., status of containment spray systems and secondary containment systems, status of possible injection systems) and about the timing of events after core damage begins (e.g., time of core uncovery, time of vessel breach, availability of water in the reactor cavity, etc.). In fact, the differences between accident sequences may not be as significant as the differences between different sets of cut sets within the same sequence. The methodology described in this section describes the process used to effect the transformation of the accident sequences into plant damage states for the Level II analysis.

2.2 General Methodology

The process used to transform the accident sequences into plant damage states can be described in terms of the following series of steps:

1. Identify and characterize as specifically as possible the dominant accident sequences that need to be analyzed in the Level II analysis.

2. Using the accident sequence definitions from step 1, the plant specific information on the reactor core, reactor vessel, and primary and secondary containments make a preliminary determination of those plant characteristics that will be needed to define the initial state of the plant for input into the accident progression analysis.

3. Construct a preliminary series of questions at the beginning of the APET to define the plant damage states.

4. Examine each cut set from the Level I accident sequences and determine the answer to each of the PDS defining questions (Some of the questions used to define the PDS can be added to the end of the Level I event trees to help in grouping the cut sets for the level II analysis even though they are not explicitly needed to determine if core damage occurs). If characteristics show up that appear to be important for the Level II analysis, revise the list of questions.

5. Create a file showing all of the unique sets of answers to these questions for all sequences and all cut sets. Each unique set of answers is a possible plant damage state.

6. In order to reduce the number of plant damage states needing to be analyzed, sort the PDSs in various ways by rearranging the order of the questions to determine differences and similarities between the PDSs.

7. Determine which information is of primary importance to the characterization of the accident evolution, which is of secondary importance, and which is not important to the accident evolution but must be carried along from other analyses to properly represent all aspects of the analysis (i.e., the fact that the sequence originates from a seismic event may not be important to the APET if all the seismic effects are included in the PDS definition; however, one may want to separate out the seismic results later).

8. Select the final set of questions to be used to define the PDSs and any sub-PDSs considered necessary. (A sub-PDS is a grouping of cut sets with some characteristic that will result in a different accident progression but the difference is not significant enough to warrant defining a new PDS.)

9. Edit the accident sequence cut set files, rearranging the cut sets into new files for each PDS and sub-PDS.

10. Perform a full Level I uncertainty calculation on each sub-PDS and PDS using the TEMAC[1] code.

11. From the uncertainty calculations, identify those events whose uncertainty contributes significantly to the uncertainty of each PDS or sub-PDS and at what level the cut sets can be truncated without significantly affecting either the estimate of the PDS or sub-PDS frequency or uncertainty distribution.

12. Construct a list of those events significant to uncertainty and include them in the Level II/III Latin Hypercube Sample[2] (LHS).

13. Truncate the cut sets for each PDS or sub-PDS as appropriate, construct a reduced Level I LHS sample, and re-run TEMAC to make sure that the distributions are still roughly the same.

14. Merge the reduced Level 1 LHS sample with the Level II/III sample, create the final LHS sample, and run TEMAC to calculate the total Level I core damage frequency, the conditional probability of each PDS, and conditional probabilities of each sub-PDS used in the final analysis.

In the sections that follow, we will describe some of these steps in more detail and give examples of the results.

2.3 Construction of the Plant Damage State Definitions

The dominant accident sequences are described in detail in Volume 2 of the LaSalle Level I report.* These sequences fall into the following general categories or groups: Transients, Transient-Induced LOCAs, LOCAs, ATWS, Seismic, Internal Fire, and Internal Flood. These groups will be tracked throughout the analysis and the results of the accident progression, source term, consequence, and risk analyses will be presented in terms of these groups. The results could be tracked in terms of the individual PDSs but there are too many to make describing every result in terms of each PDS tractable.

The accident progression analysis for the LaSalle plant follows the general method used in the NUREG-1150** analysis but with some significant differences. The basic purpose of the LaSalle PRA is to develop methods for performing integrated uncertainty analysis. To this end, the EVNTRE code,[3] which was initially developed in the Severe Accident Risk Reduction Program (SARRP), was extensively rewritten for the LaSalle PRA in order to be able to perform these detailed analyses. Because of the delays in the LaSalle analysis resulting from the priority of the NUREG-1150 analysis, the EVNTRE code in its final form was first used in the NUREG-1150 analysis. However, NUREG-1150 did not perform the accident progression analysis in an integrated fashion. First, the internal, seismic, and fire analyses were each performed individually and, second, each PDS from each analysis was passed through the APET individually.

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* A. C. Payne Jr., T. T. Sype, D. W. Whitehead, and A. W. Shiver, "Analysis of the LaSalle Unit 2 Nuclear Power Plant: Risk Methods Integration and Evaluation Program (RMIEP) Volume 2: Integrated Quantification and Uncertainty Analysis," NUREG/CR-4832/2 of 10, SAND 92-0537, Sandia National Laboratories, Albuquerque, NM, to be published.

** E. D. Gorham, J. C. Helton, R. J. Breeding, S. C. Hora, W. B. Murffin, J. L. Sprung and F. T. Harper, "Evaluation of Severe Accident Risks: Methodology," NUREG/CR-4551, Volume 1, Revision 1, SAND86-1309, Sandia National Laboratories, Albuquerque, NM, to be published.

In the LaSalle program, a procedure for performing this analysis in an integrated fashion was subsequently developed. This procedure, using the same EVNTRE code as in NUREG-1150, was applied in a limited fashion in the N-Reactor PRA.[4] In order to fully implement the procedure for the LaSalle PRA, a modification to the EVNTRE code was made which allows split fractions, which represent the conditional probabilities of the various question branches, to be input directly into the APET for questions with any number of branches. Previously this was possible only for questions with two branches. With this modification, the PDSs conditional probabilities of occurrence calculated by TEMAC can be directly input into the APET.

With the above modification, the APET can be run efficiently for the integrated problem. After the determination of the final set of PDS to be used in the analysis, the characteristics of each PDS can be programmed into the initial questions in the APET that define the PDSs using the question case structure. For each LHS sample member, the total core damage frequency is calculated, the conditional probabilities of each of the 30 PDSs and any additional split fractions for sub-PDSs are calculated for that sample member, and the APET evaluates all of the PDSs simultaneously for the sample member. This allows consistent truncation of the resulting accident progression bins (APBs) at a uniform level and results in fewer APBs that need to be analyzed further. (This may not appear to be so since there are roughly 75,000 APBs in the LaSalle analysis and roughly the same number in the Peach Bottom NUREG-1150 analysis;[5] but, given the level of detail in the LaSalle analysis, we would have expected many more).

Table 2.3-1 contains a list of the final set of questions used in the APET to define the plant damage states. These are the first twenty-two questions of the accident progression event tree. These questions define those characteristics of the plant that were considered to be important in defining the initial conditions for the accident progression analysis. Each question and the various possible answers are discussed. A detailed listing of the APET is given in Appendix B. In these questions certain descriptors are used to classify the state of the systems that are influenced by the Level I analysis:

1. failed - the system will not work because of the failure of some system or subsystem component or because of some phenomena occurring in the sequence definition, any possible recovery credit using alternate system/components was given in the Level I analysis and no further actions are possible,

2. recoverable - system has failed due to loss of offsite power or simultaneous loss of offsite and onsite power; if AC power can be restored, the system may be used given other conditions allow this (recovery of AC power is tracked in the APET),

3. available - system is not being used at the present time either because it has not been called for or because conditions do not allow its use, the APET will track the appropriate conditions and

Table 2.3-1
PDS Definition Questions From the LaSalle APET

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1 What is the Plant Damage State (PDS)?

30	EQ1	EQ2	FI1	FI2	FI3	FI4
	FI5	FI6	FL1	FL2	IA1	IA2
	IL1	IT1	IT2	IT3	IT4	IT5
	IT6	IT7	IT8	IT9	IT10	IT11
	ITL1	ITL2	ITL3	ITL4	ITL5	ITL6

2 Is there a loss of offsite power?
3 E1-sLOP E1-LOP E1nLOP

3 Is there a Station Blackout (Loss of all AC)?
2 SB nSB

4 Is dc power available?
3 E1fDC E1rDC E1-DC

5 Does a S/RV stick open early?
2 E1-SORV E1nSORV

6 Does the HPCS system fail to inject?
3 E1fHPC E1rHPC E1-HPC

7 Does the RCIC system fail to inject?
3 E1fRCIC E1rRCIC E1aRCIC

8 What is the initial status of the CRD hydraulic system?
3 E1fCRD E1rCRD E1-CRD

9 What is the initial status of the main feedwater system?
3 E1fMFW E1rMFW E1aMFW

10 What is the initial status of RPV depressurization?
3 E1fADS E1rADS E1aADS

11 What is the initial status of the low-pressure ECC systems?
3 E1fLPC E1rLPC E1aLPC

12 What is the initial status of the condensate system?
3 E1fCOND E1rCOND E1aCOND

13 What is the initial status of containment heat removal?
3 E1fRHR E1rRHR E1-RHR

14 What is the initial status of containment spray?
4 E1fCSS E1rCSS E1aCSS E1-CSS

Table 2.3-1 (Concluded)
PDS Definition Questions From the LaSalle APET

15 Does the containment fail before core damage?
2 ElnCF El-CF

16 Is the containment vented before core degradation?
4 ElfVNT ElrVNT ElaVNT El-VNT

17 Level of pre-existing leakage or isolation failure?
5 ElnCL El-CL2 El-CL3 El-CPr2 El-CPr3

18 Location of pre-existing leakage or isolation failure?
4 El-CInt El-CLD El-CLDH El-CLW

19 What is the level of preexisting suppression pool bypass?
5 ElnSPB El-SPB2 El-SPB3 El-SPD ElpSPD

20 For TC does SLC fail to inject?
3 ElfSLC ElrSLC ElaSLC

21 When does core damage occur?
6 CD1 CD8 CD10 CDVNT CDSRV CDCF

22 What type of sequence is this (summary of plant damage)?
7 EQ Fire Fld ATWS LOCA Tran TranL

allow the use of the system when needed if conditions change enough to allow operation, parts of the system could be operating but the overall system function is not being performed, and

4. working - system is performing its function.

Question 1 - What is the Plant Damage State?

Question 1 defines the the plant damage states to be analyzed and their conditional probabilities. There are 30 PDSs included in the integrated analysis. There are 2 seismic PDSs, 6 fire PDSs, 2 flood PDSs, 2 ATWS PDSs, 1 LOCA PDS, 11 transient PDSs, and 6 transient-induced LOCA PDSs. Since the conditional probabilities for each PDS are input directly into the question for each sample member, all PDSs are evaluated simultaneously.

Question 2 - Is there a loss of offsite power?

Question 2 defines the state of offsite power. There are three branches to this question: (1) loss of all offsite power occurred from a seismic event, (2) loss of offsite power occurred as a random initiator or randomly after some other initiator, and (3) no loss of offsite power has occurred. In the first case, no recovery of offsite power is allowed in the APET evaluation of the accident progression because of the severity of the damage. This arises because the methodology for treating recovery for external events is not as developed as for the internal event analysis and; therefore, conservative assumptions are made. Some fire PDSs were assigned to this branch if the fire damage was so severe that recovery of AC power was not thought to be likely. For the second case, the APET will track the accident progression timing and calculate the appropriate conditional probability of recovery of AC power for the time interval in question using the same data and method as used in the Level 1 analysis.[6] In the third case, no loss of offsite power occurs and at least one train of AC power is available.

Question 3 - Is there a Station Blackout (Loss of all AC)?

Question 3 defines whether or not a total loss of offsite and onsite AC power has occurred, with the exception of the HPCS dedicated diesel. There are two branches to this question: (1) Station Blackout has occurred or (2) Station Blackout has not occurred. If a complete loss of offsite power has occurred and train A and B of onsite power have failed (i.e., DG "0" and DG "A", respectively) then a station blackout is said to have occurred. Note that DG "B" (train C) may still be available. If a partial loss of offsite power or at least train A or B of onsite power is available then station blackout has not occurred.

Question 4 - Is DC power available?

Question 4 defines the availability state of DC power. There are three branches: (1) DC power has failed and can not be recovered, (2) DC power has failed but is recoverable, and (3) DC power is available. The first case represents hardware failure of sufficient equipment that all DC power is failed. Alternative equipment is not available and repair of the failed equipment is not allowed. In the second case, DC power has failed because of battery depletion after a loss of AC power. If AC power can be restored, DC power can be recovered through the inverters. In the third case, DC power is available throughout the accident.

Question 5 - Does an S/RV stick open early?

Question 5 defines the status of the safety relief valves. There are two branches to this question: 1) at least one SRV has stuck open and 2) no SRV has stuck open. This is part of the determination of reactor vessel pressure during the core damage process and before vessel breach occurs. If an SRV has stuck open then the vessel will depressurize before vessel breach. If an SRV has not stuck open then the vessel will be at high pressure unless some other mechanism for vessel depressurization, such as ADS, has occurred.

Question 6 - Does the HPCS system fail to inject?

Question 6 defines the status of the high pressure core spray system (HPCS). For this analysis, all of the core vulnerable sequences were resolved in the Level I analysis. This means that any system failures occurring as a result of containment pressurization, containment failure, and severe environments in the reactor building that would result in core damage have already been evaluated. This implies that in order to get core damage, HPCS can not be working. There are, therefore, only three branches to consider: (1) HPCS is mechanically failed, (2) HPCS is failed from loss of AC power but can be recovered if offsite power is restored, and (3) HPCS is available but the operator has not used it (i.e., for some reason HPCS did not automatically start or it was turned off). The APET will track AC power and allow use of the system if conditions permit. If HPCS works, it can prevent core damage (i.e., about 1000 gpm at normal operating pressure). If restarted after core damage, it may result in core damage arrest and no vessel breach. Water on the debris may affect the source term and the occurrence or magnitude of some types of phenomena (e.g., cavity failure, core-concrete interactions, ex-vessel steam explosions, long-term pedestal failure, etc.).

Question 7 - Does the RCIC system fail to inject?

Question 7 defines the status of the reactor core isolation cooling (RCIC) system. RCIC is a steam-driven DC dependent system injecting at high RPV pressure similar to HPCS but with a lower flow rate (i.e., about 600 gpm at normal operating pressure). As with HPCS continued success of RCIC has already been evaluated in the Level I analysis and some failure had to occur to get to this point. There are, therefore, only three branches that need to be considered: (1) RCIC is mechanically failed or RPV pressure is too low to drive the turbine, (2) RCIC is failed from loss of DC power but can be recovered on restoration of AC power, and (3) RCIC is available but the operator has not used it. The APET will track AC power and allow use of the system if conditions permit.

Question 8 - What is the initial status of the CRD hydraulic system?

Question 8 defines the status of the control rod drive (CRD) system. CRD is also a high pressure injection system and is dependent on offsite AC. Its capacity (about 200 gpm at maximum) is not enough to prevent core damage if all other injection is lost at the start of the accident; but, after about 6 hours, it is sufficient. As with HPCS and RCIC, the success of CRD in preventing core damage was evaluated in the Level I analysis. There are, therefore, only three branches: (1) CRD is mechanically failed, (2) CRD is failed from loss of AC power but can be recovered on restoration of AC power, and (3) CRD is available but the operator has not used it. The APET will track AC power and allow use of the system if conditions permit. Normal operation of the CRD without enhancement by the operator (i.e., starting of the second pump and/or realigning of the valves for maximum flow) was not considered significant for this analysis.

Question 9 - What is the initial status of the main feedwater system?

Question 9 defines the state of the main feedwater (MFW) system. At LaSalle, the MFW system has a motor-driven pump powered by offsite AC power. A full system fault tree was developed in the Level I analysis and credit was taken for the use of this system in the accident mitigation. This is also a high pressure injection system and as with the others only three branches exist. (1) MFW is mechanically failed, (2) MFW is failed from loss of AC power but can be recovered on restoration of AC power, and (3) MFW is available but the operator has not used it. The APET will track AC power and allow use of the system if conditions permit.

Question 10 - What is the initial status of RPV depressurization?

Question 10 is part of the determination of the state of RPV pressurization and asks about the status of the automatic depressurization system (ADS). This determines whether or not low pressure injection systems can be used to mitigate the accident, whether certain in-vessel phenomena occur, and the magnitude of some phenomena (e.g., core damage arrest, in-vessel steam explosions, recriticality, and the nature of the vessel breach). There are three branches: (1) ADS is mechanically failed, (2) ADS is unavailable because of the loss of DC power but can be recovered if AC power is restored, (3) ADS is available and can be used. The APET will track AC power and allow use of the system if conditions permit. As with the high pressure injection systems, the use of ADS to decrease reactor pressure and allow the use of low pressure injection to prevent core damage has already been taken into account in the Level I analysis. Any sequence with continued low pressure injection does not result in core damage. All sequences coming from the Level I analysis, therefore, have not had ADS for some reason or ADS occurred and the reactor vessel repressurized when ADS reclosed on high containment pressure.

Question 11 - What is the initial status of the low-pressure ECC system?

Question 11 defines the status of the low pressure coolant injection (LPCI) system and the low pressure core spray (LPCS) system. These systems are treated together because of their common support systems and interacting control and actuation systems. These are high flow, low pressure injection (LPI) systems and require that the RPV be depressurized in order to inject. Three branches exist: (1) LPI is mechanically failed, (2) LPI is failed from loss of AC power but can be recovered on restoration of AC power, and (3) LPI is available but the operator has not used it. The APET will track AC power and allow use of the system if conditions permit. As with the high pressure injection systems, the Level I analysis has already evaluated the core vulnerable sequences with low pressure injection and low pressure injection can not be working at this time.

Question 12 - What is the initial status of the condensate system?

Question 12 defines the status of the condensate system (CDS). CDS is an intermediate pressure injection system and for this analysis is treated as a low pressure injection system. As with the other low pressure injection systems only three branches exist: (1) CDS is mechanically failed, (2) CDS is failed from loss of AC power but can be recovered on restoration of AC power, and (3) CDS is available but the operator has not used it. The APET will track AC power and allow use of the system if conditions permit.

Question 13 - What is the initial status of containment heat removal?

Question 13 defines the status of the residual heat removal (RHR) system. The RHR system performs the function of containment heat removal. Three branches exist: (1) RHR is mechanically failed, (2) RHR is failed from loss of AC power but can be recovered on restoration of AC power, and (3) RHR is working. The APET will track AC power and allow use of the system if conditions permit. The RHR system can operate in any of four modes shutdown cooling system (SCS), suppression pool cooling (SPC), containment sprays (CSS) and LPCI with flow through the appropriate heat exchanger. Since the spray mode has unique decontamination mechanisms, its operation is defined separately in the next question. Operation of RHR is evaluated in the Level I analysis because of the evaluation of core vulnerable sequences. For sequences that were not core vulnerable sequences and resulted in core damage, no cases of operator failure to initiate appeared in the dominant cut sets.

Question 14 - What is the initial status of containment spray?

Question 14 defines the status of the containment sprays independent of the operation of the heat exchangers. The combination of operating heat exchangers and CSS will cool the containment and will directly affect retention of radioactive fission products in the drywell atmosphere. If only CSS operates, the retention of fission products will be affected, but differently than with cooling, and the containment will not be cooled. Since RHR can operate without CSS, there are four cases for this question: (1) CSS is mechanically failed, (2) CSS is failed from loss of AC power but can be recovered if AC power is recovered, (3) CSS is available but not being used, and (4) CSS is working. The APET will track AC power and allow use of the system if conditions permit.

Question 15 - Does the containment fail before core damage?

Question 15 defines the time of containment failure relative to the time of core damage. For certain Level I accident sequences, containment failure had to have occurred in order for core damage to occur. The timing or core damage relative to the occurrence of containment failure was, therefore, determined in the Level I analysis. There are two branches: (1) the containment fails after core damage and (2) the containment fails before core damage. The results of this question are used later in the event tree as the APET tracks the containment failure mode. This impacts the severe environment failure of various systems that have components in the reactor building.

Question 16 - Is the containment vented before core damage?

Question 16 defines the status of containment venting. There are four branches: (1) the containment venting system is mechanically failed, (2) the venting system is failed because of the loss of AC power but can be recovered if AC is recovered, (3) the venting system is available but has not been used, and (4) the containment has been vented. Venting affects the operability of systems in the reactor building and the ADS system. The APET will track AC power and allow use of the system if conditions permit.

Question 17 - Level of pre-existing leakage or isolation failure?

Question 17 defines the status of the containment at the start of the accident. There are five branches: (1) No pre-existing containment leakage or isolation failure, (2) pre-existing leak, (3) pre-existing rupture, (4) isolation failure resulting in a leak, and (5) isolation failure resulting in a rupture. For LaSalle with an inert containment and isolation system, the probability of a pre-existing failure large enough to be classified as a leak or a rupture is negligible (i.e., technical specification leakage is not classified as a leak). This question was included to allow sensitivity analysis at a later date. For this analysis, a leak is defined as any containment failure that results in greater than two hours to depressurize the containment and a rupture as any failure that depressurizes the containment in less than two hours. This definition comes from the resolution of core vulnerable sequences in the Level I analysis. Two hours is the minimum time within which the containment can be depressurized to the point where low pressure injection systems can inject into the core and still prevent core damage for long-term sequences. This is conservative and is determined by the decay heat level and water level at the time of loss of injection.

Question 18 - Location of pre-existing leakage or isolation failure?

Question 18 defines the location of the leakage in question 17. There are four branches: (1) the containment is intact, (2) failure is in the drywell and leads to the reactor building, (3) failure is in the drywell head and leads to the refueling floor, and (4) failure is in the wetwell and leads to the reactor building. For this analysis, the containment is always intact; but, the question was included to allow sensitivity analysis at a later date.

Question 19 - What is the level of pre-existing suppression pool bypass?

Question 19 defines the initial level of suppression pool bypass. There are five branches: (1) no suppression pool bypass, (2) there is a leak above the water line, (3) there is a rupture above the water line, (4) the suppression pool is completely drained, and (5) the suppression pool is partially drained. There is no initial suppression pool bypass for any of the sequences at LaSalle. The question was included to allow sensitivity analysis at a later date.

Question 20 - For TC does SLC fail to inject?

Question 20 defines the status of the standby liquid control (SBLC or SLC) system for ATWS scenarios (TC). There are three branches: (1) the SBLC system is mechanically failed, (2) the SBLC system is failed due to the loss of AC and can be recovered if AC is recovered, and (3) the SBLC system is available but not being used. The APET will track AC power and allow use of the system if conditions permit. Credit for the early use of SBLC to switch the sequence to an ordinary transient has already been taken in the Level I analysis, only long-term use of SBLC after core damage has occurred is given here.

Question 21 - When does core damage occur?

Question 21 defines the time at which core damage occurs. There are six branches: (1) at about one hour, (2) at about eight hours, (3) at about ten hours, (4) as a result of containment venting, (5) as a result of SRV reclosure on high containment pressure and the loss of low pressure injection, and (6) after containment structural failure. Cases 4-6 imply core damage at about the time of occurrence of specific events in the accident progression. For cases 1-3, core damage is not explicitly associated with any other specific event in the accident progression.

Question 22 - What type of sequence is this (summary of plant damage)?

Question 22 defines the general type of sequence that the particular plant damage state fails into. There are seven branches: (1) seismic, (2) fire, 3) flood, (4) ATWS, (5) LOCA, (6) transient, and (7) transient-induced LOCA. This classification is used throughout the analysis to simplify the discussions.

2.4 Initial PDSs From Accident Sequence Cut Sets

Once the initial set of questions that define the PDSs have been determined, the accident sequence cut sets have to be examined to evaluate the specific PDSs that will appear in the analysis. For the LaSalle PRA, the final accident sequences are reported in Volume 2 of the LaSalle Level I report and Table 2.4-1 shows the accident sequences in decreasing order of frequency. The table has some statistics of the individual sequences' frequency distributions (5th percentile, median, mean, and 95th

Table 2.4-1
LASALLE FINAL SEQUENCE CORE DAMAGE STATISTICS

SEQUENCE	5%	MEDIAN	MEAN	95%	PE	FRAC OF TOT	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-06	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	1.750[illegible]-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	1.01[illegible]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-08	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 2.4-1 (Continued)
LASALLE FINAL SEQUENCE CORE DAMAGE STATISTICS

SEQUENCE	5%	MEDIAN	MEAN	95%	PE	FRAC OF TOT	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	3.5200E-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	2.5500E-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	1.9989E-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	1.7768E-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 2.4-1 (Continued)
LASALLE FINAL SEQUENCE CORE DAMAGE STATISTICS

SEQUENCE	5%	MEDIAN	MEAN	95%	PE	FRAC OF TOT	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	8.5500E-10	1.2833E-05	9.9991E-01
T84	0.0000E+00	0.0000E+00	1.0100E-09	7.1100E-10	1.4200E-10	1.2463E-05	9.9993E-01
TL16	0.0000E+00	0.0000E+00	9.0400E-10	4.7300E-09	3.6200E-10	1.1155E-05	9.9994E-01
TL1_01.L1	2.3700E-14	2.0600E-11	8.6900E-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	1.2800E-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	4.2200E-11	4.9357E-06	9.9997E-01
TL1_01.L3	2.5300E-14	7.3000E-12	3.7900E-10	6.5400E-10	1.9000E-10	4.6765E-06	9.9997E-01
TL1_01.L4	1.2800E-14	3.2600E-12	2.5800E-10	3.6100E-10	1.1900E-10	3.1835E-06	9.9998E-01
TL1_03.L2	2.6400E-16	2.6200E-13	2.5500E-10	1.4100E-10	2.7500E-11	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	4.2400E-11	1.1673E-06	9.9999E-01
TL1_01.L6	1.7000E-15	5.6700E-13	8.2400E-11	1.1700E-10	5.2900E-11	1.0167E-06	9.9999E-01
A55	0.0000E+00	0.0000E+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 2.4-1 (Concluded)
LASALLE FINAL SEQUENCE CORE DAMAGE STATISTICS

SEQUENCE	5%	MEDIAN	MEAN	95%	PE	FRAC OF TOT	CUM FRAC
T30	2.0500E-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-06	1.3717E-05	8.1043E-05	2.2330E-04	6.8769E-05		
INTEGRATED**	5.3400E-06	2.9200E-05	1.0100E-04	2.9300E-04	6.7700E-05		

* Sequences above 1.0E-08/yr. mean frequency were considered in the Level II/III analysis.
** TOTAL is the algebraic sum of all the rows, INTEGRATED is the result of the combined calculation.

percentile), a point estimate (PE), the fraction of the total core damage frequency contributed by this individual sequence (FRAC OF TOT), and the cumulative fraction of the total core damage frequency (CUM FRAC). Detailed discussions of the characteristics of each sequence are given in Volume 2 of the LaSalle I report. For the Level II/III analysis, it was decided to analyze all sequences with mean frequencies in the Level I analysis greater than or equal to 1.0E-8/yr. This included the top 50 sequences out of the 113 sequences that were evaluated in the Level I analysis.

The following procedure was used to group the cut sets into PDSs:

1. For each accident sequence, a table was constructed listing the initial PDS defining questions across the top and listing each cut set number down the side.

2. Computer printouts of the sequence cut sets from the accident sequence TEMAC calculations were obtained and each cut set was examined. The answers to each of the defining questions were entered into the table.

3. A final table was constructed consisting of the unique PDSs for each accident sequence.

The results of following this procedure are shown in Tables 2.4-2 thru 2.4-8 for the seismic, flood, LOCA, ATWS, transient-induced LOCA, fire, and transient accident sequences, respectively. Each table presents a list of all the unique sets of responses obtained for the PDS defining questions. At this point the PDSs are not named or numbered, this was only done after the final set was determined. The questions represent the initial set of questions used to define the PDSs and have been ordered to try and facilitate the determination of groups of the PDSs (their relationship to the final set used in the APET is described below). In the next step, the questions are reduced to the final set of questions described in Section 2.3. An individual sequence may have cut sets in several PDSs since each cut set in the sequence can imply different answers to the questions. For example, some cut sets may have AC power failure while others may not, so systems may either be failed or recoverable depending on the specific cut set.

The tables have the following column headings: SEQ - only one of the many sequences which might have had this PDS in it is shown (this is for checking purposes), ANAL - from which subanalysis is this PDS, LOSP - the status of offsite power, SB - if a station blackout has occurred, L/R - the status of the containment (leak or rupture), VENT - the status of venting, CDT - the core damage time, SRV - the status of the SRVs, ADS - the status of the ADS system, RHR - the status of the heat removal heat exchangers in the RHR system, CSS, SPC, SCS - the status of the RHR modes without asking if the heat exchangers are working, MFW, CDS, HPCS, RCIC, LPCS, LPCI, CRD, DDFW - the status of the various injection systems , PCS - the status of the power conversion system, and DC, A, B - the status of DC power and the individual AC power trains.

Table 2.4-2
Plant Damage States Appearing in LaSalle Level I Dominant Sequences: Seismic Sequences

SEQ	ANAL	LOSP	SB	L/R	VENT	CDT	SRV	ADS	RHR	CSS	MFW	CDS	HPCS	RCIC	LPCS	LPCI	SPC	CRD	DDFW	PCS	DC	SBLC	A	B	SCS
S1	EQ	Y	Y	N	R	8	N	R	R	R	R	R	R	DF	R	R	R	R	A	R	R	F	F	F	R
S1	EQ	Y	Y	N	R	10,16	N	R	R	R	R	R	DR	DF	R	R	R	R	A	R	R	F	F	F	R
S1	EQ	Y	Y	N	R	10,16	N	R	DR	R	R	R	DR	DF	DR	DR	DR	R	A	R	R	F	DF	DF	R
S1	EQ	Y	Y	N	R	10,16	N	R	DR	R	R	R	R	DF	DR	DR	DR	R	A	R	R	F	DF	DF	R
S1	EQ	Y	Y	N	R	10,16	N	R	DR	R	R	R	DR	DF	DR	DR	DR	R	A	R	R	F	DF	F	R
S1	EQ	Y	Y	N	R	10,16	N	R	DR	R	R	R	R	DF	DR	DR	DR	R	A	R	R	F	DF	F	R
S1	EQ	Y	Y	N	R	10	N	A	DR	R	R	R	R	DF	R	DR	DR	R	A	R	A	F	F	DF	R
S1	EQ	Y	Y	N	R	10	N	A	DR	R	R	R	DR	DF	R	DR	DR	R	A	R	A	F	F	DF	R
S2	EQ	Y	Y	N	R	10	N	A	DR	R	R	R	R	F	DR	DR	DR	R	A	R	A	F	DF	DF	R
S2	EQ	Y	Y	N	R	10	N	A	DR	R	R	R	R	F	R	DR	DR	R	A	R	A	F	F	DF	R
S2	EQ	Y	Y	N	R	10	N	A	DR	R	R	R	DR	F	R	DR	DR	R	A	R	A	F	F	DF	R
S2	EQ	Y	Y	N	R	10	N	A	DR	R	R	R	DR	F	DR	DR	DR	R	A	R	A	F	DF	DF	R

S1 = TLOSP-01.LXX and S2 = TLOSP-03.LXX in SEQ column. The difference in the seismic sequences represented by the hazard level affects only the evacuation assumptions for the LaSalle analysis, since no seismic-induced failures except loss of offsite power were important. A -a or -b was used to represent high vs low g, respectively.

EQ = Earthquake in ANAL column.

Y = Yes, N = No or None, F = failed, R = recoverable, A = available, W = working; a D in front means delayed (i.e., DF is delayed failure).

8 or 10 = the time core damage starts in hours.

10, 16 = core damage starts at 16 hours but the sequence progresses similarly to the 10 hour sequence just shifted in time (neglecting decay heat changes.

Table 2.4-3
Plant Damage States Appearing in LaSalle Level I Dominant Sequences: Flood Sequences

SEQ	ANAL	LOSP	SB	L/R	VENT	CDT	SRV	ADS	RHR	CSS	MFW	CDS	HPCS	RCIC	LPCS	LPCI	SPC	CRD	DDFW	PCS	DC	SBLC	A	B	[illegible]CS
FL1	FL	N	N	N	A	1	N	A	A	A	F	F	F	F	F	F	A	F	A	F	A	F	A	A	F
FL2	FL	N	N	N	A	1	N	A	F	F	F	F	F	F	F	F	F	F	A	F	A	F	A	A	F

FL = Flood in ANAL column.

FL1 = flood sequence 1 (FS1) and FL2 = flood sequence 2 (FS2) in SEQ column.

1 = core damage time is about 1 hour (80 minutes).

Table 2.4-4
Plant Damage States Appearing in LaSalle Level I Dominant Sequences: LOCA Sequences

SEQ	ANAL	LOSP	SB	L/R	VENT	CDT	SRV	ADS	RHR	CSS	MFW	CDS	HPCS	RCIC	LPCS	LPCI	SPC	CRD	DDFW	PCS	DC	SBLC	A	B	SCS
L14	L	N	N	L	F	CF+X	N	A	F	F	F	F	DF	DF	F	F,A	F	DF	A	F	A	F	A	A	F
L14	L	N	N	L	F	CF+X	N	A	F	F	F	F	DF	DF	F,A	F,A	F	DF	A	F	A	F	A	A	F
L14	L	N	N	L	F	CF+X	N	A	F	F	F	F	DF	DF	F,A	F	F	DF	A	F	A	F	A	A	F
L14	L	N	N	L	F	CF+X	N	A	F	F	F	F	DF	DF	F	F	F	DF	A	F	A	F	A	A	F
L14	L	N	N	L	F	CF+X	N	A	F	F,W	F	F	DF	DF	F,A	F,A	F,A	DF	A	F	A	F	A	A	F
L14	L	N	N	L	F	CF+X	N	A	F	F,W	F	F	DF	DF	F	F,A	F,A	DF	A	F	A	F	A	A	F

L = LOCA in ANAL column and Leak in L/R column.

CF+X means core damage occurs X hours after containment failure due to injectio 'ailure from severe environ.ents in the reactor building.

F,W or F,A mean system is failed after containment failure due to the severe environments in the reactor building.

Table 2.4-5
Plant Damage States Appearing in LaSalle Level I Dominant Sequences: ATWS Sequences

SEQ	ANAL	LOSP	SB	L/R	VENT	CDT	SRV	ADS	RHR	CSS	MFW	CDS	HPCS	RCIC	LPCS	LPCI	SPC	CRD	DDFW	PCS	DC	SBLC	A	B	SCS
A120	A	N	N	N	Y	1	1	A	F,W	F,W	A	A	DF	F	DF	DF	DF	DF	A	F	A	F	A	A	DF
A120	A	N	N	N	Y	1	N	A	F,W	F,W	F	A	DF	F	DF	DF	DF	DF	A	F	A	F	A	A	DF
A120	A	N	N	N	Y	1	N	A	F,W	F,W	A	A	DF	F	DF	DF	DF	DF	A	F	A	F	A	A	DF
A120	A	N	N	N	Y	1	N	A	F,W	F,W	F	F	DF	F	DF	DF	DF	F	A	F	A	F	A	A	DF
A120	A	N	N	N	Y	1	N	A	F,W	F,W	F	F	DF	F	DF	DF	DF	DF	A	F	A	F	A	A	DF
A120	A	Y	N	N	Y	1	N	A	F,W	F,W	F	F	DF	F	DF	DF	DF	DF	A	F	A	F	A	A	DF
A123	A	N	N	L	F	1	1	A	F,W	F,W	A	A	DF	F	DF	DF	DF	DF	A	F	A	F	A	A	DF
A123	A	N	N	L	F	1	N	A	F,W	F,W	A	A	DF	F	DF	DF	DF	DF	A	F	A	F	A	A	DF
A123	A	N	N	L	F	1	N	A	F,W	F,W	A	A	DF	F	F	DF	DF	DF	A	F	A	F	F	A	F
A123	A	N	N	L	F	1	N	A	F,W	F,W	F	F	DF	F	F	DF	DF	DF	A	F	A	F	A	F	F
A123	A	N	N	L	F	1	N	A	F,W	F,W	F	F	DF	F	DF	DF	DF	DF	A	F	A	F	A	A	DF
A123	A	Y	N	L	F	1	N	A	F,W	F,W	F	F	DF	F	DF	DF	DF	DF	A	F	A	F	A	A	F
A123	A	Y	N	L	F	1	N	A	F,W	F,W	F	F	DF	F	F	DF	DF	DF	A	F	A	F	F	A	F
A123	A	Y	N	L	F	1	N	A	F,W	F,W	F	F	DF	F	F	DF	DF	DF	A	F	A	F	A	F	F
A49	A	N	N	N	Y	1	1	A	F,W	F,W	A	A	DF	F	DF	DF	DF	DF	A	F	A	F	A	A	DF
A49	A	N	N	N	Y	1	1	A	F,W	F,W	A	A	DF	F	DF	DF	DF	DF	A	F	A	F	A	A	DF
A49	A	N	N	N	Y	1	1	A	F,W	F,W	A	A	DF	F	DF	DF	DF	DF	A	F	A	F	A	A	DF
A49	A	N	N	N	Y	1	N	F	F,W	F,W	A	A	DF	F	DF	DF	DF	DF	A	F	A	F	A	A	DF
A49	A	N	N	N	Y	1	N	A	F,W	F,W	A	A	DF	F	DF	DF	DF	DF	A	F	A	F	A	A	DF
A49	A	N	N	N	Y	1	N	A	F,W	F,W	A	A	DF	F	DF	DF	DF	DF	A	F	A	F	A	A	DF
A49	A	N	N	N	Y	1	N	A	F,W	F,W	A	A	DF	F	DF	DF	DF	DF	A	F	A	F	A	A	DF

A = ATWS in ANAL column.

Table 2.4-6
Plant Damage States Appearing in LaSalle Level I Dominant Sequences: Transient-Induced LOCA Sequences

SEQ	ANAL	LOSP	SB	L/R	VENT	CDT	SRV	ADS	RHR	CSS	MFW	CDS	HPCS	RCIC	LPCS	LPCI	SPC	CRD	DDFW	PCS	DC	SBLC	A	B	SCS
TL12	TL	N	N	N	Y	V+X	1	A	F	F,W	F	F	DF	DF	F	F,A	F,A	F,A	A	F	A	F	A	A	F
TL12	TL	N	N	N	Y	V+X	1	A	F	F,W	F	F	DF	F	F	F,A	F,A	F,A	A	F	A	F	A	A	F
TL12	TL	N	N	N	Y	V+X	1	A	F	F	[illegible]	F	DF	DF	F	F,A	F	F,A	A	F	A	F	A	A	F
TL12	TL	N	N	N	Y	V+X	1	A	F	F	F	F	DF	DF	F,A	F	F	F,A	A	F	A	F	A	A	F
TL12	TL	N	N	N	Y	V+X	1	A	F	F,W	F	F	DF	DF	F,A	F,A	F,A	F,A	A	F	A	F	A	A	F
TL59	TL	N	N	N	F	8	1	A	F	F	F	F	F	DF	F	F	F	F	A	F	A	R	F	A	F
TL59	TL	Y	Y	N	R	10	1	A	DR	DR	R	R	R	DF	R	DR	DR	R	A	R	R	R	F	DF	R
TL59	TL	Y	Y	N	R	10	1	A	DR	DR	R	R	DR	DF	R	DR	DR	R	A	R	R	R	F	DF	R
TL59	TL	Y	Y	N	R	10,16	1	R	DR	DR	R	R	R	DF	DR	DR	DR	R	A	R	R	R	DF	F	R
TL59	TL	Y	Y	N	R	10,16	1	R	DR	DR	R	R	R	DF	DR	DR	DR	R	A	R	R	R	DF	DF	R
TL59	TL	Y	Y	N	R	10,16	1	R	DR	DR	R	R	DR	DF	DR	DR	DR	R	A	R	R	R	DF	F	R
TL59	TL	Y	Y	N	R	8	1	R	F	F	R	R	F	DF	F	F	F	F	A	R	A	R	F	F	F
TL59	TL	Y	Y	N	R	10	1	R	R	R	R	R	DR	DF	R	R	R	R	A	R	R	R	F	F	R
TL59	TL	Y	Y	N	R	8	1	R	R	R	R	R	R	DF	R	R	R	R	A	R	R	R	F	F	R
TL97	TL	N	N	N	F	1	1	A	F	F	F	F	F	F	F	F	F	F	A	F	A	A	A	A	F
TL97	TL	Y	N	N	R	1	1	A	R	R	R	R	R	F	R	R	R	R	A	R	A	R	F	A	R
TL97	TL	Y	Y	N	R	1	1	A	R	R	R	R	F	F	R	R	R	F	A	R	A	R	F	F	R
TL97	TL	Y	Y	N	R	1	2	A	R	R	R	R	R	F	R	R	R	R	A	R	A	R	F	F	R
TL97	TL	Y	Y	N	R	1	1	A	F	F	R	R	F	F	F	F	F	F	A	R	A	R	F	F	F
TL97	TL	Y	Y	N	R	1	1	A	R	R	R	R	R	F	R	R	R	R	A	R	A	R	F	F	R
TL97	TL	Y	Y	N	R	1	1	A	R	R	R	R	F	F	R	R	R	R	A	R	A	R	F	F	R
TL97	TL	Y	Y	N	R	10	1	A	DR	DR	R	R	R	F	DR	DR	DR	R	A	R	A	R	DF	DF	R
TL97	TL	Y	Y	N	R	10	1	A	DR	DR	R	R	R	F	R	DR	DR	R	A	R	A	R	F	DF	R
TL97	TL	Y	Y	N	R	10	1	A	DR	DR	R	R	R	F	DR	DR	DR	R	A	R	A	R	DF	F	R
TL97	TL	Y	Y	N	R	10	1	R	R	R	R	R	DR	F	R	R	R	R	A	R	A	R	F	F	R

TL = Transient-induced LOCA in ANAL column.

V+X = core damage occurs X hours after venting due to injection system failure from severe environments in reactor building.

Table 2.4-7
Plant Damage States Appearing in LaSalle Level I Dominant Sequences: Fire Sequences

SEQ	ANAL	LOSP	SB	L/R	VENT	CDT	SRV	ADS	RHR	CSS	MFW	CDS	HPCS	RCIC	LPCS	LPCI	SPC	CRD	DDFW	PCS	DC	SBLC	A	B	SCS
FIAA	FI	N	N	R	F	CF+X	N	A	F	F	F	F	DF	DF	F,A	F	F	F	A	F	A	F	A	F	F
FIAA	FI	N	N	R	F	CF+X	N	A	F	F	F	F	DF	DF	F	F	F	F	A	F	A	F	A	F	F
FIAA	FI	N	N	R	F	CF+X	N	A	F	F,W	F	F	DF	DF	F,A	F,A	F,A	F	A	F	A	F	A	F	F
FIAA	FI	N	N	R	F	CF+X+Y	N	A	DF	DF	F	F	DF	DF	F	DF	DF	F	A	F	A	F	A	DF	F
FIAA	FI	N	N	R	F	CF+X+Y	N	A	DF	DF	F	F	DF	DF	F,A	DF	DF	F	A	F	A	F	A	DF	F
FIAA	FI	N	N	R	F	CF+X+Y	N	A	DF	F,W	F	F	DF	DF	F,A	F,A	F,A	F	A	F	A	F	A	DF	F
FIAA	FI	N	N	L	F	CF+X	N	A	F	F,W	F	F	DF	DF	F,A	F,A	F,A	F	A	F	A	F	A	F	F
FIAA	FI	N	N	L	F	CF+X	N	A	F	F	F	F	DF	DF	F	F	F	F	A	F	A	F	A	F	F
FIAA	FI	N	N	L	F	CF+X	N	A	F	F	F	F	DF	DF	F,A	F	F	F	A	F	A	F	A	F	F
FIAA	FI	N	N	L	F	CF+X+Y	N	A	DF	DF	F	F	DF	DF	F	DF	DF	F	A	F	A	F	A	DF	F
FIAA	FI	N	N	L	F	CF+X+Y	N	A	DF	DF	F	F	DF	DF	F,A	DF	DF	F	A	F	A	F	A	DF	F
FIAA	FI	N	N	L	F	CF+X+Y	N	A	DF	F,W	F	F	DF	DF	F,A	F,A	F,A	F	A	F	A	F	A	DF	F
FIAC	FI	N	N	R	F	CF+X	N	A	F	F,W	F	F	DF	DF	F	F,A	F,A	F,A	A	F	A	F	A	A	F
FIAC	FI	N	N	R	F	CF+X	N	A	F	F	F	F	DF	DF	F	F	F	F,A	A	F	A	F	A	A	F
FIAC	FI	N	N	R	F	CF+X	N	A	F	F	F	F	DF	DF	F	F,A	F	F,A	A	F	A	F	A	A	F
FIAC	FI	N	N	R	F	CF+X	N	A	F	F	F	F	DF	DF	F	F	F	F	A	F	A	F	A	A	F
FIAC	FI	N	N	L	F	CF+X	N	A	F	F,W	F	F	DF	DF	F	F,A	F,A	F,A	A	F	A	F	A	A	F
FIAC	FI	N	N	L	F	CF+X	N	A	F	F	F	F	DF	DF	F	F	F	F	A	F	A	F	A	A	F
FIAC	FI	N	N	L	F	CF+X	N	A	F	F	F	F	DF	DF	F	F	F	F,A	A	F	A	F	A	A	F
FIAC	FI	N	N	L	F	CF+X	N	A	F	F	F	F	DF	DF	F	F,A	F	F,A	A	F	A	F	A	A	F
FICR	FI	N	N	N	F	1	N	A	A	A	F	F	F	A	F	A	A	F	A	F	A	F	A	A	A
FIES2	FI	Y,N	Y	N	F	1	N	A	F	F	F	F	F	F	F	F	F	F	A	F	A	F	F	F	F
FIES2	FI	Y	Y	N	F	1	N	A	F	F	F	F	F	F	F	F	F	F	A	F	A	F	F	F	F
FIES3	FI	Y,N	Y	R	F	CF+X	N	F	F	F	F	F	DF	DF	F	F	F	F	A	F	F	F	F	F	F
FIES3	FI	Y,N	Y	N	Y	V+X	N	F	F	F	F	F	DF	DF	F	F	F	F	A	F	F	F	F	F	F
FIES3	FI	Y,N	Y	L	F	CF+X	N	F	F	F	F	F	DF	DF	F	F	F	F	A	F	F	F	F	F	F
FIN	FI	N	N	R	F	CF+X	N	A	F	F	F	F	DF	DF	F,A	F	F	F	A	F	A	F	A	F	F
FIN	FI	N	N	R	F	CF+X	N	A	F	F	F	F	DF	DF	F	F	F	F	A	F	A	F	A	F	F
FIN	FI	N	N	R	F	CF+X	N	A	F	F	F	F	DF	DF	F,A	F	F	F,A	A	F	A	F	A	F	F
FIN	FI	N	N	R	F	CF+X	N	A	F	F,W	F	F	DF	DF	F,A	F,A	F,A	F,A	A	F	A	F	A	F	F
FIN	FI	N	N	R	F	CF+X	N	A	F	F	F	F	DF	DF	F	F	F	F,A	A	F	A	F	A	F	F
FIN	FI	N	N	L	F	CF+X	N	A	F	F	F	F	DF	DF	F,A	F	F	F,A	A	F	A	F	A	F	F
FIN	FI	N	N	L	F	CF+X	N	A	F	F	F	F	DF	DF	F	F	F	F	A	F	A	F	A	F	F
FIN	FI	N	N	L	F	CF+X	N	A	F	F	F	F	DF	DF	F,A	F	F	F	A	F	A	F	A	F	F
FIN	FI	N	N	L	F	CF+X	N	A	F	F	F	F	DF	DF	F	F	F	F,A	A	F	A	F	A	F	F
FIN	FI	N	N	L	F	CF+X	N	A	F	F,W	F	F	DF	DF	F,A	F,A	F,A	F,A	A	F	A	F	A	F	F
FIP	FI	N	N	R	F	CF+X	N	A	F	F	F	F	DF	DF	F	F	F	F	A	F	A	F	A	A	F
FIP	FI	N	N	R	F	CF+X	N	A	F	F	F	F	DF	DF	F	F	F	F,A	A	F	A	F	A	A	F
FIP	FI	N	N	R	F	CF+X	N	A	F	F	F	F	DF	DF	F	F,A	F	F,A	A	F	A	F	A	A	F
FIP	FI	N	N	R	F	CF+X	N	A	F	F,W	F	F	DF	DF	F	F,A	F,A	F,A	A	F	A	F	A	A	F
FIP	FI	N	N	L	F	CF+X	N	A	F	F	F	F	DF	DF	F	F	F	F	A	F	A	F	A	A	F

Table 2.4-7 (Continued)
Plant Damage States Appearing in LaSalle Level I Dominant Sequences: Fire Sequences

SEQ	ANAL	LOSP	SB	L/R	VENT	CDT	SRV	ADS	RHR	CSS	MFW	CDS	HPCS	RCIC	LPCS	LPCI	SPC	CRD	DDFW	PCS	DC	SBLC	A	B	SCS
FIP	FI	N	N	L	F	CF+X	N	A	F	F	F	F	DF	DF	F	F	F	F,A	A	F	A	F	A	A	F
FIP	FI	N	N	L	F	CF+X	N	A	F	F	F	F	DF	DF	F	F,A	F	F,A	A	F	A	F	A	A	F
FIP	FI	N	N	L	F	CF+X	N	A	F	F,W	F	F	DF	DF	F	F,A	F,A	F,A	A	F	A	F	A	A	F
FISAA	FI	N	N	R	F	CF+X	N	A	F	F	F	F	DF	DF	F,A	F	F	F	A	F	A	F	A	F	F
FISAA	FI	N	N	R	F	CF+X	N	A	F	F,W	F	F	DF	DF	F,A	F,A	F,A	F	A	F	A	F	A	F	F
FISAA	FI	N	N	R	F	CF+X	N	A	F	F	F	F	DF	DF	F	F	F	F	A	F	A	F	A	F	F
FISAA	FI	N	N	R	F	CF+X+Y	N	A	DF	F,W	F	F	DF	DF	F,A	F,A	F,A	F	A	F	A	F	A	DF	F
FISAA	FI	N	N	R	F	CF+X+Y	N	A	DF	DF	F	F	DF	DF	F	DF	DF	F	A	F	A	F	A	DF	F
FISAA	FI	N	N	R	F	CF+X+Y	N	A	DF	DF	F	F	LF	DF	F,A	DF	DF	F	A	F	A	F	A	DF	F
FISAA	FI	N	N	L	F	CF+X	N	A	F	F	F	F	DF	DF	F,A	F	F	F	A	F	A	F	A	F	F
FISAA	FI	N	N	L	F	CF+X	N	A	F	F,W	F	F	DF	DF	F,A	F,A	F,A	F	A	F	A	F	A	F	F
FISAA	FI	N	N	L	F	CF+X	N	A	F	F	F	F	DF	DF	F	F	F	F	A	F	A	F	A	F	F
FISAA	FI	N	N	L	F	CF+X+Y	N	A	DF	F,W	F	F	DF	DF	F,A	F,A	F,A	F	A	F	A	F	A	DF	F
FISAA	FI	N	N	L	F	CF+X+Y	N	A	DF	DF	F	F	DF	DF	F	DF	DF	F	A	F	A	F	A	DF	F
FISAA	FI	N	N	L	F	CF+X+Y	N	A	DF	DF	F	F	DF	DF	F,A	DF	DF	F	A	F	A	F	A	DF	F
FISW	FI	N	N	R	F	CF+X	N	A	F	F	F	F	DF	DF	F	F	F	F	A	F	A	F	A	A	F
FISW	FI	N	N	R	F	CF+X	N	A	F	F	F	F	DF	DF	F,A	F	F	F,A	A	F	A	F	A	A	F
FISW	FI	N	N	R	F	CF+X	N	A	F	F	F	F	DF	DF	F	F	F	F,A	A	F	A	F	A	A	F
FISW	FI	N	N	R	F	CF+X	N	A	F	F,W	F	F	DF	DF	F,A	F,A	F,A	F,A	A	F	A	F	A	A	F
FISW	FI	N	N	L	F	CF+X	N	A	F	F	F	F	DF	DF	F,A	F	F	F,A	A	F	A	F	A	A	F
FISW	FI	N	N	L	F	CF+X	N	A	F	F	F	F	DF	DF	F	F	F	F	A	F	A	F	A	A	F
FISW	FI	N	N	L	F	CF+X	N	A	F	F	F	F	DF	DF	F	F	F	F,A	A	F	A	F	A	A	F
FISW	FI	N	N	L	F	CF+X	N	A	F	F,W	F	F	DF	DF	F,A	F,A	F,A	F,A	A	F	A	F	A	A	F
FIT	FI	N	N	R	F	CF+X	N	A	F	F	F	F	DF	DF	F	F	F	F,A	A	F	A		A	A	F
FIT	FI	N	N	R	F	CF+X	N	A	F	F	F	F	DF	DF	F	F,A	F	F,A	A	F	A	[illegible]	A	A	F
FIT	FI	N	N	R	F	CF+X	N	A	F	F,W	F	F	DF	DF	F	F,A	F,A	F,A	A	F	A	F	A	A	F
FIT	FI	N	N	N	Y	V+X	N	A	F	F	F	F	DF	DF	F	F	F	F,A	A	F	A	F	A	A	F
FIT	FI	N	N	N	Y	V+X	N	A	F	F	F	F	DF	DF	F	F,A	F	F,A	A	F	A	F	A	A	F
FIT	FI	N	N	N	Y	V+X	N	A	F	F,W	F	F	DF	DF	F	F,A	F,A	F,A	A	F	A	F	A	A	F
FIT	FI	N	N	L	F	CF+X	N	A	F	F	F	F	DF	DF	F	F	F	F,A	A	F	A	F	A	A	F
FIT	FI	N	N	L	F	CF+X	N	A	F	F	F	F	DF	DF	F	F,A	F	F,A	A	F	A	F	A	A	F
FIT	FI	N	N	L	F	CF+X	N	A	F	F,W	F	F	DF	DF	F	F,A	F,A	F,A	A	F	A	F	A	A	F
FIW1	FI	N	N	R	F	CF+X	N	A	F	F	F	F	DF	DF	F,A	F	F	F,A	A	F	A	F	A	A	F
FIW1	FI	N	N	R	F	CF+X	N	A	F	F,W	F	F	DF	DF	F,A	F,A	F,A	F,A	A	F	A	F	A	A	F
FIW1	FI	N	N	R	F	CF+X	N	A	F	F	F	F	DF	DF	F	F	F	F	A	F	A	F	A	A	F
FIW1	FI	N	N	R	F	CF+X	N	A	F	F	F	F	DF	DF	F	F	F	F,A	A	F	A	F	A	A	F
FIW1	FI	N	N	L	F	CF+X	N	A	F	F	F	F	DF	DF	F	F	F	F	A	F	A	F	A	A	F
FIW1	FI	N	N	L	F	CF+X	N	A	F	F,W	F	F	DF	DF	F,A	F,A	F,A	F,A	A	F	A	F	A	A	F
FIW1	FI	N	N	L	F	CF+X	N	A	F	F	F	F	DF	DF	F	F	F	F,A	A	F	A	F	A	A	F
FIW1	FI	N	N	L	F	CF+X	N	A	F	F	F	F	DF	DF	F,A	F	F	F,A	A	F	A	F	A	A	F
FIW2	FI	N	N	R	F	CF+X	N	A	F	F,W	F	F	DF	DF	F,A	F,A	F,A	F,A	A	F	A	F	A	A	F

Table 2.4-7 (Concluded)
Plant Damage States Appearing in LaSalle Level I Dominant Sequences: Fire Sequences

SEQ	ANAL	LOSP	SB	L/R	VENT	CDT	SRV	ADS	RHR	CSS	MFW	CDS	HPCS	RCIC	LPCS	LPCI	SPC	CRD	DDFW	PCS	DC	SBLC	A	B	SCS
FIW2	FI	N	N	R	F	CF+X	N	A	F	F	F	F	DF	DF	F	F	F	F	A	F	A	F	A	A	F
FIW2	FI	N	N	R	F	CF+X	N	A	F	F	F	F	DF	DF	F,A	F	F	F,A	A	F	A	F	A	A	F
FIW2	FI	N	N	R	F	CF+X	N	A	F	F	F	F	DF	DF	F	F	F	F,A	A	F	A	F	A	A	F
FIW2	FI	N	N	L	F	CF+X	N	A	F	F	F	F	DF	DF	F,A	F	F	F,A	A	F	A	F	A	A	F
FIW2	FI	N	N	L	F	CF+X	N	A	F	F	F	F	DF	DF	F	F	F	F,A	A	F	A	F	A	A	F
FIW2	FI	N	N	L	F	CF+X	N	A	F	F,W	F	F	DF	DF	F,A	F,A	F,A	F,A	A	F	A	F	A	A	F
FIW2	FI	N	N	L	F	CF+X	N	A	F	F	F	F	DF	DF	F	F	F	F	A	F	A	F	A	A	F
FIY1	FI	N	N	R	F	CF+X	N	A	F	F	F	F	DF	DF	F	F,A	F	F,A	A	F	A	F	A	A	F
FIY1	FI	N	N	R	F	CF+X	N	A	F	F	F	F	DF	DF	F	F	F	F	A	F	A	F	A	A	F
FIY1	FI	N	N	R	F	CF+X	N	A	F	F,W	F	F	DF	DF	F	F,A	F,A	F,A	A	F	A	F	A	A	F
FIY1	FI	N	N	R	F	CF+X	N	A	F	F	F	F	DF	DF	F	F	F	F,A	A	F	A	F	A	A	F
FIY1	FI	N	N	L	F	CF+X	N	A	F	F	F	F	DF	DF	F	F	F	F	A	F	A	F	A	A	F
FIY1	FI	N	N	L	F	CF+X	N	A	F	F	F	F	DF	DF	F	F	F	F,A	A	F	A	F	A	A	F
FIY1	FI	N	N	L	F	CF+X	N	A	F	F	F	F	DF	DF	F	F,A	F	F,A	A	F	A	F	A	A	F
FIY1	FI	N	N	L	F	CF+X	N	A	F	F,W	F	F	DF	DF	F	F,A	F,A	F,A	A	F	A	F	A	A	F
FIY2	FI	N	N	R	F	CF+X	N	A	F	F,W	F	F	DF	DF	F	F,A	F,A	F,A	A	F	A	F	A	A	F
FIY2	FI	N	N	R	F	CF+X	N	A	F	F	F	F	DF	DF	F	F	F	F	A	F	A	F	A	A	F
FIY2	FI	N	N	R	F	CF+X	N	A	F	F	F	F	DF	DF	F	F	F	F,A	A	F	A	F	A	A	F
FIY2	FI	N	N	R	F	CF+X	N	A	F	F	F	F	DF	DF	F	F,A	F	F,A	A	F	A	F	A	A	F
FIY2	FI	N	N	L	F	CF+X	N	A	F	F,W	F	F	DF	DF	F	F,A	F,A	F,A	A	F	A	F	A	A	F
FIY2	FI	N	N	L	F	CF+X	N	A	F	F	F	F	DF	DF	F	F	F	F,A	A	F	A	F	A	A	F
FIY2	FI	N	N	L	F	CF+X	N	A	F	F	F	F	DF	DF	F	F,A	F	F,A	A	F	A	F	A	A	F
FIY2	FI	N	N	L	F	CF+X	N	A	F	F	F	F	DF	DF	F	F	F	F	A	F	A	F	A	A	F
FIZ	FI	N	N	R	F	CF+X	N	A	F	F,W	F	F	DF	DF	F	F,A	F,A	F,A	A	F	A	F	A	A	F
FIZ	FI	N	N	R	F	CF+X	N	A	F	F	F	F	DF	DF	F	F	F	F	A	F	A	F	A	A	F
FIZ	FI	N	N	R	F	CF+X	N	A	F	F	F	F	DF	DF	F	F,A	F	F,A	A	F	A	F	A	A	F
FIZ	FI	N	N	R	F	CF+X	N	A	F	F	F	F	DF	DF	F	F	F	F,A	A	F	A	F	A	A	F
FIZ	FI	N	N	L	F	CF+X	N	A	F	F	F	F	DF	DF	F	F,A	F	F,A	A	F	A	F	A	A	F
FIZ	FI	N	N	L	F	CF+X	N	A	F	F	F	F	DF	DF	F	F	F	F,A	A	F	A	F	A	A	F
FIZ	FI	N	N	L	F	CF+X	N	A	F	F,W	F	F	DF	DF	F	F,A	F,A	F,A	A	F	A	F	A	A	F
FIZ	FI	N	N	L	F	CF+X	N	A	F	F	F	F	DF	DF	F	F	F	F	A	F	A	F	A	A	F

FI = Fire analysis in ANAL column.

FIX = FIRE-X sequence in SEQ column. The "-" is removed from the sequence name.

L/R = Containment failure by Leak or Rupture in L/R column.

CF+X+Y = core damage at containment failure + X + Y hours. Y is from delayed failure, treat the same as CF+X just sifted in time, similar to 10,16 approximation in Table 2.4-2.

Table 2.4-8
Plant Damage States Appearing in LaSalle Level I Dominant Sequences: Transient Sequences

SEQ	ANAL	LOSP	SB	L/R	VENT	CDT	SRV	ADS	RHR	CSS	MFW	CDS	HPCS	RCIC	LPCS	LPCI	SPC	CRD	DDFW	PCS	DC	SBLC	A	B	SCS
T100	T	Y	Y	N	R	1	N	A	R	R	R	R	R	R	F	R	R	R	A	R	A	R	F	F	R
T100	T	N	N	N	R	1	N	A	F	F	F	F	F	F	F	F	F	F	A	F	A	F	A	A	F
T100	T	Y	Y	N	R	1	N	A	R	R	R	R	F	R	R	R	R	R	A	R	A	R	F	F	R
T100	T	Y	Y	N	R	1	N	A	R	R	R	R	R	F	R	R	R	R	A	R	A	R	F	F	R
T100	T	Y	Y	N	R	1	N	A	F	F	R	R	F	R	F	F	F	F	A	R	A	R	F	F	F
T100	T	Y	Y	N	R	1	N	A	R	R	R	R	R	F	R	R	R	R	A	R	A	R	F	F	R
T100	T	N	N	N	A	1	N	A	F	F	F	F	F	R	F	F	F	F	A	F	A	F	A	A	F
T100	T	Y	Y	N	R	1	N	A	R	R	R	R	R	R	R	R	R	R	A	R	A	R	F	F	R
T100	T	Y	Y	N	R	10	N	A	DR	DR	R	R	R	R	DR	DR	DR	R	A	R	A	R	DF	F	R
T100	T	Y	Y	N	R	10	N	A	DR	DR	R	R	R	R	R	DR	DR	R	A	R	A	R	F	DF	R
T100	T	Y	Y	N	R	10	N	A	DR	DR	R	R	R	R	DR	DR	DR	R	A	R	A	R	DF	DF	R
T100	T	Y	Y	N	R	10	N	A	DR	DR	R	R	DR	R	R	DR	DR	R	A	R	A	R	F	DF	R
T100	T	Y	Y	N	R	10	N	R	R	R	R	R	DR	F	R	R	R	R	A	R	R	R	F	F	R
T101	T	Y	Y	N	R	1	N	R	R	R	R	R	R	R	R	R	R	R	A	R	R	R	F	F	R
T16	T	N	N	N	Y	V+X	N	A	F	F	F	A	DF	DF	F	F	F	F,A	A	F	A	F	A	A	F
T16	T	N	N	N	Y	V+X	N	A	F	F	F	A	DF	F	F	F	F	F,A	A	F	A	F	A	A	F
T16	T	N	N	N	Y	V+X	N	A	F	F	F	F	DF	DF	F	F	F	F,A	A	F	A	F	A	A	F
T16	T	N	N	N	Y	V+X	N	A	F	F,W	F	A	DF	DF	F	F,A	F,A	F,A	A	F	A	F	A	A	F
T16	T	N	N	N	Y	V+X	N	A	F	F,W	F	F	DF	DF	F,A	F,A	F,A	F,A	A	F	A	F	A	A	F
T16	T	N	N	N	Y	V+X	N	A	F	F,W	F	A	DF	F	F	F,A	F,A	F,A	A	F	A	F	A	A	F
T16	T	N	N	N	Y	V+X	N	A	F	F	F	F	DF	DF	F,A	F	F	F,A	A	F	A	F	A	A	F
T16	T	N	N	N	Y	V+X	N	A	F	F	F	A	DF	F	F	F,A	F	F,A	A	F	A	F	A	A	F
T16	T	N	N	N	Y	V+X	N	A	F	F,W	F	A	DF	DF	F,A	F,A	F,A	F,A	A	F	A	F	A	A	F
T16	T	N	N	N	Y	V+X	N	A	F	F	F	A	DF	DF	F	F	F	F	A	F	A	F	A	A	F
T16	T	N	N	N	Y	V+X	N	A	F	F	F	A	DF	DF	F,A	F	F	F,A	A	F	A	F	A	A	F
T16	T	Y	N	N	Y	V+X	N	A	F	F	R	R	DF	DF	F,A	F	F	F,R	A	R	A	F	A	A	F
T18	T	N	N	L	F	CF+X	N	A	F,A	F,A	F	A	DF	DF	F	F,A	F,A	DF	A	F	A	F	F	A	F
T18	T	N	N	L	F	CF+X	N	A	F,A	F,A	F	A	DF	DF	F,A	F,A	F,A	DF	A	F	A	F	A	F	F
T18	T	N	N	L	F	CF+X	N	A	F	F	F	A	DF	DF	F,A	F	F	DF	A	F	A	F	A	A	F
T18	T	N	N	L	F	CF+X	N	A	F	F	F	A	DF	DF	F	F	F	DF	A	F	A	F	A	F	F
T18	T	N	N	L	F	CF+X	N	A	F	F	F	A	DF	DF	F	F	F	DF	A	F	A	F	A	A	F
T18	T	N	N	L	F	CF+X	N	A	F	F,W	F	A	DF	DF	F,A	F,A	F,A	DF	A	F	A	F	A	A	F
T18	T	N	N	L	F	CF+X	N	A	F	F	F	A	DF	F	F	F	F	DF	A	F	A	F	A	A	F
T18	T	N	N	L	F	CF+X	N	A	F	F	F	A	DF	DF	F	F	F	DF	A	F	A	F	F	A	F
T20	T	N	N	R	F	CF+X	N	A	F	F	F	A	DF	F	F	F	F	DF	A	F	A	F	A	A	F
T20	T	N	N	R	F	CF+X	N	A	F,A	F,A	F	A	DF	DF	F,A	F,A	F,A	DF	A	F	A	F	A	F	F
T20	T	N	N	R	F	CF+X	N	A	F	F	F	A	DF	DF	F	F	F	DF	A	F	A	F	A	F	F
T20	T	N	N	R	F	CF+X	N	A	F	F	F	A	DF	DF	F	F	F	DF	A	F	A	F	F	A	F
T20	T	N	N	R	F	CF+X	N	A	F	F,W	F	A	DF	DF	F,A	F,A	F,A	DF	A	F	A	F	A	A	F

Table 2.4-8 (Continued)
Plant Damage States Appearing in LaSalle Level I Dominant Sequences: Transient Sequences

SEQ	ANAL	LOSP	SB	L/R	VENT	CDT	SRV	ADS	RHR	CSS	MFW	CDS	HPCS	RCIC	LPCS	LPCI	SPC	CRD	DDFW	PCS	DC	SBLC	A	B	SCS
T20	T	N	N	R	F	CP+X	N	A	F	F	F	A	DF	DF	F	F	F	DF	A	F	A	F	A	A	F
T20	T	N	N	R	F	CP+X	N	A	F,A	F,A	F	A	DF	DF	F	F,A	F,A	DF	A	F	A	F	F	A	F
T20	T	N	N	R	F	CP+X	N	A	F	F	F	A	DF	DF	F,A	F	F	DF	A	F	A	F	A	A	F
T20	T	N	N	R	F	CP+X	N	A	F	F,W	F	F	DF	DF	F,A	F,A	F,A	DF	A	F	A	F	A	A	F
T22	T	N	N	L	F	CP+X	N	A	F,A	F,A	F	A	DF	DF	F	F,A	F,A	F,A	A	F	A	F	F	A	F
T22	T	N	N	L	F	CP+X	N	F	F	F	F	A	DF	F	F	F	F	F	A	F	A	F	A	F	F
T22	T	N	N	L	F	CP+X	N	A	F	F,W	F	A	DF	DF	F	F,A	F,A	F	A	F	A	F	F	A	F
T22	T	N	N	L	F	CP+X	N	F	F	F	F	A	DF	DF	F	[illegible]	F	F	A	F	A	F	F	A	F
T22	T	N	N	L	F	CP+X	N	A	F	F	F	A	DF	DF	F	F	F	F	A	F	A	F	A	F	F
T22	T	Y	N	L	F	CP+X	N	A	F	F,W	R	R	DF	DF	F,A	F,A	F,A	F,R	A	R	A	F	A	F	F
T22	T	Y	N	L	F	CP+X	N	A	F	F,W	R	R	DF	DF	F,R	F,A	F,A	F,R	A	R	A	F	F	A	F
T22	T	Y	Y	L	F	CP+X	N	R	F,R	F,R	R	R	DF	DF	F,R	F,R	F,R	F,R	A	R	A	F	F	F	F,R
T22	T	Y	N	L	F	CP+X	N	A	F,R	F,R	R	R	DF	DF	F,A	F,R	F,R	F,R	A	R	A	F	A	F	F,R
T22	T	Y	Y	L	F	CP+X	N	R	F,R	F,R	R	R	DF	DF	F,R	F,R	F,R	F,R	A	R	A	F	DF	F	F,R
T22	T	Y	Y	L	F	CP+X	N	R	F,R	F,R	R	R	DF	DF	F	F,R	F,R	F,R	A	R	A	F	F	F	F,R
T22	T	Y	N	L	F	CP+X	N	A	F,R	F,R	R	R	DF	DF	F	F,R	F,R	F,R	A	R	A	F	A	F	F,R
T22	T	Y	Y	L	F	CP+X	N	R	F,R	F,R	R	R	DF	DF	F,R	F,R	F,R	F,R	A	R	A	F	DF	DF	F,R
T22	T	Y	Y	L	F	CP+X	N	R	F,R	F,R	R	R	DF	DF	F,R	F,R	F,R	F,R	A	R	A	F	F	F	F,R
T22	T	Y	Y	L	F	CP+X	N	R	F,R	F,R	R	R	DF	DF	F,R	F,R	F,R	F,R	A	R	A	F	F	DF	F,R
T24	T	N	N	R	F	CP+X	N	F	F	F	F	A	DF	F	F	F	F	F	A	F	A	F	A	F	F
T24	T	Y	N	R	F	CP+X	N	A	F	F,W	R	R	DF	DF	F,A	F,A	F,A	F,R	A	R	A	F	A	F	F
T24	T	N	N	R	F	CP+X	N	F	F	F	F	A	DF	DF	F	F	F	F	A	F	A	F	F	A	F
T24	T	Y	N	R	F	CP+X	N	A	F	F,W	R	R	DF	DF	F,R	F,A	F,A	F,R	A	R	A	F	F	A	F
T24	T	Y	Y	R	F	CP+X	N	R	F,R	F,R	R	R	DF	DF	F,R	F,R	F,R	F,R	A	R	A	F	DF	F	F,R
T24	T	Y	Y	R	F	CP+X	N	R	F,R	F,R	R	R	DF	DF	F,R	F,R	F,R	F,R	A	R	A	F	F	DF	F,R
T24	T	Y	Y	R	F	CP+X	N	R	F,R	F,R	R	R	DF	DF	F	F,R	F,R	F,R	A	R	A	F	F	F	F,R
T24	T	Y	Y	R	F	CP+X	N	R	F,R	F,R	R	R	DF	DF	F,R	F,R	F,R	F,R	A	R	A	F	DF	DF	F,R
T24	T	N	N	R	F	CP+X	N	A	F,A	F,A	F	A	DF	DF	F	F,A	F,A	F,A	A	F	A	F	F	A	F
T24	T	Y	N	R	F	CP+X	N	A	F,R	F,R	R	R	DF	DF	F	R	F,R	F,R	A	R	A	F	A	F	F,R
T24	T	Y	Y	R	F	CP+X	N	R	F,R	F,R	R	R	DF	DF	F,R	F,R	F,R	F,R	A	R	A	F	F	F	F,R
T24	T	N	N	R	F	CP+X	N	A	F	F	F	A	DF	DF	F	F	F	F	A	F	A	F	A	F	F
T24	T	N	N	R	F	CP+X	N	A	F	F,W	F	A	DF	DF	F	F,A	F,A	F	A	F	A	F	[illegible]	A	F
T24	T	Y	N	R	F	CP+X	N	A	F,R	F,R	R	R	DF	DF	F,A	F,R	F,R	F,R	A	R	A	F	A	F	F,R
T24	T	Y	Y	R	F	CP+X	N	R	F,R	F,R	R	R	DF	DF	F,R	F,R	F,R	F,R	A	R	A	F	F	F	F,R
T38	T	N	N	L	F	CP+X	N	A	F	F	F	A	F,A	DF	F,A	F,A	F	DF	A	F	A	F	A	F	F
T38	T	N	N	L	F	CP+X	N	A	F	F,W	F	A	F	DF	F,A	F,A	F,A	DF	A	F	A	F	A	F	F
T38	T	N	N	L	F	CP+X	N	A	F	F	F	A	F,A	DF	F	F,A	F	DF	A	F	A	F	A	F	F
T38	T	N	N	L	F	CP+X	N	A	F	F,W	F	A	F	DF	F	F,A	F,A	DF	A	F	A	F	F	A	F
T38	T	N	N	L	F	CP+X	N	A	F	F	F	A	F	DF	F	F	F	DF	A	F	A	F	F	A	F
T38	T	N	N	L	F	CP+X	N	A	F	F	F	A	F,A	DF	F,A	F,A	F	DF	A	F	A	F	F	A	F

Table 2.4-8 (Continued)
Plant Damage States Appearing in LaSalle Level I Dominant Sequences: Transient Sequences

SEQ	ANAL	LOSP	S"	L/R	VENT	CDT	SRV	ADS	RHR	CSS	MFW	CDS	HPCS	RCIC	LPCS	LPCI	SPC	CRD	DDFW	PCS	DC	SBLC	A	B	SCS
T38	T	N	N	L	F	CF+X	N	A	F	F	F	A	F,A	DF	F,A	F	F	DF	A	F	A	F	A	F	F
T38	T	N	N	L	F	CF+X	N	A	F	F	F	A	F	DF	F	F	F	DF	A	F	A	F	A	F	F
T38	T	N	N	L	F	CF+X	N	A	F	F,W	F	A	F	DF	F	F,A	F,A	DF	A	F	A	F	F	A	F
T38	T	N	N	L	F	CF+X	N	A	F	F	F	A	F,A	DF	F	F,A	F	DF	A	F	A	F	F	A	F
T41	T	N	N	N	F	SRV+X	N	A	F	F	F	DF	F	DF	A	F	F	F	A	F	A	F	A	F	F
T41	T	N	N	N	F	SRV+X	N	A	F	F	F	DF	F	DF	F	F	F	F	A	F	A	F	F	A	F
T41	T	N	N	N	F	SRV+X	N	A	F	W	F	DF	F	DF	F	A	A	F	A	F	A	F	F	A	F
T41	T	N	N	N	F	SRV+X	N	A	F	F	F	DF	F	DF	F	A	F	F	A	F	A	F	F	A	F
T41	T	N	N	N	F	SRV+X	N	A	F	W	F	DF	F	DF	A	A	A	F	A	F	A	F	A	F	F
T41	T	N	N	N	F	SRV+X	N	A	F	F	F	DF	F	DF	F	F	F	F	A	F	A	F	A	F	F
T47	T	N	N	L	F	CF+X	N	A	F	F,W	F	A	F	DF	F	F,A	F,A	DF	A	F	A	F	F	A	F
T47	T	N	N	L	F	CF+X	N	A	F	F,W	F	A	F	DF	F	F,A	F,A	DF	A	F	A	F	F	A	F
T47	T	N	N	L	F	CF+X	N	A	F	F,W	F	A	F	DF	F,A	F,A	F,A	DF	A	F	A	F	A	F	F
T47	T	N	N	L	F	CF+X	N	A	F	F	F	A	F,A	DF	F	F,A	F	DF	A	F	A	F	F	A	F
T47	T	N	N	L	F	CF+X	N	A	F	F	F	A	F,A	DF	F	F,A	F	DF	A	F	A	F	F	A	F
T47	T	N	N	L	F	CF+X	N	A	F	F	F	F	F,A	DF	F	F,A	F	DF	A	F	A	F	F	A	F
T47	T	N	N	L	F	CF+X	N	A	F	F,W	F	F	F	DF	F	F,A	F,A	DF	A	F	A	F	F	A	F
T47	T	N	N	L	F	CF+X	N	A	F	F	F	F	F	DF	F	F,A	F	DF	A	F	A	F	F	A	F
T50	T	Y	N	N	R	SRV+X	N	A	F	W	R	R	R	DF	DF	A	A	F	A	R	A	R	DF	A	R
T50	T	N	N	N	F	SRV+X	N	A	F	W	F	F	F	DF	F	A	A	F	A	F	A	F	A	A	F
T50	T	Y	N	N	R	SRV+X	N	A	F	W	R	R	R	DF	A	A	A	F	A	R	A	R	A	F	R
T50	T	N	N	N	F	SRV+X	N	A	F	F	F	F	F	DF	F	A	F	F	A	F	A	F	A	A	F
T50	T	Y	N	N	R	SRV+X	N	A	F	W	R	R	R	DF	A	A	A	F	A	R	A	R	A	DF	R
T50	T	N	N	N	F	SRV+X	N	A	F	W	F	F	F	DF	F	A	A	F	A	F	A	F	F	A	F
T50	T	Y	N	N	R	SRV+X	N	A	F	W	R	R	R	DF	F	A	A	F	A	R	A	R	F	A	R
T58	T	N	N	L	F	CF+X	N	A	F,A	F,A	F	A	F,A	DF	F,A	F,A	F,A	DF	A	F	A	F	A	F	F
T59	T	Y	N	N	F	SRV+X	N	A	F	F	F	F	F	DF	A	F	F	F	A	F	A	F	A	F	F
T59	T	Y	N	N	F	SRV+X	N	A	F	F	F	F	DF	DF	A	F	F	F	A	F	A	F	A	F	F
T59	T	Y	N	N	F	SRV+X+Y	N	A	DF	DF	F	F	DF	DF	A	DF	DF	F	A	F	A	F	A	DF	F
T59	T	Y	N	N	F	SRV+X+Y	N	A	DF	DF	F	F	F	DF	A	DF	DF	F	A	F	A	F	A	DF	F
T62	T	Y	Y	N	R	8	N	R	F	F	R	R	F	DF	F	F	F	F	A	F	R	F	F	F	F
T62	T	Y	Y	N	R	8	N	R	R	R	R	R	R	DF	R	R	R	R	A	R	R	R	F	F	R
T62	T	Y,N	Y	N	F	8	N	A	F	F	R	F,A	F	DF	F	F	F	F	A	F	A	F	F	F	F
T62	T	Y,N	Y	N	F	8	N	A	F	F	R	F	F	DF	F	F	F	F	A	F	A	F	F	F	F
T62	T	Y	Y	N	R	10	N	R	R	R	R	R	DF	DF	R	R	R	R	A	R	R	R	F	F	R
T62	T	Y	Y	N	R	10	N	A	DR	DR	R	R	DF	DF	R	DR	DR	R	A	R	A	R	F	DF	R
T62	T	Y	Y	N	R	10	N	A	DR	DR	R	R	R	LF	R	DR	DR	R	A	R	A	R	F	DF	R
T62	T	Y	Y	N	R	10,16	N	R	DR	DR	R	R	R	DF	DR	DR	DR	R	A	R	R	R	DF	F	R
T62	T	Y	Y	N	R	10,16	N	R	DR	DR	R	R	R	DF	DR	DR	DR	R	A	R	R	R	DF	DF	R
T62	T	Y	Y	N	R	10,16	N	R	DR	DR	R	R	LF	DF	DR	DR	DR	R	A	R	R	R	DF	F	R

Table 2.4-8 (Concluded)
Plant Damage States Appearing in LaSalle Level I Dominant Sequences: Transient Sequences

SEQ	ANAL	LOSP	SB	L/R	VENT	CJT	SRV	ADS	RHR	CSS	MFW	CDS	HPCS	RCIC	LPCS	LPCI	SPC	CRD	DDFW	PCS	DC	SBLC	A	B	SCS
T70	T	N	N	L	F	CF+X	N	A	F,A	F,A	F	A	F,A	F	F	F,A	F,A	DF	A	F	A	F	F	A	F
T70	T	N	N	L	F	CF+X	N	A	F	F	F	A	F,A	F	F	F,A	F	DF	A	F	A	F	F	A	F
T70	T	N	N	L	F	CF+X	N	A	F	F	F	A	F	F	F,A	F	F	DF	A	F	A	F	A	F	F
T70	T	N	N	L	F	CF+X	N	A	F	F	F	A	F	F	F	F	F	DF	A	F	A	F	F	A	F
T70	T	N	N	L	F	CF+X	N	A	F	F	F	A	F	F	F	F	F	DF	A	F	A	F	A	F	F
T70	T	N	N	L	F	CF+X	N	A	F	F	F	A	F	F	F	F,A	F	DF	A	F	A	F	F	A	F
T70	T	N	N	L	F	CF+X	N	A	F	F,W	F	A	F	F	F	F,A	F,A	DF	A	F	A	F	F	A	F
T82	T	N	N	L	F	CF+X	N	A	F	F	F	F	F	F	F	F,A	F	DF	A	F	A	F	F	A	F
T82	T	N	N	L	F	CF+X	N	A	F	F,W	F	F	F,A	F	F	F,A	F,A	DF	A	F	A	F	F	A	F
T82	T	N	N	L	F	CF+X	N	A	F	F,W	F	F	F	F	F	F,A	F,A	DF	A	F	A	F	F	A	F
T82	T	N	N	L	F	CF+X	N	A	F	F	F	F	F,A	F	F	F,A	F	DF	A	F	A	F	F	A	F
T85	T	N	N	N	F	SRV+X	N	A	F	W	F	F	F,A	F	F	A	A	F	A	F	A	F	F	A	F
T85	T	N	N	N	F	SRV+X	N	A	F	F	F	F	F,A	F	F	A	F	F	A	F	A	F	F	A	F
T85	T	Y	N	N	F	SRV+X	N	A	DF	DF	F	F	F	F	F	DF	DF	F	A	F	A	F	A	DF	F
T85	T	N	N	N	F	SRV+X	N	A	F,A	F,A	F	F	F,A	F	F	A	A	F	A	F	A	F	F	A	F

T= Transient analysis in ANAL column.

Y,N = not a station blackout due to DG failure but due to other electrical system failures.

SRV+X = core damage X hours after containment pressure reaches SRV reclusure pressure and SRVs reclose failing low-pressure injection.

SRV+X+Y same as SRV+X just shifted Y hours due to delayed failures.

The above procedure is labor intensive because of the large number of cut sets involved in the LaSalle analysis - 11,656 cut sets in the top 50 sequences after truncation in the Level I analysis. Each cut set had to examined and classified. However, it is necessary to look at all of these cut sets for the following reasons:

1. to be assured of obtaining a resolution of the PDSs to the same level of detail as used in the rest of the analysis,

2. to minimize the chance of missing a significant PDS (probability or consequence) and having to go back and modify the APET,

3. it reduces significantly the chance of errors in the PDS definition, and

4. if a PDS is left out, it insures a greater chance that at least the defining characteristics will be in the APET structure (i.e, the characteristics represented by the initial questions will be in the APET).

After the cut sets were regrouped, point estimate TEMAC calculations were performed on the larger plant damage states in order to truncate the cut sets at 99% of the remaining probability. The final number of cut sets retained for the Level II/III analysis was 5508.

2.5 Final PDSs

Clearly, examination of Tables 2.4-2 thru 2.4-8 shows that there are too many PDSs defined for propagation through the analysis (roughly 300). In order to reduce the number of PDSs to a manageable number, the following procedure was followed:

1. Tables 2.4-2 thru 2.4-8 were combined into one table with all PDSs and all accident sequences.

2. The PDSs were sorted in various ways to determine if there were any natural groupings of similar PDSs. The sorts were done by rearranging the columns to put the most important characteristics or characteristics that tended to divide the PDSs into large groups first. The patterns obtained were compared and the PDSs were divide into groups which were assessed to have similar accident characteristics.

3. After sorting and comparing the groups defined by these sorts, one particular ordering of the questions was selected to be used to define the final grouping. This ordering is shown both in the list of initial PDSs given in Tables 2.4-2 thru 2.4-8 and in the final list of PDSs given in Table 2.5-1

Table 2.5-1
Final LaSalle Plant Damage States

PDS	SEQ	ANAL	LOSP	SB	L/R	VENT	CDT	SRV	ADS	RHR	CSS	MFW	CDS	HPCS	RCIC	LPCS	LPCI	SPC	CRD	DDFW	PCS	DC	SBLC	A	B	SCS
EQ1/2-a	S1	EQ	Y	Y	N	R	10	N	A	DR	R	R	R	DR	DF	R	DR	DR	R	A'	R	A	F	F	DF	R
EQ1/2-b	S1	EQ	Y	Y	N	R	10,16	N	R	DR	R	R	R	DR	DF	DR	DR	DR	R	A^1	R	R	F	DF	F	R
	S1	EQ	Y	Y	N	R	10	N	R	R	R	R	R	DR	DF	R	R	R	R	A^1	R	R	F	F	F	R
FI1	FICR	FI	N	N	N	F	1	N	A	A	A	F	F	F	A	F	A	A	F	A	F	A	F	A	A	A
FI2	FIES2	FI	Y	Y	N	F	1	N	A	F	F	F	F	F	F	F	F	F	F	A^2	F	A	F	F	F	F
	FIES2	FI	Y,N	Y	N	F	1	N	A	F	F	F	F	F	F	F	F	F	F	A^2	F	A	F	F	F	F
FI3	FIT	FI	N	N	N	Y	V+X	N	A	F	F	F	F	DF	DF	F	F	F	F,A	A^3	F	A	F	A	A	F
	FIT	FI	N	N	N	Y	V+X	N	A	F	F,W	F	F	DF	DF	F	F,A	F,A	F,A	A^3	F	A	F	A	A	F
FI4	FIES3	FI	Y,N	Y	N	Y	V+X	N	F	F	F	F	F	DF		F	F	F	F	A^3	F	F	F	F	F	F
FI5-a	FIAC	FI	N	N	L	F	CF+X	N	A	F	F	F	F	DF		F	F	F	F	A^4	F	A	F	A	A	F
	FIZ	FI	N	N	L	F	CF+X	N	A	F	F,W	F	F	DF	DF	F	F,A	F,A	F,A	A^4	F	A	F	A	A	F
	FIAA	FI	N	N	L	F	CF+X+Y	N	A	DF	DF	F	F	DF	DF	F	DF	DF	F	A^4	F	A	F	A	DF	F
	FIAA	FI	N	N	L	F	CF+X+Y	N	A	DF	F,W	F	F	DF	DF	F,A	F,A	F,A	F	A^4	F	A	F	A	DF	F
FI5-b	FIN	FI	N	N	R	F	CF+X	N	A	F	F	F	F	DF	DF	F	F	F	F	A^4	F	A	F	A	F	F
	FIY1	FI	N	N	R	F	CF+X	N	A	F	F,W	F	F	DF	DF	F	F,A	F,A	F,A	A^4	F	A	F	A	A	F
	FIAA	FI	N	N	R	F	CF+X+Y	N	A	DF	DF	F	F	DF	DF	F	DF	DF	F	A^4	F	A	F	A	DF	F
	FIAA	FI	N	N	R	F	CF+X+Y	N	A	DF	F,W	F	F	DF	DF	F,A	F,A	F,A	F	A^4	F	A	F	A	DF	F
FI6-a	FIES3	FI	Y,N	Y	L	F	CF+X	N	F	F	F	F	F	DF	DF	F	F	F	F	A^4	F	F	F	F	F	F
FI6-b	FIES3	FI	Y,N	Y	R	F	CF+X	N	F	F	F	F	F	DF	DF	F	F	F	F	A^4	F	F	F	F	F	F
FL1	FL1	FL	N	N	N	A	1	N	A	A	A	F	F	F	F	F	F	A	F	A	F	A	F	A	A	F
FL2	FL2	FL	N	N	N	A	1	N	A	F	F	F	F	F	F	F	F	F	F	A	F	A	F	A	A	F
IA1-a	A123	A	N	N	L	F	1	1	A	F,W	F,W	A	A	DF	F	DF	DF	DF	DF	A^4	F	A	F	A	A	DF
	A123	A	N	N	L	F	1	N	A	F,W	F,W	A	A	DF	F	DF	DF	DF	DF	A^4	F	A	F	A	A	DF
IA1-b	A123	A	N	N	L	F	1	N	A	F,W	F,W	F	F	DF	F	DF	DF	DF	DF	A^4	F	A	F	A	A	DF
	A123	A	Y	N	L	F	1	N	A	F,W	F,W	F	F	DF	F	DF	DF	DF	DF	A^4	F	A	F	A	A	F

Table 2.5-1 (Continued)
Final LaSalle Plant Damage States

PDS	SEQ	ANAL	LOSP	SB	L/R	VENT	CDT	SRV	ADS	RHR	CSS	MFW	CDS	HPCS	RCIC	LPCS	LPCI	SPC	CRD	DDFW	PCS	DC	SBLC	A	B	SCS
IA2-a	A49	A	N	N	N	Y	1	1	A	F,W	F,W	A	A	DF	F	DF	DF	DF	DF	A[3]	F	A	F	A	A	DF
	A49	A	N	N	N	Y	1	N	A	F,W	F,W	A	A	DF	F	DF	DF	DF	DF	A[3]	F	A	F	A	A	DF
	A120	A	N	N	N	Y	1	N	A	F,W	F,W	F	A	DF	F	DF	DF	DF	DF	A[3]	F	A	F	A	A	DF
IA2-b	A120	A	N	N	N	Y	1	N	A	F,W	F,W	F	F	DF	F	DF	DF	DF	DF	A[3]	F	A	F	A	A	DF
	A120	A	Y	N	N	Y	1	N	A	F,W	F,W	F	F	DF	F	DF	DF	DF	DF	A[3]	F	A	F	A	A	DF
IA2	A49	A	N	N	N	Y	1	N	F	F,W	F,W	A	A	DF	F	DF	DF	DF	DF	A[3]	F	A	F	A	A	DF
IL1	L14	L	N	N	L	F	CF+X	N	A	F	F	F	F	DF	DF	F	F	F	DF	A[4]	F	A	F	A	A	F
	L14	L	N	N	L	F	CF+X	N	A	F	F,W	F	F	DF	DF	F	F,A	F,A	DF	A[4]	F	A	F	A	A	F
IT1	T100	T	N	N	N	A	1	N	A	F	F	F	F	F	A[5]	F	F	F	F	A	F	A	F	A	A	F
IT2-a	T100	T	Y	Y	N	R	1	N	A	F	F	R	R	F	R	F	F	F	F	A	R	A	R	F	F	F
IT2-b	T100	T	Y	Y	N	R	1	N	A	R	R	R	R	F	R[6]	R	R	R	R	A	R	A	R	F	F	R
	T100	T	Y	Y	N	R	1	N	A	R	R	R	R	R[6]	F	R	R	R	R	A	R	A	R	F	F	R
IT2-c	T101	T	Y	Y	N	R	1	N	R	R	R	R	R	R	R	R	R	R	R	A	R	R	R	F	F	R
IT3-a	T62	T	Y	Y	N	R	8	N	R	R	R	R	R	R	DF	R	R	R	R	A	R	R	R	F	F	R
IT3-b	T62	T	Y	Y	N	R	8	N	R	F	F	R	R	F	DF	F	F	F	F	A	F	R	1	F	F	F
IT3-c	T62	T	Y,N	Y	N	F	8	N	A	F	F	F	F	F	DF	F	F	F	F	A[2]	F	A	F	F	F	F
IT4-a	T62	T	Y	Y	N	R	10	N	A	DR	DR	R	R	DF	DF	R	DR	DR	R	A	R	A	R	F	DF	F
IT4-b	T62	T	Y	Y	N	R	10,16	N	R	DR	DR	R	R	DF	DF	DR	DR	DR	R	A	R	R	R	DF	F	R
	T62	T	Y	Y	N	R	10	N	R	R	R	R	R	DF	DF	R	R	R	R	A	R	R	R	F	F	R
IT5-a	T16	T	N	N	N	Y	V+X	N	A	F	F	F	A[7]	DF	DF	F	F	F	F	A[3]	F	A	F	A	A	F
	T16	T	N	N	N	Y	V+X	N	A	F	F,W	F	A[7]	DF	DF	F	F,A	F,A	F,A	A[3]	F	A	F	A	A	F
IT5-b	T16	T	N	N	N	Y	V+X	N	A	F	F	F	F	DF	DF	F	F	F	F,A	A[3]	F	A	F	A	A	F
	T16	T	N	N	N	Y	V+X	N	A	F	F,W	F	F	DF	DF	F,A	F,A	F,A	F,A	A[3]	F	A	F	A	A	F
IT5-c	T16	T	Y	N	N	Y	V+X	N	A	F	F	R	R	DF	DF	F,A	F	F	F,R	A[3]	R	A	F	A	A	F

Table 2.5-1 (Continued)
Final LaSalle Plant Damage States

PDS	SEQ	ANAL	LOSP	SB	L/R	VENT	CDT	SRV	ADS	RHR	CSS	MFW	CDS	HPCS	RCIC	LPCS	LPCI	SPC	CRD	DDFW	PCS	DC	SBLC	A	B	SCS
IT6-a	T41	T	N	N	N	F	SRV+X	N	A	F	F	F	DF	F	DF	A	F	F	F	A^2	F	A	F	A	F	F
	T50	T	N	N	N	F	SRV+X	N	A	F	F	F	F	F	DF	F	A	F	F	A^2	F	A	F	A	A	F
	T85	T	N	N	N	F	SRV+X	N	A	F,A	F,A	F	F	F,A	F	F	A	A	F	A^2	F	A	F	F	A	F
IT6-b	T41	T	N	N	N	F	SRV+X	N	A	F	W	F	DF	F	DF	A	A	A	F	A^2	F	A	F	A	F	F
IT6-c	T41	T	N	N	N	F	SRV+X	N	A	F	F	F	DF	F	DF	F	F	F	F	A^2	F	A	F	F	A	F
IT7-a	T85	T	Y	N	N	F	SRV+X	U	A	DF	DF	F	F	F	F	F	DF	DF	F	A^2	F	A	F	A	DF	F
IT7-b	T50	T	Y	N	N	R	SRV+X	N	A	F	W	R	R	R	DF	A	A	A	F	A^2	R	A	R	A	DF	R
IT8-a	T18	T	N	N	L	F	CF+X	N	A	F	F	F	A	DF	DF	F	F	F	DF	A^4	F	A	F	A	F	F
	T22	T	N	N	L	F	CF+X	N	A	F	F,W	F	A	DF	DF	F	F,A	F,A	F	A^4	F	A	F	F	A	F
	T18	T	N	N	L	F	CF+X	N	A	F,A	F,A	F	A	DF	DF	F	F,A	F,A	DF	A^4	F	A	F	F	A	F
IT8-b	T47	T	N	N	L	F	CF+X	N	A	F	F	F	F	F	DF	F	F,A	F	DF	A^4	F	A	F	F	A	F
	T47	T	N	N	L	F	CF+X	N	A	F	F,W	F	F	F	DF	F	F,A	F,A	DF	A^4	F	A	F	F	A	F
IT8-c	T20	T	N	N	R	F	CF+X	N	A	F	F	F	A^7	DF	DF	F	F	F	DF	A^4	F	A	F	A	A	F
	T24	T	N	N	R	F	CF+X	N	A	F	F,W	F	A^7	DF	DF	F	F,A	F,A	F	A^4	F	A	F	F	A	F
	T20	T	N	N	R	F	CF+X	N	A	F,A	F,A	F	A^7	DF	DF	F	F,A	F,A	DF	A^1	F	A	F	F	A	F
IT8-d	T20	T	N	N	R	F	CF+X	N	A	F	F,W	F	F	DF	DF	F,A	F,A	F,A	DF	A^4	F	A	F	A	A	F
IT9-a	T22	T	N	N	L	F	CF+X	N	F	F	F	F	A	DF	DF	F	F	F	F	A^4	F	A	F	F	A	F
IT9-b	T24	T	N	N	R	F	CF+X	N	F	F	F	F	A^7	DF	DF	F	F	F	F	A^4	F	A	F	F	A	F
IT10-a	T22	T	Y	N	L	F	CF+X	N	A	F	F,W	R	R	DF	DF	F,A	F,A	F,A	F,R	A^4	R	A	F	A	F	F
	T22	T	Y	N	L	F	CF+X	N	A	F,R	F,R	R	R	DF	DF	F	F,R	F,R	F,R	A^4	R	A	F	A	F	F,R
IT10-b	T24	T	Y	N	R	F	CF+X	N	A	F	F,W	R	R	DF	DF	F,A	F,A	F,A	F,R	A^4	R	A	F	A	F	F
	T24	T	Y	N	R	F	CF+X	N	A	F,R	F,R	R	R	DF	DF	F	R	F,R	F,R	A^4	R	A	F	[illegible]	F	F,R
IT11-a	T22	T	Y	Y	L	F	CF+X	N	R	F,R	F,R	R	R	DF	DF	F	F,R	F,R	F,R	A	R	A^8	F	F	F	F,R
IT11-b	T24	T	Y	Y	R	F	CF+X	N	R	F,R	F,R	R	R	[illegible]	DF	F	F,R	F,R	F,R	A	R	A^8	F	F	F	F,R
ITL1	TL97	TL	N	N	N	F	1	1	A	F	F	F	F	F	F	F	F	F	F	A^2	F	A	A	A	A	F

Table 2.5-1 (Concluded)
Final LaSalle Plant Damage States

PDS	SEQ	ANAL	LOSP	SB	L/R	VENT	CDT	SRV	ADS	RHR	CSS	MFW	CDS	HPCS	RCIC	LPCS	LPCI	SPC	CRD	DDFW	PCS	DC	SBLC	A	B	SCS
ITL2-a	TL97	TL	Y	Y	N	R	1	1	A	F	F	R	R	F	F	F	F	F	F	A	R	A	R	F	F	F
ITL2-b	TL97	TL	Y	Y	N	R	1	1	A	R	R	R	R	F	F	R	R	R	F	A	R	A	R	F	F	R
	TL97	TL	Y	Y	N	R	1	1	A	R	R	R	R	R[9]	F	R	R	R	R[9]	A	R	A	R	F	F	R
ITL3	TL59	TL	Y	Y	N	R	8	1	R	F	F	R	R	F	DF	F	F	F	F	A	R	A	R	F	F	F
ITL4	TL59	TL	N	N	N	F	8	1	A	F	F	F	F	F	DF	F	F	F	F	A[2]	F	A	R	F	A	F
ITL5-a	TL59	TL	Y	Y	N	R	10	1	A	DR	DR	5	R	DR	DF	R	DR	DR	R	A	R	R	R	F	DF	R
ITL5-b	TL59	TL	Y	Y	N	R	10,16	1	R	DR	DR	R	R	DR	DF	DR	DR	DR	R	A	R	R	R	DF	F	R
	TL59	TL	Y	Y	N	R	10	1	R	R	R	R	R	DR	DF	R	R	R	R	A	R	R	R	F	F	R
ITL6	TL12	TL	N	N	N	Y	V+X	1	A	F	F	F	F	DF	DF	F	F,A	F	F,A	A[3]	F	A	F	A	A	F
	TL12	TL	N	N	N	Y	V+X	1	A	F	F,W	F	F	DF	DF	F	F,A	F,A	F,A	A[3]	F	A	F	A	A	F

1. No ADS, RPV at high pressure.

2. High containment pressure results in ADS reclosure, DDFW fails.

3. Containment venting before DDFW hookup, severe environment prevents use.

4. Containment failure before DDFW hookup, severe environment prevents use.

5. Error, system is failed.

6. Simplify by treating R as F, so many system on recovery of AC that does not matter.

7. Given credit in Level I analysis, no additional credit here as approaches random failure probability.

8. Error, system recoverable on AC recovery in this time frame.

9. Simplify by treating R as F, so many systems on recovery of AC that does not matter.

4. Once the final ordering of the questions was determined, all similar PDSs were removed from the list (i.e., many sequences had PDSs that were identical to those in another sequence). This reduced set of unique PDSs was then resorted in order to group PDSs with similar characteristics together.

5. Starting with the first question, the PDSs were grouped and subgrouped depending on whether or not the first n questions had similar answers. There were still too many PDSs remaining. The answers to the questions were examined further and, for cases where the answers to subsequent questions would not be relevant or impact significantly the accident evolution given the answers to prior questions, the PDSs were formed into larger groups.

The result was the definition of 53 PDSs and sub-PDSs (i.e., a sub-PDS results when a distinction is made in the case structure of one of the initial questions in the APET, see question 2, case 4 where PDS IT5 is split into sub-PDSs IT5-c and IT5-a+IT5-b, this distinction was not considered significant enough to be worth defining a completely different PDS) as shown in Table 2.5-1. All of these were not used in the final analysis. Because some systems were already given credit for recovery in the Level I analysis and because severe environments occurred in the reactor building and to some extent in the turbine building after containment failure, the following sub-PDSs were combined: IT5-a and IT5-b, and IT8-c and IT8-d. Also, the CDS and DDFW were changed from available to failed as noted in the table. The following PDSs, which differ only in whether or not the containment failed by leak or rupture, were combined since the LHS sample would specify the exact containment failure mode so this does not need to be explicitly called out in the PDS definition: FI5-a and FI5-b, FI6-a and FI6-b, IT8-b with IT8-c and IT8-d, IT9-a and IT9-b, IT10-a and IT10-b, and IT11-a and IT11-b. For the final analysis, 30 PDSs and 16 sub-PDSs were used. Because of the large number of PDSs defined in this analysis, each PDS is not described in detail. The characteristics of the individual PDSs can be obtained by an examination of Table 2.5-1. Summary groups are defined later and these are tracked through the analysis.

The ordering of the questions in the APET as described in Section 2.3 is different from the order used in Table 2.5-1. This is because the order in the APET conforms roughly to the time order of events (i.e., initiating event, high pressure injection, depressurization, low-pressure injection, containment heat removal, containment failure, and core damage) while the order in Table 2.5-1 was selected for purposes of defining similar groups. It should be noted that the SEQ, SPC, DDWF, PCS, A, B, and SCS questions were not used in the final definition. Either it was not necessary to carry the information through the analysis (SEQ) or the answer was determined by other questions that were used (A, B, SPC, DDFW, SCS, PCS).

2.6 Construction of Level I LHS Sample For Level II/III Analysis

In order to construct an integrated LHS sample for the Level II/III analysis, the important variables for the Level I analysis have to be selected. The following procedure was used to construct the Level I portion of the final LHS sample for the Level II/III analysis:

1. The cut sets from the sequences were regrouped into plant damage states and sub-PDSs by editing the TEMAC cut set file for each sequence. The cut sets were moved from the accident sequence files to new files for each PDS and sub-PDS. Because of the large number of cut sets and the level of detail used to define the PDSs, it is not possible to give a listing of which cut sets went where. In some cases, in sequences with thousands of cut sets, every other cut set went to a different PDS. This information is, therefore, available only on marked up computer output and in the handmade tables used to classify the cut sets.

2. After regrouping the cut sets from the accident sequences into PDSs, a full uncertainty calculation was performed for each PDS and/or sub-PDS using the TEMAC code.

3. The list of events appearing in the cut sets for the uncertainty importance measure for each PDS or sub-PDS was put in a file and all events contributing greater than or equal to a 5% reduction in the uncertainty in the log of the core damage frequency for any PDS or sub-PDS were retained for the uncertainty evaluation. In some cases, variables contributing less than 5% were retained, e.g., if no variable contributed at least 5% or if only a few variables contributed to the PDS. All other events appearing in the cut sets were frozen at their mean values.

4. This file was then sorted and duplicates removed.

5. For all events appearing in the final list, the primary LHS variables were identified. In the TEMAC input files, all other primary LHS variables were frozen at their mean values.

6. Full uncertainty calculations were again run on each PDS or sub-PDS using TEMAC and the results of this limited sample were compared to the initial calculation with all variables being sampled. If the resulting top event uncertainty distribution was similar to the original distribution then the process was complete. If the distributions were not similar, then additional variables representing events for that PDS were added to the LHS sample. The process was repeated until an accurate representation of the initial distributions was obtained for all PDSs and sub-PDSs.

Table 2.6-1 contains a complete list of the Level I variables selected for inclusion in the final Level II/III LHS sample. The table contains: the

Table 2.6-1
Variables Sampled in the Accident Frequency Analysis

Variable & Case	LHS #	Range	Distribution	Correlation	Correl. with	Description
CFM-37*6594	1	9.9E-5 0.57	Lognormal M=1.99E-2	Rank 1	5,6,9 10,14	Relay coil failure to energize and remain energized. Test interval 18 months and 24 hr. operation.
CFM-30*24	2	3.6E-7 2.0E-3	Lognormal M=7.1E-5	Rank 1	3	Circuit breaker failure to remain open or closed. 24 hr. operation.
CFM-30*28	3	4.2E-7 2.4E-3	Lognormal M=8.3E-5	Rank 1	2	Circuit breaker failure to remain open or closed. 24 hr. operation and 8 hr exposure time.
CFM-114	4	1.2E- 7.1E-2	Lognormal M=2.5E-3	None		Component control circuit failure to operate on demand (MOV, pump, circuit breaker, fan, damper).
CFM-37*192	5	2.9E-6 1.6E-2	Lognormal M=5.6E-4	Rank 1	1,6,9 10,14	Relay coil failure to energize. Test interval 2 weeks and 24 hr operation.
CFM-37*168	6	2.5E-6 1.4E-2	Lognormal M=4.8E-4	Rank 1	1,5,9 10,14	Relay coil failure to energize. Test interval 2 weeks.
M1	7	1.5E-5 8.5E-2	Lognormal M=2.9E-3	None		Motor-driven pump unavailable due to maintenance (safety pumps).
M18	8	1.2E-4 1.7E-3	Lognorm- M=5.0E-	None		Fan motor unavailable due to maintenance.
CFM-37*384	9	6.0E-6 3.4E-2	Lognormal M=1.2E-3	Rank 1	1,5,6 10,14	Relay coil failure to energize and remain energized. Test interval of 1 month and 24 hr operation.

Table 2.6-1 (Continued)
Variables Sampled in the Accident Frequency Analysis

Variable & Case	LHS #	Range	Distri-bution	Corre-lation	Correl. with	Description
CFM-37*1119	10	1.7E-5 9.6E-2	Lognormal M=3.4E-3	Rank 1	1,5,6,9 14	Relay coil failure to energize and remain energized. Test interval 3 months and 24 hr operation.
M2	11	1.5E-5 8.5E-2	Lognormal M=2.9E-3	None		Motor-driven pump unavailable due to maintenance (non-safety pump).
CFM-3	12	1.5E-4 0.85	Lognormal M=2.9E-2	None		Turbine-driven pump failure to operate on demand.
CFM-57*1095	13	3.1E-5 0.17	Lognormal M=6.0E-3	None		Heat exchanger fails due to blockage. Test interval 3 months.
CFM-37*13140	14	5.0E-4 0.75	Lognormal M=3.9E-2	Rank 1	1,5,6,9 10	Relay coil failure to energize and remain energized. Test interval 3 yrs.
T1	15	0.46 28.3	Lognormal M=4.5	None		Turbine Trip With Bypass initiating event frequency.
T5	16	0.061 3.8	Lognormal M=6.0E-1	None		Total Loss of Feedwater initiating event frequency.
T9-10	17	5.1E-4 3.1E-2	Lognormal M=5.0E-3	None		Initiating event frequency for the loss of a specific 4160 VAC or 125 VDC electrical bus
T11-12	18	3.0E-4 1.9E-2	Lognormal M=3.0E-3	None		Initiating event frequency for loss of instrument air or drywell pneumatic system.
BATT-BETA	19	4.1E-3 0.25	Lognormal M=4.0E-2	None		Conditional probability of common cause failure of batteries given one has failed.

Table 2.6-1 (Continued)
Variables Sampled in the Accident Frequency Analysis

Variable & Case	LHS #	Range	Distri-bution	Corre-lation	Correl. with	Description
H3*M18A* 13140	20	6.9E-6 4.0E-2	Lognormal M=1.4E-3	None		Probability of operator failure to restore fan correctly after maintenance. Test interval 3 yrs.
LPCI-MDP -BETA	21	0.011 0.69	Lognormal M=0.11	None		Conditional probability of common cause failure of LPCI pumps given one has failed.
CFM-31	22	2.5E-3 0.16	Lognormal M=2.5E-2	None		Diesel generator failure to start on demand.
CFM-32*8	23	9.4E-5 0.54	Lognormal M=1.9E-2	None		Diesel generator failure to run. 8 hr. operating time.
CFM-73	24	3.0E-5 1.9E-3	Lognormal M=3.0E-4	None		Fan motor failure to start on demand.
CFM-74*24	25	2.4E-5 1.5E-3	Lognormal M=2.4E-4	None		Fan motor failure to run. 24 hr operating time.
Q1	26	4.0E-4 8.0E-2	Lognormal M=8.1E-3	Rank 1	27	Probability of one SRV failing to reclose after transient initiator.
Q2	27	4.7E-7 2.7E-3	Lognormal M=9.3E-5	Rank 1	26	Probability of two SRVs failing to reclose after transient initiator.
RA-G1	28	2.7E-5 4.0E-2	Lognormal M=2.1E-3	None		Recovery action from group 1: normal operation of system or component to control a critical parameter prior to automatic initiation.

Table 2.6-1 (Continued)
Variables Sampled in the Accident Frequency Analysis

Variable & Case	LHS #	Range	Distribution	Correlation	Correl. with	Description
RA-G11	29	7.9E-6 4.5E-2	Lognormal M=1.5E-3	None		Recovery action from group 11: Local operation of manually controlled components normally operated from the control room when control room operation fails.
RA-G3	30	1.3E-5 7.4E-2	Lognormal M=2.5E-3	None		Recovery action from group 3: Manual operation of systems or components which failed to automatically operate.
BATT-FTDP	31	1.8E-6 1.0E-2	Lognormal M=3.5E-4	None		Battery failure to deliver power. 1 month test interval.
RA-G2	32	1.1E-5 6.2E-2	Lognormal M=2.1E-3	None		Recovery action from group 2: Use of low pressure systems when high pressure systems are unavailable.
RA-DGHW-27	33	0.04 1.00	Max. Entropy M=0.40	None		Failure of operators to recover diesel generator hardware failure within 27 hrs.
RPS	34	5.0E-8 2.8E-4	Lognormal M=1.0E-5	None		Failure of the reactor protection system to scram the reactor after a transient or LOCA initiating event.
OPFAILSMFW -8M	35	0.10 1.00	Max. Entropy M=0.5	None		Operator failure to control main feedwater level during an ATWS event to a level consistent with make-up from the CST tank.
RA-3-12-80M	36	1.7E-5 9.9E-2	Lognormal M=3.4E-3	None		Recovery action from group 12: manual override of a false control signal when no direct indication exists that signal is false. 80 minutes to correct.

Table 2.6-1 (Continued)
Variables Sampled in the Accident Frequency Analysis

Variable & Case	LHS #	Range	Distri-bution	Corre-lation	Correl. with	Description
RA-3-12-68M	37	8.9E-5 0.51	Lognormal M=1.8E-2	None		Recovery action from group 12: manual override of a false control signal when no direct indication exists that signal is false. 68 minutes to correct.
DGX-GEN-CC -FTR	38	1.4E-5 7.9E-2	Lognormal M=2.7E-3	None		Diesel generator control circuit failure to run. 8 hour run time.
FMS-LF	39	1.2E-4 2.?E-3	User Dist. M=5.4E-4	None		Local fault of a fan to start and run for 24 hours. Combination of CFM-73 and CFM-74.
RA-G-10	40	0.01 1.00	Max. Entropy M=0.09	None		Recovery action from group 10: Request to use last line of "Garbage" systems for level control.
MFWRF	41	5.0E-3 5.0E-2	Max. Entropy M=0.01	Rank 1	42-47	Random failure of main feedwater used in ATWS severe environment recovery event SUR-001-A-L.
IE-SEI-L1	42	3.1E-7 3.0E-2	Lognormal M=5.1E-4	Rank 1	41 43-47	Initiating event frequency for an earthquake of seismic hazard level L1.
IE-SEI-L2	43	9.3E-8 1.8E-2	Lognormal M=2.6E-4	Rank 1	41,42 44-47	Initiating event frequency for an earthquake of seismic hazard level L2.
IE-SEI-L3	44	2.6E-8 1.1E-2	Lognormal M=1.4E-4	Rank 1	41-43 45-47	Initiating event frequency for an earthquake of seismic hazard level L3.
IE-SEI-L4	45	4.9E-9 7.8E-3	Lognormal M=8.1E-5	Rank 1	41-44 46,47	Initiating event frequency for an earthquake of seismic hazard level L4.

Table 2.6-1 (Continued)
Variables Sampled in the Accident Frequency Analysis

Variable & Case	LHS #	Range	Distribution	Correlation	Correl. with	Description
IE-SEI-L5	46	6.4E-10 5.2E-3	Lognormal M=4.0E-5	Rank 1	41-45 47	Initiating event frequency for an earthquake of seismic hazard level L5.
IE-SEI-L6	47	9.0E-11 4.3E-3	Lognormal M=2.8E-5	Rank 1	41-46	Initiating event frequency for an earthquake of seismic hazard level L6.
IE-SEI-LL1	48	1.8E-6 6.8E-2	Lognormal M=1.4E-3	None		Initiating event frequency for an earthquake of seismic hazard level LL1.
LOSP-LL1	49	1.0E-8 1.0	User Dist. M=0.24	None		Conditional probability of LOSP resulting from ceramic insulator failure from seism of level LL1.
LOSP-L1	50	4.8E-8 1.0	User Dist. M=0.29	None		Conditional probability of LOSP resulting from ceramic insulator failure from seism of level L1.
LOSP-L2	51	3.2E-7 1.0	User Dist. M=0.36	None		Conditional probability of LOSP resulting from ceramic insulator failure from seism of level L2.
LOSP-L3	52	1.6E-6 1.0	User Dist. M=0.44	None		Conditional probability of LOSP resulting from ceramic insulator failure from seism of level L3.
LOSP-L4	53	5.9E-6 1.0	User Dist. M=0.50	None		Conditional probability of LOSP resulting from ceramic insulator failure from seism of level L4.
LOSP-L5	54	2.6E-5 1.0	User Dist. M=0.58	None		Conditional probability of LOSP resulting from ceramic insulator failure from seism of level L5.
LOSP-L6	55	1.2E-4 1.0	User Dist. M=0.66	None		Conditional probability of LOSP resulting from ceramic insulator failure from seism of level L6.

Table 2.6-1 (Continued)
Variables Sampled in the Accident Frequency Analysis

Variable & Case	LHS #	Range	Distribution	Correlation	Correl. with	Description
RN-CFP	56	2.3E-4 1.0	Uniform M=0.50	None		Random number, 0-1, used in the extender code to calculate the containment failure pressure.
RN-CF-MOD	57	1.6E-4 1.0	Uniform M=0.50	None		Random number, 0-1, used in the extender code to calculate the containment failure mode at a given pressure.
CONT-LEAK	58	0.0 1.0	User Dist. M=0.74	None		The conditional probability that the containment fails by leak (as opposed to rupture). Calculated in extender code from containment failure mode.
LEAKTRB	59	0.0 1.0	User Dist. M=0.21	None		The conditional probability that the containment fails by leakage to the reactor building (as opposed to rupture or leakage to the refueling floor). Calculated in extender code.
RUPTUTETRB	60	0.0 1.0	User Dist. M=0.22	None		The conditional probability that the containment fails by rupture to the reactor building (as opposed to leak or rupture to the refueling floor). Calculated in the extender code.
IE-LOSP	61	1.4E-3 0.44	User Dist. M=9.5E-2	None		Initiating event frequency for Loss of Offsite power.
SUR-001-L	62	0.0 1.0	User Dist. M=0.21	None		Severe environment failure of CRD after containment failure by leak to the reactor building.

Table 2.6-1 (Continued)
Variables Sampled in the Accident Frequency Analysis

Variable & Case	LHS #	Range	Distri-bution	Corre-lation	Correl. with	Description
SUR-002-L	63	0.0 1.0	User Dist. M=0.16	None		Severe environment failure of HPCS after containment failure by leak to the reactor building.
SUR-003-L	64	0.0 1.0	User Dist. M=0.16	None		Severe environment failure of HPCS and CRD after containment failure by leak to the reactor building.
SUR-005-L	65	0.0 3.6E-2	User Dist. M=5.0E-4	None		Severe environment failure of HPCS, CRD, and CDS after containment failure by leak to the reactor building.
SUR-006-L	66	0.00 3.6E-2	User Dist. M=5.1E-4	None		Severe environment failure of HPCS and CDS after containment failure by leak to the reactor building.
SUR-001-R	67	0.00 0.48	User Dist. M=1.5E-2	None		Severe environment failure of CRD, LPCS, and DDFW after containment failure by rupture to the reactor building.
SUR-002-R	68	0.00 0.49	User Dist. M=1.6E-2	None		Severe environment failure of LPCS and DDFW after containment failure by rupture to the reactor building.
SUR-003-R	69	0.00 0.48	User Dist. M=1.5E-2	None		Severe environment failure of CRD, LPCI, and DDFW after containment failure by rupture to the reactor building.

Table 2.6-1 (Continued)
Variables Sampled in the Accident Frequency Analysis

Variable & Case	LHS #	Range	Distri-bution	Corre-lation	Correl. with	Description
SUR-005-R	70	0.00 0.49	User Dist. M=1.6E-2	None		Severe environment failure of LPCI and DDFW after containment failure by rupture to the reactor building.
SUR-007-R	71	0.00 2.4E-2	User Dist. M=3.2E-4	None		Severe environment failure of CRD, CDS, LPCI, and LPCS after containment failure by rupture to the reactor building.
SUR-011-R	72	0.00 2.4E-2	User Dist. M=3.4E-4	None		Severe environment failure of CDS, LPCI, and DDFW after containment failure by rupture to the reactor building.
SUR-014-R	73	0.00 1.00	User Dist. M=8.4E-2	None		Severe environment failure of CRD and ADS or CDS and DDFW after containment failure by rupture to the reactor building.
SUR-015-R	74	0.00 1.00	User Dist. M=8.4E-2	None		Severe environment failure of CRD and ADS or CDS, DDFW, and LPCI after containment failure by rupture to the reactor building.
SUR-018-R	75	0.00 1.00	User Dist. M=9.5E-2	None		Severe environment failure of CRD and ADS or DDFW, LPCI, and LPCS after containment failure by rupture to the reactor building.
SUR-021-R	76	0.00 1.00	User Dist. M=8.4E-2	None		Severe environment failure of HPCS, CRD, and ADS or CDS and DDFW after containment failure by rupture to the reactor building.

Table 2.6-1 (Continued)
Variables Sampled in the Accident Frequency Analysis

Variable & Case	LHS #	Range	Distri-bution	Corre-lation	Correl. with	Description
SUR-022-R	77	0.00 1.00	User Dist. M=9.6E-2	None		Severe environment failure of HPCS, CRD, and ADS or DDFW after containment failure by rupture to the reactor building.
SUR-023-R	78	0.00 1.00	User Dist. M=8.4E-2	None		Severe environment failure of HPCS and ADS or CDS and DDFW after containment failure by rupture to the reactor building.
SUR-024-R	79	0.00 1.00	User Dist. M=9.6E-2	None		Severe environment failure of HPCS and ADS or DDFW after containment failure by rupture to the reactor building.
SUR-025-R	80	0.00 0.50	User Dist. M=1.9E-2	None		Severe environment failure of HPCS, CRD, and CDS after containment failure by rupture to the reactor building.
SUR-027-R	81	0.00 0.51	User Dist. M=1.9E-2	None		Severe environment failure of HPCS and DDFW after containment failure by rupture to the reactor building.
SUR-005-V	82	1.1E-5 4.9E-2	User Dist. M=2.1E-3	None		Severe environment failure of HPCS, and ADS or CDS and DDFW after containment venting.
SUR-006-V	83	9.8E-3 0.66	User Dist. M=8.7E-2	None		Severe environment failure of HPCS and ADS or DDFW after containment venting.
SUR-001-A-V	84	1.3E-3 1.00	User Dist. M=0.61	None		Severe environment failure of MFW and HPCS after containment venting during ATWS scenarios.

Table 2.6-1 (Continued)
Variables Sampled in the Accident Frequency Analysis

Variable & Case	LHS #	Range	Distri-bution	Corre-lation	Correl. with	Description
RA-8-1H	85	0.11 0.54	LOSP M=0.25	Rank 1	85-90 284-292	Failure to restore offsite power within 1 hour.
RA-8-80M	86	7.6E-2 0.48	LOSP M=0.20	Rank 1	85-90 284-292	Failure to restore offsite power within 80 minutes.
RA-8-8H	87	1.5E-3 0.18	LOSP M=2.7E-2	Rank 1	85-90 284-292	Failure to restore offsite power within 8 hours.
RA-8-10H	88	5.4E-4 0.16	LOSP M=2.0E-2	Rank 1	85-90 284-292	Failure to restore offsite power within 10 hours.
RA-8-23H	89	8.9E-6 0.10	LOSP M=4.5E-3	Rank 1	85-90 284-292	Failure to restore offsite power within 23 hours.
RA-8-27H	90	5.3E-6 9.2E-2	LOSP M=3.2E-3	Rank 1	85-90 284-292	Failure to restore offsite power within 27 hours.
R-OP-1	91	6.4E-3 0.64	Max. Entropy M=0.064	None		Probability that the operators will not successfully recover the plant from the remote shutdown panel after a control room fire.
FAE	92	7.6E-4 1.9E-2	Max. Entropy M=3.8E-3	None		Area fire ratio of fire in area E to that of turbine building.
FA1	93	0.06 1.00	Max. Entropy M=0.3	None		Area ratio within fire area E where a large fire can damage critical equipment.

Table 2.6-1 (Continued)
Variables Sampled in the Accident Frequency Analysis

Variable & Case	LHS #	Range	Distri-bution	Corre-lation	Correl. with	Description
FS1	94	0.01 0.30	Max. Entropy M=0.17	None		Severity ratio for a large fire in are E.
FAT	95	1.4E-2 0.34	Max. Entropy M=0.068	None		Area fire ratio of fire in area T to that of auxiliary building.
LAMBDA-CR	96	0.00 0.54	FIRE M=3.4E-3	None		Initiating event frequency for a fire starting in the control room.
LAMBDA-TB	97	1.0E-5 0.6	FIRE M=3.7E-2	None		Initiating event frequency for a fire starting in the turbine building.
LAMBDA-CSR	98	0.00 0.42	FIRE M=9.9E-3	None		Initiating event frequency for a fire starting in the cable spreading room (Area N).
LAMBDA-AUX	99	1.0E-5 0.6	FIRE M=5.3E-2	None		Initiating event frequency for a fire starting in the auxiliary building.
LAMBDA-SVGR	100	0.00 0.20	FIRE M=1.4E-2	None		Initiating event frequency for a fire starting in a switchgear room.
IE-VALVE-RUP	101	1.9E-7 1.1E-3	Lognormal M=3.8E-5	None		Internal flood initiator from break in valve in service water pipe on reactor building level 3G.
OP-FTISOL -FLOOD	102	0.022 0.30	Lognormal M=9.0E-2	None		Operator fails to isolate internal flood within 7.3 minutes.

Table 2.6-1 (Concluded)
Variables Sampled in the Accident Frequency Analysis

Variable & Case	LHS #	Range	Distri-bution	Corre-lation	Correl. with	Description
PIPE-FREQ	103	3.2E-11 0.01	User Dist M=3.[illegible]-6	None		Frequency per year of low pressure pipe failure resulting in an internal flood (per 100 feet of pipe).

1. User Dist. - Using results of expert or internal elicitation, perform some secondary calculations and gererate final distribution. Performed in LHS extender code or user distribution subroutine.
2. LOSP - Uses the LOSP codes developed for the Level I analysis.
3. FIRE - Uses the LOSP codes as in 2. but for the fire data base.

name of the variable, the number of that variable in the LHS sample, the general range of values that the variable can take, which distribution is used to model the variable and the mean value, the amount of correlation with other variables, which other variables the variable is correlated with, and a description of the variable. A detailed discussion of the setup of the Level I LHS sample is presented in Chapter 5 of Volume 2 of the LaSalle Level I report.

It is important to note at this point that, even though the same random seed was used to generate the Level II/III LHS sample as was used to generate the Level I sample, the distributions of the top level events will not be exactly the same. The numbers generated for the distribution of any individual variable that appeared in the original Level I LHS sample will be the same in the Level II/III LHS sample since the same random seed was used. However, the LHS code reorders the values of the various variables so that random correlations are minimized. If the number of variables or the order of the variables are changed (which they are in this analysis), then different values from the distributions of the individual variables will be matched up in a sample and the values obtained for any combined event will be different.

2.7 PDS Uncertainty Results

In this section, the results of the TEMAC uncertainty analyses of the plant damage states are discussed. Table 2.7-1 lists the 30 PDSs used in the Level II/III analysis, their core damage frequency distribution statistics as calculated from the reduced sample, and the integrated results showing the total core damage frequency from all events and all plant damage states combined. Table 2.7-2 lists the 30 PDSs used and their conditional core damage probability distribution statistics relative to the total core damage frequency.

Because of the large number of plant damage states and sub-plant damage states, it would not be efficient to track and discuss each of these in detail in this report. Table 2.7-3 shows the core damage frequency for two sets of summary groups which will be used in this analysis. The first set divides the PDS into seismic, fire, flood, and internal groups for each of the unique analyses done in the Level I analysis. The second set keeps the general externally initiated groups but divides the internally initiated accidents into some subgroups of interest: LOCAs, transient-induced LOCAs, transients, and ATWS. The results for the different stages of the Level II/III analysis will be presented in terms of this second summary group. More detailed results for each plant damage state can be obtained from the output files but will not be discussed here.

Figure 2.7-1 shows the total Level I core damage frequency distribution from all causes. The figure contains a plot of the cumulative distribution function (CDF) and an approximate density plot in the form of a histogram where the height of each bar is determined by the fraction of LHS

Table 2.7-1
LaSalle Plant Damage State Frequency Distributions

PDS NO.	PDS	5TH PCT	MEDIAN	95TH PCT	MEAN
1	EQ1	0.189E-11	0.311E-09	0.427E-07	0.230E-07
2	EQ2	0.717E-10	0.307E-07	0.140E-05	0.674E-06
3	FI1	0.140E-12	0.172E-11	0.813E-05	0.370E-04
4	FI2	0.184E-08	0.303E-07	0.430E-06	0.110E-06
5	FI3	0.436E-07	0.714E-06	0.858E-05	0.216E-05
6	FI4	0.508E-08	0.116E-06	0.191E-05	0.496E-06
7	FI5	0.000E+00	0.000E+00	0.388E-04	0.157E-04
8	FI6	0.000E+00	0.000E+00	0.120E-07	0.191E-08
9	FL1	0.109E-09	0.318E-08	0.728E-06	0.175E-06
10	FL2	0.111E-06	0.109E-05	0.112E-04	0.298E-05
11	IA1	0.000E+00	0.000E+00	0.819E-07	0.169E-07
12	IA2	0.114E-08	0.273E-07	0.497E-06	0.170E-06
13	IL1	0.000E+00	0.000E+00	0.567E-07	0.117E-07
14	IT1	0.225E-07	0.393E-06	0.617E-05	0.145E-05
15	IT2	0.830E-06	0.697E-05	0.977E-04	0.282E-04
16	IT3	0.245E-06	0.153E-05	0.151E-04	0.419E-05
17	IT4	0.121E-07	0.304E-06	0.102E-04	0.462E-05
18	IT5	0.147E-07	0.154E-06	0.156E-05	0.401E-06
19	IT6	0.376E-09	0.665E-08	0.288E-06	0.159E-06
20	IT7	0.682E-10	0.193E-08	0.186E-06	0.396E-07
21	IT8	0.000E+00	0.000E+00	0.217E-04	0.435E-05
22	IT9	0.000E+00	0.000E+00	0.728E-06	0.150E-06
23	IT10	0.000E+00	0.000E+00	0.101E-05	0.191E-06
24	IT11	0.000E+00	0.000E+00	0.149E-05	0.311E-06
25	ITL1	0.147E-09	0.333E-08	0.889E-07	0.257E-07
26	ITL2	0.226E-08	0.296E-07	0.621E-06	0.165E-06
27	ITL3	0.306E-10	0.866E-09	0.270E-07	0.713E-08
28	ITL4	0.709E-10	0.178E-08	0.432E-07	0.108E-07
29	ITL5	0.325E-09	0.468E-08	0.137E-06	0.441E-07
30	ITL6	0.284E-08	0.381E-07	0.848E-06	0.230E-06
TOTAL		0.574E-05	0.276E-04	0.325E-03	0.104E-03

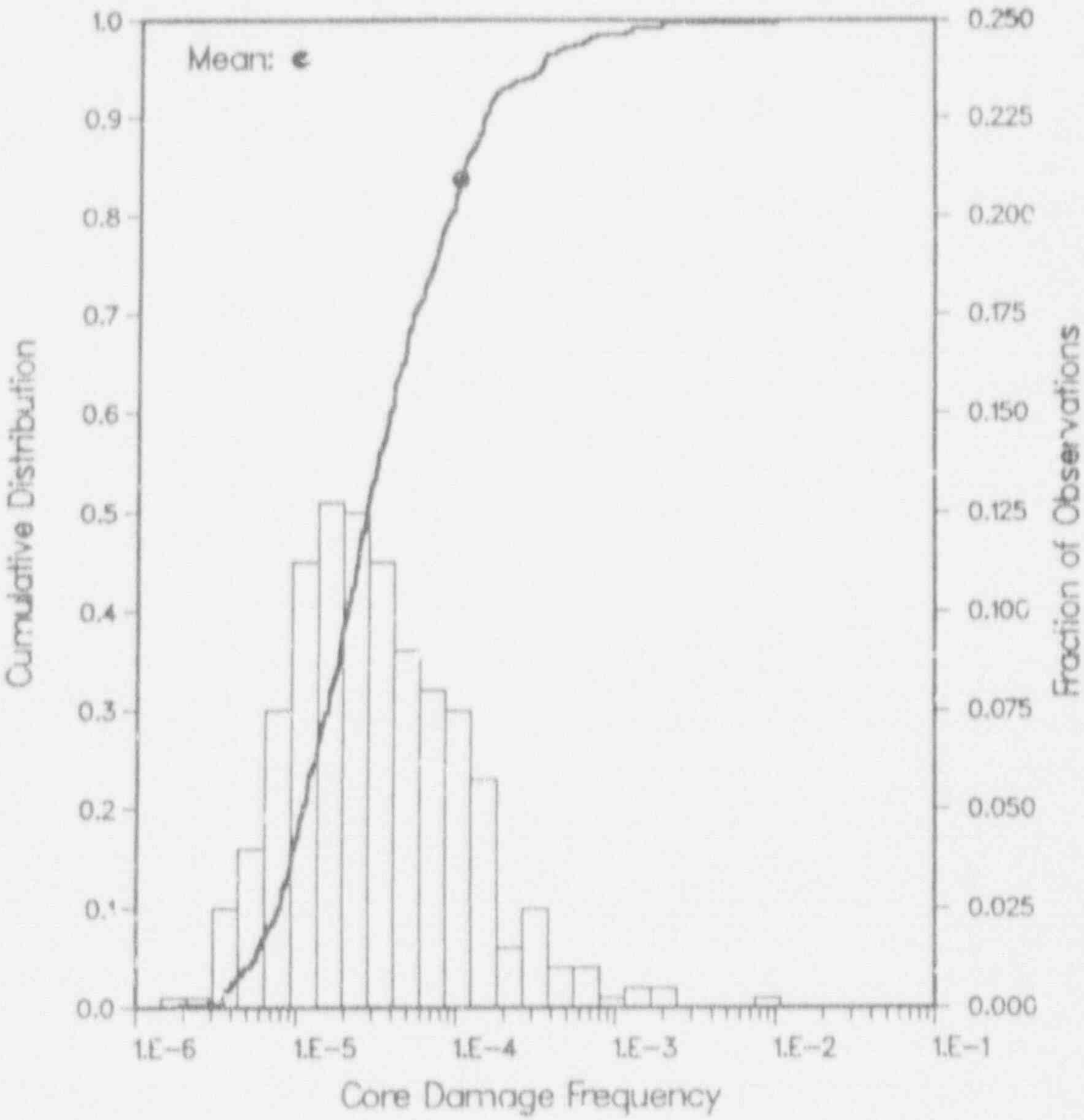

Figure 2.7-1 LaSalle Total Core Damage Frequency From All Initiators

observations that occurred in each interval. Figure 2.7-2 shows the CDFs for the first set of summary groups, described in the preceding paragraph, representing each of the individual Level I analyses. One can see from this figure that the total distribution is very close to the internal event distribution and that, overall, internal events dominate the results. However, the fire distribution turns over at the upper end and has some very high values at the end of the distribution. This occurs because the fire data is very scarce and the fire initiating event distributions are very wide with long tails. This affects the mean value of the fire results significantly. In fact, the mean result for fire is very close to that for internal events even though the bulk of the fire distribution is significantly below that for the distribution for internal initiators. For LaSalle, seismic events can be seen to make only a minor contribution to the total core damage frequency.

Table 2.7-4 shows the fractional contribution of each of the summary PDS groups in the two sets to the total core damage frequency calculated in two different ways. Section 6.3 contains a detailed description of the significance of the two different methods of calculating fractional contribution. MFCCD is the mean fractional contribution to core damage and FCMCD is the fractional contribution to mean core damage. For MFCCD the averaging over the LHS observations is done after the ratio of group core damage frequency to total core damage frequency is formed and for FCMCD the averaging over the LHS observations is done before the ratio of group core damage frequency to total core damage frequency is formed.

Notice that substantial differences in the fractional contribution exist between the two methods. The MFCCD tends to represent contributions from the whole distribution while the FCMCD tends to emphasize contributions from the upper tail of the distributions. In particular, the fire distribution has a very long upper tail and, from the table, one can see its contribution increases dramatically for the FCMCD measure. The mean core damage frequency is clearly dominated by the fire and internal transient groups. Internal floods also contribute. Seismic, ATWS, LOCAs, and transient-induced LOCAs contribute very little to the mean core damage frequency. One must remember, however, that the contribution to risk from seismic and ATWS events, in particular, can be much larger due to the characteristics of these accidents compared to some of the other groups.

Table 2.7-4
LaSalle Summary Group Fractional Contribution To Core Damage

Summary Group 1

SUM PDS	MFCCD	FCMCD
SEISMIC	0.015	0.007
FIRE	0.238	0.533
FLOOD	0.110	0.030
INTERNAL	0.637	0.430

Summary Group 2

SUM PDS	MFCCD	FCMCD
SEISMIC	0.015	0.007
FIRE	0.238	0.533
FLOOD	0.110	0.030
ATWS	0.005	0.002
LOCA	0.000	0.000
TRANSIENT	0.624	0.423
TRAN-LOCA	0.008	0.005

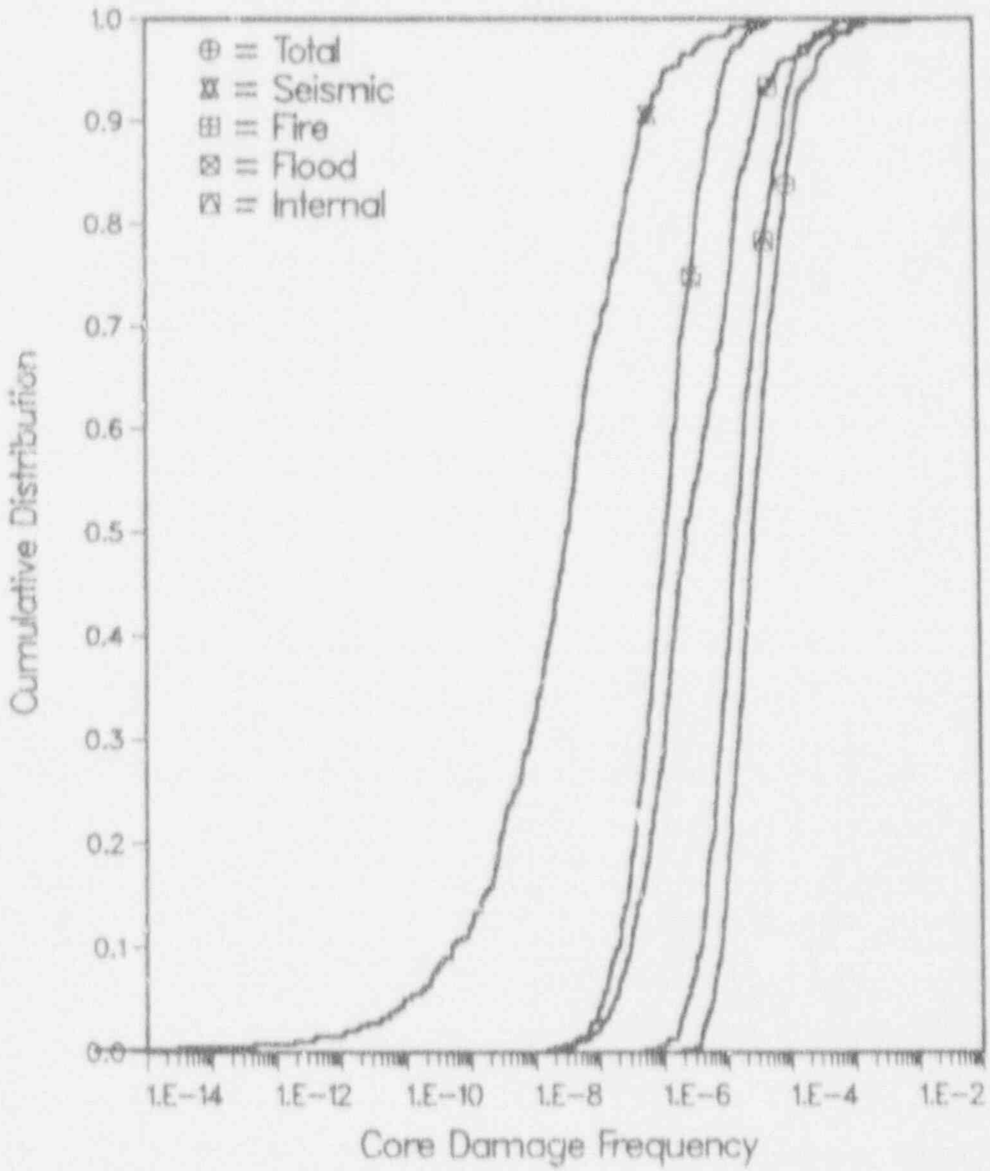

Figure 2.7-2 LaSalle Core Damage Frequencies From Individual Level I Analyses

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3.0 ANALYSIS OF THE ACCIDENT PROGRESSION

3.1 Introduction

This chapter describes the analysis of the progression of the accident, starting from significant core uncovery (i.e., collapsed water level 2 feet above the bottom of the active fuel) with imminent re-flooding of the core not expected and continuing for about 24 hours or until the bulk of the radioactive material that is going to be released has been released. The many possible paths that an accident may follow during the course of the accident are described through the use of an accident progression event tree (APET). The basic purpose of the APET is to delineate those accident progressions which will result in different radionuclide releases (i.e., source terms) from the plant for input into the source term analysis described in Chapter 4 and to quantify the conditional probability of their occurrence. When combined with the frequency of occurrence of the plant damage states, the frequency of each source term can be calculated.

The APET consists of a series of questions that address: the status of coolant injection and containment heat removal systems, the condition of the core, the status of the reactor pressure vessel, the pressure in the containment, the integrity of the containment during the various stages of the accident, and the occurrence and magnitude of phenomena that could affect the state of any of the previous issues or the character of the source term. These questions are developed to the level of detail consistent with the purpose of the analysis, the amount of resources available, and the level necessary to represent the interaction between various events and/or to calculate their probability of occurrence. Each question can have several outcomes or branches which allow the accident to progress down different paths.

The first portion of the APET defines the plant damage states (PDSs) which form the interface between the Level I and Level II portions of the PRA. A PDS describes the status of various emergency systems (e.g., injection systems, containment heat removal systems, and emergency power system), the pressure in reactor pressure vessel (RPV), and the status of containment integrity at the time of core damage. Each PDS defines an initial set of plant conditions that are common for all of the accident sequence cut sets that make up the PDS. Thus, it is assumed that within the level of detail that is used to model the accidents in this study, all of the accident sequence cut sets that form the PDS will behave similarly. This does not mean to imply that the characteristics of the PDSs are certain; as discussed in Section 2.1, there is still substantial uncertainty in the initial and boundary conditions of PDSs, in those characteristics of the plant not modeled in the Level I analysis but important to the Level II analysis, and in our understanding and modeling of the various phenomena and processes occurring during the accident evolution. This uncertainty is what leads to the many possible different accident progressions arising from one PDS.

The output from the APET is, therefore, hundreds of thousands of possible accidents with each accident corresponding to a path through the tree. While the tree is constructed to delineate accident progressions on the basis of those characteristics important to the source terms, it is not possible to know beforehand everything that what will be or will not be important or to separate all paths leading to the same source term. Many of these paths, therefore, will result in very similar source terms. Thus, these paths are grouped together based on attributes that are important to the source term analysis in the binning and rebinning processes discussed later in this chapter. These groups of paths are called accident progression bins (APBs). The APBs are the input to the source term analysis which will be discussed in the next Chapter 4.

To help the reader understand the logic behind the APET, Section 3.2 describes the plant features that are important to the accident progression. In Section 3.3, the APET is briefly described. The quantification of the APET is presented in Section 3.4. Section 3.5 includes a description of the APBs that form the input to the source term analysis. A summary of the results from the accident progression analysis is presented in Section 3.6. These results will provide the reader with greater insight into the integrated risk results that are presented in Chapter 6.

3.2 Plant Features Important to the Accident Progression at LaSalle

This section provides details on the plant features that are important to the progression of a core degradation accident and the response of the containment to the stresses placed upon it. Many of these features are described in Chapter 1; but here, their relationship to the accident progression is made more explicit. These features are:

- containment structure;
- drywell/suppression pool arrangement;
- reactor pedestal cavity;
- containment venting system;
- containment heat removal systems;
- coolant injection systems; and
- automatic depressurization system (ADS).

Containment Structure

The LaSalle plant employs a Mark II type containment to house a General Electric BWR-5 reactor. The primary containment is a post-tensioned, reinforced concrete structure with a steel liner. The containment, shown in Figure 3.2-1, consists of a lower cylindrical portion founded on the base mat and an upper portion 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 internal diameter of the

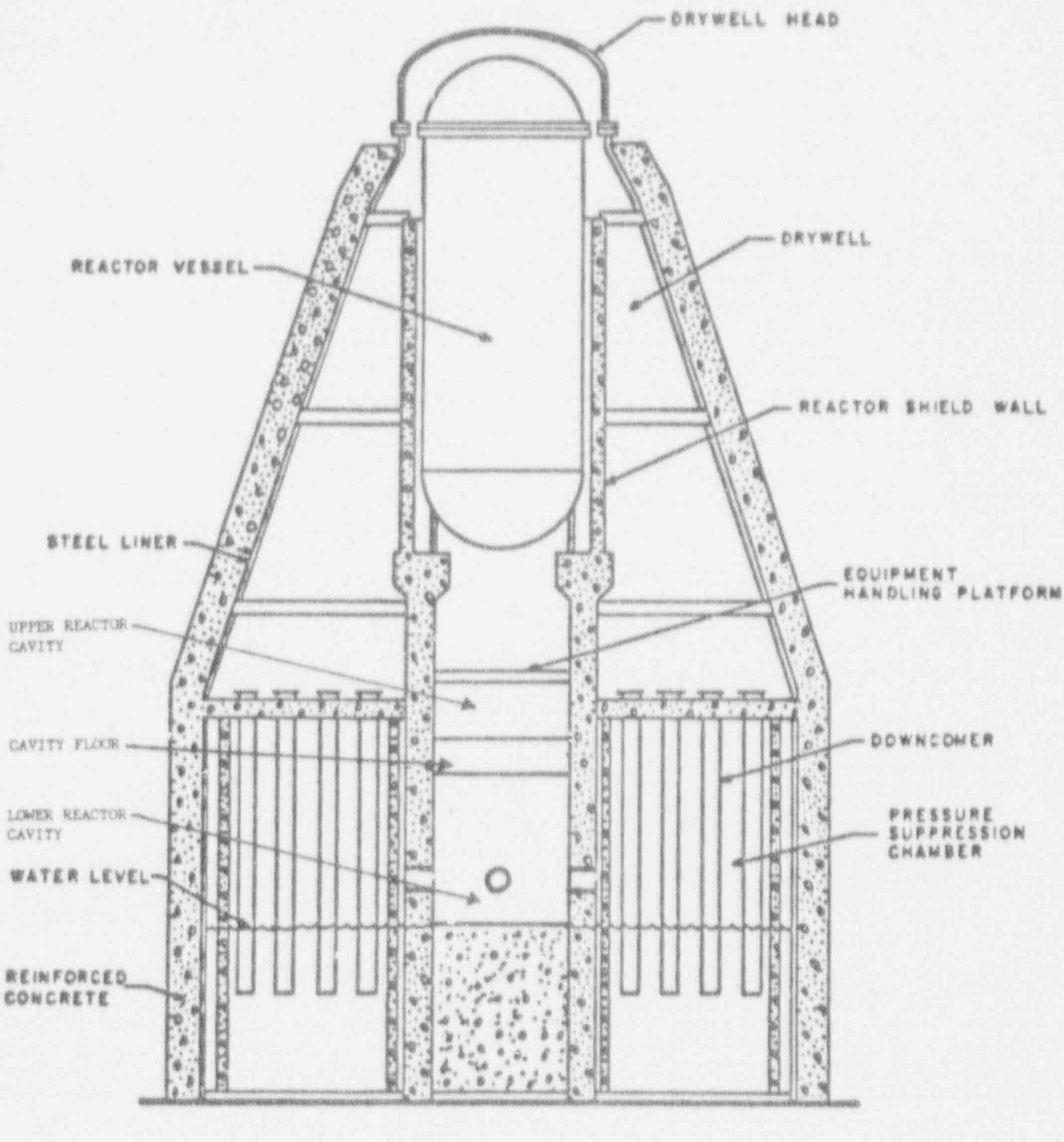

Figure 3.2-1 LaSalle Primary Containment Layout

lower cylindrical portion is approximately 87 feet and the overall height of the containment is approximately 150 feet. The basemat is a 7 foot thick reinforced concrete slab. The primary containment is enclosed by a reinforced concrete reactor building which forms the secondary containment, shown in Figure 3.2-2. 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. Because the containment failure pressure and failure location are important parameters for the accident frequency, accident progression, and source term analyses, the distributions for the ultimate failure pressure and location were assessed by the NUREG-1150 Structural Response Panel. Based on distributions provided by this panel and described in Appendix B.7 in Volume 2 of this report, the assessed mean failure pressure was 191 psig; the minimum and maximum failure pressures were 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. For each location both leaks and ruptures were considered. The containment failure location is important for two reasons. First, containment failure into the reactor building, as opposed to failure through the drywell head into the refueling floor, will flood vital locations in the reactor building with high temperature steam. These vital locations house motor-control cabinets and equipment related to emergency coolant injection systems. It is possible that the high temperature steam will fail these systems. Second, the location of containment failure affects the magnitude of the source term. The relationship between the source term magnitude and the containment failure location is discussed in Chapter 4.

Drywell/Suppression Pool Arrangement

The pressure suppression system is a over-and-under configuration. The drywell, in the form of a frustum of a cone, 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 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

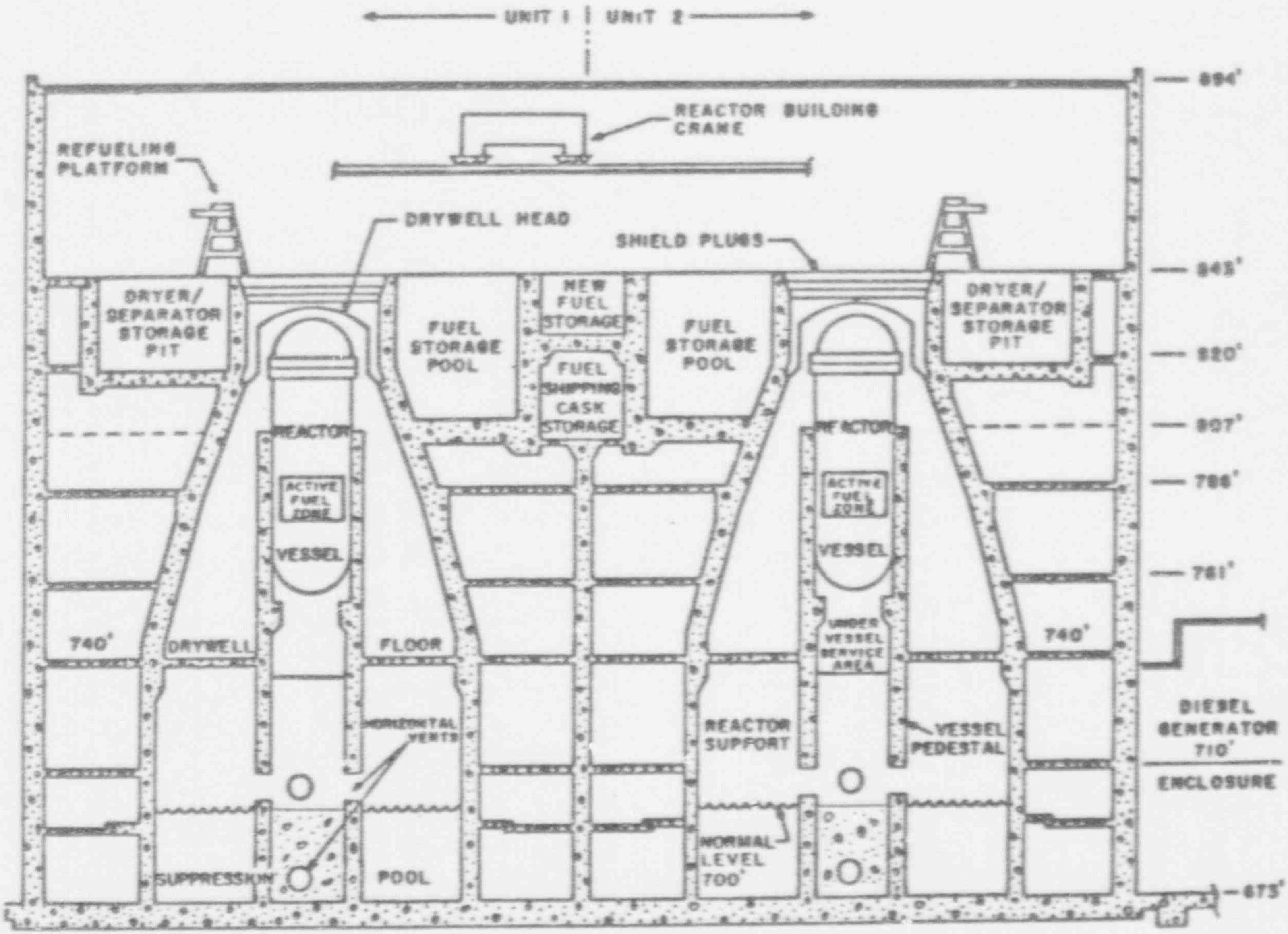

Figure 3.2-2 LaSalle Reactor Building Layout

pressure between the two volumes. The suppression pool is also used to condense the steam and cool the noncondensibles that are released through the SRV tailpipes when the RPV is depressurized or when the SRVs lift or stick open. 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. Each tailpipe has two check valves designed to relieve the pressure on the pipe when the pipe cools after the discharge of hot gases from the primary system. These valves can stick open and result in the SRV discharge being directed to the drywell instead of t e suppression pool. The nominal free volumes of the drywell and the supp ession chamber are 219,800 ft^3 and 165,100 ft^3, respectively. The nominal lume of the suppression pool is 128,800 ft^3.

Reactor Pede..al Cavity

The reactor pedestal cavity is located directly below the reactor pressure vessel. The cavity is divided into an upper and lower portion. The upper cavity is formed by the 4.83 foot thick pedestal wall. The upper cavity is essentially a right cylinder with a diameter of 20.25 feet and a depth of approximately 28 feet. The top of the upper cavity contains the control rod drive (CRD) housings. The major pedestal penetrations in the upper cavity are the CRD piping penetrations at the top of the pedestal and the CRD removal opening (essentially a door) that is located 11 feet above the upper cavity floor. The upper cavity is large enough to contain all of the core debris released at the time of vessel breach. There are scenarios, however, in which a portion the core debris may not remain in the upper cavity. For example, during a direct containment heating event, a portion of the core debris that is ejected at high pressure may be dispersed into the drywell. In other scenarios, it is possible that the upper cavity floor will fail and the core debris will be released into the lower cavity which connects directly to the wetwell. However, because a large amount of the core debris can not enter the drywell, direct attack of the drywell wall by core debris is not an issue at LaSalle as it is for the Mark I containments and possibly for other Mark II containments with different cavity designs.

In addition to holding the core debris, the upper cavity can also accumulate a large volume of water during the accident. When the upper cavity is completely flooded, a water depth of 11 feet can be established. Any additional water will drain out the CRD removal opening onto the drywell floor where it will flow down the downcomers into the suppression pool. There are two pathways by which water in the drywell can enter the upper reactor cavity. The first pathway is through the drywell floor drains that connect to the equipment drain sump in the pedestal. There are three ways water can get on the drywell floor. The first way is from the containment sprays which are located in the drywell. The second way is from condensation of water vapor in the drywell. During long term loss of containment heat removal accidents, the suppression pool becomes saturated. The steam released from the pool is vented into the drywell where it condenses on the drywell wall. The water that accumulates on the drywell floor will drain into the upper cavity rather than down the downcomers because the top of the downcomers are located approximately six inches above the drywell floor whereas the floor drains are essentially flush with the floor. The third way is from LOCAs in the piping in the drywell. The second pathway is through the RPV after vessel failure. Coolant injected into the vessel after vessel breach will drain directly into the upper cavity.

The potential for large amounts of water to be in the upper cavity has two major implications. First, when core debris is released from the vessel at the time of vessel breach, the potential exists for energetic fuel-coolant interactions (FCIs) to occur if the upper cavity is full of water. These FCIs can fail the containment directly from quasi-static pressure loads or can fail the RPV pedestal which can then lead to drywell failure (e.g., penetration failure due to distortion of the piping). An FCI can also fail the pedestal drain pipes. There are two four-inch drain pipes that drain water from the upper cavity sumps to the reactor water cleanup system. The isolation valves on these lines are outside the containment. Thus, failure of these lines outside the containment can establish a pathway for radionuclides to escape the containment. Leakage from equipment in the drywell during normal operation will drain into the sumps. Thus, it is likely that these lines will be full of water regardless of whether another source of water is available to flood the upper cavity. If the upper cavity is flooded, an FCI can occur in either the upper cavity or the pipe. On the other hand, if the upper cavity is dry, an FCI can only occur in the drain pipe. The probability of pipe failure is higher when the FCI occurs in the pipe than it is when it occurs in the upper cavity pool. However, if the pipe entrance is plugged by debris, then either the effects do not propagate to the pipe or no FCI can occur in the pipe. In addition to FCI considerations, a large amount of water in the upper cavity can cool the core debris that is released from the reactor vessel and thus mitigate the releases associated with core-concrete interactions. Thus, based on the above discussion it is obvious that the presence of water in the upper cavity can be either beneficial or detrimental, depending on the accident progression.

Even though the upper cavity can contain all of the core debris, it is possible for the upper reactor cavity floor to fail. This would result in the core debris dropping into the lower reactor cavity in the lower portion of the reactor pedestal. Unlike some other Mark II containment designs, the lower reactor pedestal at LaSalle is partially filled with concrete and contains no water. The lower pedestal is also large enough to contain the core debris and has openings to the wetwell airspace above the suppression pool water line (there will be about 1 ft. between the top of the core debris and the bottom of the openings to the wetwell airspace). The NUREG-1150 Molten Core/Containment Interaction Panel evaluated the probability distributions for the time and mechanism of reactor cavity floor failure after vessel breach at LaSalle. The results of this elicitation are discussed in Appendix B.8 of Volume 2 of this report. The APET contains question to represent the effects of reactor cavity floor failure on the accident progression.

Containment Venting System

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 vent lines using a 2-inch valve or a 26-inch valve (two sets of valves, one set in each line). The vent pipes join into one common vent line which ties into the standby gas treatment system (SGTS) which releases the gases to the stack. The vent pipe is attached to the SGTS with a rubber boot. It was assessed that this rubber boot would fail when high pressure steam was released through the vent. Therefore, the steam would be released into the reactor building rather than being directed to the stack when the containment is vented. Inundation of the reactor building with high temperature steam 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.

Coolant Injection systems

There are various injection systems that can be used to cool the core and arrest the core damage process at LaSalle. Four high pressure and four low pressure injection systems were considered in this analysis.

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). All of these systems can inject water into the primary system up to the relief valve safety setpoints (1146-1205 psig). The HPCS system has a motor-driven pump with its own dedicated diesel generator. This system draws water from either the condensate storage tank or the suppression pool. The RCIC system utilizes a turbine-driven pump. Steam from the RPV is used to drive the turbine which pumps water from either the condensate storage tank or the suppression pool back to the core. Thus, RCIC can not be used once the vessel fails and the steam supply to the turbine is lost. In addition, the RCIC system isolates on high pressure in the wetwell (25 psig). DC power is also required to control this system. The MFW system draws water from the condenser hotwell using a motor-driven pump. This pump requires offsite power (i.e., normal not emergency power). The CRD system can be used to inject water into the core through the control rod drives. The CRD has two small positive-displacement pumps and can only inject several hundred gallons per minute. It 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 train A and B of the CRD system are powered by train A and B of the emergency power system (EPS), respectively. 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. The system is powered by train A of the EPS. LPCS sprays coolant 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. Train A is powered by train A of the 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. This system requires offsite power. The last resort injection system, used when all other systems have failed, is the diesel-driven firewater system (DFWS). 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 (a seismically qualified lake built near the plant). For these low pressure systems to provide coolant to the core, the RPV must be depressurized from normal operating pressure of about 1055 psig to the 150-500 psig depending on the system operating.

Containment Heat Removal Systems

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 the core standby cooling system (CSCS) supplying water from the ultimate heat sink on the shell side. In the CSS mode, water is sprayed into the drywell and, in the SPC mode, the water is returned directly to the suppression pool. 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 taken directly from one of the primary system recirculation loops, passed through the RHR heat exchangers and then injected back into the vessel. In addition, effective containment heat removal can also be obtained by using the LPCI mode of operation. In this mode, coolant from the suppression pool is past through the heat exchangers, injected into the reactor vessel, and then flows back into the suppression pool via a LOCA or by boiloff through the SRV discharge lines. All four modes of RHR (i.e., SPC, CSS, SDC, and LPCI with heat exchangers) require AC power and are, therefore, unavailable during a station blackout. The support system trains require power from the same trains of EPS as the front-line system trains they support.

Automatic Depressurization System

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 to 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 injection pump must be running (LPCS or LPCI only). 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 of the 18 valves discharges into the suppression pool as described in the Section on Drywell/Suppression Pool Arrangement. The ADS valves are located in the drywell and drywell pressures above approximately 85 psig will prevent opening of the ADS valves or result in reclosure of the ADS valves because of the loss of the differential pressure required for valve opening. The ADS system also requires DC power; therefore, the RPV can not be depressurized in sequences that involve failure of DC power or can not remain depressurized in accidents in which the containment pressure exceeds 85 psig before vessel breach occurs.

3.3 Description of the Accident Progression Event Tree

This section describes the APET that was used to perform the accident progression analysis for LaSalle. The APET itself forms a high-level probabilistic model of the accident progression as opposed to a detailed thermal/hydraulic model as in one of the mechanistic codes. The APET is too large to be shown in a figure as smaller event trees usually are. Instead, the APET exists only as a computer input file. The APET was evaluated by the code EVNTRE, which is described elsewhere.[1]

The APET is not meant to be a substitute for mechanistic codes such as the STCP,[2] CONTAIN,[3] MELCOR,[4] and MAAP.[5] Rather, it is an integrating framework for synthesizing the representation of accident progressions as depicted by these codes supplemented by expert judgement that addresses the strengths and weaknesses of the codes. The final result of the APET is a probabilistic analysis of the likelihood of various accident progressions. The mechanistic codes are impractical to use in a large integrated study because they require too much computer time if run for all the possible initial and boundary conditions and all the possible variations in input parameters in order to generate all possible accident progressions. In addition, each code has limitations due to phenomena not included in the physics models, simplifications in the physics models, and the level of detail of the plant model being used. Therefore, the results from these codes were used as a basis for constructing the LaSalle APET by extending these results to include the possible effects of phenomena not modeled, modeling limitations, and any pertinent experimental results not included in the codes. Because it is a probabilistic model, the APET could be evaluated relatively quickly. In this way, the full diversity of possible accident progressions could be considered and the uncertainty in the many phenomena involved could be included.

The APET is a type of event tree which consists of a list of questions which represent the top events in the tree. The answer to a question is expressed as a branch in the tree. A question may have several potential answers depending on the PDS or the accident progression that is being analyzed. For example, a question in the LaSalle APET asks the status of DC power at the time of core damage. The possible answers are: (1) DC power is failed and cannot be recovered, (2) DC power is not available but can be restored once AC power is recovered, and (3) DC power is available. The probability of each branch for a question may depend on the branch taken at a previous question. Thus, if necessary, a question can have a case structure which allows the branch probabilities of a question to depend on the branches taken in previous questions. Referring back to the DC power example, the probability that DC power is available depends on the PDS that is being analyzed. Thus, the case structure for this question delineates groups of PDSs. The probability that DC power is available is different for each group and, therefore, a different branch probability is used for each case. Thus, the value used to quantify the probability of DC power being available for a seismic PDS will be different than the value used for a fire PDS. The probabilities that are assigned to the branches in a question are referred to branching split fractions. Parameter values can also be assigned to an accident path in a question. An example of a parameter used in the LaSalle APET is the containment failure pressure.

The following section contains a brief overview of the LaSalle APET. Details, including a complete listing of the APET, may be found in Appendix B.1 in Volume 2 of this report. The format of this file is described in Reference 1. Section 3.3.2 is a summary of how the APET was quantified, that is, how the many numerical values for branching ratios and parameters were derived. Section 3.3.3 presents the variables that were sampled in the accident progression analysis for LaSalle.

3.3.1 Overview of the Accident Progression Event Tree

The APET for LaSalle considers the progression of the accident from the time core damage is imminent (i.e., collapsed water level two feet above the bottom of the active fuel) through the period of active core-concrete interaction (CCI) and/or containment failure. Except in very unusual accidents, almost all of the fission products that are going to be released from the containment will have been released by 24 hours after the onset of core damage.

This event tree is based on the LaSalle containment arrangement, systems, and procedures. In addition, emphasis was placed on modeling the accident progression for the dominant plant damage states presented in Chapter 2. This means that not all theoretically possible choices are included in the question structure and some questions may default in ways that would lead to incorrect results if changes are made to the tree in other places or PDSs not considered in the original construction of the tree are analyzed. Table 3.3-1 lists the 135 questions in the LaSalle APET. In this APET seven time periods are considered. To facilitate understanding of the APET and referencing between questions, each branch of every question is assigned a mnemonic abbreviation. The mnemonic branch abbreviations for most branches start with a character or characters which indicate the time period of the question. The time periods and their abbreviations are:

Mnemonic	Time	Description
E1	Initial	Questions 1 thru 22 define the PDSs and determine the conditions just prior to core damage.
E2	Before CD	Question 23 thru 37 address the status of the plant before core damage based on attributes of the PDS; but, which were not explicitly addressed by the PDS definition.
E3	During CD	Questions 38 thru 53 address the progression of the accident from the beginning of core damage (CD) to just before vessel breach (VB). Questions in this time period consider: the pressure in the RPV, the status of various systems (e.g., coolant injection and containment heat removal), the pressure in the containment, and the status of the containment integrity (e.g., has it been vented or failed from loads present during core damage). The recovery of AC power is considered in this time period.
E4	Before VB	Questions 54 thru 72 establish the status of injection systems, pressure in the RPV, and the condition of the containment just prior to vessel breach. These questions address the effects that containment failure or venting during core damage had on the injection systems (i.e., severe environment in the reactor building), the RPV pressure, and the pressure in the reactor building. The amount of water in the reactor cavity is also determined during this time period.
I	At VB	Questions 73 thru 98 determine the progression of the accident from immediately before vessel breach to the time of significant core-concrete interaction (CCI). The potential for core damage arrest (i.e., no vessel breach) is addressed in this time period. The majority of these questions address the loads accompanying vessel breach and the containment's structural response to these loads.

Table 3.3-1
Questions in the LaSalle APET

Question Number	Question	Sampling	Quantification
1.	What is the Plant Damage State (PDS)?	SF	TEMAC
2.	Is there a loss of offsite power?	SF	TEMAC
3.	Is there a Station Blackout (Loss of all AC)?		PDS
4.	Is DC Power available?	SF	TEMAC
5.	Does a S/RV stick open early?		PDS
6.	Does the HPCS system fail to inject?	SF	TEMAC
7.	Does the RCIC system fail to inject?	SF	TEMAC
8.	What is the initial status of the CRD hydraulic system?		PDS
9.	What is the initial status of the main feedwater system?	SF	TEMAC
10.	What is the initial status of RPV depressurization?	SF	TEMAC
11.	What is the initial status of the low pressure ECC systems?	SF	TEMAC
12.	What is the initial status of the condensate system?	SF	TEMAC
13.	What is the initial status of containment heat removal?		PDS
14.	What is the initial status of containment spray?	SF	TEMAC
15.	Does the containment fail before core damage?		PDS
16.	Is the containment vented before core damage?		PDS
17.	What is the level of pre-existing leakage or isolation failure?		Internal
18.	What is the location of pre-existing leakage or isolation failure?		Internal
19.	What is the level of pre-existing suppression pool bypass?		Internal
20.	For TC does SLC fail to inject?		PDS
21.	When does core damage occur?		PDS
22.	What type of sequence is this (summary of plant damage)?		Summary
23.	What containment pressure forces reclosure of the S/RVs?		Internal
24.	What is the containment pressure when RCIC fails?		Internal
25.	What is the CF pressure and mode sample value	P	Struct
26.	What is the CF mode before CD?	ZO	UFUN-Str
27.	Is there leakage in the drywell head?		Summary

Table 3.3-1 (Continued)
Questions in the LaSalle APET

Question Number	Question	Sampling	Quantification
28.	Is there leakage in the drywell?		Summary
29.	Is there leakage in the wetwell?		Summary
30.	What is the location of early containment leakage?		Summary
31.	What is the containment leakage level before CD?		Summary
32.	Is the suppression pool drained before CD?		Internal
33.	What is the containment pressure at CD?		Internal
34.	What is the containment pressure level at CD?		Summary
35.	What is the RPV pressure before CD?		AcFrqAn
36.	Is the SP saturated at CD?		Summary
37.	Does (do) any S/RV tailpipe vacuum breaker(s) stick open?	SF	Internal
38.	Does AC power remain lost during core degradation?	SF	ROSP
39.	Is the RPV depressurized during core degradation?	SF	AcFrqAn
40.	Is there injection during core degradation?	SF	AcFrqAn
41.	What is the status of containment sprays during CD?	SF	AcFrqAn
42.	What is the level of flow to the drywell during CD?		Internal
43.	Is the core in a critical configuration following injection recovery?		Internal
44.	Total amount of H2 released in-vessel during CD?	P	In-Vessel
45.	What is the level of In-Vessel zirconium oxidation?		Summary
46.	Does at least one drywell vacuum breaker stick open?		Internal
47.	What is the pressure rise during CD?		UFUN-Int
48.	Is the vent threshold reached during CD?		Summary
49.	Does containment venting occur during CD?	SF	AcFrqAn
50.	Is DC lost during CD?		Internal
51.	Does LP injection fail because of ADS reclosure?		Summary
52.	Does the containment fail by pressure during CD?	ZO	UFUN-Str
53.	What is the CF mode during CD?	ZO	UFUN-Str
54.	Is there a leak in the drywell head prior to VB?		Summary
55.	Is there a leak in the drywell prior to VB?		Summary
56.	Is there leakage in the wetwell prior to VB?		Summary
57.	What is the location of containment leakage prior to VB?		Summary
58.	What is the level of containment leakage before VB?		Summary

Table 3.3-1 (Continued)
Questions in the LaSalle APET

Question Number	Question	Sampling	Quantification
59.	Is the suppression pool drained before VB?		Internal
60.	Does the RPV repressurize during core degradation?		Summary
61.	What is the status of HPCS prior to vessel breach?	SF	AcFrqAn
62.	What is the status of low-pressure ECC prior to VB?	SF	AcFrqAn
63.	What is the status of RCIC prior to VB?	SF	AcFrqAn
64.	What is the status of CRD prior to vessel breach?	SF	AcFrqAn
65.	Is there auto injection during vessel breach?		Summary
66.	What is the reactor building pressure after CF before vessel breach?	P	Loads
67.	What is the level of RxBldg breach/bypass before VB without a burn?	ZO	UFUN-Str
68.	Does the SGTS fail before VB?		Internal
69.	Does hydrogen burn in the RxBldg before VB?		Loads
70.	What is the level of RxBldg breach/bypass by hydrogen burn before VB?	ZO	UFUN-Str
71.	What is the level of RB bypass before VB?		Summary
72.	What is the base containment pressure before VB?		UFUN-Int
73.	Is there a large in-vessel steam explosion?		Internal
74.	Does an Alpha Mode Event fail both the vessel and the containment?	SF	Note 1
75.	Does a large in-vessel steam explosion fail the RPV?	ZO	Internal
76.	What fraction of the core debris would be mobile at vessel breach?	ZO	Internal
77.	Is there water in the reactor cavity		Summary
78.	What is the mode of vessel breach?	ZO	Internal
79.	Is there a high pressure melt ejection?	ZO	Internal
80.	Does a large ex-vessel steam explosion occur?	ZO	MCCI
81.	What is the amount of H2 released at vessel breach?	P	In-Vessel
82.	How much hydrogen is released at vessel breach?		UFUN-Int
83.	What is the peak pedestal pressure at vessel breach?	P	Loads
84.	Does the RPV pedestal fail due to pressurization at vessel breach?	ZO	UFUN-Int
85.	What is the pressure rise from VB?	P	Loads
86.	Does the RPV pedestal fail from an ex-vessel steam explosion (impulse loading)?	SF	Internal
87.	Does the cavity floor fail from core debris attack?	ZO	MCCI

Table 3.3-1 (Continued)
Questions in the LaSalle APET

Question Number	Question	Sampling	Quantification
88.	When does the cavity floor fail?		Summary
89.	Does a FCI fail the drain pipe outside the containment?	SF	Internal
90.	Does the drywell fail on pedestal failure?	ZO	Internal
91.	Does pressurization fail the containment at VB?	ZO	UFUN-Str
92.	What is the CF mode at VB from overpressure?	ZO	UFUN-Str
93.	Is there a leak in the drywell head after VB?		Summary
94.	Is there a leak in the drywell after VB?		Summary
95.	Is there leakage in the wetwell after VB?		Summary
96.	What is the location of containment failure after VB?		Summary
97.	What is the containment leakage level after VB?		Summary
98.	Is the suppression pool drained following VB?		Internal
99.	Is AC power not available late?	SF	ROSP
100.	What is the status of HPCS after vessel breach?	SF	AcFrqAn
101.	What is the status of low-pressure ECC after VB?	SF	AcFrqAn
102.	What is the status of CRD after vessel breach?	SF	AcFrqAn
103.	Is RHR operating late?	SF	AcFrqAn
104.	Do containment sprays operate following VB?	SF	AcFrqAn
105.	Is water supplied to the debris late?		Internal
106.	What is the nature of the core concrete interactions?	ZO	Internal
107.	What fraction of core not participating in HPME participates in CCI?	P	Internal
108.	How much H2, CO, and CO2 are produced during CCI?		UFUN-Int
109.	What is the level of Zirc. oxidation in the pedestal before CCI?		Summary
110.	What is the pressure rise after VB?		UFUN-Int
111.	Is the vent threshold reached after VB?		Summary
112.	Is the containment vented late?	SF	AcFrqAn
113.	How much concrete must be eroded to cause pedestal failure?	P	Internal
114.	At what time does pedestal failure occur?	ZO	MCCI
115.	Does the drywell fail from late pedestal failure before overpressure?		Internal
116.	Is AC power recovered very late?		ROSP
117.	Is RHR operating very late in the accident?	SF	AcFrqAn

Table 3.3-1 (Continued)
Questions in the LaSalle APET

Question Number	Question	Sampling	Quantification
118.	Do containment sprays operate very late in the accident?	SF	AcFrqAn
119.	What is the pressure very late in the accident?		UFUN-Int
120.	Does the containment fail at low pressure from temperature in the drywell?		Summary
121.	If the containment fails from temperature where does it fail?		Summary
122.	Does the containment fail late from overpressure?	ZO	UFUN-Str
123.	What is the CF mode late?	ZO	UFUN-Str
124.	Is there a leak in the drywell head late?		Summary
125.	Is there a leak in the drywell late?		Summary
126.	Is there a leak in the wetwell late?		Summary
127.	What is the location of late containment leakage?		Summary
128.	What is the level of late containment leakage?		Summary
129.	Is the suppression pool drained late?		Internal
130.	What is the level of late suppression pool bypass?		Summary
131.	What is the level of late RxBldg bypass w/o a burn?	ZO	UFUN-Str
132.	Does the standby gas treatment work late without a late hydrogen burn?		Internal
133.	Does hydrogen burn in the RxBldg after VB?		Loads
134.	What is the level of late RxBldg bypass from hydrogen burns?	ZO	UFUN-Str
135.	What is the level of late RxBldg bypass?		Summary

Notes to Table 3.3-1

Note 1. The Alpha mode of vessel and containment failure was previously considered by the Steam Explosion Review Group. The distribution used in this analysis is based on information contained in the report generated by this group.

Key to Acronyms in Table 3.3-1

AcFrqAn — The quantification was performed by the Accident Frequency Analysis project staff.

Table 3.3-1 (Concluded)
Questions in the LaSalle APET

Internal	The quantification was performed at Sandia National Laboratories by the project team with the assistance of other members of the laboratory staff.
In-Vessel	This question was quantified by sampling from an aggregate distribution provided by the Expert Panel on In-Vessel Issues.
Loads	This question was quantified by sampling from an aggregate distribution provided by the Expert Panel on Containment Loads Issues.
MCCI	This question was quantified by sampling from an aggregate distribution provided by the Expert Panel on Molten Core/Containment Interaction Issues.
P	A value, sampled from a distribution, is assigned to a parameter.
PDS	The quantification follows directly from the definition of the Plant Damage State.
ROSP	This question was quantified by sampling from a distribution derived from the offsite power recovery data for the plant.
SF	Split fraction sampling - the branch probabilities are real numbers between zero and one.
Struct	This question was quantified by sampling from a aggregate distribution provided by the Expert Panel on Structural Issues.
Summary	The quantification for this question follows directly from the branches taken at preceding questions, or the values of parameters defined in preceding questions.
UFUN-Str	This question is quantified by the execution of a module in the User Function subroutine, using distributions from the Structural Expert Panel
UFUN-Int	This question is quantified by the execution of a module in the User Function subroutine using models and data generated by the project staff.
ZO	Zero-One sampling - the branch probabilities are either 0.0 or 1.0.
TEMAC	Split fractions from TEMAC4 are used to quantify this question.

L	Late	Questions 99 thru 115 determine the progression of the accident during the core-concrete interaction phase. The recovery of AC power is addressed during this time period as well as the status of coolant injection systems and containment heat removal systems. This section also establishes the nature of CCI, the amount of concrete eroded in the pedestal during CCI, and the containment pressure late in the accident. The probability that the containment is vented is also considered.
VL	Very Late	Questions 116 thru 135 determine the very late conditions. Questions 116 thru 120 determine the very late progression of the accident. These questions allow for very late recovery of AC power and containment heat removal systems during accidents in which the containment has not failed. The containment pressure during this time period is also addressed. Questions 121 thru 135 determine the final status of the containment and reactor building integrity.

The clock time for each period will vary depending upon the type of accident being modeled.

This APET does not contain any questions to resolve core-vulnerable sequences (sequences for which the issue of core damage has not been decided). In NUREG-1150,[6] core vulnerable sequences from the Level I analysis were resolved in the Level II analysis and question relating to their resolution appear in the APET. For the LaSalle analysis, all of the core-vulnerable sequences were resolved in the Level I portion of the PRA. Thus, all of the sequences propagated through the Level II analysis involved core damage.

In several places in the evaluation of the APET, a user function is called. This is a FORTRAN function subroutine which is executed at that point in the evaluation of the APET. The user function allows computations to be carried out which are too complex to be treated directly in the event tree. The user function itself is listed in Appendix B.4, and the general types of calculations performed by the user function are described below. The user function for the LaSalle APET is called to:

- Determine the containment baseline pressure during the various time periods,
- Compute the amount of hydrogen released to the containment at the time of vessel breach and during CCI,
- Calculate the pressure rise due to hydrogen burns in the reactor building,
- Determine the peak pressure in the reactor building that results from containment failure or venting,
- Determine whether the containment fails and the mode of failure, and

* Determine whether the reactor building fails and the mode of failure from pressurization events associated with hydrogen burns in the reactor building or blowdown following containment failure.

3.3.2 Overview of the APET Quantification

This section summarizes the ways in which the questions in the LaSalle APET were quantified and discusses these methods briefly.

In addition to the number and name of the question, Table 3.3-1 indicates if the question was sampled, and how the question was evaluated or quantified. In the sampling column, an entry of P indicates that a parameter is sampled from a distribution. The entry ZO in the sampling column indicates that the question was sampled zero-one, and the entry SF means the questions was sampled with split fractions. The difference may be illustrated by a simple example. Consider a question that has two branches, and a uniform distribution from 0.0 to 1.0 for the probability for the first branch. If the sampling is zero-one, in half the observations, the probability for the first branch will be 1.0, and in the other half of the observations it will be 0.0. If the sampling is a split fraction, the probability for the first branch for each observation is a random fractional value between 0.0 and 1.0. The average over all the fractions in the sample is 0.50. The implications of zero-one or split fraction sampling are discussed in the methodology volume of NUREG/CR-4551,* the NUREG-1150 technical support documents. If the sampling column is blank, the branching ratios for that question, and the parameter values defined in that that question, if any, are quantified with a single fixed value.

The number of questions that were quantified using the various information sources listed in Table 3.3-1 (e.g., accident frequency analysis) are summarized in Table 3.3-2.

Many of the questions in the APET were quantified using distributions generated by the expert panels that were formed during the NUREG-1150 project. Some of the issues that were presented to these expert panels were specifically for the LaSalle plant. These issues included the containment failure pressure (Structural Response Panel) and the reactor cavity failure probability (Molten Core/Containment Interaction Panel), described in Appendix B.7 and B.8 of Vvolume 2 of this report, respectively. Other issues were general for BWRs; however, the actual

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* E. D. Gorham, J. C. Helton, R. J. Breeding, S. C. Hora, W. B. Murfin, J. L. Sprung, and F. T. Harper, "Evaluation of Severe Accident Risks: Methodology for the Accident Progression, Source Term, Consequence, Risk Integration and Uncertainty Analyses," NUREG/CR-4551, Vol. 2, Rev. 1, SAND86-1309, Sandia National Laboratories, in preparation.

Table 3.3-2
LaSalle APET Quantification Summary

Type of Quant.	Number of Questions	Comments
TEMAC/PDS	18	Determined by the Plant Damage State or TEMAC.
AcFrqAn	17	Determined by the Accident Frequency Analysis.
Internal	30	Quantified internally in this analysis.
Summary	37	The branch taken at this question follows directly from the branches taken at previous questions.
ROSP	3	This question was quantified by sampling a distribution derived from the offsite power recovery data for the plant.
UFUN-Str	11	Calculated in the User Function using distributions from the Structural Expert Panel.
UFUN-Int	7	Calculated in the User Function using models and data generated by the project staff.
In-Vessel	2	Distributions from the In-Vessel Expert Panel.
Loads	5	Distributions from the Containment Loads Expert Panel
MCCI	3	Distributions from the Molten Core-Containment Interaction Panel.
Struct	1	Distributions from the Structural Expert Panel.
Other Expert	1	See Note 1 of table 3.3-1.

values determined for LaSalle could be different. These issues included: in-vessel hydrogen production (In-Vessel Phenomenology Panel), reactor building failure pressure (Structural Response Panel), and the reactor building pressure following a hydrogen burn (Containment Loads Panel), described in Appendix B.9 thru B.11, respectively. Additional issues were quantified by modifying issues that were presented to the panel for a different plant. These issues included: containment loads accompanying vessel breach (Containment Loads Panel), peak pedestal pressure at vessel breach (Containment Loads Panel), and the probability of pedestal failure caused by concrete erosion (MCCI Panel). The issues quantified by the expert panels for the NUREG-1150 study are described in NUREG/CR-4551, Volume 2.[7]

In some cases, a question may have been quantified by more than one source. If this is the case, the entry under Quantification in Table 3.3-1 represents the major contributor to the quantification. For example, Question 66, which addresses the pressure in the reactor building following containment failure, was quantified by the Containment Loads expert panel and by the project staff of this study. The majority of cases were quantified by the expert panel. There were several cases, however, that the expert panel felt were not important. These cases were quantified internally by the project staff. However, because the majority of the cases were quantified by the expert panel, the entry in Table 3.3-1 for Question 66 indicates that this question was quantified by the Containment Loads Expert panel.

3.3.3 Variables Sampled for the Accident Progression Analysis

The LaSalle APET consists of 135 questions. However, as was described previously, each question may have a case structure which allows different branch probabilities to be used depending on the branches taken in previous questions. Thus, the number of branch probabilities used in this analysis will exceed the number of branches. In addition to branch probabilities, parameter values (e.g., containment failure pressure) can also be assigned in a question. However, not all of these probabilities and parameters were included in the uncertainty analysis. Only the variables that had broad distributions and that were assessed to be important to the accident progression were included in the uncertainty analysis. Even though all of the branch probabilities and parameters used in the APET were not included in the uncertainty analysis, the number of variables that were sampled is still large. In fact, about 172 variables were sampled for the accident progression analysis. (We use "about" because the variables can be counted differently. For Example if you count correlated and uncorrelated variables separately or if you count each case of a variable separately since the same variable can be quantified in different ways depending on the initial conditions.) That is, every time the APET was evaluated by EVNTRE, the original values of about 172 variables were replaced with values selected for the particular observation under consideration. These values were selected by the LHS program[8] from distributions that were defined before the APET was evaluated. Many of these distributions were determined by expert panels. Table 3.3-3 lists the variables in the APET which were sampled for the accident progression analysis. Some of them are branch split fractions; the others are parameter values for use in calculations performed while the APET is being evaluated.

In Table 3.3-3, the first column gives the variable abbreviation or identifier, and the question (and case if appropriate) in which the variable is used. For example, the first variable, RA-G1, in Table 3.3-3 is the recovery probability for injection, sprays, containment heat removal, and venting. This variable is used to quantify branch probabilities for the following questions: cases 5, 6, and 9 in question 40; case 4 in question 41; case 3 in question 49; case 3 in question 103; case 2 in question 104; case 3 in question 112, case 3 in question 117; and case 3 in question 119. Thus, for all of these cases the same branch probability is used. Even though the entry under "Correlation" is none, all of the cases in this example are completely correlated (i.e., correlation of one) because they use the same variable. The "Correlation" column is used to refer to correlation among different variables. Thus, the variable RA-G1 is not correlated with any other variables in this analysis. Where several variables are correlated, they are treated as one variable in the sensitivity analysis (see Chapter 6), but are different variables as far as the accident progression analysis and sampling process are concerned.

The second column lists the LHS variable number. This number is simply the variable position in the LHS sample (i.e., variable 1 is the first variable in the sample). Many of the variables in the LHS sample are used in the accident frequency analysis and the source term analysis but are not used in the accident progression analysis. Thus, Table 3.3-3 will not include all of the variables in the LHS sample (e.g., variables 1 through 27 are only used in the accident frequency analysis). This number is used in the LHS input and output presented in Appendix G. The number is also used in the sensitivity analysis.

The third column gives the minimum and maximum values of the distribution. An entry of "Zero/One" in this column indicates that the variable was sampled Zero-One, i.e., it took on only the values 0.0 and 1.0. In each observation, one of these two values would be assigned. The definition of the minimum and maximum values that appear in Table 3.3-3 depends on the type of distribution that is being described. For lognormal distributions, the minimum and maximum values correspond to the 0.001 and 0.999 quantile, respectively. For variables that are described by a user supplied distribution (e.g., distributions from an expert panel), the minimum and maximum values corresponded to the minimum and maximum values of the distribution (i.e., 0 and 1.0 quantiles, respectively). For distributions that are generated in the extender code (see Appendix G), the minimum and maximum values correspond to the minimum and maximum value in the LHS sample. For example, the variable RA-G1 is described by a lognormal distribution with a mean of 2.1E-03. The values of this variable at the 0.001 and 0.999 quantile are 2.7E-05 and 4.0E-02, respectively.

Table 3.3-3
Variables Sampled in the Accident Progression Analysis

Variable Question & Case	LHS #	Range	Distribution	Correlation	Correl. with	Description
RA-G1	28	2.7E-5 4.0E-2	Lognormal M=2.1E-3	None		Probability of recovery for injection, sprays, containment heat removal, and venting.
Q40 C5,6,9 Q41 C4, Q49 C3, Q103 C3, Q104 C2, Q112 C3, Q117 C3, Q118 C3						
RA-G11 Q40 C1	29	7.9E-6 4.5E-2	Lognormal M=1.6E-3	None		Probability that RCIC is recovered at the Remote shutdown panel during fire PDS.
RA-G3 Q40 C2,3	30	1.3E-5 7.4E-2	Lognormal M=2.6E-3	None		Probability that operators recovery mainfeed water or condensate during an ATWS.
RA-G2	32	1.1E-5 6.2E-2	Lognormal M=2.2E-3	None		Probability that the operators depressurize the RPV; probability that LPC or CDS is is recovered.
Q35 C3,8,9 Q39 C4, Q40 C8						
RA-G-10 Q40 C10	40	0.01 1.00	Max. Entropy M=0.09	None		Probability that the operators use the diesel driven firewater pumps for injection.
CFMODE Q25 C1	57	0.00 1.00	Uniform M=0.5	None		Random number used to determine the containment failure mode.
CFPress Q25 C1	104	10.6 20.0	Expert M=14.2	None		Containment failure pressure (Bar).
F-ADS-Lk	106	0.00 1.00	Internal M=0.50	None		Probability that containment leakage will fail ADS (severe environment in the containment).

Table 3.3-3 (Continued)
Variables Sampled in the Accident Progression Analysis

Variable Question & Case	LHS #	Range	Distribution	Correlation	Correl. with	Description
F-ADS-Rpt Q35 C7	107	0.00 1.00	Internal M-0.38	None		Probability that containment rupture will fail ADS (severe environment in containment).
SRVBkr1 Q37 C3	108	0.01 0.50	Uniform M-0.25	None		The failure probability of a SRV tailpipe vacuum breaker during a non-ATWS.
SRVBkr2 Q37 C2	109	0.01 0.10	Uniform M-0.055	None		The failure probability of a SRV tailpipe vacuum breaker during an ATWS.
H2INVES1 Q44 C5	110	32.4 1082	Experts M-400	Rank 1 Rank -1	110-115 163-168	The amount of H2 (Kg-moles) produced during CD. The RPV is at high pressure and coolant is not restored to the core (non-ATWS).
H2INVES2 Q44 C3	111	0.0 927	Experts M-296	Rank 1 Rank -1	110-115 163-168	The amount of H2 (Kg-moles) produced during CD. The RPV is at high pressure and coolant is restored to the core during CD (non-ATWS).
H2INVES3 Q44 C6	112	0.00 1145	Experts M-414	Rank 1 Rank -1	110-115 163-168	The amount of H_2 (Kg-moles) produced during CD. The RPV is at low pressure and coolant is not restored to the core (non-ATWS).
H2INVES4 Q44 C4	113	0.0 804	Experts M-252	Rank 1 Rank -1	110-115 163-168	The amount of H_2 (Kg-moles) produced during CD. The RPV is at low pressure and coolant is restored to the core during CD (non-ATWS).
H2INVES5 Q44 C2	114	0.00 1128	Experts M-410	Rank 1 Rank -1	110-115 163-168	The amount of H_2 (Kg-moles) produced during CD. The PDS is an ATWS and coolant is not restored to the core.

Table 3.3-3 (Continued)
Variables Sampled in the Accident Progression Analysis

Variable Question & Case	LHS #	Range	Distri-bution	Corre-lation	Correl. with	Description
H2INVES6 Q44 C1	115	0.0 850	Experts M=197	Rank 1 Rank -1	110-115 163-168	The amount of H_2 (Kg-moles) produced during CD. The PDS is an ATWS and coolant is restored to the core during CD.
F-HPCS-R Q61 C2,6	116	7.9E-2 1.0	Experts M=0.89	None		Probability HPCS will fail because of severe environments following containment rupture.
F-HPCS-L Q61 C3,7	117	0.35 1.00	Experts M=0.79	None		Probability HPCS will fail because of severe environments following containment leakage.
F-HPCS-V Q61 C4,8	118	0.25 1.00	Experts M=0.96	None		Probability HPCS will fail because of severe environments following containment venting.
F-LPI-R Q62 C2,6,10 Q63 C5	119	2.3E-3 1.0	Experts M=0.69	None		Probability LPI will fail because of severe environments following containment rupture (also used for RCIC failure).
F-LPI-L Q62 C3,7,11	120	5.5E-3 1.0	Experts M=0.73	None		Probability LPI will fail because of severe environments following containment leakage
F-LPI-V Q62 C4,7,11 Q63 C6	121	1.4E-3 1.00	Experts M=0.63	None		Probability LPI will fail because of severe environments following containment venting (also used for RCIC failure).
F-CRD-R Q64 C2,6	122	0.49 1.00	Experts M=0.89	None		Probability CRD will fail because of severe environments following containment rupture.

Table 3.3-3 (Continued)
Variables Sampled in the Accident Progression Analysis

Variable Question & Case	LHS #	Range	Distri- bution	Corre- lation	Correl. with	Description
F-CRD-L Q64 C3,7	123	0.54 1.00	Experts M=0.79	None		Probability CRD will fail because of severe environments following containment leakage.
F-CRD-V Q64 C4,8	124	0.60 1.00	Experts M=0.96	None		Probability CRD will fail because of severe environments following containment venting.
P-RB-nH2 Q66 C1	125	0.0 1.0	Uniform M=0.5	None		Random number used to select the peak pressure in the reactor building- no H2 burn.
RBFPres Q66 C1	126	0.0 1.0	Uniform M=0.5	None		Random number used to select the reactor building failure pressure.
P-RB-H2 Q66 C1	127	0.0 1.0	Uniform M=0.5	None		Random number used to select the peak pressure in the RxBldg from a H2 burn.
RBH2Brn Q69 C3	128 129	Zero One	Burn=0.83 nBrn=0 17	None		Probability of a hydrogen burn in the reactor building following CF.
Alpha1 Q74 C2	130	1.0E-7 0.76	Internal M=0.009	Rank 1	131	Probability that an alpha mode event occurs conditional on an in-vessel steam explosion. RPV is at high pressure.
Alpha2 Q74 C3	131	1.2E-7 0.88	Internal M=0.01	Rank 1	130	Probability that an alpha mode event occurs conditional on an in-vessel steam explosions. RPV is at low pressure.
F-RPV-SE Q75 C2	132 133 134 135	Zero One	BtHd 0.2 LgBrch 0.2 SmBrch 0.3 nFail 0.3	None		The probability that an in-vessel steam explosion will fail the RPV.

Table 3.3-3 (Continued)
Variables Sampled in the Accident Progression Analysis

Variable Question & Case	LHS #	Range	Distri-bution	Corre-lation	Correl. with	Description
LiqVB1 Q76 C1	136 137	Zero One	HiLiq 0.025 LoLiq 0.975	None		Probability that there is a large amount of molten core debris (HiLiq) at VB given that coolant is being supplied to the RPV.
LiqVB2 Q76 C2	138 139	Zero One	H Liq 0.10 LoLiq 0.90	None		Probability that there is a large amount of molten core debris (HiLiq) at VB given that coolant is not being supplied to the RPV.
F-RPV1 Q78 C5	140 141 142 143	Zero One	BtHd 0.125 LgBrch 0.005 SmBrch 0.370 nFail 0.500	Rank 1	140-150	The probability that the RPV will fail given that a large amount of the core is molten and coolant is being injected into the RPV.
F-RPV2 Q78 C6,C7 C9,C10	144 145 146	Zero One	BtHd 0.250 LgBrch 0.005 SmBrch 0.745 nFail 0.000	Rank 1	140-150	The probability that the RPV will fail given that there is no coolant injection.
F-RPV3 Q78 C8	147 148 159 150	Zero One	BtHd 0.062 LgBrch 0.005 SmBrch 0.188 nFail 0.745	Rank 1	140-150	The probability that the RPV will fail given that a small amount of the core is molten and coolant is being injected into the RPV.
HPME Q79 C3	151 152	Zero One	HPME 0.8 nHPME 0.2	None		The probability of an HPME event given that the RPV fails at high pressure.
E-StmEx1 Q80 C2	 153 154	Zero One	SEPool 0.00 SEPipe 0.63 nEXSE 0.37	None		The probability that ex-vessel steam explosion occurs in a dry cavity with a large amount of molten core debris (StmEx in pipe).

Table 3.3-3 (Continued)
Variables Sampled in the Accident Progression Analysis

Variable Question & Case	LHS #	Range	Distribution	Correlation	Correl. with	Description
E-StmEx1 Q80 C3	 155 156	Zero One	SEPool 0.00 SEPipe 0.54 nEXSE 0.46	None		The probability that ex-vessel steam explosion occurs in a dry cavity with a small amount of molten core debris (StmEx in pipe).
E-StmEx1 Q80 C4	157 158 159	Zero One	SEPool 0.58 SEPipe 0.14 nEXSE 0.28	None		The probability that ex-vessel steam explosion occurs in a wet cavity with a large amount of molten core debris.
E-StmEx1 Q80 C5	160 161 162	Zero One	SEPool 0.48 SEPipe 0.10 nEXSE 0.42	None		The probability that ex-vessel steam explosion occurs in a wet cavity with a small amount of molten core debris.
H2AVB1 Q81 C6	163	0.0 556	Experts M-208	Rank 1 Rank-1	163-168 110-115	The amount of H_2 (Kg-moles) produced at VB. PDS is not an ATWS, the RPV is pressurized, and coolant is not restored during CD.
H2AVB2 Q81 C4	164	0.0 232	Experts M-47	Rank 1 Rank-1	163-168 110-115	The amount of H_2 (Kg-moles) produced at VB. PDS is not an ATWS, the RPV is pressurized, and coolant is restored to the RPV during CD.
H2AVB3 Q81 C7	165	0.0 371	Experts M-55	Rank 1 Rank-1	163-168 110-115	The amount of H_2 (Kg-moles) produced at VB. PDS is a not an ATWS, the RPV pressure is low and coolant is not restored during CD.
H2AVB4 Q81 C5	166	0.0 139	Experts M-24	Rank 1 Rank-1	163-168 110-115	The amount of H_2 (Kg-moles) produced at VB. PDS is a not an ATWS, the RPV pressure is low and coolant is restored to the RPV during CD.
H2AVB5 Q81 C3	167	0.0 572	Experts M-79	Rank 1 Rank-1	163-168 110-115	The amount of H2 (Kg-moles) produced at VB during an ATWS in which coolant injection is not restored to the RPV.

Table 3.3-3 (Continued)
Variables Sampled in the Accident Progression Analysis

Variable Question & Case	LHS #	Range	Distri-bution	Corre-lation	Correl. with	Description
H2AVB6 Q81 C2	168	0.0 695	Experts M=55	Rank 1 Rank-1	163-168 110-115	The amount of H2 (Kg-moles) produced at VB during an ATWS in which coolant injection is restored to the core during CD.
PedVB1 Q83 C3	169	5.50 83.70	Experts M=35.80	Rank 1:	170,173 174,184,185,188 189,196,197,200 201	The peak pedestal cavity pressure (bar) at VB. RPV fails at high pressure into a wet cavity (Expert Case 1-HC); differential pressure.
PedVB2 Q83 C4	170	4.68 83.70	Experts M=27.80	Rank 1:	169,173 174,184,185,188 189,196,197,200 201	The peak pedestal cavity pressure (bar) at VB. RPV fails at high pressure into a wet cavity (Expert Case 1-hC).
PedVB3 Q83 C5	171	3.85 60.00	Experts M=30.80	Rank 1:	172,175 176,186,187,190 191,198,199,202 203	The peak pedestal cavity pressure (bar) at VB. RPV fails at high pressure into a dry cavity (Expert Case 2-HC).
PedVB4 Q83 C6	172	0.00 49.80	Experts M=17.20	Rank 1:	171,175 176,186,187,190 191,198,199,202 203	The peak pedestal cavity pressure (bar) at VB. RPV fails at high pressure into a dry cavity (Expert Case 2-hC).
PedVB5 Q83 C7	173	4.40 67.00	Experts M=32.50	Rank 1:	169,170 174,184,185,188 189,196,197,200 201	The peak pedestal cavity pressure (bar) at VB. RPV fails at high pressure into a wet cavity (Expert Case 1-Hc).

Table 3.3-3 (Continued)
Variables Sampled in the Accident Progression Analysis

Variable Question & Case	LHS #	Range	Distri-bution	Corre-lation	Correl. with	Description
PedVB6 Q83 C8	174	3.74 56.90	Experts M=21.70	Rank 1:	169,170 173,184,185,188 189,196,197,200 201	The peak pedestal cavity pressure (bar) at VB. RPV fails at high pressure into a wet cavity (Expert Case 1-hc).
PedVB7 Q83 C9	175	3.08 60.00	Experts M=28.50	Rank 1:	171,172 176,186,187,190 191,198,199,202 203	The peak pedestal cavity pressure (bar) at VB. RPV fails at high pressure into a dry cavity (Expert Case 2-Hc).
PedVB8 Q83 C10	176	2.62 39.90	Experts M=14.30	Rank 1:	171,172 175,186,187,190 191,198,199,202 203	The peak pedestal cavity pressure (bar) at VB. RPV fails at high pressure into a dry cavity (Expert Case 2-hc).
PedVB9 Q83 C12 Q83 C14	177	2.00 42.00	Experts M=11.20	Rank 1	178-182 192-195	The peak pedestal cavity pressure (bar) at VB. RPV fails at low pressure into a wet cavity (Expert Cases 3-OHC and 3-oHC).
PedVB10 Q83 C13	178	1.38 24.00	Experts M=7.4	Rank 1	177 179-182 192-195	The peak pedestal cavity pressure (bar) at VB. RPV fails at low pressure into a wet cavity (Expert Case 3-OhC).
PedVB11 Q83 C15	179	0.69 24.00	Experts M=5.57	Rank 1	177,178 180-182 192-195	The peak pedestal cavity pressure (bar) at VB. RPV fails at low pressure into a wet cavity (Expert Case 3-ohC).
PedVB12 Q83 C16	180	1.00 42.00	Experts M=10.00	Rank 1	177-179 181,182 192-195	The peak pedestal cavity pressure (bar) at VB. RPV fails at low pressure into a wet cavity (Expert Case 3-OHc).

Table 3.3-3 (Continued)
Variables Sampled in the Accident Progression Analysis

Variable Question & Case	LHS #	Range	Distri- bution	Corre- lation	Correl. with	Description
PedVB13 Q83 C17 Q83 C18	181	1.00 21.00	Experts M=6.06	Rank 1	177-180 182 192-195	The peak pedestal cavity pressure (bar) at VB. RPV fails at low pressure into a wet cavity (Expert Cases 3-Ohc and 3-oHc).
PedVB14 Q83 C19	182	0.69 16.00	Experts M=4.36	Rank 1	177-181 192-195	The peak pedestal cavity pressure (bar) at VB. RPV fails at low pressure into a wet cavity (Expert Case 3-ohc).
PedFail Q84 C1	183	11.4 38.6	Internal M=25.24	None		Pedestal failure pressure (bar) (differential pressure)
DWPVB1 Q85 C3	184	0.00 20.00	Experts M=4.34	Rank 1:	169,170 173,174,185,188 189,196,197,200 201	The peak drywell pressure differential (bar) at VB. RPV fails at high pressure into a wet cavity; no ped. failure (Expert Case 1-HC).
DWPVB2 Q85 C4	185	0.00 20.00	Experts M=3.33	Rank 1:	169,170 173,174,184,188 189,196,197,200 201	The peak drywell pressure differential (bar) at VB. RPV fails at high pressure into a wet cavity; no ped. failure (Expert Case 1-hC).
DWPVB3 Q85 C5	186	0.33 9.50	Experts M=3.92	Rank 1:	171,172 175,176,187,190 191,198,199,202 203	The peak drywell pressure differential (bar) at VB. RPV fails at high pressure into a dry cavity; no ped. failure (Expert Case 2-HC).
DWPVB4 Q85 C6	187	0.20 5.31	Experts M=2.42	Rank 1:	171,172 175,176,186,190 191,198,199,202 203	The peak drywell pressure differential (bar) at VB. RPV fails at high pressure into a dry cavity; no ped. failure (Expert Case 2-hC).

Table 3.3-3 (Continued)
Variables Sampled in the Accident Progression Analysis

Variable Question & Case	LHS #	Range	Distri-bution	Corre-lation	Correl. with	Description
DWPVB5 Q85 C7	188	0.00 20.00	Experts M=4.25	Rank 1:	169,170 173,174,184,185 189,196,197,200 201	The peak drywell pressure differential (bar) at VB. RPV fails at high pressure into a wet cavity; no ped. failure (Expert Case 1-Hc).
DWPVB6 Q85 C8	189	0.00 20.00	Experts M=3.10	Rank 1:	169,170 173,174,184,185 188,196,197,200 201	The peak drywell pressure differential (bar) at VB. RPV fails at high pressure into a wet cavity; no ped. failure (Expert Case 1-hc).
DWPVB7 Q85 C9	190	0.33 8.50	Experts M=3.37	Rank 1:	171,172 175,176,186,187 191,198,199,202 203	The peak drywell pressure differential (bar) at VB. RPV fails at high pressure into a dry cavity; no ped failure (Expert Case 2-Hc).
DWPVB8 Q85 C10	191	0.20 5.31	Experts M=2.22	Rank 1:	171,172 175,176,186,187 190,198,199,202 203	The peak drywell pressure differential (bar) at VB. RPV fails at high pressure into a dry cavity; no ped. failure (Expert Case 2-hc).
DWPVB9 Q85 C12	192	0.00 20.00	Experts M=2.94	Rank 1	177-182 193-195	The peak drywell pressure differential (bar) at VB. RPV fails at low pressure into a dry cavity (Expert Case 3-HC).
DWPVB10 Q85 C13	193	0.00 20.00	Experts M=2.41	Rank 1	177-182 192 194,195	The peak drywell pressure differential (bar) at VB. RPV fails at low pressure into a dry cavity (Expert Case 3-hC).
DWPVB11 Q85 C14	194	0.00 20.00	Experts M=2.90	Rank 1	177-182 192,193 195	The peak drywell pressure differential (bar) at VB. RPV fails at low pressure into a dry cavity (Expert Case 3-Hc).

Table 3.3-3 (Continued)
Variables Sampled in the Accident Progression Analysis

Variable Question & Case	LHS #	Range	Distribution	Correlation	Correl. with	Description
DWPVB12 Q85 C15	195	0.00 20.00	Experts M=2.38	Rank 1	177-182 192-194	The peak drywell pressure differential (bar) at VB. RPV fails at low pressure into a dry cavity (Expert Case 3-hc).
DWPVB13 Q85 C3	196	0.00 20.00	Internal M=8.30	Rank 1:	169,170 173,174,184,185 188,189,197,200 201	The peak drywell pressure differential (bar) at VB. RPV fails at high pressure into a wet cavity; pedestal failure (Expert Case 1-HC).
DWPVB14 Q85 C4	197	0.00 20.00	Internal M=6.36	Rank 1:	169,170 173,174,184,185 188,189,196,200 201	The peak drywell pressure differential (bar) at VB. RPV fails at high pressure into a wet cavity; pedestal failure (Expert Case 1-hC).
DWPVB15 Q85 C5	198	0.33 20.00	Internal M=8.1	Rank 1:	171,172 175,176,186,187 190,191,199,202 203	The peak drywell pressure differential (bar) at VB RPV fails at high pressure into a dry cavity; pedestal failure (Expert Case 2-HC).
DWPVB16 Q85 C6	199	0.20 20.00	Internal M=6.28	Rank 1:	171,172 175,176,186,187 190,191,198,202 203	The peak drywell pressure differential (bar) at VB. RPV fails at high pressure into a dry cavity; pedestal failure (Expert Case 2-hC).
DWPVB17 Q85 C7	200	0.00 20.00	Internal M=8.29	Rank 1:	169,170 173,174,184,185 188,189,196,197 201	The peak drywell pressure differential (bar) at VB. RPV fails at high pressure into a wet cavity; pedestal failure (Expert Case 1-Hc).

Table 3.3-3 (Continued)
Variables Sampled in the Accident Progression Analysis

Variable Question & Case	LHS #	Range	Distri-bution		Corre-lation	Correl. with	Description
DWPVB18 Q85 C8	201	0.00 20.00	Internal M=5.58		Rank 1:	169,170 173,174,184,185 188,189,196,197 200	The peak drywell pressure differential (bar) at VB. RPV fails at high pressure into a wet cavity; pedestal failure (Expert Case 1-hc).
DWPVB19 Q85 C9	202	0.33 20.00	Internal M=7.52		Rank 1:	171,172, 175,176,186,187 190,191 198,199 203	The peak drywell pressure differential (bar) at VB. RPV fails at high pressure into a dry cavity; pedestal failure (Expert Case 2-Hc).
DWPVB20 Q85 C10	203	0.20 20.00	Internal M=5.63		Rank 1:	171,172 175,176,186,187 190,191,198,199 202	The peak drywell pressure differential (bar) at VB. RPV fails at high pressure into a dry cavity; pedestal failure (Expert Case 2-hc).
PedExSE1 Q86 C1	204	0.00 1.00	Internal M=0.5		None		The probability that the pedestal does not fail given that an ExSE occurs at VB.
PedExSE2 Q86 C1	205	0.00 1.00	Internal M=0.5		None		The probability that the pedestal floor fails given that the pedestal fails from ExSE.
CavFail Q87 C3	206 207 208 209	Zero One	CavF30 CavF1h CavF2h noCavF	0.82 0.07 0.08 0.03	None		The probability that the cavity floor fails during various time intervals due to core debris attack when a large amount of core debris is involved in a ExSE in the pool.
CavFail Q87 C4	210 211 212	Zero One	CavF30 CavF1h CavF2h noCa	0.97 0.02 0.01 0.00	None		The probability that the cavity floor fails during various time intervals due to core debris attack when a large amount of core debris is involved in a ExSE in the pipe.

Table 3.3-3 (Continued)
Variables Sampled in the Accident Progression Analysis

Variable Question & Case	LHS #	Range	Distribution	Correlation	Correl. with	Description
CavFail Q87 C6	213 214 215 216	Zero One	CavF30 0.15 CavF1h 0.18 CavF2h 0.18 noCavF 0.49	None		The probability that the cavity floor fails during various time intervals due to core debris attack when a small amount of core debris is involved in a ExSE in the pool.
CavFail Q87 C7	217 218 219 220	Zero One	CavF30 0.72 CavF1h 0.11 CavF2h 0.08 noCavF 0.09	None		The probability that the cavity floor fails during various time intervals due to core debris attack when a small amount of core debris is involved in a ExSE in the pipe.
CavFail Q87 C8	221 222 223 224	Zero One	CavF30 0.69 CavF1h 0.06 CavF2h 0.08 noCavF 0.17	None		The probability that the cavity floor fails during various time intervals due to core debris attack. A small amount of core debris is released; no ExSE.
CFDPipe1 Q89 C1	225	0.0 0.1	Uniform M=0.05	Rank 1	226	The probability that a FCI in the cavity pool fails the drain pipe outside containment.
CFDPipe2 Q89 C2	226	0.0 1.0	Uniform M=0.5	Rank 1	225	The probability that a FCI in the drain pipe fails the drain pipe outside containment.
DW-Ped-F Q90 C1	227 228	Zero One	Fail 0.175 nFail 0.825	None		The probability that pedestal failure induces drywell failure given that the pedestal fails.
CCI Q106 C5 Q106 C6,7	229 230	Zero One	FldCCI 0.84 NOCCI 0.16	None		The probability that the core debris in the cavity is cooled (i.e., no CCI).
CD-CCI1 Q107 C2	231	0.6 1.0	Uniform M=0.8	Rank 1	232	The fraction of core debris that participates in CCI; given that a large amount of core debris participates in an ExSE.

Table 3.3-3 (Continued)
Variables Sampled in the Accident Progression Analysis

Variable Question & Case	LHS #	Range	Distribution	Correlation	Correl. with	Description
CD-CCI2 Q107 C3	232	0.9 1.0	Uniform M=0.95	Rank 1	231	The fraction of core debris that participates in CCI; given that a small amount of core debris participates in an ExSE.
ConErPed Q113 C1	233	0.3 1.5	Internal M=0.92	None		The depth (M) of concrete erosion that will fail the reactor pedestal.
PedF1G1 Q114 C3	234	0.00 0.53	Expert M=0.19	Rank 1	234-261	The depth of concrete eroded (M) in 1 hour during CCI--Expert Group 1.
PedF1G2 Q114 C5	235	0.00 0.53	Expert M=0.16	Rank 1	234-261	The depth of concrete eroded (M) in 1 hour during CCI--Expert Group 2.
PedF1G3 Q114 C4	236	0.00 0.39	Expert M=0.14	Rank 1	234-261	The depth of concrete eroded (M) in 1 hour during CCI--Expert Group 3.
PedF1G4 Q114 C9	237	0.02 0.60	Expert M=0.20	Rank 1	234-261	The depth of concrete eroded (M) in 1 hour during CCI--Expert Group 4.
PedF1G5 Q114 C8	238	0.02 0.61	Expert M=0.26	Rank 1	234-261	The depth of concrete eroded (M) in 1 hour during CCI--Expert Group 5.
PedF1G6 Q114 C6	239	0.02 0.61	Expert M=0.26	Rank 1	234-261	The depth of concrete eroded (M) in 1 hour during CCI--Expert Group 6.
PedF1G7 Q114 C7	240	0.02 0.43	Expert M=0.20	Rank 1	234-261	The depth of concrete eroded (M) in 1 hour during CCI--Expert Group 7.
PedF3G1 Q114 C3	241	0.00 0.75	Expert M=0.32	Rank 1	234-261	The depth of concrete eroded (M) in 3 hours during CCI--Expert Group 1.

Table 3.3-3 (Continued)
Variables Sampled in the Accident Progression Analysis

Variable Question & Case	LHS #	Range	Distribution	Correlation	Correl. with	Description
PedF3G2 Q114 C5	242	0.00 0.75	Expert M=0.29	Rank 1	234-261	The depth of concrete eroded (M) in 3 hours during CCI--Expert Group 2.
PedF3G3 Q114 C4	243	0.00 0.68	Expert M=0.26	Rank 1	234-261	The depth of concrete eroded (M) in 3 hours during CCI--Expert Group 3.
PedF3G4 Q114 C9	244	0.07 0.85	Expert M=0.41	Rank 1	234-261	The depth of concrete eroded (M) in 3 hours during CCI--Expert Group 4.
PedF3G5 Q114 C8	245	0.07 0.85	Expert M=0.47	Rank 1	234-261	The depth of concrete eroded (M) in 3 hours during CCI--Expert Group 5.
PedF3G6 Q114 C6	246	0.07 0.85	Expert M=0.47	Rank 1	234-261	The depth of concrete eroded (M) in 3 hours during CCI--Expert Group 6.
PedF3G7 Q114 C7	247	0.07 0.85	Expert M=0.40	Rank 1	234-261	The depth of concrete eroded (M) in 3 hours during CCI--Expert Group 7.
PedF6G1 Q114 C3	248	0.15 1.26	Expert M=0.55	Rank 1	234-261	The depth of concrete eroded (M) in 6 hours during CCI--Expert Group 1.
PedF6G2 Q114 C5	249	0.15 1.26	Expert M=0.52	Rank 1	234-261	The depth of concrete eroded (M) in 6 hours during CCI--Expert Group 2.
PedF6G3 Q114 C4	250	0.15 1.26	Expert M=0.49	Rank 1	234-261	The depth of concrete eroded (M) in 6 hours during CCI--Expert Group 3.
PedF6G4 Q114 C9	251	0.23 1.26	Expert M=0.66	Rank 1	234-261	The depth of concrete eroded (M) in 6 hours during CCI--Expert Group 4.

Table 3.3-3 (Continued)
Variables Sampled in the Accident Progression Analysis

Variable Question & Case	LHS #	Range	Distribution	Correlation	Correl. with	Description
PedF6G5 Q114 C8	252	0.28 1.26	Expert M=0.72	Rank 1	234-261	The depth of concrete eroded (M) in 6 hours during CCI--Expert Group 5.
PedF6G6 Q114 C6	253	0.28 1.26	Expert M=0.72	Rank 1	234-261	The depth of concrete eroded (M) in 6 hours during CCI--Expert Group 6.
PedF6G7 Q114 C7	254	0.23 1.26	Expert M=0.62	Rank 1	234-261	The depth of concrete eroded (M) in 6 hours during CCI--Expert Group 7.
PedF10G1 Q114 C3	255	0.36 1.41	Expert M=0.83	Rank 1	234-261	The depth of concrete eroded (M) in 10 hours during CCI--Expert Group 1.
PedF10G2 Q114 C5	256	0.25 1.41	Expert M=0.79	Rank 1	234-261	The depth of concrete eroded (M) in 10 hours during CCI--Expert Group 2.
PedF10G3 Q114 C4	257	0.25 1.41	Expert M=0.74	Rank 1	234-261	The depth of concrete eroded (M) in 10 hours during CCI--Expert Group 3.
PedF10G4 Q114 C9	258	0.29 1.57	Expert M=0.83	Rank 1	234-261	The depth of concrete eroded (M) in 10 hours during CCI--Expert Group 4.
PedF10G5 Q114 C8	259	0.37 1.57	Expert M=0.92	Rank 1	234-261	The depth of concrete eroded (M) in 10 hours during CCI--Expert Group 5.
PedF10G6 Q114 C6	260	0.37 1.57	Expert M=0.92	Rank 1	234-261	The depth of concrete eroded (M) in 10 hours during CCI--Expert Group 6.
PedF10G7 Q114 C7	261	0.30 1.41	Expert M=0.93	Rank 1	234-261	The depth of concrete eroded (M) in 10 hours during CCI--Expert Group 7.

Table 3.3-3 (Continued)
Variables Sampled in the Accident Progression Analysis

Variable Question & Case	LHS #	Range	Distribution	Correlation	Correl. with	Description
AC-ST-CD Q38 C3	284	0.38 0.83	Internal M=0.69	Rank 1	85-90 284-292	The probability that AC power is recovered before VB during a short-term SB given that it was not available at CD.
AC-ST-VB Q99 C3	285	0.14 0.69	Internal M=0.41	Rank 1	85-90 284-292	The probability that AC power is recovered after VB during a short-term SB given that it was not available during CD.
AC-ST LT Q116 [illegible]	286	0.39 0.99	Internal M=0.87	Rank 1	85-90 284-292	The probability that AC power is recovered late during a short-term SB given that it was not available shortly after VB.
AC-IT1-CD Q38 C5	287	0.11 0.74	Internal M=0.70	Rank 1	85-90 284-292	The probability that AC power is recovered before VB during an extended SB (w/o RPV repressurization) given that it was not available at core damage.
AC-IT1-VB Q99 C5	288	0.028 0.60	Internal M=0.29	Rank 1	85-90 284-292	The probability that AC power is recovered after VB during an extended SB (w/o RPV repressurization) given that it was not available during CD.
AC-IT1-LT Q116 C6	289	0.014 0.88	Internal M=0.74	Rank 1	85-90 284-292	The probability that AC power i[illegible]covered [illegible]ate during an extended SB (w/o R[illegible] repressurization) given that was n[illegible] available shortly after VB.
AC-IT2-CD Q38 C4	290	0.23 0.98	Internal M=0.85	Rank 1	85-90 284-292	The probability that AC power is recovered before VB during an extended SB (RPV repressurizes) given that it was not available at core damage.

Table 3.3-3 (Concluded)
Variables Sampled in the Accident Progression Analysis

Variable Question & Case	LHS #	Range	Distribution	Correlation	Correl. with	Description
AC-IT2-VB Q99 C4	291	3.1E-3 0.56	Internal M=0.27	Rank 1	85-90 284-292	The probability that AC power is recovered after VB during an extended SB (RPV repressurizes) given that it was not available during CD.
AC-IT2-LT Q116 C5	292	0.011 0.84	Internal M=0.71	Rank 1	85-90 284-292	The probability that AC power is recovered late during an extended SB (RPV repressurizes) given that was not available shortly after VB.

The fourth column in Table 3.3-3 indicates the type of distribution used. For uniform distributions from 0.0 to 1.0, the mean is obvious and so is not listed. Otherwise, the mean (M) is given if appropriate. The entry "Experts" for the distribution indicates that the distribution came from an expert panel and the entry "Internal" distribution indicates that the distribution was determined by some method other than the formal expert elicitation process. (None of the distributions obtained by aggregating the conclusions of experts can be described succinctly in words. Plots of some of the aggregate expert distributions are contained in Volume 2 of NUREG/CR-4551.[7] A listing of the input to the LHS program that contains many of these distributions in tabular form is given in Appendix G.) For Zero-One variables, an indication of the probability of each state (the average value obtained for that branch over the sample) is given in this column.

The fifth and s[illegible] columns in Table 3.3-3 show whether the variable is correlated with [illegible] variable. "Rank 1" indicates a rank correlation of 1.0. The [illegible] "[illegible]rel with" column are the LHS variable numbers. In the [illegible] H2INVES1, the numbers 110-115 indicate that H2INVES1 is correla[illegible] LHS variables 110 thru 115.

3.4 Description of the Accident Progression Bins

As each path through the APET was evaluated, the result of that evaluation is stored by assigning it to an accident progression bin. This bin describes the evaluation in enough detail that a source term (release of radionuclides) can be calculated for it. The accident progression bins are the means by which information is passed from the accident progression analysis to the source term analysis. A bin was defined by specifying the attribute or value for each of fourteen characteristics or quantities which define a certain feature of the evaluation of the APET. Section 3.4.1 describes the fourteen characteristics, and the values that each characteristic can assume. The binner itself, which is expressed as a computer input file, is listed in Appendix B.2. Section 3.4.2 contains a discussion of rebinning, a process that takes place between evaluating the APET (in which binning takes place) and the source term analysis. This is because the analyst might wish to bin initially on characteristics of interest to the accident progression to gain additional insights and then simplify the result to only those of interest in the source term calculation. Appendix B.3 contains the input of the rebinner. Section 3.4.3 describes a reduced set of binning characteristics, which is used in presenting the results of evaluating the APET.

3.4.1 Description of the Bin Characteristics

The binning scheme for LaSalle utilized fourteen characteristics. That is, fourteen types of information were used to define a path through the APET. A bin was defined by specifying a letter for each of the fourteen characteristics, where each letter for each characteristic has a meaning defined in Table 3.4-1. For a characteristic, the possible states are

Table 3.4-1
Description of Accident Progression Bin Characteristics and Attributes

Attribute Bin	Rebin	Mnemonic	Description
Characteristic 1 - Type of Accident Sequence (ASeq)			
A	A	EQHiG	Seismic initiated accident (High G) (PDS 1)
B	B	EQLoG	Seismic initiated accident (Low G) (PDS 2)
C	C	Fire	Fire initiated accident (PDS 3 - 8)
D	D	Flood	Flood initiated accident (PDS 9 and 10)
E	E	ATWS	Anticipated transient without scram initiated accident (PDS 11 and 12)
F	F	LOCA	LOCA initiated accident (PDS 13)
G	G	Tran	Transient initiated accident (PDS 14 - 24)
H	H	T-LOCA	Transient LOCA initiated accident (PDS 25 - 30)
Characteristic 2 - Time at which Core Damage Occurs (CDTime)			
A	A	ST-1hr	Short-term accidents in which core damage (CD) occurs approximately 1 hour after the initiating event. This category includes short-term SBOs, the ATWS accidents and all other accidents in which all injection is lost shortly after the initiating event.
B	B	IT Fst	Intermediate-term accidents in which CD occurs approximately 13 hours after the initiating event. This category includes sequences that involve delayed diesel generator failure and subsequent loss of injections at 8 hours.

Table 3.4-1 (Continued)
Description of Accident Progression Bin Characteristics and Attributes

Attribute - Bin	Rebin	Mnemonic	Description
Characteristic 2 continued.			
C	C	IT-Slw	Intermediate-term accidents in which CD occurs approximately 18 hours after the initiating event. This category includes sequences in which delayed loss of the the batteries and ADS occurs. In this type of accident the RPV is depressurized before core damage but then repressurizes after the loss of DC power (and the loss of injection).
D	D	Vent	Accidents in which the operators vent the containment and the ensuing severe environment in the reactor building fails the injection systems. Core damage occurs approximately 28 hours after the initiating event.
E	E	F-SRV	Accidents in which low pressure injection systems are initially working, however, the containment heat removal systems are not available. The injection systems are lost when the containment pressure exceeds 85 psig which causes the ADS to reclose and repressurize the RPV. Core damage occurs approximately 39 hours after the initiating event.
F	F	LT-CF	Accidents in which the containment fails, hot gases escape into the reactor building, and the ensuing severe environment in the reactor building fails the injection systems. Core damage occurs approximately 58 hours after the initiating event.
Characteristic 3 - Fraction of Zr Oxidized In-Vessel (ZrOxid)			
A	A	HiZrOx	High - Greater than 21 % of the In-Vessel zirconium has been oxidized before vessel breach
B	B	LoZrOx	Low - Less than 21% of the In-Vessel zirconium has been oxidized before vessel breach

Table 3.4-1 (Continued)
Description of Accident Progression Bin Characteristics and Attributes

Attribute - Bin	Rebin	Mnemonic	Description
Characteristic 4 - Vessel Condition at Vessel Breach (VB)			
A	A	HiP-nLPI	RPV is at high pressure and there is no coolant injection after vessel breach
B	B	LoP-nLPI	RPV is at low pressure and there is no coolant injection after vessel breach
C	C	HiP-LPI	RPV is at high pressure and coolant is being injected after vessel breach
D	D	LoP-LPI	RPV is at low pressure and coolant is being injected after vessel breach
E	E	nVB	There is no vessel breach (i.e., core damage arrest)
Characteristic 5 -			Fraction of Core Participating in DCH or Steam Explosions (DCH-SE)
A	A	Alpha	An Alpha mode event occurred.
B	B	HiDCH	40% of the core participates in DCH
C	C	LoDCH	10% of the core participates in DCH
D	D	HiEXSE	20% of the core participates in ex-vessel steam explosions
E	E	LoEXSE	5% of the core participates in ex-vessel steam explosions
F	F	nDCH-SE	There are no Alpha mode, DCH, or steam explosions events

Table 3.4-1 (Continued)
Description of Accident Progression Bin Characteristics and Attributes

Attribute - Bin	Rebin	Mnemonic	Description
Characteristic 6 - Containment Failure Mode before Vessel Breach (CFBVB)			
A	A	DWHR	Containment fails in the rupture mode at the drywell head.
B	B	DWR	Containment fails by a rupture in the drywell wall.
C	C	WWaWR	Containment fails in the rupture mode in the wetwell above the water line.
D	D	WWbWR	Containment fails in the rupture mode in the wetwell below the water line.
E	E	Vent	The containment is vented.
F	F	DWHL	Containment fails in the leak mode at the drywell head.
G	G	DWL	Containment fails by a leak in the drywell wall.
H	H	WWaWL	Containment fails in the leak mode in the wetwell above the water line.
I	I	WWbWL	Containment fails in the leak mode in the wetwell below the water line.
J	J	nCFBVB	The containment does not fail before vessel breach.

Table 3.4-1 (Continued)
Description of Accident Progression Bin Characteristics and Attributes

Attribute Bin	Rebin	Mnemonic	Description
Characteristic 7 - Containment Failure Mode at Vessel Breach (CFAVB)			
A	A	DWHR	Containment fails in the rupture mode at the drywell head.
B	B	DWR	Containment fails by a rupture in the drywell wall.
C	C	WWaWR	Containment fails in the rupture mode in the wetwell above the water line.
D	D	WWbWR	Containment fails in the rupture mode in the wetwell below the water line.
E	E	DWHL	Containment fails in the leak mode at the drywell head.
F	F	DWL	Containment fails by a leak in the drywell wall.
G	G	CFPipe	Containment fails in the leak mode in the wetwell above the water line. Containment failure is caused by failure of the cavity drain pipe outside the containment wall.
H	G	WWaWL	Containment fails in the leak mode in the wetwell above the water line.
I	H	WWbWL	Containment fails in the leak mode in the wetwell below the water line.
J	I	nCFAVB	The containment does not fail at vessel breach.

Table 3.4-1 (Continued)
Description of Accident Progression Bin Characteristics and Attributes

Attribute - Bin	Rebin	Mnemonic	Description
Characteristic 8 - Containment Failure Mode Late in the Accident (CFLAT)			
A	A	DWHR	Containment fails in the rupture mode at the drywell head.
B	B	DWR	Containment fails by a rupture in the drywell wall.
C	C	DWRPed	Containment fails in the rupture mode at the drywell wall. The failure is caused by late failure of the reactor pedestal.
D	D	WWaWR	Containment fails in the rupture mode in the wetwell above the water line.
E	E	WWbWR	Containment fails in the rupture mode in the wetwell below the water line.
F	F	Vent	The containment is vented.
G	G	DWHL	Containment fails in the leak mode at the drywell head.
H	H	DWL	Containment fails by a leak in the drywell wall.
I	I	WWaWL	Containment fails in the leak mode in the wetwell above the water line.
J	J	WWbWL	Containment fails in the leak mode in the wetwell below the water line.
K	K	nCFLate	The containment does not fail late in the accident.

Table 3.4-1 (Continued)
Description of Accident Progression Bin Characteristics and Attributes

Attribute - Bin	Rebin	Mnemonic	Description
Characteristic 9 - Period in which Containment Sprays Operate (SPRAYS)			
A	A	nSpray	The containment sprays do not operate during the accident
B	B	E-Spry	The sprays only operate before vessel breach (VB)
C	C	L-Spry	The sprays only operate after vessel breach
D	D	E&L-Spry	The sprays both before VB and after VB
Characteristic 10 - Type of Molten Core-Concrete Interactions (MCCI)			
A	A	DryCCI	MCCI occurs in a dry reactor pedestal cavity.
B	B	INJCCI	MCCI occurs in wet cavity with a continuous supply of water.
C	C	NOCCI	MCCI does not occur.

Table 3.4-1 (Continued)

Description of Accident Progression Bin Characteristics and Attributes

Attribute - Bin	Rebin	Mnemonic	Description
Characteristic 11 - Level of SRV Tailpipe Bypass (SRVBY)			
A	A	nSRVBY	All of the in-vessel releases pass through the SRV tailpipes and are discharged into the suppression pool.
B	B	SRVnVB1	Partial bypass of the SRV tailpipe. A fraction of the in-vessel releases escape directly into the drywell, the remaining fraction are released into the suppression pool. There are no stuck open drywell vacuum breakers. Thus, the radionuclides in the drywell will enter the suppression pool through the downcomers (assuming the drywell has not failed).
C	C	SRVnVB2	All of the in-vessel releases escape directly into the drywell (i.e., during a LOCA). There are no stuck open drywell vacuum breakers (see attribute B).
D	D	SRVoVB1	Partial bypass of the SRV tailpipe. A fraction of the in-vessel releases escape directly into the drywell, the remaining fraction are released into the suppression pool. In addition to a partial bypass of the SRV tailpipe, a drywell vacuum breaker is stuck open. Thus, the drywell releases can enter the suppression chamber air space without going through the suppression pool.
E	E	SRVoVB2	All of the in-vessel releases escape directly into the drywell (i.e., during a LOCA). Furthermore, a drywell vacuum breaker is stuck open (see attribute D).

Table 3.4-1 (Concluded)
Description of Accident Progression Bin Characteristics and Attributes

Attribute - Bin	Rebin	Mnemonic	Description
Characteristic 12 - Level of Suppression Pool Bypass (SPBY)			
A	A	CAVAVB	The reactor cavity floor fails at the time of vessel breach or shortly after vessel breach.
B	B	CAVLAT	The reactor cavity floor fails late in the accident (several hours after vessel breach).
C	C	NOSPBY	The reactor cavity floor does not fail during the accident.
Characteristic 13 - Level of Reactor Building Bypass (RBBY)			
A	A	nRBBY	There is only nominal reactor building bypass.
B	A	RBB2	There is a small partial bypass of the reactor building.
C	B	RBB3	There is a large partial bypass of the reactor building.
D	B	RBB4	The reactor building is completely bypassed.
Characteristic 14 - Occurrence of a Station Blackout (SBO)			
A	A	SBO	A station blackout occurs during the accident.
B	B	nSBO	The accident is not a station blackout.

termed attributes. Most of this information was needed by the LASSOR code to calculate the fission product source terms. The remainder of this section consists of a brief description of each characteristic and an explanation of an example bin.

Characteristic 1 summarized the groups of accident sequences or plant damage states (PDSs) that were propagated through the APET. Eight attributes were defined. These attributes were based on the initiating events that are associated with the accident sequences that make up the PDSs. The initiating events included: seismic, fire, flood, ATWS, LOCA, transients, and transient LOCAs.

Characteristic 2 addressed the time at which core damage occurred. Six attributes (times) were considered for this characteristic. The attributes depended mainly on the length of time that coolant was supplied to the RPV. The first group included accidents in which coolant was lost immediately (i.e., short-term accidents). The second group consisted of accidents that involved failure of the station batteries or failure of the diesel generators to continue to run and the RPV not repressurizing before vessel breach. The third group consisted of accidents that involved failure of the station batteries or failure of the diesel generators to continue to run and the RPV repressurizing before vessel breach due to loss of DC power. Repressurization of the RPV delays the onset of core damage. The fourth group included long-term containment heat removal accidents in which the operators vented the containment when it reached 60 psig. The ensuing severe environment in the reactor building failed vital injection systems. Accidents in which low pressure injection systems were operating until high containment pressure forced the reclosure of the ADS form the fifth group. The low pressure injection systems were lost when the RPV repressurizes. The sixth group included long-term heat removal accidents that resulted in containment failure and subsequent failure of injection systems due to severe environments in the reactor building.

Characteristic 3 addressed the fraction of in-vessel zirconium that was oxidized before vessel breach. There were two possible values for this characteristic: low and high. The demarcation point between the two ranges is 21%.

Characteristic 4 addressed the RPV pressure before vessel breach and the availability of coolant injection at vessel breach; there were five possibilities, including no vessel breach. The RPV could either be at high or low pressure before vessel breach. High pressure is system pressure (i.e., approximately 1000 psia) and low pressure is less than 200 psia. There were two possibilities for coolant injection: coolant was being injected into the RPV at or immediately after vessel breach or coolant was not being injected into the RPV at or immediately after vessel breach.

Characteristic 5 addressed energetic events that could occur at the time of vessel breach. The events considered by this characteristic were alpha mode events (i.e., large in-vessel steam explosions that fail both the RPV and the containment), direct containment heating (DCH), and ex-vessel steam explosions. In addition to alpha mode, five other attributes were defined that addressed the fraction of core participating in DCH or an ex-vessel steam explosion. There were two levels for DCH: low (10% of the core) and high (40% of the core). Similarly, there were two levels for steam explosions: low (5% of the core) and high (20% of the core). The fifth attribute was for the case with no DCH events or ex-vessel steam explosions. The following hierarchy was applied when more than one of these events occur during the accident. If an alpha mode event occurred, regardless of whether a DCH or an ex-vessel steam explosion occurred, the alpha mode event was chosen. Next, if a DCH event and an ex-vessel steam explosion both occurred during an accident, the attribute associated with the DCH event was assigned to this characteristic. The following rational was used for this hierarchy. The alpha mode event involved failure of both the RPV and the drywell head as well as an energetic release of core debris. The radionuclide releases associated with a DCH event were greater than the releases associated with an ex-vessel steam explosion. Thus, alpha mode failure was chosen over both DCH and ex-vessel steam explosions and DCH was chosen over an ex-vessel steam explosion.

Characteristic 6 addressed the mode of containment failure before vessel breach. Failures that occurred before core damage as well as failures that occur during core damage were included in this characteristic. The only credible mechanism that was identified that would result in containment failure during core damage was slow pressurization of the containment caused by the accumulation of steam and noncondensibles in the drywell and wetwell. Thus, containment failure before core damage was not exacerbated by events during core damage (i.e., a leak would arrest the slow pressure rise). There were ten attributes. Four locations were considered: drywell head (i.e., failure into the refueling floor), the drywell wall, the wetwell wall above the suppression pool, and the wetwell wall below the suppression pool surface. For each location, the mode of containment failure could be either a leak or a rupture. The remaining two attributes were venting and no containment failure before vessel breach.

Characteristic 7 addressed the mode and location of containment failure at the time of vessel breach. There were ten attributes. The four locations that were considered for failures before vessel breach were again addressed. For each location, the mode of containment failure could be either a leak or a rupture. In addition to these four locations, failure of the reactor cavity drain pipe was also addressed. The isolation valves for the drain pipe are located in the reactor building (i.e., outside the containment). Thus, failure of either the valves or the section of this pipe that is outside the containment will result in a pathway for radionuclides to escape the containment. Because the drain line is a four inch pipe, this failure mode is a leak. The remaining attribute was no containment failure at vessel breach.

Characteristic 8 addressed the mode of containment failure late in the accident. There were eleven attributes. The four locations that were considered for failures before vessel breach were again addressed. For each location, the mode of containment failure could be either a leak or a rupture. An additional attribute was defined for failures that result from late failure of the reactor pedestal and the resultant motion of the RPV (i.e, penetration failure caused by gross motion of the RPV). Pedestal failures that occur between one and six hours after VB were included in this attribute. This mode was distinguished from drywell failure because of timing differences between this mode and failures caused by late overpressurization. The remaining two attributes were venting and no late containment failure.

Characteristic 9 addressed the period in which containment sprays operate; there were four attributes. Two time periods were addressed: early and late. The early time period included spray operation before VB whereas the late time period included spray operation after VB (i.e., during CCI). Spray operation during the very late time period was only considered for accidents in which the containment failed after the very late recovery of AC power. The four possibilities were: no containment sprays, only early containment sprays, only late containment sprays, and early and late containment sprays.

Characteristic 10 addressed core-concrete interactions (CCI). Three attributes addressed the following cases: dry CCI, CCI under water, and no CCI. Accidents in which the CCI proceeded in either a dry cavity or an initially wet cavity with no replenishable source of water were included in the first attribute, dry CCI. Accidents which involved an initially coolable debris bed but which did not have a replenishable source of water were also identified with this attribute. Accidents in which a continuous source of water was injected onto the debris were addressed by the second attribute, flooded CCI. The third attribute included core damage arrest accidents (i.e., no VB) and accidents in which the core debris was coolable and there was a continuous source of water.

Characteristic 11 addressed whether the in-vessel release went directly to the suppression pool or first into the drywell and then into the suppression pool. Depending on the time and location of containment failure, this had a significant effect on the magnitude of the release. There were five attributes. Two levels of SRV tailpipe bypass were considered: partial bypass and complete bypass. In the partial bypass cases, a fraction of the in-vessel released escaped directly into the drywell and the remaining fraction passed through the tailpipe and was released into the suppression pool. All of the in-vessel releases escaped directly into the drywell when a complete bypass occurred. The in-vessel release could also bypass the suppression pool during a LOCA. For each tailpipe bypass level and for the LOCA, two levels of drywell vacuum breaker failure were considered: partial and complete. Partial failure of the drywell vacuum breakers means that a fraction of the releases in the drywell will pass through the downcomer and be released into the pool. Complete failure of the vacuum breakers means that the drywell releases will enter the suppression chamber air through the vacuum breaker rather than through the downcomer and the suppression pool. It should be noted that for this analysis no credible cases could be identified in which there would be simultaneous SRV tailpipe bypass or LOCA and drywell vacuum breaker failure. Accidents in which there was no bypass were identified by the remaining attribute. It should be noted that this bypass occurs before vessel breach.

Characteristic 12 addressed the level of suppression pool bypass after vessel breach. This was primarily determined by the failure of the reactor cavity floor. There were three attributes that addressed the timing of cavity floor failure. The first attribute was cavity floor failure at the time of vessel breach or shortly after vessel breach. Cavity failure from quasi-static loads accompanying vessel breach, dynamic loads from ex-vessel steam explosions, and thermal attack from the core debris on the drain pipes were all included in this attribute. The second attribute addressed cavity failure late in the accident from core debris interactions with the cavity floor and drain pipes. The last attribute included the cases where the cavity floor did not fail. This occurs for core damage arrest accidents and scenarios in which the core debris was cooled in a flooded cavity.

Characteristic 13 addressed the level of reactor building bypass or failure. There were four attributes which identified the following cases: no reactor building bypass, level two reactor building bypass (leak), level three reactor building bypass (rupture), and complete reactor building bypass. Reactor building failure resulted from pressure loads accompanying containment failure. These loads were a result of the containment blowdown into the reactor building or hydrogen burns in the reactor building following containment failure.

Characteristic 14 addressed the occurrence of a station blackout. The first attribute included all of the accidents that were classified as a station blackout at the time of core damage. For LaSalle, a station blackout was defined as loss of offsite power and loss of trains A and B emergency power. The second attribute identified those accidents that were not station blackouts.

A typical bin might be ACAACJBKAAAAAA which, using the information presented in Table 3.4-1, is:

A	EQHiG	High G seismic initiated accident (PDS 1)
C	IT-Slw	Core damage occurs after delayed failure of injection, batteries and ADS.
A	HiZrOx	A large fraction of the Zr was oxidized in-vessel
A	HiP-nLPI	The RPV was at high pressure before vessel breach and there was no injection to the RPV after vessel breach
C	LoDCH	A small fraction of the core participated in a direct containment heating event (DCH)
J	nCFBVB	The containment did not fail before vessel breach
B	DWR	The drywell was ruptured at the time of vessel breach
K	nCFLate	The containment did not fail during the late time regime
A	nSpray	Drywell sprays were not used during the accident.
A	DryCCI	CCI proceeded in a dry reactor cavity

A	nSRVBY	All of the in-vessel releases passed through the suppression pool and there were no stuck open drywell vacuum breakers
A	CAVAVB	The reactor cavity floor fails at the time of vessel breach or shortly there after
A	nRBBY	The reactor building was not bypassed
A	SBO	The accident is a station blackout

3.4.2 Rebinning

The binning scheme utilized for the evaluation of the APET does not necessarily have to exactly match the input information required by the LASSOR code. The additional information in the initial binning was kept because it provided a better record of the outcomes of the APET evaluation. Therefore, there was a step between the evaluation of the accident progressions by the APET and the evaluation of the source terms by the LASSOR code known as "rebinning". In the rebinning, a few attributes in some characteristics were combined because there are no significant differences between them for calculating the fission product releases.

In the rebinning for LaSalle, only two characteristics were changed. For Characteristic 7, containment failure mode at vessel breach, Attribute G, cavity drain pipe failure, was included in Attribute H, containment leak in the wetwell above the suppression pool. For either release path, the reactor cavity floor has almost certainly failed and the radionuclides would bypass both the drywell and the suppression pool and enter the reactor building. Thus, LASSOR would not make a distinction between these two release paths. For Characteristic 13, level of reactor building bypass, the four levels were reduced to two levels: small bypass and large bypass. The letters for the rebinner attributes are also shown in Table 3.4-1.

3.4.3 Summary Bins for Presentation

For presentation purposes, a set of "summary" bins was defined. These summary bins are also used to display risk results in Chapter 6. Instead of the 14 characteristics and thousands of possible bins that describe the evaluation of the APET in detail, the summary bins place the outcomes of the evaluation of the APET into a few, very general groups.

In the summary binning scheme there are three characteristics: vessel breach (VB), containment failure time, and reactor pressure vessel (RPV) pressure. Each of these characteristics and their associated attributes are defined in Table 3.4-2. The eight summary bins may now be defined as follows:

VB, Early CF, RPV at low pressure

Vessel breach occurs, the containment fails either before or at the time of vessel breach, and the reactor pressure vessel is at low pressure at the time of vessel breach.

Table 3.4-2
Description of Summary Accident Progression Bin Characteristics

Attribute	Description
Characteristic 1. Vessel Breach (VB)	
VB	Vessel breach occurs
No VB	Vessel breach does not occur.
Characteristic 2: Containment Failure Time (CF)	
Early CF	The containment fails either before or at the time of vessel breach from the development of a leak or a rupture.
Late CF	The containment fails during the late time period from the development of either a leak or a rupture.
Vent	The containment is vented either before core damage, during core damage, or late in the accident.
No CF	The containment does not fail.
Characteristic 3: Reactor Pressure Vessel (RPV) Pressure	
Low Pressure	The RPV is at low pressure (i.e., less than 200 psia) just prior to vessel breach.
High Pressure	The RPV is at high pressure (i.e., approximately 1000 psi) at the time of vessel breach.

VB, Early CF, RPV at high pressure

Vessel breach occurs, the containment fails either before or at the time of vessel breach, and the reactor pressure vessel is at high pressure at the time of vessel breach.

VB, Late CF

Vessel breach occurs and the containment fails late in the accident (i.e., hours after vessel breach).

VB, Early or Late Venting

Vessel breach occurs and the containment is either vented before vessel breach or late in the accident.

VB, no CF

Vessel breach occurs; however, the containment neither fails nor is vented during the accident.

No VB, CF

The core damage process is arrested (i.e., no vessel breach); however, the containment still fails during the accident due to the generation of steam and noncondensibles during the accident.

No VB, Venting

The core damage process is arrested before vessel failure. However, the containment is vented either before the onset of core damage or during the core damage process.

No VB, No CF, No Venting

The core damage process is arrested and the containment remains intact.

In addition to these eight summary APBs that were used to present results throughout the analysis (i.e., accident progression, source term results and risk analyses), additional groups of APBs were defined which were used to present the accident progression results in section 3.5. These summary bins are defined in Table 3.4-3.

3.5 Results of the Accident Progression Analysis

This section presents the results of evaluating the APET. As evaluating the APET produces a number of accident progression bins (APBs), the discussion is primarily in terms of APBs. However, the propagation of the PDSs through the APET resulted in thousands of APBs and, therefore, it is

Table 3.4-3
Description of Summary Accident Progression Bins Used to Present Results in Section 3.5

Summary APB	Description
Accident Progression Bins Used in Figure 3.5-3	
CF-DWH	Failure of the containment in the drywell head before vessel breach.
CF-DW	Failure of the containment in the drywell wall before vessel breach.
CF-WWaW	Failure of the containment in the wetwell above the suppression pool before vessel breach.
CF-WWbW	Failure of the containment in the wetwell below the pool level before vessel breach.
CF-Vent	The containment is vented before vessel breach.
nCFBVB	The containment does not fail before vessel breach.
Accident Progression Bins Used in Figure 3.5-4	
CF-DWH	Failure of the containment in the drywell head at vessel breach.
CF-DW	Failure of the containment in the drywell wall at vessel breach.
CF-WWaW	Failure of the containment in the wetwell above the suppression pool at vessel breach.
CF-WWbW	Failure of the containment in the wetwell below the pool level at vessel breach.
nCFAVB	The containment does not fail at vessel breach.

Table 3.4-3 (Continued)
Description of Summary Accident Progression Bins Used to
Present Results in Section 3.5

Summary APB	Description
Accident Progression Bins Used in Figure 3.5-5	
CF-DWH	Failure of the containment in the drywell head after vessel breach.
CF-DW	Failure of the containment in the drywell wall after vessel breach.
CF-WWaW	Failure of the containment in the wetwell above the suppression pool after vessel breach.
CF-WWbW	Failure of the containment in the wetwell below the pool level after vessel breach.
CF-Ped	Failure of the pedestal after vessel breach results in a drywell rupture.
CF-Vent	The containment is vented after vessel breach.
nCF-Late	The containment does not fail after vessel breach.
Accident Progression Bins Used in Figure 3.5-6	
CF-BVB	The containment fails before vessel breach.
E-Vent	The containment is vented before vessel breach.
CFVB-WWaW-Leak	The containment fails at vessel breach. The failure is a leak in the wetwell above the suppression pool.
CF-VB	The containment fails at vessel breach. The failure mode is not a leak in the wetwell above the suppression pool.
LCF	The containment fails late in the accident.
L-Vent	The containment is vented after vessel breach.
No CF	The containment neither fails nor is vented during the accident.

Table 3.4-3 (Concluded)
Description of Summary Accident Progression Bins Used to Present Results in Section 3.5

Summary APB	Description
Accident Progression Bins Used in Figure 3.5-7	
Dry CCI	Core-concrete interactions proceed in a dry cavity.
CCI w/ Injection	Core-concrete interactions occur with an overlaying pool of water. The pool of water is maintained by injection from the vessel.
VB, No CCI	The vessel fails, however, the core debris is cooled and CCI is avoided.
No VB	The core damage process is arrested and vessel breach is avoided.

not practical to discuss each APB individually. Rather, groups of APB or summary APBs will be discussed as well as key events that occur during the progression of the various accidents (e.g., containment failure).

Thirty PDSs were propagated through the APET. For presentation purposes these thirty PDSs have been grouped into 7 summary PDSs. The 7 summary PDSs are: seismic, fire, flood, ATWS, LOCA, transients, and transient LOCAs. The summary accident progression bins that are used in this discussion were defined in section 3.4.3. In addition to the summary APBs, three key events in the accident are discussed. These events are the core damage arrest, early containment failure, and the status of core-concrete interactions. These three events are critical in determining the risk to the offsite population. Therefore, this discussion will begin with these events.

3.5.1 Core Damage Arrest and Avoidance of Vessel Breach

Once core damage has begun, the only way vessel failure can be prevented is if coolant injection is restored to the RPV. Restoration of coolant injection to the RPV, however, does not necessarily preclude vessel breach. If injection is not recovered until late in the core damage process, it is unlikely that the addition of water will prevent vessel breach. In addition, there is the possibility that the core debris that slumps into the bottom head of the vessel will trigger a steam explosion that may fail the vessel. Although steam explosions do not always result in vessel failure, they do pose a significant challenge to the integrity of the RPV.

For example, from the APET, the probability that an in-vessel steam explosion occurs during core damage when the RPV has been depressurized is 0.86. The probability that this steam explosion will fail the vessel is 0.7. These values are independent of the status of injection. Thus, 60% of the time the vessel will fail regardless of whether injection is restored to the core. If injection is recovered, the APET shows that the probability that the core damage process will be arrested and, thus, avert vessel breach is approximately 0.75 (assuming the vessel did not fail from a steam explosion). Therefore, the overall probability that vessel breach will be averted given that injection is restored to a depressurized vessel is only 0.30. Figure 3.5-1 shows the probability that core damage is arrested before the lower head of the vessel fails for the seven summary PDSs groups.

The distributions displayed in Figure 3.5-1 are conditional on the occurrence of core damage. The distributions in this plot are vertical histograms displayed on a log scale. The sample size used in this analysis is 400 and, thus, there are potentially 400 values for the conditional probability of core damage arrest for each PDS group. However, the frequency of some of the PDS groups for certain observations is zero. This results from treating the containment failure as a zero/one variable (i.e., it either fails or does not fail for each observation) in the accident frequency analysis. The statistical measures (i.e., 5th percentile,

Figure 3.5-1. Distributions of Probability for Core Damage Arrest Conditional on the Occurrence of a Summary PDS Group.

median, mean, and 95th percentile) presented in this figure are based on the number of observations that had a non-zero value for the PDS frequency. That is, if a particular PDS had non-zero frequencies in 300 observations, then the statistics are based on a sample size of 300. The width of the boxes on the histogram is proportional to the number of observations that have probabilities that fall within the range defined by the height of the box. The eighth distribution in this figure labeled F.W.A. is the frequency weighted average of the seven other PDS groups in the figure. This frequency weighted average is done on an observation by observation basis. For each observation, the conditional probability of core damage arrest for a each PDS group is multiplied by the ratio of the PDS group's frequency to the total core damage frequency (these frequencies are for the particular observation that is being considered and are not mean frequencies).

For both the seismic PDS group and the LOCA PDS group, the core damage process is never arrested and vessel breach always occurs. The seismic PDSs are station blackouts (SBO). For the seismic SBOs, AC power cannot be recovered during the accident. Thus, injection cannot be restored to the RPV and vessel breach is assured. The LOCA PDS group consists of one PDS which involves long-term loss of containment heat removal. The pressurization of the containment from the boiling suppression pool results in containment failure. The ensuing severe environment in the reactor building fails the injection systems. The main feedwater system and the condensate system are not available because of a previous failure. Thus, the only injection system that is available is diesel-driven firewater system. However, once the containment has failed and core damage has begun, it is assumed that the operators will have insufficient time to align the DFWS before vessel breach occurs. It must be remembered that the reactor building will be flooded with steam after containment failure to the reactor building and the radiation level in the reactor building will also be increasing during core damage. Thus, vessel breach is assured for the LOCA PDS group.

For the fire PDS group, the mean value for the conditional probability of core damage arrest is 0.014 which is a fairly low value. The reason for this low value stems from the definitions of the PDSs that form the fire PDS group. The fire PDS group consists of six PDSs. In only one of these six PDSs, namely F11, can injection be recovered. This PDS contributes 17% of the f e groups conditional pr ability of core damage. Thus, the combination of a small likelihood that injection will be recovered with the fact that injection does not necessarily preclude vessel breach results in the low [illegible] for the conditional probability of core damage arrest.

For both the flood and the ATWS PDS groups, the likelihood of core damage arrest is considerably higher than for the fire PDS group because of the greater availability of injection systems. The mean values for the conditional probability of core damage arrest for the flood and the ATWS PDS groups are 0.27 and 0.24, respectively. The median value is closer to 0.1 for both of these PDS groups. In the flood PDS group, the diesel-driven firewater pumps are available. In this PDS group, it is likely that the operators will depressurize the RPV to allow firewater to be injected in the core. Furthermore, there is a high likelihood that the containment pressure will remain low either through the use of the containment heat removal systems or by venting. Thus, it is highly likely that the firewater system will be used to inject water into the core during core damage. In the ATWS PDS group, the only injection systems that can be recovered are the main feedwater system and the condensate system. The mean conditional probability of recovering one of these systems is about 0.5.

The mean conditional probabilities of core damage arrest for the Transient PDS group and the Transient LOCA PDS group are 0.17 and 0.10, respectively. Both of these PDS groups consist of short-term station blackouts (SBO), long-term SBOs, and long-term loss of containment heat removal accidents. For the short-term SBO PDSs, the probability of core damage arrest is driven by the probability that AC power is recovered during core damage. For the long-term station blackout PDSs, it is assumed that if AC power has not been recovered by the time of core damage (>27 hours) it will not be recovered during the accident. This assumption is based on the following rationale. The accident frequency analysis considered AC power recovery out to approximately 27 hours. Beyond this time, the probability of random failures of systems that could be recovered becomes important and limits the amount of credit that can be given to AC recovery. Furthermore, the data from which the [illegible] curves were generated only extend to 8.9 hours. Thus, the applicability of extrapolating the power recovery data beyond 27 hours is very questionable. Therefore, in the accident progression analysis, no credit is given for recovering AC power during long-term SBO accidents. For the long-term loss of containment heat removal accidents, coolant injection cannot be recovered before vessel failure either because all of the injection systems have failed or because the containment pressure is too high and low pressure injection systems cannot be used (if the containment pressure exceeds 85 psig the SRVs reclose and the RPV repressurizes).

3.5.2 Early Containment Failure

Early fatality risk depends strongly on the integrity of the containment early in the accident. The integrity can be compromised by either early containment failure (CF) or by venting the containment during core damage. Early containment failure is defined as failure of the containment either at the time of vessel breach or before vessel breach. Before vessel breach, the event that threatens the integrity of the containment is the slow pressurization of the containment caused by the generation of steam from a boiling suppression pool during accidents in which containment heat removal is lost or inadequate (i.e., long-term loss of containment heat removal accidents, long-term station blackout accidents, and ATWS accidents). At the time of vessel breach, there are several events that can result in containment failure. The first event is an alpha mode event. An alpha mode event is a large in-vessel steam explosion that results in the failure of the reactor vessel and the drywell head. The second event is the fast pressurization of the containment caused by the loads accompanying vessel breach. These loads include contributions from direct containment heating (DCH), ex-vessel steam explosions, and reactor vessel blowdown. The third event is failure of the reactor pedestal caused either by quasi-static pressurization loads accompanying vessel breach (i.e., DCH, ex-vessel steam explosions, or RPV blowdown) or by dynamic loads associated with an ex-vessel steam explosion. It is postulated that failure of the reactor support and subsequent gross motion of the RPV will place a large amount of stress on the piping penetrations. Given that the pedestal fails, there is some probability that the penetrations will fail and a failure in the drywell wall will result. The last event that can breach the integrity of the containment is an ex-vessel steam explosion that fails the cavity drain pipe beyond the containment wall.

Late in the accident, the events that result in containment failure include the slow pressurization of the containment from the steam and noncondensible gases generated during CCI and failure of the reactor pedestal caused by concrete erosion during CCI. Noticeable omissions from this list are hydrogen combustion events (which are important events for a BWR Mark III containment, such as at Grand Gulf[9]) and drywell meltthrough (a dominant mode of containment failure at Peach Bottom,[10] a Mark I containment). Because the LaSalle containment is inerted with nitrogen, hydrogen combustion events are not possible and, therefore, are not an issue at LaSalle. The reactor cavity at LaSalle is large enough to contain all of the core debris that is released at vessel breach and, thus, attack of the drywell wall from core debris is also not an issue at LaSalle.

Although not considered as a containment failure, venting can also establish a path for radionuclides to escape the containment. The operators are instructed to vent the containment when the containment pressure exceeds 60 psig. Thus, in this analysis, if AC power is available and the vent valves are operable there is a high likelihood that the operators will vent the containment when the pressure reaches 60 psig. Early venting is only a concern during long-term loss of containment heat removal accidents and during ATWS accidents. The containment pressure before vessel breach is low during a short-term accident because the suppression pool is subcooled and, thus, venting is not required. For a short-term accident the operators may vent the containment after vessel breach due to containment pressurization caused by steam and noncondensibles generated during core-concrete interactions (CCI).

Figure 3.5-2 shows the probability distribution for early CF at LaSalle. The probability distributions displayed in this figure are conditional on core damage. These distributions do not, however, include accidents in which the containment is vented. When all of the accidents are considered together (i.e., the frequency weighted average (F.W.A.) column) the mean conditional probability of early containment failure is 0.33. The events that cause these failures are described in the following paragraphs.

Figure 3.5-3 shows the mean probability of containment failure before vessel breach sorted by the various failure locations. Each location

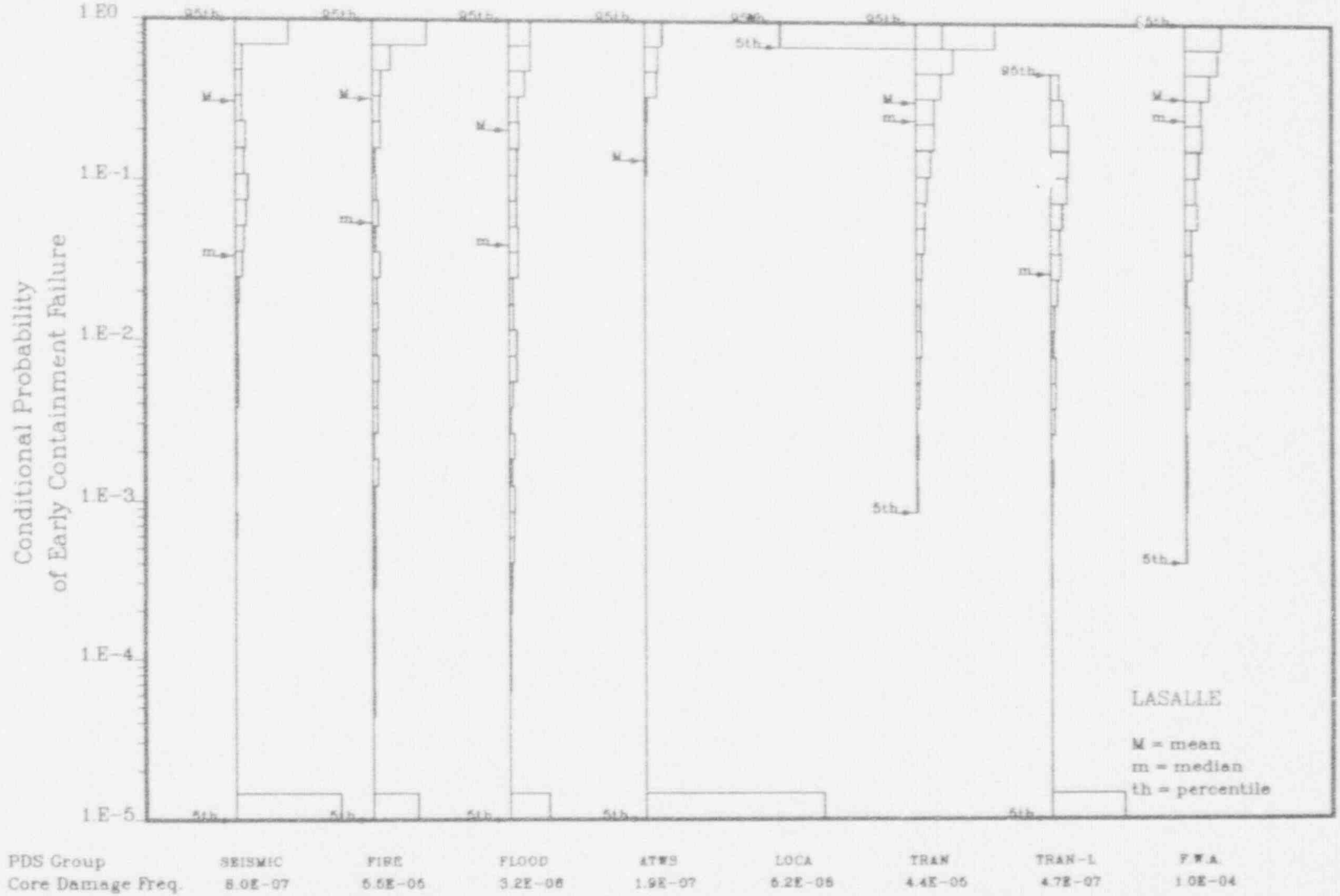

Figure 3.5-2. Distributions of Probability for Early Containment Failure Conditional on the Occurrence of a Summary PDS Group.

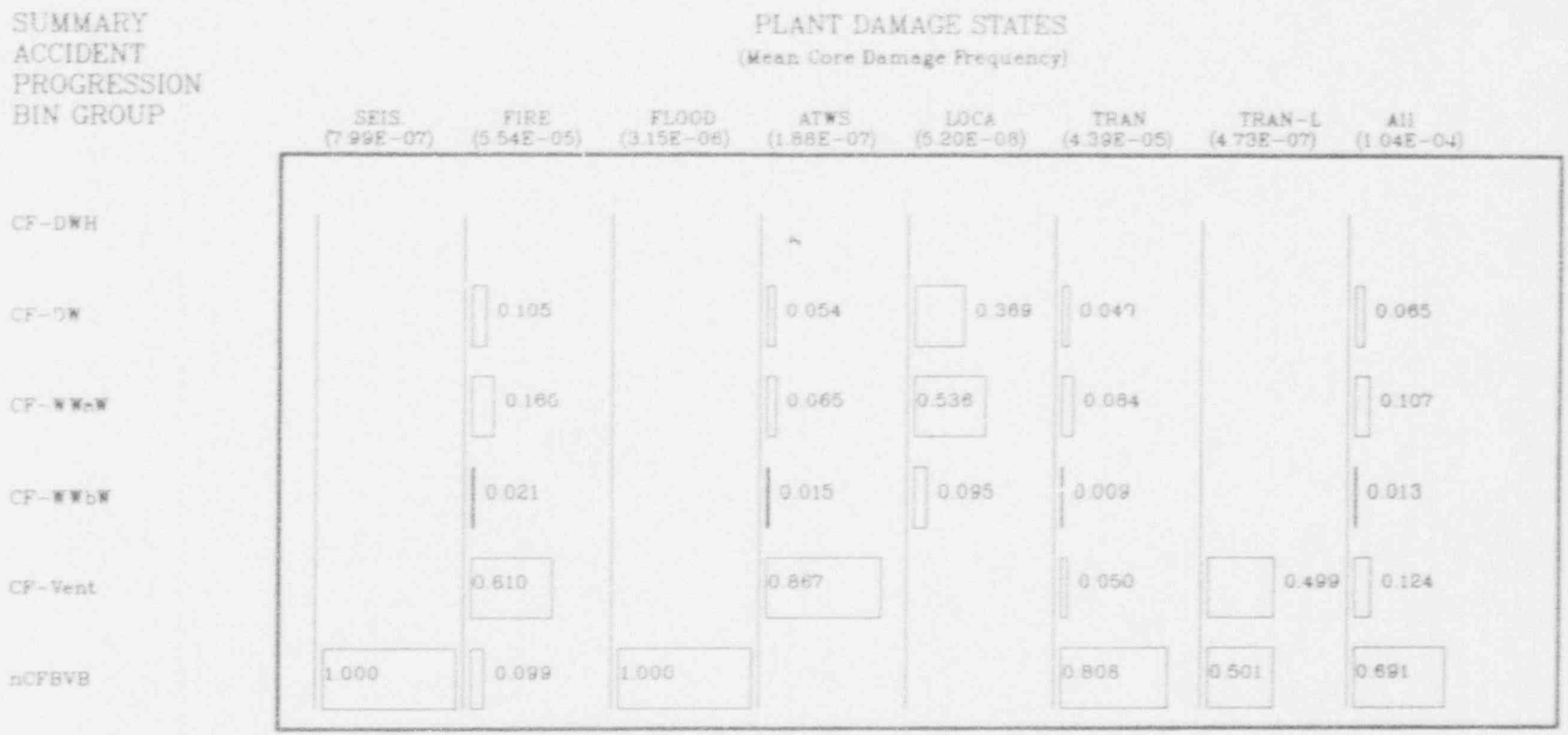

Figure 3.5-3. Mean Conditional Probability of Containment Failure Modes Before Vessel Breach.

includes both leaks and ruptures except for venting which is equivalent to a rupture. Figures 3.5-4 and 3.5-5 present the same type of information for containment failures at vessel breach and late containment failures, respectively. These figures only show the failures that occur during the given time regime (e.g., before vessel breach). Thus, if a drywell rupture occurs before vessel breach, this failure will only show up in Figure 3.5-3; it will not appear as a drywell failure in either Figure 3.5-4 or Figure 3.5-5. It should be noted, however, that multiple failure locations can occur at LaSalle. For example, it is possible that a given accident progression path can have a leak in the drywell before vessel breach (caused by a slow pressurization of the containment) and a rupture in the drywell head at vessel breach (cause by Alpha mode event). For this example, the drywell leak would be included in Figure 3.5-3 and the drywell rupture would be included in Figure 3.5-4.

Figure 3.5-6 presents a summary of the containment failures for the following time regimes: before vessel breach, at vessel breach, and late in the accident. The failures at vessel breach have been separated into two groups: wetwell leaks above the suppression pool and all other failures at vessel breach. The cavity drain pipe failures are included in the wetwell leaks above the suppression pool. Early and late venting have also been separated into two groups. Early venting includes venting before core damage as well as venting during the core degradation process. In this figure each accident progression path is only assigned to one time regime regardless of the number of containment failure modes. For classification of APBs, the order that the various modes are considered is the reverse from the order presented in this figure (e.g., No CF is the first mode considered, then late venting and so on). This is because an APB can have several containment failure modes and only the mode most important to determining the source term is considered for the final classification.

For the seismic PDSs, the containment always fails during the course of the accident. The probability of early containment failure is 0.30. All of these early containment failures occur at the time of vessel breach. The containment is not vented in any of these accidents because AC power is not recoverable for the seismic PDSs.

The mean conditional probability of early containment failure for the Fire PDSs is also approximately 0.30; nearly all of these failures occur before core damage. However, for the Fire PDSs, the mean probability of early containment venting is 0.61. Thus, for the Fire PDSs the mean probability that the containment integrity is lost early in the accident is 0.93.

The Flood PDS group consist of two PDSs. Both of these PDSs involve accidents in which all injection is lost at the beginning of the accident which result in a subcooled suppression pool and a low containment pressure during the early portion of the accident. Thus, neither containment failure nor venting occurs before vessel breach for these PDSs. The probability of early containment failure for this PDS group is 0.20; all of these failures occur at the time of vessel breach. It is interesting to note that the probability of late venting is 0.78. This high likelihood of

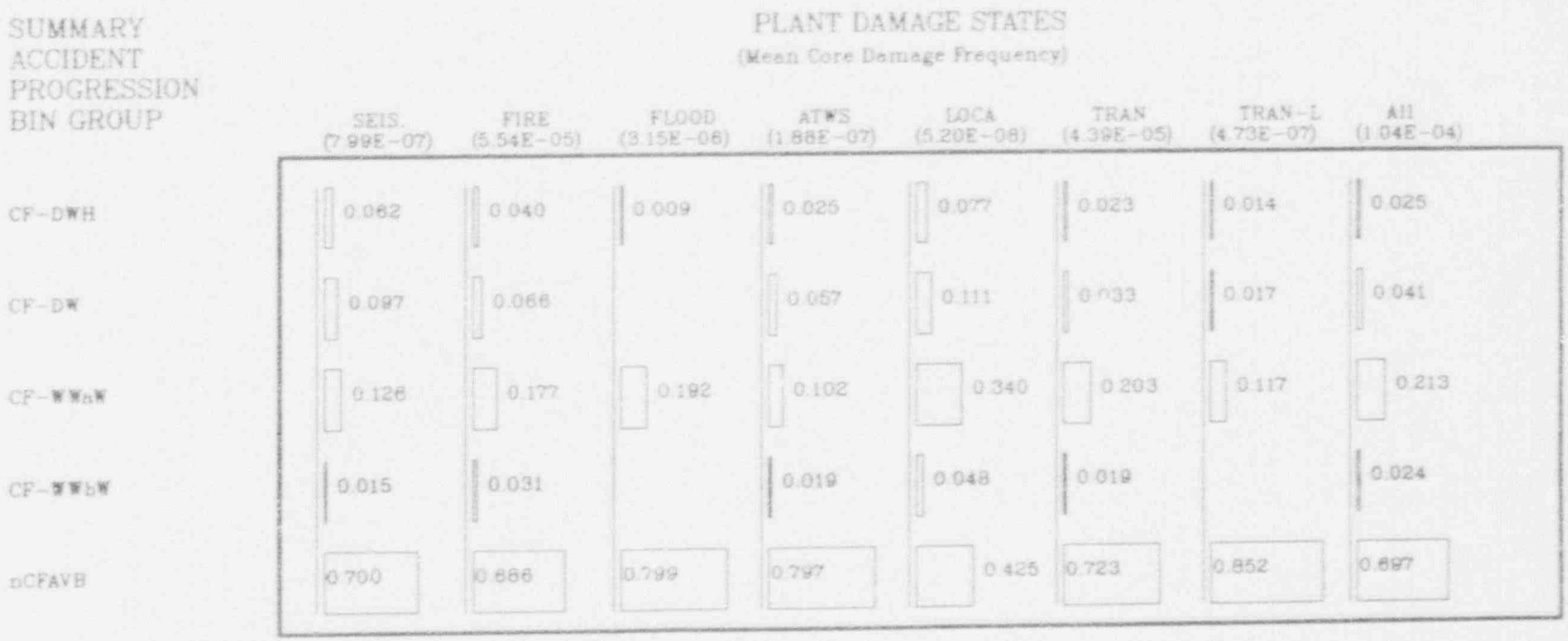

Figure 3.5-4. Mean Conditional Probability of Containment Failure Modes At Vessel Breach.

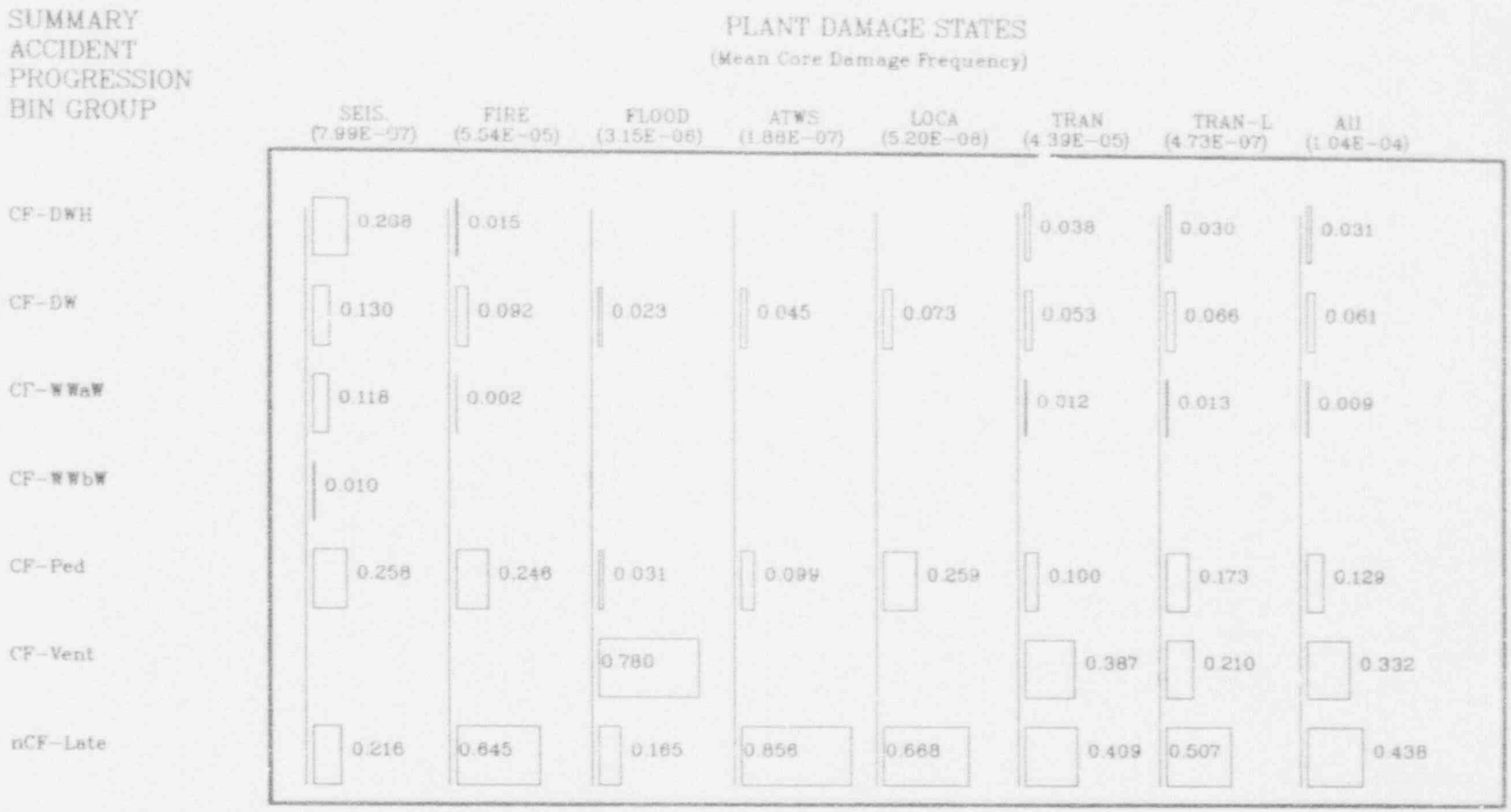

Figure 3.5-5. Mean Conditional Probability of Containment Failure Modes Late in the Accident (i.e., After Vessel Breach).

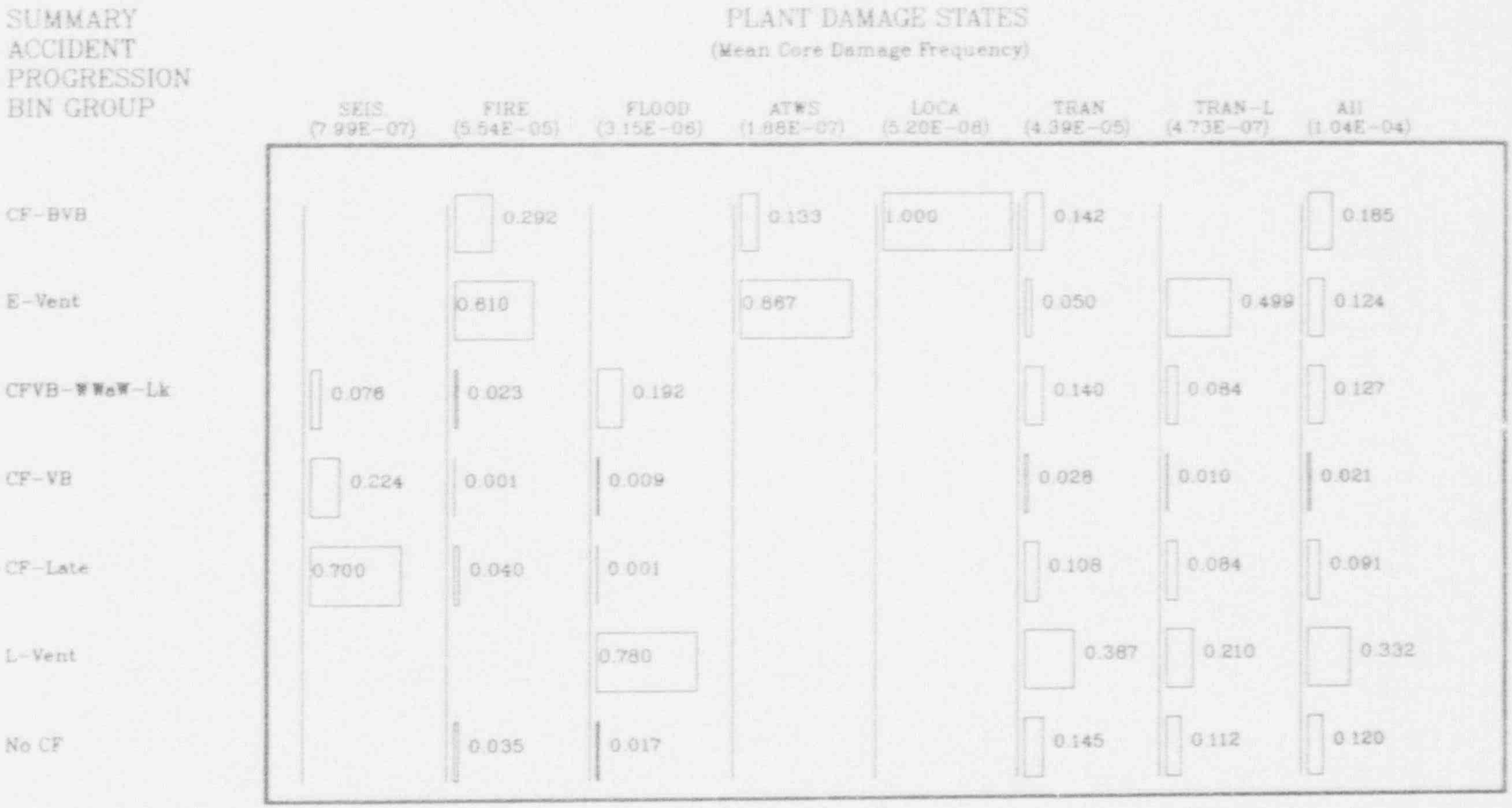

Figure 3.5-6. Mean Conditional Probability of Containment Failure Times.

late venting can be attributed to the fact that in this PDS group AC power is always available and the vent valves are operable. Thus, during the course of the accident it is very likely that the containment integrity will be lost either due to containment failure or late containment venting.

For both the ATWS PDS group and the LOCA PDS, the containment integrity is always lost before core damage. The ATWS PDS group consists of two PDSs. In the first PDS, the containment fails from the rapid generation of steam in the containment. In the second PDS, the containment is vented before core damage in an attempt to relieve the pressure in the containment. The LOCA PDS involves long-term loss of containment heat removal which eventually results in containment failure. The venting system is not available for this PDS.

For the Transient PDS group, the mean conditional probabilities of early containment failure and early venting are 0.31 and 0.05, respectively. About half of the early containment failures occur before vessel breach. This PDS group consists of short-term SBOs, delayed SBOs (injection works for approximately 8 hours), long-term SBOs, and long-term loss of containment heat removal accidents. For the long-term accidents the containment either fails or is vented before core damage. For the short-term and delayed SBOs early containment failure is caused by events at vessel breach. For the Transient PDS group the mean conditional probability of late venting is 0.39. This PDS group is dominated by short-term SBO. Before vessel breach the containment pressure is low and early venting is not required. There is a high likelihood that AC power will be recovered after core damage but before containment failure and the venting system is recoverable with restoration of AC power. Thus, it is likely that the operators will vent the containment late in the accident in an attempt to reduce the pressure caused by the gases released during CCI.

The mean conditional probability of early containment failure for the Transient LOCA PDS group is 0.085; all of these failures occur at the time of vessel breach. This low value for early containment failure can in part be attributed to the high probability of early venting, 0.50. The dominant PDS in this group, which accounts for 50% of this groups mean fractional contribution to the core damage frequency, is a long-term loss of containment heat removal accident in which the operators vent the containment before core damage.

3.5.3 Status of Core-Concrete Interactions

Figure 3.5-7 shows the mean conditional probability of core-concrete interactions (CCI). The first two summary APBs include accidents in which CCI occurs following vessel breach. Accidents in which CCI occurs in a dry cavity are grouped together in the first bin. The second bin includes the accidents in which the CCI occurs under a pool of water that is maintained by injection from the vessel. The presence of water is important because a pool of water above the core debris will scrub the releases. The third bin corresponds to cases in which a coolable debris bed forms and CCI is avoided. For CCI to be avoided two conditions must exist: a coolable

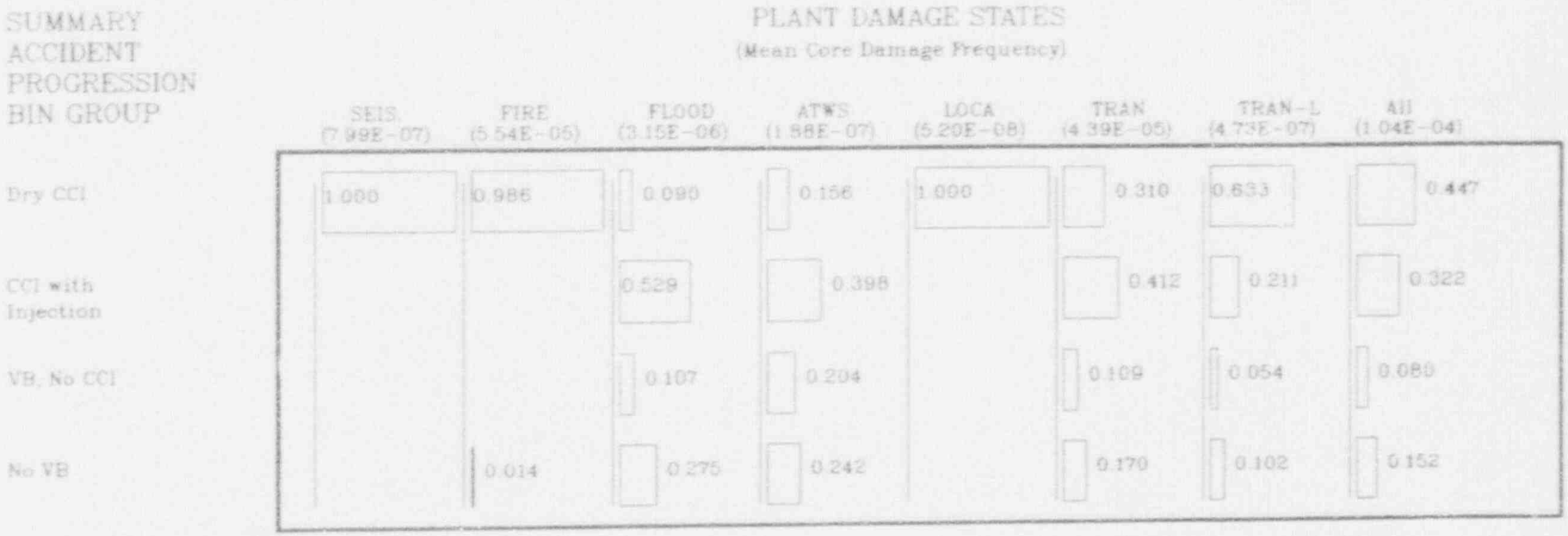

Figure 3.5-7. Mean Conditional Probability of Core-Concrete Interactions.

debris bed and a replenishable source of water (i.e., injection from the vessel). The last bin includes all of the accidents that do not proceed to vessel breach.

Because injection systems are not recoverable in either the Seismic or LOCA PDS groups, CCI always occurs in a dry cavity. Thus, the releases associated with CCI are not scrubbed by either the suppression pool or the containment sprays (sprays are not available in either of these PDSs). The Fire PDS group also has a very high probability of dry CCI because of the lack of recoverable injection systems. Although injection systems are frequently recoverable in the Flood, ATWS, and Transient PDS groups, CCI is still likely to occur for accidents that proceed to vessel breach. However, because injection is often recovered in these PDSs, it is likely that CCI will proceed with an overlying pool of water.

3.5.4 Summary Accident Progression Bins

Figure 3.5-8 shows the mean distribution among the summary accident progression bins for the summary PDS groups. Only mean values are shown, so Figure 3.5-8 gives no indication of the range of values encountered. These mean values are conditional on core damage. The distribution for core damage arrest, which consists of the last three bins in Figure 3.5-8, is shown in Figure 3.5-1. Similarly, the distribution for early containment failure, which consists of the first two bins in Figure 3.5-8, is shown in Figure 3.5-2. Nonetheless, Figure 3.5-8 gives a good idea of the relative likelihood of the results of the accident progression analysis. The summary bins were defined in section 3.4. These summary APBs will be presented again in Chapter 6 to show which types of accident are contributing the most to mean values of risk. Thus, this section provides the reader with information on the relative likelihood of the various bins while Chapter 6 will provide information on the relative importance of each bin with respect to the mean values of risk.

The summary APBs are composed of three characteristics: the occurrence of vessel breach, the containment failure time, and the RPV pressure at the time of vessel breach. The reactor pressure is only considered if the containment fails early. If the containment fails late in the accident the effect of vessel pressure on the accident progression is not as important.

The first four summary APBs group together accidents in which the core damage process is not arrested (i.e., vessel breach occurs) and the containment integrity is compromised by either containment failure or containment venting. The releases associated with these accidents tend to be large because there are both in-vessel and ex-vessel releases and a path has been established for the radionuclides to escape the containment. The first two bins involve accidents in which the containment fails early in the accident. For the first bin, the reactor pressure is low at the time of vessel breach, whereas for the second bin the reactor pressure is high. The mean probabilities of these two bins are 0.15 and 0.19, respectively. The third APB includes accidents that result in containment failure late in

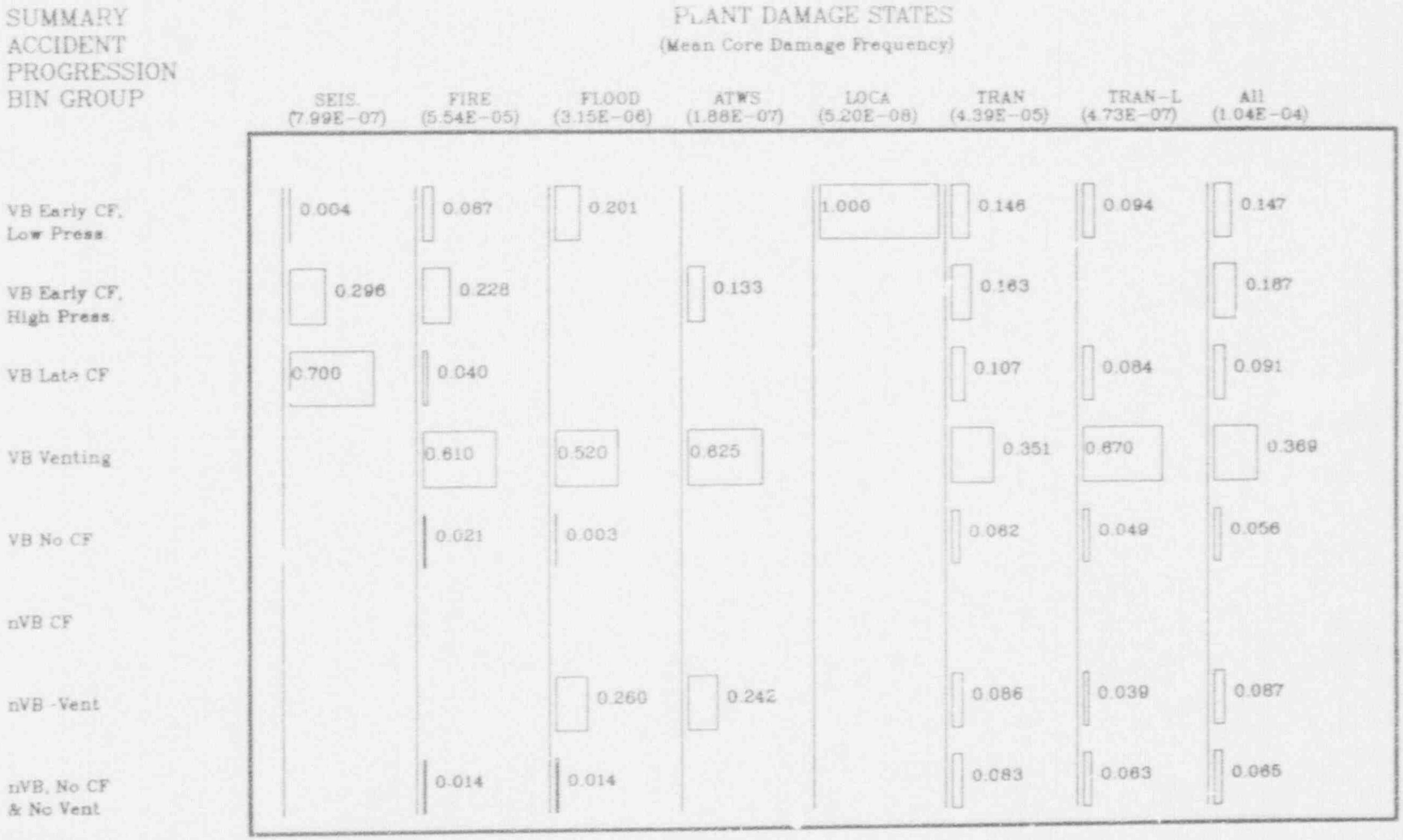

Figure 3.5-8. Mean Conditional Probability of the Summary APBs for the Summary PDSs.

the accident. This is the dominant APB for the seismic PDS group. The mean probability of this APB when all of the PDS groups are considered is 0.09. The fourth APB contains all of the accident progression paths that involve vessel failure and containment venting. Both early and late containment venting is included in this bin. The mean probability of this bin is 0.37 which is roughly equal to the sum of the early containment failure bins (i.e., the first two APBs). This is the dominant bin for all of the PDS groups except for the Seismic and the LOCA PDSs groups.

The fifth and sixth bins in Figure 3.5-8 represent accidents in which the containment remains intact; however, vessel breach occurs in the fifth bin and does not occur in the sixth bin. The mean probabilities of these two bins are 0.056 and 0.065, respectively. These bins mostly occur for the Transient and Transient-LOCA PDS groups. For these bins, the radionuclide releases to the environment are negligible. The sixth and seventh bins include accidents in which the core damage process is arrested, but the containment integrity is compromised by either containment failure or venting, respectively. For these accidents, there are only in-vessel releases and, in most cases, these releases will pass through the suppression pool where they will be scrubbed. Inspection of Figure 3.5-8 reveals that for all of the PDS groups the mean probability of core damage arrest coincident with containment failure (i.e., the sixth bin) is less than 0.001. The mean probability of the seventh bin is approximately 0.09; however, for the Flood and ATWS PDS groups this bin occurs approximately 25% of the time.

Based on the results presented in this section, the following observations can be made with regard to the accidents analyzed in this study. The majority of the accidents analyzed will proceed to vessel failure. Although notable, the mean conditional probability of core damage arrest is still fairly small, approximately 0.15. The probability of core damage arrest is driven by the recovery of AC power for the short-term station blackouts, the lack of available or recoverable injection systems for the other accidents, and the probability of occurrence of in-vessel stem explosions. Given that core damage occurs, it very likely that the containment's integrity will be compromised during the course of the accident by either containment failure or by containment venting. The mean probability that the containment will remain intact throughout the accident is only 0.12. Furthermore, it is fairly likely that the containment will fail early in the accident--the mean probability of early containment failure is 0.33. It is also likely that the operators will vent the containment during the accident--the mean probability of containment venting is 0.46. Given that core damage occurs, it is likely that the core debris released from the vessel will participate in core-concrete interactions. The mean probability of CCI, conditional on core damage, is 0.77. Thus, the potential exist for a large release late in the accident.

The events that result in containment failure before core damage are slow pressurization events that result from the accumulation of steam and noncondensibles during accidents in which containment heat removal is lost or inadequate (i.e., long-term loss of containment heat removal accidents, long-term station blackout accidents, and ATWS accidents). Events that result in containment failure around the time of vessel breach include fast pressurization of the containment from loads accompanying vessel breach (i.e., DCH, ex-vessel steam explosions, RPV blowdown), Alpha mode events, drywell failure induced by reactor pedestal failure, and cavity drain line isolation failure. Late in the accident, the events that result in containment failure include the slow pressurization of the containment from the steam and noncondensibles generated during CCI and failure of the reactor pedestal by caused concrete erosion during CCI.

3.6 References

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4.0 SOURCE TERM ANALYSIS

4.1 Introduction

This chapter describes the source term analysis. The model presented was developed to analyze the radioactive material transport within the primary system, the primary containment, and the reactor building. The model was used in the integrated risk uncertainty analysis to determine the releases arising from severe accidents at the LaSalle nuclear power plant. The analysis includes both internal and external events and resulted in tens of thousands of source terms. The resulting source terms were grouped for subsequent consequence analyses using a partitioning process described in this chapter.

The analysis tracks the transport of radioactive fission products from the fuel, through the reactor coolant system, through the primary and secondary containments to the environment. The removal and retention of fission products by natural processes, such as deposition on surfaces, and by engineered safety systems, such as sprays, are accounted for when applicable. A parametric computer model, LASSOR, was constructed to perform the source term calculations. It was executed once for each accident progression bin for each observation in the LHS sample, using parameter values drawn randomly from the distributions of all the source term variables. The results of evaluating this parametric model were the release fractions for each of nine radionuclide groups during three time regimes. The power of the releases, the timing of these releases, and the altitude of the releases are also determined, as well as the time at which the emergency response warning is given. The type of event was also distinguished as: seismic, fire, and all others. These source terms were then partitioned into groups that were expected to have similar consequences. The resulting source term groups were used in the consequence analysis.

This chapter includes a description of the method used for estimating the source terms, a description of the LASSOR code, the source term uncertainty issues, the source term results, and a description of the partitioning of the source terms.

4.2 Considerations for Estimating the Source Terms

4.2.1 Introduction

The source term for a given accident progression bin consists of the release fractions for nine radionuclide groups for the release before vessel breach, the release at vessel breach, and the release after vessel breach, as well as information about the timing of the releases, the power associated with the releases, the height of the releases, the event type, and the warning time. The radionuclide groups and the isotopes included in the groups are tabulated in Table 4.2-1.

Table 4.2-1.
Isotopes in Each Radionuclide Release Class.

Release Class	Isotopes Included
1. Inert Gases	Kr-85, Kr-85M, Kr-87, Kr-88, Xe-133, Xe-135
2. Iodine	I-131, I-132, I-133, I-134, I-135
3. Cesium	Rb-86, Cs-134, Cs-136, Cs-137
4. Tellurium	Sb-127, Sb-129, Te-127, Te-127M, Te-129, Te-129M, Te-131M, Te-132
5. Strontium	Sr-89, Sr-90, Sr-91, Sr-92
6. Ruthenium	Co-58, Co-60, Mo-99, Tc-99M, Ru-103, Ru-105, Ru-106, Rh-105
7. Lanthanum	Y-90, Y-91, Y-92, Y-93, Zr-95, Zr-97, Nb-95, La-140, La-141, La-142, Pr-143, Nd-147, Am-241, Cm-242, Cm-244
8. Cerium	Ce-141, Ce-143, Ce-144, Np-239, Pu-238, Pu-239, Pu-240, Pu-241
9. Barium	Ba-139, Ba-140

The source term analysis is performed with a relatively small computer code: LASSOR. The goal of this code is not to calculate the behavior of the fission products from their chemical and physical properties and the thermal hydraulic conditions in the reactor and the containment. Instead, the purpose is to parametrically represent the results of more detailed codes that do consider these quantities, as well as the opinions of the experts who supplied source term information. The method by which the source terms were estimated is discussed in this section.

4.2.2 Timing Considerations

Releases to the environment are divided into three time regimes: before vessel breach, at vessel breach, and after vessel breach. Releases before vessel breach include those releases associated with the in-vessel core degradation process. The vessel breach releases include the puff release that occurs when the vessel breach occurs, releases due to direct containment heating, and releases due to ex-vessel steam explosions. The late or after vessel breach releases include releases due to core-concrete interactions, revolatilization of radionuclides in the RCS following vessel breach, and volatilization of iodine from the cavity water and suppression pool.

Multiple containment failures were considered in this analysis resulting in the possibility of three different release paths for the three release times described above. For each release time, the current and any previous containment failures were considered in determining the pathway that would most accurately represent the characteristics of the release at that time. The release path was then changed so that the release would go through the appropriate pathway. Releases prior to the change in pathway were not affected. If the containment did not fail until the late time frame, then all releases went through that failure path and divided into two release segments with the first release segment represented as a puff at the time of containment failure whose relative magnitude depended on the failure mode (rupture vs. leak).

4.2.3 System and Phenomenological Considerations

The characteristics of the primary system and the primary and secondary (reactor building) containments are described in Sections 1.3 and 3.2. The primary barrier between the radionuclides released from the vessel and the outside environment is the containment structure. The containment structure has a design pressure of 45 psig and an assessed mean failure pressure of 191 psig. The causes of containment failure considered in the analysis include:

1. Overpressure (rapid and slow),

2. Containment venting through the wetwell when the containment pressure reaches 60 psig,

3. Drywell rupture due to reactor pedestal failure caused by core concrete interactions or by loads at vessel breach,

4. Drywell floor drain isolation failure, and

5. Alpha mode failures.

In the event of a containment failure, there are a number of plant characteristics that help reduce the amount of radionuclides released to the environment. The suppression pool can provide substantial mitigation of the source terms in accidents because of its ability to effectively trap radionuclides. The cavity may also fill with water trapping radionuclides during core concrete interactions. In addition to these two pools, the containment sprays may also reduce the source term.

There are two pathways by which radionuclides enter the suppression pool. The first pathway is through the SRV tailpipes from the vessel through the T-quenchers in the suppression pool. The second pathway is through the drywell and then into the suppression pool. This second pathway can occur in two ways. The first way is if an SRV tailpipe vacuum breaker sticks open during core damage. If the drywell has not failed and the drywell vacuum breakers have not failed open, then the release path is to the drywell then through the downcomers into the suppression pool. The second way is is via a LOCA where the release path is directly from the reactor vessel to the drywell and then through the downcomers into the suppression pool. The first pathway, through the tailpipes directly to the suppression pool, is more effective than the second pathway, through the drywell and then to the suppression pool, at trapping radionuclides. However, the second pathway still offers a significant mechanism for mitigating the source term.

The containment sprays, which are located in the drywell, can also be effective at reducing the amount of airborne radionuclides. The decontamination factor associated with the sprays is roughly equivalent to the decontamination factor for the suppression pool through the downcomers.

The final barrier between the radionuclides released from the fuel and the environment is the reactor building. The reactor building envelopes the containment; thus, all radionuclides released from the containment must pass through the reactor building. However, the decontamination factor varies for different containment failure locations. If the drywell head fails, then the radionuclides must only pass through the refueling floor before reaching the environment. This is in contrast to a failure anywhere else in the containment which results in the radionuclides passing through the reactor building proper with its relatively higher decontamination factor due to the more contorted pathway to the environment and more surface area for radionuclide deposition.

4.2.4 Environmental Release Considerations

In addition to the amounts of the radionuclide species released during an accident, the consequences analysis also needs input concerning the release power, time, duration, and height. Furthermore, the analysis requires an estimate of the time at which emergency warning is given by the plant operational personnel so that the time for a possible implementation of the emergency response measures can be judged. This estimation is also made in LASSOR.

4.3 scription of the LASSOR Code

4.3.1 Introduction

The LaSalle parametric source term code, LASSOR, is based on the XSOR codes developed for the NUREG-1150 analysis* and is most closely related to the source term code for Peach Bottom, PBSOR.[1] The following sections describe the major differences in the two codes and the parametric equation used to calculate radionuclide transport at LaSalle.

4.3.2 Release Phases

To improve the characterization of the source term, the release was divided into three segments; in contrast, two segments were used for the NUREG-1150 plants.[2] The three time segments were defined to be before vessel breach, at vessel breach, and after vessel breach. This characterization allows the slow releases before vessel breach to be differentiated from the short puff at vessel breach, resulting in a better representation of the source term during an evacuation. The three release phases are only necessary when the containment fails before vessel breach. If the containment does not fail before vessel breach, the second release is set to zero and the first release becomes a puff and occurs at containment failure (i.e., there are only two release segments; a puff and a tail).

4.3.3 Multiple Containment Failures

In the NUREG-1150 analysis only two release segments were used. For the BWR analyses (e.g., Peach Bottom[1] and Grand Gulf[3]), it was very difficult for the analysts to accurately represent the characteristics of the release and its impact on the evacuating population with only two release segments. This was particularly true in two cases. The first involves those accident progressions where the containment failed prior to vessel breach and a

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* H.-N. Jow, W. B. Murfin, and J. D. Johnson, "XSOR Codes User's Manual," NUREG/CR-5360, SAND89-0943, Sandia National Laboratories, Albuquerque, NM, to be published.

small initial release occurred during evacuation. Depending on the nature of the evacuation, this small release can be the dominant contributor to the early radiation effects on the evacuating population. However, the puff release at vessel breach and the long-term CCI release are the largest contributors to the long-term effects. If all three occur in an accident progression, then three release segments are needed to accurately model the effects. The second case involves accident progressions similar to the first case; but, where the containment failed a second time, usually at vessel breach, in a way that would lead to most of the radionuclides following a path with substantially different characteristics from the path resulting from the original failure. This implies that the initial release path could be substantially different from the release at vessel breach or the long-term release and again three segments would be needed to accurately model the effects. The MACCS code[4] that was used to perform the consequence calculation can handle a source term with three release segments without modification. They were just not used in the NUREG-1150 analysis. The capability of modeling multiple containment failures was incorporated into the LASSOR code to ensure that a more accurate representation of the characteristics of the source terms would be obtained and to take advantage of this capability in the MACCS code. In each time frame the radionuclides were released through the pathway that was judged to most accurately represent the source term. A detailed analysis was made of all the possible pathways for a release at a particular time. These were then analyzed using an event tree model to examine all the possible combinations of different failure modes, locations, and times. As a result of this analysis, the following hierarchy of failure locations and sizes was determined. It should be emphasized that these comparisons take place between failures occurring at different times.

1. Ruptures were always considered to be worse than leaks into the reactor building because the lowest resistance pathway would be the rupture.

2. A failure in the drywell head was always considered to be worse than a failure of the same size in the drywell because both the median and maximum reactor building decontamination factors were smaller for a drywell head failure than for any other type of drywell failure.

3. A failure in the wetwell above the normal water line was always considered to be worse than a failure of the same size in the wetwell below the water line. Even with the most severe failure below the water line, a hand calculation showed that the water level remains above the bottom of the SRV T-quenchers and the bottom of the downcomers. The result is that the radionuclides are forced to pass through the suppression pool as they leave the containment. On the other hand, radionuclides may or may not pass through the suppression pool to be released if the failure is in the wetwell above the water line.

4. If a drywell or drywell head failure and a wetwell failure above the water line of the same size occur, the choice of pathway depends on the status of the cavity floor and the sprays. If sprays are not available, the release is treated as if it all passes through the drywell or drywell head. If sprays are available and the cavity floor has not failed, the release is treated as if it is all through the pathway with the minimum decontamination factor as determined by the observation currently being processed. If the sprays are available and the cavity floor has failed, the release is treated as if it is all through the wetwell above the water line since the sprays do not affect the radionuclides in the wetwell.

5. If a drywell or drywell head failure and a wetwell failure below the water line of the same size occur, the choice of pathway depends on the status of the sprays. If the sprays are not operating, the release is treated as if it is all through the drywell or drywell head. If the sprays are operating, the release is treated as if it is all through the pathway with the minimum decontamination factor.

An example of a possible multiple failure is venting before vessel breach followed by a rupture in the drywell caused by the reactor vessel falling as a result of pedestal failure due to core-concrete interactions. In this case, releases before and at vessel breach would go through the venting pathway, and releases after vessel breach would go through either the venting pathway or the drywell, depending on which pathway had the lowest decontamination factor.

4.3.4 Parametric Equation

This section describes the parametric source term equation developed for LaSalle. Each type of release is described separately.

<u>In-Vessel Releases</u>

The source term for each of the nine radionuclide groups (denoted by i) due to releases before vessel breach is given by:

$$ST_i = FCOR_i * FVES_i * FCONV_i * (FPLBY/DF1 + (1.0 - FPLBY)/DF2) / RBDF_i .$$

Where:

$FCOR_i$ = fraction of initial inventory of species i released from the fuel prior to vessel failure,

$FVES_i$ = fraction of species i that was released from the fuel which is released from the vessel,

$FCONV_i$ = fraction of species i released from containment for releases before vessel breach, not including the effects of scrubbing by pools and sprays,

FPLBY = fraction of pool bypass before vessel breach as a result of either a LOCA or stuck-open vacuum breaker, and

$RBDF_i$ = decontamination factor for the reactor building for species i.

The decontamination factors (DF1 and DF2) applied depend on the location of the containment failure and the pathway from the reactor vessel. This release is through the first available containment failure location. If a drywell or drywell head failure occurs, then DF1 = $DFSPRV_i$ and DF2 = max($DFVPA_i$,$DFSPRV_i$) where $DFSPRV_i$ is the scrubbing decontamination factor for sprays acting on species i released into containment from the vessel before vessel breach and $DFVPA_i$ is the scrubbing decontamination factor for aerosol species i flowing directly from the vessel to the suppression pool and then out the containment failure. The value of $DFSPRV_i$ is 1.0 if the sprays are not operating. The maximum of $DFVPA_i$ and $DFSPRV_i$ is taken since both parameters could be very high in a given observation of the sample and the experts felt that using the maximum of the two values was appropriate. The second, lower, decontamination factor was not applied in addition to the first because the pools and sprays tend to remove the same size aerosol. If a wetwell failure occurs, DF2 = $DFVPA_i$ since sprays will not affect releases from the wetwell and DF1 = $DFSPRV_i$, if the drywell vacuum breakers have failed open thus bypassing the downcomers, or DF1 = max($DFSPRV_i$,$DFCPA_i$), if the drywell vacuum breakers have not failed. $DFCPA_i$ is the scrubbing decontamination factor for aerosol species i flowing from containment through the suppression pool. The reactor building DF is selected from a different distribution if the containment failure is to the refueling floor as opposed to anywhere else.

Vessel Breach Puff

The source term due to releases at vessel breach not including the effects of direct containment heating (DCH) or ex-vessel steam explosions (EVSE) is given by:

$$ST_i = FCONC_i * VBPUF_i / (RBDF_i * DF3).$$

Where:

$FCONC_i$ = fraction of species i released from containment for CCI and other releases at or after vessel breach, not including the effects of scrubbing by pools and sprays, and

$VBPUF_i$ = fraction of initial core inventory of species i that is released to the containment as a puff at the time of vessel breach (does not include releases from DCH or EVSE).

DF3 depends on the location and timing of containment failure as described below.

Wetwell Failure

If a wetwell failure occurs above the water line and the cavity floor fails at vessel breach, DF3 = 1.0 since the radionuclides will not have to pass through the suppression pool and sprays do not affect the wetwell. However, if the cavity floor does not fail at vessel breach, DF3 = $DFCPA_i$ since the radionuclides will have to pass through the pool to be released from the containment. If a wetwell failure occurs below the water line, DF3 = $DFCPA_i$ since a hand calculation shows that the suppression pool water level will still be above the outlet of the SRV T-quenchers and the downcomers and all radionuclides leaving the containment will pass through the reduced suppression pool.

Drywell Failure

If a drywell failure occurs either in the drywell itself or in the drywell head, the decontamination factor depends on the status of the sprays. If the sprays are operating, DF3 = $DFSPRC_i$; otherwise, DF3 = 1.0. $DFSPRC_i$ is the scrubbing decontamination factor for sprays acting on species i released into containment after vessel breach.

No Containment Failure

If no containment failure occurs in the accident progression bin, DF3 = $\min(DFSPRC_i, DFCPA_i)$ which is the minimum decontamination factor possible. The minimum value is chosen in order to maximize the amount of radionuclides released from the containment through leakage. This was done to simplify the calculations since these source terms are negligible compared to the other cases.

<u>Direct Containment Heating</u>

The source term due to direct containment heating is given by:

$$ST_i = (1.0 - FCOR_i - VBPUF_i) * FLV * FHPE * FDCH_i * FCONC_i / (RBDF_i * DF3).$$

Where:

FLV = fraction of the core material that leaves the vessel anytime at or after vessel breach (assumes that the ratio of species i leaving the vessel to the initial inventory of species i is the same for all species),

FHPE = fraction of core material leaving the vessel that participates in direct containment heating (assumes that the ratio of species i participating in DCH to the amount of species i leaving the vessel is the same for all species), and

FDCH$_i$ = fraction of species i in the debris that participates in DCH that is released due to DCH.

DF3 depends on the location and timing of containment failure as described above.

Ex-Vessel Steam Explosion

Similarly, the ex-vessel steam explosion source term is given by:

$$ST_i = (1.0 - FCOR_i - VBPUF_i) * FLV * EVSE * FEVSE_i * FCONC_i / (RBDF_i * DF3).$$

Where:

EVSE = fraction of core material leaving the core after vessel breach that is involved in an ex-vessel steam explosion (assumes that the ratio of species i involved in EVSE to the amount of species i leaving the core is the same for all species), and

FEVSE$_i$ = fraction of species i in the debris participating in EVSE that is released due to EVSE.

DF3 may be determined from the information above.

Molten Core Concrete Interactions

The core-concrete interaction (CCI) release is given by:

$$ST_i = (1.0 - FCOR_i - VBPUF_i) * FLV * XCCI * FCCI_i * FCONC_i / (RBDF_i * DF4).$$

Where:

XCCI = fraction of core material leaving the core after vessel breach that is available for CCI (assumes the ratio of species i available for CCI to the amount of species i leaving the core after vessel breach is the same for all species), and

FCCI$_i$ = fraction of species i available for CCI that is released during CCI.

The decontamination factor, DF4, is partially determined in the same manner as DF3, however, radionuclide scrubbing in the cavity water is possible in the latter stages of the accident. It should be noted that, while the method of determining DF3 is the same, the numerical value of DF3 may be different due to subsequent containment failure later in the accident. If a replenishable source of water exists, either sprays or low pressure injection, DF4 = max(DF3,$DFCAV_i$) where $DFCAV_i$ is the scrubbing decontamination factor for aerosol species i released into cavity water during CCI release.

Revolatilization After Vessel Breach

The equation for cesium, iodine, and tellurium release due to revolatilization after vessel breach is given by:

$$ST_i = (\delta_{12} + \delta_{13} + \delta_{14}) * FREVO_i * FCOR_i * (1.0 - FVES_i) * FCONC_i \\ / (RBDF_i * DF5).$$

Where:

$FREVO_i$ = fraction of species i that was retained in the vessel that is revolatilized following vessel breach, and

δ_{ij} = 1 if i = j and 0 otherwise. The 2, 3, and 4 subscripts denote iodine, cesium, and tellurium, respectively.

The decontamination factor, DF5, is determined in the same manner as the decontamination factor, DF3. However, this decontamination factor is not necessarily the same as DF3 since another containment failure may have occurred in a worse location later in the accident.

Late Volatilization of Iodine

The source term equation for late revolatilization of iodine from the suppression pool and the water in the reactor cavity is given by:

$$ST_i = \delta_{12} * (FLTI1 * POOLI + FLTI2 * CAVWI) * DF6.$$

Where:

FLTI1 = fraction of iodine in the suppression pool that is volatilized and released after vessel breach,

POOLI = fraction of initial core inventory of iodine scrubbed by the pool,

FLTI2 = fraction of iodine in the cavity water that is volatilized and released after vessel breach, and

CAVWI = fraction of initial iodine core inventory scrubbed by the cavity water during CCI release.

DF6 = $FCONC_i$ if there is no containment failure or DF6 = 1.0 if there is containment failure. This is based on the opinion of the experts that revolatilized iodine would behave as a noble gas. For late volatilization of iodine, the decontamination factor associated with the reactor building is not applied.

If the containment fails before vessel breach, then there are three release segments consisting of (1) an initial release - the in-vessel release, (2) a puff release - the vessel breach puff, DCH, and EVSE, and (3) a late release - MCCI, Revolatilization after VB, and late iodine. If the containment fails at vessel breach, then there are two releases consisting of (1) an initial puff release - in-vessel release, the vessel breach puff, DCH, and EVSE, and (2) a late release - MCCI, Revolatilization after VB, and late iodine. If the containment fails after vessel breach, then there are the same two releases but some of the late release is put into the initial puff depending on the accident progression timing.

4.4 LASSOR INPUT

4.4.1 Introduction

The LASSOR input consists of two types of variables; sampled and non-sampled. This section describes both types of variables and the sources of data used to quantify the variables.

4.4.2 Source Term Uncertainty Issues

4.4.2.1 Introduction

The variables discussed in this section were sampled with the Latin Hypercube sampling technique along with all other uncertain variables from the accident frequency analysis and the accident progression analysis. Many of the distributions used in this analysis are based on distributions generated by the Source Term Expert panel during the NUREG-1150 study.*

4.4.2.2 Issue Selection

The parameters chosen for the uncertainty analysis are the same as those analyzed in the Peach Bottom study with the addition of the warning time for several scenarios. The same issues were included in the LaSalle analysis because of the many similarities between the LaSalle and Peach Bottom plants. The parameters are described in Table 4.4-1 and discussed further in Section 4.4.2.3. The table contains three columns. The first column contains the mnemonic used to represent the variable. The second column contains the position of the variable in the Level II LHS sample and the third column contains a description of the variable and the source used to quantify it. Variables not chosen for the uncertainty analysis are described in Section 4.4.3.4.

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* F. T. Harper, et. al., "Evaluation of Severe Accident Risks: Quantification of Major Input Parameters, Expert Opinion Elicitation on Source Term Issues," NUREG/CR-4551, SAND86-1309, Vol. 1, Rev. 1, Part 4, Sandia National Laboratories, Albuquerque, NM, to be published.

Table 4.4-1
Variables Sampled in the Source Term Analysis

Variable	LHS	Description
TW	262	Time at which the general public is informed that they should evacuate. The distributions used for this parameter were developed by the project staff.
FCOR	263	Fraction of each fission product group released from the core to the vessel before vessel breach. There are two cases: high and low zirconium oxidation. This parameter was assessed by the Source Term Expert Panel for NUREG-1150.
FVES	264	Fraction of each fission product group released from the core which is released from the vessel. There are three cases: short-term station blackout with the RPV at system pressure, short-term station blackout with the RPV at low pressure, and ATWS with the RPV at system pressure. This parameter was assessed by the Source Term Expert Panel for NUREG-1150.
FCCI	265	Fraction of each fission product group in the core material at the start of core concrete interactions that is released to the containment. There are four cases: low zirconium oxidation in the core and no overlying water, low zirconium oxidation in the core with overlying water, high zirconium oxidation in the core and no overlaying water, and high zirconium oxidation in the core with overlying water. This parameter was assessed by the Source Term Expert Panel for NUREG-1150.
FDCH	266	Fraction of each fission product group in the core material that participates in a direct containment heating event (DCH) that is released to the containment. Given the occurrence of DCH, there is only one case. This parameter was assessed by the Source Term Expert Panel for NUREG-1150.
FEVSE	267	Fraction of each fission product group in the core material that participated in an ex-vessel steam explosion that is released to the containment. Given the occurrence of an ex-vessel steam

Table 4.4-1 (Continued)
Variables Sampled in the Source Term Analysis

Variable	LHS #	Description
		explosion, there is only one case. This parameter was not assessed by the Source Term Expert Panel for NUREG-1150. It is assumed that the release fractions for the ex-vessel steam explosion phenomena are sufficiently similar to the release fractions associated with DCH that the DCH distributions are also used to quantify this parameter.
FLTI1	268	Fraction of iodine in the suppression pool that is volatilized and released after vessel breach. There are two cases: the suppression pool is subcooled and the suppression pool is saturated. This parameter was assessed by the Source Term Expert Panel for NUREG-1150.
FREVO	269	Fraction of the deposited amount of each fission product group in the RPV which revolatilized after VB and was released to the containment. There are three cases: no water injection after vessel breach and a high drywell temperature, no water injection after vessel breach and low drywell temperature, and water injection to the vessel after vessel breach. This parameter was assessed by the Source Term Expert Panel for NUREG-1150.
RBDF	270	Decontamination factor for aerosol releases flowing from the reactor building to the environment. There are four cases: drywell rupture and suppression pool subcooled, drywell rupture and suppression pool saturated, drywell head seal leak and suppression pool subcooled, and drywell head seal leak and suppression pool saturated. This parameter was assessed by the Source Term Expert Panel for NUREG-1150.
FCONV	271	Fraction of each fission product group released from containment for material released into containment before vessel breach, not including the effects of scrubbing by pools and sprays. There are seven cases: early containment leakage and a subcooled suppression pool, early containment leakage and a saturated suppression pool, early containment

Table 4.4-1 (Continued)
Variables Sampled in the Source Term Analysis

Variable	LHS #	Description
		rupture and a subcooled suppression pool, early containment rupture and a saturated suppression pool, late containment leak, late containment rupture, and no containment failure. The parameter was assessed by the Source Term Expert Panel for NUREG-1150.
FCONC	272	Fraction of each fission product group released from containment for CCI and other releases after vessel breach, not including the effects of scrubbing by pools and sprays. There are seven cases: early containment leakage and a subcooled suppression pool, early containment leakage and a saturated suppression pool, early containment rupture and a subcooled suppression pool, early containment rupture and a saturated suppression pool, late containment leak, late containment rupture, and no containment failure. The parameter was assessed by the Source Term Expert Panel for NUREG-1150.
DFVPA	273	Decontamination factor for in-vessel releases that are released into the suppression pool through the SRV T-quenchers. This parameter was not assessed by the Source Term Expert Panel for NUREG-1150. The values from the draft Grand Gulf source term analysis[5] were used for the single case.
DFCPA	274	Decontamination factor for aerosol releases flowing from the drywell to the suppression pool through the downcomers. This parameter was not assessed by the Source Term Expert Panel for NUREG-1150. The values from the draft Grand Gulf source term analysis[5] were used for the single case.
DFCAV	275	Decontamination factor for aerosols released into the cavity water from the CCI release. This DF is applied when the core debris is not coolable and CCI proceeds under water. There is one case: the reactor cavity is flooded with a continuous supply of water. This parameter was not assessed by the Source Term Expert Panel for NUREG-1150. The distributions for this parameter were modified from the draft Grand Gulf source term analysis[5] (as discussed later in section 4.4.2.3).

Table 4.4-1 (Concluded)
Variables Sampled in the Source Term Analysis

Variable	LHS #	Description
DFSPRV	276	Decontamination factor for sprays acting on fission product groups released into the containment from the vessel. This parameter was not assessed by the Source Term Expert Panel for NUREG-1150. The distributions for this parameter were modified from the Surry source term analysis draft report[6] (as discussed later in section 4.4.2.3).
FLTI2	277	Fraction of iodine in the cavity water that is volatilized and released after vessel breach. There are three cases: the reactor cavity is dry, the suppression pool is subcooled, or the suppression pool is saturated. This parameter was assessed by the Source Term Expert Panel for NUREG-1150.
DFSPRC	278	Decontamination factor for sprays acting on fission product groups released into the containment after vessel breach. This parameter was not assessed by the Source Term Expert Panel for NUREG-1150. The distributions for this parameter were modified from the draft report of the Surry source term analysis[6] (as discussed later in this section).

4.4.2.3 Issue Discussion

TW - Warning Time

Elici ... ns of the warning time for several accident scenarios were cond ... n July 13 and 16, 1990. Subjective opinions were obtained from Thor ... Brown and Arthur C. Payne, Jr. of SNL. The elicitations were prima..ly based on LZP-1200-1, "Classification of GSEP conditions," where GSEP is an acronym for Generating Stations Emergency Plan, for the LaSalle plant.[7] Throughout this section the conditions satisfied in the GSEP are indicated in parentheses, i.e., (condition 1K). This procedure is used by the plant personnel to determine the severity of the plant condition and gives guidance as to when to call a general emergency.

The scenarios for which distributions were developed include:

1. ATWS with successful injection and successful venting,
2. ATWS with successful injection, venting failure, and containment failure due to overpressure,
3. short-term accidents with failure of all injection,
4. scenarios with high pressure injection but no containment heat removal resulting in containment failure before core damage,
5. a scenario with no containment heat removal, however, low pressure injection is operating, resulting in a high enough containment pressure to force reclosure of the SRV's, thus repressurizing the primary system which fails the low pressure injection system leading to core dam...,
6. a scenario with injection initially working followed by failure of injection, the RPV either remains depressurized or was not depressurized,
7. a scenario with injection initially working, system depressurization occurs, injection fails, and the RPV repressurizes following the loss of DC power.

The cumulative distribution function for the ATWS scenario with successful venting is shown in Table 4.4-2. As soon as the ATWS begins, the plant is in a site emergency (condition 3K). When the containment reaches 45 psig and the water level in the reactor pressure vessel is less than the -129" level, the conditions for another site emergency exist (condition 2K). RELAP[8] and LTAS[9] calculations performed for the Level I analysis and reported in Volume 4 of NUREG/CR-4832* have shown that during an ATWS the

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* A. C. Payne Jr., S. A. Eide, J. C. LaChance, and D. W. Whitehead, "Analysis of the LaSalle Unit 2 Nuclear Power Plant: Risk Methods Integration and Evaluation Program (RMIEP), Volume 4: Initiating Events and Accident Sequence Delineation," NUREG/CR-4832/4 of 10, SAND92-0537, Sandia National Laboratories, Albuquerque, NM, to be published.

water level will be below the -129" level unless the main feedwater and primary coolant systems are working. The existence of two site emergencies can result in upgrading the condition to a general emergency by the Station Director. The experts agreed that the earliest a general emergency would be called was when the the two site emergencies occurred, i.e., when the containment pressure reached 45 psig. The experts also agreed that it was most likely that the general emergency would be called by the time core damage occurred (conditions 1Y, 1Z, 1ZA, 2N, 2P, 2Q, 3N, 3O); however, since this accident occurs very quickly, a 5% probability tail was included for 0.5 hours beyond core damage to account for possible confusion due to high stress. The distribution was assumed to be uniform between the time the containment pressure reached 45 psig and the time of core damage and between the time of core damage and 0.5 hours after core damage.

The onset of core damage was taken to be 0.5 hours after venting occurred (i.e., after loss of injection from severe reactor building environments at or near time of venting). This is 0.98 hours after the beginning of the accident. Calculations performed with the MELCOR code,[10] and described in Volume 3 of this report, show that core damage begins 1.3 hours after the initiation of a short-term station blackout accident with no injection. Since the core will be at a higher power in the ATWS sequence than in the short-term station blackout until the water is boiled off, the time to core damage for an ATWS should be shorter than that for a short-term station blackout. Based on this observation, the start of core damage was estimated to be 0.5 hours after venting. The time when the containment pressure reached 45 psig was determined from the LTAS ATWS calculations mentioned above.

The cumulative distribution function for the ATWS scenario with containment failure due to overpressure is shown in Table 4.4-3. This scenario is very similar to the ATWS scenario with successful venting. The difference is that venting does not occur and the containment, therefore, fails later due to overpressure. The structure of the distribution is the same as the distribution in Table 4.4-2; however, the timing of the events is different. Based on the LTAS ATWS calculations mentioned above, the containment would fail at 4375 seconds. Core damage was again set to begin at 0.5 hours after containment failure. A 0.5 hour tail after core damage occurs was also included in the distribution to allow for confusion on the part of the plant personnel due to high stress.

The cumulative distribution function for the short-term sequences is shown in Table 4.4-4. The earliest the experts thought a general emergency would be called was when the water level in the reactor vessel reached the -129" level (0.5 hour). Prior to this time a site emergency may exist due to the loss of power (condition 3M). When the low water level is reached, the plant personnel may call a general emergency (condition 3N). The time at which low reactor water level is reached is based on the short-term station blackout calculations performed with the MELCOR code and described in

Table 4.4-2.
Cumulative Distribution Function for TW
for the ATWS Scenario With Successful Venting.

Percentile	Warning Time (sec)
0.0	1430.
0.95	3530.
1.00	5330.

Table 4.4-3.
Cumulative Distribution Function for TW
for the ATWS Scenario With Containment Failure
Due to Overpressure.

Percentile	Warning Time (sec)
0.00	1430.
0.95	6175.
1.00	7975.

Table 4.4-4.
Cumulative Distribution Function for TW
For the Short-term Sequences.

Percentile	Warning Time (sec)
0.00	1800.
0.25	4680.
0.75	4680.
1.00	7200.

Volume 3 of this report. The experts thought it was highly likely that a general emergency would be called at the onset of core damage (conditions 3O, 2O, 2P, 2Q, 1Y, 1Z, 1ZA), thus 50% of the probability was assigned to this time. To account for confusion and possibly degraded instrumentation, the warning time was allowed to be as large as two hours after the initiation of the accident.

Loss of containment integrity occurs before core damage in two scenarios of concern to the PRA. In the first, venting occurs before core damage and in the second, the containment fails due to overpressure before core damage. The logic used to determine the cumulative distribution functions for these two scenarios was the same; however, the timing was different. The cumulative distribution function for venting before core damage is shown in Table 4.4-5 and the cumulative distribution function for containment failure due to overpressure before core damage is shown in Table 4.4-6.

The experts believed the earliest a general emergency would be called would be at or near the containment failure time based on the procedures (conditions 2Q, 3N, 2N). Injection failure from severe environments in the reactor building could occur at or close to containment failure. Following containment failure, the experts thought a declaration of a general emergency would be equally likely at any time up to two hours before core damage at which time conditions 3O and 2P of the procedures for a general emergency would be satisfied. From two hours before core damage to the beginning of core damage, the experts thought a general emergency declaration would be equally likely. The time of 5 hours from containment failure to the start of core damage was based on the long-term station blackout calculation performed with the MELCOR code and described in Volume 3 of this report.

The cumulative distribution functions for the scenario in which the SRV's reclose causing the low pressure injection systems to fail and the scenario in which injection is initially available but later fails and the system pressure either remained high or remained low are shown in Tables 4.4-7 and 4.4-8 respectively. The cumulative distribution function for the scenario in which injection is initially available and the RPS is depressurized then injection fails and the RPS repressurizes is shown in Table 4.4-9. The logic involved in developing these distributions was the same; however, the timing was different for each scenario. The experts felt that the earliest a general emergency would be called would be at the loss of coolant injection based on the procedures (conditions 2Q and 3N). When the water level in the reactor vessel reaches the -129" level the conditions for two site emergencies (conditions 2K and 3J) will have been met and the experts assigned 25% of the probability that a general emergency would be called at this time. The latest time the experts felt a general emergency would be called was at the onset of core damage (condition 1Y, 1Z, 1ZA, 2P, 3O) and they felt a general emergency was equally likely to be called at any time between the -129" level and core damage. The timing for these sequences was based on the intermediate station blackout calculation with

Table 4.4-5.
Cumulative Distribution Function for TW
For Venting Before Core Damage.

Percentile	Warning Time (sec)
0.00	83520.
0.95	94320.
1.00	101520.

Table 4.4-6.
Cumulative Distribution Function for TW
For Containment Failure Due to Overpressure Before Core Damage.

Percentile	Warning Time (sec)
0.00	192600.
0.95	203400.
1.00	210600.

Table 4.4-7.
Cumulative Distribution Function for TW
For the SRV Failure Scenario.

Percentile	Warning Time (sec)
0.00	103680.
0.25	131760.
0.50	131760.
1.00	138960.

Table 4.4-8.
Cumulative Distribution Function for TW
For the Scenario With Injection Initially
But Later Loss of Injection and System Pressure
Remaining Either High or Low.

Percentile	Warning Time (sec)
0.00	28800.
0.25	39960.
0.50	39960.
1.00	47160.

Table 4.4-9.
Cumulative Distribution Function for TW
For the Scenario With Injection Initially, RPV Depressurization,
Loss of Injection, Then System Repressurization.

Percentile	Warning Time (sec)
0.00	28800.
0.25	56880.
0.50	56880.
1.00	64080.

repressurization and the long-term station blackout calculation performed with MELCOR and described in Volume 3 of this report. In all of these scenarios the water level was taken to reach the -129" level approximately two hours before the start of core damage.

External events were also considered; however, all of these scenarios were covered by the distributions already developed. The short-term fire scenarios were taken to be similar to the short-term internal events scenarios, as were the flood scenarios. The long-term and venting fire scenarios were also taken to be similar to the long-term and venting internal events. Earthquake scenarios were taken to be similar to either the scenario with injection initially followed by injection failure and remaining at high or low pressure or the scenario with injection initially followed by injection failure and system repressurization as appropriate.

FCOR

The probability distributions for this variable were developed by the Source Term Expert Panel for NUREG-1150 for BWRs and are discussed in detail in that report.* These distributions were used in the LaSalle analysis. Table 4.4-1. contains the distribution for the high zirconium oxidation case while Table 4.4-11 contains the distribution for the low zirconium oxidation case. The probability distributions for all of the radionuclides were fully correlated for this parameter. That is, the value of one LHS variable (FCOR) was used to select the same quantile of the distributions simultaneously.

FVES

The probability distributions for this variable were developed by the Source Term Expert Panel for NUREG-1150 for BWRs and are discussed in detail in that report.* Three cases were considered by the experts: short-term station blackout with the RPV at system pressure, short-term station blackout with the RPV at low pressure, and ATWS with the RPV at system pressure. The high RPV pressure station blackout case was only used to quantify the source terms for accident progression bins that involved high pressure station blackouts. The third case was used to quantify FVES for non-station blackout accident progression bins that included high pressure in the RPV. If the RPV was at low pressure, the station blackout case with low pressure was used to quantify FVES. Tables 4.4-12, 4.4-13, and 4.4-14 contain the cumulative distributions for each of these cases. The probability distributions for all of the radionuclides were fully correlated for this parameter.

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* F. T. Harper, et. al., "Evaluation of Severe Accident Risks: Quantification of Major Input Parameters, Expert Opinion Elicitation on Source Term Issues," NUREG/CR-4551, SAND86-1309, Vol. 2, Rev. 1, Part 4, Sandia National Laboratories, Albuquerque, NM, to be published.

Table 4.4-10.
Cumulative Probability Distributions for FCOR
For All Radionuclide Groups for the High Zirconium Oxidation Case.

Percentile	NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
0.00	.05	.03	.02	0.0	0.0	0.0	0.0	0.0	0.0
0.01	.073	.049	.033	3.0E-3	3.0E-5	0.0	0.0	0.0	2.2E-4
0.05	.17	.13	.07	.018	2.5E-4	0.0	0.0	0.0	1.2E-3
0.25	.56	.34	.26	.071	2.1E-3	5.0E-5	2.0E-5	2.0E-5	4.2E-3
0.50	.90	.74	.59	.15	6.4E-3	4.6E-3	1.0E-4	1.5E-4	8.6E-3
0.75	1.	.96	.89	.59	.018	.02	1.2E-3	3.0E-3	.03
0.95	1.	1.	1.	.91	.52	.081	.032	.085	.52
0.99	1.	1.	1.	.99	1.	.14	.1	.51	1.
1.00	1.	1.	1.	1.	1.	.27	.11	1.	1.

Table 4.4-11.
Cumulative Probability Distributions for FCOR
For all Radionuclide Groups for the Low Zirconium Oxidation Case.

Percentile	NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
0.00	.02	6.0E-3	5.0E-3	0.0	0.0	0.0	0.0	0.0	0.0
0.01	.033	6.6E-3	5.8E-3	2.9E-3	3.0E-5	0.0	0.0	0.0	1.1E-4
0.05	.084	9.2E-3	9.0E-3	7.3E-3	1.5E-4	0.0	0.0	0.0	2.2E-4
0.25	.41	.16	.088	.049	7.6E-4	5.0E-5	2.0E-5	2.0E-5	1.7E-3
0.50	.90	.69	.59	.14	4.0E-3	2.0E-3	1.0E-4	1.5E-4	6.5E-3
0.75	1.	.91	.83	.46	.013	.012	9.5E-4	2.5E-3	.027
0.95	1.	1.	1.	.89	.52	.058	.021	.085	.52
0.99	1.	1.	1.	.98	1.	.14	.10	.51	1.
1.00	1.	1.	1.	1.	1.	.27	.11	1.	1.

Table 4.4-12.
Cumulative Probability Distributions for FVES
For All Radionuclide Groups for Short-term Station Blackout
With the RPV at System Pressure.

Percentile	NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
0.00	1.	0.	0	0.0	0.0	0.0	0.0	0.0	0.0
0.01	1.	2.0E-5	2.5E-5	1.0E-5	1.0E-5	1.0E-5	1.0E-5	1.0E-5	1.0E-5
0.05	1.	8.0E-5	8.0E-5	5.0E-5	5.0E-5	5.0E-5	5.0E-5	5.0E-5	5.0E-5
0.25	1.	9.6E-3	5.1E-3	1.9E-3	1.9E-3	1.9E-3	1.9E-3	1.9E-3	1.9E-3
0.50	1.	.086	.033	.033	.033	.033	.033	.033	.033
0.75	1.	.33	.32	.31	.25	.25	.25	.25	.25
0.95	1.	.79	.79	.78	.77	.77	.77	.77	.77
0.99	1.	.96	.96	.96	.95	.95	.95	.95	.95
1.00	1.	1.	1.	1.	1.	1.	1.	1.	1.

Table 4.4-13.
Cumulative Probability Distributions for FVES
For All Radionuclide Groups for Short-term Station Blackout
With the RPV at Low Pressure.

Percentile	NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
0.00	1.	0.	0	0.0	0.0	0.0	0.0	0.0	0.0
0.01	1.	5.9E-3	3.3E-3	3.3E-3	3.3E-3	3.3E-3	3.3E-3	3.3E-3	3.3E-3
0.05	1.	.041	.023	.023	.023	.023	.023	.023	.023
0.25	1.	.23	.14	.14	.13	.13	.13	.13	.13
0.50	1.	.41	.30	.27	.26	.26	.26	.26	.26
0.75	1.	.63	.60	.59	.58	.58	.58	.58	.58
0.95	1.	.99	.99	.99	.99	.99	.99	.99	.99
0.99	1.	1.	1.	1.	1.	1.	1.	1.	1.
1.00	1.	1.	1.	1.	1.	1.	1.	1.	1.

Table 4.4-14.
Cumulative Probability Distributions for FVES
For All Radionuclide Groups for ATWS With the RPV
At System Pressure.

Percentile	NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
0.00	1.	0.	1.0E-5	0.0	0.0	0.0	0.0	0.0	0.0
0.01	1.	8.0E-5	8.0E-5	2.0E-5	2.0E-5	2.0E-5	2.0E-5	2.0E-5	2.0E-5
0.05	1.	.018	7.6E-3	1.0E-4	1.0E-4	1.0E-4	1.0E-4	1.0E-4	1.0E-4
0.25	1.	.089	.052	4.9E-3	4.8E-3	4.8E-3	4.8E-3	4.8E-3	4.8E-3
0.50	1.	.28	.25	.10	.078	.078	.078	.078	.078
0.75	1.	.75	.63	.39	.29	.29	.29	.29	.29
0.95	1.	.95	.90	.70	.70	.70	.70	.70	.70
0.99	1.	.99	.99	.88	.88	.88	.88	.88	.88
1.00	1.	1.	1.	.98	.98	.98	.98	.98	.98

FCCI

The probability distributions for this variable were developed by the Source Term Expert Panel for NUREG-1150 for BWRs. Four cases were considered: low zirconium content in the melt and a dry cavity, low zirconium content in the melt and water over the debris, high zirconium content and a dry cavity, and high zirconium content and water over the debris. The distributions for Grand Gulf were used since the LaSalle geometry more closely resembles Grand Gulf than Peach Bottom in this region. The distributions for the four cases are shown in Tables 4.4-15, 4.4-16, 4.4-17, and 4.4-18. The probability distributions for all of the radionuclides were fully correlated for this parameter.

FDCH

The probability distributions for this variable were developed by the Source Term Expert Panel for NUREG-1150 for BWRs and are discussed in detail that report. Only one case was considered and is contained in Table 4.4-19. The probability distributions for all of the radionuclides were fully correlated for this parameter.

FFVSE

A probability distribution for this variable was not developed by the experts; however, since the phenomenon occurring during a steam explosion were thought to be similar to those occurring during a direct containment heating event, the FDCH distribution given in Table 4.4-19 was used for this parameter.

FLTI1

The probability distributions for this parameter were elicited from the Source Term Expert Panel for NUREG-1150 for BWRs and are discussed in detail in that report. Two cases were considered: the suppression pool is subcooled and the suppression pool is saturated. Tables 4.4-20 and 4.4-21 contain these two distributions.

FREVO

The probability distributions for this parameter were elicited from the Source Term Expert Panel for NUREG-1150 and are discussed in detail in that report. There are three cases: no water injection after vessel breach and a high drywell temperature, no water injection after vessel breach and low drywell temperature, and water injection to the vessel after vessel breach. The distributions for the three cases are shown in Tables 4.4-22, 4.4-23, and 4.4-24. The probability distributions for all of the radionuclides were fully correlated for this parameter.

Table 4.4-15.
Cumulative Probability Distributions for FCCI
For All Radionuclide Groups for Low Zirconium Content in the Melt
And a Dry Cavity.

Percentile	NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
0.00	1.	1.	1.	4.4E-3	0.0	1.0E-9	0.0	0.0	3.0E-5
0.01	1.	1.	1.	.012	5.0E-5	1.0E-9	0.0	0.0	1.2E-4
0.05	1.	1.	1.	.069	3.1E-4	1.2E-9	1.0E-5	3.0E-5	4.9E-4
0.25	1.	1.	1.	.32	2.6E-3	2.4E-9	2.1E-4	3.2E-4	3.2E-3
0.50	1.	1.	1.	.66	.052	5.6E-9	2.2E-3	2.9E-3	.061
0.75	1.	1.	1.	.76	.62	5.0E-6	.013	.026	.45
0.95	1.	1.	1.	.94	.95	7.3E-3	.086	.18	.88
0.99	1.	1.	1.	.99	.99	9.7E-2	.10	.20	.98
1.00	1.	1.	1.	1.	1.	.25	.10	.20	1

Table 4.4-16.
Cumulative Probability Distributions for FCCI
For All Radionuclide Groups for Low Zirconium Content in the Melt
And Water Over the Debris.

Percentile	NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
0.00	1.	1.	1.	1.2E-3	0.0	1.0E-9	0.0	0.0	1.0E-5
0.01	1.	1.	1.	4.8E-3	2.0E-5	1.0E-9	0.0	0.0	8.0E-5
0.05	1.	1.	1.	.032	2.7E-4	1.1E-9	0.0	1.0E-5	3.6E-4
0.25	1.	1.	1.	.26	2.0E-3	1.3E-9	1.9E-4	2.6E-4	2.3E-3
0.50	1.	1.	1.	.64	.036	1.7E-9	2.1E-3	2.5E-3	.032
0.75	1.	1.	1.	.74	.59	1.0E-6	.012	.02	.41
0.95	1.	1.	1.	.93	.94	2.5E-3	.084	.17	.87
0.99	1.	1.	1.	.99	.99	5.8E-2	.099	.20	.98
1.00	1.	1.	1.	1.	1.	.15	.10	.20	1.

Table 4.4-17.
Cumulative Probability Distributions for FCCI
For All Radionuclide Groups for High Zirconium Content in the Melt
And a Dry Cavity.

Percentile	NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
0.00	1.	1.	1.	4.4E-3	0.0	1.0E-9	0.0	0.0	3.0E-5
0.01	1.	1.	1.	.012	5.0E-5	1.0E-9	0.0	0.0	1.2E-4
0.05	1.	1.	1.	.069	3.1E-4	1.2E-9	1.0E-5	3.0E-5	4.9E-4
0.25	1.	1.	1.	.40	2.6E-3	2.4E-9	2.1E-4	3.2E-4	3.2E-3
0.50	1.	1.	1.	.67	.052	5.6E-9	2.2E-3	2.9E-3	.061
0.75	1.	1.	1.	.79	.65	5.0E-6	.02	.031	.51
0.95	1.		1.	.96	.97	7.3E-3	.11	.18	.90
0.99	[illegible]	1.	1.	.99	1.	9.7E-2	.15	.20	.98
1.00	1.	1.	1.	1.	1.	.25	.16	.20	1.

Table 4.4-18.
Cumulative Probability Distributions for FCCI
For All Radionuclide Groups for High Zirconium Content in the Melt
And Water Over the Debris.

Percentile	NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
0.00	1.	1.	1.	1.2E-3	0.0	1.0E-9	0.0	0.0	1.0E-5
0.01	1.	1.	1.	4.8E-3	2.0E-5	1.0E-9	0.0	0.0	8.0E-5
0.05	1.	1.	1.	.032	2.7E-4	1.1E-9	0.0	1.0E-5	3.6E-4
0.25	1.	1.	1.	.26	2.0E-3	1.3E-9	1.9E-4	2.6E-4	2.3E-3
0.50	1.	1.	1.	.64	.036	1.7E-9	2.1E-3	2.5E-3	.032
0.75	1.	1.	1.	.74	.59	1.0E-6	.012	.02	.41
0.95	1.	1.	1.	.93	.94	2.5E-3	.084	.17	.87
0.99	1.	1.	1.	.99	.99	5.8E-2	.099	.20	.98
1.00	1.	1.	1.	1.	1.	.15	.10	.20	1.

Table 4.4-19.
Cumulative probability distributions
for all radionuclide grou, for direct containment heating.

Percentile	NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
0.00	1.	.063	.063	0.0	0.0	0.0	0.0	0.0	0.0
0.01	1.	.15	.15	0.0	0.0	0.0	0.0	0.0	0.0
0.05	1.	.50	.50	.001	.001	.001	.001	.001	.001
0.25	1.	1.	1.	.008	.002	.007	.002	.002	.004
0.50	1.	1.	1.	.043	.012	.020	.011	.011	.012
0.75	1.	1.	1.	.600	.030	.063	.040	.040	.067
0.95	1.	1.	1.	.975	.751	.700	.087	.087	.863
0.99	1.	1.	1.	1.	.980	.900	.200	.280	.980
1.00	1.	1.	1.	1.	1.	.950	.230	.330	1.

Table 4.4-20.
Cumulative Probability Distribution for FLTI1
For the Subcooled Suppression Pool.

Percentile	Fraction released
0.0	0.0
0.01	0.0
0.05	0.0
0.25	5.00E-4
0.50	1.55E-3
0.75	.0278
0.95	.085
0.99	.097
1.00	.100

Table 4.4-21.
Cumulative Probability Distribution for FLTI1
For the Saturated Suppression Pool.

Percentile	Fraction released
0.0	0.0
0.01	1.00E-6
0.05	4.06E-5
0.25	9.36E-4
0.50	4.63E-3
0.75	.173
0.95	.759
0.99	.950
1.00	1.00

Table 4.4-22.
Cumulative Probability Distributions for FREVO
For All Radionuclide Groups for the No Water Injection
After Vessel Breach and a High Drywell Temperature Case.

Percentile	NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
0.00	1.	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
0.01	1.	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
0.05	1.	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
0.25	1.	.03	.001	0.0	0.0	0.0	0.0	0.0	0.0
0.50	1.	.115	.051	0.0	0.0	0.0	0.0	0.0	0.0
0.75	1.	.306	.132	.024	0.0	0.0	0.0	0.0	0.0
0.95	1.	.557	.284	.224	0.0	0.0	0.0	0.0	0.0
0.99	1.	.800	.535	.413	0.0	0.0	0.0	0.0	0.0
1.00	1.	1.	.750	.800	0.0	0.0	0.0	0.0	0.0

Table 4.4-23.
Cumulative Probability Distributions for FREVO
For All Radionuclide Groups for the No Water Injection
After Vessel Breach and a Low Drywell Temperature Case.

Percentile	NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
0.00	1.	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
0.01	1.	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
0.05	1.	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
0.25	1.	.03	.001	0.0	0.0	0.0	0.0	0.0	0.0
0.50	1.	.114	.050	0.0	0.0	0.0	0.0	0.0	0.0
0.75	1.	.261	.122	.024	0.0	0.0	0.0	0.0	0.0
0.95	1.	.486	.236	.209	0.0	0.0	0.0	0.0	0.0
0.99	1.	.800	.438	.413	0.0	0.0	0.0	0.0	0.0
1.00	1.	1.	.750	.800	0.0	0.0	0.0	0.0	0.0

Table 4.4-24.
Cumulative Probability Distributions for FREVO
For All Radionuclide Groups for the Water Injection
After Vessel Breach Case.

Percentile	NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
0.00	1.	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
0.01	1.	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
0.05	1.	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
0.25	1.	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
0.50	1.	.03	.001	0.0	0.0	0.0	0.0	0.0	0.0
0.75	1.	.117	.061	.024	0.0	0.0	0.0	0.0	0.0
0.95	1.	.439	.200	.209	0.0	0.0	0.0	0.0	0.0
0.99	1.	.800	.287	.413	0.0	0.0	0.0	0.0	0.0
1.00	1.	1.	.750	.800	0.0	0.0	0.0	0.0	0.0

RBDF

The probability distributions for this parameter were elicited for Peach Bottom from the Source Term Expert Panel for NUREG-1150. Based on the similarity of the LaSalle and Peach Bottom reactor building and on the characteristics used by the experts in their discussions to derive these distributions, it was decided that the Peach Bottom distributions were valid for LaSalle. Four cases were considered: drywell rupture and suppression pool subcooled, drywell rupture and suppression pool saturated, drywell head seal leak and suppression pool subcooled, and drywell head seal leak and suppression pool saturated. In the LASSOR code, the reactor building DF used for all drywell head failures corresponds to the drywell head seal leak cases since the releases would pass only through the refueling bay. All other containment failures (i.e., drywell and wetwell failures) resulted in the use of the reactor building DF's for the drywell rupture cases since the releases would have to pass through comparable portions of the reactor building to be released to the environment. The distributions for these cases are shown in Tables 4.4-25, 4.4-26, 4.4-27, and 4.4-28. The probability distributions for all of the radionuclides were fully correlated for this parameter.

FCONV

The probability distributions for this parameter were developed by the Source Term Expert Panel for NUREG-1150 for the LaSalle plant and are discussed in detail in Appendix C of this report. Seven cases were addressed: early containment leak with a subcooled suppression pool, early containment leak with a saturated suppression pool, early rupture with a subcooled pool, early rupture with a saturated pool, late containment leak, late containment rupture, and no containment failure. The values for the late leak and late rupture cases were assumed to be the same as the values for these cases for FCONC since only one expert addressed these cases. In the context of this parameter, early is before or at vessel breach and late is after vessel breach. The distributions are shown in Tables 4.4-29, 4.4-30, 4.4-31, 4.4-32, 4.4-33, 4.4-34, 4.4-35. The probability distributions for all of the radionuclides were fully correlated for this parameter.

FCONC

The probability distributions for this parameter were developed by the Source Term Expert Panel for NUREG-1150 for the LaSalle plant and are discussed in detail in Appendix C of this report. Seven cases were addressed: early containment leak with a subcooled suppression pool, early containment leak with a saturated suppression pool, early rupture with a subcooled pool, early rupture with a saturated pool, late containment leak, late containment rupture, and no containment failure. The values for the late leak and late rupture cases were assumed to be the same as the values for these cases for FCONV since only one expert addressed these cases. In the context of this parameter, early is before or at vessel breach and late is after vessel breach. The distributions are shown in Tables 4.4-36, 4.4-37, 4.4-38, 4.4-39, 4.4-40, 4.4-41, and 4.4-42. The probability distributions for all of the radionuclides were fully correlated for this parameter.

Table 4.4-25.
Cumulative Probability Distributions for RBDF
For All Radionuclide Groups for the Drywell Rupture
With a Subcooled Suppression Pool Case.

Percentile	NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
0.00	1.	1.	1.	1.	1.	1.	1.	1.	1.
0.01	1.	1.04	1.04	1.06	1.06	1.06	1.06	1.06	1.06
0.05	1.	1.1	1.1	1.11	1.11	1.11	1.11	1.11	1.11
0.25	1.	1.41	1.41	1.48	1.48	1.48	1.48	1.48	1.48
0.50	1.	2.3	2.3	2.62	2.62	2.62	2.62	2.62	2.62
0.75	1.	4.28	4.28	4.91	4.91	4.91	4.91	4.91	4.91
0.95	1.	9.88	9.88	10.2	10.2	10.2	10.2	10.2	10.2
0.99	1.	56.	56.	64.	64.	64.	64.	64.	64.
1.00	1.	470.	470.	507.	507.	507.	507.	507.	507.

Table 4.4-26.
Cumulative Probability Distributions for RBDF
For All Radionuclide Groups for the Drywell Rupture
With a Saturated Suppression Pool Case.

Percentile	NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
0.00	1.	1.	1.	1.	1.	1.	1.	1.	1.
0.01	1.	1.04	1.04	1.06	1.06	1.06	1.06	1.06	1.06
0.05	1.	1.14	1.14	1.13	1.13	1.13	1.13	1.13	1.13
0.25	1.	1.67	1.67	1.56	1.56	1.56	1.56	1.56	1.56
0.50	1.	2.84	2.84	2.61	2.61	2.61	2.61	2.61	2.61
0.75	1.	4.94	4.94	5.48	5.48	5.48	5.48	5.48	5.48
0.95	1.	10.9	10.9	12.8	12.8	12.8	12.8	12.8	12.8
0.99	1.	77.	77.	82.	82.	82.	82.	82.	82.
1.00	1.	428.	428.	461.	461.	461.	461.	461.	461.

Table 4.4-27.
Cumulative probability Distributions for RBDF
For All Radionuclide Groups for the Drywell Head Seal Leak
With a Subcooled Suppression Pool Case.

Percentile	NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
0.00	1	1.	1.	1.	1.	1.	1.	1.	1.
0.01	1.	1.	1.	1.	1.	1.	1.	1.	1.
0.05	1.	1.02	1.02	1.02	1.02	1.02	1.02	1.02	1.02
0.25	1.	1.10	1.10	1.10	1.10	1.10	1.10	1.10	1.10
0.50	1.	1.35	1.35	1.40	1.40	1.40	1.40	1.40	1.40
0.75	1.	1.73	1.73	1.88	1.88	1.88	1.88	1.88	1.88
0.95	1.	6.65	6.65	6.65	6.65	6.65	6.65	6.65	6.65
0.99	1.	9.63	9.63	9.63	9.63	9.63	9.63	9.63	9.63
1.00	1.	36.	36.	36.	36.	36.	36.	36.	36.

Table 4.4-28.
Cumulative Probability Distributions for RBDF
For All Radionuclide Groups for the Drywell Head Seal Leak
With a Saturated Suppression Pool Case.

Percentile	NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
0.00	1.	1.	1.	1.	1.	1.	1.	1.	1.
0.01	1.	1.	1.	1.	1.	1.	1.	1.	1.
0.05	1.	1.02	1.02	1.02	1.02	1.02	1.02	1.02	1.02
0.25	1.	1.10	1.10	1.10	1.10	1.10	1.10	1.10	1.10
0.50	1.	1.46	1.46	1.48	1.48	1.48	1.48	1.48	1.48
0.75	1.	2.17	2.17	2.57	2.57	2.57	2.57	2.57	2.57
0.95	1.	6.65	6.65	6.65	6.65	6.65	6.65	6.65	6.65
0.99	1.	9.63	9.63	9.63	9.63	9.63	9.63	9.63	9.63
1.00	1.	36.	36.	36.	36.	36.	36.	36.	36.

Table 4.4-29.
Cumulative Probability Distributions for FCONV
For All Radionuclide Groups for the Early Leak
With a Subcooled Suppression Pool Case.

Percentile	NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
0.00	1.	.001	.001	.001	.001	.001	.001	.001	.001
0.01	1.	.005	.005	.005	.005	.005	.005	.005	.005
0.05	1.	.017	.017	.017	.017	.017	.017	.017	.017
0.25	1.	.119	.119	.119	.119	.119	.119	.119	.119
0.50	1.	.248	.248	.248	.248	.248	.248	.248	.248
0.75	1.	.464	.464	.464	.464	.464	.464	.464	.464
0.95	1.	.688	.688	.688	.688	.688	.688	.688	.688
0.99	1.	.792	.792	.792	.792	.792	.792	.792	.792
1.00	1.	.991	.991	.991	.991	.991	.991	.991	.991

Table 4.4-30.
Cumulative Probability Distributions for FCONV
For All Radionuclide Groups for the Early Leak
With a Saturated Suppression Pool Case.

Percentile	NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
0.00	1.	.004	.004	.004	.004	.004	.004	.004	.004
0.01	1.	.012	.012	.012	.012	.012	.012	.012	.012
0.05	1.	.040	.040	.040	.040	.040	.040	.040	.040
0.25	1.	.160	.160	160	.160	.160	.160	.160	.160
0.50	1.	.284	.284	.284	.284	.284	.284	.284	.284
0.75	1.	.492	.492	.492	.492	.492	.492	.492	.492
0.95	1.	.717	.717	.717	.717	.717	.717	.717	717
0.99	1.	.837	.837	.837	.837	.837	.837	.837	.837
1.00	1.	.991	.991	.991	.991	.991	.991	.991	.991

Table 4.4-31.
Cumulative Probability Distributions for FCONV
For All Radionuclide Groups for the Early Rupture
With a Subcooled Suppression Pool Case.

Percentile	NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
0.00	1.	.010	.010	.010	.010	.010	.010	.010	.010
0.01	1.	.100	.100	.100	.100	.100	.100	.100	.100
0.05	1.	.222	.222	.222	.222	.222	.222	.222	.222
0.25	1.	.4995	.4995	.4995	.4995	.4995	.4995	.4995	.4995
0.50	1.	.721	.721	.715	.715	.715	.715	.715	.715
0.75	1.	.8635	.8635	.842	.842	.842	.842	.842	.842
0.95	1.	.959	.959	.959	.959	.959	.959	.959	.959
0.99	1.	.988	.988	.988	.988	.988	.988	.988	.988
1.00	1.	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00

Table 4.4-32.
Cumulative Probability Distributions for FCONV
For All Radionuclide Groups for the Early Rupture
With a Saturated Suppression Pool Case.

Percentile	NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
0.00	1.	.026	.026	.026	.026	.026	.026	.026	.026
0.01	1.	.160	.160	.160	.160	.160	.160	.160	.160
0.05	1.	.279	.279	.279	.279	.279	.279	.279	.279
0.25	1.	.5375	.5375	.5375	.5375	.5375	.5375	.5375	.5375
0.50	1.	.753	.753	.737	.737	.737	.737	.737	.737
0.75	1.	.8965	.8965	.881	.881	.881	.881	.881	.881
0.95	1.	.979	.979	.979	.979	.979	.979	.979	.979
0.99	1.	.995	.995	.995	.995	.995	.995	.995	.995
1.00	1.	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00

Table 4.4-33.
Cumulative Probability Distributions for FCONV
For All Radionuclide Groups for the Late Leak Case.

Percentile	NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
0.00	1.	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
0.01	1.	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
0.05	1.	0.00	0.00	.001	.001	.001	.001	.001	.001
0.25	1.	.006	.006	.013	.009	.013	.009	.009	.009
0.50	1.	.036	.036	.058	.045	.058	.045	.051	.051
0.75	1.	.096	.096	.119	.102	.119	.102	.115	.115
0.95	1.	.174	.174	.281	[illegible]	.281	.241	.260	.260
0.99	1.	.260	.260	.459	.441	.459	.441	.452	.452
1.00	1.	.474	.474	.538	.495	.538	.495	.496	.496

Table 4.4-34.
Cumulative Probability Distributions for FCONV
For All Radionuclide Groups for the Late Rupture Case.

Percentile	NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
0.00	1.	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
0.01	1.	0.00	0.00	.001	.001	.001	.001	.001	.001
0.05	1.	.001	.001	.003	.003	.003	.003	.003	.003
0.25	1.	.015	.015	.036	.020	.036	.020	.020	.020
0.50	1.	.076	.076	.104	.093	.104	.093	.093	.093
0.75	1.	.183	.183	.256	.225	.256	.225	.225	.225
0.95	1.	.331	.331	.755	.750	.755	.750	.750	.750
0.99	1.	.430	.430	.911	.911	.911	.911	.911	.911
1.00	1.	.464	.464	.947	.947	.947	.947	.947	.947

Table 4.4-35.
Cumulative Probability Distributions for FCONV
For All Radionuclide Groups for the No Containment Failure Case.

Percentile	NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
0.00	0.005	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6
0.01	0.005	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6
0.05	0.005	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6
0.25	0.005	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6
0.50	0.005	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6
0.75	0.005	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6
0.95	0.005	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6
0.99	0.005	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6
1.00	0.005	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6

Table 4.4-36.
Cumulative Probability Distributions for FCONC
For All Radionuclide Groups for the Early Leak
With a Subcooled Suppression Pool Case.

Percentile	NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
0.00	1.	.001	.001	.001	.001	.001	.001	.001	.001
0.01	1.	.017	.017	.005	.005	.005	.005	.005	.005
0.05	1.	.039	.039	.016	.016	.016	.016	.016	.016
0.25	1.	.116	.116	.092	.092	.092	.092	.092	.092
0.50	1.	.274	.274	.243	.243	.243	.243	.243	.243
0.75	1.	.464	.464	.430	.430	.430	.430	.430	.430
0.95	1.	.689	.689	.689	.689	.689	.689	.689	.689
0.99	1.	.800	.800	.800	.800	.800	.800	.800	.800
1.00	1.	.991	.991	.991	.991	.991	.991	.991	.991

Table 4.4-37.
Cumulative Probability Distributions for FCONC
For All Radionuclide Groups for the Early Leak
With a Saturated Suppression Pool Case.

Percentile	NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
0.00	1.	.004	.004	.004	.004	.004	.004	.004	.004
0.01	1.	.012	.012	.012	.012	.012	.012	.012	.012
0.05	1.	.038	.038	.030	.030	.030	.030	.030	.030
0.25	1.	.151	.151	.121	.121	.121	.121	.121	.121
0.50	1.	.286	.286	.261	.261	.261	.261	.261	.261
0.75	1.	.490	.490	.459	.459	.459	.459	.459	.459
0.95	1.	.721	.721	.721	.721	.721	.721	.721	.721
0.99	1.	.852	.852	.852	.852	.852	.852	.852	.852
1.00	1.	.982	.982	.982	.982	.982	.982	.982	.982

Table 4.4-38.
Cumulative Probability Distributions for FCONC
For All Radionuclide Groups for the Early Rupture
With a Subcooled Suppression Pool Case.

Percentile	NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
0.00	1.	.010	.010	.011	.010	.010	.010	.010	.010
0.01	1.	.100	.100	.040	.040	.040	.040	.040	.040
0.05	1.	.225	.225	.163	.163	.163	.163	.163	.163
0.25	1.	.515	.515	.460	.460	.460	.460	.460	.460
0.50	1.	.737	.737	.712	.712	.712	.712	.712	.712
0.75	1.	.862	.862	.838	.838	.838	.838	.838	.838
0.95	1.	.957	.957	.957	.957	.957	.957	.957	.957
0.99	1.	.987	.987	.987	.987	.987	.987	.987	.987
1.00	1.	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00

Table 4.4-39.
Cumulative Probability Distributions for FCONC
For All Radionuclide Groups for the Early Rupture
With a Saturated Suppression Pool Case.

Percentile	NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
0.00	1.	.026	.026	.011	.011	.011	.011	.011	.011
0.01	1.	.156	.156	.041	.041	.041	.041	.041	.041
0.05	1.	.272	.272	.158	.158	.158	.158	.158	.158
0.25	1.	.532	.532	.478	.478	.478	.478	.478	.478
0.50	1.	.750	.750	.706	.706	.706	.706	.706	.706
0.75	1.	.887	.887	.870	.870	.870	.870	.870	.870
0.95	1.	.975	.975	.975	.975	.975	.975	.975	.975
0.99	1.	.995	.995	.995	.995	.995	.995	.995	.995
1.00	1.	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00

Table 4.4-40.
Cumulative Probability Distributions for FCONC
For All Radionuclide Groups for the Late Leak Case.

Percentile	NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
0.00	1.	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
0.01	1.	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
0.05	1.	0.00	0.00	.001	.001	.001	.001	.001	.001
0.25	1.	.006	.006	.013	.009	.013	.009	.009	.009
0.50	1.	.036	.036	.058	.045	.058	.045	.051	.051
0.75	1.	.096	.096	.119	.102	.119	.102	.115	.115
0.95	1.	.174	.174	.281	.241	.281	.241	.260	.260
0.99	1.	.260	.260	.459	.441	.459	.441	.452	.452
1.00	1.	.474	.474	.538	.495	.538	.495	.496	.496

Table 4.4-41.
Cumulative Probability Distributions for FCONC
For all Radionuclide Groups for the Late Rupture Case.

Percentile	NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
0.00	1.	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
0.01	1.	0.00	0.00	.001	.001	.001	.001	.001	.001
0.05	1.	.001	.001	.003	.003	.003	.003	.003	.003
0.25	1.	.015	.015	.036	.020	.036	.020	.020	.020
0.50	1.	.076	.076	.104	.093	.104	.093	.093	.093
0.75	1.	.183	.183	.256	.225	.256	.225	.225	.225
0.95	1.	.331	.331	.755	.750	.755	.750	.750	.750
0.99	1.	.430	.430	.911	.911	.911	.911	.911	.911
1.00	1.	.464	.464	.947	.947	.947	.947	.947	.947

Table 4.4-42.
Cumulative Probability Distributions for FCONC
For All Radionuclide Groups for the No Containment Failure Case.

Percentile	NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
0.00	0.005	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6
0.01	0.005	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6
0.05	0.005	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6
0.25	0.005	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6
0.50	0.005	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6
0.75	0.005	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6
0.95	0.005	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6
0.99	0.005	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6
1.00	0.005	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6	1.0E-6

DFVPA

The probability distribution for this parameter was taken to be the same as that of Grand Gulf[3] since the water depth above the SRV T-quencher at LaSalle (24') is closer to that at Grand Gulf (15') than at Peach Bottom (10'). Only one case was considered and the cumulative probability distribution for this case is shown in Table 4.4-43. The probability distributions for all of the radionuclides were fully correlated for this parameter.

DFCPA

The probability distribution for this parameter was taken to be the same as that of Grand Gulf since the water depth above the downcomers at LaSalle (12') is closer to that at Grand Gulf (7') than at Peach Bottom (4'). Only one case was considered and the cumulative probability distribution for this case is shown in Table 4.4-44. The probability distributions for all of the radionuclides were fully correlated for this parameter.

DFCAV

The probability distribution for this parameter is the same as that used in the Peach Bottom analysis which was modified from that in the draft report of the Grand Gulf analysis.[5] Only one case was considered: the reactor cavity is flooded with a continuous supply of water. This decontamination factor was applied to the release fraction for the after vessel breach releases. Table 4.4-45 contains the cumulative probability distributions for this parameter. The probability distributions for all of the radionuclides were fully correlated for this parameter.

DFSPRV

The probability distributions for this parameter are the same as that used in the Peach Bottom analysis which was was modified from the draft report of the Surry analysis.[6] The possibility of saturated sprays was considered; however, the primary factc in determining the spray DF is whether the sprays are operating, not the temperature of the water. The current distribution was believed to include the possibility of saturated sprays. Therefore, separate cumulative probability distributions were not used for saturated and subcooled sprays. The cumulative probability distribution for this parameter is shown in Table 4.4-46. The probability distributions for all of the radionuclides were fully correlated for this parameter.

FLTI2

The probability distribution for this parameter was developed by the Source Term Expert Panel for NUREG-1150 and are discussed in detail in that report. In the elicitation only two cases were considered: the reactor cavity is dry and there is replenishable water over the debris. However,

Table 4.4-43.
Cumulative Probability Distributions
For All Radionuclide Groups for DFVPA.

Percentile	NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
0.00	1.	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
0.01	1.	1.1	1.1	1.1	1.1	1.1	1.1	1.1	1.1
0.05	1.	1.8	1.8	1.8	1.8	1.8	1.8	1.8	1.8
0.25	1.	16.	16.	16.	16.	16.	16.	16.	16.
0.50	1.	56.	56.	56.	56.	56.	56.	56.	56.
0.75	1.	180.	180.	180.	180.	180.	180.	180.	180.
0.95	1.	2500.	2500.	2500.	2500.	2500.	2500.	2500.	2500.
0.99	1.	4300.	4300.	4300.	4300.	4300.	4300.	4300.	4300.
1.00	1.	5000.	5000.	5000.	5000.	5000.	5000.	5000.	5000.

Table 4.4-44.
Cumulative probability Distributions
For All Radionuclide Groups for DFCPA.

Percentile	NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
0.00	1.	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
0.01	1.	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
0.05	1.	1.2	1.2	1.2	1.2	1.2	1.2	1.2	1.2
0.25	1.	2.6	2.6	2.6	2.6	2.6	2.6	2.6	2.6
0.50	1.	6.8	6.8	6.8	6.8	6.8	6.8	6.8	6.8
0.75	1.	20.	20.	20.	20.	20.	20.	20.	20.
0.95	1.	72.	72.	72.	72.	72.	72.	72.	72.
0.99	1.	94.	94.	94.	94.	94.	94.	94.	94.
1.00	1.	100.	100.	100.	100.	100.	100.	100.	100.

Table 4.4-45.
Cumulative Probability Distributions
For All Radionuclide Groups for DFVPA.

Percentile	NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
0.00	1.	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
0.01	1.	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
0.05	1.	1.1	1.1	1.1	1.1	1.1	1.1	1.1	1.1
0.25	1.	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0
0.50	1.	4.4	4.4	4.4	4.4	4.4	4.4	4.4	4.4
0.75	1.	11.	11.	11.	11.	11.	11.	11.	11.
0.95	1.	41.	41.	41.	41.	41.	41.	41.	41.
0.99	1.	65.	65.	65.	65.	65.	65.	65.	65.
1.00	1.	73.	73.	73.	73.	73.	73.	73.	73.

Table 4.4-46.
Cumulative Probability Distributions
For All Radionuclide Groups for DFSPRV.

Percentile	NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
0.00	1.	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
0.01	1.	1.1	1.1	1.1	1.1	1.1	1.1	1.1	1.1
0.05	1.	1.3	1.3	1.3	1.3	1.3	1.3	1.3	1.3
0.25	1.	4.2	4.2	4.2	4.2	4.2	4.2	4.2	4.2
0.50	1.	11.	11.	11.	11.	11.	11.	11.	11.
0.75	1.	29.	29.	29.	29.	29.	29.	29.	29.
0.95	1.	78.	78.	78.	78.	78.	78.	78.	78.
0.99	1.	95.	95.	95.	95.	95.	95.	95.	95.
1.00	1.	100.	100.	100.	100.	100.	100.	100.	100.

because of the size of the reactor cavity at LaSalle, a replenishable water supply implies that the cavity water will overflow into the suppression pool and can then be used for sprays or injection. This means the conditions in the cavity (i. e., the concentration of radionuclides) will be very similar to those in the suppression pool if there is a replenishable water supply. Three cases were used for the LaSalle analysis: the reactor cavity is dry, the suppression pool is subcooled, and the suppression pool is saturated. For the first case, the Source Term Expert Panel elicitation was used and is shown in Table 4.4-47. The other two cases use the same distributions as FLTI1 and are shown in Tables 4.4-20 and 4.4-21.

DFSPRC

The probability distributions for this parameter were modified from the draft report of the Surry analysis[6] and are the same as those used in the Peach Bottom analysis. The possibility of saturated sprays was considered; however, the primary factor in determining the spray DF is whether the sprays are operating, not the temperature of the water. The current distribution was believed to include the effect of saturated sprays. Therefore, separate cumulative probability distributions were not used for saturated and subcooled sprays. The cumulative probability distributions for this parameter are shown in Table 4.4-48. The probability distributions for all of the radionuclides were fully correlated for this parameter.

4.4.3 Non-Sampled Variables

4.4.3.1 Introduction

Several parameters were included in the source term analysis as best estimates rather than being sampled, including the energy release rates during an accident and the timing of the release of radioactive material. These parameters are described and the quantification sources discussed in this section.

4.4.3.2 Energy Release Rates

The rate of energy release depends on several factors including the accident scenario, whether a containment rupture or leak has occurred, whether venting has occurred, and whether or not DCH has occurred. The cases considered for LaSalle are:

1. short-term scenario with a ruptured containment,

2. short-term scenario with containment leakage,

Table 4.4-47.
Cumulative Probability Distribution
For the Dry Cavity Case for FLTI2.

Percentile	Fraction Released
0.00	1.00
0.01	1.00
0.05	1.00
0.25	1.00
0.[illegible]	1.00
0.[illegible]	1.00
0.9[illegible]	1.00
0.99	1.00
1.00	1.00

Table 4.4-48.
Cumulative Probability Distributions
For all Radionuclide Groups for DFSPRC.

Percentile	NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
0.00	1.	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
0.01	1.	1.1	1.1	1.1	1.1	1.1	1.1	1.1	1.1
0.05	1.	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5
0.25	1.	7.8	7.8	7.8	7.8	7.8	7.8	7.8	7.8
0.50	1.	17.	17.	17.	17.	17.	17.	17.	17.
0.75	1.	29.	29.	29.	29.	29.	29.	29.	29.
0.95	1.	480.	480.	480.	480.	480.	480.	480.	480.
0.99	1.	860.	860.	860.	860.	860.	860.	860.	860.
1.00	1.	1000.	1000.	1000.	1000.	1000.	1000.	1000.	1000.

3. scenarios with injection initially working followed by failure of injection with a ruptured containment,

4. scenarios with injection initially working followed by failure of injection with containment leakage,

5. long-term scenarios,

6. ATWS scenarios,

7. a scenario with no containment heat removal; however, low pressure injection is operating which results in a high enough containment pressure to result in reclosure of the SRV's, thus repressurizing the primary system which results in failure of the low pressure injection systems to cool the core, thus leading to core damage, and

8. a scenario with venting before vessel breach.

In addition to these cases, extra energy was released if a DCH event occurred. The following paragraphs document the source and derivation of the various energy release rates for the scenarios described above.

The short-term accident release rates were based on short-term station blackout calculations performed with the MELCOR code and described in Volume 3 of this report. Although calculations were performed with ADS working and failed (i.e., low and high RPV pressure), the resulting energy release rates were not significantly different; therefore, the energy release rates were not differentiated on the basis of RPV pressure. Calculations with varying containment failure sizes did result in a difference, the release rates were, therefore, differentiated on the basis of failure size (leak or rupture). Also, only two release segments are necessary to characterize the release for these accidents since the containment does not fail before vessel breach.

The actual value of the energy release rates was determined by dividing the total energy increase to the environmental Volume in the MELCOR model by the duration of the release. Table 4.4-49 contains the energy release rates for the short-term accident scenarios.

Table 4.4-49.
Energy Release Rates for the Short-term Accident Scenarios
For Containment Rupture and Containment Leakage.

Failure Type	Release Phase First Release (W)	Second Release (W)
Rupture	7.28E8	3.00E8
Leak	4.89E8	1.25E8

Scenarios with injection initially working followed by failure of injection were further divided into two scenarios: (1) the RPV either was depressurized during the injection phase and remained at low pressure or was not depressurized and remained at high pressure, or (2) the RPV was depressurized during the injection phase and repressurized followi , injection failure. The energy release rates for both scenarios were assessed to be the same and are based on the intermediate station blackout calculation performed with the MELCOR code, described in Volume 3 of this report, in which the RPV repressurized following injection failure. This calculation included containment rupture; therefore, it was necessary to use the ratio of the short-term containment rupture and containment leakage energy release rates to determine the energy release rates for the containment leakage cases. Only two release rates were necessary since the containment does not fail before vessel breach. Table 4.4-50 contains the two energy release rates for this scenario.

The energy release rates for long-term scenarios were based on the long-term station blackout calculation performed with MELCOR and described in Volume 3 of this report. The containment failed by rupture in the calculation. Because of the slow nature of this accident, ruptures and leaks were not differentiated in the release rates, since the difference between ruptures and leaks, in this time frame, were assessed to be insignificant to risk. In contrast to the two previous accident scenarios, three release phases are necessary to characterize the release during the long-term accident since the containment fails before vessel breach. Table 4.4-51 contains the energy release rates for the long-term scenarios.

The release rates for the ATWS scenarios were based on those for the long-term accident scenarios because the containment conditions are similar (e.g., suppression pool is saturated) although the decay heat is much higher than in the long-term scenarios. To compensate for the difference in decay heats, the release rates for the long-term scenarios were multiplied by the ratio of the decay heat at the beginning of the release for the ATWS scenario to the decay heat at the beginning of the release for the long-term scenario. Table 4.4-52 contains the release rates for the ATWS scenarios.

The same ratioing technique was used to determine the release rates for the scenario with no containment heat removal. However, since low pressure injection is operating in this scenario, the containment continues to pressurize until the SRVs reclose. The primary system repressurizes which fails the low pressure injection system for both the ruptured containment and containment leakage cases. The release rates from the scenarios with injection initially working and then later failing were used as the basis. Since containment does not fail before vessel breach, only two releases are specified. Table 4.4-53 contains the release fractions for this scenario.

The release rates for the scenarios with venting before vessel breach were determined exactly as the ATWS release rates were; by ratioing the decay heats at the times in question and using the long-term calculation as a basis. Table 4.4-54 contains these release rates.

Table 4.4-50.
Energy Release Rates for the Accident Scenarios
With Injection Initially Operating Followed by
Injection Failure For Containment Rupture
And Containment Leakage.

Failure Type	Release Phase First Release (W)	Second Release (W)
Rupture	4.44E8	3.18E7
Leak	2.98E8	1.32E7

Table 4.4-51.
Energy Release Rates for the Long-term Accident Scenario.

	Release Phase	
First Release (W)	Second Release (W)	Third Release (W)
7.15E6	1.26E7	1.66E7

Table 4.4-52.
Energy Release Rates for the ATWS Accident Scenario.

	Release Phase	
First Release (W)	Second Release (W)	Third Release (W)
2.00E7	3.53E7	4.65E7

Table 4.4-53.
Energy Release Rates for the Accident Scenarios With
No Containment Heat Removal and Low Pressure Injection Initially.

Failure Type	Release Phase First Release (W)	Second Release (W)
Rupture	5.64E8	4.04E7
Leak	3.79E8	1.68E7

Table 4.4-54.
Energy Release Rates for the Venting Accident Scenario.

	Release Phase	
First Release (W)	Second Release (W)	Third Release (W)
9.37E6	1.65E7	2.17E7

In addition to the above release rates dependence on accident scenario and containment failure mode, the energy release due to a DCH event was assessed. The energy released during a DCH event was thought to be important only if it caused containment failure. Therefore, the energy necessary to increase the containment pressure to the median containment failure pressure (195 psig) was calculated. This amount of energy was then divided by the puff duration (15 minutes) to determine the energy release rate due to DCH. The puff duration was taken to be consistent with the assumptions used in analyzing the NUREG-1150 plants. This value is 2.22E+07 W.

4.4.3.3 Release Times

The release timing was determined for the same accident scenarios for which warning time distributions were developed. These scenarios include ATWS with and without successful venting, short-term scenarios, scenarios with injection available initially followed by injection failure with the RPV either remaining at high pressure, remaining at low pressure, or repressurizing following the injection failure, venting before vessel breach scenarios, scenarios with no containment heat removal although low pressure injection is available until the containment pressure is high enough to fail the SRV's resulting in low pressure injection failure, and long-term scenarios. For most of these scenarios, containment failure can occur at various times during the accident. This section describes the basis for the assigned release times for the various scenarios.

First Release

The first release for an ATWS scenario will begin when the containment fails or is vented. A detailed discussion of the containment failure time is included in section 4.4.2.3. The first release begins at 3530 seconds for an ATWS scenario with successful venting and at 6175 seconds for an ATWS scenario without successful venting. The duration of the first release was assessed to be 9720 seconds which is the time from core damage to vessel breach in the short-term station blackout without ADS calculation performed with the MELCOR code and described in Volume 3 of this report. In the case of no containment failure during the accident, the first release for ATWS without successful venting is used since some leakage from the containment will occur.

The first release for a short-term scenario begins at containment failure which can occur at vessel breach, when late venting occurs, following a drywell rupture caused by a pedestal failure, or late due to overpressurization of the containment. The time of vessel breach was determined from short-term station blackout calculations performed with the MELCOR code. Late venting was assessed to be 2 hours after vessel breach. This was based on the MELCOR calculations and the allowance of 2 hours for AC recovery necessary to vent. An early (after vessel breach) pedestal failure could occur from 1 to 6 hours following vessel breach; therefore, for the source term analysis the average of 3.5 hours after vessel breach was used. Late containment failure was determined from the short-term station blackout calculation without ADS. Table 4.4-55 contains the possible first release times for the short-term scenarios. The duration of the first release was 15 minutes. This length of time was taken to represent the containment failure puff. If the containment does not fail for this accident progression bin, the first release time is set equal to the first release time for a containment failure at vessel breach since some leakage from the containment will occur during the accident.

First release times for scenarios with injection initially available followed by injection failure with the RPV either remaining at high or low pressure are very similar to those for the short-term scenarios. Containment failure can occur at vessel breach (either due to the vessel breach or due to early venting), due to late venting, due to a pedestal failure causing drywell rupture, or late due to overpressurization. The time of vessel breach is based on an intermediate station blackout calculation performed with the MELCOR code and described in Volume 3 of this report. In this calculation, the RPV repressurized following loss of injection; therefore, the timing had to be modified for use as a release time. The time calculated to repressurize the RPV was subtracted from the vessel breach time to estimate the vessel breach time for an accident with the RPV remaining at either high or low pressure. The times after vessel breach for late venting and pedestal failure are the same as for the short-term scenarios. The time of late containment failure was also based on the intermediate station blackout calculation performed with MELCOR. Table 4.4-56 contains the possible first release times for these scenarios. The duration of the first release was taken to be 15 minutes which represents the containment failure puff. If the containment does not fail for this bin, the first release time is taken as the release time at vessel breach since some leakage from the containment will occur during the accident.

Another scenario is similar to the above except that the RPV depressurizes during the injection phase then repressurizes after injection failure. To account for the repressurization phase, 4.7 hours is added to each of the above times for this scenario. The duration of the first release for this scenario is the same as that above.

The first release for the scenarios with no containment heat removal, but with low pressure injection operating until the SRV's reclose due to high containment pressure may, occur at vessel breach or late due to overpressurization. Both of these times are based on the long-term and intermediate term station blackout calculations performed with the MELCOR code and described in Volume 3 of this report. The time injection is lost was estimated from the long-term calculation as the time the containment reaches 85 psig (the SRV's reclose at this pressure). The intermediate calculation was then used to estimate the time from loss of injection to vessel breach as 13.9 hours. The amount of time between vessel breach and late containment failure was determined by estimating the time for the containment to reach 195 psig in the intermediate calculation. This time was 6 hours resulting in a release time of 49 hours for this scenario if the containment fails due to overpressurization. The duration of the first

Table 4.4-55.
Possible First Release Times for the Short-term Scenarios.

Containment Failure Mode	First Release Time (sec)
At vessel breach	14400.
Late venting	21600.
Pedestal failure	27000.
Late failure	64800.

Table 4.4-56.
Possible First Release Times for the Scenarios
With Injection Initially Followed by Injection Failure
With the RPV Remaining at High or Low Pressure.

Containment Failure Mode	First Release Time (sec)
At vessel breach	61200.
Late venting	68400.
Pedestal failure	73800.
Late failure	100800.

release is 15 minutes which is the containment failure puff. If the containment does not fail, the first release is taken to begin at vessel breach since some leakage from the containment will occur.

As with the above scenario, the first release time for a late venting scenario was based on the long-term and intermediate term station blackout calculations performed with the MELCOR code. The time of venting (i.e., the time the containment pressure reaches 60 psig) was estimated from the long-term calculation, while the time to core damage from loss of injection (at which point the release begins) was estimated from the intermediate calculation. The value used for the first release time was 28.3 hours. The duration of the first release is 4.1 hours which is the time from core damage to vessel breach, based on the intermediate station blackout calculation performed with MELCOR.

The first release in the long-term scenarios begins at core damage since the containment has already failed. This value was based on the long-term station blackout calculation performed with MELCOR and was assessed to be 58.3 hours. The duration of the first release was also based on this calculation and was 7.3 hours.

Second Release

The second release begins when the first release is completed. For this reason only the duration of the second release will be discussed in this chapter.

As noted earlier, several scenarios do not need three release phases to characterize the release since the containment does not fail before core damage. These scenarios include the short-term scenario and scenarios with injection initially available followed by failure of injection caused by battery failure, failure of the diesel generators to continue to run, or failure of low pressure injection caused by high containment pressure. All of these scenarios have a second release duration of zero. All other scenarios considered in the analysis have containment failure before core damage. These include ATWS, early venting, and the long-term scenarios. For each of these scenarios, the second release will be the vessel breach puff which is taken to be 15 minutes.

Third Release

The third release begins when the second release is completed. The third release is dependent only on the containment failure size and the occurrence of core-concrete interactions. The release durations chosen were based primarily on MELCOR calculations described in Volume 3 of this report and on engineering judgement. The long-term station blackout calculation was used in determining the release duration for the rupture cases with core-concrete interactions. Releases with no CCI were assessed to be four hours shorter than releases with CCI and releases through leaks were taken to be longer than those through ruptures. Table 4.4-57 contains the release durations used for the various combinations. If the containment does not fail, the cases of containment leakage were used for the release duration.

Table 4.4-57.
Third Release Durations.

Containment Failure Mode	CCI?	Release duration (sec)
rupture	yes	21600.
rupture	no	7200.
leak	yes	25200.
leak	no	10800.

4.4.3.4 Other Non-sampled Variables

Several other variables were not sampled. These include:

1. the delay time from when a evacuation was called to when the population actually begins to leave,

2. the fraction of the radionuclides released in the initial puff when the containment fails late in the accident,

3. the release elevation,

4. the fraction of the core participating in direct containment heating (FHPE),

5. the fraction of the core participating in ex-vessel steam explosions (EVSE),

6. the fraction of initial iodine core inventory scrubbed by the cavity water during CCI release (CAVWI),

7. fraction of the core material that leaves the vessel anytime at or after vessel breach (FLV),

8. fraction of initial core inventory of iodine scrubbed by the pool (POOLI),

9. fraction of core material leaving the core after vessel breach that is available for CCI (XCCI),

10. the fraction of the radionuclides that go directly into the drywell without passing through the suppression pool due to a stuck open SRV tailpipe vacuum breaker or a LOCA (FPLBY), and

11. the release fractions from the vessel at vessel breach (VBPUF).

The delay time of 35 minutes was based on a study performed by NUS.* This time was not sampled since the uncertainty in the warning time was assessed to include the uncertainty in the delay time.

The fraction of the radionuclides released in the puff when the containment fails late depends on the failure mode of the containment. If a late containment rupture occurs, 90% of the radionuclides are released in the puff; however, if a late containment leak occurs or the containment does not fail, 50% of the radionuclides are released in the initial release. Radionuclides are released because of technical specification leakage even if the containment does not structurally fail. The above fractions were taken to be the same as those used in the Peach Bottom analysis.[11]

The release elevation was fixed at 30 m although releases could actually occur at different elevations (i.e., the location of the reactor building failure is a random variable). This variable was not sampled since the relative importance of the release elevation was determined to be small in comparison to the energy of release as determined in sensitivity studies performed for the Peach Bottom analysis.[11]

Two cases for the fraction of the core participating in direct containment heating are considered. The first case is that of high core mobility, for which the fraction participating is 0.4. The second case is that of low core mobility for which the fraction participating is 0.1.**

The fraction of the core participating in ex-vessel steam explosions is sampled in the Accident Progression Event Tree and the nominal values are 0.2 for the high ex-vessel steam explosion case and 0.05 for the low ex-vessel steam explosion case.

The fraction of iodine scrubbed by the cavity water during CCI release is taken to be the same as the fraction scrubbed by the pool.

The fraction of the core material that leaves the vessel at or after vessel breach is assumed to be 1.0 if vessel breach has occurred and 0.0 if vessel breach has not occurred.

The fraction of iodine scrubbed by the pool is determined from the cumulative amount in the pool by mass conservation from what is not accounted for elsewhere.

The fraction of core material leaving the core after vessel breach that is available for CCI is determined by conservation of mass from what is left over from ex-vessel steam explosions or DCH.

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* Letter from G. D. Kaiser (NUS) to D. J. Alpert (SNL), January 13, 1986.

** Memo from J. E. Kelly (SNL) to E. D. Gorham-Bergeron (SNL), "Mobility at Vessel Breach Question," February 8, 1989.

The fraction of the release that is directed into the drywell due to failed-open SRV tailpipe vacuum breakers or a LOCA depends on the pressure of the RPV. If the pressure of the RPV is high, the fraction released directly into the drywell is 0.39. If the pressure of the RPV is low, then the fraction is 1.0. These numbers are the same as those used in the Peach Bottom analysis.[11]

The release fractions from the vessel during the 15-minute vessel breach puff are based on the short-term station blackout calculation performed with MELCOR and described in Volume 3 of this report. The fractions are shown in Table 4.4-58.

Table 4.4-58.
Vessel Breach Puff Release Fractions.

Radionuclide Group	Release Fraction
Noble Gases	1.49E-4
I	2.45E-5
Cs	4.56E-3
Te	1.16E-4
Sr	4.29E-6
Ru	0.0
La	0.0
Ce	0.0
Ba	4.29E-6

4.5 RESULTS

The mean source terms for the dominant early fatality and latent cancer risk bins based on the fractional contribution to mean risk, FCMR, (see Chapter 6 for a discussion on FCMR) for each of the summary plant damage state (PDS) groups are tabulated below. The PDS groups are: (1) seismic, (2) fire, (3) flood, (4) ATWS, (5) transient, (6) transient-LOCA, and (7) LOCA. The mean source terms were calculated using the LASSOR code by arithmetically averaging over all the source terms calculated for that bin in all of the 400 LHS observations.

Table 4.5-1 contains the mean source terms for the early fatality risk dominant bins for the seismic sequences. All of these dominant bins are characterized by two releases (i.e., first and third) and thus, the second release is set to zero. The mean source terms for the early fatality risk dominant bins for the fire sequences are shown in Table 4.5-2. Table 4.5-3 contains the mean source terms for the early fatality risk dominant bins for the flood sequences. As for the seismic sequences, the fire and flood dominant bins are characterized by two releases. The mean source terms for the early fatality risk dominant bins for the ATWS sequences and transient sequences are shown in Tables 4.5-4 and 4.5-5, respectively. The mean source terms for the early fatality risk dominant bins for the transient-LOCA sequences are shown in Table 4.5-6. The LOCA sequences did not appear among the dominant sequences for early fatalities.

Table 4.5-1.
The Mean Source Terms for the FCMR Early Fatality Risk Dominant Bins
For the Seismic Sequences.

ORDER	BIN	WARN (S)	ELEV (M)	POWER (W)	START (S)	DUR (S)	RELEASE FRACTIONS NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
1	BBBACJIAAAAABA	4.1E+04	3.0E+01	4.7E+08	1.0E+05	9.0E+02	9.0E-01	1.2E-01	1.2E-01	1.9E-01	1.6E-01	2.0E-03	1.4E-02	2.2E-02	1.3E-01
				0.0E+00	1.0E+05	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.2E+07	1.0E+05	2.2E+04	1.0E-01	1.3E-02	1.4E-02	2.1E-02	1.8E-02	2.2E-04	1.6E-03	2.5E-03	1.5E-02
2	ACBAEJBKAAAAAA	3.7E+04	3.0E+01	4.4E+08	7.8E+04	9.0E+02	1.5E-01	1.6E-02	1.8E-02	1.3E-04	3.2E-05	6.4E-05	3.0E-05	3.0E-05	4.6E-05
				0.0E+00	7.9E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.2E+07	7.9E+04	2.2E+04	8.5E-01	4.9E-01	4.9E-01	3.4E-01	2.4E-01	5.4E-07	7.3E-03	1.1E-02	1.5E-01
3	BCBACJIAAAAABA	5.7E+04	3.0E+01	4.7E+08	1.2E+05	9.0E+02	9.0E-01	1.1E-01	1.1E-01	2.1E-01	1.5E-01	5.9E-03	1.1E-02	1.7E-02	1.1E-01
				0.0E+00	1.2E+05	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.2E+07	1.2E+05	2.2E+04	1.0E-01	1.2E-02	1.2E-02	2.3E-02	1.7E-02	6.6E-04	1.2E-03	1.9E-03	1.2E-02
4	ACBACJIAAAAABA	5.6E+04	3.0E+01	4.7E+08	1.2E+05	9.0E+02	9.0E-01	1.6E-01	1.7E-01	2.6E-01	2.2E-01	2.0E-03	1.9E-02	3.0E-02	1.8E-01
				0.0E+00	1.2E+05	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.2E+07	1.2E+05	2.2E+04	1.0E-01	1.8E-02	1.9E-02	2.9E-02	2.4E-02	2.2E-04	2.1E-03	3.4E-03	2.0E-02
5	BBBBFJIAAAAABA	3.8E+04	3.0E+01	4.4E+08	1.0E+05	9.0E+02	9.0E-01	2.2E-01	2.3E-01	4.4E-01	4.4E-01	2.4E-04	3.9E-02	6.4E-02	3.7E-01
				0.0E+00	1.0E+05	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.2E+07	1.0E+05	2.2E+04	1.0E-01	2.4E-02	2.5E-02	4.9E-02	4.9E-02	2.6E-05	4.3E-03	7.1E-03	4.1E-02
6	BCBACJIAAAABBA	5.5E+04	3.0E+01	4.7E+08	1.2E+05	9.0E+02	9.0E-01	1.3E-01	1.3E-01	2.2E-01	2.2E-01	1.6E-03	2.0E-02	3.2E-02	1.8E-01
				0.0E+00	1.2E+05	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.2E+07	1.2E+05	2.2E+04	1.0E-01	1.4E-02	1.4E-02	2.5E-02	2.4E-02	1.8E-04	2.2E-03	3.6E-03	2.0E-02
7	ACBACJAKAAAABA	3.7E+04	3.0E+01	4.7E+08	7.8E+04	9.0E+02	1.0E+00	2.0E-01	[illegible]	1.8E-01	1.6E-01	2.3E-02	1.5E-02	7.4E-02	1.6E-01
				0.0E+00	7.9E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.2E+07	7.9E+04	2.2E+04	0.0E+00	1.7E-03	4.3E-05	5.6E-03	7.5E-04	7.6E-10	2.4E-04	2.1E-04	3.5E-04
	ACBACJGKAAAAAA	5.4E+04	3.0E+01	3.2E+08	7.8E+04	9.0E+02	5.7E-01	9.5E-03	1.1E-02	5.2E-04	3.1E-04	3.0E-04	1.6E-04	2.1E-04	3.3E-04
				0.0E+00	7.9E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				1.3E+07	7.9E+04	2.5E+04	4.3E-01	9.1E-02	8.9E-02	6.9E-02	6.8E-02	2.4E-04	7.0E-03	1.1E-02	6.1E-02
9	ACBACJIAAAABBA	5.5E+04	3.0E+01	4.7E+08	1.2E+05	9.0E+02	9.0E-01	1.3E-01	1.3E-01	2.2E-01	2.2E-01	1.6E-03	2.0E-02	3.2E-02	1.8E-01
				0.0E+00	1.2E+05	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.2E+07	1.2E+05	2.2E+04	1.0E-01	1.4E-02	1.4E-02	2.5E-02	2.4E-02	1.8E-04	2.2E-03	3.6E-03	2.0E-02
10	ABBACJIAAAAABA	3.8E+04	3.0E+01	4.7E+08	1.0E+05	9.0E+02	9.0E-01	2.2E-01	2.3E-01	4.0E-01	3.9E-01	1.1E-03	3.6E-02	5.8E-02	3.4E-01
				0.0E+00	1.0E+05	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.2E+07	1.0E+05	2.2E+04	1.0E-01	2.4E-02	2.6E-02	4.5E-02	4.4E-02	1.2E-04	4.0E-03	6.4E-03	3.7E-02
11	BCBACJAKAABABA	4.6E+04	3.0E+01	4.7E+08	7.8E+04	9.0E+02	9.9E-01	3.0E-01	3.0E-01	2.3E-01	1.7E-01	2.4E-02	1.5E-02	7.6E-02	1.7E-01
				0.0E+00	7.9E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.2E+07	7.9E+04	2.2E+04	7.8E-03	6.7E-02	8.5E-02	1.5E-01	2.1E-02	2.5E-09	9.8E-04	1.3E-03	2.2E-02
12	BCAACJICAAAABA	5.5E+04	3.0E+01	4.7E+08	9.1E+04	9.0E+02	9.0E-01	5.6E-02	5.6E-02	8.9E-02	5.7E-02	4.3E-03	2.7E-03	4.3E-03	4.2E-02
				0.0E+00	9.2E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.2E+07	9.2E+04	2.2E+04	1.0E-01	6.2E-03	6.2E-03	9.9E-03	6.3E-03	4.8E-04	3.0E-04	4.7E-04	4.6E-03
13	BCBACJBCAAAABA	5.1E+04	3.0E+01	4.7E+08	7.8E+04	9.0E+02	3.7E-01	6.7E-02	7.2E-02	6.2E-02	4.5E-02	4.3E-02	5.6E-03	5.6E-03	5.2E-02
				0.0E+00	7.9E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.2E+07	7.9E+04	2.2E+04	6.3E-01	6.2E-01	6.3E-01	5.9E-01	6.2E-01	5.0E-03	7.1E-02	1.2E-01	5.8E-01
14	BBBACJAKAACAAA	4.0E+04	3.0E+01	4.7E+08	6.1E+04	9.0E+02	1.0E+00	6.2E-01	6.2E-01	5.7E-01	5.1E-01	7.0E-02	4.7E-02	2.3E-01	5.1E-01
				0.0E+00	6.2E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.2E+07	6.2E+04	2.2E+04	0.0E+00	1.1E-03	3.7E-05	4.8E-03	6.3E-04	6.5E-10	2.0E-04	1.8E-04	2.9E-04
15	ACAACJBKAAAAAA	5.6E+04	3.0E+01	4.7E+08	7.8E+04	9.0E+02	8.2E-01	8.9E-03	1.0E-02	6.4E-03	4.6E-03	4.5E-03	1.0E-03	1.3E-03	5.0E-[illegible]
				0.0E+00	7.9E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+[illegible]
				3.2E+07	7.9E+04	2.2E+04	1.8E-01	1.2E-01	1.1E-01	9.1E-02	5.5E-02	5.0E-04	4.7E-03	3.0E-03	4.6E-[illegible]

Table 4.5-2.
The Mean Source Terms for the FCMR Early Fatality Risk Dominant Bins
For Fire Sequences.

ORDER	BIN	WARN (S)	ELEV (M)	POWER (W)	START (S)	DUR (S)	RELEASE FRACTIONS NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
1	CAABFJICCAAAAB	4.0E+03	3.0E+01	7.3E+08	2.7E+04	9.0E+02	9.0E-01	1.1E-01	1.1E-03	7.1E-04	1.7E-04	1.2E-08	6.8E-06	6.5E-06	1.2E-04
				0.0E+00	2.8E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.0E+08	2.8E+04	2.2E+04	1.0E-01	1.2E-02	1.3E-04	7.9E-05	1.9E-05	1.4E-09	7.6E-07	7.2E-07	1.4E-05
2	CFBADGCKAAABAB	2.0E+05	3.0E+01	7.2E+06	2.1E+05	2.6E+04	4.0E-01	1.1E-03	2.6E-04	1.7E-05	2.6E-07	1.7E-08	6.8E-09	6.8E-09	5.8E-07
				1.3E+07	2.4E+05	9.0E+02	1.2E-01	2.0E-04	2.9E-04	1.9E-06	7.1E-08	0.0E+00	0.0E+00	0.0E+00	7.1E-08
				1.7E+07	2.4E+05	2.2E+04	4.8E-01	6.2E-01	6.2E-01	4.5E-01	2.3E-01	1.2E-07	7.3E-03	1.1E-02	1.2E-01
3	CFBBECICAAABBB	1.9E+05	3.0E+01	7.2E+06	2.1E+05	2.6E+04	3.0E-01	8.1E-05	2.8E-05	2.0E-05	3.2E-07	2.0E-08	8.0E-09	8.0E-09	7.1E-07
				1.3E+07	2.4E+05	9.0E+02	3.5E-02	6.0E-04	7.1E-04	5.5E-06	1.4E-06	3.0E-06	1.3E-06	1.3E-06	2.1E-06
				1.7E+07	2.4E+05	2.2E+04	6.7E-01	6.5E-01	6.6E-01	6.2E-01	6.5E-01	5.3E-03	7.5E-02	1.2E-01	6.1E-01
4	CDABEEICAAABAB	9.0E+04	3.0E+01	9.4E+06	1.0E+05	1.5E+04	9.0E-01	1.1E-02	6.9E-03	2.7E-03	6.9E-04	1.6E-04	5.0E-05	2.3E-04	7.2E-04
				1.7E+07	1.2E+05	9.0E+02	5.3E-03	6.6E-04	1.1E-03	8.4E-05	3.7E-05	5.6E-05	1.9E-05	1.9E-05	4.4E-05
				2.2E+07	1.2E+05	2.2E+04	9.8E-02	1.1E-01	1.2E-01	1.2E-01	9.5E-02	8.2E-04	3.3E-03	5.8E-03	6.5E-02
5	CDBBEEIKAAABAB	8.9E+04	3.0E+01	9.4E+06	1.0E+05	1.5E+04	7.1E-01	5.6E-03	3.8E-03	2.1E-03	6.9E-04	9.0E-05	3.0E-05	1.2E-04	7.1E-04
				1.7E+07	1.2E+05	9.0E+02	1.5E-02	1.5E-03	2.2E-03	7.3E-04	3.9E-04	4.2E-04	9.8E-05	9.6E-05	4.6E-04
				2.2E+07	1.2E+05	2.2E+04	2.8E-01	1.4E-01	1.4E-01	9.2E-02	7.8E-02	1.9E-04	7.9E-03	1.2E-02	6.5E-02
6	CFAACGCKAAAAAB	2.0E+05	3.0E+01	2.9E+07	2.1E+05	2.6E+04	7.4E-01	1.8E-03	8.0E-04	1.2E-04	4.9E-05	1.1E-05	5.7E-06	3.5E-05	5.0E-05
				1.3E+07	2.4E+05	9.0E+02	2.6E-02	9.2E-03	1.2E-02	5.5E-03	3.2E-03	7.8E-03	1.4E-03	8.2E-04	3.8E-03
				1.7E+07	2.4E+05	2.2E+04	2.3E-01	1.4E-01	1.2E-01	8.9E-02	7.4E-02	1.1E-03	7.0E-03	5.3E-03	6.9E-02
7	CFAACGCKAAABAB	2.0E+05	3.0E+01	2.9E+07	2.1E+05	2.6E+04	3.0E-01	2.3E-06	5.2E-07	1.0E-08	2.6E-10	5.5E-12	2.2E-12	2.2E-12	6.2E-10
				1.3E+07	2.4E+05	9.0E+02	7.1E-02	1.1E-03	1.3E-03	3.0E-04	2.3E-05	4.4E-05	2.6E-05	2.6E-05	4.0E-05
				1.7E+07	2.4E+05	2.2E+04	6.3E-01	2.2E-01	2.3E-01	2.1E-01	2.2E-01	3.4E-03	2.0E-02	1.6E-02	2.0E-01
8	CFABECBKAAAAAB	2.0E+05	3.0E+01	7.2E+06	2.1E+05	2.6E+04	5.9E-01	1.5E-02	1.2E-02	3.1E-03	1.2E-04	7.7E-05	1.8E-06	2.7E-06	1.8E-04
				1.3E+07	2.4E+05	9.0E+02	2.1E-02	1.0E-02	1.3E-02	5.5E-03	4.1E-03	3.9E-03	5.4E-04	5.4E-04	4.8E-03
				1.7E+07	2.4E+05	2.2E+04	3.9E-01	2.1E-01	2.3E-01	1.7E-01	1.2E-01	1.3E-02	1.2E-02	2.3E-02	1.1E-01
9	CDABEEIKAAAAAB	9.0E+04	3.0E+01	9.4E+06	1.0E+05	1.5E+04	7.4E-01	7.9E-03	5.4E-03	2.5E-03	7.6E-04	1.4E-04	6.0E-05	2.6E-04	7.9E-04
				1.6E+07	1.2E+05	9.0E+02	1.3E-02	5.6E-03	8.1E-03	3.4E-03	2.3E-03	2.6E-03	5.1E-04	5.5E-04	2.7E-03
				2.2E+07	1.2E+05	2.2E+04	2.5E-01	1.4E-01	1.5E-01	1.1E-01	7.4E-02	1.0E-03	4.1E-03	4.6E-03	5.5E-02
10	CFBBEBIKAAABAB	2.0E+05	3.0E+01	7.2E+06	2.1E+05	2.6E+04	7.3E-01	3.0E-03	2.5E-03	9.4E-04	2.6E-05	2.1E-05	1.6E-06	4.1E-06	5.2E-05
				1.3E+07	2.4E+05	9.0E+02	1.4E-02	7.5E-03	1.0E-02	1.2E-03	2.0E-04	3.7E-04	2.1E-04	2.1E-04	2.7E-04
				1.7E+07	2.4E+05	2.2E+04	2.6E-01	1.5E-01	1.6E-01	1.6E-01	1.6E-01	5.7E-05	1.3E-02	2.1E-02	1.3E-01
11	CFBABGCKAAAAAB	2.0E+05	3.0E+01	2.9E+07	2.1E+05	2.6E+04	9.8E-01	3.3E-05	1.3E-05	7.4E-07	2.0E-08	1.8E-08	1.4E-09	3.8E-09	4.2E-08
				1.3E+07	2.4E+05	9.0E+02	6.2E-03	3.0E-02	5.1E-02	1.2E-01	1.5E-01	1.4E-01	1.8E-02	1.8E-02	1.7E-01
				1.7E+07	2.4E+05	2.2E+04	9.1E-03	8.6E-02	8.5E-02	1.7E-01	2.8E-01	2.3E-04	2.6E-02	4.3E-02	2.4E-01
12	CDBBEEIKAAAAAB	8.9E+04	3.0E+01	9.4E+06	1.0E+05	1.5E+04	7.0E-01	4.7E-03	3.1E-03	1.7E-03	5.1E-04	6.8E-05	2.2E-05	9.2E-05	5.3E-04
				1.[illegible]E+07	1.2E+05	9.0E+02	1.5E-02	6.0E-03	8.2E-03	2.2E-03	1.4E-03	1.5E-03	3.0E-04	3.1E-04	1.6E-03
				2.2E+07	1.2E+05	2.2E+04	2.9E-01	1.5E-01	1.6E-01	9.3E-02	6.7E-02	1.3E-04	6.0E-03	9.6E-03	5.3E-02
13	CFBADGCKAABBAB	2.0E+05	3.0E+01	7.2E+06	2.1E+05	2.6E+04	4.0E-01	3.8E-03	8.9E-04	5.8E-05	8.8E-07	5.5E-08	2.3E-08	2.3E-08	2.0E-06
				1.3E+07	2.4E+05	9.0E+02	1.2E-01	2.0E-04	2.9E-04	1.9E-06	7.1E-08	0.0E+00	0.0E+00	0.0E+00	7.1E-08
				1.7E+07	2.4E+05	2.2E+04	4.8E-01	6.2E-01	6.2E-01	4.5E-01	2.3E-01	1.2E-07	7.3E-03	1.1E-02	1.2E-01
14	CDABEEICAACBAB	9.0E+04	3.0E+01	9.4E+06	1.0E+05	1.5E+04	8.9E-01	1.5E-02	1.1E-02	5.0E-03	1.6E-03	3.2E-04	1.4E-04	6.7E-04	1.7E-03
				1.7E+07	1.2E+05	9.0E+02	5.7E-03	6.6E-04	1.0E-03	8.1E-05	4.1E-05	6.0E-05	2.1E-05	2.0E-05	4.8E-05
				2.2E+07	1.2E+05	2.2E+04	1.1E-01	1.2E-01	1.3E-01	1.3E-01	9.6E-02	4.7E-04	2.8E-03	5.0E-03	6.3E-02
15	CFBBACACAAAABB	1.9E+05	3.0E+01	2.9E+07	2.1E+05	2.6E+04	6.5E-01	2.9E-03	2.7E-03	2.0E-03	9.3E-04	1.1E-04	3.[illegible]E-05	1.5E-04	9.4E-04
				1.3E+07	2.4E+05	9.0E+02	1.4E-01	1.4E-01	1.5E-01	1.3E-01	9.2E-02	8.9E-02	1.4E-02	1.4E-02	1.1E-01
				1.7E+07	2.4E+05	2.2E+04	2.1E-01	2.5E-01	2.4E-01	2.4E-01	3.0E-01	1.8E-03	3.8E-02	6.0E-02	2.7E-01

Table 4.5-3.
The Mean Source Terms for the FCMR Early Fatality Risk Dominant Bins
For the Flood Sequences.

ORDER	BIN	WARN (S)	ELEV (M)	POWER (W)	START (S)	DUR (S)	RELEASE FRACTIONS NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
1	DABDFJIFABAAAB	4.7E+03	3.0E+01	7.3E+08	2.2E+04	9.0E+02	9.0E-01	7.8E-02	8.6E-03	1.1E-02	6.5E-03	1.0E-04	4.3E-04	8.7E-04	4.7E-03
				0.0E+00	2.3E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.0E+08	2.3E+04	2.2E+04	1.0E-01	8.6E-03	9.6E-04	1.2E-03	7.3E-04	1.1E-05	4.8E-05	9.7E-05	5.3E-04
2	DAABFJ AAAAAB	4.6E+03	3.0E+01	7.3E+08	2.2E+04	9.0E+02	9.0E-01	5.8E-02	2.1E-02	3.0E-02	2.0E-02	2.3E-04	1.1E-03	1.1E-03	1.5E-02
				0.0E+00	2.3E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.0E+08	2.3E+04	2.2E+04	1.0E-01	6.5E-03	2.3E-03	3.3E-03	2.2E-03	2.6E-05	1.2E-04	1.2E-04	1.7E-03
3	DABBFJIFAAAAAB	4.8E+03	3.0E+01	7.3E+08	2.2E+04	9.0E+02	9.0E-01	5.4E-02	2.2E-02	3.0E-02	2.0E-02	2.5E-04	1.8E-03	2.9E-03	1.7E-02
				0.0E+00	2.3E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.0E+08	2.3E+04	2.2E+04	1.0E-01	6.0E-03	2.5E-03	3.4E-03	2.3E-03	2.8E-05	2.0E-04	3.2E-04	1.8E-03
4	DAADFJIFABAAAB	4.6E+03	3.0E+01	7.3E+08	2.2E+04	9.0E+02	9.0E-01	9.1E-02	9.7E-03	9.8E-03	4.8E-03	5.5E-05	2.3E-04	4.6E-04	3.1E-03
				0.0E+00	2.3E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.0E+08	2.3E+04	2.2E+04	1.0E-01	1.0E-02	1.1E-03	1.1E-03	5.4E-04	6.1E-06	2.6E-05	5.1E-05	3.5E-04
5	DABEFJIFACACAB	4.7E+03	3.0E+01	7.3E+08	2.2E+04	9.0E+02	6.4E-01	3.8E-02	7.4E-04	7.2E-04	2.8E-04	4.4E-05	2.0E-05	9.5E-05	2.9E-04
				0.0E+00	2.3E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.0E+08	2.3E+04	7.2E+03	7.1E-02	4.2E-03	8.2E-05	8.0E-05	3.1E-05	4.9E-06	2.2E-06	1.1E-05	3.2E-05
6	DABDFJGKABAAAB	4.7E+03	3.0E+01	4.9E+08	1.4E+04	9.0E+02	6.8E-01	4.0E-03	3.5E-03	1.7E-03	8.8E-04	1.3E-04	7.6E-05	3.8E-04	9.0E-04
				0.0E+00	1.5E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				1.2E+08	1.5E+04	2.5E+04	3.2E-01	1.1E-01	2.4E-02	1.6E-02	1.2E-02	5.1E-05	8.2E-04	1.5E-03	8.2E-03
7	DAAEFJIFACACAB	4.6E+03	3.0E+01	7.3E+08	2.2E+04	9.0E+02	6.6E-01	3.6E-02	7.7E-04	7.7E-04	2.6E-04	4.8E-05	1.8E-05	6.3E-05	2.6E-04
				0.0E+00	2.3E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.0E+08	2.3E+04	7.2E+03	7.3E-02	4.0E-03	8.5E-05	8.6E-05	2.9E-05	5.3E-06	2.0E-06	7.0E-06	2.9E-05
8	DABDFJGKABAABB	4.8E+03	3.0E+01	4.9E+08	1.4E+04	9.0E+02	5.8E-01	3.3E-04	2.1E-03	1.3E-04	3.7E-05	4.5E-06	2.7E-06	1.3E-05	3.7E-05
				0.0E+00	1.5E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				1.3E+08	1.5E+04	2.5E+04	4.2E-01	2.0E-01	1.1E-01	3.7E-02	5.1E-03	7.2E-10	3.8E-04	4.5E-04	1.9E-03
9	DAADAJAKACAABB	4.6E+03	3.0E+01	7.5E+08	1.4E+04	9.0E+02	9.1E-01	7.9E-02	9.0E-02	4.9E-02	3.6E-02	3.7E-02	6.4E-03	7.6E-03	4.1E-02
				0.0E+00	1.5E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.0E+08	1.5E+04	7.2E+03	0.0E+00	1.1E-01	3.5E-02	[illegible]	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
10	DAAPFJGKAAAAAB	4.7E+03	3.0E+01	4.9E+08	1.4E+04	9.0E+02	7.0E-01	4.0E-03	3.5E-03	1.2E-03	1.7E-04	4.6E-05	1.0E-05	2.9E-05	1.9E-04
				0.0E+00	1.5E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				1.2E+08	1.5E+04	2.5E+04	3.0E-01	9.6E-02	6.5E-02	4.6E-02	3.2E-02	7.2E-05	1.7E-03	1.5E-03	2.3E-02
11	DABRFJGKAAAAAB	4.6E+03	3.0E+01	4.9E+08	1.4E+04	9.0E+02	7.3E-01	2.8E-03	2.7E-03	1.7E-03	1.2E-03	1.7E-04	1.1E-04	5.4E-04	1.2E-03
				0.0E+00	1.5E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				1.3E+08	1.5E+04	2.5E+04	2.7E-01	1.0E-01	6.6E-02	6.0E-02	5.0E-02	5.1E-04	5.1E-03	7.9E-03	4.1E-02
12	DABDFJIFCBAAAB	4.7E+03	3.0E+01	7.3E+08	2.2E+04	9.0E+02	9.0E-01	7.3E-02	8.8E-03	9.3E-03	4.4E-03	3.5E-05	2.0E-04	3.8E-04	2.7E-03
				0.0E+00	2.3E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.0E+08	2.3E+04	2.2E+04	1.0E-01	8.1E-03	9.8E-04	1.0E-03	4.8E-04	3.9E-06	2.3E-05	4.2E-05	3.0E-04
13	DAADFJGKABAAAB	4.6E+03	3.0E+01	4.9E+08	1.4E+04	9.0E+02	6.7E-01	2.9E-03	2.7E-03	1.1E-03	2.3E-04	5.4E-05	1.4E-05	4.1E-05	2.5E-04
				0.0E+00	1.5E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	[illegible]	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				1.3E+08	1.5E+04	2.5E+04	[illegible]	1.1E-01	2.3E-02	[illegible]	6.6E-03	1.9E-05	2.6E-04	5.0E-04	3.8E-03
14	DAADAJAKABAABB	4.6E+03	3.0E+01	7.5E+08	1.4E+04	9.0E+02	8.3E-01	1.3E-01	1.4E-01	7.4E-02	4.4E-02	4.3E-02	9.2E-03	1.0E-02	5.0E-02
				0.0E+00	1.5E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.0E+08	1.5E+04	2.2E+04	1.7E-01	1.2E-01	6.4E-02	3.8E-02	2.0E-02	5.3E-05	9.4E-04	1.7E-03	1.2E-02
15	DABDFJGBABAAAB	4.7E+03	3.0E+01	7.3E+08	1.4E+04	9.0E+02	6.6E-01	4.2E-03	3.8E-03	1.8E-03	9.9E-04	1.4E-04	8.5E-05	4.3E-04	1.0E-03
				0.0E+00	1.5E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.0E+08	1.5E+04	2.2E+04	3.1E-01	1.1E-01	2.4E-02	1.7E-02	1.2E-02	5.7E-05	8.8E-04	1.7E-03	8.6E-03

Table 4.5-4.
The Mean Source Terms for the FCMR Early Fatality Risk Dominant Bins
For ATWS Sequences.

ORDER	BIN	WARN (S)	ELEV (M)	POWER (W)	START (S)	DUR (S)	RELEASE FRACTIONS NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
1	EAADFEIKABAAAB	2.6E+03	3.0E+01	2.0E+07	3.5E+03	9.7E+03	7.5E-01	9.7E-03	7.3E-03	4.0E-03	9.5E-04	2.1E-04	6.6E-05	2.3E-04	1.0E-03
				3.5E+07	1.3E+04	9.0E+02	1.5E-04	6.2E-05	1.2E-03	2.8E-05	1.0E-06	0.0E+00	0.0E+00	0.0E+00	1.0E-06
				4.7E+07	1.4E+04	2.2E+04	2.5E-01	1.3E-01	3.1E-02	2.2E-02	1.9E-02	5.8E-05	1.2E-03	2.4E-03	1.3E-02
2	EABDEEIKABABAB	2.6E+03	3.0E+01	2.0E+07	3.5E+03	9.7E+03	7.2E-01	8.4E-03	6.2E-03	3.2E-03	4.1E-04	8.6E-05	1.8E-05	6.6E-05	4.8E-04
				3.5E+07	1.3E+04	9.0E+02	1.4E-02	5.5E-04	9.1E-04	3.4E-04	4.2E-04	4.5E-04	6.7E-05	6.6E-05	4.8E-04
				4.7E+07	1.4E+04	2.2E+04	2.7E-01	1.6E-01	4.3E-02	2.3E-02	1.4E-02	3.2E-06	7.9E-04	1.5E-03	1.0E-02
3	EAADFEIKABAABB	2.6E+03	3.0E+01	2.0E+07	3.5E+03	9.7E+03	5.5E-01	6.0E-02	4.5E-02	1.6E-02	5.8E-04	5.1E-04	2.2E-05	5.1E-05	8.7E-04
				3.5E+07	1.3E+04	9.0E+02	1.5E-04	1.2E-04	2.3E-03	4.9E-05	1.8E-06	0.0E+00	0.0E+00	0.0E+00	1.8E-06
				4.7E+07	1.4E+04	2.2E+04	4.5E-01	2.7E-01	2.0E-01	5.7E-02	3.9E-02	1.6E-04	2.9E-03	5.8E-03	3.1E-02
4	EABDEEIKABAAAB	2.6E+03	3.0E+01	2.0E+07	3.5E+03	9.7E+03	6.8E-01	4.8E-03	3.5E-03	2.1E-03	5.5E-04	8.2E-05	2.5E-05	1.0E-04	5.8E-04
				3.5E+07	1.3E+04	9.0E+02	1.6E-02	5.2E-03	7.0E-03	2.3E-03	1.7E-03	1.8E-03	3.3E-04	3.6E-04	2.0E-03
				4.7E+07	1.4E+04	2.2E+04	3.0E-01	1.7E-01	4.8E-02	2.7E-02	1.7E-02	1.2E-04	1.0E-03	2.0E-03	1.2E-02
5	EABDFEIKABAAAB	2.6E+03	3.0E+01	2.0E+07	3.5E+03	9.7E+03	7.1E-01	7.9E-03	6.1E-03	3.3E-03	1.7E-03	2.4E-04	1.3E-04	6.3E-04	1.7E-03
				3.5E+07	1.3E+04	9.0E+02	1.5E-04	6.0E-05	1.1E-03	2.7E-05	1.0E-06	0.0E+00	0.0E+00	0.0E+00	1.0E-06
				4.7E+07	1.4E+04	2.2E+04	2.9E-01	1.2E-01	3.7E-02	2.9E-02	1.9E-02	1.4E-04	1.4E-03	2.5E-03	1.4E-02
6	EAABFEIKAAAAAB	2.6E+03	3.0E+01	2.0E+07	3.5E+03	9.7E+03	7.5E-01	9.5E-03	6.7E-03	2.8E-03	5.6E-04	1.4E-04	4.0E-05	1.4E-04	6.1E-04
				3.5E+07	1.3E+04	9.0E+02	1.5E-04	7.4E-05	1.4E-03	3.3E-05	1.2E-06	0.0E+00	0.0E+00	0.0E+00	1.2E-06
				4.6E+07	1.4E+04	2.2E+04	2.5E-01	1.5E-01	1.4E-01	1.3E-01	1.1E-01	5.7E-04	6.8E-03	5.1E-03	6.7E-02
7	EABDFEIKACACAB	2.8E+03	3.0E+01	2.0E+07	3.5E+03	9.7E+03	6.6E-01	1.0E-02	9.5E-03	8.4E-03	6.5E-03	9.1E-04	5.9E-04	2.9E-03	6.6E-03
				3.5E+07	1.3E+04	9.0E+02	1.5E-04	1.7E-05	3.1E-04	7.6E-06	2.8E-07	0.0E+00	0.0E+00	0.0E+00	2.8E-07
				4.7E+07	1.4E+04	7.2E+03	0.0E+00	8.4E-02	3.5E-04	9.0E-05	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
8	EABDEEBKABAAAB	2.5E+03	3.0E+01	2.0E+07	3.5E+03	9.7E+03	6.3E-01	5.3E-03	3.0E-03	1.2E-03	1.0E-04	2.7E-05	7.9E-06	3.6E-05	1.3E-04
				3.5E+07	1.3E+04	9.0E+02	1.9E-02	8.7E-03	1.1E-02	2.8E-03	1.7E-03	1.9E-03	3.8E-04	3.8E-04	2.0E-03
				4.7E+07	1.4E+04	2.2E+04	3.5E-01	1.8E-01	1.1E-01	6.7E-02	4.2E-02	1.2E-03	2.6E-03	5.1E-03	3.1E-02
9	EAADFEIBABAAAB	2.4E+03	3.0E+01	2.0E+07	3.5E+03	9.7E+03	7.2E-01	1.2E-02	9.1E-03	4.7E-03	1.1E-03	2.5E-04	8.0E-05	2.9E-04	1.2E-03
				3.5E+07	1.3E+04	9.0E+02	1.5E-04	6.2E-05	1.2E-03	2.8E-05	1.1E-06	0.0E+00	0.0E+00	0.0E+00	1.1E-06
				4.7E+07	1.4E+04	2.2E+04	2.8E-01	1.6E-01	3.1E-02	2.2E-02	1.9E-02	8.2E-05	1.4E-03	2.7E-03	1.5E-02
10	EAAACHCKAAAAAB	2.7E+03	3.0E+01	4.2E+07	6.2E+03	9.7E+03	7.2E-01	1.9E-03	1.4E-03	7.9E-04	3.2E-04	5.5E-05	2.0E-05	5.2E-05	3.2E-04
				3.5E+07	1.6E+04	9.0E+02	2.8E-02	1.2E-02	1.7E-02	7.6E-03	3.0E-03	3.2E-03	7.1E-04	7.1E-04	3.8E-03
				4.7E+07	1.7E+04	2.2E+04	2.5E-01	1.5E-01	1.8E-01	6.7E-02	1.4E-02	3.4E-06	8.6E-0[illegible]	4.9E-04	1.1E-02
11	EAADEEIKABAAAB	2.7E+03	3.0E+01	2.0E+07	3.5E+03	9.7E+03	7.1E-01	6.0E-03	4.2E-03	2.1E-03	7.0E-04	1.3E-04	4.2E-05	1.1E-04	7.2E-04
				3.5E+07	1.3E+04	9.0E+02	1.5E-02	7.1E-03	9.4E-03	3.2E-03	1.8E-03	2.2E-03	4.2E-04	4.2E-04	2.2E-03
				4.7E+07	1.4E+04	2.2E+04	2.7E-01	1.7E-01	6.0E-02	4.9E-02	2.4E-02	5.2E-05	1.6E-03	3.1E-03	1.7E-02
12	EABEFEIKACACAB	2.5E+03	3.0E+01	2.0E+07	3.5E+03	9.7E+03	7.0E-01	6.7E-03	5.0E-03	3.0E-03	1.5E-03	2.1E-04	1.1E-04	5.6E-04	1.5E-03
				3.5E+07	1.3E+04	9.0E+02	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				4.6E+07	1.4E+04	7.2E+03	0.0E+00	3.8E-02	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00

Table 4.[illegible]5.
The Mean Source Terms for the FCM Early Fatality Risk Dominant Bins
For Transient Sequences.

ORDER	BIN	WARN (S)	ELEV (M)	POWER (W)	START (S)	DUR (S)	RELEASE FRACTIONS NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
1	GAbDFJIFABAAAA	4.7E+03	3.0E+01	7.3E+08	2.2E+04	9.0E+02	9.0E-01	7.9E-02	8.4E-03	1.0E-02	8.3E-03	9.8E-05	4.2E-04	8.5E-04	4.6E-03
				0.0E+00	2.3E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.0E+08	2.3E+04	2.2E+04	1.0E-01	8.8E-03	9.3E-04	1.1E-03	7.0E-04	1.1E-05	4.7E-05	9.4E-05	5.1E-04
2	GAABFJICAAAABA	4.6E+03	3.0E+01	7.3E+08	2.7E+04	9.0E+02	9.0E-01	8.9E-02	4.4E-02	6.6E-02	5.6E-02	1.9E-03	3.8E-03	4.9E-03	4.4E-02
				0.0E+00	2.8E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.0E+08	2.8E+04	2.2E+04	1.0E-01	9.9E-03	4.9E-03	7.3E-03	6.2E-03	2.1E-04	4.2E-04	5.4E-04	4.9E-03
3	GABDFJIFCBAAAA	4.6E+03	3.0E+01	7.3E+08	2.2E+04	9.0E+02	9.0E-01	7.8E-02	8.4E-03	1.0E-02	5.3E-03	9.8E-05	4.2E-04	8.5E-04	4.6E-03
				0.0E+00	2.3E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.0E+08	2.3E+04	2.2E+04	1.0E-01	8.7E-03	9.3E-04	1.1E-03	7.0E-04	1.1E-05	4.7E-05	9.4E-05	5.1E-04
4	GBBDEJBKBBAAAA	4.0E+04	3.0E+01	4.4E+08	6.1E+04	9.0E+02	7.4E-01	1.0E-02	1.2E-02	3.0E-03	2.2E-03	2.2E-03	4.0E-04	4.3E-04	2.5E-03
				0.0E+00	6.2E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.2E+07	6.2E+04	2.2E+04	2.6E-01	1.3E-01	1.3E-01	7.1E-02	4.0E-02	1.3E-03	2.7E-03	5.3E-03	3.0E-02
5	GABDFJGKABAAAA	4.7E+03	3.0E+01	4.9E+08	1.4E+04	9.0E+02	6.7E-01	3.9E-03	3.5E-03	1.6E-03	8.3E-04	1.2E-04	7.1E-05	3.6E-04	8.4E-04
				0.0E+00	1.5E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				1.2E+08	1.5E+04	2.5E+04	3.3E-01	1.2E-01	2.3E-02	1.6E-02	1.1E-02	5.9E-05	7.9E-04	1.5E-03	7.9E-03
6	GAABFJIFAAAAAA	4.6E+03	3.0E+01	7.3E+08	2.2E+04	9.0E+02	9.0E-01	5.8E-02	2.1E-02	2.9E-02	2.0E-02	2.3E-04	1.1E-03	1.1E-03	1.5E-02
				0.0E+00	2.3E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.0E+08	2.3E+04	2.2E+04	1.0E-01	6.4E-03	2.3E-03	3.2E-03	2.2E-03	2.5E-05	1.2E-04	1.2E-04	1.6E-03
7	GAAACJICAAAABA	4.7E+03	3.0E+01	7.5E+08	2.7E+04	9.0E+02	9.0E-01	7.0E-02	5.3E-02	7.4E-02	5.0E-02	4.2E-03	3.0E-03	4.6E-03	3.9E-02
				0.0E+00	2.8E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.0E+08	2.8E+04	2.2E+04	1.0E-01	7.7E-03	5.8E-03	8.2E-03	5.5E-03	4.7E-04	3.4E-04	5.1E-04	4.3E-03
8	GBABEJICAAABBA	4.1E+04	3.0E+01	4.4E+08	7.4E+04	9.0E+02	9.0E-01	6.0E-02	6.0E-02	1.2E-01	9.7E-02	2.4E-03	3.8E-03	6.9E-03	7.0E-02
				0.0E+00	7.5E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.2E+07	7.5E+04	2.2E+04	1.0E-01	6.7E-03	6.7E-03	1.4E-02	1.1E-02	2.7E-04	4.2E-04	7.7E-04	7.8E-03
9	GAADFJIFABAAAA	4.6E+03	3.0E+01	7.3E+08	2.2E+04	9.0E+02	9.0E-01	8.5E-02	8.0E-03	9.1E-03	4.6E-03	8.4E-05	2.5E-04	4.8E-04	3.0E-03
				0.0E+00	2.3E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.0E+08	2.3E+04	2.2E+04	1.0E-01	9.4E-03	8.9E-04	1.0E-03	5.1E-04	9.3E-06	2.8E-05	5.4E-05	3.4E-04
10	GAABFJICAAAAAA	4.5E+03	3.0E+01	7.3E+08	2.7E+04	9.0E+02	9.0E-01	7.1E-02	2.1E-02	3.2E-02	2.5E-02	3.4E-04	1.4E-03	1.3E-03	1.9E-02
				0.0E+00	2.8E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.0E+08	2.8E+04	2.2E+04	1.0E-01	7.9E-03	2.3E-03	3.6E-03	2.7E-03	3.8E-05	1.5E-04	1.5E-04	2.1E-03
11	GAADFJIFCBAAAA	4.6E+03	3.0E+01	7.3E+08	2.2E+04	9.0E+02	9.0E-01	8.5E-02	8.0E-03	9.1E-03	4.6E-03	8.4E-05	2.5E-04	4.8E-04	3.0E-03
				0.0E+00	2.3E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	[illegible]
				3.0E+08	2.3E+04	2.2E+04	1.0E-01	9.4E-03	8.9E-04	1.0E-03	5.1E-04	9.3E-06	2.8E-05	5.4E-05	3.4E-04
12	GABDFJIFABAAAB	4.7E+03	3.0E+01	7.3E+08	2.2E+04	9.0E+02	9.0E-01	7.8E-02	8.8E-03	1.1E-02	6.4E-03	1.0E-04	3.9E-04	8.0E-04	4.6E-03
				0.0E+00	2.3E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	[illegible]	0.0E+00
				3.0E+08	2.3E+04	2.2E+04	1.0E-01	8.7E-03	9.8E-04	1.2E-03	7.1E-04	1.1E-05	4.4E-05	[illegible]	5.1E-04
13	GFAAFCGCAAAAAB	1.9E+05	3.0E+01	7.2E+06	2.1E+05	2.6E+04	1.9E-01	1.6E-05	3.4E-06	1.3E-08	2.4E-10	2.5E-12	9.8E-13	[illegible]	8.4E-10
				1.3E+07	2.4E+05	9.0E+02	1.5E-04	1.1E-04	2.0E-03	4.6E-05	1.7E-06	0.0E+00	0.0E+00	[illegible]	1.7E-06
				1.7E+07	2.4E+05	2.2E+04	8.1E-01	4.0E-01	4.1E-01	3.5E-01	3.5E-01	6.4E-04	2.8E-02	[illegible]	3.0E-01
14	GABBFJGKAAAAAA	4.6E+03	3.0E+01	4.9E+08	1.4E+04	9.0E+02	7.3E-01	2.7E-03	2.7E-03	1.6E-03	1.1E-03	1.6E-04	9.9E-05	5.0E-04	1.1E-03
				0.0E+00	1.5E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				1.3E+08	1.5E+04	2.5E+04	2.7E-01	1.0E-01	6.4E-02	5.9E-02	4.9E-02	4.7E-04	4.9E-03	7.6E-03	4.0E-02
15	GAAACJIFAAAAAA	4.6E+03	3.0E+01	7.5E+08	2.2E+04	9.0E+02	9.0E-01	4.0E-02	2.3E-02	2.9E-02	1.9E-02	1.2E-03	1.1E-03	1.1E-03	1.5E-02
				0.0E+00	2.3E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.0E+08	2.3E+04	2.2E+04	1.0E-01	4.4E-03	2.5E-03	3.3E-03	2.1E-03	1.4E-04	1.3E-04	1.2E-04	1.6E-03

Table 4.5-6.
The Mean Source Terms for the FCMR Early Fatality Risk Dominant Bins
For the Transient-LOCA Sequences.

ORDER	BIN	WARN (S)	ELEV (M)	POWER (W)	START (S)	DUR (S)	RELEASE FRACTIONS NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
1	HDBBEEIKAAABAB	8.9E+04	3.0E+01	9.[illegible]E+06	1.0E+05	1.5E+04	6.9E-01	4.[illegible]E-03	3.1E-03	1.9E-03	7.7E-04	9.6E-05	3.4E-05	1.4E-04	7.8E-04
				1.7E+07	1.2E+05	9.0E+02	1.6E-02	1.1E-03	1.6E-03	6.2E-04	3.3E-04	3.6E-04	8.7E-05	8.5E-05	3.8E-04
				2.2E+07	1.2E+05	2.2E+04	2.9E-01	1.4E-01	1.4E-01	8.5E-02	8.0E-02	1.6E-04	8.1E-03	1.3E-02	6.6E-02
2	HDABFEIKAAAAAB	8.9E+04	3.0E+01	9.4E+06	1.0E+05	1.5E+04	7.3E-01	1.1E-02	7.8E-03	3.2E-03	6.3E-04	1.5E-04	4.2E-05	1.4E-04	6.7E-04
				1.7E+07	1.2E+05	9.0E+02	1.5E-04	7.1E-05	1.3E-03	3.0E-05	1.1E-06	0.0E+00	0.0E+00	0.0E+00	1.1E-06
				2.2E+07	1.2E+05	2.2E+04	2.7E-01	1.2E-01	1.3E-01	9.5E-02	7.5E-02	4.0E-04	4.7E-03	3.8E-03	5.8E-02
3	HABBFJIBAAAAAA	4.6E+03	3.0E+01	7.3E+08	6.5E+04	9.0E+02	9.0E-01	7.1E-02	2.8E-02	3.9E-02	2.9E-02	1.7E-04	2.1E-03	3.7E-03	2.3E-02
				0.0E+00	6.6E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.0E+08	6.6E+04	2.2E+04	1.0E-01	7.9E-03	3.1E-03	4.4E-03	3.2E-03	1.8E-05	2.4E-04	4.1E-04	2.6E-03
4	HDBBEEIKAAAAAB	8.5E+04	3.0E+01	9.4E+06	1.0E+05	1.5E+04	7.0E-01	4.7E-03	3.1E-03	1.7E-03	5.2E-04	6.9E-05	2.3E-05	9.4E-05	5.4E-04
				1.7E+07	1.2E+05	9.0E+02	1.5E-02	5.9E-03	8.0E-03	2.1E-03	1.4E-03	1.4E-03	2.9E-04	3.0E-04	1.6E-03
				2.2E+07	1.2E+05	2.2E+04	2.9E-01	1.5E-01	1.6E-01	9.3E-02	6.9E-02	1.4E-04	6.3E-03	1.0E-02	5.5E-02
5	HDBBEEIBAAABAB	8.9E+04	3.0E+01	9.4E+06	1.0E+05	1.5E+04	6.5E-01	3.0E-03	1.7E-03	7.2E-04	3.4E-05	1.3E-05	1.5E-06	4.6E-06	5.0E-05
				1.7E+07	1.2E+05	9.0E+02	1.8E-02	1.3E-0[illegible]	1.8E-03	6.8E-04	3.5E-04	3.7E-04	7.9E-05	8.3E-05	4.2E-04
				2.2E+07	1.2E+05	2.2E+04	3.4E-01	1.5E-01	1.6E-01	9.0E-02	6.9E-02	7.2E-05	5.5E-03	9.0E-03	5.6E-02
6	HDABEEICAAABAB	9.0E+04	3.0E+01	9.4E+06	1.0E+05	1.5E+04	8.8E-01	1.2E-02	7.7E-03	3.1E-03	7.7E-04	1.8E-04	5.6E-05	2.6E-04	8.1E-04
				1.7E+07	1.2E+05	9.0E+02	5.9E-03	7.4E-04	1.2E-03	9.3E-05	4.1E-05	5.8E-05	2.1E-05	2.0E-05	4.9E-05
				2.2E+07	1.2E+05	2.2E+04	1.1E-01	1.2E-01	1.3E-01	1.3E-01	1.0E-01	9.2E-04	3.6E-03	6.4E-03	7.1E-02
7	HBBDEJBKBBAAAA	3.9E+04	3.0E+01	4.4E+08	6.1E+04	9.0E+02	6.3E-01	1.1E-02	1.4E-02	4.4E-03	3.5E-03	3.6E-03	5.6E-04	5.9E-04	4.1E-03
				0.0E+00	6.2E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.2E+07	6.2E+04	2.2E+04	3.7E-01	1.7E-01	1.7E-01	8.0E-02	6.9E-02	2.7E-03	5.2E-03	1.0E-02	5.8E-02
8	HDABEEBKAAAAAB	9.0E+04	3.0E+01	9.4E+06	1.0E+05	1.5E+04	6.6E-01	6.3E-03	4.9E-03	1.7E-03	1.7E-04	5.6E-05	8.9E-06	2.3E-05	2.0E-04
				1.7E+07	1.2E+05	9.0E+02	1.7E-02	8.9E-03	1.2E-02	2.8E-03	1.6E-03	1.7E-03	[illegible].4E-04	3.4E-04	1.9E-03
				2.2E+07	1.2E+05	2.2E+04	3.3E-01	1.9E-01	2.0E-01	1.3E-01	6.7E-02	3.7E-03	4.7E-03	7.8E-03	5.5E-02
9	HDABEEIKAAAAAB	9.0E+04	3.0E+01	9.4E+06	1.0E+05	1.5E+04	7.3E-01	8.2E-03	5.6E-03	2.5E-03	7.6E-04	1.4E-04	6.0E-05	2.6E-04	8.0E-04
				1.7E+07	1.2E+05	9.0E+02	1.4E-02	5.5E-03	8.0E-03	3.3E-03	2.3E-03	2.6E-03	5.2E-04	5.5E-04	2.6E-03
				2.2E+07	1.2E+05	2.2E+04	2.5E-01	1.4E-01	1.5E-01	1.1E-01	7.4E-02	1.1E-03	4.0E-03	4.7E-03	5.5E-02
10	HDBBEEICAAABBB	8.6E+04	3.0E+01	9.4E+06	1.0E+05	1.5E+04	3.0E-01	8.1E-05	2.8E-05	2.0E-05	3.2E-07	2.0E-08	8.0E-09	8.0E-09	7.1E-07
				1.7E+07	1.2E+05	9.0E+02	3.5E-02	6.0E-04	7.1E-04	5.5E-06	1.4E-06	3.0E-06	1.3E-06	1.3E-06	2.1E-06
				2.2E+07	1.2E+05	2.2E+04	6.7E-01	6.5E-01	6.6E-01	6.2E-01	6.5E-01	5.3E-03	7.5E-02	1.2E-01	6.1E-01
11	HABJFJIFABAAAA	4.7E+03	3.0E+01	7.3E+08	2.2E+04	9.0E+02	9.0E-01	7.9E-02	9.2E-03	1.1E-02	6.6E-03	1.0E-04	4.1E-04	8.1E-04	4.8E-03
				0.0E+00	2.3E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.0E+08	2.3E+04	2.2E+04	1.0E-01	8.8E-03	1.0E-03	1.2E-03	7.4E-04	1.1E-05	4.5E-05	9.0E-05	5.4E-04
12	HABDFJIFCBAAAA	4.7E+03	3.0E+01	7.3E+08	2.[illegible]E+04	9.0E+02	9.0E-01	7.3E-02	8.7E-03	1.1E-02	6.9E-03	1.2E-04	4.7E-04	9.6E-04	5.1E-03
				0.0E+00	2.3E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.0E+08	2.3E+04	2.2E+04	1.0E-01	6.1E-03	9.7E-04	1.2E-03	7.7E-04	1.4E-05	5.2E-05	1.1E-04	5.7E-04
13	HDABFEICAAAAAB	8.8E+04	3.0E+01	9.4E+06	1.0E+05	1.5E+04	7.9E-01	5.2E-03	3.5E-03	1.4E-03	2.2E-04	5.4E-05	1.2E-05	3.2E-05	2.4E-04
				1.7E+07	1.2E+05	9.0E+02	1.5E-04	9.0E-05	1.7E-03	3.8E-05	1.4E-06	0.0E+00	0.0E+00	0.0E+00	1.4E-06
				2.2E+07	1.2E+05	2.2E+04	2.1E-01	1.5E-01	1.5E-01	1.3E-01	1.1E-01	8.0E-05	6.3E-03	3.8E-03	8.6E-02
14	HABBFJIBAAAAAB	4.9E+03	3.0E+01	7.3E+08	6.5E+04	9.0E+02	9.0E-01	6.5E-02	2.7E-02	3.6E-02	2.2E-02	1.1E-04	1.6E-03	2.8E-03	1.8E-02
				0.0E+00	6.6E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.0E+08	6.6E+04	2.2E+04	1.0E-01	7.3E-03	3.0E-03	4.1E-03	2.5E-03	1.2E-05	1.8E-04	3.1E-04	2.0E-03
15	HDBBFEIKAAAA[illegible]	8.9E+04	3.0E+01	9.4E+06	1.0E+05	1.5E+04	7.1E-01	5.0E-03	4.0E-03	2.6E-03	1.6E-03	2.1E-04	1.2E-04	5.7E-04	1.6E-03
				1.7E+07	1.2E+05	9.0E+02	1.5E-04	7.3E-05	1.4E-03	3.1E-05	1.1E-06	0.0E+00	0.0E+00	0.0E+00	1.1E-06
				2.2E+07	1.2E+05	2.2E+04	2.9E-01	1.4E-01	1.4E-01	1.0E-01	7.5E-02	1.7E-04	7.0E-03	1.1E-02	6.2E-02

The mean source terms for the latent cancer risk dominant bins are contained in Tables 4.5-7 through 4.5-13. Table 4.5-7 contains the mean source terms for the latent cancer dominant bins for seismic sequences. As for the early fatality dominant bins, these source terms are characterized by two releases. The mean source terms for the latent cancer risk dominant bins for the fire sequences are shown in Table 4.5-8. The mean source terms for the latent cancer risk dominant bins for the flood and ATWS sequences are shown in Tables 4.5-9 and 4.5-10, respectively. Table 4.5-11 contains the mean source terms for the latent cancer risk dominant bins for LOCA sequences. The mean source terms for latent cancer risk dominant bins for the transient and transient-LOCA sequences are shown in Table 4.5-12 and 4.5-13, respectively.

Table 4.5-14 contains the mean source terms for the seven supergroups: 1) seismic, 2) fire, 3) flood, 4) ATWS, 5) LOCA, 6) transient, and 7) transient-LOCA. The mean source terms were determined by frequency weighting all of the source terms from all of the observations for each supergroup.

The total complementary cumulative distribution functions, CCDFs, for the I, Cs, Sr, and La radionuclide classes are shown in Figure 4.5-1. This figure presents information on both source term size and frequency. It indicates the frequency with which different values of the release fraction are exceeded, and displays the uncertainty in that frequency. The curves in Figure 4.5-1 are derived in the following manner: for each observation, a frequency for each APB is obtained. Calculation of the source terms for the APBs gives a total release fraction for each APB. When all the APBs are considered, a curve of exceedance frequency vs. release fraction can be plotted for each observation.

Figure 4.5-1 is a summary presentation of these curves for the 400 observations in the sample. Instead of placing all 400 curves on one figure, only four statistical measures are shown. These measures are generated by analyzing the curves as follows. For each release fraction on the abscissa, there are 400 values of the exceedance frequency (one for each observation). From these 400 values, it is possible to calculate mean, median (50th quantile), 95th quantile and 5th quantile values. When this is done for each value of the release fraction and the results plotted, the curves in Figure 4.5-1 are obtained. Thus, Figure 4.5-1 provides information on the relationship between the size of the release fractions and the frequency at which these release fractions are exceeded, as well as the variation in that relationship between the observations in the sample.

As an illustration of the information in Figure 4.5-1, the mean frequency (yr^{-1}) at which a release fraction of 1.E-05 is exceeded is roughly, 1.0E-04, 7.0E-05, 2.0E-05, and 5.0E-05 for the I, Cs, Sr, and La release classes, respectively. For a release fraction of 0.1, the corresponding mean exceedance frequencies are 2.0E-05, 2.0E-05, 8.0E-06, and <<1.0E-08, respectively. The flat portion of the CCDF shown for the I and Cs release

Table 4.5-7.
The Mean Source Terms for the FCMR Latent Cancer Risk Dominant Bins
For Seismic Sequences.

ORDER	BIN	WARN (S)	ELEV (M)	POWER (W)	START (S)	DUR (S)	RELEASE FRACTIONS NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
1	BBAAFJGKAAAABA	4.3E+04	3.0E+01	3.0E+08	6.1E+04	9.0E+02	6.6E-01	1.6E-04	1.9E-03	6.3E-05	5.1E-06	9.8E-07	2.7E-07	7.3E-07	6.3E-06
				0.0E+00	6.2E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				1.3E+07	6.2E+04	2.5E+04	3.4E-01	3.0E-01	3.3E-01	5.0E-02	7.7E-04	6.6E-10	5.1E-05	7.9E-05	5.7E-04
2	BCBAEJBKAAAAAA	5.0E+04	3.0E+01	4.4E+08	7.8E+04	9.0E+02	5.0E-01	6.5E-03	7.9E-03	8.5E-05	2.3E-05	5.6E-05	2.5E-05	2.5E-05	3.2E-05
				0.0E+00	7.9E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.2E+07	7.9E+04	2.2E+04	5.0E-01	2.2E-01	2.2E-01	1.7E-01	1.4E-01	2.2E-04	9.7E-03	1.6E-02	1.1E-01
3	BCAACJIBAAAAAA	5.4E+04	3.0E+01	4.7E+08	1.2E[illegible]	[illegible].0E+02	9.0E-01	3.1E-02	2.8E-02	3.3E-02	2.3E-02	1.0E-03	1.3E-03	1.2E-03	1.7E-02
				0.0E+00	1.2E+[illegible]	0.0E+00	0.0E+00	0.0E+00	0.[illegible]+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.2E+07	1.2E+05	2.2E+04	1.0E-01	3.5E-03	3.1E-03	3.6E-03	2.5E-03	1.1E-04	1.5E-04	1.4E-04	1.9E-03
4	BCAACJIGAAAABA	5.5E+04	3.0E+01	3.2E+08	1.2E+05	9.0E+02	5.0E-01	2.2E-02	2.1E-02	2.6E-02	1.6E-02	9.5E-04	8.3E-04	1.1E-03	1.3E-02
				0.0E+00	1.2E+05	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				1.3E+07	1.2E+05	2.5E+04	5.0E-01	2.2E-02	2.1E-02	2.6E-02	1.6E-02	9.5E-04	8.3E-04	1.1E-03	1.3E-02
5	BBAACJIBAAAAAA	4.0E+04	3.0E+01	4.7E+08	1.0E+05	9.0E+02	9.0E-01	3.1E-02	3.1E-02	3.7E-02	2.7E-02	1.2E-03	1.4E-03	1.2E-03	2.0E-02
				0.0E+00	1.0E+05	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.2E+07	1.0E+05	2.2E+04	1.0E-01	3.4E-03	3.5E-03	4.1E-03	3.1E-03	1.3E-04	1.5E-04	1.4E-04	2.2E-03
6	BBAAFJGKAACABA	4.0E+04	3.0E+01	3.0E+08	6.1E+04	9.0E+02	3.3E-01	2.0E-04	3.5E-03	9.0E-05	3.3E-06	2.6E-10	1.0E-10	1.0E-10	3.3E-08
				0.0E+00	6.2E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				1.3E+07	6.2E+04	2.5E+04	6.7E-01	6.0E-01	6.5E-01	9.8E-02	9.9E-04	1.2E-09	7.4E-05	1.2E-04	8.4E-04
7	BBABFJGKAAAABA	3.9E+04	3.0E+01	3.0E+08	6.1E+04	9.0E+02	5.3E-01	1.5E-02	1.3E-02	4.4E-03	1.6E-04	1.4E-04	6.6E-06	1.5E-05	2.4E-04
				0.0E+00	6.2E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				1.3E+07	6.2E+04	2.5E+04	4.7E-01	2.1E-01	2.3E-01	8.2E-02	2.4E-02	8.4E-05	1.0E-03	1.2E-03	1.5E-02
8	BBBAFJICAAAABA	3.9E+04	3.0E+01	4.4E+08	7.4E+04	9.0E+02	9.0E-01	6.5E-02	6.8E-02	9.6E-02	4.8E-02	4.6E-04	3.7E-03	5.6E-03	3.8E-02
				0.0E+00	7.5E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.2E+07	7.5E+04	2.2E+04	1.0E-01	7.2E-03	7.6E-03	1.1E-02	5.4E-03	5.1E-05	4.1E-04	6.3E-04	4.2E-03
9	BBAAFJGBAAAABA	4.3E+04	3.0E+01	4.4E+08	6.1E+04	9.0E+02	6.6E-01	1.6E-04	1.9E-03	6.3E-05	6.1E-06	9.8E-07	2.7E-07	7.3E-07	6.3E-06
				0.0E+00	6.2E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.2E+07	6.2E+04	2.2E+04	3.4E-01	3.0E-01	3.3E-01	5.0E-02	7.7E-04	6.6E-10	5.1E-05	7.9E-05	5.7E-04
10	BCBACJAKAAAABA	4.6E+04	3.0E+01	4.7E+08	7.8E+04	9.0E+02	9.9E-01	1.0E-01	1.1E-01	9.1E-02	8.2E-02	1.2E-02	7.7E-03	3.7E-02	8.[illegible]E-02
				0.0E+00	7.9E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.2E+07	7.9E+04	2.2E+04	7.8E-03	6.7E-02	8.5E-02	1.5E-01	2.1E-02	2.5E-09	9.8E-04	1.3E-03	2.2E-02
11	BCAACJIGAAABBA	5.3E+04	3.0E+01	3.2E+08	1.2E+05	9.0E+02	5.0E-01	3.0E-02	2.9E-02	2.7E-02	1.2E-02	6.2E-04	5.6E-04	7.5E-04	8.9E-03
				0.0E+00	1.2E+05	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				1.3E+07	1.2E+05	2.5E+04	[illegible].0E-01	3.0E-02	2.9E-02	2.7E-02	1.2E-02	6.2E-04	5.6E-04	7.5E-04	8.9E-03
12	BCBACJICAAAABA	5.7E+04	3.0E+01	4.7E+08	9.1E+04	9.0E+02	9.0E-01	5.4E-02	5.5E-02	4.2E-02	1.6E-02	3.9E-03	2.0E-03	2.6E-03	1.4E-02
				0.0E+00	9.2E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.2E+07	9.2E+04	2.2E+04	1.0E-01	6.0E-03	6.1E-03	4.6E-03	1.8E-03	4.3E-04	2.2E-04	2.9E-04	1.6E-03
13	BCAAFJIBAAABAA	4.4E+04	3.0E+01	4.4E+08	1.2E+05	9.0E+02	9.0E-01	6.3E-03	6.8E-03	5.0E-03	2.7E-03	4.0E-05	2.5E-04	1.1E-04	2.5E-03
				0.0E+00	1.2E+05	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.2E+07	1.2E+05	2.2E+04	1.0E-01	7.0E-04	7.6E-04	5.5E-04	3.0E-04	4.4E-06	2.8E-05	1.3E-05	2.8E-04
14	BBBACJGKAAAAAA	3.9E+04	3.0E+01	3.2E+08	6.1E+04	9.0E+02	7.0E-01	9.3E-03	1.0E-02	3.0E-03	7.5E-04	9.2E-04	2.8E-04	2.8E-04	9.9E-04
				0.0E+00	6.2E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				1.3E+07	6.2E+04	2.5E+04	3.0E-01	9.1E-02	9.1E-02	5.7E-02	4.3E-02	1.8E-04	4.0E-03	6.4E-03	3.5E-02
15	BCBACJB[illegible]AAAAAA	5.8E+04	3.0E+01	4.7E+08	7.6E+04	9.0E+02	7.5E-01	1.1E-02	1.4E-02	5.7E-03	5.2E-03	5.8E-03	1.3E-03	1.5E-03	5.7E-03
				0.0E+00	7.9E+04	0.0E+00	0.0E+00	0.0E+00	0.0E[illegible]	[illegible]E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.2E+07	7.9E+04	2.2E+04	2.5E-01	1.6E-01	1.5E-0[illegible]	9.[illegible]E-02	6.6E-02	5.1E-06	3.7E-03	5.8E-03	4.9E-02

Table 4.5-8.
The Mean Source Terms for the FCMR Latent Cancer Risk Dominant Bins
For the Fire Sequences.

ORDER	BIN	WARN (S)	ELEV (M)	POWER (W)	START (S)	DUR (S)	RELEASE FRACTIONS NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
1	CABBFJGKCAAAAB	4.1E+03	3.0E+01	4.9E+08	1.4E+04	9.0E+02	5.8E-01	2.9E-03	2.5E-03	9.0E-04	3.4E-05	1.5E-05	2.5E-06	8.8E-06	7.3E-05
				0.0E+00	1.5E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				1.3E+08	1.5E+04	2.5E+04	4.1E-01	1.3E-01	8.1E-02	5.5E-02	4.2E-02	1.6E-05	3.4E-03	5.4E-03	3.4E-02
2	CABBFJGKAAAAAB	4.4E+03	3.0E+01	4.9E+08	1.4E+04	9.0E+02	5.7E-01	2.4E-03	2.1E-03	7.4E-04	4.4E-05	1.3E-05	2.1E-06	7.2E-06	6.0E-05
				0.0E+00	1.5E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				1.3E+08	1.5E+04	2.5E+04	4.3E-01	1.3E-01	7.5E-02	5.1E-02	3.9E-02	1.3E-05	2.9E-03	4.6E-03	3.1E-02
3	CFAACGCKAAAAAB	2.0E+05	3.0E+01	2.9E+07	2.1E+05	2.6E+04	7.4E-01	1.8E-03	8.0E-04	1.2E-04	4.9E-05	1.1E-05	5.7E-06	3.5E-05	5.0E-05
				1.3E+07	2.4E+05	9.0E+02	2.6E-02	9.2E-03	1.2E-02	5.5E-03	3.2E-03	7.8E-03	1.4E-03	8.2E-04	3.8E-03
				1.7E+07	2.4E+05	2.2E+04	2.3E-01	1.4E-01	1.2E-01	6.9E-02	7.4E-02	1.1E-03	7.0E-03	5.3E-03	6.9E-02
4	CFAACGCKAAABAB	2.0E+05	3.0E+01	2.9E+07	2.1E+05	2.6E+04	3.0E-01	2.3E-06	5.2E-07	1.0E-08	2.6E-10	5.5E-12	2.2E-12	2.2E-12	6.2E-10
				1.3E+07	2.4E+05	9.0E+02	7.1E-02	1.1E-03	1.3E-03	3.0E-04	2.3E-05	4.4E-05	2.6E-05	2.6E-05	4.0E-05
				1.7E+07	2.4E+05	2.2E+04	6.3E-01	2.7E-01	2.3E-01	2.1E-01	2.2E-01	3.4E-03	2.0E-02	1.6E-02	2.0E-01
5	CFBACICKAAAAAB	2.0E+05	3.0E+01	2.9E+07	2.1E+05	2.6E+04	5.2E-01	2.0E-04	1.2E-04	3.9E-05	8.3E-07	3.0E-07	2.1E-08	2.8E-08	1.5E-06
				1.3E+07	2.4E+05	9.0E+02	4.9E-02	2.1E-03	2.5E-03	1.0E-03	6.1E-05	1.2E-04	7.8E-05	7.8E-05	1.3E-04
				1.7E+07	2.4E+05	2.2E+04	4.4E-01	1.9E-02	2.1E-02	1.5E-02	9.3E-03	8.0E-09	2.9E-04	4.5E-04	5.0E-03
6	CFBADGCKAAABAB	2.0E+05	3.0E+01	7.2E+06	2.1E+05	2.6E+04	4.0E-01	1.1E-03	2.6E-04	1.7E-05	2.6E-07	1.7E-08	6.8E-09	6.8E-03	5.8E-07
				1.3E+07	2.4E+05	9.0E+02	1.2E-01	2.0E-04	2.9E-04	1.9E-06	7.1E-08	0.0E+00	0.0E+00	0.0E+00	7.1E-08
				1.7E+07	2.4E+05	2.2E+04	4.8E-01	6.2E-01	8.2E-01	4.5E-01	2.3E-01	1.2E-07	7.3E-03	1.1E-02	1.2E-01
7	CFAACHCKAAAAAB	2.0E+05	3.0E+01	2.9E+07	2.1E+05	2.6E+04	7.2E-01	2.2E-03	1.7E-03	7.7E-04	3.1E-04	5.3E-05	1.9E-05	5.0E-05	3.1E-04
				1.3E+07	2.4E+05	9.0E+02	2.8E-02	1.3E-02	1.8E-02	7.6E-03	3.0E-03	3.2E-03	7.1E-04	7.1E-04	3.8E-03
				1.7E+07	2.4E+05	2.2E+04	2.5E-01	1.6E-01	1.8E-01	7.0E-02	1.5E-02	3.6E-06	9.0E-04	5.2E-04	1.2E-02
8	CABBFJGBAAAAAB	4.5E+03	3.0E+01	7.3E+08	1.4E+04	9.0E+02	5.9E-01	1.6E-03	1.7E-03	4.9E-04	4.3E-05	1.3E-05	2.1E-06	7.6E-06	5.5E-05
				0.0E+00	1.5E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.0E+08	1.5E+04	2.2E+04	4.1E-01	1.3E-01	7.5E-02	5.6E-02	4.2E-02	1.4E-05	3.2E-03	5.1E-03	3.4E-02
9	CFAAEGEKAAABBB	2.0E+05	3.0E+01	7.2E+06	2.1E+05	2.6E+04	9.6E-01	2.6E-05	1.6E-05	3.8E-06	1.8E-07	1.0E-07	9.6E-09	2.7E-08	2.3E-07
				1.3E+07	2.4E+05	9.0E+02	2.1E-03	6.9E-04	3.0E-03	5.3E-04	5.1E-05	9.7E-05	6.6E-05	6.4E-05	8.7E-05
				1.7E+07	2.4E+05	2.5E+04	3.7E-02	5.1E-02	1.9E-02	4.3E-02	6.5E-02	5.1E-08	2.9E-03	5.1E-03	3.2E-02
10	CABBFJGCAAAAAB	4.5E+03	3.0E+01	7.3E+08	1.4E+04	9.0E+02	1.0E+00	1.1E-03	1.2E-03	5.9E-04	2.3E-04	3.0E-05	9.5E-06	3.8E-05	2.4E-04
				0.0E+00	1.5E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.0E+08	1.5E+04	2.2E+04	0.0E+00	1.9E-01	2.2E-02	2.0E-02	2.5E-02	2.1E-05	3.1E-03	4.9E-03	2.1E-02
11	CAABFJICCAAAAB	4.0E+03	3.0E+01	7.3E+08	2.7E+04	9.0E+02	9.0E-01	1.1E-01	1.1E-03	7.1E-04	1.7E-04	1.2E-08	6.8E-06	6.5E-06	1.2E 04
				0.0E+00	2.8E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.0E+08	2.8E+04	2.2E+04	1.0E-01	1.2E-02	1.3E-04	7.9E-05	1.9E-05	1.4E-09	7.6E-07	7.2E-07	1.4E-05
12	CFAACCICAAAAAB	2.0E+05	3.0E+01	2.9E+07	2.1E+05	2.6E+04	8.9E-01	1.2E-02	8.7E-03	1.8E-03	6.8E-05	4.4E-05	1.3E-06	1.9E-06	9.7E-05
				1.3E+07	2.4E+05	9.0E+02	1.1E-02	7.6E-03	1.1E-02	8.6E-03	5.3E-03	5.2E-03	9.1E-04	9.1E-04	6.7E-03
				1.7E+07	2.4E+05	2.2E+04	9.8E-02	9.8E-02	1.0E-01	7.8E-02	2.4E-02	1.2E-06	6.5E-04	1.1E-03	1.7E-02
13	CFAACHGKAAAAAB	2.0E+05	3.0E+01	2.9E+07	2.1E+05	2.6E+04	8.0E-01	3.6E-03	2.6E-03	8.3E-04	2.4E-05	2.2E-05	1.2E-06	2.8E-06	3.9E-05
				1.3E+07	2.4E+05	9.0E+02	2.0E-02	2.8E-03	4.0E-03	7.7E-04	1.1E-03	1.3E-03	2.0E-04	2.0E-04	1.3E-03
				1.7E+07	2.4E+05	2.5E+04	1.8E-01	3.9E-02	3.9E-02	3.1E-02	1.7E-02	1.0E-04	1.4E-03	4.0E-04	1.5E-02
14	CFAAEGEKAAAABB	2.0E+05	3.0E+01	7.2E+06	2.1E+05	2.6E+04	9.6E-01	2.6E-05	1.6E-05	3.8E-06	1.8E-07	1.0E-07	9.6E-09	2.7E-08	2.3E-07
				1.3E+07	2.4E+05	9.0E+02	2.1E-03	6.9E-04	3.0E-03	5.3E-04	5.1E-05	9.7E-05	6.6E-05	6.4E-05	8.7E-05
				1.7E+07	2.4E+05	2.5E+04	3.7E-02	5.1E-02	1.9E-02	4.3E-02	6.5E-02	5.1E-08	2.9E-03	5.1E-03	3.2E-02
15	CFAABHDKAAAAAB	2.0E+05	3.0E+01	2.9E+07	2.1E+05	2.6E+04	5.5E-01	2.7E-04	2.3E-04	1.4E-04	6.1E-05	9.7E-06	3.7E-06	9.9E-06	6.1E-05
				1.3E+07	2.4E+05	9.0E+02	1.8E-01	4.1E-03	5.4E-03	3.8E-03	9.0E-04	3.2E-03	2.1E-03	2.0E-03	1.6E-03
				1.7E+07	2.4E+05	2.2E+04	2.7E-01	4.3E-02	2.9E-02	2.2E-02	1.2E-02	2.0E-08	3.1E-04	5.6E-04	6.6E-03

Table 4.5-9.
The Mean Source Terms for the FCMR Latent Cancer Risk Dominant Bins
For the Flood Sequences.

ORDER	BIN	WARN (S)	ELEV (M)	POWER (W)	START (S)	DUR (S)	RELEASE FRACTIONS NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
1	DABDFJIFABAAAB	4.7E+03	3.0E+01	7.3E+08	2.2E+04	9.0E+02	9.0E-01	7.8E-02	8.6E-03	1.1E-02	6.5E-03	1.0E-04	4.3E-04	8.7E-04	4.7E-03
				0.0E+00	2.3E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.0E+08	2.3E+04	2.2E+04	1.0E-01	8.6E-03	9.8E-04	1.2E-03	7.3E-04	1.1E-05	4.8E-05	9.7E-05	5.3E-04
2	DABDFJGKABAAAB	4.7E+03	3.0E+01	4.9E+08	1.4E+04	9.0E+02	6.8E-01	4.0E-03	3.5E-03	1.7E-03	8.8E-04	1.3E-04	7.6E-05	3.8E-04	9.0E-0[illegible]
				0.0E+00	1.5E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				1.2E+08	1.5E+04	2.5E+04	3.2E-01	1.1E-01	2.4E-02	1.6E-02	1.2E-02	5.1E-05	8.2E-04	1.5E-03	8.2E-03
3	DAADFJIFABAAAB	4.6E+03	3.0E+01	7.3E+08	2.2E+04	9.0E+02	9.0E-01	9.1E-02	9.7E-03	9.8E-03	4.8E-03	5.5E-05	2.3E-04	4.6E-04	3.1E-03
				0.0E+00	2.3E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.0E+08	2.3E+04	2.2E+04	1.0E-01	1.0E-02	1.1E-03	1.1E-03	5.4E-04	6.1E-06	2.6E-05	5.1E-05	3.5E-04
4	DAABFJIFAAAAAB	4.6E+03	3.0E+01	7.3E+08	2.2E+04	9.0E+02	9.0E-01	5.8E-02	2.1E-02	3.0E-02	2.0E-02	2.3E-04	1.1E-03	1.1E-03	1.5E-02
				0.0E+00	2.3E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.0E+08	2.3E+04	2.2E+04	1.0E-01	6.5E-03	2.3E-03	3.3E-03	2.2E-03	2.6E-05	1.2E-04	1.2E-04	1.7E-03
5	DABDFJGKABAABB	4.8E+03	3.0E+01	4.9E+08	1.4E+04	9.0E+02	5.8E-01	3.3E-04	2.1E-03	1.3E-04	3.7E-05	4.5E-06	2.7E-06	1.6E-05	3.7E-05
				0.0E+00	1.5E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				1.3E+08	1.5E+04	2.5E+04	4.2E-01	2.0E-01	1.1E-01	3.7E-02	5.1E-03	7.2E-10	3.8E-04	4.5E-04	1.9E-03
6	DAAEFJIFACACAB	4.6E+03	3.0E+01	7.3E+08	2.2E+04	9.0E+02	8.6E-01	3.6E-02	7.7E-04	7.7E-04	2.6E-04	4.8E-05	1.8E-05	6.3E-05	2.6E-04
				0.0E+00	2.3E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.0E+08	2.3E+04	7.2E+03	7.3E-02	4.0E-03	8.5E-05	8.6E-05	2.9E-05	5.3E-06	2.0E-06	7.7E-06	2.9E-05
7	DAADFJGKABAAAB	4.6E+03	3.0E+01	4.9E+08	1.4E+04	9.0E+02	6.7E-01	2.9E-03	2.7E-03	1.1E-03	2.3E-04	5.4E-05	1.4E-05	4.1E-05	2.5E-04
				0.0E+00	1.5E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				1.3E+08	1.5E+04	2.5E+04	3.3E-01	1.1E-01	2.3E-02	1.4E-02	6.6E-03	1.9E-05	2.8E-04	5.0E-04	3.8E-03
8	DABDFJGBABAAAB	4.[illegible]E+03	3.0E+01	7.3E+08	1.4E+04	9.0E+02	6.9E-01	4.2E-03	3.8E-03	1.8E-03	9.9E-04	1.4E-04	8.5E-05	4.3E-04	1.0E-03
				0.0E+00	1.5E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.[illegible]	1.5E+04	2.2E+04	3.1E-01	1.1E-01	2.4E-02	1.7E-02	1.2E-02	5.7E-05	8.8E-04	1.7E-03	8.6E-03
9	DAABFJGKAAAAAB	4.7E+03	3.0E+01	[illegible]	1.4E+04	9.0E+02	7.0E-01	4.0E-03	3.5E-03	1.2E-03	1.7E-04	4.6E-05	1.0E-05	2.9E-05	1.9E-04
				0.0E+00	1.5E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				1.2E+08	1.5E+04	2.5E+04	3.0E-01	9.6E-02	6.5E-02	4.6E-02	3.2E-02	7.2E-0[illegible]	1.7E-03	1.5E-03	2.3E-02
10	DABEFJIFACACAB	4.7E+03	3.0E+01	7.3E+08	[illegible].2E+04	9.0E+02	6.4E-01	3.8E-02	7.4E-04	7.2E-04	2.8E-04	4.4E-05	2.0E-05	9.5E-05	2.9E-04
				0.0E+00	2.3E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.0E+08	2.3E+04	7.2E+03	7.1E-02	4.2E-03	8.2E-05	6.0E-05	3.1E-05	4.9E-06	2.2E-06	1.1E-05	3.2E-05
11	DABDFJGCABAAAB	4.5E+03	3.0E+01	7.3E+08	1.4E+04	9.0E+02	8.2E-01	2.5E-03	2.2E-03	6.8E-04	8.4E-05	2.0E-05	5.2E-06	2.9E-05	9.7E-05
				0.0E+00	1.5E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.0E+08	1.5E+04	2.2E+04	1.8E-01	7.5E-02	2.1E-02	1.4E-02	6.2E-03	2.1E-05	3.1E-04	5.1E-04	3.3E-03
12	DABBFJIFAAAAAB	4.8E+03	3.0E+01	7.3E+08	2.2E+04	9.0E+02	9.0E-01	5.4E-02	2.2E-02	3.0E-02	2.0E-02	2.5E-04	1.8E-03	2.9E-03	1.7E-02
				0.0E+00	2.3E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.0E+08	2.3E+04	2.2E+04	1.0E-01	6.0E-03	2.5E-03	3.4E-03	2.3E-03	2.8E-05	2.0E-04	3.2E-04	1.8E-03
13	DABBFJGKAAAAAB	4.6E+03	3.0E+01	4.9E+08	1.4E+04	9.0E+02	7.3E-01	2.8E-03	2.7E-03	1.7E-03	1.2E-03	1.7E-04	1.1E-04	5.4E-04	1.2E-03
				0.0E+00	1.5E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				1.3E+08	1.5E+04	2.5E+04	2.7E-01	1.0E-01	6.6E-02	6.0E-02	5.0E-02	5.1E-04	5.1E-03	7.9E-03	4.1E-02
14	DAADAJAKABAABB	4.6E+03	3.0E+01	7.5E+08	1.4E+04	9.0E+02	8.3E-01	1.3E-01	1.4E-01	7.4E-02	4.4E-02	4.3E-02	9.2E-03	1.0E-02	5.0E-02
				0.0E+00	1.5E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.0E+08	1.5E+04	2.2E+04	1.7E-01	1.2E-01	6.4E-02	3.8E-02	2.0E-02	5.3E-05	9.4E-04	1.7E-03	1.2E-02
15	DABDFJIFACAAAB	4.6E+03	3.0E+01	7.3E+08	2.2E+04	9.0E+02	6.5E-01	4.6E-02	8.9E-04	4.7E-04	9.4E-05	3.4E-06	4.0E-07	1.4E-06	1.3E-05
				0.0E+00	2.3E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.0E+08	2.3E+04	7.2E+03	7.3E-02	5.1E-03	9.9E-05	5.2E-05	1.0E-06	3.8E-07	4.4E-08	1.5E-07	1.5E-06

Table 4.5-10.
The Mean Source Terms for the FCMR Latent Cancer Risk Dominant Bins
For the ATWS Sequences.

ORDER	BIN	WARN (S)	ELEV (M)	POWER (W)	START (S)	DUR (S)	RELEASE FRACTIONS NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
1	EABDFEIKABAAAB	2.6E+03	3.0E+01	2.0E+07	3.5E+03	9.7E+03	7.1E-01	7.9E-03	6.1E-03	3.3E-03	1.7E-03	2.4E-04	1.3E-04	6.3E-04	1.7E-03
				3.5E+07	1.3E+04	9.0E+02	1.5E-04	6.0E-05	1.1E-03	2.7E-05	1.0E-06	0.0E+00	0.0E+00	0.0E+00	1.0E-06
				4.7E+07	1.4E+04	2.2E+04	2.9E-01	1.2E-01	3.7E-02	2.9E-02	1.9E-02	1.4E-04	1.4E-03	2.5E-03	1.4E-02
2	EABDFEICABAAAB	2.3E+03	3.0E+01	2.0E+07	3.5E+03	9.7E+03	8.0E-01	3.8E-03	2.9E-03	1.1E-03	1.9E-04	3.1E-05	7.5E-06	2.9E-05	2.1E-04
				3.5E+07	1.3E+04	9.0E+02	1.5E-04	7.2E-05	1.3E-03	3.4E-05	1.2E-06	0.0E+00	0.0E+00	0.0E+00	1.2E-06
				4.7E+07	1.4E+04	2.2E+04	2.0E-01	1.2E-01	3.0E-02	2.3E-02	1.7E-02	5.7E-06	1.2E-03	2.2E-03	1.3E-02
3	EABDEEIKABAAAB	2.5E+03	3.0E+01	2.0E+07	3.5E+03	9.7E+03	6.8E-01	4.8E-03	3.5E-03	.1E-03	5.5E-04	8.2E-05	2.5E-05	1.0E-04	5.8E-04
				3.5E+07	1.3E+04	9.0E+02	1.6E-02	5.2E-03	7.0E-03	2.3E-03	1.7E-03	1.8E-03	3.3E-04	3.6E-04	2.0E-03
				4.7E+07	1.4E+04	2.2E+04	3.0E-01	1.3E-01	4.8E-02	2.7E-02	1.7E-02	1.2E-04	1.0E-03	2.0E-03	1.2E-02
4	EABEFEIKACACAB	2.5E+03	3.0E+01	2.0E+07	3.5E+03	9.7E+03	7.0E-01	6.7E-03	5.0E-03	3.0E-03	1.5E-03	2.1E-04	1.1E-04	5.6E-04	1.5E-03
				3.5E+07	1.3E+04	9.0E+02	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				4.6E+07	1.4E+04	7.2E+03	0.0E+00	3.8E-02	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
5	EAAACHCKAAAAAB	2.7E+03	3.0E+01	4.2E+07	6.2E+03	9.7E+03	7.2E-01	1.9E-03	1.4E-03	7.9E-04	3.2E-04	5.5E-05	2.0E-05	5.2E-05	3.2E-04
				3.5E+07	1.6E+04	9.0E+02	2.8E-02	1.2E-02	1.7E-02	7.6E-03	3.0E-03	3.2E-03	7.1E-04	7.1E-04	3.8E-03
				4.7E+07	1.7E+04	2.2E+04	2.5E-01	1.5E-01	1.8E-01	6.7E-02	1.4E-02	3.4E-06	8.6E-04	4.9E-04	1.1E-02
6	EAADFEIKABAAAB	2.6E+03	3.0E+01	2.0E+07	3.5E+03	9.7E+03	7.5E-01	8.7E-03	7.3E-03	4.0E-03	9.5E-04	2.1E-04	6.6E-05	2.3E-04	1.0E-03
				3.5E+07	1.3E+04	9.0E+02	1.5E-04	6.2E-05	1.2E-03	2.8E-05	1.0E-06	0.0E+00	0.0E+00	0.0E+00	1.0E-06
				4.7E+07	1.4E+04	2.2E+04	2.5E-01	1.3E-01	3.1E-02	2.2E-02	1.9E-02	5.8E-05	1.2E-03	2.4E-03	1.3E-02
7	EABDFEGKABAAAB	2.7E+03	3.0E+01	2.0E+07	3.5E+03	9.7E+03	6.2E-01	3.4E-03	2.3E-03	1.3E-03	4.7E-04	7.8E-05	4.1E-05	2.0E-04	4.9E-04
				3.5E+07	1.3E+04	9.0E+02	1.5E-04	9.7E-05	1.8E-03	4.5E-05	1.7E-06	0.0E+00	0.0E+00	0.0E+00	1.7E-06
				4.7E+07	1.4E+04	2.2E+04	3.8E-01	7.1E-02	5.5E-02	3.6E-02	2.2E-02	6.4E-04	1.0E-03	1.9E-03	1.4E-02
8	EABDFEIKACACAB	2.8E+03	3.0E+01	2.0E+07	3.5E+03	9.7E+03	6.6E-01	1.0E-02	9.5E-03	8.4E-03	6.5E-03	9.1E-04	5.9E-04	2.9E-03	6.6E-03
				3.5E+07	1.3E+04	9.0E+02	1.5E-04	1.7E-05	3.1E-04	7.6E-06	2.8E-07	0.0E+00	0.0E+00	0.0E+00	2.8E-07
				4.7E+07	1.4E+04	7.2E+03	0.0E+00	6.4E-02	3.5E-04	9.0E-05	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
9	EABDEEIKACACAB	2.4E+03	3.0E+01	2.0E+07	3.5E+03	9.7E+03	7.0E-01	8.0E-03	5.1E-03	2.7E-03	6.6E-04	1.0E-04	2.9E-05	1.1E-04	7.0E-04
				3.5E+07	1.3E+04	9.0E+02	1.5E-02	1.9E-03	2.5E-03	1.0E-03	6.6E-04	6.7E-04	1.4E-04	1.6E-04	7.5E-04
				4.7E+07	1.4E+04	7.2E+03	0.0E+00	5.0E-02	4.2E-04	1.3E-04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
10	EABDFEIBABAAAB	2.6E+03	3.0E+01	2.0E+07	3.5E+03	9.7E+03	7.4E-01	9.7E-03	7.4E-03	3.9E-03	2.1E-03	2.9E-04	1.7E-04	8.1E-04	2.2E-03
				3.5E+07	1.3E+04	9.0E+02	1.5E-04	6.5E-05	1.2E-03	2.9E-05	1.1E-06	0.0E+00	0.0E+00	0.0E+00	1.1E-06
				4.7E+07	1.4E+04	2.2E+04	2.8E-01	1.1E-01	3.8E-02	2.9E-02	1.6E-02	1.6E-04	8.3E-04	1.5E-03	1.0E-02
11	EABDEEIKABABAB	2.6E+03	3.0E+01	2.0E+07	3.5E+03	9.7E+03	7.2E-01	8.4E-03	6.2E-03	3.2E-03	4.1E-04	8.6E-05	1.8E-05	6.6E-05	4.8E-04
				3.5E+07	1.3E+0[illegible]	9.0E+02	1.4E-02	5.5E-04	9.1E-04	3.4E-04	4.2E-04	4.5E-04	6.7E-05	6.6E-05	4.8E-04
				4.7E+07	1.4E+04	2.2E+04	2.7E-01	.6E-01	4.3E-02	2.3E-02	1.4E-02	3.2E-06	7.9E-04	1.5E-03	1.0E-02
12	EABDEEBKABAAAB	2.5E+03	3.0E+01	2.0E+07	3.5E+03	9.7E+03	6.3E-01	5.3E-03	3.0E-03	1.2E-03	1.0E-04	2.7E-05	7.9E-06	3.6E-05	1.3E-04
				3.5E+07	1.3E+04	9.0E+02	1.9E-02	8.7E-03	1.1E-02	2.8E-03	1.7E-03	1.9E-03	3.8E-04	3.8E-04	2.0E-03
				4.7E+07	1.4E+04	2.2E+04	3.5E-01	1.8E-01	1.1E-01	6.7E-02	4.2E-02	1.2E-03	2.6E-03	5.1E-03	3.1E-02
13	EAAEFEIKACACAB	2.5E+03	3.0E+01	2.0E+07	3.5E+03	9.7E+03	7.5E-01	8.2E-03	6.0E-03	2.7E-03	5.6E-04	1.3E-04	3.8E-05	1.4E-04	6.0E-04
				3.5E+07	1.3E+04	9.0E+02	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				4.7E+07	1.4E+04	7.2E+03	0.0E+00	3.8E-02	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
14	EAAACHCBAAAAAB	2.6E+03	3.0E+01	4.2E+07	6.2E+03	9.7E+03	8.6E-01	2.8E-03	2.2E 3	1.4E-03	5.5E-04	9.5E-05	3.4E-05	9.0E-05	5.5E-04
				3.5E+07	1.6E+04	9.0E+02	1.4E-02	1.1E-02	1.8E-02	1.3E-02	5.3E-03	5.5E-03	1.2E-03	1.2E-03	6.6E-03
				4.7E+07	1.7E+04	2.2E+04	1.3E-01	1.3E-01	1.6E-01	9.0E-02	2.4E-02	5.9E-06	1.5E-03	8.0E-04	1.9E-02
15	EAADFEIBABAAAB	2.4E+03	3.0E+01	2.0E+07	3.5E+03	9.7E+03	7.2E-01	1.2E-02	9.1E-03	4.7E-03	1.1E-03	2.5E-04	8.0E-05	2.9E-04	1.2E-03
				3.5E+07	1.3E+04	9.0E+02	1.5E-04	6.2E-05	1.2E-03	2.8E-05	1.1E-06	0.0E+00	0.0E+00	0.0E+00	1.1E-06
				4.7E+07	1.4E+04	2.2E+04	2.8E-01	1.6E-01	3.1E-02	2.2E-02	1.9E-02	8.2E-05	1.4E-03	2.7E-03	1.5E-02

Table 4.5-11.
The Mean Source Terms for the FCMR Latent Cancer Risk Dominant Bins
For LOCA Sequences.

ORDER	BIN	WARN (S)	ELEV (M)	POWER (W)	START (S)	DUR (S)	RELEASE FRACTIONS NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
1	FFABFBICAABAAB	1.9E+05	3.0E+01	7.2E+06	2.1E+05	2.6E+04	4.9E-01	4.5E-03	3.1E-03	8.4E-04	3.0E-05	1.3E-05	3.9E-07	5.4E-07	4.8E-05
				1.3E+07	2.4E+05	9.0E+02	1.5E-04	3.6E-05	6.6E-04	1.5E-05	5.5E-07	0.0E+00	0.0E+00	0.0E+00	5.5E-07
				1.7E+07	2.4E+05	2.2E+04	5.1E-01	9.0E-02	1.0E-01	8.0E-02	7.6E-02	9.2E-06	4.2E-03	2.4E-03	5.9E-02
2	FFABEHBKAABAAB	2.0E+05	3.0E+01	7.2E+06	2.1E+05	2.6E+04	8.1E-01	8.2E-03	4.7E-03	1.1E-03	7.1E-05	3.5E-05	2.5E-06	5.8E-06	8.8E-05
				1.3E+07	2.4E+05	9.0E+02	9.6E-03	7.8E-03	1.1E-02	1.4E-03	1.9E-03	2.3E-03	3.9E-04	3.8E-04	2.2E-03
				1.7E+07	2.4E+05	2.2E+04	1.8E-01	1.7E-01	1.9E-01	1.6E-01	9.4E-02	1.7E-06	3.2E-07	3.6E-03	6.9E-02
3	FFBBFGIKAABAAB	2.0E+05	3.0E+01	7.2E+06	2.1E+05	2.6E+04	9.2E-01	9.9E-03	5.9E-03	2.5E-03	1.8E-04	6.1E-05	8.4E-06	2.9E-05	2.4E-04
				1.3E+07	2.4E+05	9.0E+02	1.5E-04	5.0E-05	9.3E-04	2.1E-05	7.9E-07	0.0E+00	0.0E+00	0.0E+00	7.9E-07
				1.7E+07	2.4E+05	2.5E+04	7.5E-02	6.0E-02	6.7E-02	9.5E-02	6.9E-02	5.3E-05	6.3E-03	1.0E-02	6.0E-02
4	FFBBEHGKAABAAB	2.0E+05	3.0E+01	7.2E+06	2.1E+05	2.6E+04	8.7E-01	7.0E-03	6.5E-03	5.4E-03	4.9E-03	6.6E-04	4.4E-04	2.2E-03	4.9E-03
				1.3E+07	2.4E+05	9.0E+02	6.5E-03	7.3E-04	2.0E-03	5.9E-04	3.7E-04	5.9E-04	2.1E-04	1.5E-04	4.5E-04
				1.7E+07	2.4E+05	2.5E+04	1.2E-01	1.9E-02	2.0E-02	3.9E-02	8.7E-02	6.1E-04	1.4E-02	2.3E-02	7.9E-02
5	FFABEHGKAABBAB	2.0E+05	3.0E+01	7.2E+06	2.1E+05	2.6E+04	6.3E-01	3.0E-03	2.4E-03	1.3E-03	4.9E-04	8.4E-05	3.0E-05	7.8E-05	5.0E-04
				1.3E+07	2.4E+05	9.0E+02	1.9E-02	6.7E-04	8.1E-04	1.1E-04	7.9E-05	7.9E-05	1.3E-05	1.3E-05	9.1E-05
				1.7E+07	2.4E+05	2.5E+04	3.5E-01	6.9E-02	7.9E-02	3.4E-02	1.7E-02	5.5E-05	7.6E-04	8.5E-04	1.3E-02
6	FFBBFHIKAABAAB	1.9E+05	3.0E+01	7.2E+06	2.1E+05	2.6E+04	8.4E-01	4.3E-03	2.7E-03	1.5E-03	4.6E-04	6.1E-05	2.7E-05	1.2E-04	4.7E-04
				1.3E+07	2.4E+05	9.0E+02	1.5E-04	3.1E-05	5.9E-04	1.3E-05	4.6E-07	0.0E+00	0.0E+00	0.0E+00	4.6E-07
				1.7E+07	2.4E+05	2.5E+04	1.6E-01	5.5E-02	4.4E-02	4.4E-02	6.2E-02	3.9E-04	1.0E-02	1.5E-02	5.5E-02

Table 4.5-12.
The Mean Source Terms for the FCMR Latent Cancer Risk Dominant Bins
For Transient Sequences.

ORDER	BIN	WARN (S)	ELEV (M)	POWER (W)	START (S)	DUR (S)	RELEASE FRACTIONS MG	I	Cs	Te	Sr	Ru	La	Ce	Ba
1	GABDFJIFABAAAA	4.7E+03	3.0E+01	7.3E+08	2.2E+04	9.0E+02	9.0E-01	7.9E-02	8.4E-03	1.0E-02	6.3E-03	9.8E-05	4.2E-04	8.5E-04	4.6E-03
				0.0E+00	2.3E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.0E+08	2.3E+04	2.2E+04	1.0E-01	8.8E-03	9.3E-04	1.1E-03	7.0E-04	1.1E-05	4.7E-05	9.4E-05	5.1E-04
2	GABDFJIFCBAAAA	4.6E+03	3.0E+01	7.3E+08	2.2E+04	9.0E+02	9.0E-01	7.8E-02	8.4E-03	1.0E-02	6.3E-03	9.8E-05	4.2E-04	8.5E-04	4.6E-03
				0.0E+00	2.3E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.0E+08	2.3E+04	2.2E+04	1.0E-01	8.7E-03	9.3E-04	1.1E-03	7.0E-04	1.1E-05	4.7E-05	9.4E-05	5.1E-04
3	GBBDEJBKBBAAAA	4.0E+04	3.0E+01	4.4E+08	6.1E+04	9.0E+02	7.4E-01	1.0E-02	1.2E-02	3.0E-03	2.2E-03	2.2E-03	4.0E-04	4.3E-04	2.5E-03
				0.0E+00	6.2E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.2E+07	6.2E+04	2.2E+04	2.5E-01	1.3E-01	1.3E-01	7.1E-02	4.0E-02	1.3E-03	2.7E-03	5.3E-03	3.0E-02
4	GAADFJIFABAAAA	4.6E+03	3.0E+01	7.3E+08	2.2E+04	9.0E+02	9.0E-01	8.5E-02	8.0E-03	9.1E-03	4.6E-03	8.4E-05	2.5E-04	4.8E-04	3.0E-03
				0.0E+00	2.3E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.0E+08	2.3E+04	2.2E+04	1.0E-01	9.4E-03	8.9E-04	1.0E-03	5.1E-04	9.3E-06	2.8E-05	5.4E-05	3.4E-04
5	GABDFJGKABAAAA	4.7E+03	3.0E+01	4.9E+08	1.4E+04	9.0E+02	6.7E-01	3.9E-03	3.5E-03	1.6E-03	8.3E-04	1.2E-04	7.1E-05	3.6E-04	8.4E-04
				0.0E+00	1.5E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				1.2E+08	1.5E+04	2.5E+04	3.3E-01	1.2E-01	2.3E-02	1.6E-02	1.1E-02	5.9E-05	7.9E-04	1.5E-03	7.9E-03
6	GAADFJIFCBAAAA	4.6E+03	3.0E+01	7.3E+08	2.2E+04	9.0E+02	9.0E-01	8.5E-02	8.0E-03	9.1E-03	4.6E-03	8.4E-05	2.5E-04	4.8E-04	3.0E-03
				0.0E+00	2.3E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.0E+08	2.3E+04	2.2E+04	1.0E-01	9.4E-03	8.9E-04	1.0E-03	5.1E-04	9.3E-06	2.8E-05	5.4E-05	3.4E-04
7	GBADFJIFDBAAAA	4.0E+04	3.0E+01	4.4E+08	6.8E+04	9.0E+02	9.0E-01	1.7E-02	5.9E-03	6.0E-03	[illegible]	6.4E-05	2.3E-04	4.8E-04	2.6E-03
				0.0E+00	6.9E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	[illegible]	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.2E+07	6.9E+04	2.2E+04	1.0E-01	1.9E-03	6.6E-04	6.6E-04	[illegible]	7.2E-06	2.6E-05	5.3E-05	2.9E-04
8	GAABFJGCAAAAAA	4.6E+03	3.0E+01	7.3E+08	1.4E+04	9.0E+02	7.7E-01	3.9E-03	3.3E-03	1.1E-03	[illegible]	5.0E-05	1.3E-05	6.2E-05	2.1E-04
				0.0E+00	1.5E+04	[illegible]	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.0E+08	1.5E+04	[illegible]	[illegible]	1.0E-01	6.8E-02	5.6E-02	4.5E-02	1.2E-04	2.3E-03	1.8E-03	3.3E-02
9	GAABFJGKAAAAAA	4.7E+03	3.0E+01	4.9E+08	1.4E+04	[illegible]	[illegible]	4.0E-03	3.4E-03	1.2E-03	1.7E-04	4.7E-05	1.0E-05	2.9E-05	1.9E-04
				0.0E+00	1.5E+04	0.0E+00	[illegible]	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				1.2E+08	1.5E+04	2.5E+04	3.1E-01	9.7E-02	6.5E-02	4.6E-02	3.4E-02	1.4E-04	1.9E-03	1.7E-03	2.5E-02
10	GABBFJGKAAAAAA	4.6E+03	3.0E+01	4.9E+08	1.4E+04	9.0E+02	7.3E-01	2.7E-03	2.7E-03	1.6E-03	1.1E-03	1.6E-04	9.9E-05	5.0E-04	1.1E-03
				0.0E+00	1.5E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				1.3E+08	1.5E+04	2.5E+04	2.7E-01	1.0E-01	6.4E-02	5.9E-02	4.9E-02	4.7E-04	4.9E-03	7.6E-03	4.0E-02
11	GAADAJAKABAABA	4.6E+03	3.0E+01	7.5E+08	1.4E+04	9.0E+02	8.2E-01	1.3E-01	1.5E-01	5.8E-02	3.1E-02	3.3E-02	7.0E-03	7.5E-03	3.6E-02
				0.0E+00	1.5E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.0E+08	1.5E+04	2.2E+04	1.8E-01	1.3E-01	6.2E-02	3.8E-02	1.8E-02	3.2E-05	7.6E-04	1.4E-03	1.1E-02
12	GABBFJIBAAAAAA	4.7E+03	3.0E+01	7.3E+08	6.5E+04	9.0E+02	9.0E-01	6.2E-02	2.5E-02	3.5E-02	2.5E-02	3.2E-04	2.3E-03	3.6E-03	2.0E-02
				0.0E+00	6.6E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.0E+08	6.6E+04	2.2E+04	1.0E-01	6.8E-03	2.7E-03	3.9E-03	2.7E-03	3.6E-05	2.5E-04	4.0E-04	2.3E-03
13	GCAACJCKAAAABA	5.0E+04	3.0E+01	4.7E+08	7.8E+04	9.0E+02	7.2E-01	2.8E-02	3.5E-02	2.1E-02	2.5E-03	3.8E-03	1.8E-03	1.8E-03	4.1E-03
				0.0E+00	7.9E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.2E+07	7.9E+04	2.2E+04	2.8E-01	3.6E-01	3.3E-01	2.1E-01	1.6E-02	2.0E-09	7.2E-04	9.5E-04	9.4E-03
14	GFAAFCGCAAAAAB	1.9E+05	3.0E+01	7.2E+06	2.1E+05	2.6E+04	1.9E-01	1.6E-05	3.4E-06	1.3E-08	2.4E-10	2.5E-12	9.8E-13	9.8E-13	8.4E-10
				1.3E+07	2.4E+05	9.0E+02	1.5E-04	1.1E-04	2.0E-03	4.6E-05	1.7E-06	0.0E+00	0.0E+00	0.0E+00	1.7E-06
				1.7E+07	2.4E+05	2.2E+04	8.1E-01	4.0E-01	4.1E-01	3.5E-01	3.5E-01	6.4E-04	2.8E-02	7.7E-03	3.0E-01
15	GABBFJIGAAAABA	4.9E+03	3.0E+01	4.8E+08	6.5E+04	9.0E+02	5.0E-01	3.2E-02	1.9E-02	2.1E-02	1.3E-02	3.1E-04	1.2E-03	2.0E-03	1.2E-02
				0.0E+00	6.6E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				1.3E+08	6.6E+04	2.5E+04	5.0E-01	3.2E-02	1.9E-02	2.1E-02	1.3E-02	3.1E-04	1.2E-03	2.0E-03	1.2E-02

Table 4.5-13.
The Mean Source Terms for the FCMR Latent Cancer Risk Dominant Bins for the Transient-LOCA Sequences.

ORDER	BIN	WARN (S)	ELEV (M)	POWER (W)	START (S)	DUR (S)	RELEASE FRACTIONS NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
1	HDABFEICAAAAAB	8.8E+04	3.0E+01	9.4E+06	1.0E+05	1.5E+04	[illegible].8E-01	5.2E-03	3.5E-03	1.4E-03	2.2E-04	5.4E-05	1.2E-05	3.2E-05	2.4E-04
				1.7E+07	1.2E+05	9.0E+02	1.5E-04	9.0E-05	1.7E-03	3.8E-05	1.4E-06	0.0E+00	0.0E+00	0.0E+00	1.4E-06
				2.2E+07	1.2E+05	2.2E+04	2.1E-01	1.5E-01	1.5E-01	1.3E-01	1.1E-01	8.0E-[illegible]	6.3E-03	3.8E-03	8.6E-02
2	HDABFEIKAAAAAB	8.9E+04	3.0E+01	9.4E+06	1.0E+05	1.5E+04	7.3E-01	1.1E-02	7.8E-03	3.2E-03	6.3E-04	1.5E-04	4.2E-05	1.4E-04	6.7E-04
				1.7E+07	1.2E+05	9.0E+02	1.5E-04	7.1E-05	1.3E-03	3.0E-05	1.1E-06	0.0E+00	0.0E+00	0.0E+00	1.1E-06
				2.2E+07	1.2E+05	2.2E+04	2.7E-01	1.2E-01	1.3E-01	9.5E-02	7.5E-02	4.0E-04	4.7E-03	3.8E-03	5.8E-02
3	HDABEEIKAAAAAB	9.0E+04	3.0E+01	9.4E+06	1.0E+05	1.5E+04	7.3E-01	8.2E-03	5.6E-03	2.5E-03	7.6E-04	1.4E-04	6.0E-05	2.6E-04	8.0E-04
				1.7E+07	1.2E+05	9.0E+02	1.4E-02	5.6E-03	8.0E-03	3.3E-03	2.3E-03	2.6E-03	5.2E-04	5.5E-04	2.6E-03
				2.2E+07	1.2E+05	2.2E+04	2.5E-01	1.4E-01	1.5E-01	1.1E-01	7.4E-02	1.1E-03	4.0E-03	4.7E-03	5.5E-02
4	HDBBEEIKAAABAB	8.9E+04	3.0E+01	9.4E+06	1.0E+05	1.5E+04	6.9E-01	4.3E-03	3.1E-03	1.9E-03	7.7E-04	9.6E-05	3.4E-05	1.4E-04	7.8E-04
				1.7E+07	1.2E+05	9.0E+02	1.6E-02	1.1E-03	1.6E-03	6.2E-04	3.2E-04	3.6E-04	8.7E-05	8.[illegible]E-05	3.8E-04
				2.2E+07	1.2E+05	2.2E+04	2.9E-01	1.4E-01	1.4E-01	8.5E-02	8.0E-02	1.6E-04	8.1E-03	1.3E-02	6.6E-02
5	HDBBEEIKAAAAAB	8.9E+04	3.0E+01	9.4E+06	1.0E+05	1.5E+04	7.0E-01	4.7E-03	3.1E-03	1.7E-03	5.2E-04	6.9E-05	2.3E-05	9.4E-05	5.4E-04
				1.7E+07	1.2E+05	9.0E+02	1.5E-02	5.9E-03	8.0E-03	2.1E-03	1.4E-03	1.4E-03	2.9E-04	3.0E-04	1.6E-03
				2.2E+07	1.2E+05	2.2E+04	2.9E-01	1.5E-01	1.6E-01	9.3E-02	6.9E-02	1.4E-04	6.3E-03	1.0E-02	5.5E-02
6	HDBBFZIKAAAAAB	8.9E+04	3.0E+01	9.4E+06	1.0E+05	1.5E+04	7.1E-01	5.0E-03	4.0E-03	2.6E-03	1.6E-03	2.1E-04	1.2E-04	5.7E-04	1.6E-03
				1.7E+07	1.2E+05	9.0E+02	1.5E-04	7.3E-05	1.4E-03	3.1E-05	1.1E-06	0.0E+00	0.0E+00	0.0E+00	1.1E-06
				2.2E+07	1.2E+05	2.2E+04	2.9E-01	1.4E-01	1.4E-01	1.0E-01	7.5E-02	1.7E-04	7.0E-03	1.1E-02	6.2E-02
7	HDBBFEICAAAAAB	9.0E+04	3.0E+01	9.4E+06	1.0E+05	1.5E+04	6.9E-01	2.6E-03	1.6E-03	5.6E-04	4.9E-05	1.4E-05	2.1E-06	7.5E-06	6.1E-05
				1.7E+07	1.2E+05	9.0E+02	1.5E-04	6.0E-05	1.1E-03	2.5E-05	9.1E-07	0.0E+00	0.0E+00	0.0E+00	9.1E-07
				2.2E+07	1.2E+05	2.2E+04	3.0E-01	1.5E-01	1.5E-01	1.1E-01	6.6E-02	1.1E-03	4.2E-03	6.3E-03	4.9E-02
8	HDBBFEICAAABAB	9.0E+04	3.0E+01	9.4E+06	1.0E+05	1.5E+04	6.8E-01	5.4E-04	4.8E-04	2.5E-04	6.9E-05	1.1E-05	2.9E-06	1.1E-05	7.3E-05
				1.7E+07	1.2E+05	9.0E+02	1.5E-04	9.7E-06	1.8E-04	3.7E-06	1.4E-07	0.0E+00	0.0E+00	0.0E+00	1.4E-07
				2.2E+07	1.2E+05	2.2E+04	3.2E-01	5.2E-02	4.2E-02	3.0E-02	1.9E-02	2.9E-03	3.0E-03	3.8E-03	1.8E-02
9	HDBBFEIKAAABAB	9.0E+04	3.0E+01	9.4E+06	1.0E+05	1.5E+04	7.9E-01	1.8E-02	1.7E-02	1.3E-02	1.0E-02	1.4E-03	9.0E-04	4.5E-03	1.0E-02
				1.7E+07	1.2E+05	9.0E+02	1.5E-04	2.0E-05	3.7E-04	8.4E-06	3.1E-07	0.0E+00	0.0E+00	0.0E+00	3.1E-07
				2.2E+07	1.2E+05	2.2E+04	2.1E-01	1.6E-01	1.6E-01	1.1E-01	1.2E-01	2.1E-04	1.0E-02	1.6E-02	8.9E-02
10	HDABFEIKAAABAB	8.9E+04	3.0E+01	9.4E+06	1.0E+05	1.5E+04	6.4E-01	5.2E-03	3.5E-03	1.0E-03	9.9E-05	1.5E-05	4.9E-06	1.3E-05	1.2E-04
				1.7E+07	1.2E+05	9.0E+02	1.5E-04	1.5E-05	2.8E-04	6.2E-06	2.3E-07	0.0E+00	0.0E+00	0.0E+00	2.3E-07
				2.2E+07	1.2E+05	2.2E+04	3.6E-01	1.2E-01	1.3E-01	7.4E-02	5.2E-02	2.1E-04	2.7E-03	2.6E-03	3.8E-02
11	HBBDEJBKBBAAAA	3.9E+04	[illegible]E+01	4.4E+08	6.1E+04	9.0E+02	6.3E-01	1.1E-02	1.4E-02	4.4E-03	3.5E-03	3.6E-03	5.6E-04	5.9E-04	4.1E-03
				0.0E+00	6.2E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.2E+07	6.2E+04	2.2E+04	3.7E-01	1.7E-01	1.7E-01	9.0E-02	6.9E-02	2.7E-03	5.2E-03	1.0E-02	5.8E-02
12	HDABEEIKAAABAB	9.0E+04	3.0E+01	9.4E+06	1.0E+05	1.5E+04	7.1E-01	6.8E-03	4.9E-03	2.4E-03	8.0E-04	1.4E-04	6.6E-05	2.8E-04	8.3E-04
				1.7E+07	1.2E+05	9.0E+02	1.5E-02	1.8E-03	2.5E-03	1.2E-03	9.1E-04	9.9E-04	1.8E-04	2.0E-04	1.0E-03
				2.2E+07	1.2E+05	2.2E+04	2.8E-01	1.5E-01	1.6E-01	1.0E-01	6.1E-02	1.1E-03	3.7E-03	4.0E-03	4.6E-02
13	HDABEEICAAABAB	9.0E+04	3.0E+01	9.4E+06	1.0E+05	1.5E+04	8.8E-01	1.2E-02	7.7E-03	3.1E-03	7.7E-04	1.8E-04	5.6E-05	2.6E-04	8.1E-04
				1.7E+07	1.2E+05	9.0E+02	5.9E-03	7.4E-04	1.2E-03	9.3E-05	4.1E-05	5.8E-05	2.1E-05	2.0E-05	4.9E-05
				2.2E+07	1.2E+05	2.2E+04	1.1E-01	1.2E-01	1.3E-01	1.3E-01	1.0E-01	9.2E-04	3.6E-03	6.4E-03	7.1E-02
14	HDABEEICAAAAAB	8.9E+04	3.0E+01	9.4E+06	1.0E+05	1.5E+04	7.5E-01	1.3E-02	8.0E-03	2.8E-03	5.6E-04	1.3E-04	4.1E-05	2.0E-04	6.1E-04
				1.7E+07	1.2E+05	9.0E+02	1.3E-02	4.3E-03	6.6E-03	1.4E-03	5.6E-04	5.8E-04	1.9E-04	1.8E-04	7.0E-04
				2.2E+07	1.2E+05	2.2E+04	2.4E-01	1.2E-01	1.2E-01	1.1E-01	9.6E-02	1.1E-03	4.6E-03	6.1E-03	7.0E-02
15	HABDFJIFABAAAA	4.7E+03	3.0E+01	7.3E+08	2.2E+04	9.0E+02	9.0E-01	7.9E-02	9.2E-03	1.1E-02	6.6E-03	1.0E-04	4.[illegible]E-04	8.1E-04	4.8E-03
				0.0E+00	2.3E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
				3.0E+08	2.3E+04	2.2E+04	1.0E-01	8.8E-03	1.0E-03	1.2E-03	7.4E-04	1.1E-05	4.5E-05	9.0E-05	5.4E-04

Table 4.5-14.

The Mean Source Terms for the Supergroups (e.g. Seismic, Fire).

Mean Source Terms for Seismic

WARN (S)	ELEV (M)	POWER (W)	START (S)	DUR (S)	NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
					RELEASE FRACTIONS								
5.0E+04	3.0E+01	4.1E+08	9.3E+04	9.0E+02	7.5E-01	2.9E-02	2.9E-02	3.1E-02	1.9E-02	1.9E-03	1.5E-03	2.5E-03	1.6E-02
		0.0E+00	9.4E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
		2.5E+07	9.4E+04	2.3E+04	2.5E-01	4.4E-02	4.4E-02	3.0E-02	1.9E-02	2.5E-04	1.1E-03	1.5E-03	1.4E-02

Mean Source Terms for Fire

WARN (S)	ELEV (M)	POWER (W)	START (S)	DUR (S)	NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
					RELEASE FRACTIONS								
9.1E+04	3.0E+01	1.3E+08	1.1E+05	1.4E+04	7.3E-01	1.7E-02	1.1E-02	8.3E-03	4.4E-03	5.8E-04	3.6E-04	7.2E-04	3.9E-03
		1.3E+07	1.2E+05	7.3E+02	1.8E-02	8.4E-03	1.1E-02	4.7E-03	2.9E-03	3.1E-03	6.6E-04	7.0E-04	3.4E-03
		6.0E+07	1.2E+05	2.2E+04	2.4E-01	1.3E-01	1.3E-01	9.4E-02	6.7E-02	7.4E-04	4.9E-03	6.3E-03	5.2E-02

Mean Source Terms for Flood

WARN (S)	ELEV (M)	POWER (W)	START (S)	DUR (S)	NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
					RELEASE FRACTIONS								
4.7E+03	3.0E+01	6.3E+08	2.4E+04	9.0E+02	7.2E-01	4.3E-02	1.7E-02	1.3E-02	7.7E-03	2.3E-03	8.1E-04	1.0E-03	6.6E-03
		0.0E+00	2.4E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
		2.4E+08	2.4E+04	1.9E+04	1.6E-01	4.7E-02	2.3E-02	1.7E-02	1.0E-02	5.4E-05	6.7E-04	9.1E-04	7.5E-03

Mean Source Terms for ATWS

WARN (S)	ELEV (M)	POWER (W)	START (S)	DUR (S)	NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
					RELEASE FRACTIONS								
2.7E+03	3.0E+01	2.3E+07	3.8E+03	9.7E+03	7.2E-01	9.4E-03	7.0E-03	4.0E-03	3.4E-03	2.2E-04	9.3E-05	4.1E-04	1.5E-03
		3.5E+07	1.4E+04	9.0E+02	1.2E-02	5.0E-03	6.7E-03	2.6E-03	1.4E-03	1.5E-03	3.3E-04	3.4E-04	1.6E-03
		4.6E+07	1.4E+04	1.7E+04	1.8E-01	1.0E-01	5.5E-02	3.8E-02	2.5E-02	1.7E-04	1.6E-03	2.1E-03	1.9E-02

Mean Source Terms for LOCA

WARN (S)	ELEV (M)	POWER (W)	START (S)	DUR (S)	NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
					RELEASE FRACTIONS								
2.0E+05	3.0E+01	7.9E+06	2.1E+05	2.6E+04	7.4E-01	1.5E-02	1.2E-02	7.1E-03	2.9E-03	4.8E-04	2.1E-04	9.1E-04	3.0E-03
		1.3E+07	2.4E+05	9.0E+02	1.3E-02	4.9E-03	6.4E-03	1.8E-03	1.2E-03	1.5E-03	2.8E-04	2.7E-04	1.4E-03
		1.7E+07	2.4E+05	2.3E+04	2.5E-01	8.0E-02	8.1E-02	5.1E-02	3.6E-02	1.5E-04	3.0E-03	3.7E-03	2.8E-02

Mean Source Terms for Transient

WARN (S)	ELEV (M)	POWER (W)	START (S)	DUR (S)	NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
					RELEASE FRACTIONS								
4.7E+04	3.0E+01	3.9E+08	7.4E+04	3.4E+03	6.7E-01	2.6E-02	1.6E-02	1.3E-02	7.5E-03	1.5E-03	7.2E-04	1.1E-03	6.5E-03
		1.9E+06	7.8E+04	1.2E+02	3.2E-03	1.5E-03	1.9E-03	7.3E-04	4.2E-04	4.6E-04	1.0E-04	1.1E-04	5.0E-04
		8.7E+07	7.8E+04	2.0E+04	1.8E-01	4.7E-02	4.0E-02	2.7E-02	1.7E-02	2.2E-04	1.2E-03	1.6E-03	1.3E-02

Mean Source Terms for Transient-LOCA

WARN (S)	ELEV (M)	POWER (W)	START (S)	DUR (S)	NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
					RELEASE FRACTIONS								
3.1E+04	3.0E+01	3.9E+08	5.6E+04	3.3E+03	6.4E-01	2.6E-02	1.1E-02	1.1E-02	7.4E-03	6.5E-04	5.6E-04	8.5E-04	5.9E-03
		2.9E+06	5.9E+04	1.6E+02	3.1E-03	1.5E-03	2.0E-03	8.1E-04	5.0E-04	5.4E-04	1.1E-04	1.2E-04	5.8E-04
		1.2E+08	5.9E+04	2.0E+04	1.8E-01	5.4E-02	4.4E-02	3.0E-02	2.0E-02	2.2E-34	1.4E-03	1.9E-03	1.6E-02

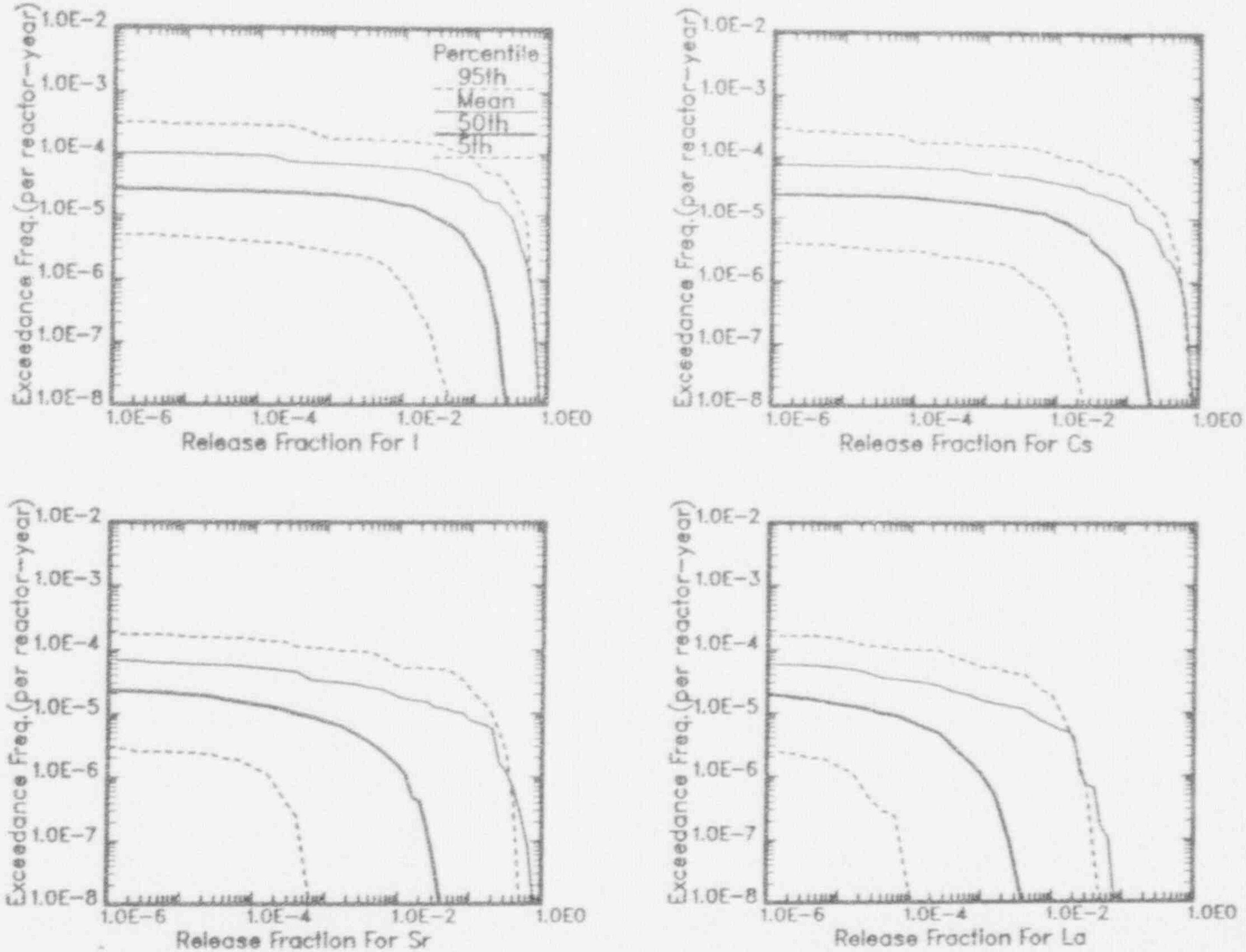

Figure 4.5-1. CCDFs for Release Fractions for I, Cs, Sr, and La Classes For the Total Analysis.

fractions implies that very small releases rarely occur. For example, the mean I release fractions are between 1.0E-03 and 0.8. This behavior is not as prevalent in Sr and La since they are less volatile.

The same information is shown in Figures 4.5-2 thru 4.5-8 for the summary PDS groups: seismic, fire, flood, ATWS, LOCA, transient, and transient-LOCA, respectively. A comparison of these CCDFs shows that the highest exceedance frequencies occur for fire and transient scenarios while the lowest exceedance frequencies occur for LOCA scenarios. This is due to the relative contribution of each PDS group to the core damage (i.e., fire and transient scenarios contribute significantly to the core damage frequency).

The CCDFs for the collapsed APBs are shown in Figures 4.5-9 thru 4.5-16. The collapsed APBs are: (1) vessel breach, early containment failure, low RPV pressure, (2) vessel breach, early containment failure, high RPV pressure, (3) vessel brea , late containment failure, (4) vessel breach, successful venting, (5) vessel breach, no containment failure, (6) no vessel breach, containment failure, (7) no vessel breach, successful venting, and (8) no vessel breach, no containment failure.

A comparison of the CCDFs for the vessel breach, early containment failure cases with high RPV pressure (Figure 4.5-10) with those with low RPV pressure (Figure 4.5-9) shows that the difference in the source term is not significant. The fraction of a species that was released from the fuel that is subsequently released from the vessel depends on the pressure of the RPV. However, the fraction of a species that was retained in the vessel is available for revolatilization after vessel breach, which does occur in these cases. Thus, the source terms due to different RPV pressures are comparable since the vessel does breach.

Comparing the CCDFs for the vessel breach, early containment failure, low RPV pressure collapsed bin (Figure 4.5-9) with those for the vessel breach with late containment failure collapsed bin (Figure 4.5-11) shows many similarities. The maximum magnitude of the release fractions are approximately the same and the ranges of the exceedance frequency are also approximately the same. However, in the late containment failure case, the source terms are confined to a smaller range of magnitude as evidenced by the flatter curves at lower release fractions. This is occurs because all of the releases (before vessel breach, at vessel breach, and after vessel breach) reach the environment through the same path. In contrast, an early containment failure may be followed by a later failure, depending on the accident progression bin, and thus the releases may pass through different release paths (and different scrubbing mechanisms) yielding a larger spread in the magnitude of the source terms.

The importance of the release path can be demonstrated by comparing the CCDFs for the vessel breach case with early containment failure and low pressure (Figure 4.5-9) to the case of vessel breach with venting (Figure 4.5-12). The venting pathway is through the wetwell above the water line. If the cavity floor has failed, radionuclides may leave the containment

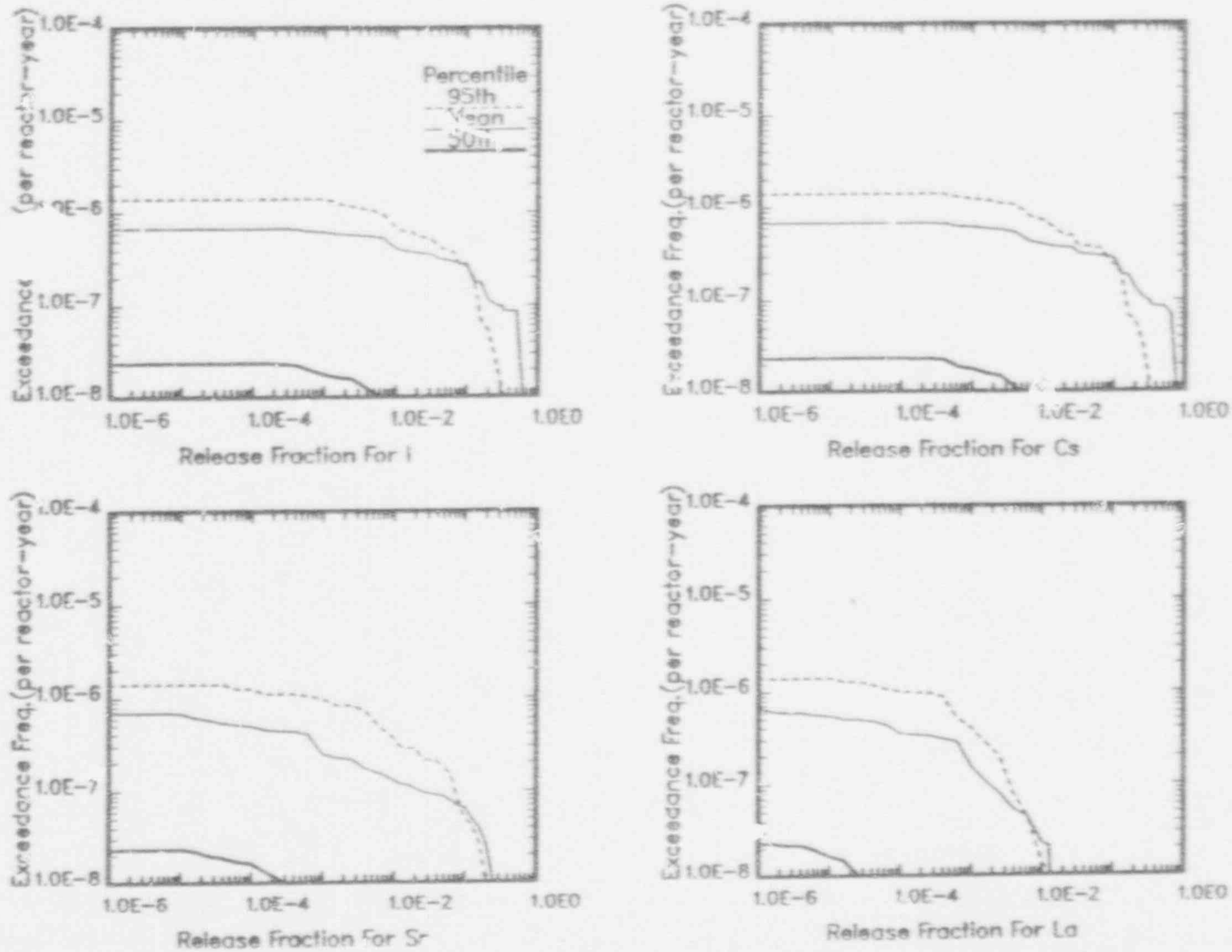

Figure 4.5-2. CCDFs for Release Fractions for I, Cs, Sr, and La Classes For the Seismic Analysis.

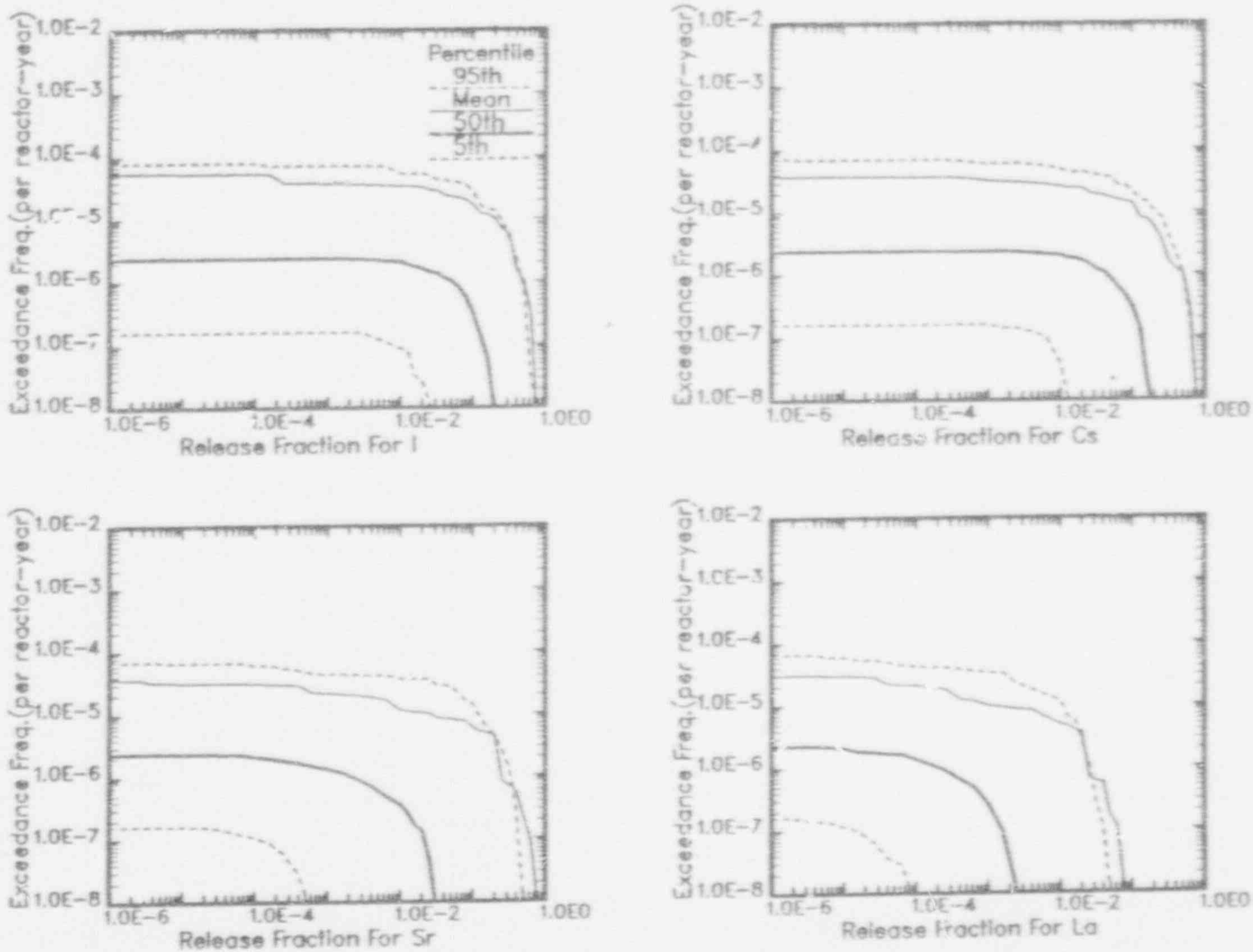

Figure 4.5-3. CCDFs for Release Fractions for I, Cs, Sr, and La Classes For the Fire Analysis.

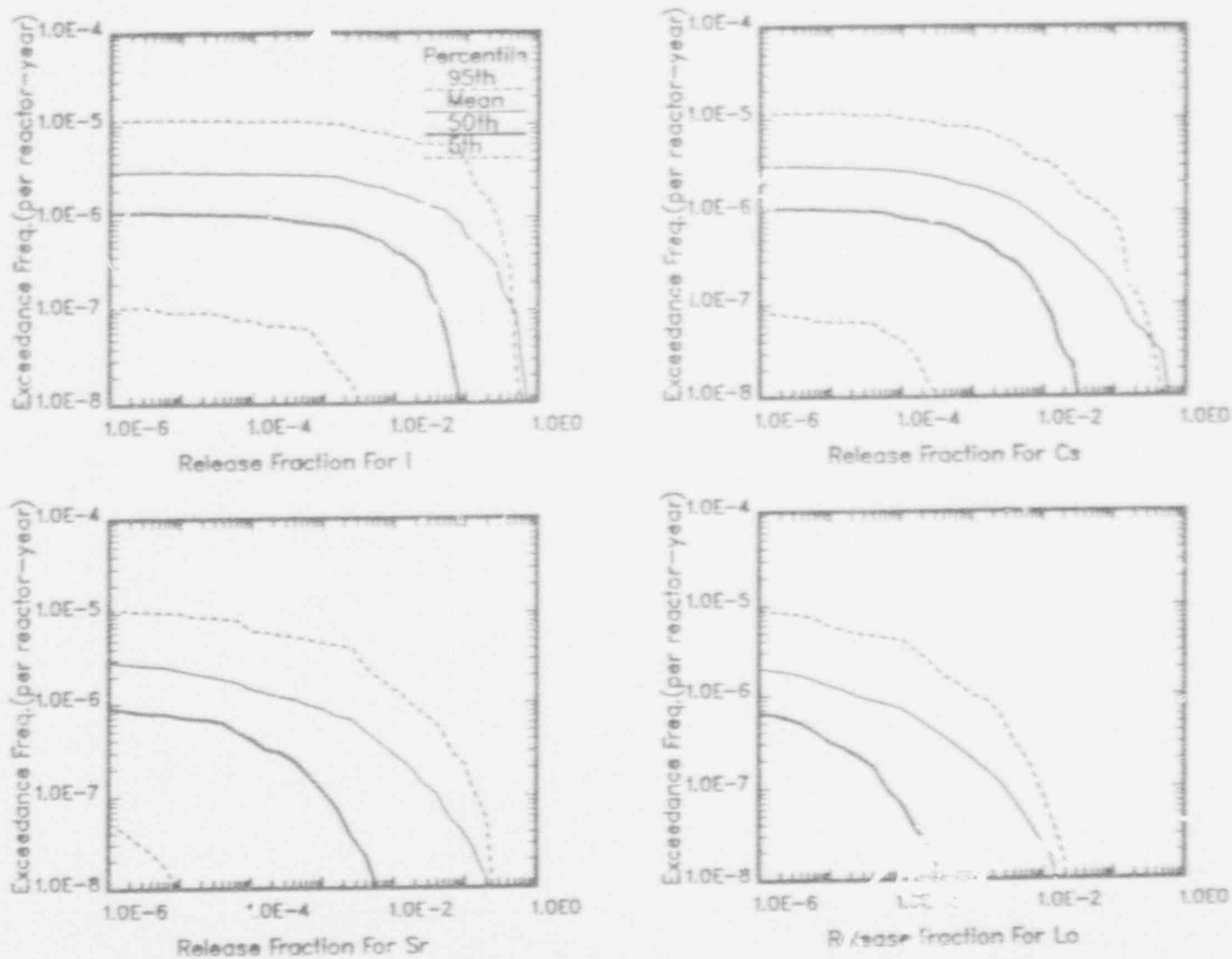

Figure 4.5-4. CCDFs for Release Fractions for I, Cs, Sr, and La Classes For the Flood Analysis.

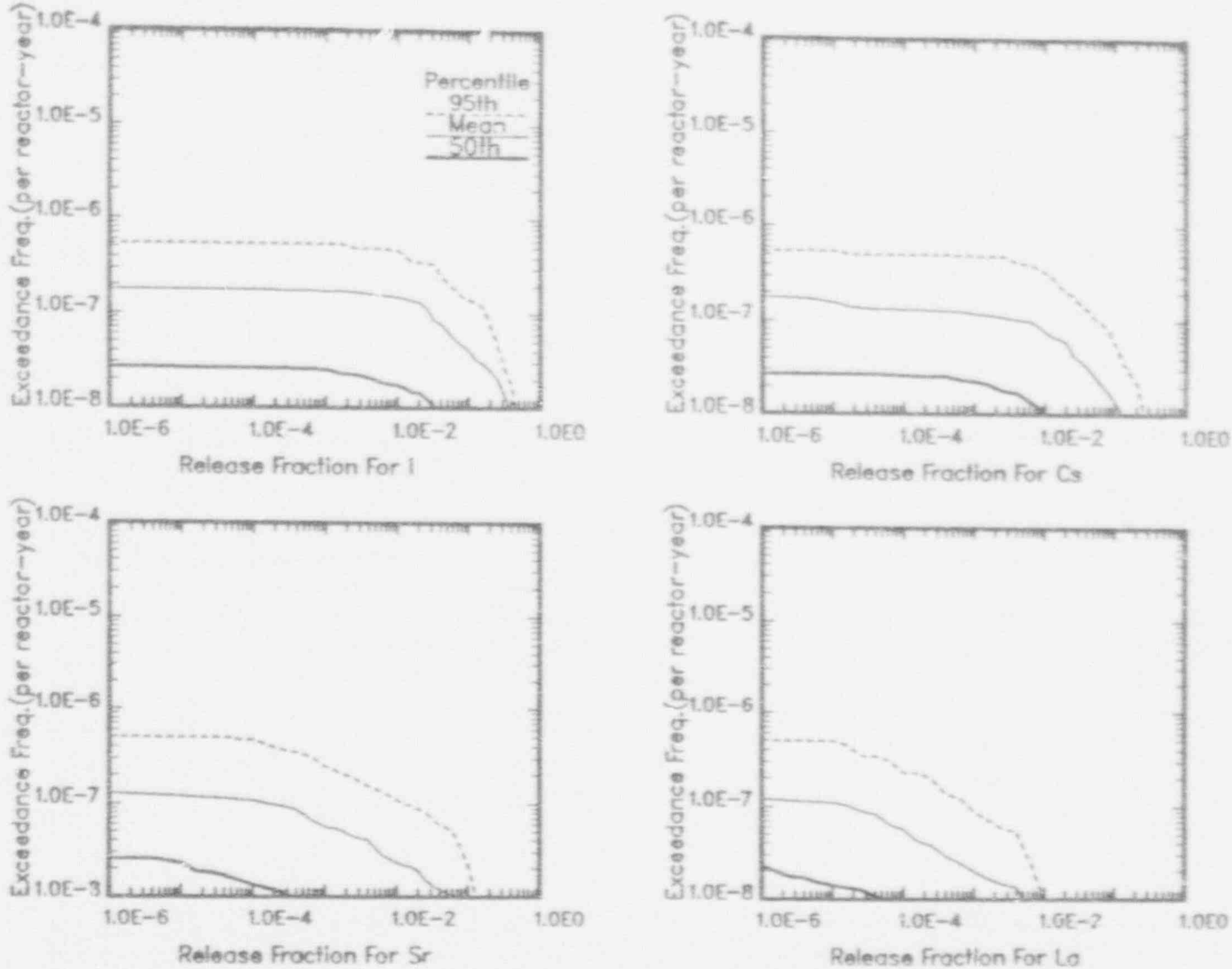

Figure 4.5-5. CCDFs for Release Fractions for I, Cs, Sr, and La Classes For the ATWS Analysis.

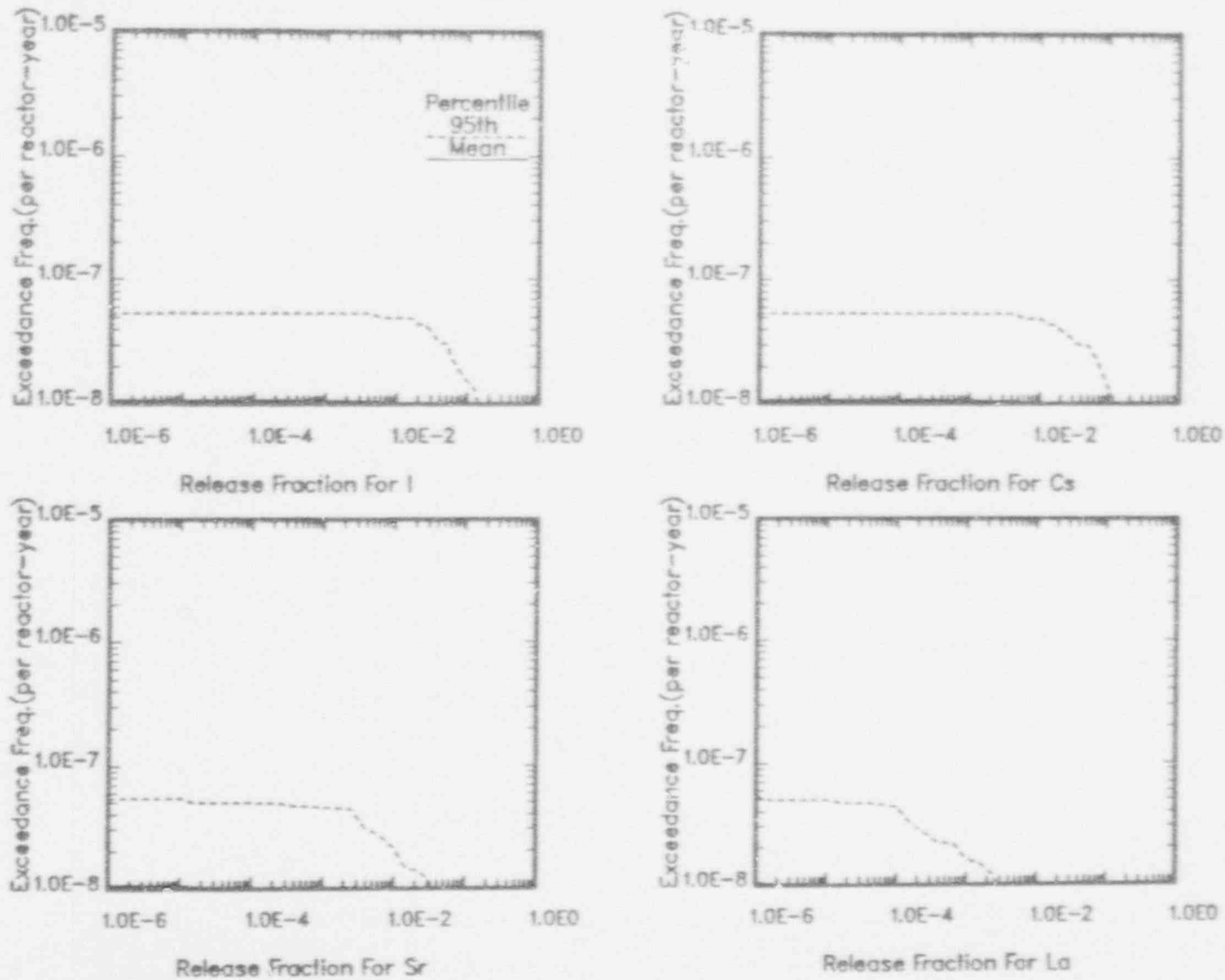

Figure 4.5-6. CCDFs for Release Fractions for I, Cs, Sr, and La Classes For the LOCA Analysis.

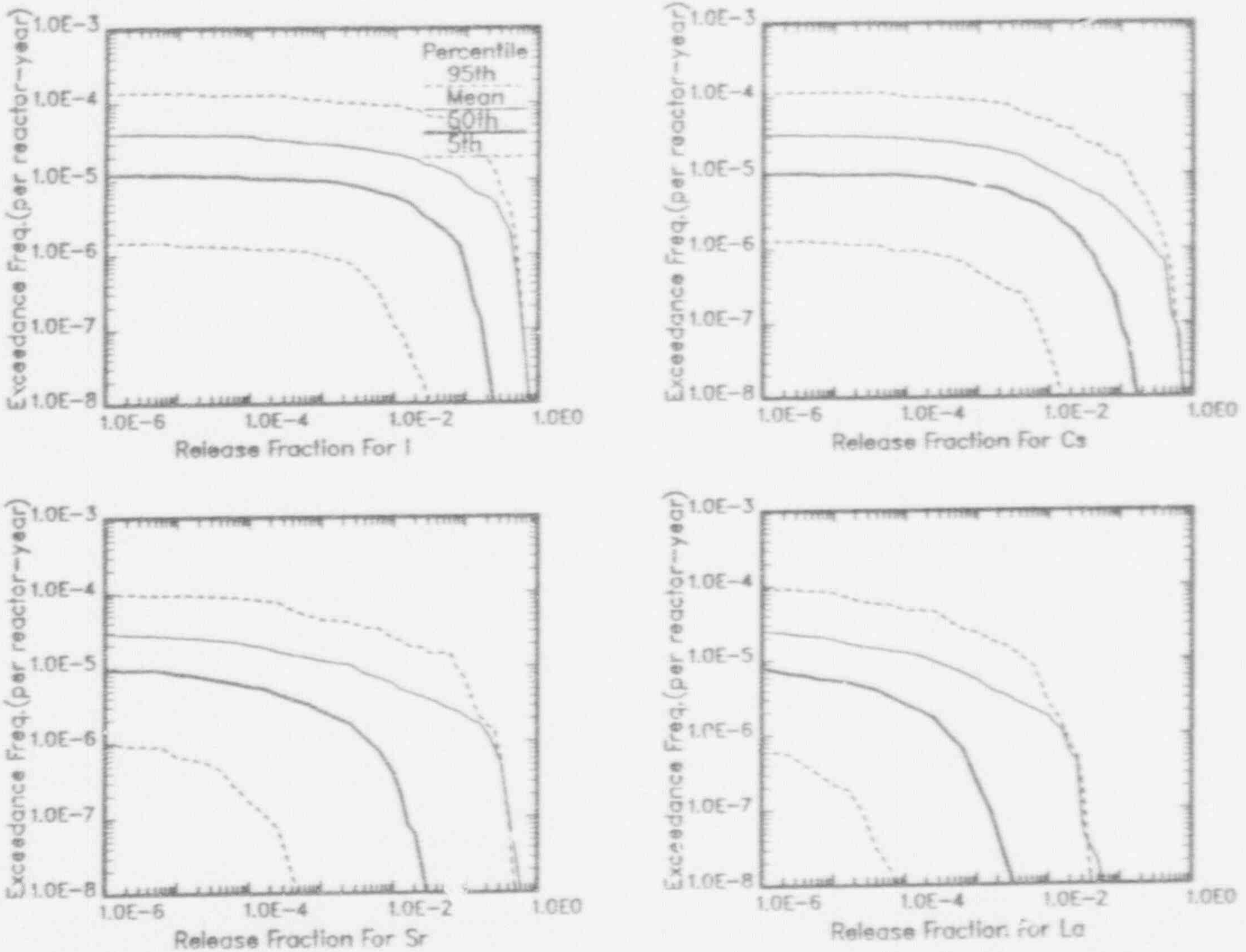

Figure 4.5-7. CCDFs for Release Fractions for I, Cs, Sr, and La Classes For the Transient Analysis.

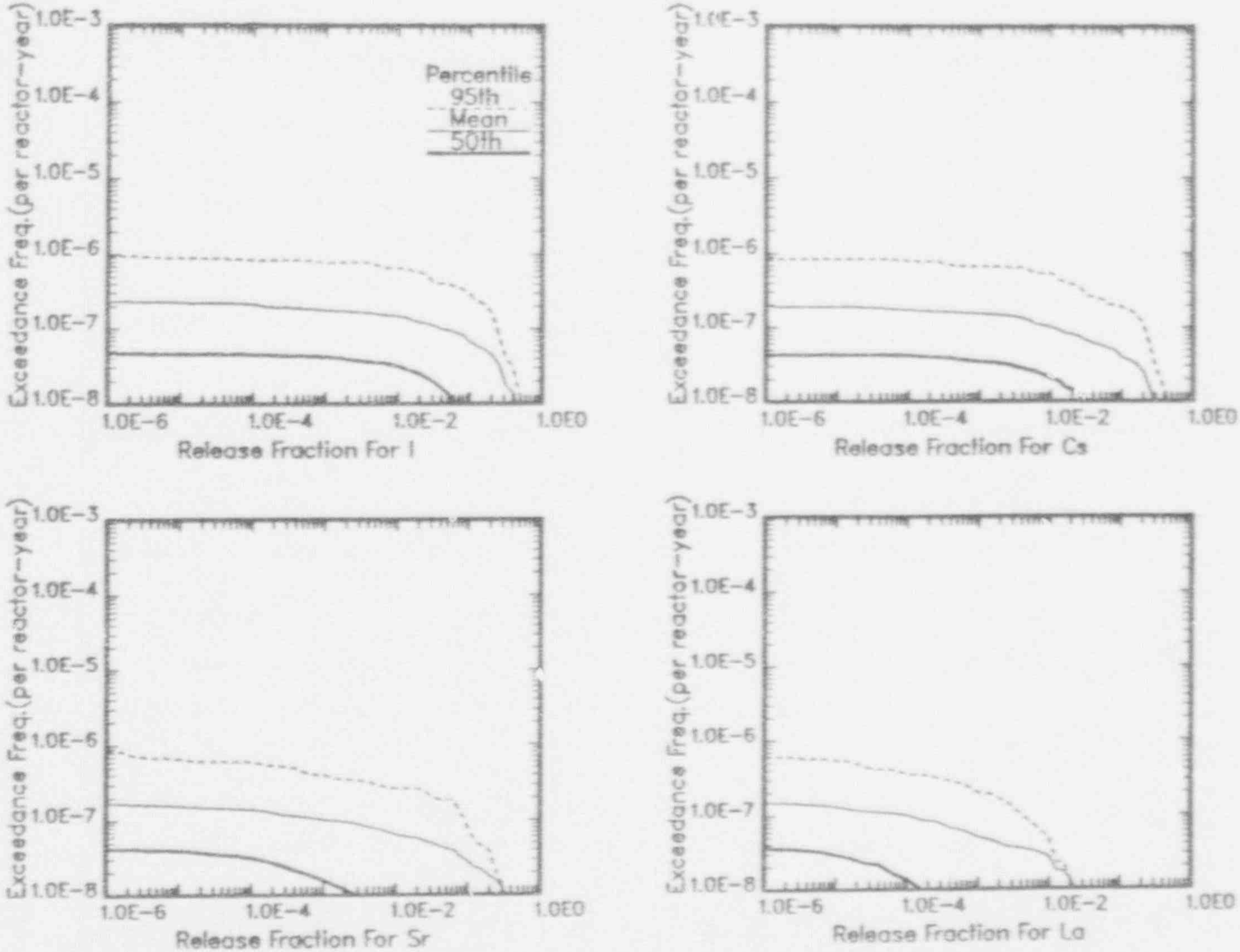

Figure 4.5-8. CCDFs for Release Fractions for I, Cs, Sr, and La Classes For the Transient-Induced LOCA Analysis.

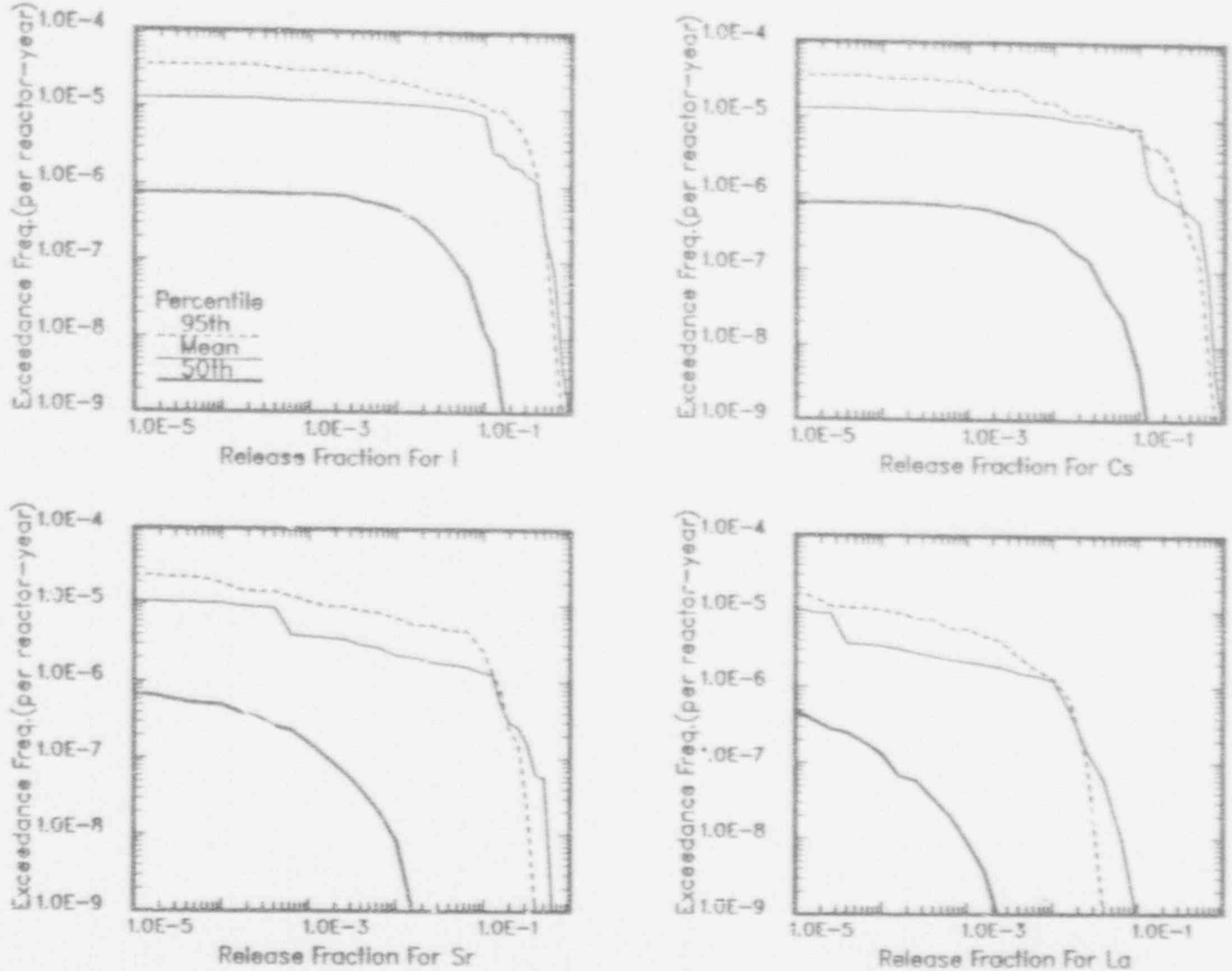

Figure 4.5-9. CCDFs for Release Fractions for I, Cs, Sr, and La Classes For the Summary Accident Progression Bin: VB, Early CF, Low Press.

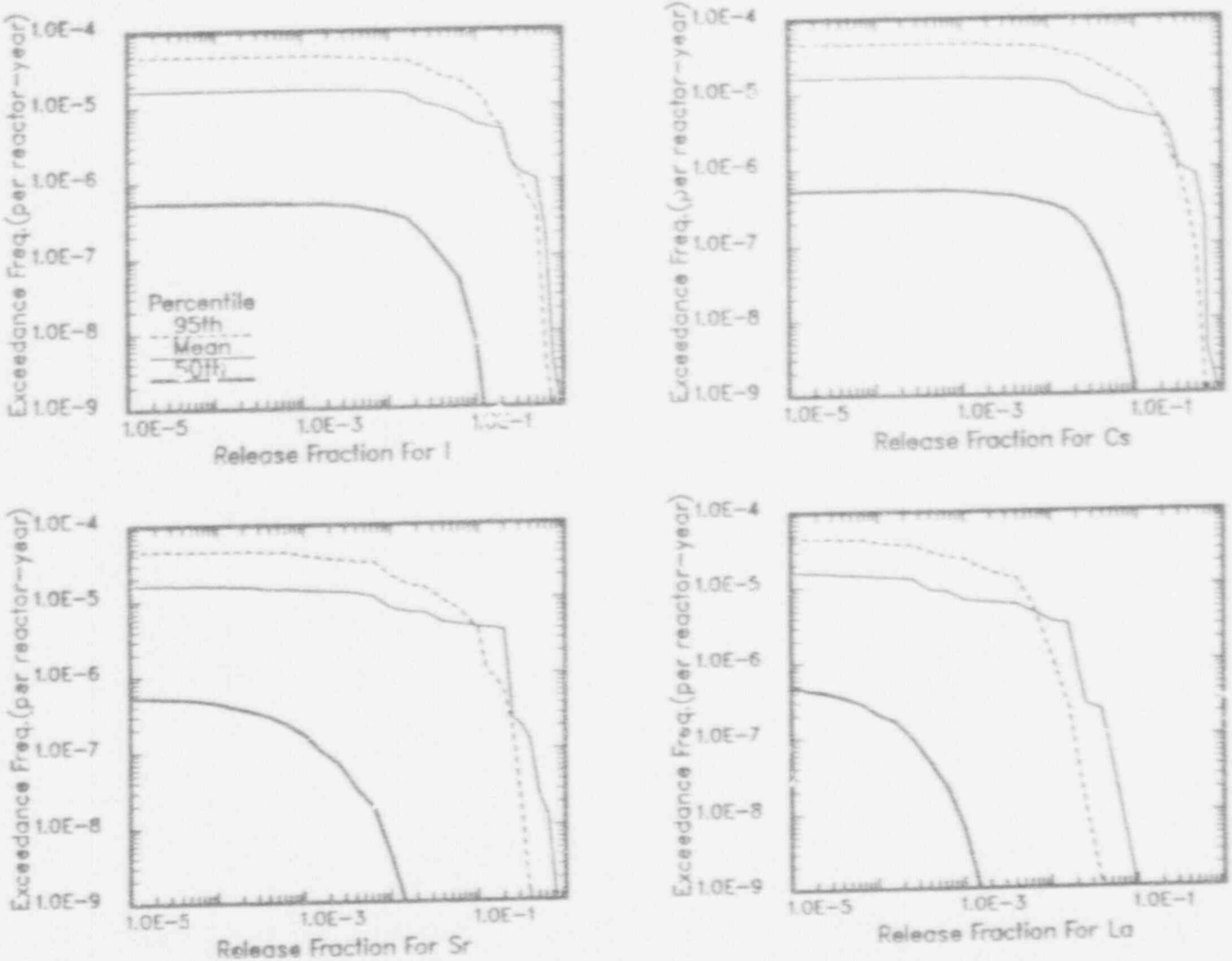

Figure 4.5-10. CCDFs for Release Fractions for I, Cs, Sr, and La Classes For the Summary Accident Progression Bin: VB, Early CF, High Press.

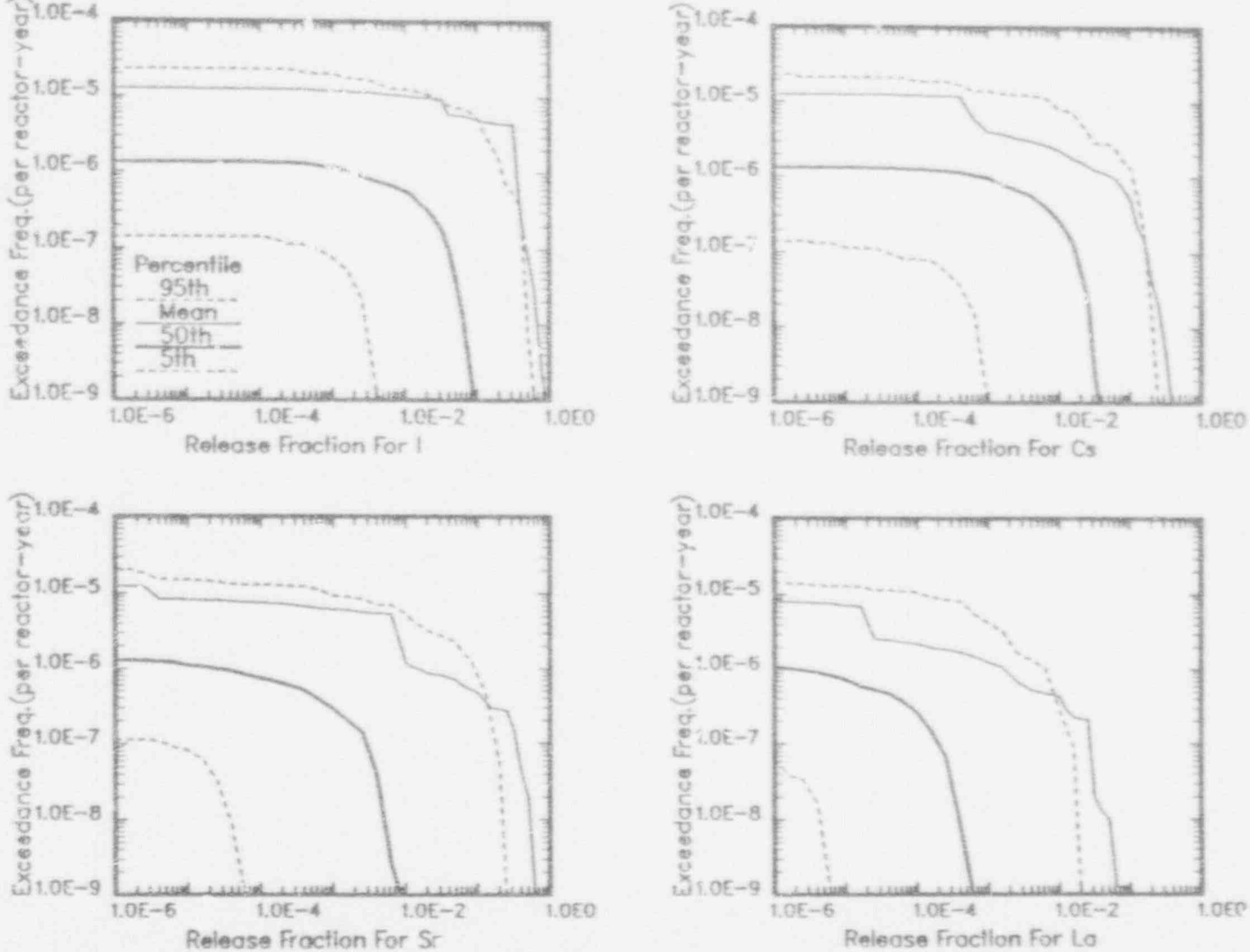

Figure 4.5-11. CCDFs for Release Fractions for I, Cs, Sr, and La Classes For the Summary Accident Progression Bin: VB, Late CF.

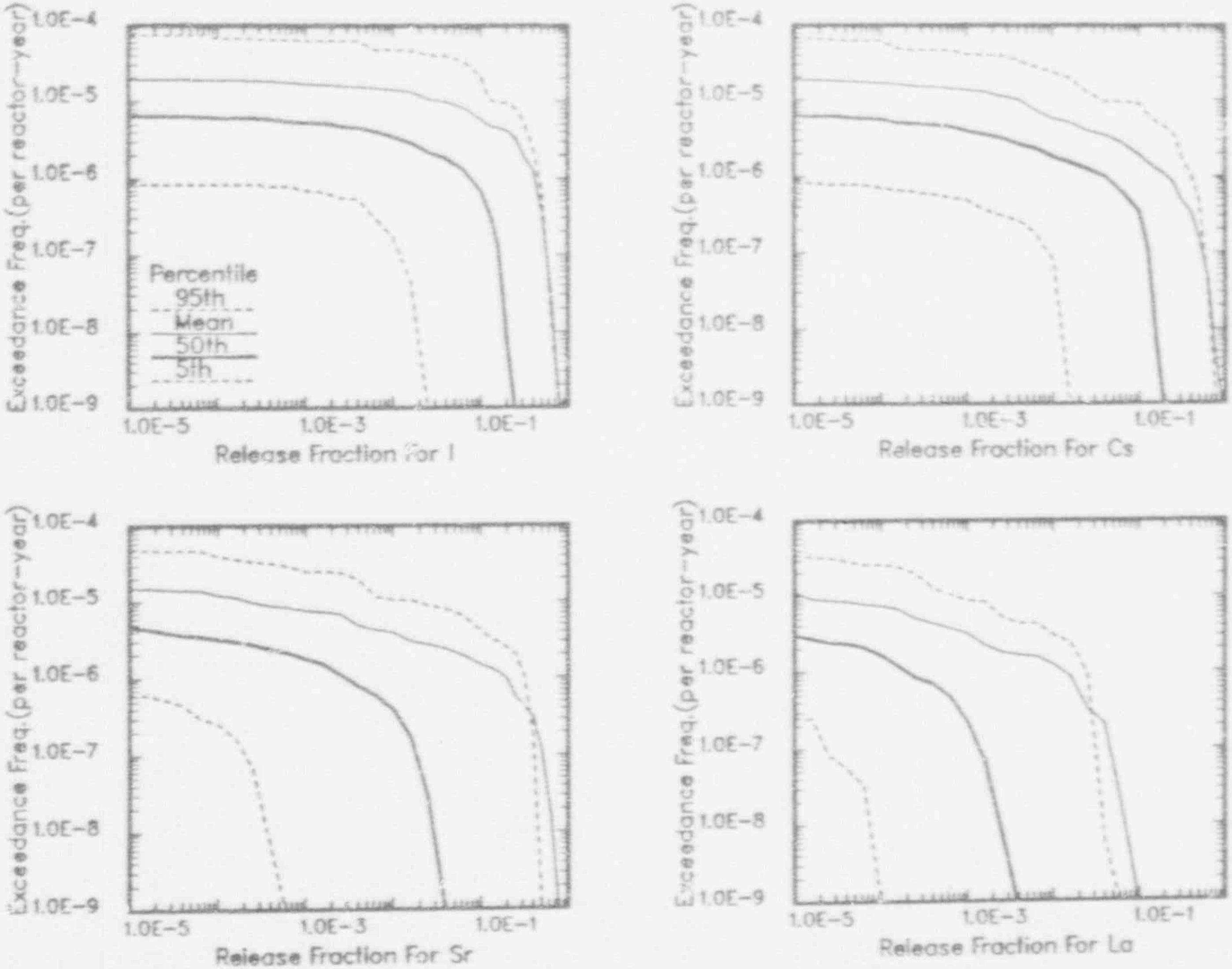

Figure 4.5-12. CCDFs for Release Fractions for I, Cs, Sr, and La Classes For the Summary Accident Progression Bin: VF Vent.

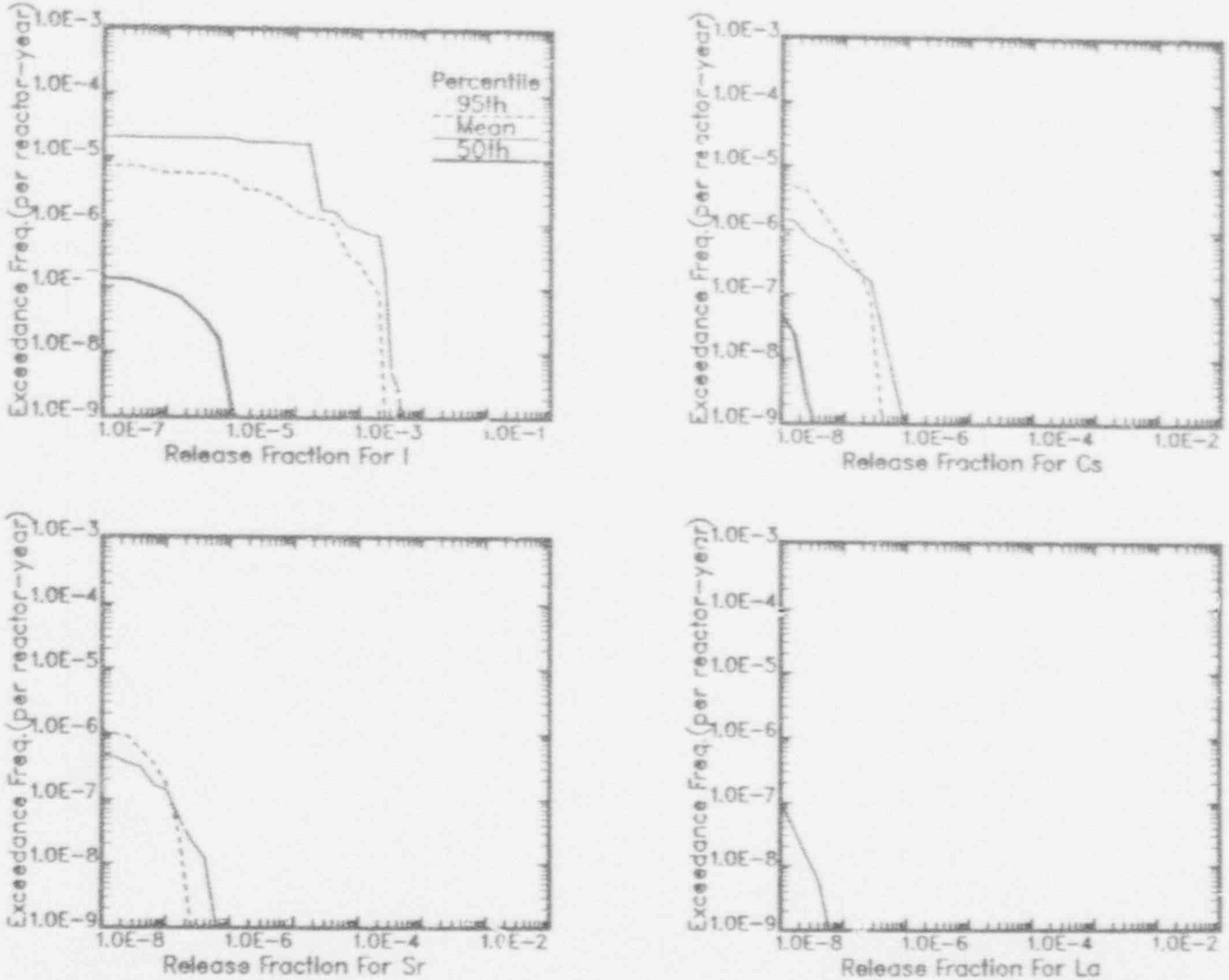

Figure 4.5-13. CCDFs for Release Fractions for I, Cs, Sr, and La Classes For the Summary Accident Progression Bin: VB, No CF.

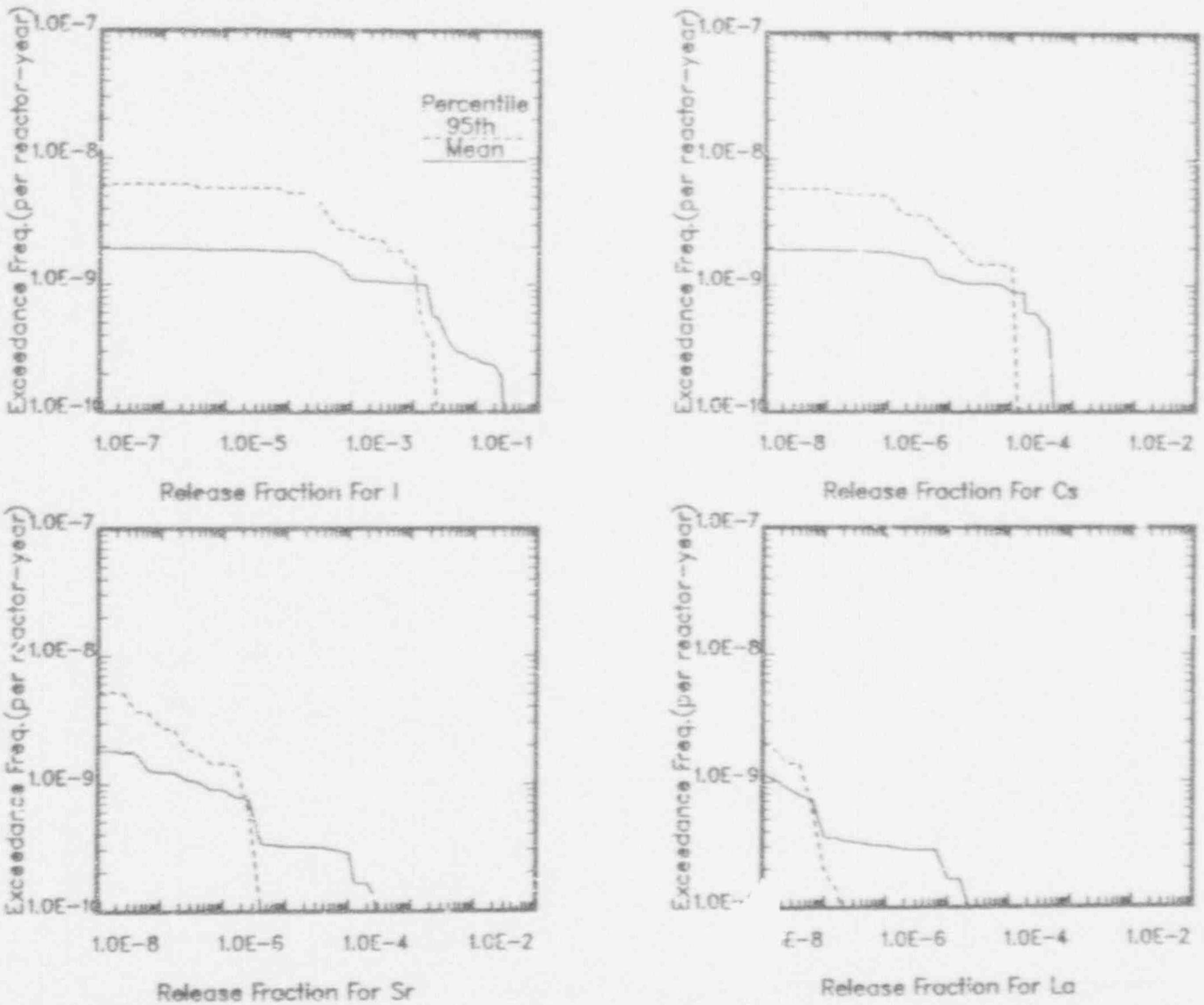

Figure 4.5-14. CCDFs for Release Fractions for I, Cs, Sr, and La Classes For the Summary Accident Progression Bin: No VB, CF.

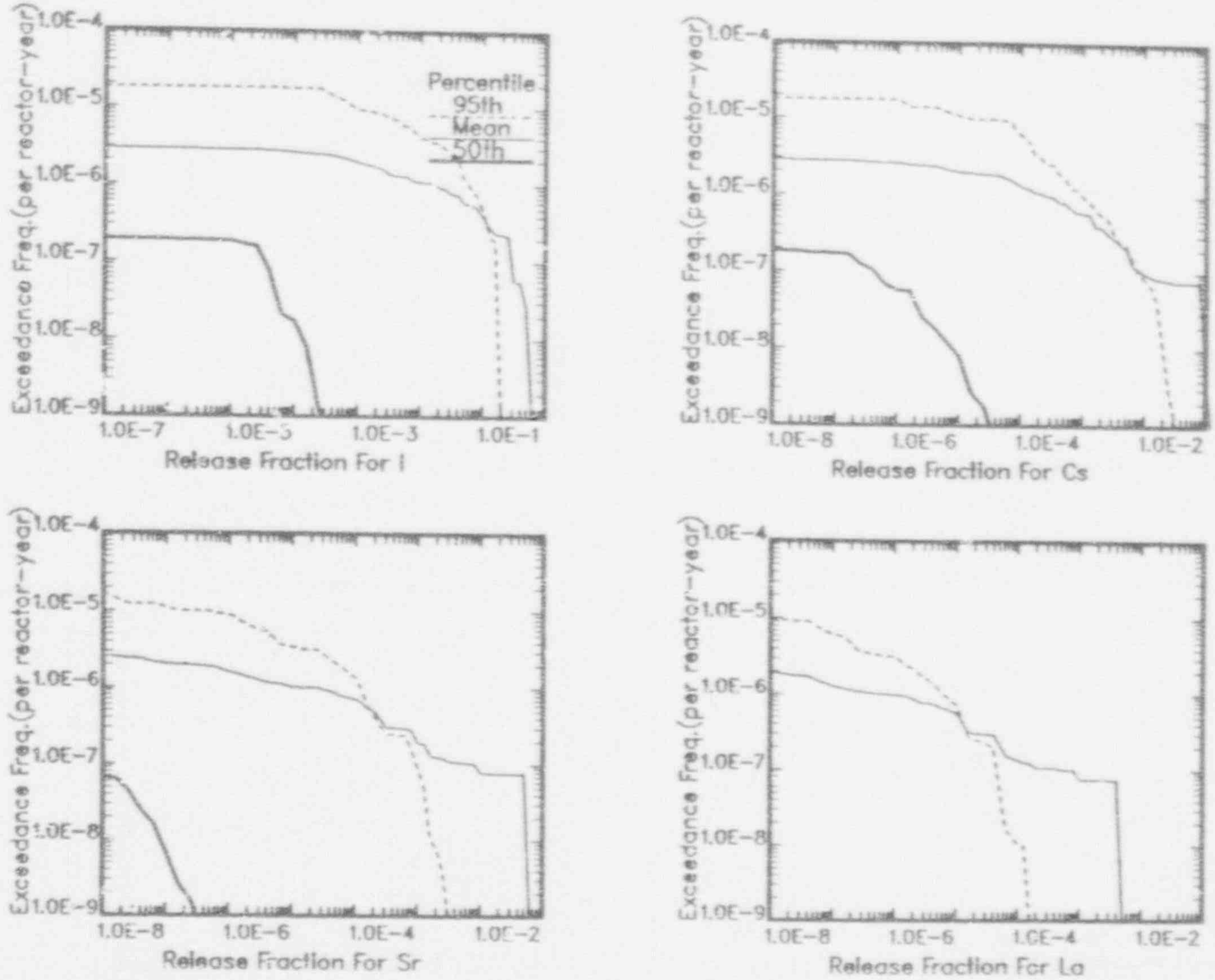

Figure 4.5-15. CCDFs for Release Fractions for I, Cs, Sr, and La Classes For the Summary Accident Progression Bin: No VB, Vent.

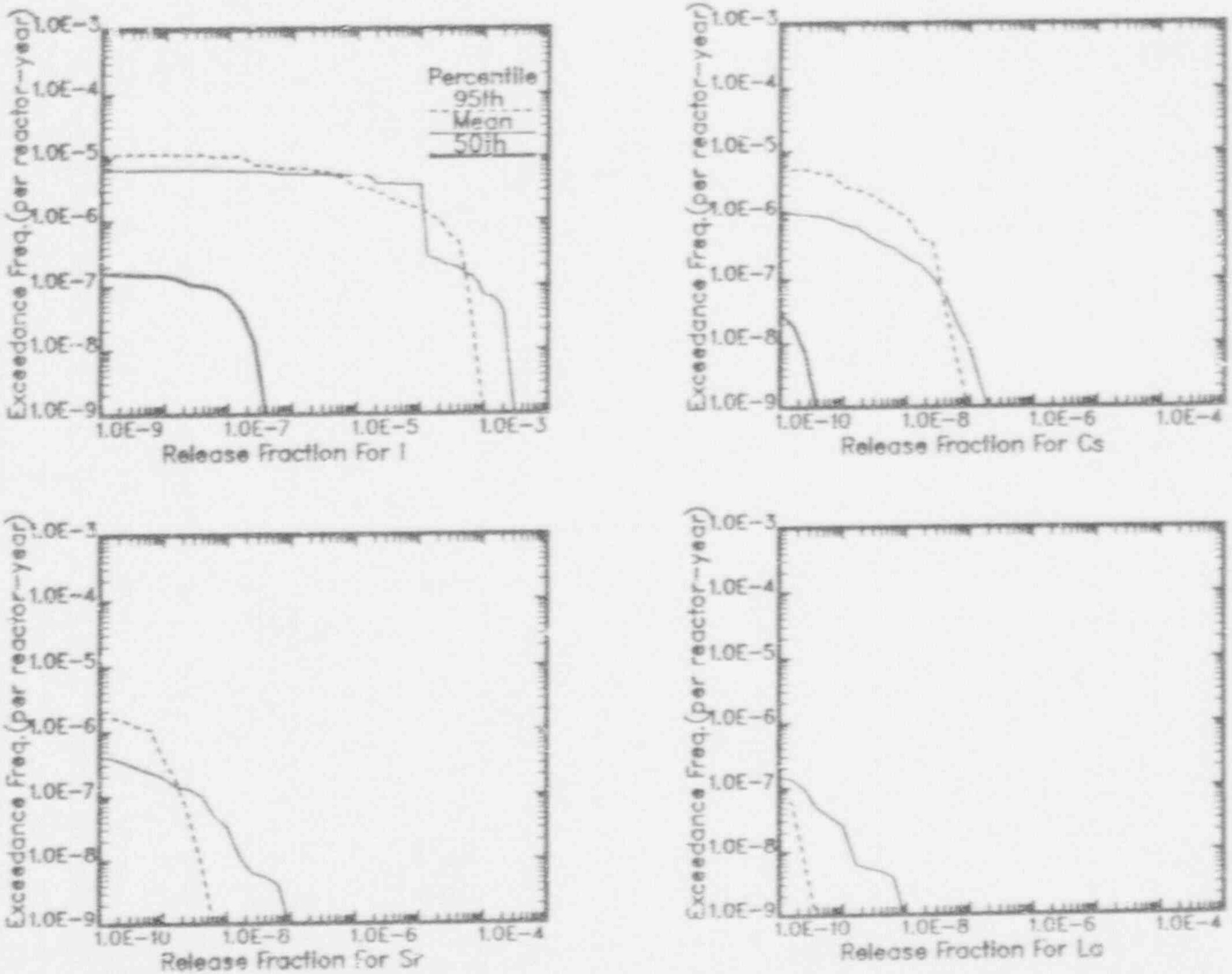

Figure 4.5-16. CCDFs for Release Fractions for I, Cs, Sr, and La Classes For the Summary Accident Progression Bin: No VB, No CF.

without being scrubbed by the pool or the sprays, resulting in a larger source term.

The CCDF for the vessel breach with no containment failure case (Figure 4.5-13) shows that the magnitude of the source term is drastically decreased since the containment remains intact. The seemingly high mean release fractions of I is due to a few high source terms. The I released during the accident may be scrubbed by the pool and later revolatilized. After revolatilizing, the I is not scrubbed by any mechanism (based on the expert's opinions) as it passes out of the containment by normal leakage paths. This is true even with the assumption of minimum DF used in this case.

Comparing the CCDFs for the vessel breach with no containment failure case (Figure 4.5-13) to the no vessel breach with containment failure case (Figure 4.5-14) shows that the relative importance of the containment is greater than that of the vessel in retaining radionuclides. This is as one would expect since there are many large surfaces for the radionuclides to deposit and settle on in the containment.

Comparing the no vessel breach with containment failure case (Figure 4.5-14) to the no vessel breach with successful venting case (Figure 4.5-15) shows that the latter results in significantly higher source terms. This is due to the difference in deposition in the containment for the two cases resulting from the longer residence time of radionuclides in the containment in cases with leakage instead of rupture. Venting always results in a rupture with a relatively small amount of deposition in the containment. On the other hand the containment failure case also includes leaks with a higher amount of radionuclide deposition in the containment.

Figure 4.5-16 shows that, even though the containment and vessel do not fail, releases may still be expected to the environment due to technical specification leakage from the containment.

4.6 Partitioning of the Source Terms for the Consequence Analysis

4.6.1 Introduction

As a result of the accident progression analysis for LaSalle, 75,680 accident progression bins were retained for further analysis. Using the methods described in Sections 4.0-4.5, source terms were estimated for all of these accident progression bins using the LASSOR code. The consequences resulting from these source terms were calculated with the MACCS consequence model[4] and are described in Section 5 of this report.

If all the possible consequence measures are calculated, the MACCS code takes about 30 minutes for each source term calculation. By reducing the consequence measures calculated to only those specifically needed, the time for the calculation of an individual source term can be reduced to about 7.5 minutes. Even with this reduction in calculation time, if all 75,680 source terms were individually calculated, the amount of effort required to determine the risk would be inordinate. Some simplification to bring the level of effort into line with the other stages of the analysis is, therefore, required.

In this analysis, the source terms were grouped on the bases of similar characteristics and a frequency-weighted mean source term was determined for each group. Then, a single MACCS calculation was performed for each mean source term and the consequences determined for each group. The consequence results for a group were then assigned to each source term in the group. If a process can be defined that limits the number of groups to be analyzed and also represents the unique characteristics of the source terms in sufficient detail, then the simplification will adequately represent the actual expected result. This grouping of the source terms was performed with the PARTITION program, and the process is referred to as "partitioning the source terms" or just "partitioning."

Section 4.6.2 presents an overview of the partitioning process. Section 4.6.3 discusses the partitioning process in some detail, and the partitioning results for the integrated analysis are given in Section 4.6.4.

4.6.2 Partitioning Process

The initial partitioning process developed for the LaSalle analysis was used in NUREG-1150 and is described in NUREG/CR-4551, Vol. 1,* and in the user's manual for the PARTITION program.[12] For the final LaSalle analysis, further improvements have been made in the partitioning process to better determine the actual consequences from the potential consequences. This section describes the general method used to partition the source terms generated in this analysis.

The partitioning process used for the LaSalle analysis consists of the following steps:

1. Site specific MACCS calculations were performed to create the early health (EH) effect and chronic health (CH) effect weights which are the bases of the partitioning process. This is described in Section 4.6.3.1. This was the same as in the initial process applied in NUREG-1150.

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* E. D. Gorham, J. C. Helton, R. J. Breeding, S. C. Hora, W. B. Murfin, and J. L. Sprung, " Evaluation of Severe Accident Risks Volume 1, Part 1: Methodology," NUREG/CR-4551, SAND86-1309, Sandia National Laboratories, Albuquerque, NM, to be published.

2. Each accident progression bin was assigned an event type parameter which indicated the general class of evacuation assumptions that were used in the MACCS calculation. The accident progressions were grouped into three categories for this analysis: (1) high-g seismic, (2) low-g seismic, and (3) all others. For each of these groups, a unique set of evacuation assumptions was used. This was new and is an improvement over the process used in NUREG-1150.

3. The source term and evacuation parameters, whose values are important to the MACCS consequence calculation, were selected and prioritized in terms of their importance in determining the consequences. These parameters, which are independent of the health effect weights, were used to construct groups of source terms that were more homogeneous in their characteristics than would have been obtained solely by using the health effect weights. Thus, the MACCS results more accurately represent the source terms in each group. This was new and is an improvement over the process used in NUREG-1150.

4. For each event type, the PARTITION code was run. The code calculated, for each accident progression bin, the appropriate early and chronic health effect weight risks, constructed a two-dimensional grid based on these risks, and then created source term groups within each of the cells on the grid consisting of accident progression bins with similar values of the source term and evacuation parameters selected in step 3, if possible. The groups were defined in terms of their fractional contribution to the total early health effect weight risk, total chronic health effect weight risk, and total source term frequency in that order. The remaining source terms, that could not be grouped, were then assigned to the closest defined bin as described in Section 4.6.3.6. The use of frequency in addition to early and chronic health effect weight risks was new and the use of the source term and evacuation parameters to define the subgroups was new and is an improvement over the process used in NUREG-1150.

5. In order to perform the integrated calculation, the source term groups for the different event types, which were each numbered 1 to n_i, were combined into one set with the groups renumbered 1 to n_t and the pointers to the individual source terms arranged into one file. This allowed all of the source terms to be calculated in one MACCS calculation, instead of having to make several different runs, and simplified the data handling. This was new and is an improvement over the process used in NUREG-1150.

6. For each source term group, the MACCS input files were constructed showing the specific mean source term for that group, the mean values of the source term and evacuation parameters used in the partitioning process, and the general evacuation assumptions for the event type to which the group belongs. This was the same as the process used in NUREG-1150.

4.6.3 Partitioning Process Summary

The new partitioning process defines partitions based on:

1. early health effect weight risk (product of early health effect weight and frequency for a source term),

2. chronic health effect weight risk (product of chronic health effect weight and frequency for a source term),

3. source term frequency, and

4. source term phenomenology parameters (i.e., release time, duration, and energy) and evacuation characteristics (i.e., warning time, evacuation delay time, and initiator type) which are partitioned to achieve less variation in these parameters for the partition definitions.

The following six sections describe the partition definition process.

4.6.3.1 Calculation of the Health Effect Weights

The early health effect weight is based on converting the radionuclide release associated with a source term into an equivalent I-131 release and then estimating the number of early fatalities that would result from this equivalent I-131 release. This estimated number of early fatalities is the early health effect weight. The relationship between early fatalities and equivalent I-131 releases is shown in the PARTITION input file listed in Appendix D.1 and is based on site-specific MACCS calculations for different-sized releases of I-131 with the conservative assumptions described below. The procedure used to convert each isotopes' release into an equivalent iodine release is described in the PARTITION user's manual.

The chronic health effect weight is based on an assumed linear relationship between cancer fatalities due to a radionuclide and the amount of that radionuclide released. Specifically, a site-specific MACCS calculation, with the conservative assumptions described below, is performed for a fixed release of each of the 60 radionuclides included in the LaSalle consequence calculations. The results of these calculations and the assumed linear relationship between the amount released and cancer fatalities for each radionuclide are then used to estimate the total number of chronic fatalities associated with a source term. This estimated number of chronic fatalities is the chronic health effect weight. The results of the MACCS calculations used in the determination of CH weight are shown in Appendix D.2 and the input file for PARTITION containing the site-specific data used in the calculation of CH weight is shown in Appendix D.1.

The site-specific MACCS calculations that underlie the early and chronic health effect weights were performed with conservative assumptions with respect to the energy and timing of the releases and also with respect to the emergency responses taken. As a result, these weights should be regarded as a measure of the potential of a source term to cause early and chronic fatalities rather than as an estimate of the fatalities that would actually result from a source term. The primary consideration of the partitioning process to be used is to group the source terms so that the conversion of these potential consequences into actual consequences is done in such a way that the effects of the consequence parameters are not masked by the grouping process.

4.6.3.2 Partitioning Process Initialization

The partitioning process was performed by the PARTITION code. For this analysis, a new version of the PARTITION code was developed. For each event type that has unique evacuation assumptions (i.e., high-g seismic, low-g seismic, and all others for this analysis), a separate PARTITION calculation was performed. The source terms for each event type were divided into a grid based on the logarithm of the chronic health effect weights on one axis (horizontal) and the logarithm of the early health effect weights on the other axis (vertical). Table 4.6-2, which will be discussed later, shows an example of this grid for the event type consisting of high-g seismic scenarios. The user defines the number of grid divisions to be used for each axis and the event type to be partitioned. Each cell in this grid has four associated values:

1. number of source terms for that event type in each grid cell,

2. fraction of the total frequency of the source terms in each grid cell relative to the total frequency of all source terms for that event type (i.e., the total frequency of all source terms for that event type is the sum of the frequencies for all source terms of that type and over all LHS observations, it has no direct physical meaning),

3. fraction of the total early health effect weight risk of the source terms in each grid cell relative to the total early health effect weight risk for all source terms for that event type, and

4. fraction of the total chronic health effect weight risk of the source terms in each grid cell relative to the total chronic health effect weight risk for all source terms for that event type in the grid.

All of these fractions are related to the source terms for the event type under consideration, not to the total of all event types, unless the user selects to partition all the source terms at once.

The early health effect weight risk within a grid cell is defined as the sum of the products of the early health effect weight and frequency for all source terms within that grid cell. The chronic health effect weight risk within a grid cell is defined as the sum of the products of the chronic health effect weight and frequency for all source terms within that grid cell. The user defines the minimum fractional values of the following parameters that a group of source terms must have in order for the group to be defined as a partition:

1. early health effect weight risk,

2. chronic health effect weight risk, and

3. source term frequency.

For example, the fraction of the total early health effect weight risk in a group must be at least 0.02 of the total, the fraction of the chronic health effect weight risk must be at least 0.01 of the total, and the fraction of the source term frequency must be at least 0.05 of the total).

In order to be considered to belong to a potential group, the source terms should also have similar characteristics in those parameters that are important to the conversion of potential early and chronic health effects into actual early and chronic health effects when the actual evaluation of the consequences is done in the MACCS code. The analyst, therefore, selects a set of source term parameters for which less variation is desired and prioritizes theses in the partition algoritum prior to program execution. These source term parameters allow more specific partitions to be defined according to source term phenomenology such as timing and energy release rates. The actual parameters used in the LaSalle analysis are, in order of their priority: the event type, the warning time, the delay time, the first release time, the first release duration, the first release energy, the second release time, the second release duration, the second release energy, the third release time, the third release duration, and the third release energy. Other parameters or combinations of parameters can be defined and used if desired.

4.6.3.3 Partitioning According to Early Health Effect Weight Risk

Once the initialization of the grid for the source terms of a particular event type has been completed, the identification and division of the source terms into partitions can begin. The partitioning process starts by defining partitions based on early health effect weight risk. The grid cell having the maximum fraction of the early health effect weight risk is first located. Ranges for the previously prioritized list of source term and evacuation parameters are calculated for all the source terms in this grid cell. Each of these ranges is subdivided logarithmically into a user-defined number of subranges. The early health effect weight risk for each of the source terms in the grid cell is then accumulated into the subrange corresponding to the value of each source term parameter (e.g., 1) the range of warning times for all the source terms within a cell is divided into 3 subranges, 2) the value of the warning time for each source term in the cell is compared to the subrange values and each source term is assigned to one subrange, and 3) the fraction of the early health effect weight risk corresponding to all of the source terms in that subrange are added up). The set of subranges having the highest accumulated early health effect weight risk for each source term parameter defines a possible partition.

Next, since the parameter values for a particular source term do not necessarily all fall within the set of subranges having the highest accumulated early health effect weight risks, the subset of source terms that have all their source term parameter values within the defined subranges of the possible partition is identified. If the early health effect weight risk for this subset of source terms relative to the total original early health effect weight risk in the grid is equal to or greater than the user-defined minimum for a partition definition (i.e., if the fraction is greater than the user defined minimum determined in the initialization process described above), this subset of source terms defines a partition.

If the early health effect risk for this subset of source terms relative to the total original early health effect weight risk in the grid is less than the user-defined minimum for a partition definition, the source term parameter with the least priority is removed from the set of parameter ranges and the set of source terms with parameter values within this reduced set of subranges is identified. It is expected that the number of source term should increase since the requirements are less restrictive. The early health effect weight risk for the source terms having all their parameter values within the defined subranges for this reduced set of parameters is calculated. If the fraction is equal to or greater than the user-defined minimum then a partition is defined. If not, the next lowest priority parameter is dropped and the process repeated. The process of removing additional source term parameters from consideration is repeated until the early health effect weight risk relative to the total original early health effect weight risk in the grid, for the subset of source terms in the subranges, exceeds or equals the user-defined minimum for a partition definition or until no parameters are left. If the fraction of the total early health effect weight risk of all the source terms within the grid cell is greater than the user-defined minimum and no parameters besides the event type are used, then one partition, which includes all the source terms in the cell, is defined. If the fraction of the total early health effect risk within the cell is less than the user-defined minimum, then no partition is defined for that cell and the calculation moves to the next cell.

When a partition has been defined, source term parameter values for the partition are calculated based on frequency-weighted mean values corresponding to the subset of grid cell source terms used to define the partition. The contributions of the source terms making up a partition are then removed from the grid (i.e., the source term count, the fraction of the frequency, fraction of early health effect weight risk, and fraction of chronic health effect weight risk for that cell are recalculated for that cell and the total remaining number of source terms and fraction of frequency, early health effect weight risk, and chronic health effect weight risk are recalculated for the entire grid (i.e., the grid total fraction remaining is less than 100% by the amount contributed by the source terms in the partition just defined)).

The partitioning process continues by again locating the grid cell having the maximum remaining early health effect risk and succeeding partitions are defined in a similar manner. When the maximum grid cell early health effect weight risk relative to the total original early health effect weight risk in the grid is less than the user-defined minimum for a partition definition, the partition process based on early health effect weight risk is complete. Early health effect weight risk source terms remain in the grid but no single grid cell contains early health effect weight risk exceeding the user-defined minimum required for a partition definition. The partitioning process continues based on chronic health effect weight risk.

4.6.3.4 Partitioning According to Chronic Health Effect Weight Risk

The partitioning process continues by defining partitions based on the chronic health effect weight risk using the source terms remaining in the grid after the application of the previous step. The process is exactly the same as for the early health effect weight risk except that chronic health effect weight risk is used. When no grid cell contains more than the user-defined minimum chronic health effect weight risk, the partitioning process continues based on frequency.

4.6.3.5 Partitioning According to Frequency

The partitioning process continues by defining partitions based on the fractional contribution to the total frequency of all source terms in the grid using only those source terms remaining in the grid after the application of the two previous steps. The frequency of each source term is the sum of the frequencies of the originating plant damage states times the conditional probabilities of all the accident progressions resulting from that plant damage state that fall into the accident progression bin that defines the source term (since each accident progression bin is a source term) and summed over all LHS observations. The total source term frequency has no direct physical interpretation.

When the maximum grid cell fraction of frequency relative to the total original frequency in the grid is less than the user-defined minimum for a partition definition, the partition process based on frequency is complete. Source terms remain in the grid but no single grid cell contains a fraction of the early health effect weight risk, chronic health effect weight risk, or original frequency exceeding the user-defined minimums required for a partition definition. The partitioning process is then over.

4.6.3.6 Pooling of Remaining Source Terms in Grid

Each of the source terms remaining in the grid following the partition definition process is assigned to the closest previously defined partition based on the minimum Euclidean distance (logarithmic scale) between the individual source term early health effect weight and chronic health effect weight and the corresponding frequency-weighted mean values for the partition definitions. Parameters for these source terms are not used to alter the partition definitions but the individual source term frequencies are accumulated into the frequencies associated with the appropriate partition definitions.

4.6.3.7 Combination of Event Types

The process described in Sections 4.6.3.1 thru 4.6.3.6 is repeated for each event type that is to be considered to have unique evacuation assumptions in the consequence analysis. For the LaSalle analysis, three event types were selected. Event type 1 represented high-g seismic scenarios, event type 2 represented low-g seismic scenarios, and event type 3 represented all other scenarios (i.e., all internal, fire, and flood scenarios). The specific evacuation assumptions for each event type are discussed in Section 5.

Each PARTITION run creates a file defining the source term partition groups for the particular event type evaluated. This file contains the following information for each partition group: the mean values of the release fractions, the mean values of the source term and evacuation parameters used both in the partition process and those that are not used for partitioning but are needed to complete the source term definition, and a pointer relating the partition to the individual accident progression bins that make up the partition group. These pointers are necessary in order that the core damage frequency, conditional probability of the accident progression bin, source term, and consequence can be tracked through the calculation and risk values calculated.

In order to perform an integrated calculation, we need one set of partition groups numbered sequentially from 1 to n. The COMBIN code, written for this analysis, reads in the three event type partition output files, renumbers the groups, and renumbers the pointers relating each accident progression bin to the correct partition group.

4.6.3.8 Construction of the MACCS Input Files

The STER code takes generic MACCS ATMOS and EARLY input files and appends the information to define the source term and emergency response assumptions. The code creates a set of MACCS input files for each partition group defined in the analysis.

4.6.3.9 Input Modifications

The input file specifying the files to be used as input to the PARTITION program was altered from that used in the NUREG-1150 analysis for the LaSalle analysis. Each file in the list is now preceded by a 3-character identification so that the files may be specified in any order. The 3-character identifications are:

MBL -- master bin list
WGT -- dose factors and effect weights
SOR -- source terms to be partitioned
APB -- accident bin conditional probabilities by plant damage state and by sample
PDS -- plant damage state frequencies by sample

4.6.4 Results for the Integrated Analysis

As mention previously, the accident progression analysis and the subsequent source term analysis resulted in the generation of 75,680 source terms for all initiators. Table 4.6-1 shows the division of the source terms into three cases, EH>0 and CH>0, EH=0 and CH>0, and EH=0 and CH=0, and lists the ranges of EH and CH for each case. Also, the twelve parameters used in the partitioning process are listed.

As shown in Table 4.6-1, for the case where EH and CH are both greater than 0, log CH ranges from 0.7746 to 5.3030 and log EH ranges from -0.8180 to 1.6595. For the case where EH=0 and CH>0, log CH ranges from -3.5270 to 4.7281. Figure 4.6-1 shows a plot of the pairs (CH,EH) for the 25,772 source terms for which both EH and CH are nonzero. The partitioning process is based on laying a grid on the (CH,EH) space shown in Figure 4.6-1 and with additional row for the (0,0) and (CH,0) cases (see Tables 4.6-1 thru 4.6-3). Specifically, the number of subdivisions of the grid is selected by the user for the EH and CH dimensions and for the parameters used in the partitioning process. The grid is selected so that the ratio between the maximum and minimum value for CH in any cell and also the ratio between the maximum and minimum value for EH in any cell will be equal. The result of placing the selected grid on the (CH,EH) space is also shown in Figure 4.6-1, neglecting the (CH,0) row.

A summary of the partitioning process for event type 1, high-g seismic scenarios is given in Table 4.6-2. The table is divided into two parts. The first page is labeled "BEFORE PARTITIONING" and shows the distribution of the source terms before the partitioning process. As in Figure 4.6-1, the abscissa and ordinate correspond to CH and EH, respectively, with the ranges given in Table 4.6-1. The bottom row shows the EH=0, CH>=0 source terms. The top grid shows the number of source terms in each cell for the event type under consideration, the second grid shows the fraction of the

Table 4.6-1
Summary of Early and Chronic Health Effect Weights
for All Initiators

	Number of Source Terms	Percent of Total Frequency
EH>0 AND CH>0	25772	31.81
EH=0 AND CH>0	49908	68.19
EH=0 AND CH=0	0	0.00
TOTAL	75680	100.00

FOR EH>0 AND CH>0, RANGE LOG10(CH)= 0.7746 TO 5.3030
RANGE LOG10(EH)= -0.8180 TO 1.6595

FOR EH=0 AND CH>0, RANGE LOG10(CH)= -3.5270 TO 4.7281

NUMBER OF SOURCE TERM PHENOMENOLOGY PARAMETERS DEFINED = 12:

10-EVNTYPE	1-TW	2-TDELAY	3-T1	4-DT1
11-E1	5-T2	6-DT2	21-E2	7-T3
8-DT3	31-E3			

EVNTYPE = Event type, group of source term with unique evacuation assumptions to be partitioned separately.

TW = Warning time, time at which population receives evacuation order.

TDELAY = Delay time, time interval before population begins evacuating.

T1, T2, and T3 = Start time of the first, second, and third releases.

DT1, DT2, and DT3 = Duration of the first, second, and third releases.

E1, E2, and E3 = Energy of the first, second, and third releases.

Table 4.6-2
Distribution of Source Terms for Event Type 1
High-g Seismic Initiators

BEFORE PARTITIONING:

NUMBER OF SOURCE TERMS IN GRID = 570

	0	1	2	3	4	5
5	0	0	0	0	0	2
4	0	0	0	0	0	16
3	0	0	0	0	0	33
2	0	0	0	0	3	65
1	0	0	0	0	29	24
0	0	0	0	7	172	219

FRACTION OF ORIGINAL FREQUENCY REMAINING = 1.00000

	0	1	2	3	4	5
5	0.00000	0.00000	0.00000	0.00000	0.00000	0.00018
4	0.00000	0.00000	0.00000	0.00000	0.00000	0.04237
3	0.00000	0.00000	0.00000	0.00000	0.00000	0.08880
2	0.00000	0.00000	0.00000	0.00000	0.00021	0.06230
1	0.00000	0.00000	0.00000	0.00000	0.01032	0.17170
0	0.00000	0.00000	0.00000	0.00333	0.19300	0.42779

FRACTION OF ORIGINAL EH RISK REMAINING = 1.00000

	0	1	2	3	4	5
5	0.00000	0.00000	0.00000	0.00000	0.00000	0.00583
4	0.00000	0.00000	0.00000	0.00000	0.00000	0.39060
3	0.00000	0.00000	0.00000	0.00000	0.00000	0.43626
2	0.00000	0.00000	0.00000	0.00000	0.00017	0.10128
1	0.00000	0.00000	0.00000	0.00000	0.00419	0.06166
0	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000

FRACTION OF ORIGINAL CH RISK REMAINING = 1.00000

	0	1	2	3	4	5
5	0.00000	0.00000	0.00000	0.00000	0.00000	0.00124
4	0.00000	0.00000	0.00000	0.00000	0.00000	0.14170
3	0.00000	0.00000	0.00000	0.00000	0.00000	0.30274
2	0.00000	0.00000	0.00000	0.00000	0.00019	0.11638
1	0.00000	0.00000	0.00000	0.00000	0.00805	0.21859
0	0.00000	0.00000	0.00000	0.00000	0.00303	0.20806

Table 4.6-2 (Concluded)
Distribution of Source Terms for Event Type 1
High-g Seismic Initiators

AFTER PARTITIONING:

NUMBER OF SOURCE TERMS IN GRID = 69

	0	1	2	3	4	5
5	0	0	0	0	0	2
4	0	0	0	0	0	0
3	0	0	0	0	0	0
2	0	0	0	0	3	0
1	0	0	0	0	0	15
0	0	0	0	7	25	17

FRACTION OF ORIGINAL FREQUENCY REMAINING = 0.02188

	0	1	2	3	4	5
5	0.00000	0.00000	0.00000	0.00000	0.00000	0.00018
4	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000
3	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000
2	0.00000	0.00000	0.00000	0.00000	0.00021	0.00000
1	0.00000	0.00000	0.00000	0.00000	0.00000	0.00462
0	0.00000	0.00000	0.00000	0.00333	0.00921	0.00433

FRACTION OF ORIGINAL EH RISK REMAINING = 0.00763

	0	1	2	3	4	5
5	0.00000	0.00000	0.00000	0.00000	0.00000	0.00583
4	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000
3	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000
2	0.00000	0.00000	0.00000	0.00000	0.00017	0.00000
1	0.00000	0.00000	0.00000	0.00000	0.00000	0.00162
0	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000

FRACTION OF ORIGINAL CH RISK REMAINING = 0.00786

	0	1	2	3	4	5
5	0.00000	0.00000	0.00000	0.00000	0.00000	0.00124
4	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000
3	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000
2	0.00000	0.00000	0.00000	0.00000	0.00019	0.00000
1	0.00000	0.00000	0.00000	0.00000	0.00000	0.00540
0	0.00000	0.00000	0.00000	0.00000	0.00020	0.00082

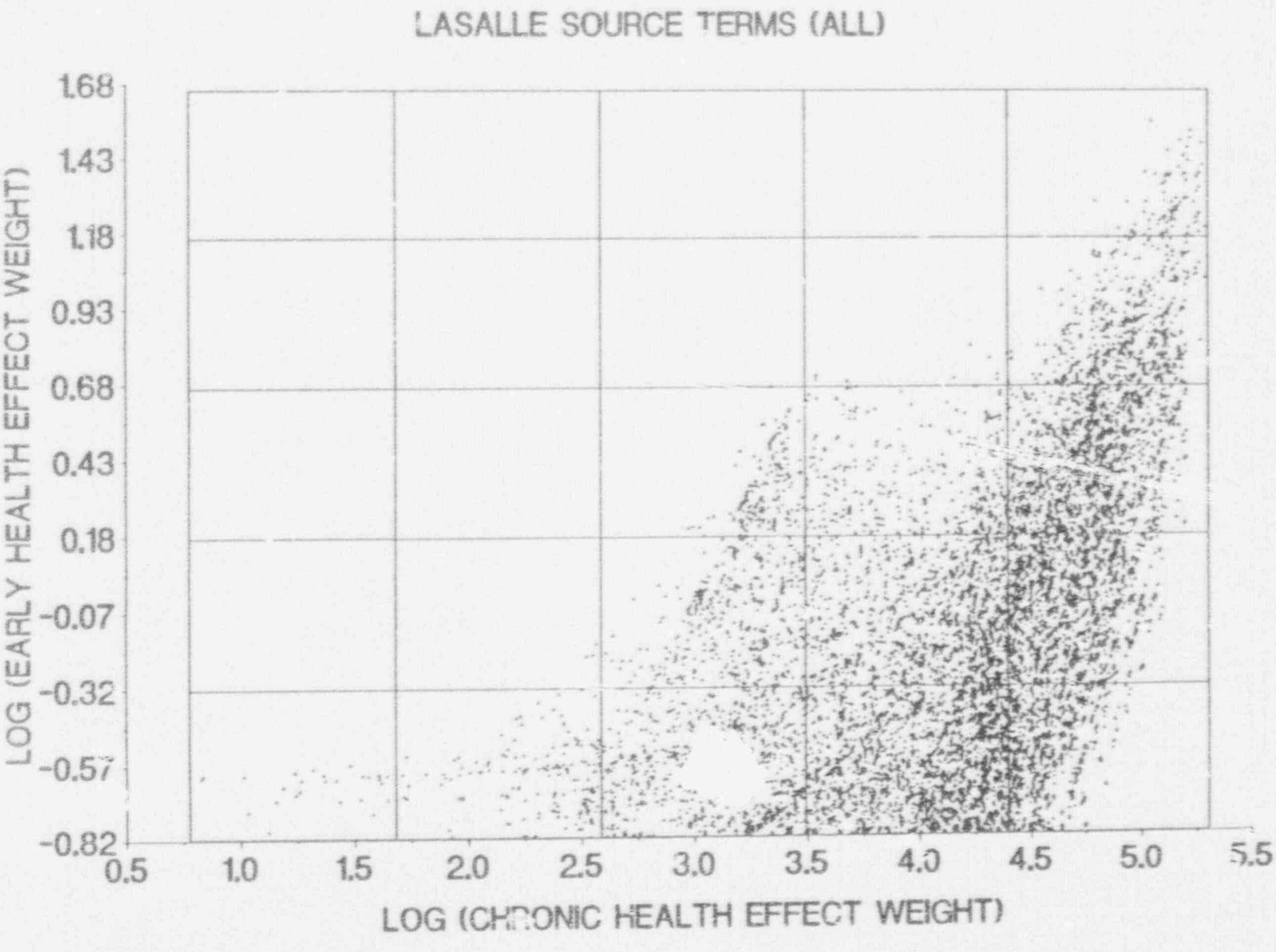

Figure 4.6-1. Distribution of Nonzero Early and Chronic Health Effect Weights for All Initiators

total frequency of all source terms of that event type that is in the each cell, the third grid shows the fraction of the total EH effect weight risk that is in each cell, and the fourth grid shows the fraction of CH effect weight risk that is in each cell. The second page of Table 4.6-2 is labeled "AFTER PARTITIONING" and shows the distribution of the source terms remaining after the partitioning is completed. As described above in Section 4.6.3.6, these remaining source terms are assigned to the nearest defined partition group of a similar event type. The FRACTION OF ORIGINAL FREQUENCY REMAINING shows, for the event type under consideration (i.e., 570 source terms for high-g seismic case), the fraction of frequency for all the source terms remaining in grid after removal of source terms in each partition group that is defined compared to the original 570 source terms. For example, before partitioning all source terms are in the grid and the fraction is 1.0; after the definition of twenty partitions only source terms contributing 2.188% of the original frequency remain in the grid. Similar fractions are shown for the EH and CH effect weights.

Similar summaries are shown in Tables 4.6-3 and 4.6-4, for the seismic low-g scenarios and the internal, fire, and flood scenarios combined. In Appendix D.4, D.5, and D.6, the actual partition output is listed for each event type. Shown in this output is the location of the group being defined, the number of source terms in the group, the accident progression bin attributes of the source terms composing the group, the fraction of total frequency in the group, the fraction of early health effect weight risk in the group, the fraction of chronic health effect weight risk in the group, and the mean, minimum, and maximum of the evacuation and source term parameters and release fractions for the group. A total of 20 high-g, 23 low-g, and 54 other partitions were defined for a grand total of 97 partitions. Table 4.6-5 lists the final mean source terms for these 97 partition groups. By looking in the appropriate appendix, one can see exactly which cell each partition group is being defined for, and how many source terms and parameters are being used in the definition.

Although not part of the source term definition, Table 4.6-5 also contains the mean frequency for the source term group, the event type, and the mean value for the difference between the time at which release starts and the time at which evacuation starts (labeled dEVAC in the table). A positive value of dEVAC indicates that the evacuation starts before the release and a negative value of dEVAC indicates that the evacuation starts after the release. The mean frequency for a source term partition group is obtained by summing the frequencies of all source terms assigned to the partition group and then dividing by the sample size (400 in this analysis).

The highest mean release fractions are associated with partition groups LAS-2, 5, 21, 22, 23, 25, 27, 32, 46, 49, 56, 61, and 73. These partition groups are all in the upper right quadrant of the grid, i.e., the (3,5), (4,5), or (5,5) cells, and represent source terms with large early and chronic health potential. Groups 2 and 5 are from the high-g seismic analysis, groups 21, 22, 23, 25, 27, and 32 are from the low-g seismic analysis, and groups 46, 49, 56, 61, and 73 are from the internal, fire,

Table 4.6-3
Distribution of Source Terms for Event Type 2
Low-g Seismic Initiators

BEFORE PARTITIONING:

NUMBER OF SOURCE TERMS IN GRID = 2972

	0	1	2	3	4	5
5	0	0	0	0	0	7
4	0	0	0	0	0	42
3	0	0	0	0	0	163
2	0	0	0	0	40	259
1	0	0	0	0	196	122
0	0	0	0	45	644	1454

FRACTION OF ORIGINAL FREQUENCY REMAINING = 1.00000

	0	1	2	3	4	5
5	0.00000	0.00000	0.00000	0.00000	0.00000	0.00034
4	0.00000	0.00000	0.00000	0.00000	0.00000	0.00803
3	0.00000	0.00000	0.00000	0.00000	0.00000	0.15466
2	0.00000	0.00000	0.00000	0.00000	0.00105	0.04100
1	0.00000	0.00000	0.00000	0.00000	0.03217	0.05525
0	0.00000	0.00000	0.00000	0.00769	0.33987	0.35995

FRACTION OF ORIGINAL EF RISK REMAINING = 1.00000

	0	1	2	3	4	5
5	0.00000	0.00000	0.00000	0.00000	0.00000	0.01643
4	0.00000	0.00000	0.00000	0.00000	0.00000	0.11119
3	0.00000	0.00000	0.00000	0.00000	0.00000	0.73743
2	0.00000	0.00000	0.00000	0.00000	0.03152	0.08707
1	0.00000	0.00000	0.00000	0.00000	0.01946	0.02690
0	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000

FRACTION OF ORIGINAL CF RISK REMAINING = 1.00000

	0	1	2	3	4	5
5	0.00000	0.00000	0.00000	0.00000	0.00000	0.00244
4	0.00000	0.00000	0.00000	0.00000	0.00000	0.02868
3	0.00000	0.00000	0.00000	0.00000	0.00000	0.65072
2	0.00000	0.00000	0.00000	0.00000	0.00092	0.07468
1	0.00000	0.00000	0.00000	0.00000	0.02502	0.07517
0	0.00000	0.00000	0.00000	0.00000	0.00836	0.13401

Table 4.6-3 (Concluded)
Distribution of Source Terms for Event Type 2
Low-g Seismic Initiators

AFTER PARTITIONING:

NUMBER OF SOURCE TERMS IN GRID = 210

	0	1	2	3	4	5
5	0	0	0	0	0	1
4	0	0	0	0	0	
3	0	0	0	0	0	26
2	0	0	0	0	40	0
1	0	0	0	0	33	0
0	0	0	0	45	56	0

FRACTION OF ORIGINAL FREQUENCY REMAINING = 0.01538

	0	1	2	3	4	5
5	0.00000	0.00000	0.00000	0.00000	0.00000	0.00001
4	0.00000	0.00000	0.00000	0.00000	0.00000	0.00007
3	0.00000	0.00000	0.00000	0.00000	0.00000	0.00164
2	0.00000	0.00000	0.00000	0.00000	0.00105	[illegible]
1	0.00000	0.00000	0.00000	0.00000	0.00120	0.00000
0	0.00000	0.00000	0.00000	0.00769	0.00372	0.00000

FRACTION OF ORIGINAL EF RISK REMAINING = 0.01389

	0	1	2	3	4	5
5	0.00000	0.00000	0.00000	0.00000	0.00000	0.00047
4	0.00000	0.00000	0.00000	0.00000	0.00000	0.00116
3	0.00000	0.00000	0.00000	0.00000	0.00000	0.00985
2	0.00000	0.00000	0.00000	0.00000	0.00152	0.00000
1	0.00000	0.00000	0.00000	0.00000	0.00089	0.00000
0	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000

FRACTION OF ORIGINAL CF RISK REMAINING = 0.00591

	0	1	2	3	4	5
5	0.00000	0.00000	0.00000	0.00000	0.00000	0.00006
4	0.00000	0.00000	0.00000	0.00000	0.00000	0.00033
3	0.00000	0.00000	0.00000	0.00000	0.00000	0.00370
2	0.00000	0.00000	0.00000	0.00000	0.00092	0.00000
1	0.00000	0.00000	0.00000	0.00000	0.00082	0.00000
0	0.00000	0.00000	0.00000	0.00000	0.00008	0.00000

Table 4.6-4
Distribution of Source Terms for Event Type 3
Internal, Fire, and Flood Initiators

BEFORE PARTITIONING:

NUMBER OF SOURCE TERMS IN GRID = 72138

	0	1	2	3	4	5
5	0	0	0	0	[illegible]	192
4	0	0	0	0	20	1521
3	0	0	0	473	811	4322
2	0	0	3	819	2788	4393
1	0	29	272	1785	4717	2626
0	0	1571	4674	3494	15246	22382

FRACTION OF ORIGINAL FREQUENCY REMAINING = 1.00000

	0	1	2	3	4	5
5	0.00000	0.00000	0.00000	0.00000	0.00000	0.00016
4	0.00000	0.00000	0.00000	0.00000	0.00002	0.01231
3	0.00000	0.00000	0.00000	0.01330	0.00389	0.02676
2	0.00000	0.00000	0.00000	0.06061	0.06716	0.04519
1	0.00000	0.00003	0.00409	0.02994	0.04032	0.01450
0	0.00000	0.01499	0.09585	0.21219	0.17709	0.18161

FRACTION OF ORIGINAL EF RISK REMAINING = 1.00000

	0	1	2	3	4	5
5	0.00000	0.00000	0.00000	0.00000	0.00000	0.00785
4	0.00000	0.00000	0.00000	0.00000	0.00026	0.22936
3	0.00000	0.00000	0.00000	0.07613	0.02128	0.19612
2	0.00000	0.00000	0.00000	0.14881	0.12283	0.14344
1	0.00000	0.00002	0.00212	0.02045	0.02490	0.01244
0	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000

FRACTION OF ORIGINAL CF RISK REMAINING = 1.00000

	0	1	2	3	4	5
5	0.00000	0.00000	0.00000	0.00000	0.00000	0.00269
4	0.00000	0.00000	0.00000	0.00000	0.00003	0.09438
3	0.00000	0.00000	0.00000	0.00279	0.00333	0.22049
2	0.00000	0.00000	0.00000	0.00911	0.12898	0.26903
1	0.00000	0.00000	0.00014	0.00475	0.06147	0.06607
0	0.00000	0.00000	0.00000	0.00006	0.00757	0.12910

Table 4.6-4 (Concluded)
Distribution of Source Terms for Event Type 3
Internal, Fire, and Flood Initiators

AFTER PARTITIONING:

NUMBER OF SOURCE TERMS IN GRID = 1581

	0	1	2	3	4	5
5	0	0	0	0	0	192
4	0	0	0	0	20	0
3	0	0	0	130	198	0
2	0	0	3	0	0	0
1	0	29	272	0	0	737
0	0	0	0	0	0	0

FRACTION OF ORIGINAL FREQUENCY REMAINING = 0.00748

	0	1	2	3	4	5
5	0.00000	0.00000	0.00000	0.00000	0.00000	0.00016
4	0.00000	0.00000	0.00000	0.00000	0.00002	0.00000
3	0.00000	0.00000	0.00000	0.00065	0.00153	0.00000
2	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000
1	0.00000	0.00003	0.00409	0.00000	0.00000	0.00100
0	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000

FRACTION OF ORIGINAL EF RISK REMAINING = 0.02184

	0	1	2	3	4	5
5	0.00000	0.00000	0.00000	0.00000	0.00000	0.00785
4	0.00000	0.00000	0.00000	0.00000	0.00026	0.00000
3	0.00000	0.00000	0.00000	0.00387	0.00695	0.00000
2	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000
1	0.00000	0.00002	0.00212	0.00000	0.00000	0.00077
0	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000

FRACTION OF ORIGINAL CF RISK REMAINING = 0.00712

	0	1	2	3	4	5
5	0.00000	0.00000	0.00000	0.00000	0.00000	0.00269
4	0.00000	0.00000	0.00000	0.00000	0.00003	0.00000
3	0.00000	0.00000	0.00000	0.00017	0.00078	0.00000
2	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000
1	0.00000	0.00000	0.00014	0.00000	0.00000	0.00331
0	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000

Table 4.6-5
Mean Source Terms Resulting from Partitioning - LaSalle

Source Term	Freq. (1/yr)	Event Type	Warn (s)	dEvac (s)	Elev (m)	Energy (w)	Start (s)	Dur (s)	Release Fractions NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
LAS-01	1.1E-09	1	3.7E+04	3.9E+04	30.	4.4E+08	7.8E+04	9.0E+02	1.5E-01	1.6E-02	1.8E-02	1.3E-04	3.2E-05	6.4E-05	3.0E-05	3.0E-05	4.6E-05
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						3.2E+07	7.9E+04	2.2E+04	8.5E-01	4.9E-01	4.9E-01	3.4E-01	2.4E-01	5.4E-07	7.3E-03	1.1E-02	1.5E-01
LAS-02	7.3E-10	1	5.2E+04	6.3E+04	30.	4.7E+08	1.2E+05	9.0E+02	9.0E-01	2.2E-01	2.3E-01	4.0E-01	3.9E-01	1.8E-03	3.5E-02	5.8E-02	3.4E-01
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						3.2E+07	1.2E+05	2.2E+04	9.8E-02	[illegible]	2.5E-02	4.4E-02	4.3E-02	1.2E-04	3.9E-03	6.2E-03	3.7E-02
LAS-03	6.7E-10	1	5.7E+04	2.1E+04	30.	4.7E+08	8.1E+04	9.0E+02	9.6E-01	1.1E-01	1.1E-01	1.2E-01	1.2E-01	1.2E-02	1.0E-02	3.7E-02	1.1E-01
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						3.2E+07	8.2E+04	2.2E+04	4.4E-02	1.2E-01	1.0E-01	1.2E-01	8.3E-02	6.4E-05	4.3E-03	[illegible]	5.9E-02
LAS-04	4.6E-10	1	5.8E+04	5.4E+04	30.	3.2E+08	1.1E+05	9.0E+02	5.3E-01	5.9E-02	6.8E-02	1.1E-01	9.3E-02	1.9E-04	3.5E-03	3.6E-03	7.1E-02
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						1.3E+07	1.2E+05	2.5E+04	4.7E-01	6.7E-02	7.7E-02	1.2E-01	1.1E-01	1.7E-04	4.1E-03	4.6E-03	8.0E-02
LAS-05	2.0E-10	1	3.8E+04	6.1E+04	30.	4.6E+08	1.0E+05	9.0E+02	9.0E-01	2.2E-01	2.3E-01	4.1E-01	4.0E-01	9.3E-04	3.6E-02	5.9E-02	3.4E-01
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						3.2E+07	1.0E+05	2.2E+04	1.0E-01	2.4E-02	2.5E-02	4.6E-02	4.5E-02	1.0E-04	4.0E-03	6.5E-03	3.8E-02
LAS-06	3.6E-09	1	3.9E+04	7.7E+04	30.	3.2E+08	1.2E+05	9.0E+02	5.0E-01	8.8E-02	9.2E-02	5.4E-02	1.4E-03	6.0E-04	3.9E-04	4.3E-04	1.1E-03
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						1.3E+07	1.2E+05	2.5E+04	5.0E-01	8.8E-02	9.2E-02	5.4E-02	1.4E-03	6.0E-04	[illegible]	4.3E-04	[illegible]
LAS-07	6.1E-10	1	3.6E+04	5.8E+04	30.	4.5E+08	9.6E+04	9.0E+02	3.7E-01	1.9E-01	2.0E-01	1.8E-01	6.2E-02	9.7E-04	2.8E-03	4.0E-03	4.0E-02
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						3.2E+07	9.7E+04	2.2E+04	1.3E-01	3.5E-02	3.6E-02	2.6E-02	8.7E-03	1.1E-04	3.6E-04	[illegible]	5.5E-03
LAS-08	7.4E-11	1	5.7E+04	2.2E+04	30.	3.2E+08	8.1E+04	9.0E+02	3.1E-01	3.0E-02	3.3E-02	1.6E-02	1.4E-02	1.9E-04	5.9E-04	5.9E-04	1.1E-02
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						1.3E+07	8.2E+04	2.5E+04	6.9E-01	2.1E-01	2.2E-01	1.9E-01	1.9E-01	8.8E-04	1.6E-02	2.9E-02	1.7E-01
LAS-09	1.5E-10	1	3.9E+04	3.7E+04	30.	[illegible]	7.8E+04	9.0E+02	2.3E-01	2.6E-02	2.8E-02	7.0E-03	5.1E-04	9.9E-04	6.0E-04	6.0E-04	9.1E-04
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	[illegible]	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						1.3E+07	7.9E+04	2.5E+04	7.7E-01	3.7E-01	3.8E-01	[illegible]	3.3E-03	1.0E-09	1.7E-04	2.4E-04	1.8E-03
LAS-10	6.7E-11	1	3.9E+04	2.2E+04	30.	4.6E+08	6.3E+04	9.0E+02	9.5E-01	1.0E-01	1.1E-01	8.9E-02	7.9E-02	1.2E-02	8.0E-03	3.5E-02	8.0E-02
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						3.1E+07	6.4E+04	2.2E+04	4.9E-02	1.5E-01	1.3E-01	1.7E-01	7.6E-02	1.4E-03	3.3E-03	4.8E-03	4.1E-02

Table 4.6-5 (Continued)
Mean Source Terms Resulting from Partitioning - LaSalle

Source Term	Freq. (1/yr)	Event Type	Warn (s)	dEvac (s)	Elev (m)	Energy (w)	Start (s)	Dur (s)	Release Fractions NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
LAS-11	1.5E-10	1	5.2E+04	3.8E+04	30.	4.6E+08	9.2E+04	9.0E+02	8.2E-01	6.6E-02	7.2E-02	1.0E-01	7.6E-02	1.1E-03	4.4E-03	7.3E-03	5.2E-02
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						3.1E+07	9.2E+04	2.2E+04	1.8E-01	1.1E-01	1.1E-01	7.3E-02	3.2E-02	3.1E-05	1.1E-03	2.0E-03	2.2E-02
LAS-12	8.3E-09	1	5.5E+04	5.0E+04	30.	4.6E+08	1.1E+05	9.0E+02	9.0E-01	6.7E-02	6.6E-02	2.4E-02	5.0E-03	8.0E-04	5.7E-04	6.2E-04	4.0E-03
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						3.2E+07	1.1E+05	2.2E+04	1.0E-01	8.2E-03	7.9E-03	3.2E-03	7.1E-04	8.6E-05	7.1E-05	8.0E-05	5.3E-04
LAS-13	2.3E-10	1	3.9E+04	7.7E+04	30.	4.7E+08	1.2E+05	9.0E+02	9.0E-01	1.6E-01	1.9E-01	1.4E-01	4.1E-03	1.6E-03	1.1E-03	1.2E-03	3.2E-03
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						3.2E+07	1.2E+05	2.2E+04	1.0E-01	2.1E-02	2.1E-02	1.6E-02	4.5E-04	1.8E-04	1.3E-04	1.3E-04	3.5E-04
LAS-14	1.4E-09	1	5.7E+04	5.1E+04	30.	3.1E+08	1.1E+05	9.0E+02	5.9E-01	1.1E-03	5.3E-04	6.5E-04	2.5E-04	1.7E-04	4.3E-05	6.6E-05	2.0E-04
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						1.3E+07	1.1E+05	2.5E+04	4.1E-01	1.9E-03	1.2E-03	1.0E-03	2.7E-04	1.7E-04	4.1E-05	4.7E-05	1.9E-04
LAS-15	2.0E-09	1	5.9E+04	4.7E+04	30.	4.7E+08	1.1E+05	9.0E+02	9.0E-01	2.1E-03	1.1E-03	1.5E-03	7.6E-04	8.8E-05	1.8E-04	2.6E-04	6.6E-04
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						3.2E+07	1.1E+05	2.2E+04	1.0E-01	3.6E-04	2.2E-04	2.3E-04	9.4E-05	9.6E-06	2.4E-05	3.1E-05	7.8E-05
LAS-16	5.4E-10	1	6.0E+04	5.4E+04	30.	3.2E+08	1.2E+05	9.0E+02	5.3E-01	1.0E-02	1.0E-02	1.2E-02	1.0E-02	2.3E-04	4.0E-04	7.5E-04	8.4E-03
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						1.3E+07	1.2E+05	2.5E+04	4.7E-01	1.1E-02	1.1E-02	1.3E-02	1.1E-02	2.2E-04	4.0E-04	7.5E-04	8.5E-03
LAS-17	9.8E-10	1	3.4E+04	8.1E+04	30.	3.2E+08	1.2E+05	9.0E+02	5.0E-01	3.8E-03	1.6E-03	1.4E-03	4.6E-05	1.2E-05	7.4E-06	8.5E-06	4.3E-05
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						1.3E+07	1.2E+05	2.5E+04	5.0E-01	3.8E-03	1.6E-03	1.4E-03	4.6E-05	1.2E-05	7.4E-06	8.5E-06	4.3E-05
LAS-18	3.0E-10	1	3.4E+04	6.9E+04	30.	4.0E+08	1.1E+05	9.0E+02	8.4E-01	1.9E-02	1.9E-02	7.3E-03	9.7E-04	7.8E-05	7.9E-05	1.2E-04	7.5E-04
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						2.4E+07	1.1E+05	2.3E+04	1.6E-01	1.3E-02	1.1E-02	5.8E-03	5.9E-04	1.0E-05	4.5E-05	7.7E-05	5.2E-04
LAS-19	2.4E-10	1	6.4E+04	2.5E+04	30	4.6E+08	9.1E+04	9.0E+02	9.0E-01	5.8E-03	7.1E-03	1.2E-02	4.9E-03	6.4E-04	6.5E-04	8.8E-04	4.2E-03
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						3.2E+07	9.2E+04	2.2E+04	9.8E-02	1.0E-03	1.3E-03	2.0E-03	5.5E-04	6.8E-05	7.1E-05	9.7E-05	4.8E-04
LAS-20	2.3E-10	1	4.2E+04	5.3E+04	30.	3.8E+08	9.7E+04	9.0E+02	7.2E-01	5.9E-02	6.0E-02	7.0E-02	4.3E-02	1.4E-03	1.9E-03	3.6E-03	2.9E-02
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						2.3E+07	9.8E+04	2.3E+04	2.8E-01	2.4E-02	2.4E-02	2.5E-02	1.6E-02	1.7E-04	8.3E-04	1.5E-03	1.2E-02

Table 4.6-5 (Continued)
Mean Source Terms Resulting from Partitioning - LaSalle

Source Term	Freq. (1/yr)	Event Type	Warn (s)	Evac (s)	Elev (m)	Energy (w)	Start (s)	Dur (s)	Release Fractions NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
LAS-21	1.5E-08	2	4.0E+04	1.8E+04	30.	4.5E+08	6.1E+04	9.0E+02	4.6E-01	3.3E-02	3.6E-02	3.0E-02	2.7E-02	3.8E-03	2.5E-03	1.2E-02	2.7E-02
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E‹00
						3.2E+07	6.2E+04	2.2E+04	5.4E-01	4.9E-01	5.3E-01	8.2E-02	2.7E-03	7.8E-08	1.4E-04	2.1E-04	1.8E-03
LAS-22	5.6E-08	2	4.0E+04	1.8E+04	30.	3.0E+08	6.1E+04	9.0E+02	3.3E-01	6.4E-04	3.9E-03	9.4E-05	4.2E-06	9.2E-07	8.1E-07	8.1E-07	4.3E-06
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						1.3E+07	6.2E+04	2.5E+04	6.7E-01	6.0E-01	6.5E-01	9.8E-02	2.4E-03	1.0E-07	1.1E-04	1.8E-04	1.6E-03
LAS-23	1.3E-08	2	3.7E+04	3.9E+04	30.	4.4E+08	7.8E+04	9.0E+02	1.5E-01	1.6E-02	1.8E-02	9.3E-04	5.9E-04	8.7E-05	6.3E-05	7.8E-05	4.5E-04
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						3.2E+07	7.9E+04	2.2E+04	8.5E-01	4.9E-01	4.8E-01	3.4E-01	2.3E-01	6.7E-06	7.4E-03	1.1E-02	1.5E-01
LAS-24	1.6E-08	2	5.5E+04	2.2E+04	30.	4.7E+08	7.9E+04	9.0E+02	8.4E-01	1.0E-01	1.1E-01	9.6E-02	8.4E-02	1.2E-02	8.1E-03	3.3E-02	8.3E-02
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						3.2E+07	8.0E+04	2.2E+04	1.6E-01	2.0E-01	1.9E-01	1.5E-01	6.7E-02	3.7E-05	3.3E-03	3.6E-03	4.4E-02
LAS-25	2.8E-09	2	3.8E+04	6.1E+04	30.	4.6E+08	1.0E+05	9.0E+02	9.0E-01	2.2E-01	2.3E-01	4.1E-01	4.0E-01	9.3E-04	3.6E-02	5.9E-02	3.4E-01
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						3.2E+07	1.0E+05	2.2E+04	1.0E-01	2.4E-02	2.5E-02	4.6E-02	4.5E-02	1.0E-04	4.0E-03	6.5E-03	3.8E-02
LAS-26	1.7E-06	2	3.2E+04	4.6E+04	30.	4.5E+08	8.1E+04	9.0E+02	8.9E-01	1.6E-01	1.6E-01	1.7E-01	9.1E-02	2.4E-04	3.1E-03	4.7E-03	5.7E-02
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						3.2E+07	8.1E+04	2.2E+04	1.1E-01	2.3E-02	2.3E-02	2.1E-02	1.1E-02	2.7E-05	3.8E-04	5.7E-04	7.0E-03
LAS-27	2.6E-09	2	5.4E+04	5.5E+04	30.	4.7E+08	1.1E+05	9.0E+02	9.0E-01	2.2E-01	2.4E-01	3.9E-01	3.8E-01	9.6E-03	3.2E-02	5.1E-02	3.2E-01
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						3.2E+07	1.1E+05	2.2E+04	1.0E-01	4.0E-02	4.3E-02	5.8E-02	4.9E-02	4.0E-04	3.6E-03	5.8E-03	3.9E-02
LAS-28	4.3E-09	2	3.6E+04	2.3E+04	30.	4.7E+08	6.1E+04	9.0E+02	9.4E-01	2.0E-02	3.0E-02	2.4E-02	8.1E-04	1.4E-03	7.3E-04	7.3E-04	8.3E-04
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						3.2E+07	6.2E+04	2.2E+04	6.3E-02	3.2E-01	2.9E-01	3.6E-01	1.6E-01	1.4E-06	5.1E-03	7.7E-03	7.8E-02
LAS-29	6.9E-09	2	4.8E+04	3.9E+04	30.	4.6E+08	9.0E+04	9.0E+02	8.0E-01	1.3E-01	1.4E-01	1.1E-01	5.5E-02	7.8E-04	1.9E-03	3.1E-03	3.2E-02
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						3.2E+07	9.1E+04	2.2E+04	2.0E-01	9.7E-02	9.9E-02	6.4E-02	3.5E-02	2.8E-05	9.6E-04	1.6E-03	2.2E-02
LAS-30	2.2E-08	2	3.9E+04	7.7E+04	30.	3.2E+08	1.2E+05	9.0E+02	5.0E-01	8.8E-02	9.2E-02	5.4E-02	1.4E-03	6.0E-04	3.9E-04	4.3E-04	1.1E-03
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						1.3E+07	1.2F 05	2.5E+04	5.0E-01	8.8E-02	9.2E-02	5.4E-02	1.4E-03	6.0E-04	3.9E-04	4.3E-04	1.1E-03

Table 4.6-5 (Continued)
Mean Source Terms Resulting from Partitioning - LaSalle

Source Term	Freq. (1/yr)	Event Type	Warn (s)	dEvac (s)	Elev (m)	Energy (w)	Start (s)	Dur (s)	Release Fractions NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
LAS-31	1.3E-08	2	5.2E+04	3.9E+04	30.	4.2E+08	9.3E+04	9.0E+02	7.9E-01	4.1E-02	4.9E-02	9.2E-02	2.3E-02	8.6E-04	1.4E-03	2.3E-03	1.7E-02
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						2.9E+07	9.4E+04	2.2E+04	2.1E-01	2.3E-02	2.6E-02	3.2E-02	2.4E-02	1.4E-04	2.0E-03	3.1E-03	2.0E-02
LAS-32	2.3E-10	2	4.6E+04	2.3E+04	30.	4.7E+08	7.1E+04	9.0E+02	6.6E-01	3.2E-01	3.3E-01	3.0E-01	2.6E-01	5.7E-02	2.5E-02	1.1E-01	2.7E-01
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						3.2E+07	7.1E+04	2.2E+04	3.4E-01	3.4E-01	3.4E-01	3.2E-01	3.4E-01	3.0E-03	3.9E-02	6.3E-02	3.2E-01
LAS-33	4.5E-09	2	5.3E+04	4.4E+04	30.	3.3E+08	9.8E+04	9.0E+02	4.8E-01	4.7E-02	5.2E-02	6.2E-02	4.7E-02	9.7E-04	2.3E-03	2.6E-03	3.6E-02
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						1.5E+07	9.9E+04	2.5E+04	5.2E-01	1.5E-01	1.6E-01	1.1E-01	6.5E-02	4.4E-04	2.6E-03	3.4E-03	4.8E-02
LAS-34	1.5E-08	2	4.8E+04	4.6E+04	30.	3.6E+08	9.6E+04	9.0E+02	5.7E-01	8.2E-02	8.8E-02	3.5E-02	1.8E-03	7.3E-04	2.6E-04	3.1E-04	1.5E-03
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						2.0E+07	9.7E+04	2.4E+04	4.3E-01	1.0E-01	1.1E-01	3.9E-02	1.8E-03	2.8E-05	9.2E-05	1.2E-04	1.5E-03
LAS-35	1.6E-07	2	5.1E+04	4.9E+04	30.	4.6E+08	1.0E+05	9.0E+02	9.0E-01	5.4E-02	5.2E-02	2.1E-02	4.8E-03	8.8E-04	6.1E-04	7.3E-04	4.0E-03
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						3.2E+07	1.0E+05	2.2E+04	1.0E-01	8.2E-03	7.6E-03	3.3E-03	9.4E-04	8.2E-05	9.1E-05	1.2E-04	7.1E-04
LAS-36	4.0E-08	2	3.1E+04	4.2E+04	30.	3.6E+08	7.4E+04	9.0E+02	9.2E-01	6.7E-03	7.0E-03	4.4E-03	2.3E-03	9.9E-05	1.2E-04	1.8E-04	1.3E-03
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						1.6E+07	7.5E+04	2.4E+04	7.9E-02	2.7E-02	2.1E-02	8.2E-03	1.2E-03	1.2E-05	9.0E-05	1.5E-04	9.3E-04
LAS-37	4.0E-08	2	5.5E+04	3.6E+04	30.	3.7E+08	9.3E+04	9.0E+02	7.2E-01	1.0E-02	9.3E-03	8.9E-03	3.8E-03	7.7E-04	4.5E-04	7.5E-04	3.3E-03
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						2.0E+07	9.4E+04	2.4E+04	2.8E-01	1.7E-02	1.6E-02	1.4E-02	3.9E-03	3.1E-04	2.8E-04	3.4E-04	3.0E-03
LAS-38	8.3E-09	2	3.1E+04	5.9E+04	30.	3.7E+08	9.3E+04	9.0E+02	6.9E-01	6.8E-02	6.8E-02	6.6E-02	4.1E-02	5.3E-06	1.3E-03	2.1E-03	2.7E-02
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						2.2E+07	9.4E+04	2.3E+04	3.1E-01	2.8E-02	2.8E-02	2.7E-02	1.6E-02	1.1E-06	[illegible].4E-04	9.2E-04	1.1E-02
LAS-39	1.1E-07	2	5.6E+04	5.8E+04	30.	[illegible].0E+08	1.2E+05	9.0E+02	5.2E-01	2.2E-03	2.1E-03	7.0E-04	9.6E-05	5.3E-05	1.9E-05	2.1E-05	8.1E-05
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						1.3E+07	1.2E+05	2.5E+04	4.8E-01	2.4E-03	2.2E-03	7.7E-04	1.0E-04	5.2E-05	2.1E-05	2.0E-05	8.4E-05
LAS-40	5.7E-08	2	5.5E+04	4.3E+04	30.	4.6E+08	1.0E+05	9.0E+02	9.0E-01	2.0E-03	1.5E-03	1.8E-03	7.0E-04	1.9E-04	4.7E-04	3.9E-04	6.3E-04
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						3.2E+07	1.0E+05	2.2E+04	9.7E-02	3.3E-04	2.4E-04	3.4E-04	2.8E-04	2.0E-05	8.0E-05	6.5E-05	1.9E-04

Table 4.6-5 (Continued)
Mean Source Terms Resulting from Partitioning - LaSalle

Source Term	Freq. (1/yr)	Event Type	Warn (s)	dEvac (s)	Elev (m)	Energy (w)	Start (s)	Dur (s)	Release Fractions NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
LAS-41	5.0E-08	2	3.4E+04	7.4E+04	30.	3.2E+08	1.1E+05	9.0E+02	5.0E-01	2.7E-03	1.2E-03	1.1E-0	1.1E-04	1.3E-05	1.5E-05	2.2E-05	1.1E-04
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						1.3E+07	1.1E+05	2.5E+04	5.0E-01	2.7E-03	1.2E-03	1.1E-03	1.1E-04	1.3E-05	1.5E-05	2.2E-05	1.1E-04
LAS-42	1.1E-08	2	4.0E+04	1.9E+04	30.	4.6E+08	6.1E+04	9.0E+02	1.0E+00	4.6E-04	2.1E-0	3.3E-04	2.5E-04	1.6E-04	6.8E-05	2.0E-04	2.5E-04
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						3.2E+07	6.2E+04	2.2E+04	6.7E-04	4.2E-04	9.1E-06	1.0E-03	4.4E-07	2.8E-09	9.6E-04	4.0E-04	2.6E-07
LAS-43	8.2E-09	2	3.4E+04	7.1E+04	30.	4.6E+08	1.1E+05	9.0E+02	9.0E-01	6.7E-03	2.4E-03	2.2E-03	5.5E-05	2.7E-05	1.1E-05	1.2E-05	4.5E-05
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						3.2E+07	1.1E+05	2.2E+04	1.0E-01	7.5E-04	2.6E-04	2.5E-04	6.1E-06	3.0E-06	1.2E-06	1.3E-08	5.0E-06
LAS-44	9.6E-08	3	2.0E+03	6.1E+04	30.	7.3E+08	6.5E+04	9.0E+02	9.0E-01	3.0E-01	7.0E-02	2.7E-01	2.9E-01	1.2E-04	2.5E-02	4.0E-02	2.4E-01
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						3.0E+08	6.6E+04	2.2E+04	1.0E-01	3.4E-02	7.8E-03	3.0E-02	3.2E-02	1.3E-05	2.8E-03	4.5E-03	2.7E-02
LAS-45	5.1E-07	3	2.0E+03	1.5E+04	30.	6.7E+08	2.0E+04	9.6E+02	9.0E-01	2.1E-01	4.3E-02	1.5E-01	1.6E-01	3.5E-04	1.2E-02	2.2E-02	1.3E-01
						3.5E+07	1.3E+04	9.0E+02	6.9E-05	4.1E-05	6.3E-05	4.0E-05	1.8E-05	1.9E-05	3.9E-06	3.9E-06	2.2E-05
						2.6E+08	2.1E+04	2.2E+04	1.0E-01	1.3E-01	4.8E-02	9.3E-02	9.9E-02	2.5E-05	7.3E-03	1.3E-02	8.0E-02
LAS-46	1.4E-06	3	1.7E+05	1.1E+04	30.	1.5E+07	1.8E+05	2.3E+04	5.8E-01	6.6E-03	4.4E-03	2.0E-03	7.4E-04	2.7E-04	7.1E-05	2.3E-04	7.9E-04
						1.4E+07	2.0E+05	9.0E+02	6.1E-02	7.5E-03	1.2E-02	1.5E-02	1.6E-02	1.5E-02	2.1E-03	2.1E-03	1.6E-02
						1.8E+07	2.0E+05	2.2E+04	3.6E-01	4.2E-01	4.2E-01	3.5E-01	2.6E-01	4.2E-04	1.5E-02	1.8E-02	1.6E-01
LAS-47	4.0E-06	3	4.7E+03	2.0E+04	30.	7.3E+08	2.7E+04	9.0E+02	9.0E-01	2.9E-01	5.1E-04	9.3E-04	4.2E-04	1.4E-07	1.9E-05	1.6E-05	3.2E-04
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						3.0E+08	2.3E+04	2.2E+04	1.0E-01	3.2E-02	5.6E-05	1.0E-04	4.7E-05	1.5E-08	2.1E-06	1.7E-06	3.5E-05
LAS-48	3.4E-06	3	2.0E+05	1.2E+04	30.	2.9E+07	2.1E+05	2.6E+04	4.0E-01	7.5E-04	4.4E-04	1.8E-04	6.4E-05	9.9E-06	4.8E-06	2.1E-05	6.6E-05
						1.3E+07	2.4E+05	9.0E+02	6.0E-02	1.3E-02	1.5E-02	7.0E-03	4.3E-03	4.5E-03	8.0E-04	8.0E-04	5.1E-03
						1.7E+07	2.4E+05	2.2E+04	5.4E-01	2.2E-01	2.2E-01	2.0E-01	2.0E-01	2.8E-03	1.8E-02	1.5E-02	1.8E-01
LAS-49	3.9E-07	3	1.3E+05	1.2E+04	30.	1.1E+07	1.4E+05	1.9E+04	6.5E-01	1.9E-02	9.8E-03	2.7E-03	1.7E-04	7.9E-05	9.2E-06	2.9E-05	2.1E-04
						1.5E+07	1.6E+05	9.0E+02	2.5E-02	1.7E-02	2.0E-02	1.7E-02	1.4E-02	1.3E-02	1.8E-03	1.9E-03	1.6E-02
						2.0E+07	1.6E+05	2.2E+04	3.2E-01	3.9E-01	4.2E-01	5.0E-01	5.2E-01	8.2E-03	4.3E-02	7.1E-02	4.4E-01
LAS-50	5.1E-06	3	1.9E+03	1.0E+04	30.	4.9E+08	1.4E+04	9.0E+02	7.8E-01	8.1E-05	8.9E-04	3.3E-05	2.3E-06	1.3E-06	2.4E-07	2.5E-07	2.7E-06
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						1.2E+08	1.5E+04	2.5E+04	2.2E-01	1.1E-01	1.1E-01	6.5E-02	8.9E-04	2.7E-07	6.0E-05	8.5E-05	8.8E-04

Table 4.6-5 (Continued)
Mean Source Terms Resulting from Partitioning - LaSalle

Source Term	Freq. (1/yr)	Event Type	Warn (s)	dEvac (s)	Elev (m)	Energy (w)	Start (s)	Dur (s)	Release Fractions NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
LAS-51	1.8E-07	3	5.0E+03	7.3E+03	30.	7.4E+08	1.4E+04	9.0E+02	6.6E-01	[illegible]6E-02	[illegible]	7.[illegible]E-03	2.3E-03	2.7E-03	8.6E-04	9.3E-04	2.8E-03
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	[illegible]	[illegible]	0.0E+00	0.[illegible]E+00	0.0E+00	0.0E+00	0.0E+00	[illegible].0E+00
						3.0E+08	1.5E+04	2.2E+04	3.4E-01	[illegible]	[illegible]-01	1.0E-01	4.0E-02	8.2E-05	2.3E-03	2.1E-03	2.9E-02
LAS-52	2.8E-07	3	3.3E+04	2.6E+04	30.	4.4E+08	6.1E+04	9.0E+02	3.0E-01	1.9E-[illegible]	2.2E-02	6.9E-03	2.7E-03	2.3E-03	6.6E-04	9.1E-04	3.3E-03
						0.0E+00	0.0E+00	0.0E+00	3.7E-04	3.2E-04	4.5E-04	2.0E-04	6.1E-05	7.8E-05	1.9E-05	1.9E-05	7.9E-05
						3.1E+07	6.2E+04	2.2E+04	7.0E-01	4.0E-01	4.0E-01	2.7E-01	1.7E-01	1.4E-05	4.6E-03	7.7E-03	9.3E-02
LAS-53	4.8E-07	3	5.5E+03	1.4E+04	30.	7.3E+08	2.2E+04	9.0E+02	9.0E-01	4.6E-0[illegible]	1.4E-03	1.6E-03	1.0E-03	6.8E-05	1.2E-04	2.4E-04	1.0E-03
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						3.0E+08	2.2E+04	2.2E+04	1.0E-01	5.1E-02	1.5E-04	1.8E-04	1.2E-04	9.7E-06	1.3E-05	2.6E-05	1.1E-04
LAS-54	3.1E-0[illegible]	3	5.6E+03	6.7E+03	30	4.9E+06	1.4E+04	9.0E+02	7.2E-01	6.1E-03	6.0E-03	[illegible].8E-03	1.2E-04	1.2E-04	3.3E-05	3.5E-05	1.[illegible]E-04
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						1.2E+08	1.5E+04	2.5E+04	2.8E-01	2.7E-01	1.5E-01	1.2E-01	7.2E-02	1.5E-[illegible]	5.1E-03	5.3E-03	[illegible].4E-02
LAS-55	1.4E-06	3	4.8E+03	[illegible].5E+04	30.	[illegible].3E+08	2.2E+04	9.0E+02	9.0E-01	2.2E-01	2.7E-03	4.0E-03	3.1E-03	9.9E-05	1.9E-04	3.6E-04	2.4E-03
						[illegible].0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						3.0E+08	2.2E+04	2.2E+04	1.0E-01	2.5E-02	2.9E-04	4.5E-04	3.4E-04	1.1E-05	2.1E-05	4.0E-05	2.7E-04
LAS-56	9.9E-08	3	4.7E+03	7.6E+03	30.	7.4E+08	1.4E+04	9.0E+02	7.0E-01	9.4E-02	1.0E-01	2.8E-02	1.7E-02	1.6E-02	3.1E-03	3.8E-03	1.9E-02
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						3.0E+08	1.5E+04	2.2E+04	3.0E-01	3.9E-01	3.1E-01	1.8E-01	1.1E-01	6.5E-05	6.2E-03	9.5E-03	8.4E-02
LAS-57	4.6E-07	3	3.7E+03	1.8E+04	30.	7.3E+08	2.2E+04	9.0E+02	9.0E-01	3.9E-01	4.4E-05	1.7E-05	2.5E-07	2.6E-08	2.0E-08	3.3E-08	2.2E-07
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						3.0E+08	2.3E+04	2.2E+04	1.0E-01	4.4E-02	4.9E-0[illegible]	1.9E-06	2.8E-08	2.9E-09	2.2E-09	3.7E-09	2.4E-08
LAS-58	2.2E-07	3	4.5E+04	4.1E+04	30.	4.6E+08	8.8E+04	9.0E+02	8.5E-01	5.8E-02	6.4E-02	1.5E-01	1.2E-01	5.[illegible]E-03	1.0E-02	1.6E-02	9.8E-02
						0.0E+00	0.0E+00	0.0E+00	6.5E-05	4.1E-05	6.1E-05	1.5E-06	1.9E-06	3.[illegible]E-06	6.5E-07	6.2E-07	2.3E-06
						3.2E+07	8.9E+04	2.2E+04	1.5E-01	2.4E-01	2.0E-01	1.8E-01	5.7E-02	2.8E-04	3.1E-03	4.7E-03	4.0E-02
LAS-59	5.7E-07	3	4.7E+03	7.6E+03	30.	7.3E+08	1.4E+04	9.0E+02	9.6E-01	4.2E-03	4.6E-03	2.6E-03	2.2E-03	1.8E-03	3.9E-04	5.5E-04	2.2E-03
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						3.0E+08	1.5E+04	2.2E+04	3.6E-02	6.7E-02	4.6E-02	4.4E-02	4.2E-02	4.1E-05	4.4E-03	6.9E-03	3.5E-02
LAS-60	5.6E-07	3	9.2E+04	8.3E+03	30.	9.4E+06	1.0E+05	1.5E+04	7.1E-01	1.5E-02	1.3E-02	9.3E-03	6.5E-03	8.5E-04	4.8E-04	2.4E-03	6.5E-03
						1.7E+07	1.2E+05	9.0E+02	5.6E-03	2.4E-03	3.9E-03	9.0E-04	4.3E-04	4.4E-04	9.7E-05	1.0E-04	5.0E-04
						2.2E+07	1.2E+05	2.2E+04	2.9E-01	2.4E-01	2.5E-01	1.6E-01	8.2E-02	2.6E-04	5.8E-03	5.7E-03	6.0E-02

Table 4.6-5 (Continued)
Mean Source Terms Resulting from Partitioning - LaSalle

Source Term	Freq. (1/yr)	Event Type	Warn (s)	dEvac (s)	Elev (m)	Energy (w)	Start (s)	Dur (s)	Release Fractions NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
LAS-61	1.3E-07	3	4.6E+03	7.7E+03	30.	4.9E+08	1.4E+04	9.0E+02	8.4E-01	1.7E-03	3.3E-03	1.1E-03	7.6E-04	5.6E-04	8.3E-05	1.0E-04	8.5E-04
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						1.2E+08	1.5E+04	2.5E+04	1.6E-01	4.8E-01	1.7E-01	1.3E-01	1.7E-01	5.5E-05	1.4E-02	2.1E-02	1.3E-01
LAS-62	6.4E-07	3	5.1E+03	1.3E+04	30.	6.0E+08	2.0E+04	9.4E+02	8.9E-01	7.3E-02	2.0E-02	2.4E-02	1.2E-02	4.2E-04	7.3E-04	1.2E-03	8.8E-03
						3.5E+07	1.5E+04	9.0E+02	4.4E-05	1.4E-05	2.1E-05	6.6E-06	2.1E-06	2.5E-06	7.6E-07	7.6E-07	2.7E-06
						2.1E+08	2.1E+04	2.3E+04	1.1E-01	1.0E-01	1.6E-02	1.6E-02	1.4E-02	5.2E-05	9.2E-04	1.3E-03	1.0E-02
LAS-63	2.1E-06	3	5.0E+03	1.1E+04	30.	6.4E+08	1.8E+04	9.1E+02	6.4E-01	2.5E-02	1.7E-02	1.3E-02	6.2E-03	2.5E-04	3.0E-04	4.9E-04	3.5E-03
						3.5E+07	1.6E+04	9.0E+02	2.5E-05	1.0E-05	1.3E-06	5.6E-07	4.3E-08	9.7E-08	6.2E-08	6.1E-08	8.8E-08
						2.4E+08	1.9E+04	2.2E+04	3.6E-01	3.8E-02	3.1E-02	9.1E-03	2.9E-03	4.4E-05	1.3E-04	2.1E-04	1.9E-03
LAS-64	2.2E-07	3	6.9E+03	1.6E+04	30.	7.3E+08	2.5E+04	9.0E+02	9.0E-01	4.3E-01	1.1E-04	1.1E-04	2.4E-05	1.5E-08	8.1E-07	1.4E-06	1.6E-05
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						3.0E+08	2.6E+04	2.1E+04	1.0E-01	4.7E-02	1.2E-05	1.3E-05	2.6E-06	1.7E-07	8.9E-08	1.5E-07	1.7E-06
LAS-65	1.9E-07	3	5.0E+03	1.4E+04	30.	7.3E+08	2.2E+04	9.0E+02	9.0E-01	8.4E-02	8.2E-02	1.5E-01	1.4E-01	2.6E-04	9.8E-03	1.9E-02	1.1E-01
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						3.0E+08	2.2E+04	2.2E+04	1.0E-01	9.3E-03	9.1E-03	1.7E-02	1.5E-02	2.9E-05	1.1E-03	2.1E-03	1.2E-02
LAS-66	2.6E-07	3	4.7E+03	1.0E+04	30.	6.1E+08	1.7E+04	1.2E+03	7.5E-01	1.4E-01	1.6E-02	2.0E-02	1.9E-02	3.6E-03	1.9E-03	3.4E-03	1.8E-02
						3.5E+07	1.3E+04	9.0E+02	2.2E-04	5.2E-05	1.1E-04	1.3E-05	5.7E-06	7.2E-06	2.0E-06	2.1E-06	7.0E-06
						2.2E+08	1.6E+04	2.1E+04	2.5E-01	2.5E-01	1.6E-02	1.8E-02	1.7E-02	1.9E-04	1.6E-03	2.6E-03	1.4E-02
LAS-67	1.5E-06	3	4.7E+03	1.5E+04	30.	7.3E+08	2.2E+04	9.0E+02	9.7E-01	8.9E-02	3.7E-03	5.8E-03	4.6E-03	4.8E-04	4.5E-04	1.3E-03	4.0E-03
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						3.0E+08	2.3E+04	1.9E+04	1.0E-01	9.8E-03	4.1E-04	6.5E-04	5.1E-04	5.4E-05	5.0E-05	1.4E-04	4.4E-04
LAS-68	8.4E-07	3	3.3E+03	1.8E+04	30.	7.0E+08	2.3E+04	1.0E+03	8.0E-01	1.9E-01	5.8E-04	4.5E-04	3.8E-04	4.0E-05	4.2E-05	7.1E-05	3.4E-04
						3.5E+07	1.3E+04	9.0E+02	1.9E-04	4.9E-06	8.3E-06	5.0E-06	1.8E-06	3.4E-06	1.5E-06	1.5E-06	2.6E-06
						2.8E+08	2.4E+04	1.8E+04	1.7E-01	9.8E-02	4.9E-04	4.0E-04	2.8E-04	4.5E-06	1.5E-05	2.7E-05	2.0E-04
LAS-69	7.0E-07	3	9.3E+04	2.1E+04	30.	3.2E+08	1.1E+05	1.0E+04	8.1E-01	4.8E-02	4.9E-02	4.4E-02	3.0E-02	1.7E-03	2.6E-03	6.1E-03	2.8E-02
						1.4E+07	2.0E+05	9.0E+02	5.8E-03	3.4E-03	4.8E-03	8.5E-04	3.5E-04	4.1E-04	1.2E-04	1.2E-04	4.2E-04
						8.2E+07	1.2E+05	2.2E+04	1.8E-01	1.2E-01	1.1E-01	8.6E-02	8.6E-02	6.6E-05	6.1E-03	5.8E-03	6.4E-02
LAS-70	3.5E-07	3	4.3E+03	9.7E+03	30.	6.0E+08	1.7E+04	9.0E+02	7.4E-01	1.1E-01	1.1E-03	6.9E-04	3.1E-04	4.7E-05	1.6E-05	5.1E-05	3.1E-04
						3.5E+07	1.3E+04	9.0E+02	1.9E-08	8.4E-10	1.9E-09	1.2E-09	9.1E-11	1.9E-10	1.2E-10	1.2E-10	1.9E-10
						2.1E+08	1.8E+04	2.1E+04	2.5E-01	4.0E-01	5.3E-04	5.2E-04	1.6E-04	5.3E-06	1.4E-05	1.4E-05	1.5E-04

Table 4.6-5 (Continued)
Mean Source Terms Resulting from Partitioning - LaSalle

Source Term	Freq. (1/yr)	Event Type	Warn (s)	dEvac (s)	Elev (m)	Energy (w)	Start (s)	Dur (s)	Release Fractions NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
LAS-71	1.5E-06	3	1.6E+05	9.0E+03	30.	2.2E+07	1.7E+05	2.2E+04	4.7E-01	8.8E-03	7.6E-03	4.5E-03	3.0E-03	4.3E-04	2.8E-04	1.3E-03	3.0E-03
						1.4E+07	2.0E+05	9.0E+02	7.9E-02	1.2E-02	1.4E-02	3.5E-03	1.6E-03	1.6E-03	4.0E-04	1.3E-04	1.8E-03
						1.8E+07	2.0E+05	2.2E+04	4.5E-01	1.9E-01	2.1E-01	1.1E-01	4.4E-02	3.2E-05	1.7E-03	2.1E-03	2.9E-02
LAS-72	2.1E-06	3	1.5E+05	1.4E+04	30.	1.4E+08	1.8E+05	2.0E+04	7.5E-01	1.2E-02	9.5E-03	8.7E-03	4.5E-03	4.6E-04	2.8E-04	4.3E-04	3.2E-03
						1.3E+07	2.2E+05	9.0E+02	1.8E-02	4.9E-03	6.4E-03	5.4E-04	4.1E-04	4.7E-04	1.5E-04	1.7E-04	4.2E-04
						4.7E+07	1.8E+05	2.2E+04	2.3E-01	6.1E-02	6.4E-02	5.8E-02	6.8E-02	9.5E-04	5.9E-03	4.3E-03	5.6E-02
LAS-73	5.3E-08	3	1.7E+04	2.[illegible]	30.	6.4E+08	4.4E+04	9.0E+02	8.4E-01	1.4E-01	1.5E-01	3.7E-01	3.5E-01	1.2E-02	1.3E-02	7.0E-02	2.6E-01
						3.5E+07	1.4E+04	9.0E+02	5.2E-04	5.3E-04	8.4E-04	4.3E-04	3.1E-04	3.0E-04	4.4E-05	4.4E-05	3.6E-04
						2.1E+08	4.5E+04	2.2E+04	1.6E-01	6.2E-02	6.7E-02	9.4E-02	8.6E-02	3.0E-03	5.4E-03	9.6E-03	7.1E-02
LAS-74	1.6E-07	3	3.2E+03	3.1E+04	30.	6.8E+08	3.6E+04	1.2E+03	6.3E-01	7.9E-02	6.0E-02	1.2E-01	1.1E-01	2.2E-03	8.3E-03	1.3E-02	8.7E-02
						3.5E+07	1.5E+04	9.0E+02	6.5E-04	6.5E-04	9.5E-04	8.6E-04	3.4E-04	3.5E-04	7.7E-05	7.7E-05	4.2E-04
						2.7E+08	3.7E+04	2.2E+04	3.7E-01	9.7E-02	1.0E-01	8.3E-02	6.3E-02	1.1E-04	2.6E-03	4.5E-03	4.6E-02
LAS-75	5.1E-07	3	8.8E+03	1.2E+04	30.	6.4E+08	2.3E+04	2.0E+03	8.3E-01	1.6E-02	1.4E-02	1.6E-02	1.2E-02	3.6E-04	6.4E-04	9.3E-04	8.3E-03
						2.6E+07	7.0E+04	9.0E+02	2.6E-04	1.6E-04	4.4E-04	3.7E-04	1.1E-03	1.1E-03	2.4E-04	3.1E-04	1.1E-03
						2.6E+08	2.5E+04	2.1E+04	1.7E-01	8.5E-02	7.6E-02	4.8E-02	9.9E-03	2.5E-04	9.7E-04	1.2E-03	7.8E-03
LAS-76	8.6E-06	3	2.0E+05	8.6E+03	30.	2.9E+07	2.1E+05	2.6E+04	6.3E-01	3.0E-03	2.3E-03	1.1E-03	3.8E-04	6.5E-05	2.5E-05	6.7E-05	3.9E-04
						1.3E+07	2.4E+05	9.0E+02	4.9E-02	2.9E-03	2.9E-03	2.3E-03	8.2E-04	1.6E-03	6.4E-04	5.8E-04	1.2E-03
						1.7E+07	2.4E+05	2.2E+04	3.2E-01	3.3E-02	2.7E-02	1.6E-02	7.3E-03	7.0E-06	4.1E-04	6.4E-04	4.6E-03
LAS-77	3.3E-06	3	2.0E+05	7.8E+03	30.	7.2E+06	2.1E+05	2.6E+04	7.7E-01	1.3E-03	9.4E-04	3.6E-04	6.9E-05	1.6E-05	6.6E-06	2.3E-05	7.5E-05
						1.3E+07	2.4E+05	9.0E+02	7.4E-03	1.3E-03	2.3E-03	4.9E-04	5.8E-05	1.1E-04	5.8E-05	5.8E-05	9.5E-05
						1.7E+07	2.4E+05	2.4E+04	2.2E-01	5.8E-02	3.8E-02	3.2E-02	2.0E-02	1.4E-04	8.1E-04	1.4E-03	1.2E-02
LAS-78	1.5E-06	3	1.1E+05	1.2E+04	30.	4.8E+07	1.2E+05	1.6E+04	7.7E-01	5.9E-03	4.6E-03	2.3E-03	7.7E-04	4.5E-04	9.7E-05	1.6E-04	8.3E-04
						1.6E+07	1.3E+05	9.0E+02	1.2E-02	1.9E-03	3.0E-03	1.1E-03	7.5E-04	8.3E-04	2.0E-04	2.2E-04	8.4E-04
						2.2E+07	1.4E+05	2.0E+04	1.7E-01	5.1E-02	4.5E-02	2.5E-02	9.3E-03	6.2E-06	3.6E-04	5.7E-04	5.5E-03
LAS-79	2.5E-06	3	4.9E+03	2.0E+04	30.	6.4E+08	2.7E+04	9.0E+02	6.5E-01	1.2E-02	6.4E-03	6.2E-03	2.9E-03	8.9E-05	1.8E-04	2.8E-04	2.2E-03
						3.5E+07	1.6E+04	9.0E+02	2.2E-05	2.0E-08	2.9E-08	6.5E-10	3.1E-10	3.1E-10	9.5E-11	9.5E-11	3.5E-10
						2.3E+08	2.8E+04	2.2E+04	3.4E-01	1.8E-02	1.6E-02	4.0E-03	1.1E-03	1.5E-05	7.3E-05	1.1E-04	8.7E-04
LAS-80	2.8E-06	3	2.8E+04	3.1E+04	30.	5.0E+08	6.1E+04	9.1E+02	7.6E-01	1.7E-02	1.2E-02	1.1E-02	4.2E-03	2.7E-04	2.6E-04	4.1E-04	2.9E-03
						3.5E+07	1.4E+04	9.0E+02	3.8E-04	1.3E-04	2.2E-04	6.8E-05	3.2E-05	3.3E-05	7.0E-06	7.0E-06	3.8E-05
						1.0E+08	6.2E+04	2.2E+04	2.3E-01	1.5E-02	1.2E-02	6.7E-03	1.8E-03	4.1E-05	8.7E-05	1.4E-04	1.3E-03

Table 4.6-5 (Continued)
Mean Source Terms Resulting from Partitioning - LaSalle

Source Term	Freq. (1/yr)	Event Type	Warn (s)	dEvac (s)	Elev (m)	Energy (w)	Start (s)	Dur (s)	Release Fractions NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
LAS-81	1.6E-05	3	2.0E+03	1.0E+04	30.	0.0E+00	1.4E+04	9.0E+02	2.5E-03	8.9E-05	3.5E-09	1.4E-09	8.6E-11	3.2E-13	5.7E-12	9.1E-12	7.4E-11
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						0.0E+00	1.5E+04	2.5E+04	2.5E-03	8.9E-05	3.5E-09	1.4E-09	8.6E-11	3.2E-13	5.7E-12	9.1E-12	7.4E-11
LAS-82	3.8E-06	3	2.1E+03	6.1E+04	30.	7.2E+08	6.3E+04	9.0E+02	8.8E-01	2.8E-02	5.1E-04	4.9E-04	5.9E-05	8.4E-07	3.1E-06	4.9E-06	4.2E-05
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						3.0E+08	6.6E+04	2.2E+04	1.1E-01	3.4E-03	6.7E-05	7.0E-05	9.6E-06	1.7E-07	6.6E-07	1.1E-06	7.4E-06
LAS-83	2.8E-06	3	5.0E+03	1.4E+04	30.	7.3E+08	2.2E+04	9.0E+02	9.0E-01	1.6E-02	1.7E-03	1.8E-03	5.0E-04	3.5E-05	3.8E-05	6.9E-05	3.7E-04
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						3.0E+08	2.2E+04	2.2E+04	1.0E-01	1.7E-03	1.9E-04	1.8E-04	5.5E-05	3.9E-06	4.3E-05	7.7E-06	4.1E-05
LAS-84	3.5E-06	3	2.6E+03	1.4E+04	30.	7.0E+08	1.9E+04	1.0E+03	8.4E-01	4.0E-03	6.5E-04	5.9E-04	9.0E-05	5.6E-06	5.2E-06	1.2E-05	5.9E-05
						3.5E+07	1.3E+04	9.0E+02	1.2E-05	3.7E-07	1.1E-06	4.3E-08	1.7E-08	1.7E-08	4.6E-09	5.7E-09	1.8E-08
						2.9E+08	2.0E+04	1.9E+04	1.1E-01	9.0E-03	6.5E-04	4.5E-04	1.6E-05	5.6E-07	9.9E-07	1.9E-06	1.1E-05
LAS-85	5.5E-06	3	4.1E+03	8.2E+03	30.	0.0E+00	1.4E+04	9.0E+02	1.4E-03	3.6E-05	8.2E-10	3.3E-10	4.7E-11	8.1E-12	3.7E-12	1.7E-11	5.3E-11
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						0.0E+00	1.5E+04	1.1E+04	1.4E-03	3.6E-05	8.2E-10	3.3E-10	4.7E-11	8.1E-12	3.7E-12	1.7E-11	5.3E-11
LAS-86	3.5E-06	3	5.0E+03	2.2E+04	30.	6.8E+08	3.0E+04	9.0E+02	7.7E-01	1.0E-02	8.1E-04	7.0E-04	2.5E-04	3.2E-05	2.4E-05	4.3E-05	2.2E-04
						3.5E+07	1.6E+04	9.0E+02	4.9E-08	6.3E-10	6.0E-10	9.7E-10	8.3E-10	7.7E-10	9.7E-11	9.7E-11	9.5E-10
						2.0E+08	3.0E+04	1.6E+04	1.5E-01	3.2E-03	6.3E-04	3.0E-04	1.2E-04	6.7E-06	1.1E-05	1.6E-05	8.8E-05
LAS-87	2.3E-06	3	5.1E+03	1.4E+04	30.	7.3E+08	2.2E+04	9.0E+02	5.1E-01	5.8E-04	1.6E-05	1.1E-05	4.7E-07	2.6E-07	2.9E-08	7.6E-08	6.3E-07
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						3.0E+08	2.3E+04	7.2E+03	5.7E-02	6.4E-05	1.8E-06	1.2E-06	5.2E-08	2.9E-08	3.3E-09	8.5E-09	7.0E-08
LAS-88	2.5E-06	3	4.0E+04	3.1E+04	30.	4.5E+08	7.2E+04	9.0E+02	9.0E-01	1.3E-02	3.6E-03	3.1E-03	3.7E-04	6.3E-05	5.3E-05	6.6E-05	2.7E-04
						0.0E+00	0.0E+00	0.0E+00	2.6E-06	9.9E-07	2.8E-06	1.3E-06	1.2E-07	2.6E-07	1.7E-07	1.6E-07	2.5E-07
						3.2E+07	7.3E+04	2.2E+04	1.0E-01	1.6E-03	4.1E-04	3.5E-04	4.4E-05	6.7E-06	6.1E-06	7.5E-06	3.2E-05
LAS-89	2.2E-06	3	4.7E+03	7.6E+03	30.	0.0E+00	1.4E+04	9.0E+02	2.5E-03	5.9E-06	9.7E-09	6.1E-09	3.3E-09	2.8E-11	2.1E-10	3.4E-10	2.6E-09
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						0.0E+00	1.5E+04	2.5E+04	2.5E-03	5.9E-06	9.7E-09	6.1E-09	3.3E-09	2.8E-11	2.1E-10	3.4E-10	2.6E-09
LAS-90	1.9E-06	3	4.9E+03	7.4E+03	30.	0.0E+00	1.4E+04	9.0E+02	2.5E-03	4.3E-04	1.8E-08	2.3E-08	2.8E-08	4.9E-10	2.1E-09	3.2E-09	2.3E-08
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						0.0E+00	1.5E+04	1.8E+04	2.5E-03	4.3E-04	1.8E-08	2.3E-08	2.8E-08	4.9E-10	2.1E-09	3.2E-09	2.2E-08

Table 4.6-5 (Concluded)
Mean Source Terms Resulting from Partitioning - LaSalle

Source Term	Freq. (1/yr)	Event Type	Warn (s)	dEvac (s)	Elev (m)	Energy (w)	Start (s)	Dur (s)	Release Fractions								
									NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
LAS-91	1.1E-06	3	4.1E+04	1.8E+04	30.	0.0E+00	6.1E+04	9.0E+02	1.8E-03	2.2E-05	7.5E-09	3.8E-09	8.8E-10	2.1E-10	7.5E-11	3.3E-10	9.6E-10
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						0.0E+00	6.2E+04	1.1E+04	1.8E-03	2.2E-05	7.5E-09	3.8E-09	8.8E-10	2.1E-17	7.5E-11	3.3E-10	9.6E-10
LAS-92	1.2E-06	3	2.0E+05	9.2E+03	30.	2.9E+07	2.1E+05	2.6E+04	8.5E-01	6.4E-04	3.3E-04	7.0E-05	3.2E-05	4.8E-06	2.6E-06	1.0E-05	3.3E-05
						1.3E+07	2.4E+05	9.0E+02	1.6E-02	4.3E-04	7.0E-04	1.9E-04	9.0E-05	1.3E-04	4.1E-05	3.7E-05	1.1E-04
						1.7E+07	2.4E+05	2.0E+04	4.8E-02	7.0E-03	1.5E-03	1.1E-03	5.6E-04	2.1E-06	1.4E-04	5.7E-05	3.7E-04
LAS-93	1.8E-06	3	2.3E+04	1.8E+04	30.	5.4E+08	4.3E+04	2.6E+03	5.5E-01	2.1E-04	4.0E-05	3.7E-05	6.6E-06	9.1E-07	3.2E-07	5.8E-07	4.0E-06
						1.4E+07	2.3E+05	9.0E+02	2.9E-03	3.3E-06	4.6E-06	4.3E-07	4.8E-08	8.6E-08	5.0E-08	5.0E-08	6.7E-08
						1.9E+08	4.6E+04	1.5E+04	9.4E-02	2.2E-04	4.4E-06	9.7E-06	1.4E-06	1.6E-07	7.1E-08	1.1E-07	8.4E-07
LAS-94	2.0E-06	3	4.7E+03	1.2E+04	30.	6.3E+08	1.8E+04	1.1E+03	8.3E-01	4.4E-02	3.0E-03	2.0E-03	4.6E-04	5.8E-05	2.9E-05	5.2E-05	3.3E-04
						3.5E+07	1.3E+04	9.0E+02	4.9E-05	5.0E-06	1.5E-05	4.7E-06	2.8E-06	3.9E-06	1.2E-06	1.0E-06	3.4E-05
						2.3E+08	1.9E+04	2.0E+04	1.4E-01	3.5E-02	2.6E-03	2.2E-03	1.5E-03	6.1E-06	1.1E-04	1.9E-04	1.0E-03
LAS-95	1.5E-06	3	1.9E+04	1.2E+04	30.	0.0E+00	3.4E+04	9.0E+02	7.5E-04	9.5E-08	7.8E-10	3.3E-10	5.4E-11	2.2E-11	8.0E-12	1.3E-11	5.8E-11
						0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						0.0E+00	3.5E+04	1.1E+04	7.5E-04	9.5E-08	7.8E-10	3.3E-10	5.4E-11	2.2E-11	8.0E-12	1.3E-11	5.8E-11
LAS-96	1.1E-06	3	2.4E+04	2.0E+04	30.	4.7E+07	4.7E+04	9.0E+02	6.5E-03	1.6E-05	2.2E-08	3.0E-08	3.6E-09	2.3E-10	3.0E-10	2.?E-10	3.4E-09
						3.5E+07	1.3E+04	9.0E+02	4.7E-09	9.5E-13	1.8E-11	3.9E-13	1.4E-14	0.0E+00	0.0E+00	0.0E+00	1.4E-14
						1.9E+07	4.7E+04	1.2E+04	2.3E-03	1.5E-05	9.0E-09	7.3E-09	3.3E-09	2.3E-10	2.9E-10	2.7E-10	2.9E-03
LAS-97	1.0E-06	3	7.9E+04	2.7E+04	30.	2.7E+08	1.1E+05	6.7E+03	5.9E-01	2.9E-03	8.0E-04	6.4E-04	2.1E-04	1.0E-04	2.4E-05	4.8E-05	2.1E-04
						1.4E+07	1.9E+05	9.0E+02	4.2E-03	3.2E-04	4.6E-04	3.7E-05	3.2E-05	5.0E-05	1.1E-05	1.1E-05	3.7E-05
						2.0E+07	1.2E+05	1.9E+04	2.1E-01	7.0E-07	1.5E-03	8.7E-04	3.4E-04	5.7E-05	5.6E-05	5.7E-05	2.7E-04

and flood analysis. All of these partition groups appear as one of the top five of the groups contributing to the release fractions for one of the release classes i.e., NG, I, Cs, Te, Sr, Ru, La, Ce, or Ba. It is not possible to pick one group that has the largest release fractions for all the classes at once; however, 21 and 22 are largest for I and Cs, 5 and 40 are largest for Te and Sr, and [illegible] and 49 and largest for Ru, La, Ce, and Ba.

For the 20 high-g seismic partition groups, 3 groups were defined using 1 parameter, 1 using 5 parameters, and 16 using all 12 parameters. Eleven of the partition groups were formed based on EH risk, 2 on CH risk, and 7 on frequency. For the 23 low-g seismic partition groups, 3 groups were defined using 1 parameter, 2 groups using 3 parameters, 2 groups using 5 parameters, and 16 groups using all 12 parameters. Fourteen partition groups were formed based on EH risk, 4 on CH risk, and 5 on frequency. For the 54 internal , fire, and flood partition groups, 12 groups were defined using 1 parameter, 9 groups using 3 parameters, 3 groups using 4 parameters, 4 groups using 6 parameters, 3 groups using 10 parameters, 2 groups using 11 parameters, and 21 groups using all 12 parameters. Thirty-two partition groups were formed on EH risk, 5 on CH risk, and 17 on frequency.

By examining the partition output in Appendix D, one can determine the homogeneity of the groups both in terms of the parameter variability and in terms of the accident progression bin attributes of the source terms that make up the group definition. In many cases, even though only a few parameters were used to define the group, all of the parameters actually have a small variation for that group. This comes about because, even though the source term parameters could not be grouped into one of the parameter subranges, they most likely fell into adjacent subranges.

In general, we feel that this new partition process results in a substantial improvement in the accuracy of the representation of source term characteristics for input into the consequence calculation and a substantial improvement over the initial process used in NUREG-1150. However, there are still some limitations in the current process such as;

1. Due to resource limitations the codes are still not easy to run and require several steps. The COMBIN code should be made part of the PARTITION code with options to specify which combinations of event types are to be partitioned separately and the STER code should also be made part of PARTITION with options to define the evacuation assumptions for each event type so that the MACCS input files can be created directly from the partition run. These changes would greatly simplify that interface between the partitioning process and the consequence calculation make the process much easier to perform.

2. The parameters to be used in the partition process need more evaluation before a final group can be selected. The purpose of

these parameters is to make the source term groups defined in the partition process homogeneous enough so that the consequences calculated by MACCS reasonably approximate the actual results that would be expected if a MACCS calculation was performed for every source term. Since the partitioning is done on potential effects, it is important that any parameters that will impact the conversion of this potential into actuality in the MACCS calculation should be as homogenous as possible.

It is clear that dEVAC, which is the first release time minus the warning time minus the delay time (i.e., T1 - TW - TDELAY), is a critical determiner of the consequences to be expected from a particular source term. This parameter determines whether or not people will be caught by the release and is critical to the determination of the early fatalities. For this analysis, it was specified that the partitioning be done on the basic parameters T1, TDELAY, and TW, rather than on the combined parameter dEVAC. By examining the PARTITION code output in Appendix D.6, one sees, for example, that partition group 2 for event type 3 has a dEVAC ranging from -1.883E+03 to +2.212E+04 sec. Since the mean value of dEVAC for this group is +1.577E+04 sec, it is clear that the source terms which have negative dEVACs are not dominant in the group definition. However, these source terms are the ones which will come closer to realizing their health effect potential and will, therefore, contribute to the final EH risk much more than their original fraction might indicate. Therefore, a nonconservative result is likely. It is our opinion, that the combined parameter should be used in future calculations.

3. As can be seen in the full PARTITION output shown in Appendix D, the number of parameters used in the definition of a group can be very small. A better process would require that a minimum number of parameters be used in the initial definition of the partition groups and then the number could be relaxed for the definition of the partition groups for the remaining source terms.

4.7 References

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5.0 CONSEQUENCE ANALYSIS

5.1 Introduction

Offsite consequences were calculated with the MACCS[1,2,3] code for each of the source term groups defined in the partitioning process. This code has been in use for some time and will not be described here in any detail. Although the variables thought to be the largest contributors to the uncertainty in risk were sampled from distributions in the accident frequency analysis, the accident progression analysis, and the source term analysis, there was no analogous treatment of uncertainties in the consequence analysis. Variability in the weather was accounted for; but, the uncertainty in other parameters such as the dry deposition velocity or the evacuation speed was not considered. The uncertainty in the warning time was assessed in the source term analysis and was used to define the source term groups analyzed in the consequence analysis. The impact of the uncertainty in the warning time is, therefore, indirectly accounted for.

5.2 Description of the Consequence Analysis

MACCS tracks the dispersion of the radioactive material in the atmosphere as it travels away from the plant. During its transport, the material is subject to deposition on the ground. MACCS then calculates the effects of this radioactivity on the population and the environment. Doses and the ensuing health effects from 60 radionuclides are computed.

MACCS treats atmospheric dispersion by the use of multiple, straight-line Gaussian plumes. Each plume can have a different direction, duration, and initial radionuclide inventory. Cross-wind dispersion is treated by a multi-step function approximating the Gaussian. Dry and wet deposition are treated as independent processes. The weather variability is treated by means of a stratified sampling process which divides the year of weather into 32 categories.

For early exposure, the following pathways are modeled: cloudshine, inhalation from the plume, groundshine, deposition on the skin, and inhalation of resuspended ground contamination. For the long-term exposure, the following four pathways are modeled: groundshine, inhalation of resuspended ground contamination, ingestion of contaminated water, and ingestion of contaminated food. The direct exposure pathways groundshine, and inhalation of resuspended ground contamination, produce doses in the population living in the area surrounding the plant. The indirect exposure pathways, ingestion of contaminated water and food, produce doses in those who ingest food or water emanating from the area around the accident site. The contamination of water bodies is estimated for the washoff of land-deposited material as well as direct deposition. The food pathway model includes direct deposition onto crop and uptake from the soil.

Both short-term and long-term mitigative measures are modeled in MACCS. Short-term actions include evacuation, sheltering, and emergency relocation out of the emergency planing zone (EPZ) based on estimated dose (i.e., hot-spot or normal relocation). Long-term actions include later relocation and agricultural restrictions on land use and crop disposition. Relocation and land decontamination, interdiction, and condemnation are based on projected long-term doses from groundshine and inhalation of resuspended radioactivity. The disposal of agricultural products and the removal of farmland from crop production are based on ground contamination criteria.

The health effects models use the dose received by an organ to predict morbidity (i.e., injury) or mortality. The models used in MACCS calculate both short-term and long-term effects for a number of organs.

The MACCS consequence model calculates a large number of different consequence measures. Results for the following six consequence measures are given in this report: early fatalities, total latent cancer fatalities, population dose within 50 miles, population dose for the entire region, early fatality risk within 1 mile, and latent cancer fatality risk within 10 miles. These consequence measures are described in Table 5.2-1. For the non-seismic accidents, 99.5% of the population is assumed to evacuate and 0.5% of the population is assumed not to evacuate, continuing normal activity during an accident. The treatment of the seismic accidents is somewhat different and is described in the next section. Details of the methods used to incorporate the consequence results for the source term groups into the integrated risk analysis are the same as for NUREG-1150 which were described in Volume 1 of NUREG/CR-4551.*

5.3 MACCS Input for LaSalle

The values of most MACCS input parameters (e.g., aerosol dry deposition velocity, health effects model parameter values, food pathway transfer factors) are not strongly dependent on site characteristics when a large region surrounding the site is being considered. For those parameters that depend on site characteristics (e.g., evacuation speed, shielding factors, farmland usage), the methods used to calculate the parameters are essentially the same for all sites. Unless specifically mentioned in this chapter, all MACCS input for LaSalle is the same as was used for NUREG-1150 which is documented in Volume 2, Part 7 of NUREG/CR-4551.[4]

Table 5.3-1 lists the MACCS input parameters that have strong site dependencies and presents the values of these parameters used in the calculations for the LaSalle site. The evacuation delay period begins when general emergency conditions occur and ends when the general public starts

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* E. D. Gorham, J. C. Helton, R. J. Breeding, S. C. Hora, W. B. Murfin, J. L. Sprung, and F. T. Harper, "Evaluation of Severe Accident Risks: Methodology for the Accident Progression, Source Term, Consequence, Risk Integration and Uncertainty Analyses," NUREG/CR-4551, Vol. 1, Rev. 1, SAND86-1309, Sandia National Laboratories, in preparation.

Table 5.2-1
Definition of Consequence Analysis Results

Variable	Definition
Early Fatalities	Total number of fatalities occurring within 1000 miles of the plant within one year of the accident.
Total Latent Cancer	Total number of latent cancer fatalities due to both early and chronic exposure within 1000 miles of the plant.
Population Dose Within 50 Miles	Population dose, expressed in effective dose equivalents for whole body exposure (person-rem), due to early and chronic exposure pathways within 50 miles of the reactor. Due to the nature of the chronic pathways models, the actual exposure due to food and water consumption may take place beyond 50 miles.
Population Dose Within Entire Region	Population dose, expressed in effective dose equivalents for whole body exposure (person-rem), due to early and chronic exposure pathways within 1000 miles of the plant.
Individual Early Fatality Probability Within One Mile	The probability of dying within one year of the accident for an individual within one mile of the exclusion boundary (i.e., Σ (ef/pop)p, where ef is the number of early fatalities, pop is the population size, p is the weather condition probability, and the summation is over all weather conditions). The appropriate evacuation assumptions are also applied to the population used in this calculation.
Individual Latent Cancer Probability Within 10 Miles	The probability of dying from cancer due to the accident for an individual within 10 miles of the plant (i.e., Σ (cf/pop)p, where cf is the number of cancer fatalities due to direct exposure in the resident population, pop is the population size, p is the weather condition probability, and the summation is over all weather conditions; chronic exposure does not include ingestion but does include integrated groundshine and inhalation exposure from $t = 0$ to $t = \infty$). The appropriate evacuation assumptions are also applied to the population used in this calculation.

Table 5.3-1
Site Specific Input Data for LaSalle MACCS Calculations

Parameter	
Reactor Power Level (MWt)	3292
Containment Height (m)	50
Containment Width (m)	50
Exclusion Zone Distance (km)	0.52
Evacuation Delay (h)	
Internal, Fire, and Flood	0.583
Seismic	0.875
Evacuation Speed (m/s)	
Internal, Fire, and Flood	2.3
Seismic	1.15
Farmland Fractions by Crop Categories	
Pasture	0.47
Stored Forage	0.10
Grains	0.26
Green Leafy Vegetables	0.0003
Legumes and Seeds	0.13
Roots and Tubers	0.002
Other Food Crops	0.001
Non-Farm Wealth ($/person)	86,000
Farm Wealth	
Value ($/hectare)	3305
Fraction in Improvements	0.19

to evacuate. Non-farm wealth includes personnel, business, and public property. The farmland fractions do not add to one because not all farmland is under cultivation in any given year. In addition to the site specific data presented in Table 5.3-1, the calculations used one year of meteorological data from the LaSalle site and regional population data developed from the 1980 census tapes modified by updated NRC data for the 0-10 mile region. Table 5.3-2 gives the population within certain distances of the plant as summarized from the MACCS demographic input. Table 5.3-3 lists the shielding parameters used in this analysis.

The evacuation parameters for the seismic risk analyses differed from those for the fire, flood, and internal events analyses. For the purpose of defining appropriate emergency response scenarios, the source term groups were segregated into two groups: (1) internal, fire, and flood initiators, and (2) seismic initiators (both low and high-g). For group 1, the NUREG-1150 approach was utilized. That is, 99.5% of the emergency planning zone (EPZ) population was evacuated from the region, 0.5% of the EPZ population remained behind and was subject to hot-spot and normal relocation, and a supplementary sheltering case was included for the purpose of comparison. For the seismic cases, group 2, a new approach was devised which represents an improvement over NUREG-1150.

The need for modifying the NUREG-1150 approach for seismic events resulted from concerns over the handling of the low-g cases. For the low-g cases in NUREG-1150, the shielding factors for normal activity were modified to reflect the effect of broken windows by increasing the shielding factors (leading to higher doses) for inhalation and groundshine. Those shielding factors were increased for the entire geographic area (1609 km radius) and were used for calculating the resultant dose over a million year exposure period. Since the extent of earthquake damage is likely to be limited to some tens of kilometers and recovery actions would be taken to repair broken windows, it is not realistic to postulate broken windows everywhere and forever. For this analysis, it was decided that using the same values as in the corresponding non-seismic cases would be more appropriate. The more accurate treatment outside the EPZ would more than compensate for some non-conservatism in the treatment within the EPZ.

For LaSalle, three different emergency response assumptions were utilized for the seismic initiators. Although the seismic source terms were segregated into two classes depending on the type of initiating event (either low-g or high-g), the same three scenarios (i.e., cohorts) were used for all seismic initiators. MACCS calculates the consequences for each cohort assuming the whole population is subject to that cohort's emergency response assumptions. The results for each seismic class are then calculated by weighting the individual cohort results by the fraction of the population participating in each emergency response for that seismic class. This approach allows flexibility since modifications of the weighting fractions do not entail rerunning MACCS, alternative results can be obtained by rerunning only the post-processor PRPOST (i.e., MACCS does not, itself, combine the different emergency responses for the portions of

Table 5.3-2

Population Within Different Radii From the Plant

Radius From Plant (km)	(miles)	Population
1.6	1.0	24
4.8	3.0	309
16.1	10.0	14,730
48.3	30.0	217,620
160.9	100.0	10,372,934
563.3	350.0	48,584,604
1609.3	1000.0	179,831,712

There is considerable variation in the sector populations (out to 1000 miles) as well. The E sector has a population of about 53 million and the ESE, SE, and SSE sectors each have populations of 15-18 million each, while the N and NNW sectors have populations of about 2 million each.

Table 5.3-3
Shielding Factors Used for LaSalle MACCS Calculations

	Population Response		
Radiation Pathway	Evacuate	Normal Activity	Take Shelter
Internal, Fire, and Flood Initiators			
Cloudshine	1.0	0.75	0.50
Groundshine	0.5	0.33	0.10
Inhalation	1.0	0.41	0.33
Skin	1.0	0.41	0.33
Seismic Initiators - Degraded Evacuation Cohort			
Cloudshine	1.0	0.75	
Groundshine	0.5	0.33	
Inhalation	1.0	0.41	
Skin	1.0	0.41	

Seismic Initiators - 12 and 24 hour Exposure Cohorts

	Within EPZ	Beyond EPZ
Cloudshine	1.0	0.75
Groundshine	0.5	0.33
Inhalation	1.0	0.41
Skin	1.0	0.41

the population affected by each emergency response assumption, this is done in PRPOST). PRPOST is a modification of PRAMIS[5] which allows for the processing of the full consequence distributions, not just the means, and the plotting of the risk complementary cumulative distribution functions (CCDFs) shown in Chapter 6. Currently the PRPOST code is undocumented and not available for release. For the LaSalle analysis, PRPOST was modified slightly from the version used in NUREG-1150 to allow the user to input different weighting fractions for each event type: non-seismic, low-g seismic, and high-g seismic for this analysis).

The three seismic cohorts were defined within the 10-mile EPZ as follows: (1) a degraded evacuation where the speed was cut in half and the delay time increased by 50% as done in NUREG-1150, (2) a 12-hour exposure period starting from accident initiation, and (3) a 24 hour exposure period starting from accident initiation. Evacuation speed and delay time for the seismic cases are shown in Table 5.3-1. Shielding factors utilized for the seismic cases are shown in Table 5.3-3. The weighting fractions used by MACCS for the seismic cases were as follows:

	Low-g	High-g
Degraded Evacuation	1.0	0.7
12 hour Relocation	0.0	0.2
24 hour Relocation	0.0	0.1

These fractions are used to determine the number of people that participate in the various evacuation assumptions (e.g., for high-g seismic accidents, 70% of the population participates in a degraded evacuation).

5.4 Results of MACCS Consequence Calculations

The results given in this section are conditional on the occurrence of a release. That is, given that a release takes place, with release fractions and other characteristics as defined by one of the source term groups; then the consequences reported in this section are calculated. The tables and figures in this section contain no information about the frequency with which these consequences may be expected. Information about the frequencies of consequences of various magnitudes is contained in the risk results described in Chapter 6. The integration of the probabilistic risk assessment uses the results of the MACCS consequence calculations in two forms.

In the first form, a single mean (over weather variation) result is reported for each consequence measure. This produces a nSTG x nC matrix of mean consequence measures, where nSTG is the number of source term groups and nC is the number of consequence measures under consideration. For LaSalle (internal and external initiators), nSTG = 98 (97 non-zero partitions and 1 zero partition) and nC = 6. The 98th partition is in case any source terms fall into the special case of no release. The resultant 98 x 6 matrix of mean consequence measures is shown in Table

5.4-1. The source terms that give rise to these mean consequence measures are given in Table 4.6-5. Because of the new partitioning process, none of source terms indicated in Table 4.6-5 have a zero frequency, unlike the NUREG-1150 results, and only the 98th partition has zero consequences. The mean consequence measures in Table 5.4-1 are used by PRAMIS in the calculation of the mean risk results for LaSalle. The population dose is the effective dose equivalent to the population in the region indicated. Table E.1-1 in Appendix E provides a breakdown of mean consequence results between individuals who evacuate, continue normal activities, and actively shelter; information on the division of results between early and chronic exposure is also given. In addition to the six consequence measures which are reported in the text of this report, Table E.1-1 contains results for early injuries (prodromal vomiting), economic cost, and individual early fatality risk at 1 mile.

In the second form, a complementary cumulative distribution function (CCDF) is used for each consequence measure. Conditional on the occurrence of a source term, each of these CCDFs gives the probability that individual consequence values will be exceeded due to the uncertainty in the weather conditions that exist at the time of an accident. These CCDFs are given in Figure 5.4-1a thru 5.4-1f. Each frame in this figure displays the CCDFs for a single consequence measure for all the subgroup source terms (LS-I-J) in Table 4.6-5 which have a non-zero frequency. The CCDFs were generated using the estimate that 99.5% of the population evacuates and 0.5% of the population continues normal activities. Each of the mean consequence results in Table 5.4-1 is the result of reducing one of the CCDFs in Figure 5.4-1a thru 5.4-1f to a single number. The CCDFs in Figure 5.4-1a thru 5.4-1f will subsequently be used to create CCDFs for risk, with the PRPOST code. The CCDFs for risk are presented in the next chapter; they relate consequence values with the frequency at which these values are exceeded.

Table 5.4-1
Mean Consequence Results for All Initiators
(Population Doses in Sv)

SOURCE TERM	EARLY FATALITIES	TOTAL LATENT CANCERS	EDEWBODY POP DOSE, (SV)<50 MI	EDEWBODY POP DOSE, (SV)<1000 MI	EARLY FATALITY RISK, 0-1 MI	TOTAL LATENT CANCER RISK, 0-10 MI
LAS-01	1.24E-01	1.34E+04	2.78E+04	8.09E+05	1.53E-03	6.720E-04
LAS-02	7.95E-02	1.17E+04	3.29E+04	7.12E+05	2.22E-04	5.360E-04
LAS-03	5.54E-02	1.24E+04	2.48E+04	7.58E+05	6.94E-04	9.230E-04
LAS-04	1.34E-02	9.56E+03	1.87E+04	5.96E+05	2.38E-04	3.980E-04
LAS-05	9.43E-02	1.18E+04	3.35E+04	7.13E+05	2.38E-04	5.580E-04
LAS-06	3.73E-06	8.70E+03	1.56E+04	4.97E+05	7.62E-08	2.070E-04
LAS-07	4.50E-03	9.91E+03	1.71E+04	5.88E+05	[illegible].20E-05	2.730E-04
LAS-08	3.10E-01	1.15E+04	2.70E+04	6.99E+05	[illegible].79E-03	2.180E-03
LAS-09	2.42E-02	1.24E+04	2.28E+04	7.04E+05	4.[illegible]7E-04	3.250E-04
LAS-10	6.06E-02	1.26E+04	2.[illegible]7E+04	7.66E+05	[illegible]	9.240E-04
LAS-11	6.76E-03	1.02E+04	1.76E+04	6.16E+05	[illegible]	2.880E-04
LAS-12	3.01E-04	4.82E+03	9.71E+03	2.78E+05	6.08E-06	1.430E-04
LAS-13	1.54E-04	9.[illegible]9E+03	1.52E+04	5.40E+05	3.13E-06	2.250E-04
LAS-14	0.00E+00	2.64E+02	3.10E+03	1.57E+04	0.00E+00	6.530E-05
LAS-15	0.00E+00	2.68E+02	2.48E+03	1.67E+04	0.00E+00	3.620E-05
LAS-16	1.17E-05	2.79E+03	9.20E+03	1.72E+05	2.38E-07	1.710E-04
LAS-17	0.00E+00	4.16E+02	3.46E+03	2.40E+04	0.00E+00	8.610E-05
LAS-18	0.00E+00	2.97E+03	1.00E+04	1.70E+05	0.00E+00	1.490E-04
LAS-19	1.08E-04	1.34E+03	5.53E+03	8.11E+04	2.21E-06	8.520E-05
LAS-20	1.24E-03	6.39E+03	1.31E+04	3.91E+05	2.12E-05	2.350E-04
LAS-21	0.00E+00	1.43E+04	2.56E+04	8.34E+05	0.00E+00	1.800E-04
LAS-22	0.00E+00	1.33E+04	2.58E+04	7.55E+05	0.00E+00	2.040E-04
LAS-23	0.00E+00	1.34E+04	2.73E+04	8.07E+05	0.00E+00	1.760E-04
LAS-24	0.00E+00	1.38E+04	2.53E+04	8.28E+05	0.00E+00	1.490E-04
LAS-25	6.41E-02	1.18E+04	3.33E+04	7.13E+05	0.00E+00	1.870E-04
LAS-26	0.00E+00	9.60E+03	1.57E+04	5.75E+05	0.00E+00	[illegible].060E-04
LAS-27	4.97E-02	1.23E+04	3.38E+04	7.42E+05	0.00E+00	[illegible].020E-04
LAS-28	0.00E+00	1.20E+04	2.31E+04	7.20E+05	0.00E+00	1.510E-04

Table 5.4-1 (Continued)
Mean Consequence Results for All Initiators
(Population Doses in Sv)

SOURCE TERM	EARLY FATALITIES	TOTAL LATENT CANCERS	EDEWBODY POP DOSE, (SV)<50 MI	EDEWBODY POP DOSE, (SV)<1000 MI	EARLY FATALITY RISK, 0-1 MI	TOTAL LATENT CANCER RISK, 0-10 MI
LAS-29	0.00E+00	1.14E+04	1.77E+04	6.79E+05	0.00E+00	1.770E-04
LAS-30	0.00E+00	8.70E+03	1.56E+04	4.97E+05	0.00E+00	1.990E-04
LAS-31	0.00E+00	6.18E+03	1.35E+04	3.76E+05	0.00E+00	1.760E-04
LAS-32	5.62E-02	2.13E+04	6.30E+04	1.23E+06	0.00E+00	1.780E-04
LAS-33	0.00E+00	1.01E+04	1.92E+04	6.13E+05	0.00E+00	1.660E-04
LAS-34	0.00E+00	7.28E+03	1.53E+04	5.29E+05	0.00E+00	1.920E-04
LAS-35	0.00E+00	4.41E+03	9.64E+03	2.56E+05	0.00E+00	1.330E-04
LAS-36	0.00E+00	2.53E+03	9.80E+03	1.47E+05	0.00E+00	1.890E-04
LAS-37	0.00E+00	2.72E+03	9.41E+03	1.61E+05	0.00E+00	1.730E-04
LAS-38	0.00E+00	6.74E+03	1.33E+04	4.10E+05	0.00E+00	2.010E-04
LAS-39	0.00E+00	5.44E+02	4.07E+03	3.10E+04	0.00E+00	9.800E-05
LAS-40	0.00E+00	3.41E+02	3.07E+03	2.12E+04	0.00E+00	4.230E-05
LAS-41	0.00E+00	3.33E+02	3.11E+03	1.94E+04	0.00E+00	7.450E-05
LAS-42	0.00E+00	3.64E+02	2.82E+03	2.20E+04	0.00E+00	3.960E-05
LAS-43	0.00E+00	3.76E+02	2.40E+03	2.18E+04	0.00E+00	4.790E-05
LAS-44	1.49E-02	8.61E+03	2.18E+04	5.54E+05	4.23E-06	7.150E-05
LAS-45	2.06E-03	9.94E+03	2.08E+04	6.49E+05	4.13E-06	1.920E-04
LAS-46	9.35E-04	1.26E+04	2.78E+04	7.69E+05	1.76E-05	2.640E-04
LAS-47	1.36E-04	5.28E+02	4.04E+03	4.73E+04	1.65E-06	1.800E-05
LAS-48	2.13E-04	9.40E+03	2.18E+04	5.84E+05	4.31E-06	1.990E-04
LAS-49	7.75E-03	1.42E+04	4.07E+04	8.76E+05	8.55E-05	3.180E-04
LAS-50	5.05E-07	6.41E+03	1.19E+04	3.64E+05	1.03E-08	1.410E-04
LAS-51	5.75E-05	1.25E+04	1.79E+04	7.33E+05	1.07E-06	1.850E-04
LAS-52	1.05E-03	1.28E+04	2.46E+04	7.68E+05	1.93E-05	1.650E-04
LAS-53	4.40E-04	8.36E+02	5.80E+03	7.06E+04	2.52E-06	4.780E-05
LAS-54	7.35E-05	8.94E+03	1.66E+04	5.40E+05	1.47E-06	1.420E-04
LAS-55	9.95E-05	8.50E+02	4.46E+03	6.50E+04	1.40E-06	4.080E-05
LAS-56	1.66E-04	1.52E+04	2.39E+04	9.13E+05	2.58E-06	1.490E-04

Table 5.4-1 (Continued)
Mean Consequence Results for All Initiators
(Population Doses in Sv)

SOURCE TERM	EARLY FATALITIES	TOTAL LATENT CANCERS	EDEWBODY POP DOSE, (SV)<50 MI	EDEWBODY POP DOSE, (SV)<1000 MI	EARLY FATALITY RISK, 0-1 MI	TOTAL LATENT CANCER RISK, 0-10 MI
LAS-57	2.39E-04	5.79E+02	4.62E+03	5.42E+04	2.2[illegible]E-06	1.440E-05
LAS-58	3.87E-04	1.28E+04	2.32E+04	7.82E+05	5.45E-06	1.510E-04
LAS-59	2.39E-07	5.16E+03	1.13E+04	3.22E+05	4.88E-09	1.020E-04
LAS-60	2.17E-04	9.96E+03	2.17E+04	5.95E+05	4.38E-06	2.380E-04
LAS-61	4.30E-04	9.23E+03	2.05E+04	5.82E+05	7.50E-06	1.220E-04
LAS-62	3.52E-05	4.37E+03	1.06E+04	2.74E+05	6.70E-07	1.120E-04
LAS-63	5.65E-06	4.40E+03	1.00E+04	2.58E+05	1.15E-0[illegible]	1.140E-04
LAS-64	2.44E-04	5.96E+02	4.92E+03	5.47E+04	2.26E-06	1.530E-05
LAS-65	6.22E-04	7.84E+03	1.48E+04	4.92E+05	2.56E-06	8.570E-05
LAS-66	9.60E-05	4.21E+03	1.17E+04	2.82E+05	1.40E-06	8.930E-05
LAS-67	2.65E-05	9.61E+02	4.12E+03	6.73E+04	5.20E-07	3.540E-05
LAS-68	5.35E-05	5.65E+02	4.20E+03	4.88E+04	9.40E-07	2.930E-05
LAS-69	1.42E-07	1.06E+04	2.06E+04	6.48E+05	2.91E-09	2.120E-04
LAS-70	4.97E-05	8.28E+02	6.34E+03	7.11E+04	9.30E-07	6.720E-05
LAS-71	3.18E-07	9.44E+03	1.94E+04	5.56E+05	6.50E-09	2.320E-04
LAS-72	0.00E+00	6.16E+03	1.62E+04	3.81E+05	0.00E+00	[illegible].050E-04
LAS-73	2.79E-02	1.22E+04	2.48E+0[illegible]	7.59E+05	4.85E-06	[illegible].530E-04
LAS-74	2.08E-04	1.14E+04	1.82E+04	7.02E+05	2.15E-06	1.540E-04
LAS-75	3.98E-06	6.48E+03	1.39E+04	3.87E+05	8.15E-08	1.750E-04
LAS-76	0.00E+00	2.96E+03	1.[illegible]5E+04	1.73E+05	0.00E+00	2.250E-04
LAS-77	0.00E+00	3.43E+03	1.07E+04	2.06E+05	0.00E+00	2.340E-04
LAS-78	0.00E+00	3.86E+03	1.25E+04	2.26E+05	0.00E+00	2.350E-04
LAS-79	1.28E-07	2.74E+03	8.03E+03	1.59E+05	2.60E-09	9.350E-05
LAS-80	0.00E+00	2.90E+03	8.44E+03	1.69E+05	0.00E+00	1.090E-04
LAS-81	0.00E+00	5.32E-01	2.80E+01	5.93E+01	0.0[illegible]00	1.090E-07
LAS-82	5.35E-09	1.49E+02	9.05E+02	1.19E+04	1.09E-10	9.930E-06
LAS-83	2.32E-08	3.39E+02	2.07E+03	2.17E+04	4.74E-10	2.320E-05
LAS-84	0.00E+00	2.24E+02	1.58E+03	1.42E+04	0.00E+00	2.330E-05

Table 5.4-1 (Concluded)
Mean Consequence Results for All Initiators
(Population Doses in Sv)

SOURCE TERM	EARLY FATALITIES	TOTAL LATENT CANCERS	EDEWBODY POP DOSE, (SV)<50 MI	EDEWBODY POP DOSE, (SV)<1000 MI	EARLY FATALITY RISK, 0-1 MI	TOTAL LATENT CANCER RISK, 0-10 MI
LAS-85	0.00E+00	2.33E-01	1.29E+01	2.60E+01	0.00E+00	4.440E-08
LAS-86	0.00E+00	2.61E+02	1.89E+03	1.67E+04	0.00E+00	2.830E-05
LAS-87	0.00E+00	6.48E+00	9.89E+01	4.75E+02	0.00E+00	3.850E-07
LAS-88	0.00E+00	5.67E+02	3.35E+03	3.38E+04	0.00E+00	6.570E-05
LAS-89	0.00E+00	6.03E-02	3.01E+00	5.76E+00	0.00E+00	7.200E-08
LAS-90	0.00E+00	2.15E+00	8.93E+01	2.39E+02	0.00E+00	5.210E-07
LAS-91	0.00E+00	1.42E-01	8.27E+00	1.59E+01	0.00E+00	2.770E-08
LAS-92	0.00E+00	3.36E+02	3.64E+03	2.02E+04	0.00E+00	9.330E-05
LAS-93	0.00E+00	1.08E+01	1.69E+02	6.84E+02	0.00E+00	1.430E-06
LAS-94	5.05E-06	9.43E+02	4.65E+03	6.21E+04	1.03E-07	5.630E-05
LAS-95	0.00E+00	3.89E-03	8.66E-02	2.62E-01	0.00E+00	1.870E-09
LAS-96	0.00E+00	1.23E-01	5.70E+00	1.29E+01	0.00E+00	1.450E-08
LAS-97	0.00E+00	3.98E+02	3.97E+03	2.39E+04	0.00E+00	8.700E-05
LAS-98	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.000E+00

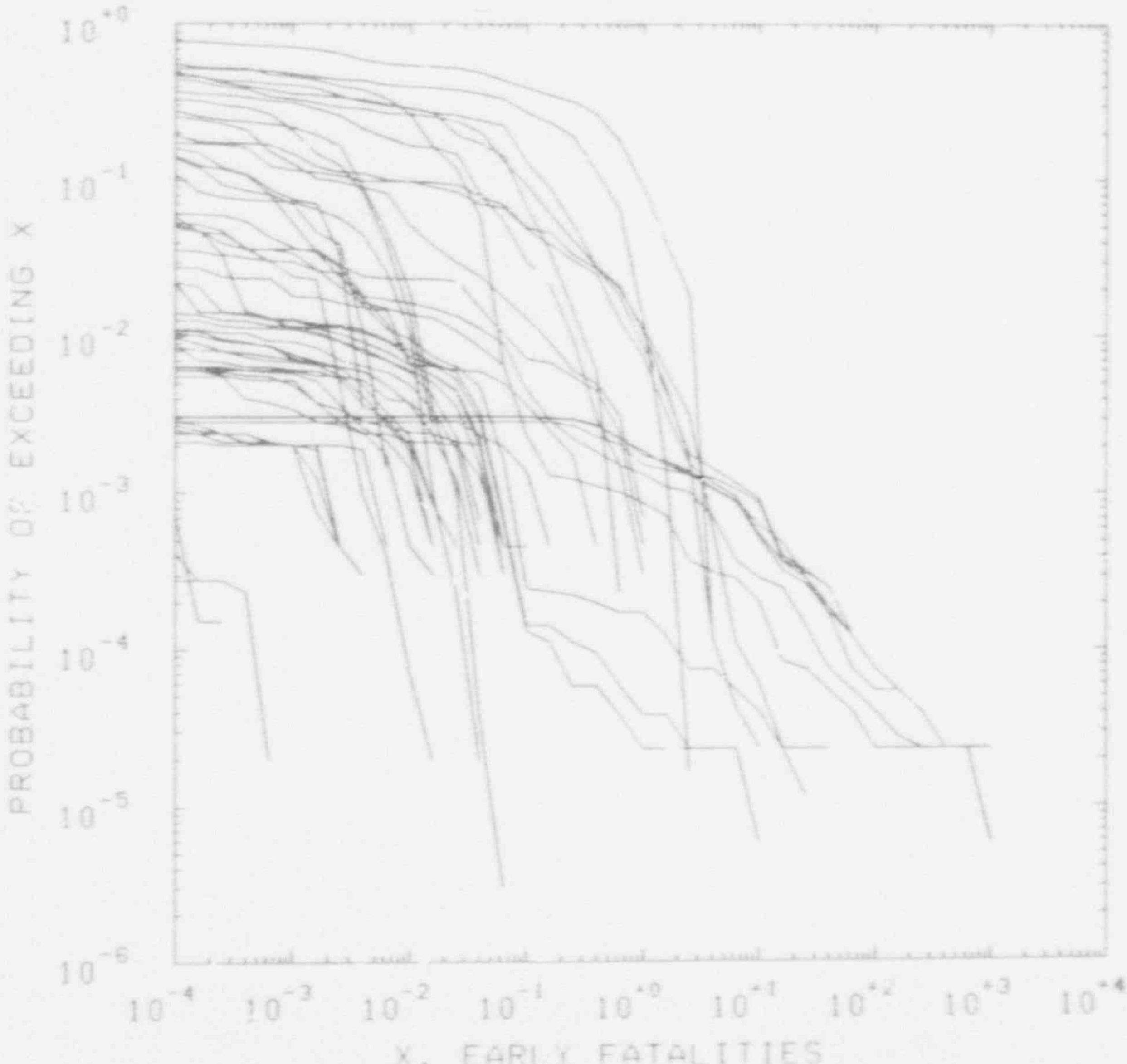

Figure 5.4-1a. Exceedance Frequency Curves for Early Fatalities for All Source Terms That Have Non-zero Frequency, Conditional on the Occurrence of the Release.

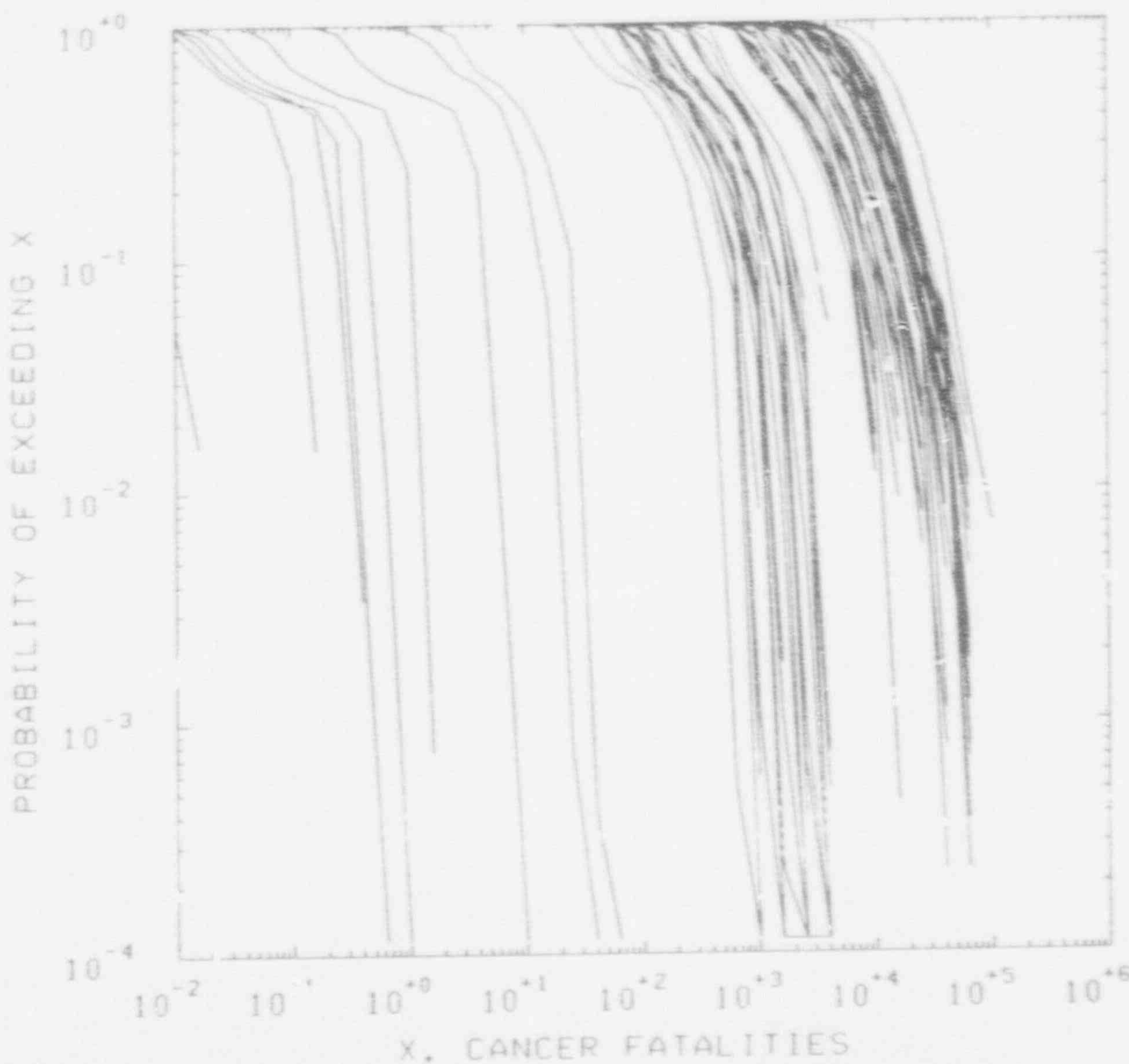

Figure 5.4-1b. Exceedance Frequency Curves for Latent Cancer Fatalities for All Source Terms That Have Non-zero Frequency, Conditional on the Occurrence of the Release.

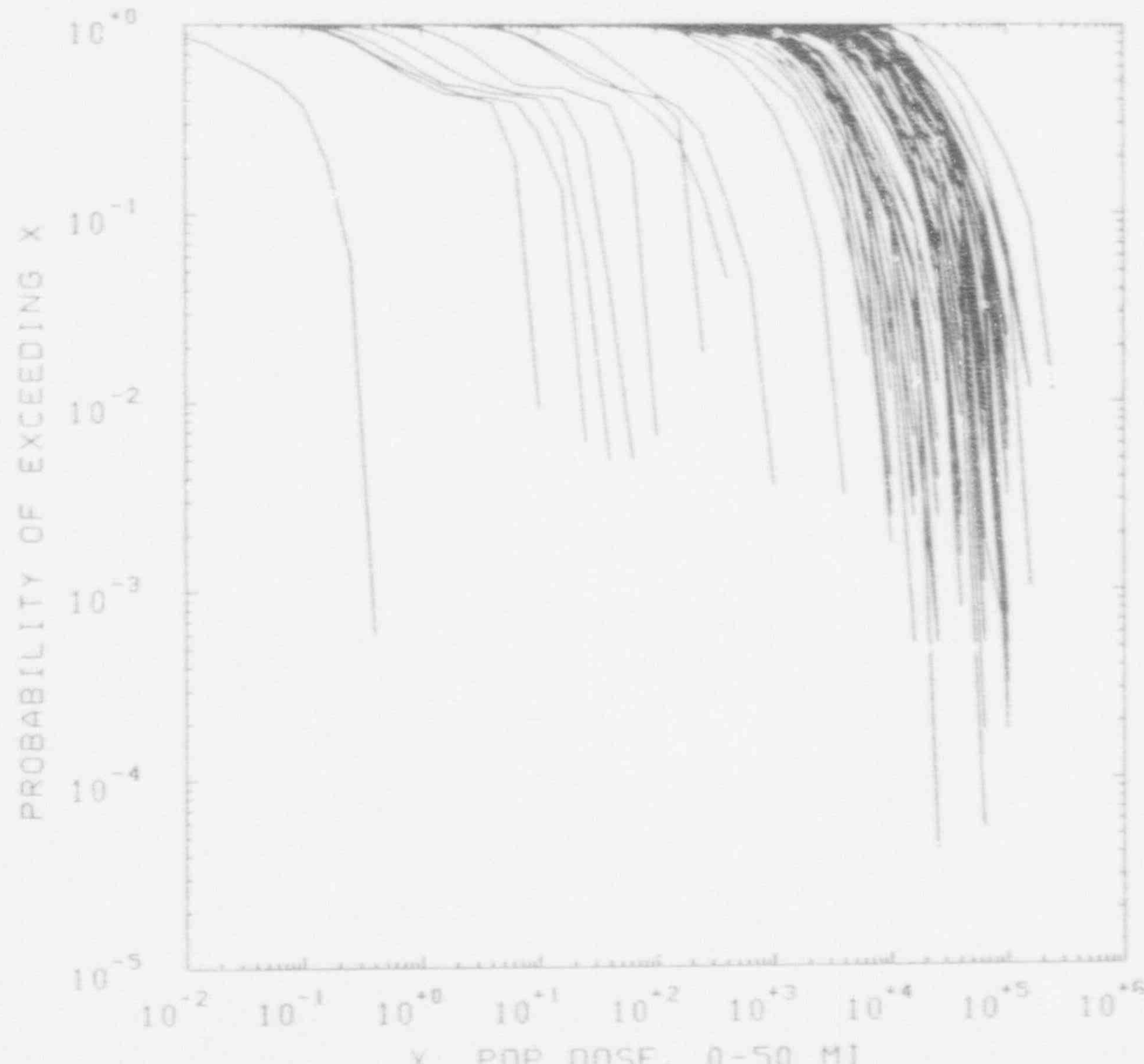

Figure 5.4-1c. Exceedance Frequency Curves for Population Dose Within 50 Miles for All Source Terms That Have Non-zero Frequency, Conditional on the Occurrence of the Release.

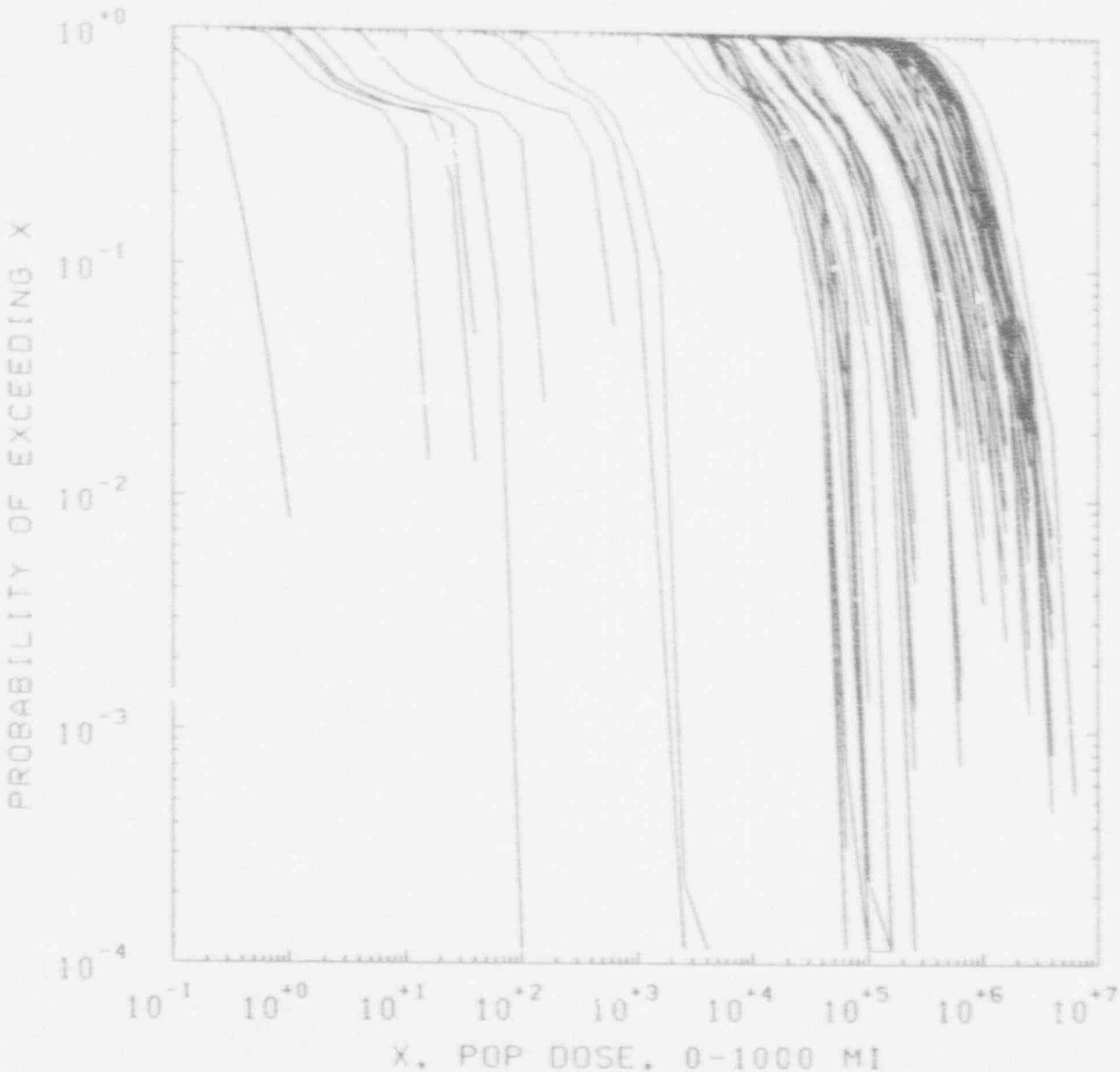

Figure 5.4-1d. Exceedance Frequency Curves for Population Dose Within Entire Region for All Source Terms That Have Non-zero Frequency, Conditional on the Occurrence of the Release.

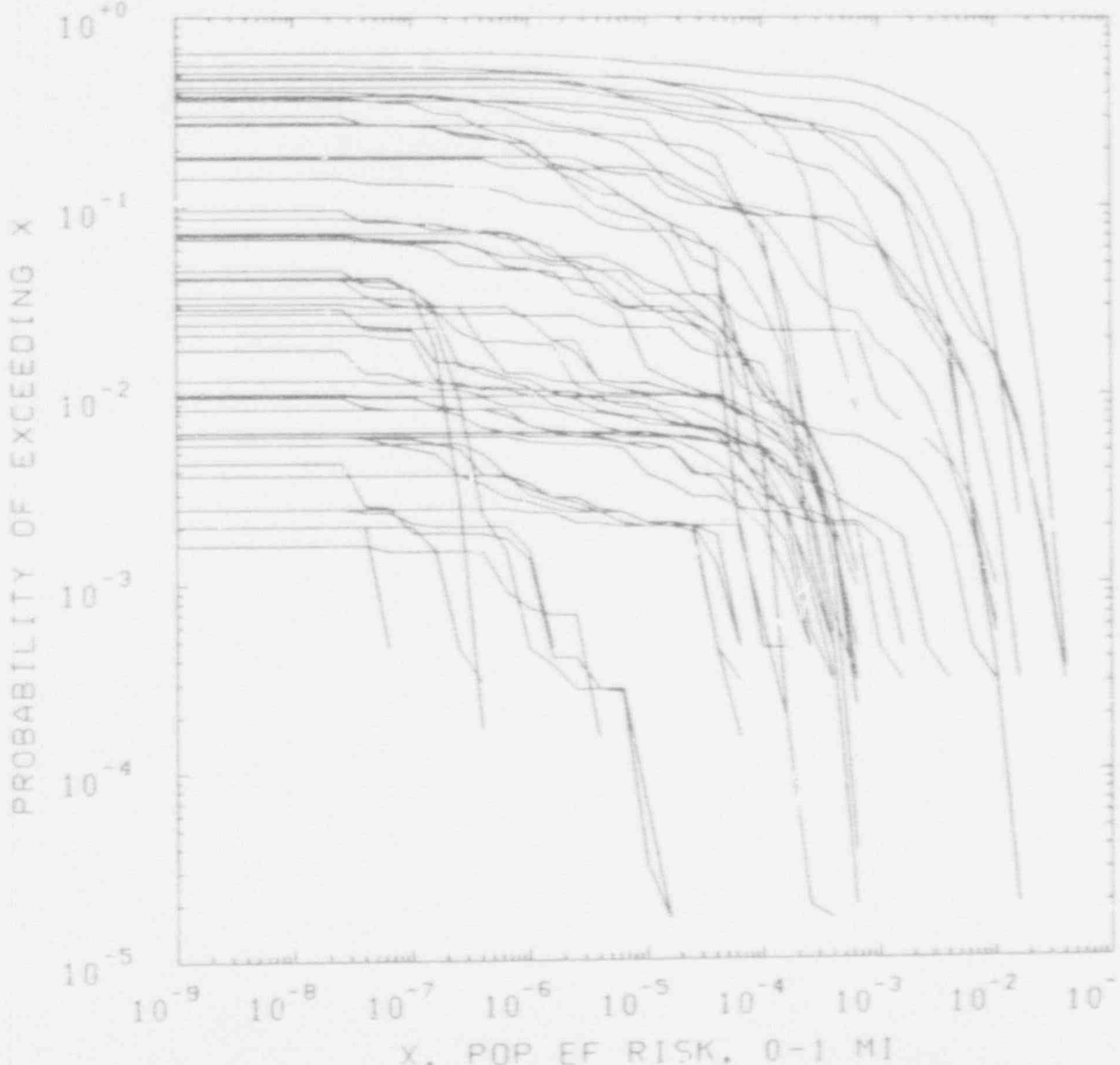

Figure 5.4-1e. Exceedance Frequency Curves for Individual Early Fatality Risk 0-1 Mile for All Source Terms That Have Non-zero Frequency, Conditional on the Occurrence of the Release.

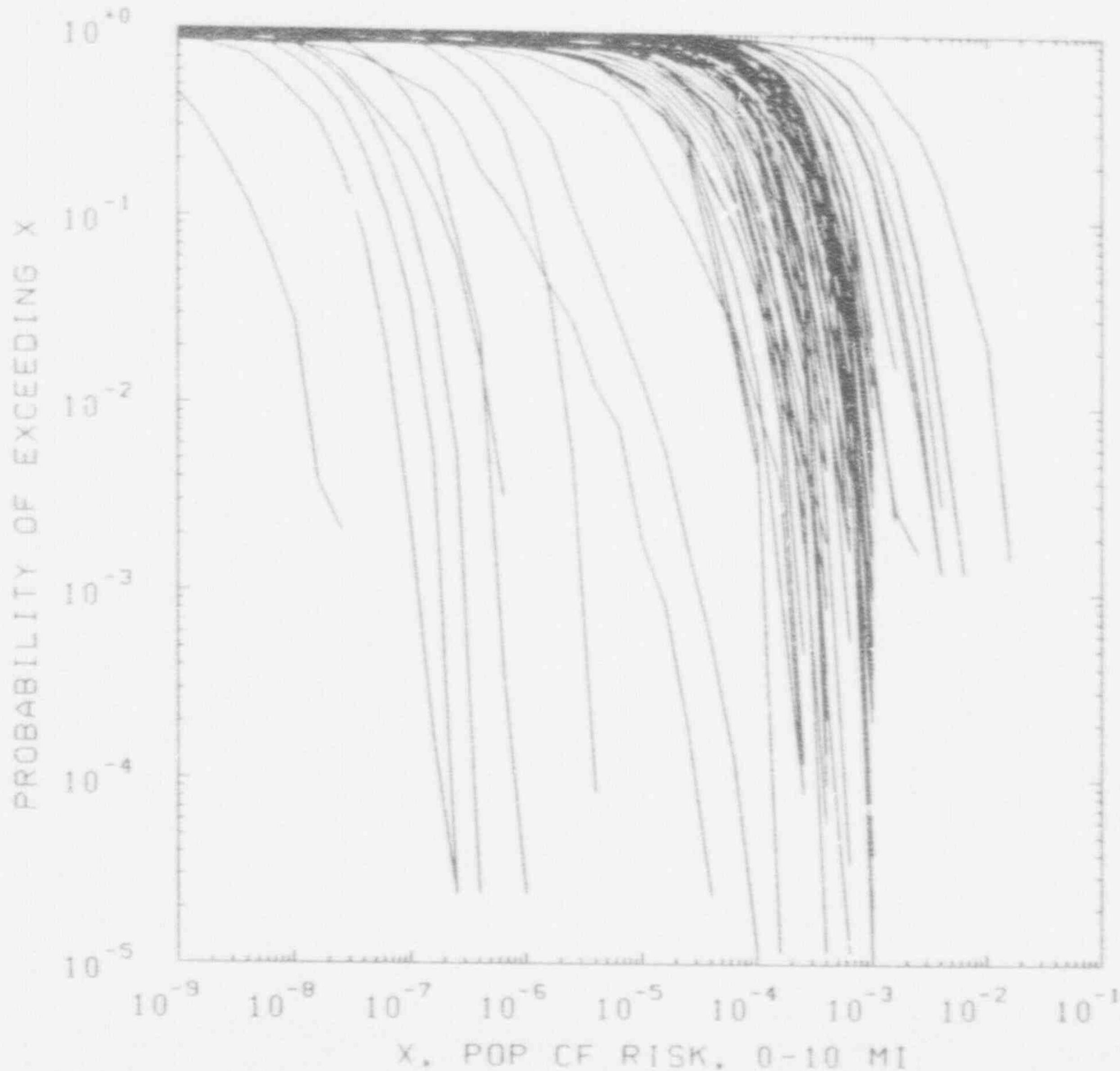

Figure 5.4-1f. Exceedance Frequency Curves for Individual Latent Cancer Fatality Risk 0-10 Miles For All Source Terms That Have Non-zero Frequency, Conditional on the Occurrence of the Release.

5.5 References

1. D. I. Chanin, J. L. Sprung, L. T. Ritchie, and H.-N. Jow, "MELCOR Accident Consequence Code System (MACCS): User's Guide," NUREG/CR-4691, SAND86-1562, Volume 1, Sandia National Laboratories, Albuquerque, NM, February 1990.

2. H.-N. Jow, J. L. Sprung, J. A. Rollstin, and D. I. Chanin, "MELCOR Accident Consequence Code System (MACCS): Model Description," NUREG/CR-4691, SAND86-1562, Volume 2, Sandia National Laboratories, Albuquerque, NM, February 1990.

3. J. A. Rollstin, D. I. Chanin, and H.-N. Jow, "MELCOR Accident Consequence Code System (MACCS): Programmer's Reference Manual," NUREG/CR-4691, SAND86-1562, Volume 3, Sandia National Laboratories, Albuquerque, NM, February 1990.

4. J. L. Sprung, J. A. Rollstin, J. C. Helton, and H-N Jow, "Evaluation of Severe Accident Risks: Quantification of Major Input Parameters -- MACCS Input," NUREG/CR-4551, Vol. 2, Rev. 1, Part 7, SAND86-1309, Sandia National Laboratories, Albuquerque, NM, December 1990.

5. R. L. Iman, J. D. Johnson, and J. C. Helton, "PRAMIS: Probabilistic Risk Assessment Model Integration System. User's Guide," NUREG/CR-5262, SAND88-3093, Sandia National Laboratories, Albuquerque, NM, May 1990.

6.0 RISK RESULTS FOR LASALLE

6.1 Introduction

This chapter describes the results of the integrated risk analysis for internal and external initiators at the LaSalle plant. Risk is determined by bringing together the results of four constituent analyses: accident frequency, accident progression, source term, and consequence analysis. The phrase integrated risk analysis has the following meanings in this study. First, it is used to refer to the result when all four analyses are combined and, second, to the expression of risk which contains all initiators (i.e., both internal and external initiators). Separate risk results are not presented for internal initiators and external initiators in this chapter. Details on the methods used in calculating risk can be found in Volume 1 of NUREG/CR-4551,* the methodology report, which is the technical support document for NUREG-1150.[1]

Section 6.2 is a discussion of basic risk results. Section 6.3 addresses the types of accidents and plant features which are important in determining the risk at LaSalle, and finally, Section 6.4 presents the results of a regression analysis performed to determine the important contributors to the uncertainty in risk. The figures and tables in Chapter 6 present only a small portion of the total risk output available. Additional data are presented in Appendix F.

6.2 Risk Results (Internal and External Initiators)

Figures 6.2-1 thru 6.2-3 show the basic results of the integrated risk analysis for internal and external initiators at LaSalle. These figures show the complementary cumulative distribution functions (CCDFs) for early fatalities, latent cancer fatalities, population dose within 50 miles, population dose within the entire region, individual early fatality probability within one mile of the site boundary, and individual latent cancer fatality probability within 10 miles. The CCDFs display the relationship between the frequency of the consequence and the magnitude of the consequence.

As there are 400 observations in the LHS[2] sample for LaSalle, the complete set of risk results, at the most basic level, consists of 400 CCDFs for each consequence measure. Plots showing these 400 curves are contained in Appendix F; only four statistical measures of the 400 curves are shown in Figures 6.2-1 thru 6.2-3. These measures are generated as follows. For

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* E. D. Gorham, J. C. Helton, R. J. Breeding, S. C. Hora, W. B. Murfin, J. L. Sprung, and F. T. Harper, "Evaluation of Severe Accident Risks: Methodology for the Accident Progression, Source Term, Consequence, Risk Integration and Uncertainty Analyses," NUREG/CR-4551, Vol. 1, Rev. 1, SAND86-1309, Sandia National Laboratories, in preparation.

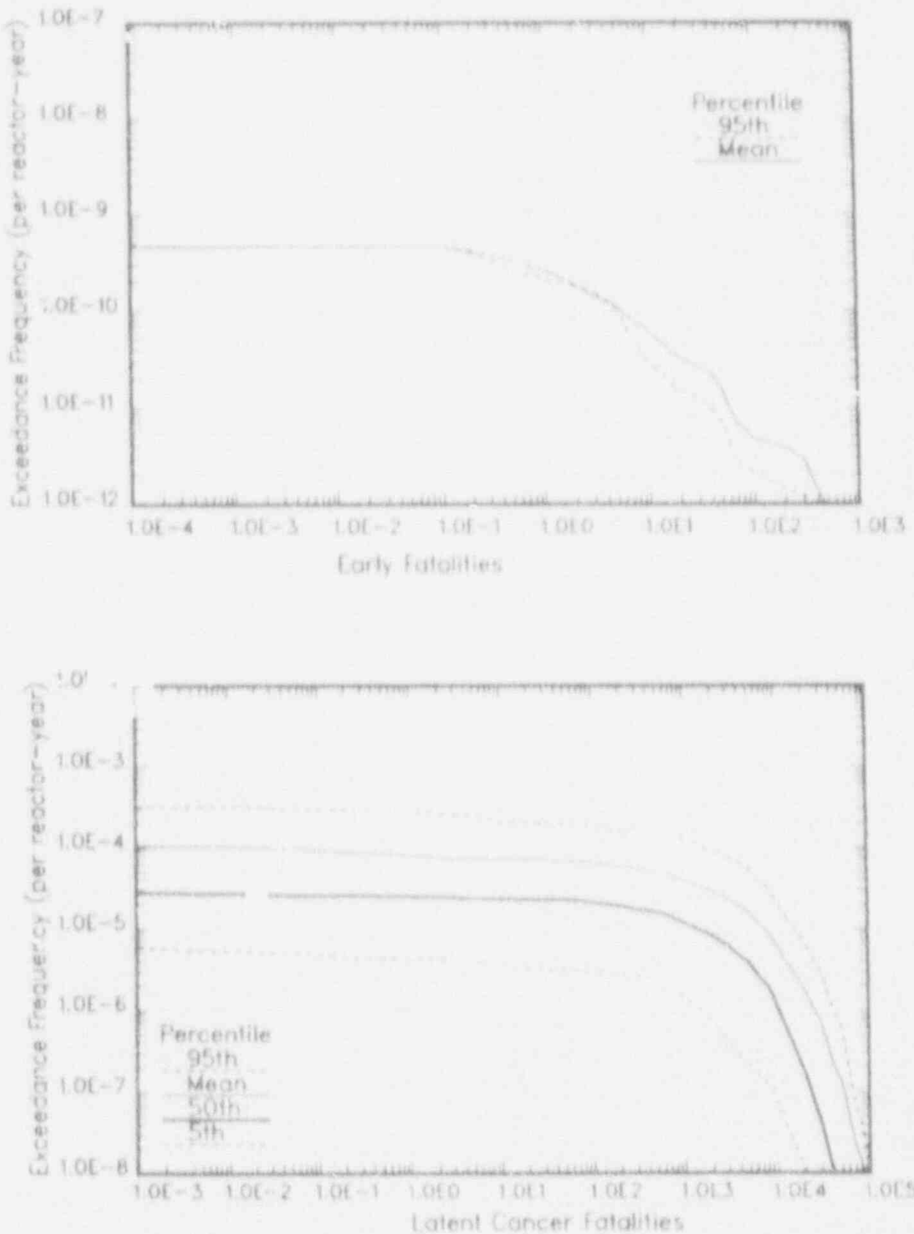
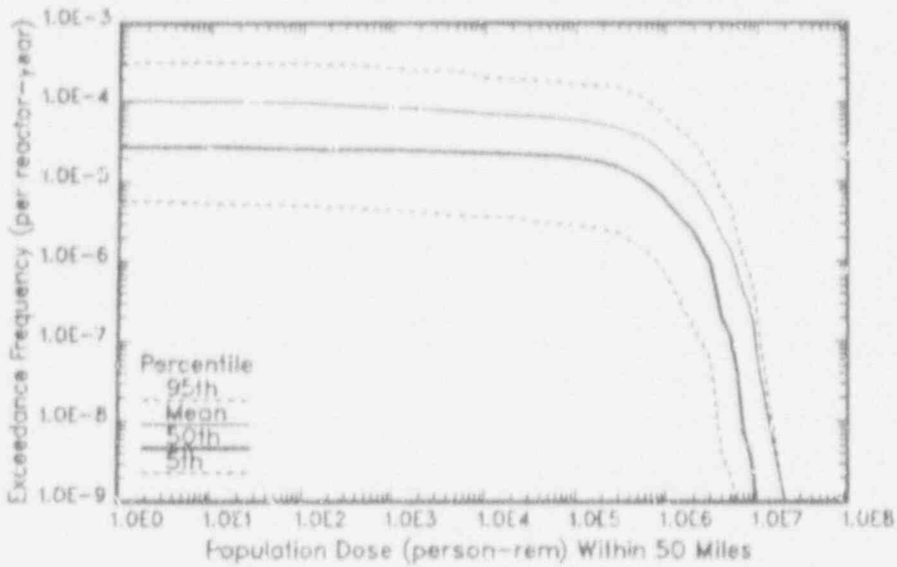


Figure 6.2-1. Results of the Integrated Risk Analysis for All Initiators at LaSalle: Statistical Measures of the 400 Excnedance Frequency Curves for Early and Latent Cancer Fat/lities.



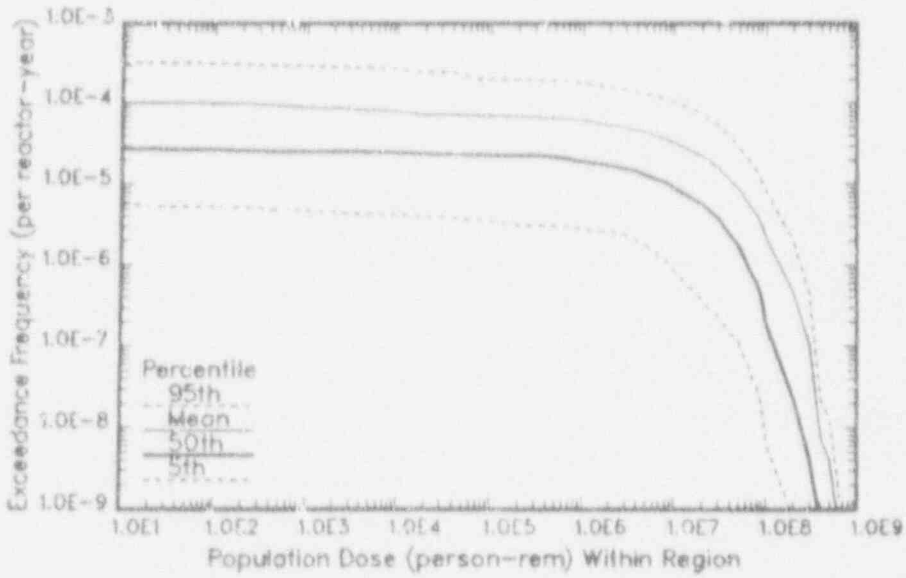


Figure 6.2-2. Results of the Integrated Risk Analysis for All Initiators at LaSalle: Statistical Measures of the 400 Exceedance Frequency Curves for Population Dose Within 50 Miles and Entire Region.

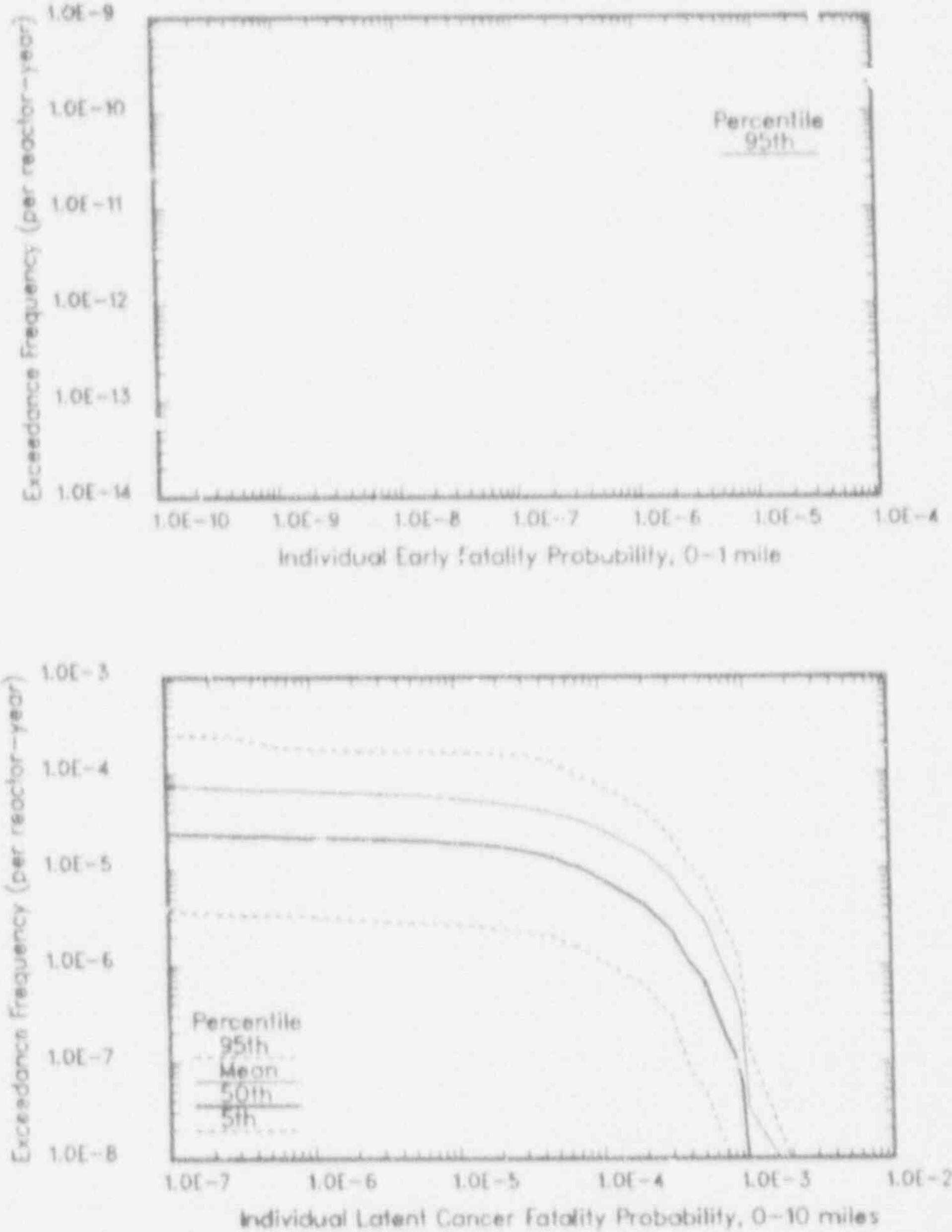


Figure 6.2-3. Results of the Integrated Risk Analysis for All Initiators at LaSalle: Statistical Measures of the 400 Exceedance Frequency Curves for Individual Early and Latent Cancer Fatality Probability.

each consequence value on the abscissa, there are 400 values of the exceedance frequency (one for each observation or sample element) and from these 400 values the mean, median, 95th percentile, and 5th percentile values are calculated. When this is done for each value of the consequence measure and plotted, the curves in Figures 6.2-1 thru 6.2-3 are obtained. For some of the plots, all the curves are not shown because the exceedance frequencies all fall below the minimum value shown on the plots. Figures 6.2-1 thru 6.2-3 give the relationship between the magnitude of the consequence and the frequency at which the consequence is exceeded, as well as the variation in that relationship. The percentile curves in Figures 6.2-1 thru 6.2-3 and similar figures are only valid when read from the abscissa.

The abscissas in Figure 6.2-3 are different from those in Figures 6.2-1 and 6.2-2 because they represents conditional probability rather than the number of fatalities or the dose; specifically, the probability that an individual, randomly located in the spatial interval according to the population distribution, will die given that the accident occurs. The ordinate gives the frequency of an accident that produces a conditional probability that exceeds the value on the abscissa. Curves for individual early fatality probability are not shown because the frequency of obtaining a value greater than 1.0E-10 for individual early fatality probability is less than 1.0E-14/R-yr. at the 95th percentile. The absolute value of these numbers becomes meaningless at this level and, therefore have not been presented.

The variation from the 5th to the 95th percentiles indicates the uncertainty in the risk estimates due to uncertainty in the basic parameters in the three constituent analyses (the accident frequency, accident progression, and source term analyses) that are sampled using the LHS formalism. The variation along a curve in Figures 6.2-1 thru 6.2-3 (or along one of the individual curves in Appendix F) is indicative of the variation in risk due to different types of accidents (i.e., uncertainty represented as variations in the outcome of the different questions making up the APET) and due to different weather conditions at the time of the accident. The significance of this distinction will be discussed further in Section 6.4. As the magnitude of the consequence measure increases, the mean curve typically approaches or exceeds the 95th percentile curve. This results when the mean is dominated by a few large observations, which often happens for large values of the consequences because only a few observations have nonzero exceedance frequencies for these large consequences. Figure 6.2-1 shows the following mean and median exceedance frequencies for fixed values of early fatalities (EF) and latent cancer fatalities (LCF):

	Exceedance Frequency (./R-yr)	
Consequence	Mean	Median
1 EF	2E-10	< 1E-12
100 EF	5E-12	< 1E-12
100 LCF	6E-5	1E-5
5000 LCF	2E-5	4E-6

Although the CCDF for each observation conveys the most information about risk, a single number may be generated for each consequence measure for each observation. This value, denoted annual risk, is determined by summing the product of the frequencies and consequences for all the points that are used to construct the CCDF for each observation in the sample. The construction of annual risk has the effect of averaging over the different weather states and the contributions from all the different types of accidents that can occur. Since the complete analysis consisted of a sample of 400 observations, there are 400 values of annual risk for each consequence measure. These 400 values may be ordered and plotted as histograms, which is done in Figures 6.2-4 thru 6.2-9. The cumulative distribution function for each of these histograms is also presented in Figures 6.2-4 thru 6.2-9. (Please note that the scale of the histogram vis-a-vis the cumulative distribution has been exaggerated for clarity.) The histograms for the six consequence measures are also presented in Figure 6.2-10 along with the four statistical measures utilized in Figures 6.2-1 thru 6.2-3 and in Table 6.2-1. The histograms presented in Figure 6.2-3 are consistent with the risk plots presented in NUREG-1150. Note that considerable information has been lost in going from the CCDFs in Appendix F to the histograms of annual risk in Figures 6.2-4 thru 6.2-9; the relationship between the size of the consequence and its frequency has been sacrificed to obtain a single value for risk for each observation.

The plots in Figures 6.2-4 thru 6.2-9 show the variation in the annual risk for six consequence measures. Where the mean is close to the upper end of the distribution (e.g., 95th percentile), it may be inferred that a relatively small number of observations dominate the mean value. This is more likely to occur for the early fatality consequence measures than for the latent cancer fatality or population dose consequence measures due to the threshold effect for early fatalities. In essence, Figures 6.2-4 thru 6.2-9 show the probability density functions of the logarithms of the consequence measures. Equivalent density functions could be generated for the consequence measures themselves, but would appear quite different due to the change in scale.

The safety goals are expressed in terms of individual's fatality risks, which are really an individual's probability of becoming a casualty of a reactor accident in a given year. The individual early fatality risk within one mile is the probability per year that a person living within one mile of the site boundary will die within a year due to an accident. To obtain an average value for a person located anywhere within one mile of the site boundary, the risks for the entire population within one mile of the site are averaged. The individual latent cancer fatality risk within 10 miles is the probability per year that a person living within 10 miles of the plant will die from cancer due to radiation exposure received from the accident. To obtain an average value for a person located anywhere within ten miles of the site boundary, the risks for the entire population within ten miles of the site are averaged. A single value for individual fatality risk for each observation is obtained by reducing the CCDF for each observation to a single value. The density distribution of these 400

Table 6.2-1
Distributions for Annual Risk at LaSalle for
Internal and External Initiators
(All values per reactor-year)
(Population doses in person-rem)

Risk Measure	5th%tile	Median	Mean	95th%tile
Core Damage	5.7E-6	2.7E-5	1.0E-4	3.2E-4
Early Fatalities	1.9E-13	1.5E-10	1.2E-8	2.5E-8
Latent Cancer Fatalities	7.3E-3	6.5E-2	2.5E-1	8.4E-1
Population Dose-50 miles	2.7E+0	1.9E+1	6.6E+1	2.3E+2
Population Dose Entire Region	4.3E+1	3.9E+2	1.5E+3	5.2E+3
Individual Early Fatality Risk--0 to 1 mile	3.6E-15	2.5E-12	1.1E-10	3.0E-10
Individual Latent Cancer Fatality Risk--0 to 10 miles	3.8E-10	[illegible].6E-09	8.5E-09	2.6E-08

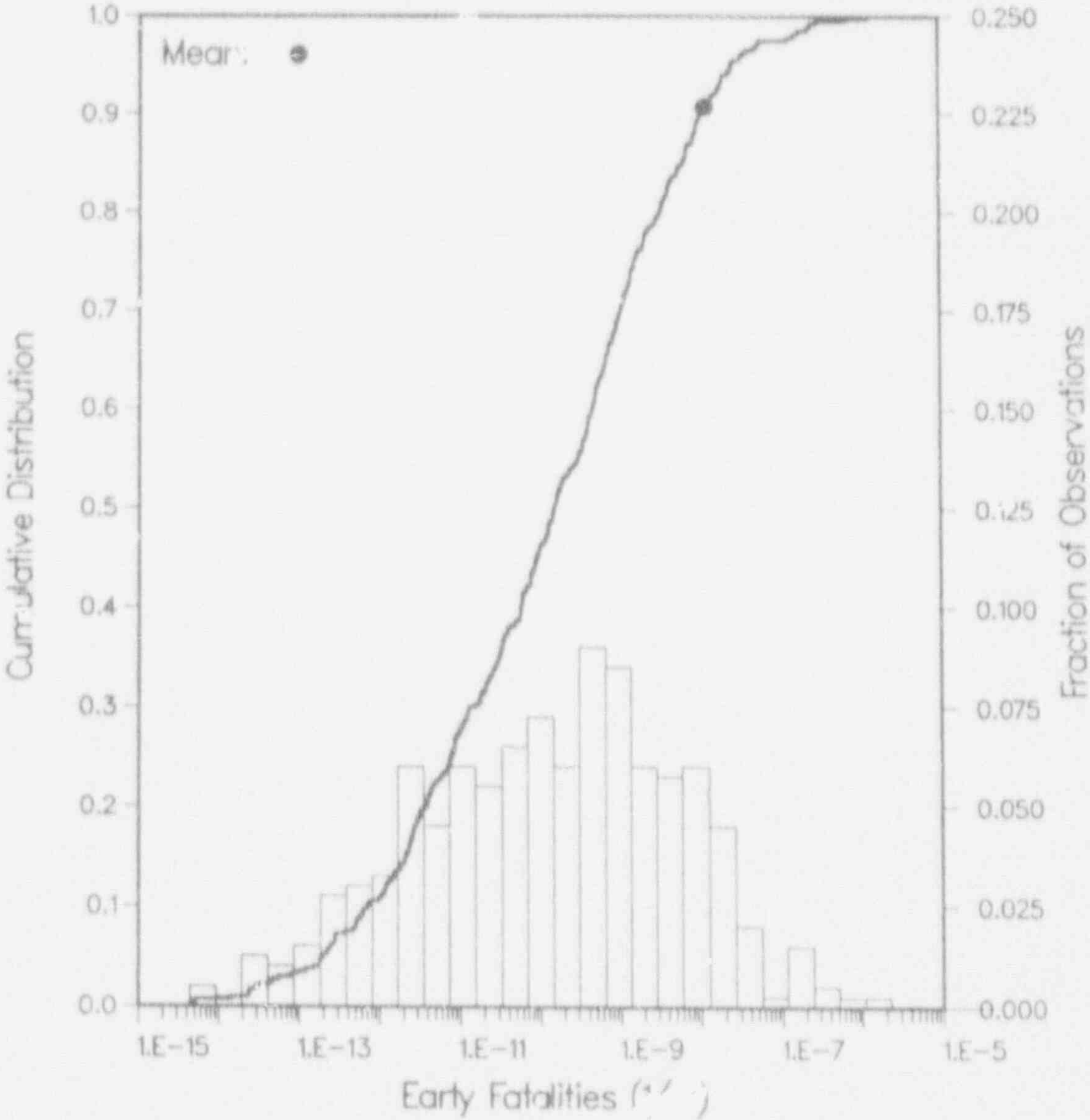


Figure 6.2-4. Histogram and Cumulative Distribution Function of Annual Risk at LaSalle for Early Fatalities.

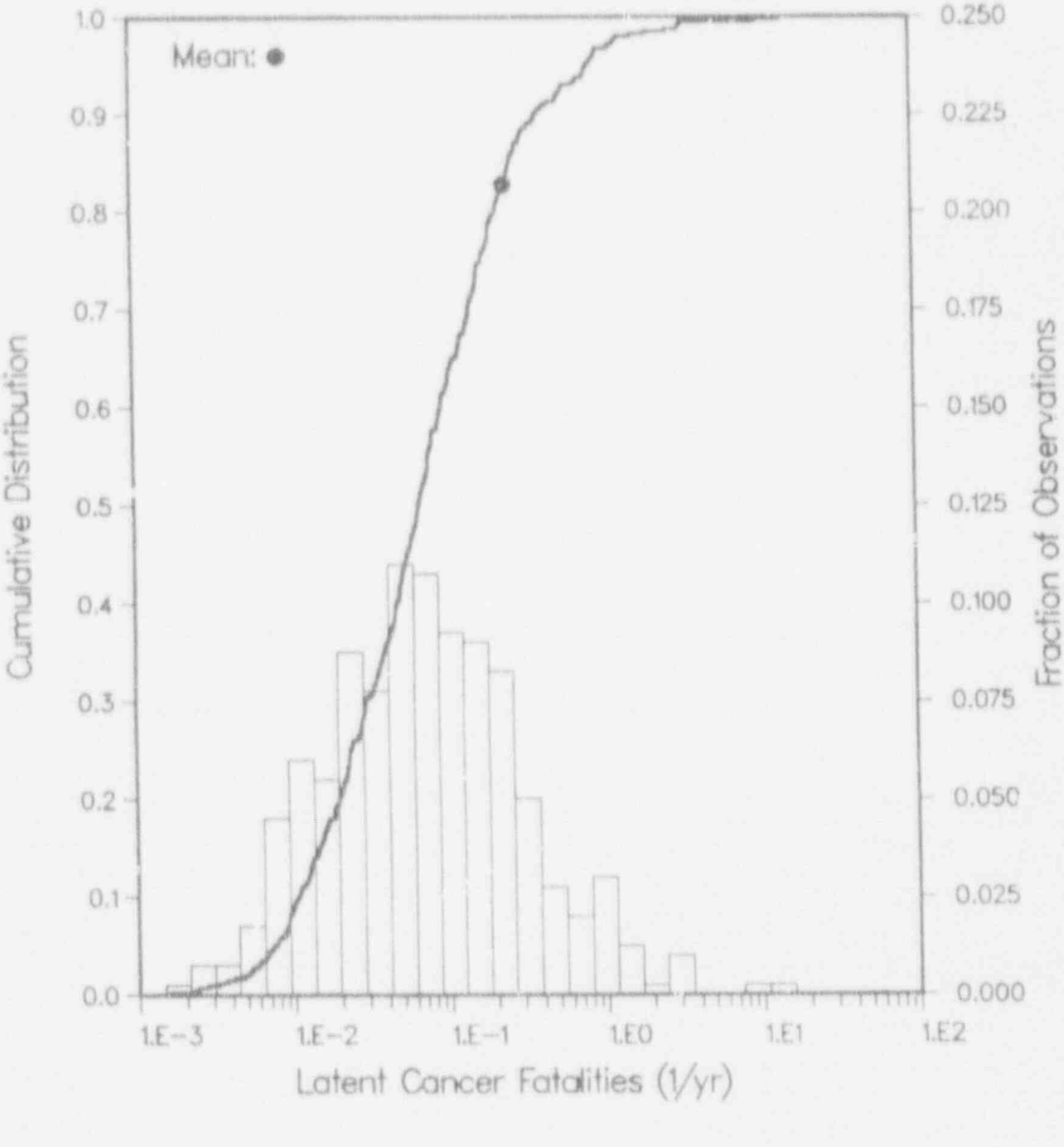


Figure 6.2-5. Histogram and Cumulative Distribution Function of Annual Risk at LaSalle for Latent Cancer Fatalities.

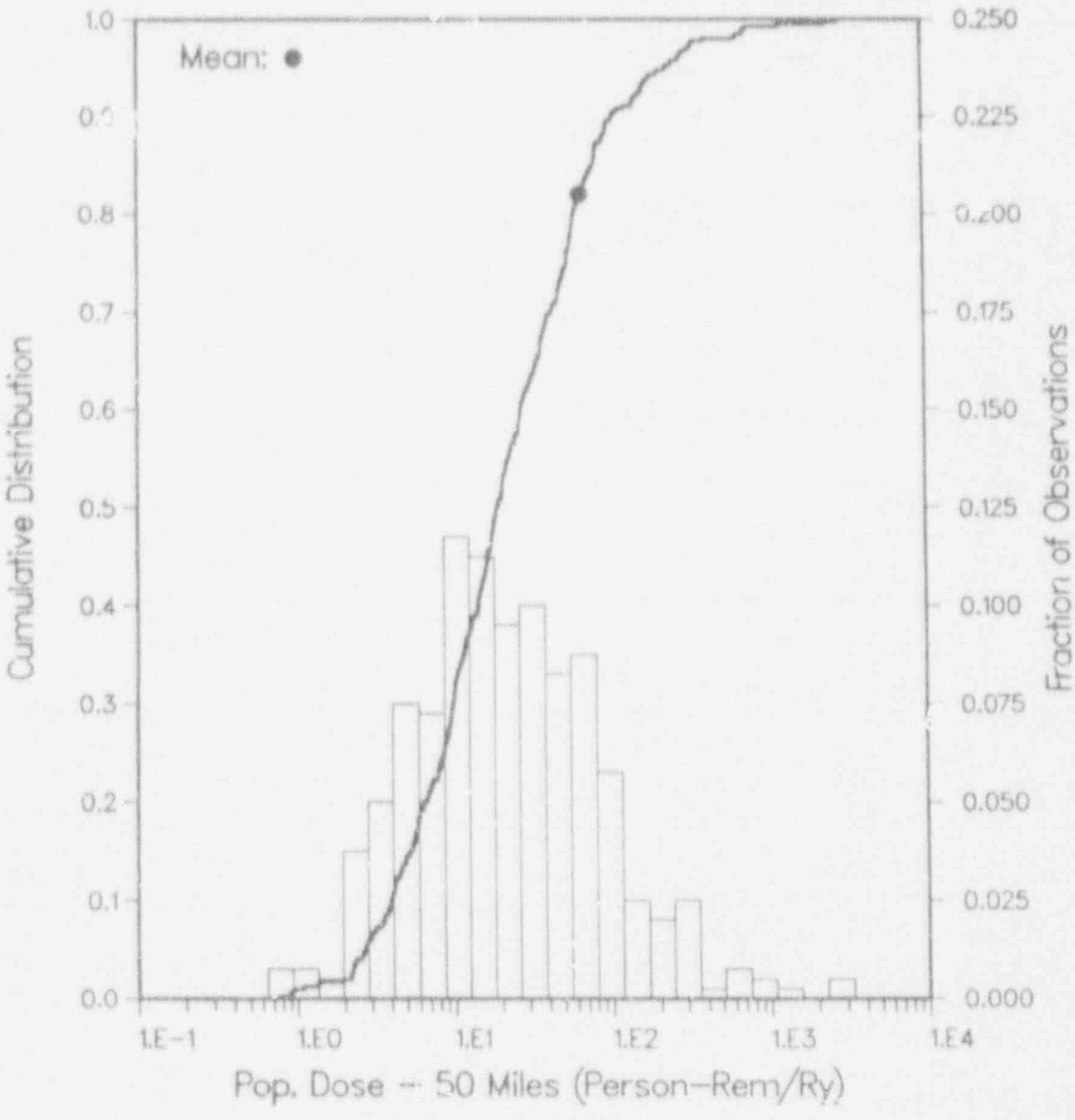


Figure 6.2-6. Histogram and Cumulative Distribution Function of Annual Risk at LaSalle for Population Dose Within 50 Miles.

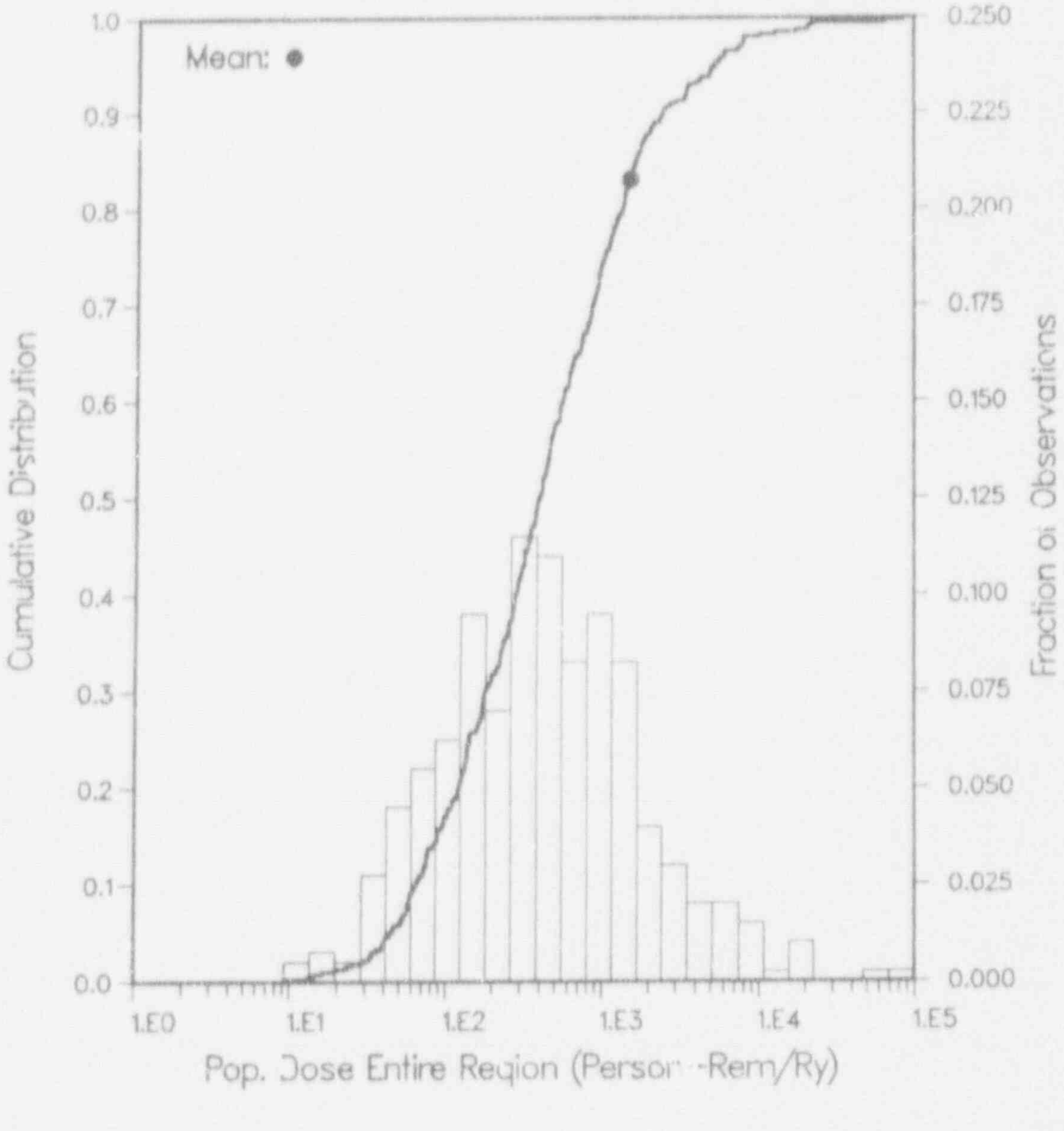


Figure 6.2-7. Histogram and Cumulative Distribution Function of Annual Risk at LaSalle for Population Dose Within Entire Region.

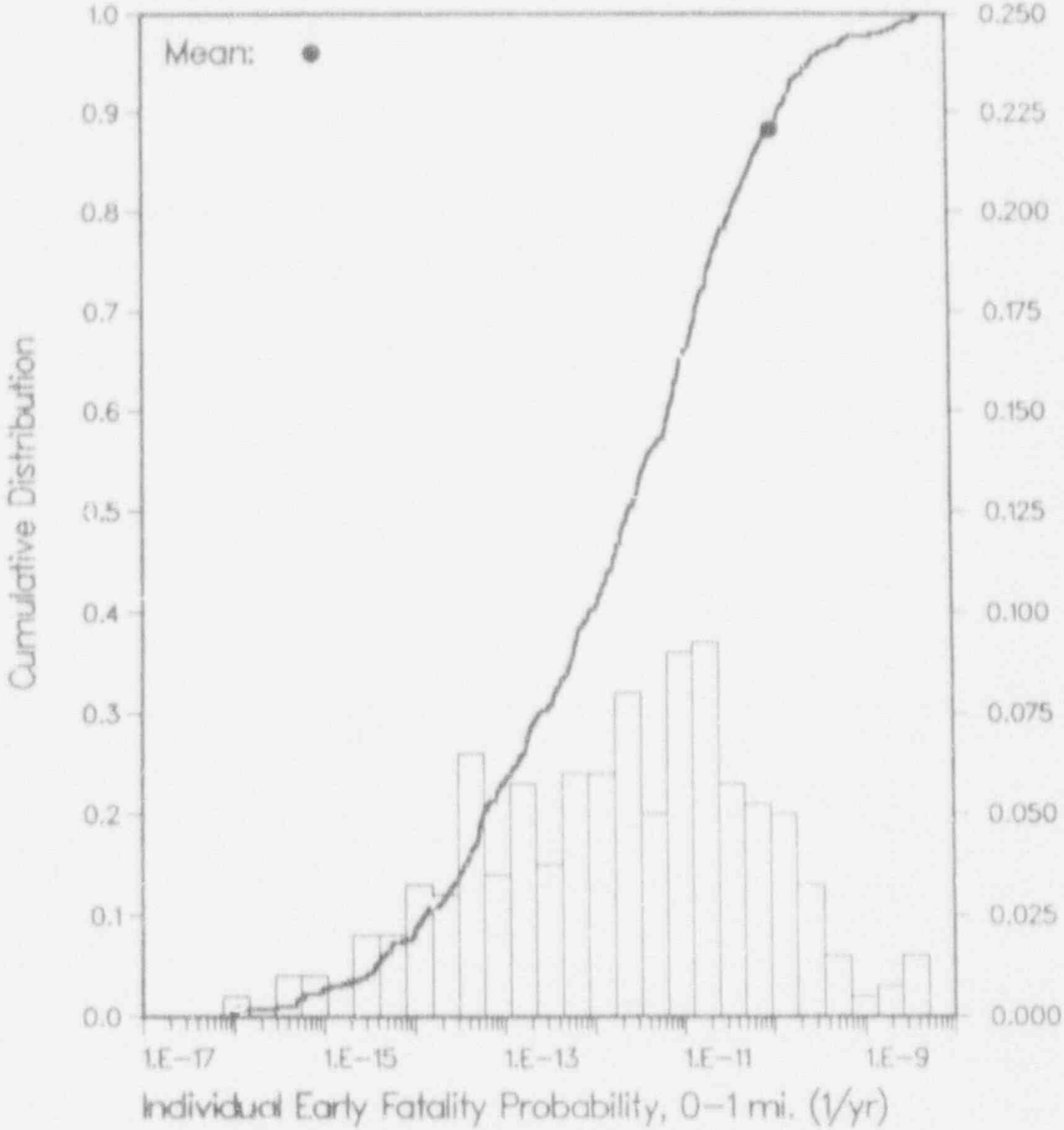


Figure 6.2-8. Histogram and Cumulative Distribution Function of Annual Risk at LaSalle for Individual Early Fatality Probability.

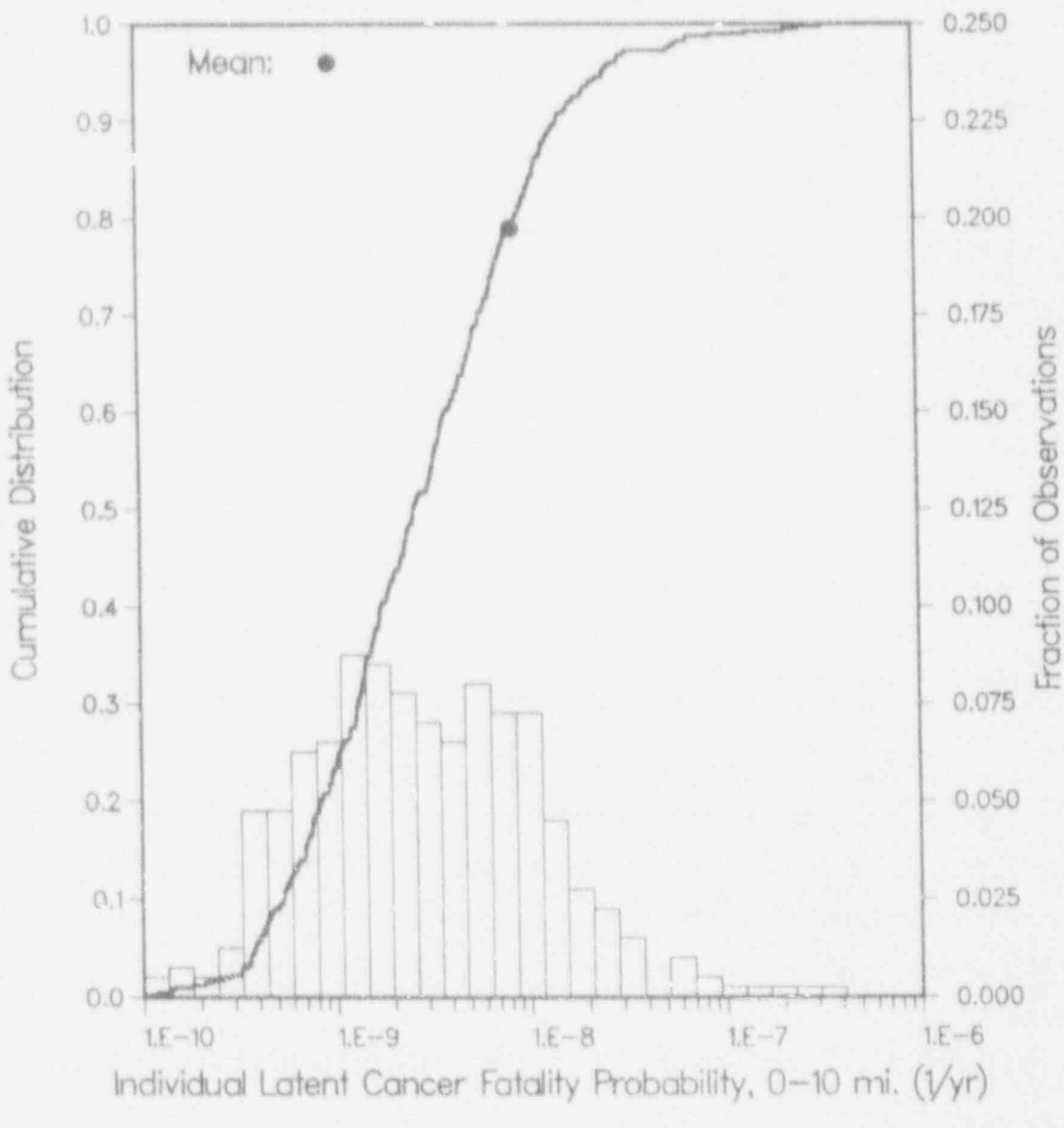


Figure 6.2-9. Histogram and Cumulative Distribution Function of Annual Risk at LaSalle for Individual Latent Cancer Probability.

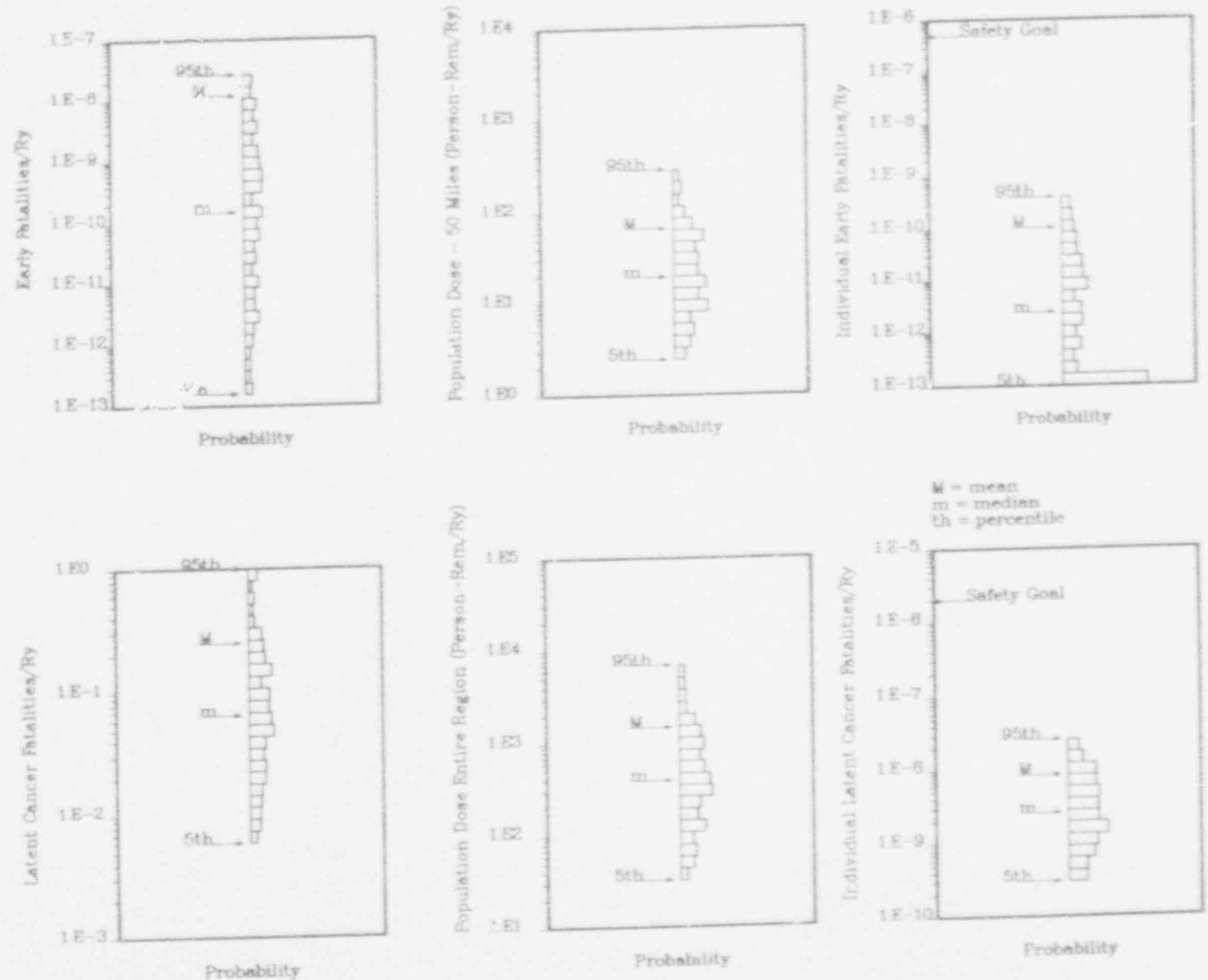


Figure 6.2-10. Distributions of Annual Risk at LaSalle.

values is plotted in Figures 6.2-8, 6.2-9 and the last two frames in Figure 6.2-10. The plots for individual risk in Figure 6.2-10 show that both risk distributions for LaSalle fall well below the safety goals.

A single measure of risk for each consequence measure for the entire sample may be obtained by taking the average values from the histograms in Figures 6.2-4 thru 6.2-9. These measures of risk are commonly called mean risks. The mean risk values for the six consequence measures reported here are displayed in Figures 6.2-4 thru 6.2-9 and Figure 6.2-10. The important contributors to mean risk are considered in Section 6.3.

The offsite risks at LaSalle are relatively low, especially with respect to the safety goals. The mean individual early fatality risk is more than 3 orders of magnitude below the safety goal. Similarly, the mean individual latent cancer fatality risk is slightly more than 2 orders of magnitude below the safety goal. In fact, the entire distributions for these two risk measures lie below the safety goals. The mean values for early fatality risk and for latent cancer fatality risk are 1.2E-08 fatalities/yr. and 0.25 fatalities/yr., respectively.

There are several factors that lead to these low values for risk. First, the total core damage frequency for LaSalle is not especially high. The mean core damage frequency is 1.0E-04/yr. At a first glance this core damage frequency may appear high, however, it must be remembered that this value includes both internal and external initiators. For the Peach Bottom analysis in NUREG-1150,[3] the total core damage frequency from both internal and external events was also about 1.0E-04/yr. using the seismic results from the Lawrence Livermore National Laboratory (LLNL) seismic hazard curve which most closely resembles the methodology used to generate the LaSalle hazard curves (i.e., as opposed to the Electric Power Research Institute (EPRI) hazard curve used for sensitivity analysis).

Second, although the mean conditional probability that the containment's integrity will be compromised during the accident is fairly high, 0.88, there are several features of the LaSalle plant and surrounding area that tend to reduce the consequences. The early fatality risk typically depends on both the magnitude of the release and on the timing of containment failure. If the containment fails early in the accident, it is more likely that a portion of the population in the emergency planning zone (EPZ) will be exposed to the release than if the containment fails after the nearby population has been evacuated. The low early fatality risk can in part be attributed to the fast evacuation of the population around the plant and to the fact that, for many of the possible accidents evaluated in this analysis, containment failure occurs very late into the accident. Since the population in the vicinity of the plant is fairly sparse, this implies a short evacuation delay, a fast evacuation speed, and time for the population to clear the area before the release of radionuclides to the environment. For all of the non-seismic source term groups that were generated during the partitioning process (see section 4.6), the population in the EPZ began evacuation before the start of the release. Thus, the dose received by the evacuating population was generally small and no early fatalities resulted from this fraction of the population. Because of this, the timing of containment failure does not show up explicitly as an important factor in the LaSalle analysis. It would be very difficult to obtain the importance of containment failure timing directly from the results because there is no single parameter by which these effect can be represented; but, instead, a combination of different parameters. Changes in the evacuation assumptions could make the representation of the timing effects more explicit. In addition, as mentioned in the Section 4.6.4 on the partitioning results, some smearing of the timing effects occurred when some source term groups had both positive and negative values for dEVAC, which represents the difference between the release time and the time the population begins to move. Furthermore, there is a threshold effect associated with early fatalities (i.e., to cause an early fatality the release must be of a certain magnitude).

Third, two features of the LaSalle plant that reduce the magnitude of the source term are the suppression pool and the reactor building. In the majority of the accidents analyzed, the in-vessel releases (i.e., an early release) are scrubbed by the suppression pool. In addition to the suppression pool, the reactor building surrounding the LaSalle containment traps a portion of the radionuclides that escape the containment depending on the location of the containment failure.

The latent cancer fatalities are generally associated with the population that is beyond the emergency evacuation zone. Thus, this risk measure is not particularly sensitive to the timing of containment failure, but rather to whether the containment fails or not. Therefore, the latent cancer fatality risk is primarily dependent on the frequency of containment failure and the magnitude of the radionuclide release. Because of the plant features that reduce the magnitude of the source term and the relatively low core damage frequency, the latent cancer fatality risk remains low even though there is a high likelihood that the containment will fail in most of the core damage accidents that were analyzed.

6.3 Contributors to Risk

There exist two distinct ways to calculate contribution to risk. To facilitate their definition, the following quantities are introduced:

rC_j = risk (units: consequences/reactor-year) for consequence measure j,

rC_{ij} = value for rC_j obtained for observation i,

rC_{jk} = risk (units: consequences/reactor-year) for consequence measure j due to PDS group k,

rC_{ijk} = value for rC_{jk} obtained for observation i, and

nLHS = number of observations in the Latin Hypercube Sample (400 for LaSalle).

The result of the first method for computing the contribution to risk is denoted as the fractional contribution to mean risk and abbreviated FCMR. The contribution of PDS group k to the risk for consequence measure j, $FCMR_{jk}$, is defined as the ratio of the annual risk due to PDS group k to the total annual risk. That is, $FCMR_{jk}$ is defined by

$$FCMR_{jk} = E(\ rC_{jk}\ ) \ /\ E(\ rC_j\ ),$$

where E(x) is the expectation value of x and represents the annual value of x in this context. Computationally, $FCMR_{jk}$ is found by use of the relation

$$FCMR_{jk} = [\ \Sigma\ \sigma\Gamma_{ijk}\ /\ nLHS\ ]\ /\ [\ \Sigma\ \sigma\Gamma_{ij}\ /\ nLHS\ ]$$

$$= \Sigma\ \sigma\Gamma_{ijk}\ /\ \Sigma\ \sigma\Gamma_{ij},$$

where the summations are from i = 1 to i = nLHS.

The result of the second method for computing contribution to risk is denoted the mean fractional contribution to risk and abbreviated MFCR. The contribution of PDS group k to the risk for consequence measure j, $FCMR_{jk}$, is defined as the annual value of ratio of the risk due to PDS group k to the total risk. That is:

$$MFCR_{jk} = E(\ rC_{jk}\ /\ rC_j\ ).$$

Computationally, $MFCR_{jk}$ is found by use of the relation

$$MFCR_{jk} = \Sigma\ (\ \sigma\Gamma_{ijk}\ /\ rC_{ij}\ )\ /\ nLHS,$$

where the summation again is from i = 1 to i = nLHS.

For FCMR, the averaging over the LHS observations is done before the ratio of group risk to total risk is formed; for MFCR the averaging over the observations is done after the ratio of group risk to total risk is formed.

The reproducibility of the distributions for the integrated risk analyses and the stability of the two measures for calculating contribution were investigated in NUREG-1150. The results of this investigation are briefly described below. A second sample was run through the entire integrated risk analyses for Surry.[4] The second sample is just as valid as the first sample and differs from the first sample only in the fact that a different random seed was used in the LHS program. Therefore, the differences in the results between the two samples are an indication of the robustness of the analysis methods. In addition, a comparison of the two samples provides an indication of which method of calculating the contribution to risk tends to be more stable. The results from the second sample and a comparison of the two samples are presented in the Surry report. Several insights gleaned from this comparison are summarized below.

First, considering the early fatality and latent cancer fatality risk distributions, the agreement between the two samples is remarkably good. This agreement indicates that the methods used for this integrated risk analysis are stable. Differences between the two samples can generally be found at the extremes of the distribution which is not surprising since the extremes are determined by a relatively few observations. Second, the variations between samples are higher for FCMR than for MFCR, indicating that MFCR is a more robust measure of the risk results than FCMR.

The FCMR measure of the contribution to mean risk tends to be less stable than the MFCR measure because the annual risk for each observation is typically dominated by a few APBs which have both high frequency and high source terms and the mean risk is dominated by a few observations which have very large values of annual risk. For example, 85% of the mean early fatality risk for LaSalle is determined by only 20 observations (5 [illegible] observations). While the sample as a whole is reproducible, the [illegible] observations that control mean risk are generally not reproducible [illegible] it is [illegible] exact nature of these 20 or so observations that dete[illegible] the contributors to mean risk, it is not surprising that FCMR is not [illegible] robust measure of the entire risk analysis. The MFCR tends to represent the contributions from the whole distribution while FCMR tends to emphasize contributions from the upper tails of the distributions.

I[illegible] is an appropriate place to remind the reader of an important caveat: a mean value is a summary measure and information is lost in generating it. Thus, considerable caution should be used in drawing conclusions solely from mean values. A mean is obtained by reducing an entire distribution to a single number.

Table 6.3-1 gives the values of FCMR and MFCR for the summary PDS groups. Even though the measures for determining the contributors to mean risk are only approximate, the types of accidents that are the largest contributors to offsite risk at LaSalle are clear. For all of the consequence measures, the risk is dominated by the Fire PDS group and the Transient PDS group. The contributions of the summary PDS groups to the mean core damage frequency are also presented in this table. Not surprisingly, these groups are also the dominant contributors to the core damage frequency. The LOCA and Transient LOCA PDS groups, on the other hand, are very minor contributors to the risk.

The contributions of the summary accident progression bins (APBs) to mean risk can also be computed in two ways. Table 6.3-2 displays the results of these calculations. The probabilities of the summary APBs, conditional on core damage, are also presented in this table.

For early fatality risk and individual risk of early fatality within one mile, the risk is dominated by accidents that progress to vessel breach and

Table 6.3-1
Fractional PDS Contributions (in percent) to Annual
Risk at LaSalle Due to All Initiators

Summary PDS Group	Method	Core Damage	Early Fatalities	Latent Cancer Fatalities	Population Dose 50 miles	Population Dose Region	Ind. E. F. Risk-1 mile	Ind. L.C.F. Risk-10 mile
Seismic	FCMR	0.7	5.2	1.3	1.1	1.2	2.9	1.0
	MFCR	1.5	4.6	2.3	2.1	2.3	4.4	2.2
Fire	FCMR	53.3	46.2	63.1	60.6	62.7	71.8	61.9
	MFCR	23.8	22.0	37.8	35.9	37.5	23.7	37.5
Flood	FCMR	3.0	0.6	1.7	2.0	1.7	0.6	1.7
	MFCR	11.0	9.1	6.5	7.4	6.6	9.3	6.8
ATWS	FCMR	0.2	<0.1	0.2	0.1	0.2	<0.1	0.1
	MFCR	0.5	10.0	0.7	0.6	0.7	9.8	0.5
LOCA	FCMR	0.0	<0.1	<0.1	<0.1	<0.1	<0.1	<0.1
	MFCR	0.0	<0.1	<0.1	<0.1	<0.1	<0.1	<0.1
Transients	FCMR	42.3	46.2	32.1	51.7	32.5	23.4	33.5
	MFCR	62.4	52.4	50.2	34.4	50.4	51.2	50.7
Trans-LOCA	FCMR	0.5	0.8	0.7	0.7	0.7	1.0	0.7
	MFCR	0.8	1.4	1.5	1.3	1.5	1.5	1.4

Table 6.3-?
Fractional APB Contributions (in percent) to Annual Risk at LaSalle Due to All Initiators

Summary Accident Progression	Method	Prob. Cond. CD	Early Fatalities	Latent Cancer Fatalities	Population Dose 50 miles	Population Dose Region	Ind. E. F. Risk-1 mile	Ind. L.C.F. Risk-10 mile
VB, Early CF, RPV at Low Press.	FCMR	-	18.2	28.7	23.2	27.9	26.4	20.1
	MFCR	0.15	26.8	19.2	18.2	19.3	28.5	17.1
VB, Early CF, RPV at High Press.	FCMR	-	17.9	42.7	43.7	42.6	34.3	51.2
	MFCR	0.19	13.6	25.5	24.5	25.3	[illegible]	25.8
VB, Late CF	FCMR	-	30.0	6.6	8.6	7.1	8.5	6.4
	MFCR	0.09	10.3	[illegible]	10.4	10.4	[illegible]	10.5
VB, Early or Late Venting	FCMR	-	34.7	20.6	22.4	20.9	[illegible]	20.3
	MFCR	0.37	44.3	41.9	[illegible]	42.1	46.9	42.8
VB, no CF	FCMR	-	<0.1	<0.1	<0.1	<0.1	<0.1	<0.1
	MFCR	0.06	<0.1	<0.1	<0.1	<0.1	<0.1	<0.1
No VB, CF	FCMR	-	<0.1	<0.1	<0.1	<0.1	<0.1	<0.1
	MFCR	0.0	<0.1	<0.1	<0.1	<0.1		<0.1
No VB, Vent	FCMR	-	0.2	0.5	0.9	0.5	0.4	0.9
	MFCR	0.09	4.3	1.9	2.9	2.0	4.5	2.7
No VB, No CF, and No Vent	FCMR	-	<0.1	<0.1	<0.1	<0.1	<0.1	<0.1
	MFCR	0.06	<0.1	<0.1	<0.1	<0.1	<0.1	<0.1

that involve loss of containment integrity. Vessel breach is important because it affects the magnitude of the source term. If the core damage process is arrested, there are only in-vessel releases which are likely to be scrubbed by the suppression pool. If, on the other hand, vessel breach occurs, there are both in-vessel releases and ex-vessel releases. The releases from core-concrete interactions, which is an ex-vessel release, can be a significant contributor to the total release. Loss of containment integrity is important because, if the containment remains intact, the release to the environment through normal leakage is extremely small and is a negligible contributor to risk. For the non-seismic PDS groups, the early fatalities are associated with the 0.5% of the population that does not evacuate and in some cases with the population beyond the emergency planning zone (EPZ). For the low-g seismic PDS group, the early fatalities are only associated with the population beyond the EPZ (for this PDS all of the population within the EPZ evacuates). Thus, as mentioned previously, the timing of the release did not show up as an important factor for early fatalities because the dose received by the evacuating population is not large enough to result in an early fatality. What does show up as important is the frequency of containment failure and the magnitude of the source term. Thus, the first four APBs in Table 6.3-2, which all involve vessel breach and containment failure, are significant contributors to the early fatality risk.

Latent cancer fatalities depend primarily on the frequency of containment failure (or venting) and the total amount of radioactivity released. Thus, like early fatality risk, accidents that progress to vessel failure and involve a loss of containment integrity are the dominant contributors to the latent cancer fatality risk. Again, the timing of containment failure does not show up as particularly important.

6.4 Contributors to Uncertainty

The uncertainty analysis described in this section follows the methodology used in the NUREG-1150 study. The uncertainty analysis was performed in this manner to be consistent with the NUREG-1150 study and because resource and scheduling limitations precluded further methodology development. While this analysis represents the state-of-the-art for Level III regression analysis, we feel that there are significant shortcomings with this method. A discussion of some of these problems is presented below.

The uncertainty in the risk results is represented in the model in two different ways. The first way is via the branching of the questions in the APET and the second way is via the sampling of parameter distributions in the LHS sample. The uncertainty resulting from the LHS sampling of the parameter distributions is, itself, represented in the model in three different ways as discussed below.

The branching in the APET represents those uncertainties that can make a difference in the nature of the outcome of the accident progression, can improve the representation of the effects of one phenomena on another, and can improve the quantification of the probabilities of the various possibilities. The result is a multitude of different paths through the APET each representing a different possible future evolution of the accident. Many of these paths will also have different outcomes for the consequence measures; but, because some of the branching in the APET does not affect those characteristics of the accident progression important to the determination of the consequences, many will also have very similar consequences (i.e., we group the paths through the APET into accident progression bins for the source term analysis and then further into source term groups for the consequence analysis). In the Level I analysis, since the only outcome of interest is core damage, all paths that lead to core damage have the same consequence "core damage" (i.e., we are not interested in the level of core damage; this is a degenerate case).

The uncertainty represented via the LHS sampling of the parameter distributions is incorporated into the model in three different ways. The first way is via the additional uncertainty that we have in the probabilities assigned to the various question branches in the APET. For some of the branches, the probability of the branches is not very well known and distributions are generated which are then sampled using the LHS formalism. The second way is via various parameters that are tracked by the APET and used to determine the the probability of occurrence or magnitude of various phenomena during the accident progression. This may result in different accident progression bins for different sample members. The third way is via the uncertainty in the parameters used to calculate the source term for each of the accident progressions resulting from the analysis of the APET. This results in different source terms for an accident progression bin that appears in several different sample members.

The end result of the analysis for a particular risk measure is a set of outcomes (O_i), a set of frequencies for these outcomes (F_{ij}), and a set of consequences for each outcome (C_{ij}) where each outcome has many possible frequencies and consequences arising from the variability in the LHS sample. Because the weather variability is not included in the LHS sample, it is represented in the analysis as an increase in the possible set of outcomes (i.e., the weather is treated the same way possible accident outcomes are treated in the APET; that is, each outcome from the APET is effectively split into 2048 outcomes, one for each weather scenario evaluated in the consequence analysis). The variation in the outcomes and the variation in the frequencies and consequences of the outcomes arising from the LHS sample all together constitute the uncertainty in a risk measure.

For this analysis, this set of results is represented as a set of exceedance frequency curves (CCDFs) shown in Appendix F and in Figures 6.2-1 thru 6.2-3. These figures provide information on the frequency at which values for individual consequence measures will be exceeded. Specifically, mean, median, 5th percentile, and 95th percentile values are shown for these exceedance frequencies. As mentioned in Section 6.2, the uncertainty represented by the branching in the APET (i.e., the fact that many different accidents are possible) and by the weather variability is displayed by the set of points constituting one of the curves and the uncertainty represented by the LHS sampling of the parameter distributions is displayed by the set of different curves.

There is considerable disagreement within the PRA community on the meaning of the uncertainty being represented in these two different ways. One camp views each CCDF curve as representing stochastic variability and the family of curves as representing modeling uncertainty. There are some problems with this interpretation because one can easily represent any uncertainty either way (i.e., either as branching in the APET or as variation in the LHS sample) and so can switch uncertainties from stochastic to modeling and visa versa at will. Since many of the issues represented in the APET are so complicated that it is impossible to say that a particular issue is only stochastic or only modeling based, the mode of representation of a particular issues in the APET will, therefore, be determined by the analyst. It is clear to us, therefore, that the fundamental issue of what the APET represents and how a particular issue should be represented in the model has not yet been adequately addressed.

An example of both of these issues is the representation of the hydrogen concentration in the containment during core damage. For a given set of initial conditions determined by the answers to previous questions in the APET, the effects of the uncertainty in the hydrogen concentration can be represented in the APET in different ways. First, it could be represented by a question which has branches that each represent a range of hydrogen concentrations (determined by the level of resolution the analyst considers important). The conditional probability of each branch could either be fixed or uncertain. Second, it could be represented by a parameter for which a value of the hydrogen concentration is selected, some calculation is performed, and a decision is made as to which branch to take.

For the case where the branch probabilities are fixed, one requires a single distribution on the hydrogen concentration which is then integrated to obtain the conditional probability of each range of hydrogen concentrations. This distribution represents both the stochastic variability due to all the initial and boundary conditions not explicitly included in the model but which might affect the amount of hydrogen produced in-vessel and the modeling uncertainties due to uncertainties in the representation of the core melt process in the thermal/hydraulic codes, limitations of the physics model used in the codes, and the limited representation of the plant used in the input decks to these codes. For the case where the branch probabilities are uncertain, one requires a family of curves where each curve can have some level of stochastic and modeling uncertainty intermixed and the family represents any additional uncertainty (modeling or stochastic) not included in an individual curve. The amount of separation depends on how the analyst views the model. In one case, the uncertainty contributes to the outcomes and would be called "stochastic"; in the other, the uncertainty represents sample to sample variability and would be called "modeling." In order to completely separate the stochastic and modeling uncertainties, the experts would have to explicitly define sets of consistent modeling assumptions and then assess the stochastic variability for each set of assumptions. We contend that not only didn't the experts in NUREG-1150 do this; but, that this would be extremely difficult in practice because all the unique sets of assumptions that could significantly affect the results would have to be enumerated and this is virtually impossible.

The regression analysis presented in this section was developed at the end of the NUREG-1150 analysis when people were looking for a way to make statements about the importance of various issues to the uncertainty in the risk results. It was felt that numerical sensitivity analysis techniques might provide a systematic way of investigating the uncertainties in the risk results and determining the dominant contributors to the uncertainty in risk. However, it is difficult (but not impossible) to represent the contribution to uncertainty from the various outcomes (i.e., branching or weather) in a regression model. Also, the fact that the problem is really highly non-linear means that a non-linear regression model should be used. However, it takes a substantial effort and some experimenting to find an appropriate non-linear model. The analyses performed here and in NUREG-1150, therefore, use simple linear models that only reflect the contribution of the variables that were sampled using LHS (i.e., the uncertainty in the family of curves) and not the contribution to uncertainty represented by the variables that contributed to the enumeration of the different outcomes (i.e., the curve itself). The results of this analysis should, therefore, be viewed somewhat skeptically. The results will only tell one the dominant contributors to the uncertainty in risk averaged over all the possible outcomes and conditional on the accuracy of a linear model. Before one makes any decisions based on these results, other analysis techniques and engineering judgement should be used to validate the results.

As the curves in Figure 6.2-1 thru 6.2-3 and in Appendix F show, there is significant uncertainty in the frequency at which a given consequence value will be exceeded. Due to the complexity of the underlying analysis and the concurrent variation of a large number of variables within this analysis, it is difficult to ascertain the cause of this uncertainty by a simple inspection of the results.

This section presents the results of using simple linear model in the variables appearing in the LHS sample with regression-based sensitivity analysis techniques to examine the variability in the consequences of accidents at LaSalle. The dependent variable is the annual risk (units: consequences/year) for each consequence measure. For a given observation in the sample, this variable is obtained by multiplying each consequence value by its frequency and then summing these products. This variable can be viewed as the result of reducing each of the curves in Figure F.1 to a single number.

The uncertainty analysis techniques used in this study can be viewed as creating a mapping from analysis input to analysis results. The variables sampled in the generation of this mapping are presented in Tables 2.6-1, 3.2-3, and 4.4-1. These variables are the independent variables in the sensitivity studies presented in this section. Variables that are fully correlated to each other are treated as a single variable in sensitivity analysis. For example, in Table 3.2-3 the variables H2INVES1 through H2INVES6 ar all correlated and, therefore, in the sensitivity analysis they are treated as a single variable (i.e., H2INVES).

Sensitivity analysis results for the six consequence measures used to express risk are presented in Table 6.4-1. This table contains the results of performing a stepwise linear regression on the risk as expressed by: early fatalities, latent cancer fatalities, population dose within 50 miles, population dose within the entire region, individual risk of early fatality within 1 mile, and individual risk of latent cancer fatality within 10 miles. The statistical package SAS[5] was used to perform the regression.

For each consequence measure, Table 6.4-1 lists the variables in the order that they entered the regression analysis, gives the sign (i.e, positive or negative) on regression coefficients for the variables in the final regression model, and shows the R^2 values that result with the entry of successive variables into the model. The order that the variables enter the model is based on the fraction of the uncertainty that is explained by each of the variables. First, the fraction of the uncertainty that is explained by each variable individually is determined. The variable with the largest fraction is the first variable that is entered into the model. Next, the variability that is explained by this variable is removed and the process is repeated with the remaining variables. The tendency of a dependent variable to increase and decrease with an independent variable is indicated by a positive regression coefficient, and the tendency of a dependent variable to decrease when an independent variable increases is indicated by a negative regression coefficient.

The regression analyses for early fatalities and individual risk of early fatality within 1 mile only account for about 49% of the observed variability. The nine independent variables that account for this variability are those that determine the frequency and the magnitude of the release. The top four variables (the release from CCI, the retention in the containment, reactor building decontamination factor, and the late release of iodine) determine the magnitude of the source term. The regression analyses for the other four consequence measures are somewhat more successful as they are able to account for about 60% of the variability. For latent cancer fatality risk, 18 independent variables account for this variability. Twelve of these 18 variables are from the Level I analysis and are used to determine the frequency of core damage. Five variables are from the source term analysis and the remaining variable is from the accident progression analysis.

Table 6.4-1
Summary of Regression Analyses for
Annual Risk at LaSalle (All Initiators)

Step	Early Fatalities			Latent Cancer Fatalities			Population Dose--50 miles		
	VAR[a]	RC[b]	R^2[c]	VAR	RC	R^2	VAR	RC	R^2
1	FCCI	Pos	0.12	CFM-114	Pos	0.13	CFM-114	Pos	0.16
2	FCONC	Pos	0.22	SUR-006-L	Pos	0.23	SUR-06-L	Pos	0.29
3	RBDF	Neg	0.31	SUR-022-R	Pos	0.32	SUR-22-R	Pos	0.39
4	FLTI	Pos	0.37	FCONC	Pos	0.37	IE-LOSP	Pos	0.41
5	CFM-114	Pos	0.41	RBDF	Neg	0.41	CFM-37	Pos	0.43
6	FCOR	Neg	0.44	FCOR	Neg	0.44	RBDF	Neg	0.45
7	ALPHA	Pos	0.46	SUR-27-R	Pos	0.46	SUR-27-R	Pos	0.47
8	DFCAV	Pos	0.48	FCCI	Pos	0.48	FCONC	Pos	0.49
9	MODE VB	Pos	0.49	IE-LOSP	Pos	0.50	AC RECOV	Pos	0.50
10				CFM-37	Pos	0.51	LPCI-BET	Pos	0.52
11				AC RECOV	Pos	0.53	N-IStmEx	Neg	0.53
12				LPCI-BETA	Pos	0.54	ExStmEx	Pos	0.54
13				LAM-CR	Pos	0.55	LAM-CR	Pos	0.55
14				No-IStmEx	Neg	0.56	LAM-SVGR	Pos	0.56
15				IE-VAL-R	Pos	0.57	IE-VAL-R	Pos	0.57
16				LAM-SVGR	Pos	0.58	FCOR	Neg	0.58
17				LAM-AUX	Pos	0.59	SUR-3-R	Neg	0.58
18				DFCPA	Neg	0.60	FCCI	Pos	0.59
19							No VB	Neg	0.60

Table 6.4-1 (Concluded)
Summary of Regression Analyses for
Annual Risk at LaSalle (All Initiators)

	Population Dose Entire Region			Individual Early Fat. Risk 0-1 mile			Individual Latent Can. Fat. Risk 0-10 mi.		
Step	VAR[a]	RC[b]	R^2[c]	VAR	RC	R^2	VAR	RC	R^2
1	CFM-114	Pos	0.14	FCCI	Pos	0.11	LEAKTRB	Pos	0.21
2	SUR-6-L	Pos	0.23	RBDF	Neg	0.21	CFM-114	Pos	0.34
3	SUR-22-R	Pos	0.32	FCONC	Pos	0.30	SUR-22-R	Pos	0.46
4	FCONC	Pos	0.36	FLTI	Pos	0.37	SUR-27-R	Pos	0.47
5	RBDF	Neg	0.41	CFM-114	Pos	0.41	IE-LOSP	Pos	0.49
6	FCOR	Neg	0.44	FCOR	Neg	0.44	CFM-37	Pos	0.51
7	SUR-27-R	Pos	0.46	ALPHA	Pos	0.46	AC RECOV	Pos	0.52
8	FCCI	Pos	0.48	DFCAV	Pos	0.48	LPCI-BET	Pos	0.54
9	IE-LOSP	Pos	0.50	SUR-25-R	Pos	0.49	NoIStmEx	Neg	0.55
10	CFM-37	Pos	0.51				LAM-SVGR	Pos	0.56
11	AC RECOV	Pos	0.53				SUR-3-R	Neg	0.57
12	LPCI-BET	Pos	0.54				IE-VAL-R	Pos	0.58
13	LAM-CR	Pos	0.55				LAM-CR	Pos	0.59
14	N-IStmEx	Neg	0.56						
15	IE-VAL-R	Pos	0.57						
16	LAM-SVGR	Pos	0.58						
17	LAM-AUX	Pos	0.59						
18	NO VB	Neg	0.60						

[a] Variables listed in the order that they entered the regression analysis.
[b] Sign on the regression coefficients (RCs) in final regression model.
Pos: Increase in independent variable increases dependent variable
Neg: Increase in independent variable decreases dependent variable
[c] R^2 values with the entry of successive variables into the regression model.

As demonstrated above, the regression analysis results do not explain a substantial portion of the uncertainty. We believe that this arises from the fact that a substantial portion of the uncertainty in the annual risk result comes from the variation in the nature of the accident progression bins contributing to the risk for a particular sample member (i.e., along a curve) and that it is, therefore, impossible to explain all of the uncertainty in the annual risk by just examining the LHS sample variability. As you go from curve to curve, the nature and weighting of the various accident progression bins contributing to the result change and some substantial part of the uncertainty results from the so-called "stochastic" variability which is not included in the regression analysis. The only case where most of the uncertainty can be explained by these regression techniques is the case where one set of accident progression bins is so dominant that the variation in the bins from sample to sample (i.e., curve to curve) is not significant.

6.5 References

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7.0 ANALYSIS AND METHODOLOGY INSIGHTS AND CONCLUSIONS

7.1 Analysis Conclusions and Limitations

7.1.1 Conclusions Arising From the Plant Analysis

7.1.1.1 Plant Damage State Definition Results

For the LaSalle analysis, 30 plant damage states were defined with some 16 additional sub-plant damage states included in the analysis. These PDSs resulted from a detailed examination of all of the cut sets from the top 50 dominant accident sequences from the Level I analysis.[1]

The total core damage frequency distribution from the Level I analysis had a mean value of 1.01E-04/R-yr. with a 5th percentile of 5.34E-06/R-yr., a median of 2.92E-05/R-yr., and a 95th percentile of 2.93E-04/R-yr. The Level II/III LHS sample resulted in a mean of 1.04E-04/R-yr., a 5th percentile of 5.74E-06/R-yr., a median of 2.76E-05/R-yr., and a 95th percentile of 3.25E-04/R-yr. The difference between the two distributions is less than 10%. Given that the Level I analysis used 270 primary variables and that the Level II/III analysis used only 103 of the Level I variables for the uncertainty analysis (the other Level I variables were fixed at their mean values), this is a very good approximation to the Level I distribution. Individually, the PDS distributions may show more variation than this; but, the dominant PDSs are very close to the Level I results as a result of the variable selection process described in Chapter 2.

The dominant plant damage states are IT2, FI5, and FL2 with 0.368, 0.107, and 0.105 mean fractional contributions of the total core damage frequency, respectively. IT2 is a transient-initiated short-term station blackout PDS with core damage beginning at about 80 minutes after the accident initiation. At the onset of core damage, containment failure has not yet occurred and venting is recoverable if AC power is restored. There are three sub-PDSs: (1) ADS and almost all injection systems are recoverable if AC power is restored, (2) ADS is available during the core damage process and almost all injection is recoverable if AC power is restored, and (3) ADS is available during the core damage process but only MFW, CDS, and RCIC are recoverable if AC power is restored.

FI5 is a fire-initiated accident resulting directly in a partial loss of containment heat removal. Random failures complete the loss of containment heat removal, and a long-term loss of containment heat removal sequence results. Primary injection into the RPV is available and works mainly using the HPCS system although other systems may be used for some part of the time. Containment pressurizes, RCIC isolates at 25 psig, the ADS valves reclose at 85 psig, and any low pressure injection fails. High pressure injection continues, and the containment pressurizes until structural failure of the containment to the reactor building occurs anywhere from 140 to 275 psig (mean value 191 psig). The severe environment created in the reactor building by the blowdown results in failure of any remaining injection systems, and core damage occurs with a failed containment.

FL2 is an internal flood initiated by a service water pipe break on the ground floor of the reactor building. The flood fails all injection systems except diesel-driven firewater which is not used in time to prevent core damage. ADS and containment venting are available. This sequence is a short-term loss of all injection sequence.

The mean fractional contributions to the total core damage frequency of the different accident classes are: seismic 1.5%, fire 2[illegible].8%, flood 11%, ATWS 0.5%, LOCAs 0.2%, transients 62.4%, and transients-induced LOCAs 0.8%.

7.1.1.2 Accident Progression Analysis Results

Based on the results presented in Section 3.4 the following observations can be made with regard to the accidents analyzed in this study. The majority of the accidents analyzed will proceed to vessel failure. Although notable, the mean conditional probability of core damage arrest is fairly small, approximately 0.13. The probability of core damage arrest is driven by the recovery of AC power for the short-term station blackouts and the lack of available or recoverable injection systems for the other accidents. Given that core damage occurs, it very likely that the containment's integrity will be compromised during the course of the accident by either containment failure or by containment venting. The mean conditional probability that the containment will remain intact throughout the accident is only 0.12. Furthermore, it is fairly likely that the containment will fail early in the accident--the mean conditional probability of early containment failure is 0.33. It is also likely that the operators will vent the containment during the accident--the mean conditional probability of containment venting is 0.46. Given that core damage occurs, it is likely that the core debris released from the vessel will participate in core-concrete interactions. The mean probability of CCI, conditional on core damage, is 0.77. Thus, the potential exists for a large release late in the accident.

The events that result in containment failure before core damage are slow pressurization events that result from the accumulation of steam and noncondensibles during accidents in which containment heat removal is lost or inadequate (i.e., long-term loss of containment heat removal accidents, long-term station blackout accidents, and ATWS accidents). Events that result in containment failure around the time of vessel breach include fast pressurization of the containment from loads accompanying vessel breach (i.e. DCH, ex-vessel steam explosions, RPV blowdown), Alpha mode events, drywell failure induced by reactor pedestal failure, and cavity drain line isolation failure. Late in the accident, the events that result in containment failure include the slow pressurization of the containment from the steam and noncondensibles generated during CCI and failure of the reactor pedestal caused by concrete erosion during CCI.

7.1.1.3 Source Term Analysis Results

The source term results showed that the pressure of the RPV does not significantly affect the source term for a particular accident progression if vessel breach occurs. This is due to the differences in the fraction of a species released from the fuel before vessel breach being negated when the vessel breaches and the radionuclides in the vessel revolatilize (i.e., low release before vessel breach results in high release after vessel breach and vice versa).

The release path through which the radionuclides pass was determined to be important. The path that resulted in the highest release to the environment is through the wetwell above the water line. If the cavity floor has failed, radionuclides may leave the containment without being scrubbed by sprays or the suppression pool. The venting pathway is also through the wetwell above the water line. In most cases, containment failures in the drywell or drywell head were accompanied by successful operation of containment sprays which significantly reduced the amount of radionuclides being released. Also, for many of the cases for which containment failure in the drywell head occurred, core damage did not occur because severe environments were not created in the reactor building and the injection systems did not subsequently fail.

The analysis showed that late iodine revolatilization results in significant releases due to much of the iodine being scrubbed by the pools initially and revolatilized later when the removal mechanisms are not as effective. For this reason, the modeling of late iodine revolatilization may need to be investigated further.

The regression analysis shows that much of the uncertainty in the final risk results is due to the uncertainty in the source term parameters. This may be due to the fact that the distributions being used were elicited for cases with large uncertainties in the initial and boundary conditions. Some of this uncertainty might be eliminated by a more detailed case structure in the APET and LASSOR codes with new distributions for these more specific cases.

7.1.1.4 Partitioning Results

The new partitioning process resulted in a total of 97 non-zero partitions being defined. There were 20 high-g seismic, 23 low-g seismic, and 54 fire, flood, and internal partitions defined. Out of 75,680 source terms only 1,860, or less than 2.5%, of the source terms were not partitioned directly and had to be distributed to the nearest defined partition. All of the partitions defined were important either to early fatalities, latent cancers, or frequency.

The result of the new partitioning process was that most of the partitions were defined using many, if not all, of the parameters selected and that each partition is important to the final result (i.e., only one empty partition was defined). The total number of partitions is less than that produced in the old partition process[2] used in NUREG-1150; but, the level of resolution is substantially increased as the source terms are grouped much more homogeneously. The risk distribution is more accurately modeled since the increased homogeneity of the partition groups allows a more accurate representation of the final consequences for each group for input into the consequence calculation.

7.1.1.5 Risk Analysis Results

The offsite risk at LaSalle is relatively low, especially with respect to the NRC safety goals (i.e., individual early fatality risk safety goal = 5.0E-07/R-yr., individual latent cancer fatality risk safety goal = 2.0E-05/R-yr.). The mean individual early fatality risk, 1.1E-10/R-yr, is more than three orders of magnitude below the safety goal. Similarly, the mean individual latent cancer fatality risk, 8.5E-09/R-yr, is slightly more than two orders of magnitude below the safety goal. In fact, the entire distributions for these two risk measures lie below the safety goals. The mean values for early fatality risk and for latent cancer fatality risk are 1.2E-08/R-yr and 0.25/R-yr, respectively.

For all of the consequence measures, the risk is dominated by the Fire PDS group and the Transient PDS group. Not surprisingly, these groups are also the dominant contributors to the core damage frequency. The LOCA and Transient-Induced LOCA PDS groups, on the other hand, are very minor contributors to the risk. The Seismic, ATWS, and Flood groups are intermediate contributors.

For early fatality risk and individual risk of early fatality within one mile, the risk is dominated by accidents that progress to vessel breach and that involve loss of containment integrity. Vessel breach is important because it affects the magnitude of the source term. If the core damage process is arrested, there are only in-vessel releases which are likely to be scrubbed by the suppression pool. If, on the other hand, vessel breach occurs, there are both in-vessel releases and ex-vessel releases. Releases from core-concrete interaction, which is ex-vessel, can be a significant contributor to the total release. Loss of containment integrity is important because, if the containment remains intact, the release to the environment through normal leakage is extremely small and is a negligible contributor to risk. For this analysis, the timing of the release did not significantly influence risk. As discussed in Section 6.2, there are two reasons for this. The first is that the timing of containment failure is only indirectly represented in the risk results since the evacuating population almost always leaves before the release, whenever it occurs, and an analysis of the results will not show a significant variation due to containment failure timing. The second is that some deficiencies in the partitioning process averaged some of the few cases where the population did not evacuate before the release with cases where they did. This reduced the impact of the cases where the population did not evacuate before the release on the risk results.

Latent cancer fatalities depend primarily on the frequency of containment failure (or venting) and the total amount of radioactivity released. Thus, like early fatality risk, accidents that progress to vessel failure and involve a loss of containment integrity are the dominant contributors to the latent cancer fatality risk. The timing of containment failure is not particularly important for this risk measure.

There are several factors that lead to these low values for risk. First, while the core damage frequency for LaSalle, mean = 1.0E-04/R-yr., appears high when compared to the NUREG-1150 plants, it must be remembered that this is an integrated value and represents not only accidents from internal initiators but also accidents from fire, flood, and seismic initiators. In fact, if ones adds up the mean core damage frequencies for the internal, fire, and seismic analyses at the Peach Bottom plant analyzed in NUREG-1150,[3] the mean core damage frequency is 1.4E-04/R-yr. with the LLNL seismic hazard curve and 6.6E-05/R-yr. with the EPRI seismic hazard curve. This is in the same range as the LaSalle core damage frequency and this still does not include internal flooding while the LaSalle analysis does.

Second, although the mean conditional probability that the containment's integrity will be compromised during the accident is fairly high, 0.88. The fact that about half of these are predominantly late (0.5) failures, the fast evacuation speed of the population around the plant, the fact that many of the sequences which had containment failure at or before vessel breach are long-term sequences with containment failure still many hours into the accident, and the impact of other plant features that reduce the magnitude of the source term; all tend to reduce the consequences. Two plant features that reduce the source term are the suppression pool and the reactor building. In the majority of the accidents analyzed, the in-vessel releases (i.e., an early release) are scrubbed by the suppression pool. In addition to the suppression pool, the reactor building surrounding the LaSalle containment also traps a portion of the radionuclides that escape the containment.

For all of the non-seismic source term groups that were generated during the partitioning process (see Section 4.6), the population in the emergency planning zone, EPZ, began evacuation before the start of the release. Thus, the dose received by the evacuating population was generally small and no early fatalities resulted from this fraction of the population. Because of the rapid evacuation, the timing of containment failure does not show up as important in the LaSalle analysis (even though it is since, if the population could not evacuate before the containment failure, the risk results would be significantly higher). Changes in the evacuation assumptions could change this conclusion.

7.1.2 Modeling Limitations/Technical Deficiencies

7.1.2.1 Accident Progression Analysis Deficiencies

After the LaSalle analysis was completed, an error in the APET binner was discovered. The following is a description of this error and the potential impact it may have on the results.

Following vessel breach, CCI erodes the pedestal wall. Depending on the rate of erosion, the time at which the pedestal fails can vary. Failure of the pedestal can induce drywell failure. In order to determine the probability of a drywell failure shortly after vessel breach, the probability of early pedestal failure must be combined with the conditional probability that, given pedestal failure, the drywell will fail. The conditional probability of drywell failure induced by pedestal failure is 0.175. In the binner, however, the logic for the bin representing the conditional probability of drywell failure given pedestal failure was inadvertently dropped. This effectively resulted in a drywell failure probability of 1.0 conditional on the failure of the pedestal. This error only occurs for scenarios where the pedestal fails within 6 hours after vessel breach. This error overestimates the probability of drywell failure and will tend to overestimate risk (i.e., be conservative).

The impact of this error on the results, however, is fairly minor. First, for the majority of the PDS groups, the probability that pedestal failure shortly after vessel breach causes containment failure to a previously intact containment is low (i.e, 0.05). In many of the accidents, the containment failed before vessel breach or at the time of vessel breach. This failure mode is only significant for the seismic PDS group (i.e., approximately 0.20 probability). However, for this PDS group, the containment would have failed from overpressure even if the pedestal had not failed. Thus, although the failure mode may have been different had the pedestal not failed, the containment would still have failed. Furthermore, the seismic PDS group has a very low core damage frequency and is not a significant contributor to risk at LaSalle.

Quantification of Loads at Vessel Breach

The loads at vessel breach are based on distributions used in the Grand Gulf analysis in NUREG-1150.[4] The distributions used in the Grand Gulf analysis were provided by the Containment Load Expert Panel.[5] The Grand Gulf distributions were not used directly in the LaSalle analysis for several reasons.

First, the LaSalle containment is inerted, whereas the Grand Gulf containment contains air. Thus, hydrogen burns are not an issue at LaSalle. Second, the arrangement of the drywell and wetwell is different at the two plants. At Grand Gulf, the drywell is located inside the containment. The drywell communicates to the containment through passive horizontal vents which direct gases in the drywell into the suppression pool. CONTAIN calculations* performed using the Grand Gulf geometry indicated that during a direct containment heating event the drywell rapidly pressurizes before the vents clear. Once the vents clear, however, the drywell begins to depressurize (i.e., the pressure was relieved into the suppression pool). In these calculations, the peak drywell pressure occurs just prior to clearing the vents. At LaSalle, the drywell is located above the wetwell and is separated from the wetwell by the drywell floor. The drywell communicates with the wetwell through vertical vents called downcomers. The design pressure of the drywell floor is 25 psig whereas the design pressure of the containment is 45 psig. Thus, the drywell floor is not as strong as the drywell wall. The rapid pressurization of the drywell (i.e., before the vents clear) will result in a large pressure differential across the drywell floor. If this pressurization is sufficient to fail the drywell, it is likely that drywell floor will fail before the containment fails. Failure of the floor will equalize the pressure between the drywell and the wetwell and, therefore, steam and noncondensibles released at vessel breach will not enter the suppression pool. Thus, the inability to vent into the suppression pool at LaSalle could potentially result in peak pressures that are higher than what was predicted in the Grand Gulf analysis. Also, the fact that the containment at Grand Gulf to which the drywell vents through the suppression pool is much larger than the suppression pool volume at LaSalle will result in larger pressure increases at LaSalle. To account for this possibility, the upper tails of the Grand Gulf distributions were increased. The upper ends of the modified distributions were based on bounding type calculations. However, the quantification of this issue is less than ideal. Ideally, distributions should be developed that are based on the LaSalle specific calculations; however, resource limitations prevented the performance of such calculations for the LaSalle analysis.

Quantification of Drain Pipe Failure

In the LaSalle containment, there is a drain line that goes from the pedestal cavity through the wetwell and out of the containment into the reactor building. The isolation valves are located outside the containment. At vessel breach, there is the potential for an ex-vessel steam explosion to occur in either the cavity or the drain line itself (4-inch pipe). The loads from the steam explosion could potentially fail the isolation valve and effectively fail the containment integrity.

This issue came to light during the MCCI expert meeting during the NUREG-1150 project. Although the experts assessed the likelihood of ex-vessel steam explosions, they did not quantify the probability of pipe failure outside the containment given a steam explosion since they were not experts in pipe failure. Some of the experts did indicate that this failure mode

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* D. A. Powers, et.al., "Evaluation of Severe Accident Risks: Quantification of Major Input Parameters, Supporting Calculations," NUREG/CR-4551, SAND86-1309, Vol.2, REV. 1, Part 5, Sandia National Laboratories, Albuquerque, NM, to be published.

would be more likely if the steam explosion occurs in the pipe rather than in the pool because of attenuations associated with explosions in the pool. There is a large amount of uncertainty associated with this issue because it is not clear where in the pipe the steam explosion will occur or how much material will be involved in the explosions. Again, because of resource limitations, more detailed investigation of this issue was not performed. Thus, a maximum entropy distribution between 0 and 1.0 was used to quantify the case in which a steam explosion occurs in the drain pipe (i.e., mean probability of failure is 0.50). When the steam explosion occurs in the cavity pool, a maximum entropy distribution between 0 and 0.1 was used to account for the reduced probability of pipe/valve failure. This failure mode is treated as a leak in the wetwell above the suppression pool surface. The mean conditional probability of this failure mode is 0.13 (this value also includes failures from pressurization events at vessel breach that result in wetwell leaks although these are a small fraction). Because of the high probability that the containment will be vented during the accident and the fact that containment failure time is not particularly important at LaSalle, this failure mode is not the dominant contributor to risk. Changes in the evacuation assumptions or the probability of venting could certainly change the importance of this failure mode in which case more attention to the quantification of this issue would be warranted.

7.1.2.2 Source Term Analysis Deficiencies

Two input errors were discovered after the first draft of the analysis was complete. The first of these errors was in the distribution of the fractional release from the vessel (FVES) for the high RPV pressure, non-station blackout case for the Lanthanum group. The 50th percentile was inadvertently input as 0.78 instead of 0.078. This will only affect the release fraction when the distribution is sampled between the 25th and 75th percentile. Since this error is only for the FVES release, for a very specific case, and for only one radionuclide class; the effect of this error on the risk was assessed to be small. This error will result in overprediction of the Lanthanum group release fraction and is, therefore, conservative.

The second input error was in the distribution of the fractional release from the core-concrete interaction (FCCI) for the low zirconium content in the debris and a dry cavity case for the Cerium group. The 95 percentile was input as 0.018 instead of 0.18. This will affect the release fraction when the distribution is sampled between the 75th and 99th percentile. This error will result in an underprediction of the release fraction of the Cerium group. However, the error is not considered substantial enough to warrant rerunning the analysis since the release fraction of the Cerium group, as calculated in the analysis, still covered the entire range of the distribution although the upper tail was slightly low.

7.1.2.3 Partitioning Deficiencies

As described in Section 4.6, the partitioning method used for the LaSalle analysis is an improvement over that used in NUREG-1150[4]. The new method forms subgroups of the source terms based: (1) on the fraction of the total early health effect weight, latent health effect weight, and frequency that they represent and (2) on the similarity of the values of parameters important to determining the consequences. This includes both emergency response and release parameters. The correctness of the method depends critically on the appropriate selection of parameters to form the subgroups. This will determine how accurately the method represents the transformation of potential consequences represented by the weights into actual consequences calculated by the MACCS code.[6]

It is clear that dEVAC, which is the first release time minus the warning time minus the delay time (i.e., T1 - TW - TDELAY), is a critical determiner of the consequences to be expected from a particular source term. This parameter determines whether or not people will be caught by the release while they are evacuating and is critical to the determination of the early fatalities. For this analysis, it was specified that the partitioning of the source terms would be done separately on the basic parameters T1, TDELAY, and TW. By examining the PARTITION code output in Appendix D.6, one sees, for example, that partition group 2 for event type 3 has a dEVAC ranging from -1.88E+03 to +2.21E+04 sec. Since the mean value of dEVAC for this group is +1.587E+04 sec, it is clear that the source terms which have negative dEVACs are not dominant in the group definition. However, these source terms are the ones which will come closer to realizing their potential and could, therefore, contribute to the final EH risk much more than their original fraction might indicate. Because the mean value is used to calculate the consequences in the MACCS code, the negative values were not be used in the calculations.

It is difficult to estimate the effect of this problem on the distribution of consequences obtained though it is clearly non-conservative. The magnitude of the resultant bias depends critically on two aspects of the analysis: (1) the relative magnitude of the frequency of the source terms with negative dEVAC to those with positive dEVAC and (2) the amount of potential health effects converted to final health effects for the two types of source terms. By examining the consequence output one sees that for LaSalle the early fatalities are very low and come mostly from the 0.5% of the population that does not evacuate. In no subgroup was the mean value of dEVAC negative (i.e., all evacuees began moving before release). If the plume from the first release caught up with the evacuating population, then significantly more early fatalities might be expected and the risk profile could change substantially.

7.1.2.4 Consequence Analysis Deficiencies

It was specified that many of the MACCS parameters used in this study should be the same as those used in NUREG-1150[7] in order to facilitate comparison. The values used in this version of the LaSalle analysis were, therefore, the same as those used in the NUREG-1150 analysis except for site-specific parameters. We believe that future studies should consider improved treatment of some of these parameters, including: (1) the dry deposition velocity, (2) the red marrow death LD50 (lethal dose to 50% of the exposed individuals), (3) Gastro-Intestinal (GI) death threshold dose, (4) the long-term food dose projection period, and (5) the fraction of the population that evacuates.

For the dry deposition velocity, the NUREG-1150 value is 0.01 m/s. This was selected because:* (1) it was used in WASH-1400 and in previous NRC studies using other consequence codes (CRAC[8] and CRAC2[9]), (2) the Chernobyl data indicates that this value is appropriate, and (3) the use of this value is conservative. The analysis team currently believes that a value of 0.003 m/s would be a more appropriate value based on the published literature and the manner in which the parameter is used in the MACCS code. First, many papers have been published since 1975 on dry deposition velocity. A lower value for the dry deposition velocity is consistent with values used in the PNL GENII[10] (0.001 m/s), INEL RSAC-4** (0.001 m/s), KFK UFOMOD[11] (0.001 m/s), and NRPB MARC[12] (0.003 m/s) codes. Second, the release mechanisms for Chernobyl (criticality explosion and graphite fire) are quite different from the mechanisms expected in a commercial light water reactor in the U.S. (where the nature of the containment failure and aerosol production processes will result in completely different phenomenology) and it is not possible to conclude that the Chernobyl result are applicable to this problem. Finally, the assumption that the use of the 0.01 m/s value would be conservative is based on results calculated by other codes. With the MACCS code and the currently defined NRC emergency response scenarios, early fatalities result principally from the inhalation pathway and, in particular, from bone marrow deaths. Increased deposition velocity, in this instance, results in less material being available for inhalation with a consequent decrease in early fatality predictions.

For the red marrow death LD50, NUREG-1150 used a 50/50 mix of "minimal" and "supportive" treatment based on NUREG/CR-4214, Rev 0.[13] Because of LaSalle's proximity to Chicago hospitals, low population density, high degree of emergency preparedness by the state of Illinois, and source term timing characteristics; the analysis team believes that 100% supportive treatment would be most appropriate for LaSalle. NUREG/CR-4214, Rev 1,[14] gives an LD50 of 4.5 sieverts for this level of treatment while Rev 0 gives 4.0 sieverts for the 50/50 mixed treatment level. The NUREG-1150 values

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* Letter from M. C. Cunningham, NRC, to F. T. Harper, SNL, dated June 25, 1990. Subject: MACCS Input Parameter Values for the LaSalle Consequence Analysis.

** Wenzel, D. R., "Interim User's Manual for RSAC-4, Radiological Safety Analysis Computer Program," Westinghouse Idaho Nuclear Company, Idaho Falls, ID, draft of 10 April 1990.

and assumptions were used in this analysis. This results in somewhat higher early fatality predictions over what would result from using our current recommendation.

For the GI death threshold, the analysis team believes that the NUREG-1150 value of 7.5 sieverts should be changed to 8.0 sieverts. This parameter is insignificant compared to other sources of death and would not impact any results.

NUREG-1150 uses an infinite period for the long-term food dose projection period.* Examination of the footnoted reference indicates that the FDA was addressing an exposure period on the order of 30 days as given on page 47082 of the Federal Register.[15] The FDA exposure period appears to us to be more consistent with a one year time frame than the infinite time frame chosen in NUREG-1150. We currently believe that a one year period based on U.S. and International health physics practices would be more appropriate. This would lead to less stringent actions that would result in higher cancer fatalities and lower economic costs.

In addition, there is a significant uncertainty concerning the fraction of the population that does not participate in the evacuation. In the first draft of NUREG-1150,[10] the non-evacuating fraction was 5% and was based on SNL's general understanding of evacuations for various other accidents. This was changed in the final NUREG-1150 to 0.5%. No systematic analysis of the possible values this parameter might take was done for the LaSalle study.

The concerns and recommendations noted above highlight the fact that many uncertainties exist in the data used for the consequence analysis and points out that this portion of the analysis needs to be included in future uncertainty studies.

Although consistent with the assumptions in NUREG-1150, there is an inconsistency between the evacuation assumptions used for the non-seismic and seismic events. For the non-seismic events, it is assumed that 99.5% of the population within the emergency planning zone (EPZ) is evacuated. The remaining 0.5% are subject to hot-spot relocation after 12 hours of exposure. For the low-g seismic case, a degraded evacuation scenario is used where the evacuation speed is halved and the evacuation delay time is doubled. For this scenario, however, it was assumed that 100% of the population participates in the evacuation. For the non-seismic events, all of the early fatalities were either associated with the 0.5% of the population that did not evacuate or with the population beyond the EPZ.

For the low-g seismic scenarios, the early fatalities are only associated with the population beyond the EPZ. Even though these scenarios have a degraded evacuation, the dose to the population that was evacuated was

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* Letter from Chris Ryder, NRC, to A. C. Payne Jr., SNL, 8/16/90.

insufficient to result in an early fatalities and because none of the population remain behind (as was the case in the non-seismic scenarios) there were no early fatalities within 10 miles of the plant. This difference in the evacuation assumptions must be recognized when comparing the non-seismic results with the low-g seismic results.

7.2 Methodology Insights and Conclusions

As mentioned in Chapter 1, four out of five of the primary objectives of the LaSalle analysis 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 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 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.

All of these objectives are primarily concerned with methodology development. The fifth objective was to apply this methodology to a BWR, MARK II nuclear power plant. Because of these objective, much of the presentation in both the Level I and Level II/III reports has concerned details of the implementation, perceived strengths and weaknesses, and possible improvements in the methodology. In this section we would like to summarize some of the improvements made to the Level II/III methodology since their initial development in this program and their first application in the NUREG-1150 analysis (Section 7.2.1), discuss some of the weaknesses of the current methodology (Section 7.2.2), and possible ways of overcoming them (Section 7.2.3).

7.2.1 Methodology Improvements Over NUREG-1150

7.2.1.1 Plant Damage State Definition Improvements

A systematic method, as described in Chapter 2 of this report, was developed to define the interface between the Level I analysis in terms of accident sequences and the Level II/III analysis in terms of PDSs. A preliminary version of this method was also used for the Peach Bottom plant analysis in NUREG-1150.[3]

1

IMAGE EVALUATION TEST TARGET (MT-3)

1.0 1.1 1.25 1.4 1.6 1.8 2.0 2.2 2.5 2.8

150mm

6"

1

IMAGE EVALUATION
TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION
TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION
TEST TARGET (MT-3)

1.0 1.1 1.25 1.4 1.6 1.8 2.0 2.2 2.5 2.8

150mm

6"

1

IMAGE EVALUATION
TEST TARGET (MT-3)

150mm

6"

7.2.1.2 Event Tree Construction Improvements

As described in Chapter 2 of this report, the EVNTRE[17] code was significantly improved for the LaSalle analysis over the original version developed in the SARRP program. This version of the code was used in the NUREG-1150 analysis and only a slight additional improvement was made for the LaSalle analysis. The code used in the NUREG-1150 analysis could only put split fractions directly into a question with two branches in the sampling mode. To put split fractions into a multiple branch question, a complicated procedure for calculating complements of split fractions had to be used. This original method was difficult to implement and verify because complements of split fractions appeared in the input files rather than the split fractions themselves. The code was modified so that split fractions could be input directly into any number of branches. This may seem like a small change but it allows for a simple method of performing an integrated analysis. This project, as opposed to NUREG-1150, was charged with performing an integrated analysis where the results of the internal, seismic, fire, and flood analyses were all evaluated in an integrated and consistent fashion. In order to do this, a method was devised for including in the APET a question which asks which plant damage state was being evaluated. This question had 30 branches, one for each of the 30 PDSs being analyzed for LaSalle. These PDSs came from all of the constituent analyses (fire, flood, seismic, and internal). The TEMAC code[18] calculated split fractions based on the ratio of the PDS frequency to the total core damage frequency and these were put directly into the question branches. The subsequent questions which specified the PDS characteristics had case structures that directly referenced the appropriate PDSs.

The result was that (1) this allowed testing of the APET or evaluation of one PDS by simply putting in 1.0 for the appropriate branch in the first question instead of changing all of the initial questions and having to keep track of multiple files as in NUREG-1150, (2) all of the PDSs could be run through the APET at once and the accident progression bins could be truncated at a consistent value, and (3) the integrated output could then be passed on to subsequent stages of the analysis.

A simplified version of this method was applied in the N Reactor PRA.[19] Because multiple split fractions could not be directly input into the tree at the time it was done, complicated logic was required to calculate the correct splits as described above. The new method is much more straightforward and understandable.

7.2.1.3 Source Term Analysis Improvements

For LaSalle analysis, a new code, RELTRAC,* was developed to perform the source term analysis. This code uses rate parameters instead of integral parameters as do the XSOR codes.** The use of a tunable rate-based model allows more accurate simulation of the results from more detailed thermal/hydraulic codes than is possible with an integral-based code. This allows more flexibility for the analyst in the interpolation and extrapolation of the detailed code results to APBs for which code results were not available.

Because of the timing of the NUREG-1150 analysis, the first application of this code was for the original Peach Bottom analysis in first draft NUREG-1150.[20] RELTRAC was not used in the final LaSalle analysis for several reasons. First, the interface to determine how the adjustable parameters would be modified to model each APB is currently hardwired and would require a significant effort to convert the model from Peach Bottom to some other plant. Second, the code in its current form is not fast enough to estimate source terms for the hundred thousand or so accident progression bins now being generated by the APET. Third, because of delays in getting the MELCOR[21] code to perform the integrated calculations, an interfacing code to directly extract MELCOR data and use it to tune the RELTRAC model to the detailed MELCOR calculations was not completed. Finally, the experts in the NUREG-1150 expert elicitation process gave integral results for the source term parameters, not rates, and to convert the integral form to a rate form would require a large effort. To make all of these upgrades would have required the expenditure of resources beyond what was available to complete this study.

RELTRAC is still very fast when compared to detailed thermal/hydraulic codes and can evaluate tens of thousands of accident progressions and source terms with the same resources used to make one detailed calculation. It would be good for more detailed simulation of the results of thermal/hydraulic codes such as MELCOR or STCP[22] and could be used to examine the effects of various parameter uncertainties on the accident progression and source term.

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* J. C. Helton, J. D. Johnson, and J. M. Griesmeyer," User's Guide For the RELTRAC Reactor Accident Source Term Model," NUREG/CR-5445, SAND89-2174, Sandia National Laboratories, Albuquerque, NM, Draft.

J. C. Helton, J. M. Griesmeyer, F. E. Haskin, and J. D. Johnson," Incorporation of Uncertainties Into Reactor Accident Source Term Estimates," NUREG/CR-5444, SAND89-2173, Sandia National Laboratories, Albuquerque, NM, Draft.

** H.-N. Jow, W. B. Murfin, and J. D. Johnson, "XSOR Codes User's Manual," NUREG/CR-5360, SAND89-0943, Sandia National Laboratories, Albuquerque, NM, to be published.

As a result, it was decided to create a version of the XSOR codes for the final LaSalle analysis (i.e., the LASSOR code); however, in NUREG-1150 certain limitations of the XSOR codes became apparent in the BWR analyses of Grand Gulf and Peach Bottom. First, two release segments were not enough to capture the release variability over time or interactions of the release and the evacuating population appropriately. If the containment fails before vessel breach there are really three releases: (1) the relatively small in-vessel release, (2) the puff release at vessel breach, and (3) the late release associated with CCI. Grouping the in-vessel release with the release at vessel breach may miss the evacuating population. On the other hand, releasing these source terms before vessel breach overstates the in-vessel portion and spreads out the vessel breach release. In addition, the in-vessel and vessel breach releases may have two distinctly different release paths and containment failure modes. The result was that different parts of the release had substantially different release paths and other characteristics but still had to be grouped together. For the LaSalle analysis, three release segments were defined as described in Chapter 4 of this report, and MACCS options were selected to run three release segments. The LASSOR output and PARTITION input and output were modified to pass this information to MACCS. This worked much better than the process used in NUREG-1150 in describing the unique characteristics of the individual parts of the release.

Second, in anticipation of the incorporation of consequence uncertainties, the LASSOR code was modified to include an event type parameter which could be used to specify groups of accident progression bins that had unique evacuation responses due to the characteristics of the initiating events such as high-g seismic, low-g seismic, and station blackout accidents where certain weather patterns may occur more or less often than obtained by random association. For this analysis, only the seismic sequences were singled out using the event type parameter because the interaction between station blackout accidents and weather type is not well defined at the current state-of-the-art. In addition, the code was modified to sample the evacuation parameters: warning time and evacuation delay time. Distributions were developed for the warning time based on the emergency plan and the specific characteristics of the accident progression bins.

Third, more detailed models of the release paths were developed, as described in Chapter 4, to try and capture some of the differences thought to be important as a result of the different possible containment decontamination mechanisms and the different paths resulting from multiple containment failures as described above.

7.2.1.4 Partitioning Improvements

A new method for grouping source terms for evaluation by the MACCS code was developed for the LaSalle analysis. The PARTITION code implements this method. An initial version of the PARTITION code[2] was used in the final NUREG-1150 analysis. As a result of its use in NUREG-1150, certain limitations became apparent. First, many of the partition groups were empty (20 out of about 60 used). Second, of the ones that were not empty, many did not contribute significantly to the final risk and, third, the process did not allow for the incorporation of consequence uncertainties.

The PARTITION code was therefore rewritten for the final LaSalle analysis. As described in Chapter 4 of this report, the code was changed to (1) allow accident progression bins coming from initiators with unique evacuation characteristics to be partitioned together, (2) allow partition groups to be defined not only on the basis of early and latent health effect weights but also frequency, (3) allow the definition of groups on the basis of similar parameter values for parameters important in the consequence analysis, and (4) allow the passing of these parameters to the MACCS code for each group defined. These changes were made so that the conversion of the potential consequences represented by the weights to the actual consequences calculated by MACCS would be more accurate.

7.2.1.5 Consequence Analysis Improvements

In order to perform the integrated analysis, the pre-processor to the MACCS code, STER, was changed to allow the creation and evaluation of partition groups with different evacuation assumptions in one calculation. In the final NUREG-1150 analysis, MACCS calculated a large number of consequence measures, not all of which were needed for the analysis. For LaSalle, computational cost was reduced by reducing the number of consequence measures calculated (this resulted in a factor of two decrease in calculation time). This was important because the new method of partitioning resulted in more partitions because of the increased resolution, the incorporation of all the constituent analyses, and the fact that no empty partitions were defined.

The evacuation assumptions for the non-seismic cases are the same as those used in NUREG-1150 but the evacuation assumptions for the seismic cases were changed. In the NUREG-1150 low-g seismic cases, the shielding factors for normal activity were modified to reflect broken windows as a result of the seismic event; however, this modification took place throughout the entire spatial grid (i.e., 1000 mi. radius). This assumption is clearly very unrealistic. For the LaSalle analysis, three different emergency response assumptions (cohorts) were utilized in the seismic analysis. The seismic source terms were divided into high-g and low-g cases as for NUREG-1150, but the fraction of people falling into the different cohorts was different for the different g levels. Since the fractional weighting is not done in the MACCS code but in the post-processor, this allows the analyst to easily investigate alternate assumptions. The three cases were degraded evacuation, 12 hour relocation, and 24 hour relocation. A more detailed description is presented in Chapter 5 of this report.

7.2.1.6 Risk Analysis Improvements

The PRAMIS[23] and PRPOST (PRAMIS modified to accept the full consequence distributions; not just the means as in PRAMIS) codes that perform the risk integration were developed for the LaSalle analysis. These codes were also used in the NUREG-1150 analysis and no substantial improvements were made in them for the final LaSalle analysis. The STRIP code (which extracts the mean values of the consequences from MACCS output), the SAVE code (saves subset of MACCS results used by PRPOST), the PRAMIS code, and the PRPOST code were all modified to perform their calculations using the new cohort structure with the event type parameter and the use of only one total core damage frequency from the Level I analysis instead of a separate core damage frequency for each plant damage state.

7.2.2 Methodology Limitations

7.2.2.1 Quantification Limitations

The use of Zero/One sampling is still controversial (see Section 3.2.2 for a discussion of Zero/One sampling). Some of the people involved in the NUREG-1150 expert judgement elicitation process felt that, if they could sufficiently specify the accident sequence characteristics, then they could specify precisely whether or not a phenomena would or would not occur. This led them to believe that the phenomena either always would occur or never would occur. This resulted in their selecting Zero/One sampling as the correct way to treat these phenomena in the analysis. However, since each accident represents a class of accidents and many characteristics of the accidents which could affect the occurrence of various phenomena will never be modeled in a PRA, it is our opinion that the accidents can never be specified that precisely.

While the use of Zero/One sampling does significantly speed up the accident progression analysis by eliminating the need to evaluate many of the possible paths in the APET for a particular sample member, it has a negative impact on the regression analysis. With Zero/One sampling discontinuous processes are explicitly introduced into a linear regression model. A linear regression model is most effective with continuous processes whose effects appear in every observation. In contrast, it is more difficult to represent discontinuous processes whose effects appear only in certain sample members and not in others. This can significantly affect the accuracy of the regression.

It seems to us that a more appropriate way to treat these parameters, that is consistent with the way the Level I analysis is performed and with the general philosophy of PRA, is to use split fractions and to put uncertainty distributions on the various branches. In this study, Zero/One sampling was used in spite of these objections for consistency with the NUREG-1150 analysis.

7.2.2.2 Consequence Analysis Limitations

The main methodological limitation remaining in this study is the failure to include consequence uncertainties in the integrated uncertainty analysis. One of the primary purposes of the LaSalle PRA was to develop methods for performing integrated uncertainty analyses and for propagating these uncertainties throughout the analysis. Several plans were developed for performing uncertainty analysis on the consequences and preliminary changes were made in the LASSOR and PARTITION codes to allow the LHS sampling of the emergency response parameters as described above. However, due to resource and scheduling limitations the LaSalle analysis was completed without considering consequence uncertainties.

7.2.3 Possible Methodology Improvements

7.2.3.1 Plant Damage State Definition Improvements

There are several possible improvements that could be made in the interface between the Level I and Level II/III analyses.

1. Some of the Level II/III questions about system operability could be evaluated directly from the Level I accident sequence event trees. For example, in the Level I analysis, after a system has successfully prevented core damage, the status of other systems is not further investigated. Another example is that, if core damage has occurred, the status of systems on the event tree that are unable to prevent core damage is not further investigated. It is possible for those systems for which fault trees are available, that the Level I analysts could separate the cut sets within a sequence on the bases of the status of these system using the Level I computer codes. This would remove some of the human error potential in classification of the cut sets for purposes of defining the plant damage states. Additional questions could be added to the Level I event trees to allow more explicit evaluation of the core damage sequences retained for the Level II/III analysis.

2. It should be possible to develop a code that would allow the analyst to specify various characteristics or events occurring in the Level I accident sequence cut sets that would indicate the status of the systems required in the Level II/III PDS definition and automatically put the cut sets into separate files. The current process requires that the analyst examine individually all of the cut sets to evaluate the answers to the PDS defining questions, and then to edit the sequence cut set files manually moving the cut sets for each PDS into new files. This is a very time consuming and error prone process. A least one code exists that was developed to apply recovery events to Level I cut sets that might be adapted to this purpose.

3. It should be possible to use the EVNTRE code to construct a Level I accident sequence event tree in addition to the Level II accident progression event tree. After the system fault trees are solved using some fault tree code, the cut sets could be inserted into a user function and the sequence frequencies could be calculated directly. One problem with this is how to make sure that commonalities and success states are handled correctly.

7.2.3.2 Accident Progression Analysis Improvements

There are several methodology improvements that would be useful in the accident progression analysis:

1. A restart capability should be put into the EVNTRE code. With the long run times required both for individual PDSs and for the integrated calculations, this could save a lot of resources.

2. Currently there is no way to check that the parameters are being treated correctly in the user function. The EVNTRE code could be modified to allow the mean parameter value to be included in the output file.

3. Some method of allowing the user to specify a reduced number of questions and of plotting these questions as an event tree with multiple branches and summary split fractions on the branches could be devised. This would aid the development of simplified event trees for investigation and presentation of results for specific phenomenology of interest.

7.2.3.3 Source Term Analysis Improvements

There are several methodology improvements that would be useful in the source term analysis:

1. There are several interfacing codes between the APET and the XSOR and PARTITION codes that could be combined into one code. The PSTEVNT code[24] is a post-processor of the EVNTRE code output file, and the rebinning occurs here. This code produces a new master bin list file. The MASTERK code (undocumented) produces a listing of bins with frequencies in two formats: (1) by observation for input into the XSOR code and (2) aggregated over all observations for input into the PARTITION code. The XXXFRQ code (undocumented) produces conditional probabilities and frequencies for all PDSs and is input into the XSOR code.

2. A more radical approach would be to put the XSOR code directly into the APET as a user function after constructing questions to perform the binning process directly in the APET.

7.2.3.4 Partitioning Improvements

There are several methodology improvements that would be useful in the partitioning process:

1. Due to resource limitations, the codes are still not easy to run and require several steps. The COMBIN code (developed for this

analysis) should be made part of the PARTITION code with options to specify which combinations of event types are to be partitioned separately and the STER code (undocumented) should also be made part of PARTITION with options to define the evacuation assumptions for each event type so that the MACCS input files can be created directly from the partition run.

2. The parameters to be used in the partition process need more evaluation before a final group can be selected. The purpose of these parameters is to make the source term groups defined in the partition process homogeneous enough so that the consequences calculated by MACCS reasonably approximate the actual results that would be expected if a MACCS calculation was performed just for that source term. Since the partitioning is done on potential effects, it is important that any parameters that will impact the conversion of this potential into actuality in the MACCS calculation should be as homogenous as possible.

3. As can be seen in the full PARTITION output shown in Appendix D, in some cases the number of parameters used in the definition of a group can be very small. In this case, the process will not be any worse than what was used before but will not be any better either. It seems clear that a better process would require that a minimum number of parameters be used in the initial definition of the groups and then a relaxed definition only on the remaining source terms. The actual implementation of this strategy remains to be determined.

   By examining the partition output in Appendix D, one can determine the homogeneity of the groups both in terms of the parameter variability and in terms of the accident progression bin attributes of the source terms that make up the group definition. In many cases, even though only a few parameters were used to define the group, the parameters actually have a small variation for that group. This comes about because, even though the source terms could not be grouped into one of the parameter subcells, they most likely fell into adjacent cells.

4. Another approach would be to get rid of partitioning altogether. This could be accomplished by creating a MACCSSOR code which would do for MACCS what the XSOR codes do for the thermal-hydraulic codes. In this case, consequences could be calculated for each source term just as source terms are calculated for each accident progression bin. This could even be integrated with the APET as a user function calculation along with the XSOR code as mentioned above.

7.2.3.5 Consequence Analysis Improvements

There are several methodology improvements that would be useful in the consequence analysis:

1. It is clear that a major deficiency of the LaSalle PRA is the failure to incorporate consequence uncertainties into the integrated evaluation process. Several methods have been developed for doing this but they need to be evaluated and a method selected. There are two major areas to be investigated: parameter uncertainty distributions and the sampling technique to be used to incorporate the uncertainty directly into the overall analysis structure.

2. Certain initiating events imply some general conditions on the evacuation assumptions and on the types of weather that might be likely to be occurring. For example, seismic events where the seism can affect the roads and bridges in the surrounding area and LOSP events which occur as a result of severe weather. While the new partitioning process allows for the incorporation of the impact of these events on evacuation assumptions, no consideration has yet been given to modifying the class of weather scenarios used by MACCS depending on initiator type.

3. Weather uncertainties are currently being treated differently from all other uncertainties and the current method of incorporating weather uncertainty needs to be examined to see if a method more consistent with its use in a PRA can be developed.

7.2.3.6 Risk Analysis Improvements

1. The current method of performing regression analysis with a simple linear regression model needs to be improved. The analysis is highly non-linear with many cases of the accident propagating down certain paths in which many of the parameters used in the model are no longer relevant. The method of creating the model should be improved to allow the incorporation of splits in the APET that may or may not be sampled but which determine the general characteristics of the accident progression to be included in the regression model. This would allow the creation of linear submodels for those subsets of the accident progressions with unique characteristics. The result would be a semi-linear model that would be able to more accurately represent the interactions occurring in the accident progression, source term, and consequence analysis. The basic idea is to evaluate the contribution to uncertainty of those discrete variables which are not included in the LHS sample but whose variation is represented as branching of the APET.

7.2.3.7 Latin Hypercube Sample Construction Improvements

1. From the description in the Level I[1] report and in Appendix G of this report, it should be clear that the creation of the LHS

sample is very complicated involving many steps and codes. This whole process needs to be streamlined and made more user friendly in order to be understandable and more easily reviewed.

7.2.3.8 General Improvements

1. One major deficiency in the current process is the fact that the codes are fairly user unfriendly. An example of this is the production of various tables and plots of the results which are currently done by running many individual programs to produce the needed information. Consideration should be given to incorporating many of these programs directly into the main codes so that the appropriate output can be generated directly during the runs.

2. If the XXSOR and MACCSSOR codes are incorporated directly into the APET then it would be convenient to be able to truncate on absolute frequency or at any intermediate conditional probability at any step in the analysis (i.e., CD, APET, SOR, MACCSSOR, or RISK).

3. Thermal/Hydraulic analyses. In order to evaluate a large number of sequences and to explore the output of a particular sequence over the range of assessed values for the input parameters, it is important to be able to efficiently set-up and run large numbers of calculations. Currently the MELCOR code is very sensitive to changes in the input deck. Every time a new sequence or a variation on a sequence already run was run, large numbers of new errors would appear that would result in termination of the calculation or other problems. These problems are typical of large scale thermal/hydraulic codes not just MELCOR. However, this is not acceptable for use in a PRA where we wish to make many different calculations. It seems clear that, in order to make the MELCOR code more usable in PRA analysis, the code must be made more robust. Currently, the set of parameter values for which integrated calculations can be performed without code problems is very small (compared to the whole state space). Some set of LHS calculations needs to be performed to exercise the models in an integrated fashion over a wider range.

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11. ABSTRACT (200 words or less)

A Level III probabilistic risk assessment (PRA) was performed for the LaSalle Unit 2 nuclear power plant. The objective of this study was to provide an estimate of the risk to the offsite population during full power operation of the plant and to included a characterization of the uncertainties in the calculated risk values. Uncertainties were included in the accident frequency analysis, accident progression analysis, and the source term analysis. Only weather uncertainties were included in the consequence analysis. The risk estimates presented in this report include contributions from both internal and external initiators.

The offsite risk to the public due to the operation of LaSalle County Station is relatively low, especially with respect to the NRC safety goals. The mean individual early fatality risk within 1 mile is 1.1E-10/R-yr which is more than three orders of magnitude below the safety goal. Similarly, the mean individual latent cancer fatality risk is 8.5E-09/R-yr which is slightly more than two orders of magnitude below the safety goal. In fact, the entire uncertainty distributions for these two risk measures lie below the safety goals. The mean values for early fatality risk and for latent cancer fatality risk are 1.2E-08/R-yr and 0.25/R-yr, respectively.

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