EXECUTIVE SUMMARY OF THE LASALLE INDIVIDUAL PLANT EXAMINATION AND INDIVIDUAL PLANT EXAMINATION (EXTERNAL EVENTS)

OVERVIEW

The LaSalle Individual Plant Examination (IPE) and Individual Plant Examination (External Events) (IPEEE) is the result of a detailed review of the NRC's Risk Methods Integration Evaluation Program (RMIEP) (NUREG/CR-4832) analysis. This review process, conducted by Commonwealth Edison (CECo), determined that no vulnerabilities regarding severe accident issues were indicated by the results of the RMIEP analysis of LaSalle County Station Unit 2. Furthermore, it is CECo's position that the objectives of Generic Letter 88-20 have been accomplished for both internal and external events through this review process. The RMIEP results are well within the safety goals established by the Nuclear Regulatory Commission (NRC).

The following paragraphs present more detailed information on the details of the review of the RMIEP study as it relates to IPE/IPEEE.

LASALLE RMIEP IPE/IPEEE REVIEW

The review process developed and used by Commonwealth Edison for the LaSalle County Station RMIEP IPE/IPEEE project to address the main objectives of Generic Letter 88-20 included the following steps:

- Gaining an appreciation for the behavior of the LaSalle plant under severe accident conditions through:
  - Review of the physical layout/configuration of the plant and the procedures in use at the LaSalle County Nuclear Power Station.
  - Examination of IPEs submitted for other BWR/5, Mark II plants.
  - Assessment of each of the severe accident phenomena listed in NUREG-1335 for applicability to LaSalle County Station (LSCS).
  - Limited use of the Modular Accident Analysis Program (MAAP) code to obtain a best estimate characterization of accident sequence progression.
  - Review of the RMIEP representations of LaSalle severe accident behavior.

- Reviewing and analyzing the dominant sequences (top 95% of core damage frequency) and key component failures from the internal events analysis in the RMIEP study. This involved:

  - Identifying the RMIEP dominant sequences and developing a functional understanding and description of each sequence.

  - Identifying the failed frontline systems for each dominant sequence and developing a description of the propagation of the dominating initiating events and subsequent system failures.

  - Analyzing the dominant cutsets for each sequence, determining the key component failures, and understanding the "Common Cause" and "Operator Action" treatment in the RMIEP Study.

- Identifying and evaluating observations (insights) regarding the station configuration or practices which may affect the risk profile of the plant. There were several sources of these observations: the RMIEP study, plant information review in conjunction with the RMIEP review, specific analyses of similar plants (Nine Mile Point 2 and WNP2) and previous CECo IPEs.

- Reviewing the external events portion of the RMIEP study to identify key component failures and dominant sequences, examining methodologies used, comparing the analyses to other IPEEE processes.

- Identifying and documenting technical issues which CECo will address in the future update of the LaSalle probabilistic risk assessment (PRA).

As a result of performing these reviews of the RMIEP analysis, CECo has attained a new level of understanding of the plant and its behavior under a variety of postulated accident scenarios. During this review process, numerous insights applicable to LaSalle were obtained from the previous CECo BWR IPEs, IPEs of similar plants and directly from the RMIEP analysis. These insights will be provided to the station for disposition.

Although it is CECo's position that the intent of Generic Letter 88-20, as it applies to both internal and external events, has been met with this detailed review of the RMIEP analysis, several areas of technical concern with the RMIEP analysis have been identified, including:

- The RMIEP human reliability assessment results appear to be non-conservative.

- The "Beta factor" common cause analysis process is too conservative.

- Considering main feedwater in medium and large loss of coolant accidents appears to be non-conservative.

- Considering that the emergency core cooling pumps would be made unavailable due to low net positive suction head is too conservative.

- The data used in the RMIEP analysis was generic rather than plant specific.
- The plant configuration represented in the RMIEP analysis was "frozen" in 1985.
- The fire initiating event frequencies used in the RMIEP are very conservative.

Taken collectively, these technical concerns are expected to significantly impact the current set of dominant sequences. Recognizing these concerns, CECo will limit the application of the RMEIP analysis to narrow scope probabilistic evaluations. CECo is currently finalizing the project plan to perform an independent, comprehensive Level II internal events LaSalle PRA, which will provide an analysis of LaSalle that can be considered to be an update to the applicable portions of the RMIEP analyses. This "updated" LaSalle internal events PRA will be comparable to the other CECO IPEs. Upon its completion, CECo intends to periodically evaluate changes in equipment reliability, plant design and operation as part of a periodic review and update of this latter LaSalle PRA.

The RMIEP analysis only examined Unit 2 at the LaSalle County Station. The units at LaSalle share the services of a swing diesel generator. Therefore, at least one dual unit initiator exists, Dual Unit Loss of Offsite Power, which would be considered in a complete "station PRA". In updating the PRA on LaSalle, CECo will perform a detailed examination for unit-to-unit system differences and other events that could potentially be simultaneous initiators in both units. However, based upon the results of unit-to-unit system comparisons on it's other BWRs, similar units generally retain that similarity. Furthermore, similar units would be expected to display similar risk profiles for initiators that do not impact the units differently through shared (swing) components.

Plant modifications have occurred since the RMIEP models were developed in 1985 and the impact of these modifications on the RMIEP plant model has not been quantified. A top level review of these modifications was performed by CECo personnel and no modifications which would have a significant, adverse impact on the LaSalle County Station risk profile were identified. The detailed impact of these modifications on the plant models will be assessed in the future update of the LaSalle PRA analysis.

During the conduct of the LaSalle RMIEP IPE/IPEEE program, 218 LaSalle-related IPE and accident management insights were identified. IPE insights deal with plant procedures, hardware, training, information, and test/maintenance. Accident management insights address issues involving Accident Management strategies, organization, training, computational tools, and information systems. The insights will be provided to the station for disposition.

CONCLUSIONS

The LaSalle County Station RMIEP project resulted in a very comprehensive PRA. It has provided CECo with a new level of understanding of the plant and its behavior under a variety of potential accident scenarios.

The LaSalle mean core damage frequency due to internal events was determined by the RMIEP analysis to be 4.4E-05 per year. Of the total core damage frequency, over 64% is due to one sequence in which a transient is followed by failure of all high and low pressure injection. The contribution to core damage due to external events is composed of contributions for fire, seismic and for internal flooding. The mean core damage frequency for fire events is 3.2E-05 per year. The mean core damage frequency for internally initiated flood events is 3.4E-06 per year. The mean core damage frequency for seismic events is 7.6E-07 per year. These results are within the NRC's safety goals.

Commonwealth Edison's RMIEP analysis team has performed the review of the RMIEP study for applicability as CECo's response to the objectives of Generic Letter 88-20 for LaSalle County Station. As a result, CECo personnel have developed a unique understanding of the behavior of the LaSalle plant under accident conditions and of the total plant capabilities to respond to accidents.

Because of the plant configuration cutoff date of 1985 and because of methodology differences, CECo feels that reanalysis would show that the dominant contributors to core damage would change and that sequences that are currently important would be significantly less important or eliminated. Specifically, the contribution from the dominant RMIEP sequences (representing 95% of the internal events core damage frequency) would be significantly reduced by the RCIC "sneak circuit" modification, a more realistic model of the common cause failures of diesel generator cooling, and credit for the ECCS pumps operability under low NPSH conditions.

Although CECo identified several technical concerns with the RMIEP analysis process, the principal purpose of the LaSalle County Station RMIEP IPE/IPEEE was to develop an understanding of the severe accident behavior of LaSalle and of the severe accidents postulated by the analysis. It accomplished this purpose. CECo has gained a better understanding of the probability of core damage at the LaSalle County Station as a result of this review. The numerous insights developed during the LaSalle RMIEP IPE/IPEEE process will be provided to the station for disposition. Those insights dealing with accident management will form the basis for future development and implementation of the LaSalle County Station Accident Management program.

LASALLE COUNTY NUCLEAR POWER STATION

INDIVIDUAL PLANT EXAMINATION
and
INDIVIDUAL PLANT EXAMINATION (EXTERNAL EVENTS)
DRAFT SUBMITTAL REPORT

APRIL 1994

Submitted By

COMMONWEALTH EDISON COMPANY

INDIVIDUAL PLANT EXAMINATION
and
INDIVIDUAL PLANT EXAMINATION (EXTERNAL EVENTS)
SUBMITTAL REPORT

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SUMMARY OF THE LASALLE RMIEP IPE/IPEEE

This document provides a summary of the LaSalle County Station Individual Plant Examination (IPE) and Individual Plant Examination (External Events) (IPEEE) for which Commonwealth Edison is submitting the LaSalle Unit 2 Risk Methods Integration and Evaluation Program (RMIEP) analysis. (Refer to Appendix A for definition of the acronyms used in this document.) This document specifically presents the documentation of the RMIEP analysis (including the final published portions as well as the draft portions) as of 1993; this information represents the plant as it was configured in 1985. The LaSalle RMIEP IPE/IPEEE project concludes that the RMIEP study is an acceptable representation (for IPE purposes) of the risk associated with internal and external events to which the LaSalle County Station was susceptible. The technical information presented in this summary can be found in greater detail in the RMIEP documentation.

1.0 Project Organization

Commonwealth Edison (CECo) created an organization for the performance of this project which utilizes its internal personnel resources. The CECo personnel assigned to conduct the LaSalle RMIEP IPE/IPEEE project collectively have extensive experience in plant operations and systems engineering, as well as probabilistic risk assessment (PRA) experience. CECo personnel performed the basic modeling review and analysis, as well as the Level II review and analysis using a CECo-specific version of the Modular Accident Analysis Program (MAAP) code.

The LaSalle County RMIEP IPE/IPEEE project team consisted of Messrs. Bruce Momsen (project manager and team leader), George Klopp, Phillip Cretens, James Hawley, Richard Johnson, Milad Kalache, Leland Raney, Douglas Swartz, and Kong Wang. The work of this team was assisted and reviewed by LaSalle County Station personnel. As noted in the initial CECo response to the Nuclear Regulatory Commission (NRC) on Generic Letter 88-20, and as requested in the generic letter, no separate "independent review" of the LaSalle County Station IPE/IPEEE was performed under CECo auspices.

Biographical Sketches of RMIEP IPE/IPEEE Team

Mr. Bruce Momsen is the CECo IPE/AM program manager. Mr. Momsen has 18 years of experience at CECo. Much of this experience was in the nuclear design area where he was responsible for the first CECo-performed nuclear reload design of Zion Station as well as supervision of all internal and external reactor physics analyses for any CECo plants. He also supervised the computer methods development area of Nuclear Fuel Services where he directed tasks associated with implementation, maintenance, modification and benchmarking of analytical methods and software necessary for neutronic, thermal-hydraulic, transient and core follow analyses.

Prior to taking over as supervisor of the PRA Group in February 1993, Mr. Momsen spent a year as the Risk Management Coordinator, acting as the liaison between the nuclear

stations and the PRA Group and having responsibility for the implementation of PRA technology throughout the CECo Nuclear Division.

Mr. George T. Klopp is the CECo Senior Technical Expert in PRA and the IPE technical lead. Mr. Klopp has many years of experience at CECo in the design of CECo plants and later in the performance of PRAs on CECo plants. Mr. Klopp managed the CECo IPE program from 1987 through 1992. He was the project manager for the original Zion Probabilistic Safety Study and has been responsible for various PRA analyses of the Byron and Braidwood Stations. He has participated in the peer review of PRAs conducted on Sequoyah and other nuclear plants, as well as the peer review of the NUREG-1150 study done on Zion. He was also a participant on numerous IDCOR expert review groups. Mr. Klopp is thoroughly familiar with the design of all CECo nuclear plants, and has directed the technical aspects of the IPE/AM efforts on each plant. This provides consistency to all of the CECo evaluations. Mr. Klopp has served as a member of a team assembled to review insights for both the Byron and Quad Cities IPE projects.

Mr. Phil Cretens has 20 years of experience at CECo in all areas of nuclear plant operations including tech. staff, maintenance, radwaste, work planning, startup and operations. He has held Senior Reactor Operator (SRO) licenses on three CECo plants - Quad Cities, LaSalle County, and Braidwood, and also has U.S. Navy nuclear submarine reactor operator experience. Mr. Cretens has assisted during the CECo boiling water reactor (BWR) IPEs performing a variety of tasks including system analysis review, quantification review and assisting with project planning and scheduling.

Mr. Hawley participated in the development of the Quad Cities IPE/AM success criteria and developed the Level II portion of the IPE, including the accident progression response analysis using a CECo-specific version of the MAAP computer code. Mr. Hawley performed a similar role for the Dresden IPE project and has extensive experience with the use of the MAAP code and its development gained during his five years as a staff member of Fauske and Associates, Inc. Mr. Hawley evaluated severe accident behavior addressed in the LaSalle RMIEP study, Dresden Station IPE, Quad Cities Station IPE, and other non-CECo IPEs for applicability to LaSalle County Station.

Mr. Rich Johnson has more than 12 years of CECo experience as an instructor and engineer. He supervised in-service testing, thermal and safety system performance monitoring, replacement parts evaluations, and operational experience reviews at Dresden. Prior to joining CECo, he taught nuclear engineering and conducted and analyzed reactor physics experiments. Mr. Johnson has recently participated in PRA applications for Dresden, Quad Cities, and Zion.

Mr. Milad Kalache joined CECo with more than 13 years experience in nuclear power plant systems and operation. He has more than seven years experience in IPE/PRA performance. Mr. Kalache is involved in the success criteria development, the accident progression response analysis using MAAP computer code, and the review of various notebooks for other CECo IPE/AM Projects. For the LaSalle RMIEP study, Mr. Kalache served as the CECo liaison with Sandia National Laboratories, (SNL) and as the accident sequence analyst.

Mr. Lee Raney has 23 years of experience with CECo as well as test reactor operational experience. Much of his experience was obtained performing engineering/design and safety analysis on CECo's nuclear plants. As part of the Zion, Dresden and Quad Cities IPE/AM projects, he participated in success criteria development using the MAAP code, review of MAAP analysis results, and development of common cause failure data. Mr. Raney has served as a member of the insight review team and as Insights Coordinator on the Dresden, Quad Cities and LaSalle IPE projects.

Mr. Doug Swartz is the CECo Risk Management Coordinator recently replacing Mr. Momsen in this liaison function. In this role, he has been stressing the deployment of PRA applications at the stations, as well as facilitation of Accident Management formation across the Nuclear Operating Division. Mr. Swartz came to the PRA Group with more than 21 years of CECo experience; 17 plus years in the engineering areas associated with LaSalle County, Carroll County, Zion, Byron and Braidwood Stations, and 4 years in the licensing area associated with aspects of all of CECo's nuclear power plants. For the last 3 years, he directly supervised engineering activities involving modifications, operability assessments and configuration control associated with Byron and Braidwood Stations.

Mr. Kong Wang joined CECo with nine years experience in nuclear power plant systems and fundamental research in thermal-hydraulics related to nuclear reactor safety. In the past he has worked as a safety engineer for CECo's Byron and Braidwood stations while employed by an architectural engineering firm. Most recently, he was a research engineer in the areas of developing and analyzing experiments/models/codes that investigated most of the severe accident issues of interest for nuclear plants. Currently, he reviews the phenomenological position papers developed to support CECo's IPE/AM programs and is responsible for CECo's IPEEE program. Mr. Wang evaluated IPEEE issues addressed in the LaSalle RMIEP study, Dresden Station IPE insights, Quad Cities Station IPE insights, and other non-CECo IPE insights for applicability to LaSalle County Station.

2.0 Philosophy and Conformance with Generic Letter 88-20

Commonwealth Edison Company (CECo) presents the LaSalle County Station Level III PRA performed under the NRC's Risk Methodology Integration and Evaluation Program to be in compliance with the intent of NRC Generic Letter 88-20 and its Supplement 4.

The four main objectives of the IPE requirements stated in the NRC Generic Letter 88-20, are:

- Develop an appreciation of severe accident behavior.
- Understand the severe accident sequences that are postulated for the plant.
- Gain a more quantitative understanding of the overall probabilities of core damage and fission products releases, and,
- If necessary, to reduce the overall probabilities of core damage and fission product release by modifying, where appropriate, hardware and procedures that would help prevent or mitigate severe accidents.

This summary document describes the accomplishment of these objectives through the CECo activities centered around the review, analysis, understanding and evaluation of the RMIEP study.

3.0 Process

This section describes the overall process developed and used by Commonwealth Edison for the LaSalle County Station RMIEP IPE/IPEEE project. This process, consisting of the steps described in the following subsections, addressed the main objectives of Generic Letter 88-20.

- Gain an appreciation for the behavior of the LaSalle plant under severe accident conditions through:
  - Review of the physical layout/configuration of the plant and the procedures in use at the LaSalle County Nuclear Power Station.
  - Examination of IPEs submitted for other BWR/5, Mark II plants.
  - Assessment of each of the severe accident phenomena listed in NUREG-1335 for applicability to LaSalle County Station (LSCS).
  - Limited use of the MAAP code to obtain a best estimate characterization of accident sequence progression.
  - Review of the RMIEP representations of LaSalle severe accident behavior.
- Review and analyze the dominant sequences (top 95% of core damage frequency) and key component failures from the internal events analysis in the RMIEP study. This activity involved:
  - Identifying the RMIEP dominant sequences and developing a functional understanding and description of each sequence.
  - Identifying the failed frontline systems for each dominant sequence and developing a description of the propagation of the dominating initiating events and subsequent system failures.
  - Analyzing the dominant cutsets for each sequence, determining the key component failures, and understanding the "Common Cause" and "Operator Action" treatment in the RMIEP Study.
- Identify and evaluate observations (insights) regarding the station configuration or practices which may affect the risk profile of the plant. There were potentially several sources of these observations, such as the RMIEP study, plant information review in conjunction with the RMIEP review, specific analyses of similar plants (Nine Mile Point 2 and WNP2) and previous CECo IPEs.
- Review the seismic/fire portion of the RMIEP study to identify key component failures and dominant sequences, examining specifically the seismic and fire methodologies used, comparing the analyses to other IPEEE processes.

- Identify and document technical issues which CECo will address in the future update of the LaSalle PRA.

The following subsections describe these process steps as implemented and discuss the satisfaction of the Generic Letter 88-20 objectives .

3.1 Severe Accident Behavior

To enhance CECo's current appreciation of the severe accident behavior of the LaSalle County station, the plant's behavior presented in the RMIEP study was reviewed and analyzed in detail. This review task was divided in several main sub-tasks.

Review of Plant Description

Detailed knowledge of the physical layout/ configuration and familiarity with the procedures in use are prerequisite to understanding possible severe accident progressions at the LaSalle County Nuclear Power Station. Plant details and procedural instructions can have significant effects on core debris distribution in containment, fission product distribution and retention in containment, and time intervals available to operators to mitigate a sequence progression or fission product release potential. Some of the important plant features include: routing and operational characteristics of relief valves; pedestal configuration; compartment interconnections; downcomer arrangement and submergence; structural capabilities of the drywell shell, drywell head, hatches, penetrations, and drywell floor; and existence and routing of pedestal sump drain piping. Important procedure directions relate primarily to restoring vessel injection, flooding containment, opening wetwell and/or drywell vents, and initiating drywell and wetwell sprays. These types of information were collected and used in developing a plant specific model for use with the CECo-MAAP code.

Review of Other Plant IPEs

Many features of BWR/5, Mark II plants that can significantly affect severe accident progression may be similar between the various sites that exist in this country. For this reason, IPEs submitted for such plants were obtained and reviewed to help focus CECo efforts in collecting plant data for developing a LSCS plant description and in performing best-estimate assessments of the applicability of severe accident phenomena at LSCS.

Specific areas of interest for such review efforts include: which severe accident phenomena were judged to be important for each site; which plant configuration details were identified to be important in promoting or limiting the occurrence of severe accident phenomena; which procedural instructions were identified to be important in promoting or limiting the occurrence of severe accident phenomena; and what methods were used to quantitatively assess the severe accident phenomena.

The results of the review efforts described above were used in developing a plant model for use with the CECo-MAAP code, and in assessing severe accident phenomena at LSCS.

Investigation of Severe Accident Phenomena

Each of the severe accident phenomena listed in NUREG-1335 were assessed for applicability to LSCS from a best-estimate perspective and Phenomenological Evaluation Summaries were developed to document the assessments. Each summary described the phenomenon of interest, provided the reasoning used to assess the applicability of the phenomenon to LSCS, identified the major uncertainties in the assessment, and discussed the impact of the phenomenon and its uncertainties on accident sequence progression. This effort incorporated the results of review of other similar plants' IPEs as well as the information collected for the plant description activity.

Analysis of Accident Progression

CECo-MAAP analyses performed for the Dresden and Quad Cities IPEs in conjunction with other BWR/5, Mark II plant IPEs were used to verify the Level I system success criteria adopted by the RMIEP analysis (as the need was identified by other tasks). A limited number of CECo-MAAP analyses were performed to compare core damage occurrence and timing, wetwell venting and/or containment failure occurrence and timing, and fission product releases for the sequences that were analyzed by MELCOR for the RMIEP project. Applicable severe accident phenomena and containment performance characteristics determined in the development of the Phenomenological Evaluation Summaries, were considered in the performance of the accident progression analyses. These analyses used the plant model developed as part of the plant description activity with input from the results of the phenomenology evaluations.

Identification of Accident Progression Related Insights

After the CECo-MAAP analyses were performed and evaluated, the CECo-MAAP results were further reviewed to determine whether any additional insights could be gleaned from plant behavior during severe accident conditions. The insights collected were combined with other plant insights for evaluation.

3.2 Key Dominant Sequence and Equipment Failures

To assure the proper understanding of system and component performance during the projected severe accident sequences for the LaSalle County Station, the dominant sequences and key component failures identified by the internal events analysis in the RMIEP study were reviewed and analyzed. This task was further subdivided into the three sub-tasks described below.

a. The dominant sequences resulting from the RMIEP analysis, representing 95% of the core damage frequency, were identified and reviewed. A functional description of each sequence was developed.

b. Each dominant sequence was analyzed in the following manner:

   1) The frontline systems that failed in the sequence were identified;

   2) The propagation of the driving initiating events and subsequent system failures involved in each sequence were described; and

   3) Any insights arising from this analysis were documented.

c. The dominant cutsets for each dominant accident sequence were evaluated using the following process:

   1) The key component failures were identified; and

   2) Any insights arising from this analysis were documented.

In addition to ensuring a thorough understanding of system and component performance, the analytical processes used in the performance of common cause and human reliability analyses were also reviewed in detail. Detailed descriptions of the results of the investigations into "Common Cause" and "Operator Action" treatment in the RMIEP study were developed and are presented later in this document.

3.3 LaSalle Insights Development

Throughout the CECo IPE development process on all its plants, observations regarding the station configuration or practices which may affect the risk profile of the plant have been gathered for evaluation. These observations, called insights, can suggest changes to enhance the capability of the plant and the plant operators to respond to an initiating event to either prevent or to mitigate the consequences of a severe accident. Insights can also document "good features" which have been identified during the IPE process. For the LaSalle County Station IPE/IPEEE, insights were obtained from several sources. Those sources are described below:

a. A review of Dresden and Quad Cities insights was performed to initially determine those insights that were applicable to LSCS. These preliminary insights will be provided to the LSCS insight review team for final determination of their applicability and for evaluation of their potential impact and benefit.

b. The RMIEP analysis was reviewed to identify potential plant insights. The following sections of this documentation package were focused upon:

   <u>NUREG/CR-4832</u> Volume 1, Sections 3 and 4
   Volume 3, Part 1, Section 7
   Volume 4, and

   <u>NUREG/CR 5305</u> Volume 1, Sections 6 and 7.

c. Insights from two other plant IPE submittals were reviewed for applicability to LSCS. These plants (9 Mile Point and WNP2) are similar in design to the LaSalle County station.

d. Insights were also obtained from the analyses performed of LSCS using the CECo-specific version of the MAAP code.

In addition to the above "insight specific" tasks, insights identified by all PRA Engineers working on the LSCS IPE/IPEEE effort were accumulated and processed by the LSCS analysis team. The insights obtained through all these mechanisms were documented and provided to the Insights Coordinator for entry into the LSCS Insights database. A "Tiger Team", composed of individuals from the CECo PRA Group and LaSalle County Station, will subsequently review and evaluate the insights. The Tiger Team, including personnel knowledgeable in LaSalle County plant systems and operation, severe accident phenomena, emergency procedures, and the emergency response organization will evaluate the insights from a broad perspective and to make technical assessments of their potential benefit and impact. Insights that are processed through this evaluation process will be provided to the station for disposition.

3.4 External Events Analyses

The objectives of this task were to review the external events portion of the RMIEP study (Volumes 7, 8 and 9) to assure a thorough understanding of the dominant sequences and key component failures associated with these external events.

Seismic and Fire Methodologies

In general, review of the external event portion of the RMIEP study focused on the key results rather than on the methodology. However, full knowledge of the methodologies used in the IPEEE portion of the RMIEP study was prerequisite to gaining a detailed understanding of the results. For the seismic portion of the analysis, this involved identification and understanding the fundamental approach used and a detailed review of the seismic input (hazard curves), the site response analysis, the structural/equipment fragilities, the plant logic models (event/fault trees) used and the quantification of the results. Similarly, for the fire portion of the analysis, it was necessary to identify the fundamental approach taken in the study and then to review the inputs to the analytical process: the fire area analysis, the initial fire frequencies, the screening of fire areas, the fire propagation models, the recovery analysis, and the event/fault trees.

Comparison with Other IPEEE Work

This sub-task included reviewing other utilities' IPEEE processes, comparing differences in the fundamental methodology, and understanding significant differences.

IPEEE Documentation

A summary of IPEEE issues addressed in the RMIEP study was developed and is included as part of this LSCS IPE/IPEEE Submittal.

3.5 Identification of Technical Concerns

During the development of this submittal, CECo PRA engineers identified and documented concerns identified in a comparison of RMIEP processes with the analytical processes used in the performance of the previous CECo IPEs. In this way, CECo has documented its understanding of the RMIEP report and shown that issues raised in the development of this submittal have been identified and qualitatively evaluated. The process for documentation of these issues is described below:

a. CECo PRA Engineers documented issues of concern during their review and analysis of respective topics.

b. All issues/concerns were evaluated and the most significant items arising from this process have been included in this submittal report in Section 4.

4.0 Assessment of Key RMIEP Processes

In review of the RMIEP study, assessments of the RMIEP processes for accident sequence analysis and for source term analysis have been conducted. These assessments were focused on the objectives of Generic Letter 88-20 and were conducted to gain a better understanding of the accident sequences that could occur at the LaSalle plant and the resultant plant behavior. In the process of these assessments, a better understanding of the potential scenarios was obtained and issues of technical concern with the RMIEP analysis were identified that will be addressed in an update to the LaSalle PRA. The following sections describe the assessments of the RMIEP accident sequence and source term analyses.

4.1 Accident Sequence Analysis

An accident sequence starts with an event which disrupts normal operation of the plant and requires systems to respond to the event to protect the plant. This initiating event (IE), in conjunction with the combination of successes and failures of systems responding to the event constitute an accident sequence. The interaction between each initiating event and the corresponding mitigating systems is modeled using fault trees and event trees. The quantification process used in this analysis links the system fault tree models together in one large tree. After some "clipping" and "pruning" of the trees to reflect the system performance under the conditions created by the initiator, this large fault tree is quantified to calculate the core damage frequency (CDF).

It is recognized that the RMIEP effort was a state-of-the-art analysis when it was initiated. At that time, however, most of the severe accident thermal-hydraulic codes were yet to be developed and computer capabilities were limited. UFSAR Chapter 15 was, therefore, used as the basis for success criteria for accident sequence analysis. At that time LaSalle County Station was in its early start up and testing phases so very little actual plant data was available to aid in realistically modeling critical systems such as reactor core isolation cooling (RCIC), high pressure core spray (HPCS) and residual heat removal (RHR).

The RMIEP developed functional event trees and systemic event trees to generate analytical and statistical results. RMIEP provides the following information for each individual accident sequence and for the integrated risk where all accident sequences are combined together:

(1) Descriptive statistics for the distribution of the frequency of a sequence.

(2) Risk reduction statistics for each event contributing to an accident sequence.

(3) Risk increase statistics for each event contributing to an accident sequence.

(4) Uncertainty importance for each event contributing to an accident sequence.

(5) Quantification results for each accident sequence.

(6) Core damage frequency from the quantification of the integrated risk.

(7) A listing of the major accident sequences and their cumulative contributions to the total Core Damage Frequency.

The mean CDF from internal events was found to be 4.41E-05/yr and the top five dominant sequences were found to contribute 95.3% of that total. The major contributors to those top five sequences were determined to be RCIC failure and common cause failures of the core standby cooling system (CSCS) pumps that cool the emergency diesel generators.

Analysis of the RMIEP dominant accident sequences revealed that there are several technical issues which significantly impact the results of the top contributors to core damage. The following subsections describe those technical issues related to accident sequence analysis:

Human Reliability Analysis

The human reliability analysis (HRA) performed for the RMIEP study is markedly different from the analyses performed for the other Commonwealth Edison IPE's. The RMIEP and CECo's other IPEs used the Technique for Human Error Rate Prediction (THERP) technology. However, RMIEP used THERP to evaluate the "action" portion of each operator task, but, to assess the diagnostic portion of the task, RMIEP used data from LaSalle simulator exercises. The other CECo IPEs used THERP to evaluate both the "action" and the "diagnostic" portions of the tasks.

The RMIEP models include top-level, emergency operating procedure (EOP) mandated actions and a number of recovery actions focused on failed components. The modeled EOPs were included in the RMIEP event tree and fault tree logic structures. The operator recovery actions were added to the models after the sequence cutsets were developed and prior to quantification. Therefore, the quantification of the models takes credit for performance of these recovery actions.

Significant effort was expended during the RMIEP study in determining time available for effecting the modeled human actions, including consideration of the level of anxiety induced by a variety of different initiators (such as earthquakes of differing severity), and other relevant factors. The use of simulator data for assessing the diagnostic portion of human response was unique and forward-looking at the time RMIEP started and is still at the edge of the technology.

A comparison was performed to examine the differences between the LaSalle HRA results and the results of previous CECo IPE HRA analyses, specifically the Dresden and the Quad Cities IPE results. This comparison was performed on a small number of events that were found to have been important in the Dresden and Quad Cities analyses. This comparison revealed that the human error probabilities (HEPs) used in the LaSalle

analysis were generally lower than the comparable HEPs in the previous CECo BWR analyses and that the LaSalle HEPs displayed relatively little variation for differing human actions. Recent critical reviews of LaSalle performance reinforce that the HEPs used at LaSalle should be higher than comparable HEPs at other CECo BWRs.

It is expected that the difference in HEPs is due in large part to the methods used to incorporate the use of plant procedures (including both the nature of the emergency procedures and the culture at the plant) in performance of these human reliability analyses. The RMIEP analysis process did not specifically attempt to include all the operator actions in the LaSalle EOPs as implemented during the progression of an accident. The CECo IPE process does model the EOPs as they would be implemented and an update to the LaSalle PRA will also include this detailed EOP modeling.

Summary Technical Concerns with HRA

- **The RMIEP HRA results are expected to be non-conservative.**
- **The plant model should only include proceduralized actions.**
- **The HRA should reflect the content and employment of the plant procedures that are in effect at the LaSalle County Station.**

Common Cause Analysis

The common cause analysis (CCA) used in the RMIEP assumed that common cause events for two similar components were quantified with Beta factors. A Beta factor is a conditional probability that a component will fail if a similar redundant component has failed. This method implies that some portion of the actual or potential failures observed in individual components represent mechanisms which act on all redundant components simultaneously. The Beta factors used in both the screening analysis and final quantification were calculated from analysis of component failure data which exhibited common cause characteristics.

A generic common cause failure database was used in the development of the common cause factors for the RMIEP analysis, EPRI NP-3967, "Classification and Analysis of Reactor Operating Experience Involving Dependent Events." Today more recent information is available, for example the June 1992 EPRI report entitled "A Database of Common Cause Events for Risk and Reliability Evaluations," (EPRI TR-100382). In the future update of the LaSalle analysis, and consistent with the other CECo IPE analyses, a LaSalle-specific screening will be conducted and LaSalle specific common cause factors will be developed. The common cause factors currently used in the RMIEP study are recognized to be too conservative[1]. It is expected that re-analysis, consistent with the other CECo IPEs, would result in reduced common cause factors for some key components.

[1] NRC letter dated August 7, 1991; Subject: 'Summary of Meeting with Commonwealth Edison to Discuss the Results of the NRC Sponsored Probabilistic Risk Assessment of LaSalle Station, Unit 2"; Docket Nos. 50-373 and 50-374.

Summary Technical Concerns with Common Cause Analysis

- **The CCA is not specific for LaSalle**
- **Beta factor CCA is too conservative; the analysis should be more realistic.**

Main Feedwater (MFW) Modeling

The MFW system is considered a viable injection source in the analysis of loss of coolant accidents in the RMIEP analysis. In other PRAs, including the other CECo IPEs, it has been "conservatively" assumed that MFW is not available for medium loss of coolant accidents (MLOCAs) or large LOCAs (LLOCA). The reason for this assumption is that the pipe break is assumed to be "worst case" and to be located in the recirculation piping. This piping is connected to the downcomer region of the reactor vessel which is the same region supplied by the MFW system. The water supplied by the MFW system would therefore flow to the break rather than to the core. Additionally, MFW availability depends on the water available in the hotwell and the capability to makeup water to the hotwell from the condensate storage tanks. This makeup capacity is not expected to be able to compensate for losses through the break, especially for large LOCAs. Therefore, it is expected that the pumps would run to destruction, given the work and stress load on the operators in this scenario.

In the future update to this analysis, the operation of the main feedwater system during loss of coolant accident conditions would be evaluated and removed from the event trees, if appropriate.

Summary Technical Concerns with Main Feedwater Modeling

- **Considering MFW as a viable injection source in MLOCA and LLOCA is expected to be non-conservative.**

Low Pressure ECCS Pump Modeling

The RMIEP analysis assumes that high suppression pool temperatures can result in failure of pumps taking suction from the saturated pool. This analysis also assumes that the emergency core cooling system (ECCS) pumps fail when the containment fails, presumably also due to loss of net positive suction head (NPSH). However, consistent with NRC design requirements, the low pressure pumps at LaSalle can operate at zero NPSH. In the future update of the LaSalle analysis, credit would be taken for the low pressure ECCS pumps under the conditions of low NPSH and containment failure.

Summary Technical Concerns with ECCS Pump Modeling

- **Assuming that the ECCS pumps would be unavailable due to low NPSH is expected to be too conservative.**

- **Assuming that the ECCS pumps would be unavailable due to containment failure is expected to be too conservative.**

Containment Venting Modeling

The RMIEP analysis assumes that the rubber boot connecting the standby gas treatment (SBGT) system to the containment vents will fail if the 24" vent valves are opened. The study further assumes that the environment resulting from this boot failure will be so severe that accident mitigating equipment in the reactor building will fail. Due to CECo-MAAP analyses performed during the IPE analyses for Dresden and for Quad Cities, it is expected that the two-inch vents (which are opened before the large vents) will suffice to maintain pressure in the containment below the 64-psig venting pressure. The location of this potential rupture is 47 feet (six floors) above the location of the ECCS pumps of concern. It is therefore expected that the pumps would not be as likely to fail as a result of the environment as is indicated in the RMIEP analysis.

Summary Technical Concerns with Containment Venting Modeling

- **Assuming that the boot failure will occur whenever venting is implemented and further that the accident mitigative systems would most probably be unavailable due to the resultant environment is expected to be too conservative.**

RMIEP Data

LaSalle was a very new plant at the time the RMIEP was assembling data for it's analysis and little operating experience was available. Generic data, therefore, was used extensively in this analysis for both component unavailabilities and for initiating event frequencies. Generic data may not properly represent the operational characteristics of a specific plant. In the future update of this analysis CECo will incorporate plant specific data in a manner consistent with the other CECo IPEs.

Summary Technical Concerns with RMIEP Data

- **The data used in the RMIEP analysis was generic; plant specific data should be used whenever available to properly characterize plant behavior.**

RMIEP Plant Model

Plant modifications have occurred since the RMIEP models were developed in 1985 and the impact of these modifications on the RMIEP plant model has not been quantified. A top level review of these modifications was performed by CECo personnel and no modifications which would have a significant, adverse impact on the LaSalle risk profile were identified. The detailed impact of these modifications on the plant models will be assessed in the future update of the LaSalle PRA analysis.

Additionally, the RMIEP analysis only examined Unit 2 at the LaSalle County Station. The units at LaSalle share the services of a swing diesel generator. Therefore, at least one dual unit initiator exists, Dual Unit Loss of Offsite Power, which would be considered in a complete "station PRA". In updating the PRA on LaSalle, CECo will perform a detailed

examination for unit-to-unit system differences and other events that could potentially be simultaneous initiators in both units. However, based upon the results of unit-to-unit system comparisons on it's other BWRs, similar units generally retain that similarity. Furthermore, similar units would be expected to display similar risk profiles for initiators that do not impact the units differently through shared (swing) components.

Summary Technical Concerns with the RMIEP Plant Model

- **The plant configuration represented in the RMIEP analysis has changed due to plant modifications.**

- **The plant evaluation considered only Unit 2; the RMIEP charter did not include an evaluation or a comparison of similar systems in Units 1 and 2. Dual unit initiating events were outside the scope of RMIEP.**

4.2 Level II Analysis

The Level II portion of the PRA performed as part of the RMIEP study, and incorporated, therefore, into the LaSalle IPE/IPEEE, uses a fundamentally different approach than Level II analyses performed and submitted in other CECo IPE studies. The Level II calculations link the Level I CDF results to the Level III health risk analyses performed as part of the RMIEP study, by determining accident progressions following the onset of core damage and estimating the amounts of radioactive materials released from the primary and secondary containments. This section contains a brief description of the analysis methodology used in the Level II portion of the RMIEP study, the key results obtained by these Level II analyses, and an assessment of these analyses.

Methodology Overview

Under the sponsorship of the Phenomenology and Risk Uncertainty Evaluation Program (PRUEP), Sandia National Laboratories was involved in the development and application of methods for conducting fully-integrated Level II/III PRAs. As a result of this program focus, the following five steps were developed to constitute a Level II analysis and they were performed as part of the RMIEP study:

1. Identifying Level I cutsets that provide similar unique combinations of initial and boundary conditions for post-core damage accident progression analyses. Each group so defined is called a Plant Damage State (PDS). The final 53 PDSs and sub-PDSs were developed by: manually classifying the cutsets associated with the top 50 sequences after truncation in the Level I analysis into about 300 intermediate PDSs; truncating the cutsets at 99% of the remaining probability based on TEMAC computer code point-estimate calculations for the intermediate PDSs, and consolidating the intermediate PDSs based on attributes that were assessed to have similar accident characteristics.

2. Developing core damage frequency statistics for the final set of PDSs determined. An iterative set of uncertainty calculations were performed for each PDS and sub-

PDS using TEMAC to identify a limited set of Level I variables to adequately represent the Level I calculation result obtained by sampling all Level I variables. This reduced set of Level I variables was included in the final Level II/III RMIEP analysis to assure full integration between the Level I and Level II/III efforts in the determination of the final PDS statistics.

3. Creating a logical structure for identifying the many possible accident progressions after the onset of core melt and estimating conditional probabilities for the branches in the resulting Accident Progression Event Tree (APET). The APET consisted of 135 top events, and required sampling of about 172 variables each time the APET was evaluated. The phenomenological issues that were included in the APET and evaluated by sampling distributions based on expert panel opinion, include: containment failure pressure, reactor cavity failure probability, magnitude of in-vessel hydrogen production, reactor building failure pressure, reactor building pressure following a hydrogen burn, containment loads accompanying vessel breach, peak pedestal pressure at vessel breach, and the probability of pedestal failure due to concrete erosion.

4. Identifying APET path end states that provide unique boundary conditions for source term analyses. Each such group is called an Accident Progression Bin (APB) and is defined to describe the accident progression in sufficient detail that radionuclide releases can be estimated. The 14 characteristics that were used in the definition of the APBs include: type of accident sequence, core damage timing, fraction of zirconium (Zr) oxidized in-vessel, reactor pressure vessel (RPV) condition at breach, fraction of core participating in direct containment heating (DCH) or steam explosions, containment failure mode before RPV breach, containment failure mode at RPV breach, containment failure mode late in the accident, timing of containment spray operation, type of molten core-concrete interaction, level of safety relieve valve (SRV) tailpipe bypass, level of suppression pool bypass, level of reactor building bypass, and occurrence of station blackout.

5. Estimating the magnitudes of radionuclide masses released and the timing of such releases for each APB. The radionuclides that were considered in the source term analysis include: inert gases, iodine, cesium, tellurium, strontium, ruthenium, lanthanum, cerium, and barium. Mass releases of these radionuclides were estimated parametrically rather than mechanistically. Probability distributions for sampled parameters were obtained from expert panel elicitations performed for NUREG-1150. The releases were divided into three time periods: before RPV failure, at RPV failure, and after RPV failure. Release energetics were estimated from MELCOR results. Probability distributions for the timing for various types of sequences were determined from expert opinion.

Key Results of the RMIEP Level II PRA Analyses

Level II results are summarized in the RMIEP study by the use of "summary PDS groups" and "summary APB groups" that relate to high level characteristics of interest, such as

initiating event and containment failure location, respectively. The important Level II results that are presented are summarized as follows.

1. Probability distributions are provided for arresting core damage and preventing RPV failure for external and internal initiating events. These probability distributions are conditional on the occurrence of core damage. The overall frequency-weighted-average mean value for in-vessel recovery is shown to be about 15%. The anticipated transient without scram (ATWS) and flood PDS groups are most likely to permit in-vessel recoveries (27% and 24% of the time, respectively), while in-vessel recoveries never occur for the seismic and LOCA PDS groups.

2. Probability distributions are provided for the timing of containment failure for external and internal initiating events. These probability distributions are conditional on the occurrence of core damage. The overall frequency-weighted-average mean value for early containment failure is shown to be about 33%. This value includes a contribution of 18.5% from containment failures occurring before RPV failure due to loss of containment heat removal or, in the case of ATWS sequences, insufficient containment heat removal capacity, and a contribution of 14.8% from containment failures occurring directly after RPV failure. Operation of the containment vent prior to core damage is shown to occur in 12.4% of the core damage sequences. Late breaches of containment integrity are shown to occur in 9.1% and 33.2% of the core damage sequences due to failures and venting, respectively. Finally, the containment is predicted to remain intact for 12% of the core damage sequences.

3. Mean value distributions are provided for core-concrete interaction (CCI) for external and internal initiating events. These mean values are conditional on the occurrence of core damage. CCI occurring beneath a replenishable water pool is shown to occur in 32.2% of the core damage sequences, while dry CCI is shown to occur in 44.7% of the core damage sequences. Coolable ex-vessel core debris configurations occur in 8% of the core damage sequences. The remaining core damage sequences are arrested in the vessel.

4. Complementary Cumulative Distribution Functions (CCDFs) are provided for releases of four representative radionuclides for external and internal initiating events, and mean source terms are listed for various risk-dominant APBs. The CCDFs indicate the frequency with which a release magnitude will be exceeded, and, in general, for the volatile fission product species, the structure of these CCDFs imply "that very small releases rarely occur" [NUREG/CR-5305, Vol. 1, p. 4-69].

There are obvious philosophical differences between the Level II approach developed by the RMIEP team and the approach used for CECo IPEs to-date. The two primary areas of differences are the use and quantification of the APET, and the parametric calculation of source terms. The RMIEP approach attempts to determine possible sequence progressions after core damage in an extremely comprehensive manner by using the APET construct, which treats the occurrence of possible severe accident phenomena in

a probabilistic manner. A similar comprehensive approach is taken for source term calculations by randomly sampling probability distributions for coefficients in the parametric source term code LASSOR. Although these probabilistic approaches allow the assignment of quantitative rankings to outcomes, the results are limited by the probability distributions that are assumed for and sampled during the various calculations. If one or more of the assumed probability distributions are not representative of reality, the results may be misleading.

On the other hand, the CECo BWR IPEs to-date have relied on bounding deterministic analyses to conclude whether or not a particular severe accident phenomenon will challenge containment integrity. Whichever outcome was determined to be most likely was then used as part of a so-called best estimate, or highest confidence, baseline for determining post-core damage progression and source term behavior for the plants. Sensitivity analyses were performed by assuming the opposite outcomes during accident sequences to characterize the effects on source term magnitudes. The uncertainties in phenomenological assessments are explicitly recognized and discussed in a series of papers for each CECo plant. The discussion is qualitative since data is lacking for quantified uncertainty assessment. The CECo approach is deliberately much more limited in terms of quantifying the extent of uncertainty in the answers obtained, but it is to be hoped that as understanding of severe accident phenomenology increases, the results of the two approaches will converge. Although CECo has used both approaches (the "APET" approach in the Zion Probabilistic Safety Study and the bounding approach in CECo's IPE program), the bounding approach is favored and will be used in the update of the LaSalle PRA.

In addition to the preceding philosophical differences, the following are several specific technical concerns regarding the RMIEP Level II results:

- The mean containment failure pressure used in the RMIEP study, (195 psig) appears to be too high and there is no accounting for containment strength degradation at high temperatures. LaSalle-specific studies performed for CECo suggest that the mean failure pressure is only about 147 psig, and at this pressure the failure occurs in the personnel access airlock about 2/3 of the time and in the drywell head closure the other 1/3 of the time.[1]

- It is overly conservative to assume, as the RMIEP study does, that containment failure during the ATWS and LOCA sequences always causes a sufficiently harsh reactor building environment to fail all ECCS pumps. As the preceding paragraph indicated, nearly all containment failures will occur in the drywell, well above the location of the ECCS pumps. The reference cited above[1] shows that containment failures do not significantly affect the environmental conditions below the failure site, due to the strong chimney effects created.

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[1] "Consequences of Containment Bypass Scenarios," EPRI NP-6586-L.

- The probability of containment failure due to phenomena associated with RPV failure is too high. Bounding LaSalle-specific calculations performed by CECo indicate that RPV thrust forces at vessel failure cannot move the reactor vessel. Bounding assessments performed for other CECo BWRs suggest that pressure loads associated with ex-vessel steam explosions, direct containment heating or vessel blowdown will not challenge LaSalle containment integrity unless a very high containment pressure exists prior to vessel failure.

- The α-mode (steam explosion) mean failure probability is too high. Assessments performed for other CECo BWRs have adopted the conclusion reached by the Steam Explosion Review Group and form the basis for the treatment of in-vessel steam explosions. Results of analyses performed in accordance with significant-scale experiments and expansion characteristics of shock waves forms the basis for the treatment of ex-vessel steam explosions.

  It has been concluded that the slumping of molten debris into the RPV lower plenum could not result in sufficient energy release to threaten the vessel integrity and, hence would not lead directly to containment failure. Likewise, evaluations of both the steam generation rate and shock waves induced by ex-vessel steam explosions show that these would not be of sufficient magnitude to threaten the containment integrity.

- Finally, the source term calculations take no credit for fission product retention in the drywell if containment sprays are not operating. However, a number of mechanisms are known to remove airborne fission products, including gravitational sedimentation, impaction, diffusiophoresis, and Stefan flows. Thus, the RMIEP approach is considered to be too conservative.

Summary Technical Concerns with RMIEP Level II Analyses

The containment failure pressure assumed in the RMIEP analysis is high; a lower value is more realistic.

A number of conservatisms considered in the analysis could be made more realistic:

- **Containment failure leads directly to environmental conditions failing ECCS pumps.**
- **The probability of containment failure given RPV failure is high.**
- **The probability of containment failure due to steam explosion is too high.**
- **The fission product source term model in the RMIEP study is too conservative.**

5.0 RMIEP Results

This section provides a discussion and explanation of the LaSalle RMIEP IPE/IPEEE accident sequence results. Traditional results based on the mission time of 24 hours are reported. These include overall core damage frequency, with a subsequent breakdown of core damage frequency by dominant accident sequence, initiating event contribution to dominant accident sequence, and dominant contributors to important sequences. It must be noted that many of the items identified as concerns in the previous section deal specifically with dominant contributors to the dominant sequences described below. It is therefore expected that an update to the LaSalle PRA will reveal a different set of dominant sequences than those identified by the RMIEP study.

5.1 Key Dominant Sequence Descriptions

The accident sequences developed through the RMIEP analysis are described in NUREG/CR-4832 Vol. 3, Part 1. An accident sequence is defined as a path through an event tree given any of a set of initiators to which that event tree is applicable. For example, the transient event tree is applicable to several initiators (such as main steam isolation valve (MSIV) closure or loss of condenser vacuum or loss of offsite power or a loss of a vital DC or AC bus). The results provided below represent the event tree path results for all the respective initiators that are applicable to the sequence path. The five most likely sequences, which account for 95.3% of the total core damage frequency from internal initiators, were evaluated. The sequence frequencies and contribution to total CDF is provided in Table 5-1. This section provides the results of this evaluation.

The dominant sequence is T100 which has a frequency of 2.87E-05 per year. This single sequence contributes 64.1% of the core damage frequency due to internal events. In this sequence, a transient initiating event is followed by successful scram and successful SRV operation. All core coolant makeup systems fail and core damage results from the loss of all injection in an intact containment.

The second most dominant sequence is T62 with a frequency of 6.53E-06 per year. This sequence contributes 14.6% of the total internal event core damage frequency. In this sequence, a transient initiating event is followed by a successful scram and successful SRV operation. All high pressure injection fails except RCIC. The containment and primary system heat removal functions fail. The automatic depressurization system (ADS) works but the low pressure coolant makeup systems fail.

The next three sequences (T18, T20, and T22) are essentially similar. Their combined frequency is 6.41E-06 per year, contributing 16.6% of the internal event core damage frequency. In these sequences, a transient initiating event is followed by a successful scram and successful SRV operation. Subsequently, the MFW system fails and the high pressure core spray (HPCS) system provides the high pressure injection. The normal containment and primary heat removal systems fail; all venting capability also fails. Containment pressure increases until a leak develops and this is assumed to produce an environment that causes the failure of the injection systems.

5.2 Key Dominant Sequence Analysis

Each of the dominant sequences are analyzed in this section. The frontline systems associated with each sequence are first identified. Next, the top cutsets and the driving initiating events are evaluated. The propagation of each initiating event and the failures of support or frontline systems are then summarized.

5.2.1 Sequence T100

According to the RMIEP systemic transient event tree, the sequence T100 is the combination of failures of the following frontline systems:

- Main feedwater system (MFW)
- High pressure core spray system (HPCS)
- Reactor core isolation cooling system (RCIC)
- Condensate system (CDS)
- Low pressure injection system (LPCI) in its RHR mode
- Low pressure core spray (LPCS)

The individual initiating events associated with the top cutsets for T100 are shown in Table 5-2. Additional initiators contribute to this sequence, however, they are all smaller contributors to the total sequence frequency.

The impact on the plant of each of the initiators that are identified in Table 5-2 and that are contributors to T100 is described below:

<u>IE-LOSP - Loss of Offsite Power</u>

The IE-LOSP event affects the operation of the balance of plant (BOP) systems. The loss of power results in a load rejection and turbine trip. The loss of circulating water results in a subsequent loss of the main condenser and condenser vacuum, which initiates MSIV closure. A reactor scram is initiated. All BOP systems requiring AC power trip off. These systems include the condensate system, the motor-driven feedwater pump and the recirculation pumps. Initiation of the diesel generators will occur automatically to provide power to required safety systems.

The turbine-driven feedwater pumps stop due to loss of steam flow; an independent trip due to loss of condenser vacuum is also received by these pumps. The reactor vessel water level decreases to the Level 2 setpoint which initiates RCIC and HPCS. If both RCIC and HPCS fail, the reactor water level will continue to decrease to the Level 1 setpoint at which LPCS and LPCI are initiated. ADS would initiate when the water level reaches the low-low level and the confirmatory low level signal was received. The operator would attempt to manually depressurize the RPV, if automatic depressurization were to fail .

Decay heat removal is accomplished in this scenario by the RHR system.

SBO - Station Blackout[1]

In the case of IE-LOSP and failure of all diesel generators (station blackout (SBO)), coolant makeup can still be provided by the steam-driven RCIC system. However, RCIC requires DC power for control of the system and room cooling for the pump. DC power is provided by a dedicated battery in case of an SBO; DC power is designed to provide the necessary loads for a period of about six hours. If AC is not restored during this period or other recovery actions are not taken, RCIC will fail when the batteries become depleted and DC power is lost.

The RCIC pump room cooling system requires AC power for operation; however, the high temperature trip is inoperative if both Train A and B of onsite power have failed. In this case, the RCIC system can operate without room cooling.

The RHR system cannot function without AC power, therefore, the containment pressure and temperature would increase during an SBO. The containment is expected to fail at 190 psig, based on expert panel judgement obtained during the RMIEP analysis. Given this plant condition and no containment heat removal, it is expected to take about 50 hours to reach this pressure.

IE-T5 - Complete Loss of Feedwater

A complete loss of feedwater results in decreased subcooling in the core and a subsequent reduction in core power. The water level in the core drops to Level 2, initiating HPCS and RCIC and tripping the recirculation pumps. If both RCIC and HPCS fail, the reactor water level would continue to drop to Level 1. LPCS and LPCI would automatically initiate and MSIV closure would occur. ADS would initiate when the water level reached the low-low level and the confirmatory low level signal was received. If ADS fails, the operator would attempt to manually open SRVs to depressurize the reactor and allow low pressure coolant injection.

IE-T9A - Loss of 125VDC Bus 2A

A loss of 125VDC Bus 2A results in a drywell chiller isolation and subsequent high drywell pressure. The high drywell pressure causes a reactor scram. A loss of this bus results in the loss of feedwater control and subsequent loss of the feedwater and/or condensate systems. Finally, a loss of 125VDC Bus 2A results in LPCS unavailability and also losses of portions of the ADS, RHR, and LPCI systems.

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[1] Station blackout is not an initiator; it is a condition which develops due to diesel failures subsequent to a loss of offsite power. It is discussed separately here because it has unique characteristics.

5.2.2 Sequence T62

According to RMIEP systemic event tree, accident sequence T62 is the combination of failures of the following frontline systems:

- Main feedwater system (MFW)
- High pressure core spray system (HPCS)
- Power conversion system (PCS)
- Shutdown cooling system (SCS)
- Suppression pool cooling (SPC)
- Containment spray system (CSS)
- Control rod drive (CRD)
- Condensate system (CDS)
- Low pressure injection system (LPCI) in its RHR mode
- Low pressure core spray (LPCS)

The individual initiating events associated with the top cutsets for T62 are shown in Table 5-3. Additional initiators contribute to this sequence, however, they are all smaller contributors to the total sequence frequency.

The impact on the plant of each of the initiators that are identified in Table 5-3 and that are contributors to T62 is described below:

IE-LOSP and IE-T5 were discussed above, therefore, only IE-T101 will be addressed here.

<u>IE-T101(2) Loss of 4160VAC Bus 241(2)Y</u>

A loss of 4160VAC Bus 241Y results in a drywell isolation (except MSIVs) and loss of drywell cooling which, if not promptly restored, subsequently leads to a high drywell pressure, causing a reactor scram. CRD pump 1A, RHR pump 1A, and the LPCS pump will be made unavailable by this bus failure. The other drywell chiller, CRD pump 1B, and RHR pumps 1B and 1C would be lost if Bus 242Y were lost.

5.2.3 Sequences T18/T20/T22

According to the RMIEP systemic event tree, each of the three sequences T18, T20 and T22 is essentially the combination of failures of the following frontline systems:

- Power conversion systems (PCS)
- Shutdown cooling system (SCS)
- Suppression pool cooling (SPC)
- Containment venting (Vent)

The individual initiating events associated with the top cutsets for the combination of T18, T20 and T22 are shown in Table 5-4. Additional initiators contribute to these sequences, however, they are all smaller contributors to the total sequence frequency.

The impact on the plant of each of the initiators that are identified in Table 5-4 and that are contributors to T18, T20 and T22 is described below:

IE-T9A, IE-T101(2) and IE-T5 were discussed above, therefore, only IE-T1, IE-T2 and IE-T3 will be addressed here.

IE-T1 - Turbine Trip with Turbine Bypass Available

A turbine trip is followed by an automatic scram, turbine stop valve closure and recirculation pump trip. The turbine bypass valves will open to regulate pressure but the relief valves will also be required. Feedwater will trip due to high level caused by level swell. Subsequently, the level will decrease which will cause a demand to restart feedwater. The time available for the operator to restart the motor-driven feed pumps is short and, if reactor water level is not restored by MFW, RCIC and HPCS will automatically start. If high pressure coolant makeup is not restored, level will decrease to the ADS and low pressure system initiation setpoints. RHR would be used for containment heat removal, if the condenser cannot be restored.

IE-T2 - Turbine Trip with Turbine Bypass Unavailable

This event is similar to the previous event, IE-T1, except that the bypass valve fails to open. This combination of events results in repeated demands for the SRVs to open to relieve pressure and in the unavailability of the PCS for decay heat removal. With the PCS unavailable, periodic relief valve operation is required to reduce the pressure.

IE-T3 - Total Main Steam Isolation Valve Closure

Closure of three or more of the main steam isolation valves (MSIVs) results in a scram. The loss of steam pressure causes a loss of flow from the steam-driven feed pumps. Since the steam-driven feed pumps do not trip, the motor-driven pump does not start. The water level in the reactor vessel drops to Level 2 and the RCIC and the HPCS systems initiate. If the high pressure coolant injection systems are not started, water level continues to decrease and low pressure ECCS, including ADS, will automatically initiate. Since the MSIVs are closed and are not expected to be available immediately for reopening, the PCS is not available for decay heat removal and the RHR system will be used for this purpose.

5.3 Dominant Cutset Analysis

Sequence T100

The top sixteen cutsets contributing to sequence T100 account for 80.6% of the T100 frequency, with the top two cutsets alone accounting for 61.7%. The most significant component failures and failure modes contributing to this sequence are represented by the basic events described in Table 5-5.

The first two basic events associated with T100 were linked to a RCIC "sneak circuit" issue. This issue is addressed in NUREG/CR 4832 Vol 3, Part 1, Section 7.2 and in NUREG/CR 4832 Vol 4, Section 2.6.2. This "sneak circuit" would cause the isolation of RCIC each time a loss of offsite power occurred (found during simulator exercises). Under these conditions a false, loss-of-power induced high RCIC room temperature signal was generated and the in-board AC-powered isolation valve received a signal to close. However, the valve could not close because it had no AC power. When AC was restored to the valve, a relay race ensued, and the relay associated with room high temperature was energized before the loss of power contact opened. The valve would shut isolating RCIC because it "sensed" RCIC room high temperature before it "sensed" a loss of power. This event could occur during station blackout, loss of offsite power or due to a loss of a train of AC power. The "sneak circuit" problem has been corrected by a plant modification[1].

If both of the events associated with the "sneak circuit" were eliminated from the quantification, the overall core damage frequency reported in RMIEP would be reduced by about 21%. The third basic event is the common cause beta factor used to consider the likelihood of common cause failure of the CSCS pumps. The fourth basic event is the random failure of a CSCS pump.

Sequence T62

For sequence T62, the top 100 cutsets account for 79.8% of the T62 frequency and the top 10 cutsets account for 65.62%. The most significant component failures and failure modes contributing to this sequence are described in Table 5-6. The first basic event is the common cause beta factor used to consider the likelihood of common cause failure of the CSCS pumps. The second basic event is the random failure of a CSCS pump. These two basic events are the same as discussed in relation to T100. The third and fourth basic events are random failures of the diesel generators to start or to continue to run after starting.

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[1] Modification Number M01-1-86-025 for Unit 1 (installed in May 1988) and M01-2-86-028 for Unit 2 (June 1987).

Sequence T18, T20, and T22

The top 100 cutsets for T18, T20, or T22 are similar. The most significant component failures and failure modes contributing to these sequences are described in Table 5-7. It can be seen that the survivability of components and the containment leak location are the major contributors to the cutsets; these are even more important than the RHR components and the various supporting equipment.

Summary

Because of the plant configuration cutoff date of 1985 and because of methodology differences, CECo feels that reanalysis would show that the dominant contributors to core damage would change and that sequences that are currently important would be significantly less important or eliminated. Specifically, the contribution from the dominant RMIEP sequences (representing 95% of the internal events core damage frequency) would be significantly reduced by the RCIC "sneak circuit" modification, a more realistic model of the common cause failures of diesel generator cooling, and credit for the ECCS pumps operability under low NPSH conditions.

TABLE 5-1
DOMINANT ACCIDENT SEQUENCES

Sequence Identifier	Mean Sequence CDF	Percentage of Total CDF	Cumulative % Contribution to Total CDF
T100	2.87E-05	64.1	64.1
T62	6.53E-06	14.6	78.7
T18	4.99E-06	11.2	89.9
T20	1.28E-06	2.9	92.8
T22	1.14E-06	2.6	95.3

TABLE 5-2
DOMINANT INITIATING EVENTS CONTRIBUTING TO SEQUENCE T100

Initiating Event Identifier	Initiating Event Description	Initiating Event Frequency (per year)
IE-LOSP	Loss of Offsite Power	9.6E-02
IE-9A	Loss of 125VDC Bus A	5.0E-03
IE-T5	Total Loss of Feedwater	6.0E-01

TABLE 5-3
DOMINANT INITIATING EVENTS CONTRIBUTING TO SEQUENCE T62

Initiating Event Identifier	Initiating Event Description	Initiating Event Frequency (per year)
IE-LOSP	Loss of Offsite Power	9.6E-02
IE-T5	Total Loss of Feedwater	6.0E-01
IE-T101(2)	Loss of 4160VAC Bus 241(2)Y	5.0E-03

TABLE 5-4
DOMINANT INITIATING EVENTS CONTRIBUTING TO SEQUENCES T18, T20 AND T22

Initiating Event Identifier	Initiating Event Description	Initiating Event Frequency (per year)
IE-T1	Turbine Trip with Turbine Bypass	4.5
IE-9A	Loss of 125VDC Bus A	5.0E-03
IE-T5	Total Loss of Feedwater	6.0E-01
IE-T2	Turbine Trip without Turbine Bypass	5.2E-01
IE-T101	Loss of 4160VAC Bus 241Y	5.0E-03
IE-T102	Loss of 4160VAC Bus 242Y	5.0E-03
IE-T3	Total Main Steam Isolation Valve Closure	6.1E-01

TABLE 5-5
DOMINANT CONTRIBUTORS TO T100 CUTSETS

Basic Event Name	Basic Event Description	Basic Event Probability
RCICRMCOOL-FLAG	Flag indicating failure of RCIC room cooling which in conjunction with random failures in room cooling results in closure of the RCIC isolation valve 1E-51-F063, fails RCIC.	1.0
OPFAILS-REOPEN	Operators fail to reopen RCIC isolation valve 1E-51-F063, fails RCIC.	1.0
DGCOOL-BETA	CSCS DG cooling water pump common mode beta factor, in combination with random pump failure results in loss of cooling to all DG, including the HPCS dedicated DG.	0.11
DGCOOL-PMS-CM	CSCS pump random failure for common mode DG cooling water failure, fails all emergency power.	2.5E-03

TABLE 5-6
DOMINANT CONTRIBUTORS TO T62 CUTSETS

Basic Event Name	Basic Event Description	Basic Event Probability
DGCOOL-BETA	CSCS DG cooling water pump common mode beta factor, in combination with random pump failure results in loss of cooling to all DG, including the HPCS dedicated DG.	0.11
DGCOOL-PMS-CM	CSCS pump random failure for common mode DG cooling water failure, fails all emergency power.	2.5E-03
DG(i)-GEN-LF-FTS (i)=0,2A,2B	Local fault of DG(i) results in failure to start.	2.5E-02
DG(i)-GEN-LF-FTR (i)=0,2A,2B	Local fault of DG(i) results in failure to run.	1.9E-02

TABLE 5-7
DOMINANT CONTRIBUTORS TO T18, T20 and T22 CUTSETS

Basic Event Name	Basic Event description	Basic Event Probability
CONT-LEAK	Containment overpressure without venting results in containment leakage (the complement is rupture).	7.5E-01
SUR-002-L	Severe environment induced failure of HPCS after containment failure by leak to the reactor building.	1.6E-01
SUR-003-L	Severe environment induced failure of HPCS and CRD after containment failure by leak to the reactor building.	1.6E-01
SUR-021-R	Severe environment induced failure of HPCS and CRD and either environmental failure of ADS or environmental failure of LPCS and operator failure to use diesel driven firewater after containment failure by rupture of the reactor building.	8.5E-02
1EB236B-B-UUM	AC circuit breaker to 236X and 236Y unavailable due to maintenance, fails Train B of AC power and RHR Train B.	1.0E-04
1EB236B-BCO-LF	AC circuit breaker to 236X and 236Y fails open and fails Train B of AC power and RHR Train B.	7.2E-05
RHRH01AX-HTX-LFB	RHR heat exchanger 2E12-B001A failed due to blockage, fails Train A of RHR for CHR.	6.2E-03
CSCD300A-PLG-LF	CSCS strainer 2E12-D300A faults; fails LPCS, RCIC, LPCI Train A, and DG"0".	1.2E-03

6.0 Insight Development and Evaluation

Insights are those observations regarding station configuration or practices suggested by the IPE which may affect the risk profile of the plant. Insights can suggest changes to enhance the capability of the plant and its operators to respond to an initiating event to either prevent core damage or to mitigate the consequences of core damage. The IPE insights address the capability of the existing plant to respond to an initiating event. The Accident Management insights deal with enhancements to the capability of the plant emergency response organization to respond to an accident situation, given that it has occurred. The dividing line between IPE insights and AM insights is not sharp; a distinction is made only to attempt to provide two broad categories of insights which go beyond the normal (i.e., traditional) IPE thought processes.

IPE Insight Identification

The IPE insights identified in the current study are, in many cases, significantly different from those identified in previous PRA studies. The primary difference is in completeness of the search for insights and the comprehensive coverage of all of the aspects of the RMIEP IPE/IPEEE. LaSalle IPE/IPEEE insights were identified by the analysts during the preparation for the IPE submittal (including the review of the RMIEP). IPE insights were also identified during a review of the IPE submittals for WNP-2 and NMP-2, and a review of the insights generated during the Dresden and Quad Cities IPE effort. The comprehensive efforts for identifying the IPE insights for Dresden and Quad Cities are described in Sections 1.6 and 4.7 of IPE submittals for the respective stations. The forward-looking CECo IPE program insight identification process has resulted in significantly more insights than PRA studies which looked backward from the PRA results.

IPE Insight Evaluation

An insight evaluation process has been developed to treat the 137 IPE insights identified during the IPE. The first step of the process is a distillation of the insights by a "Tiger Team", composed of individuals from the CECo PRA Group and LaSalle County Station. The distillation step consists of performing a technical review of each of the insights. The next step involves identifying the insights with the greatest impact on the risk profile. Those insights which provide a major benefit to risk reduction.

The generic procedure enhancements that were identified during the Dresden and Quad Cities IPE analyses are also applicable to the LaSalle County Station and have been forwarded to the BWR Owners Group (BWROG) and to LaSalle County Station for their consideration.

AM Insight Identification

The AM insights identified in the current study are, in many cases, significantly different from those identified in previous PRA studies. The primary difference is in completeness of the search for insights and the comprehensive coverage of all of the aspects of the IPE. The LaSalle County AM insights were principally identified by a review of the AM

insights generated during the Dresden and Quad Cities IPE effort. The process for identifying the AM insights for Dresden and Quad Cities is described in Sections 1.7 and 5.2 of IPE submittals for the respective stations. The CECo AM insight identification program resulted in a significant quantity of insights due to the extensive search for AM insights during the IPE effort.

AM Insight Evaluation

Each of the 81 individual AM insights will be evaluated by a "Tiger Team", composed of individuals from the CECo Probabilistic Risk Assessment Group and LaSalle County Station. The Team will include personnel knowledgeable in LaSalle County plant systems and operation, severe accident phenomena, emergency procedures, and the emergency response organization. The role of the review team is to evaluate the insights from a broad perspective and to make technical assessments of their potential benefit and impact. The results of the evaluation team's activities will form the basis for CECo contributions to industry and owners group AM activities.

AM insights judged to have generic procedure implications will be provided to the BWR Owners Group for further evaluation.

Insight Disposition

All insights that are developed and evaluated by the above process will be passed to the plant for disposition.

7.0 RMIEP IPEEE Review

Each of the external initiating event evaluation methods used in the RMIEP were reviewed and compared with the requirements of NUREG/CR-1407. The results of this comparison are provided in the sections below.

7.1 Seismic IPEEE

Methodology Overview

Under the Seismic Safety Margin Research Program (SSMRP) sponsored by the U. S. NRC, the Lawrence Livermore National Laboratory developed a simplified seismic PRA methodology which was concentrated on reducing the effort that is required to determine seismic hazard and to calculate seismic response for structures, systems, and components important to safety. The key elements of the LaSalle simplified seismic risk analysis include :

1. Development of the seismic hazard at the LaSalle site including the effect of local site conditions.
2. Comparisons of the best estimate seismic response of structures, components, and piping systems with design values for the purposes of specifying median responses in the seismic risk calculations.
3. Development of building and component fragilities for important structures and components.
4. Investigation of the effects of hydrodynamic loads on seismic risk.
5. Development of the system models (e.g., event and fault trees).
6. Estimation of the seismically induced core damage frequency.

The key elements and results of the LaSalle seismic analysis are briefly discussed and compared with the NRC IPEEE guideline (i.e., NUREG/CR-1407) in the following sections.

Seismic Input and Response Analysis

The methodology used to develop the hazard curves is a combination of the EUS Seismic Hazard Characterization project and the seismic experts' judgement. The mean frequency of exceedance (hazard curve) including site correction used in the RMIEP study is shown in Figure 7-1. Also shown in Figure 7-1 is the latest LLNL mean hazard curve for the LaSalle County Station. It can be seen that the RMIEP curve yields higher frequencies of exceedance than the LLNL curve for ground accelerations less than 0.3 g, but yields considerably lower frequencies of exceedance for ground accelerations greater than 0.3 g. The hazard curve was discretized to allow the convolution of the system analysis results with the seismic hazard curve. The seismic hazard curve was discretized into six intervals of peak horizontal acceleration: .18-.27, .27-.36, .36-.46, .46-.58, .58-.73, and >.73 g.

Seismic responses, together with fragilities, allow for the calculation of seismically induced failure probabilities. The SMACS methodology of the SSMRP was used in the response

analysis. SMACS analyses were performed on the LaSalle structure complex including the effects of soil-structure interaction (SSI). SMACS links together seismic input, SSI, structure response, and piping system and component response.

Hydrodynamic Load Investigation and Load Combination Approach

Although not included in the NUREG/CR-1407, the RMIEP study evaluated the probabilities of failure of a particular structure or equipment due to earthquake occurrence by including the effect of the hydrodynamic loads which may occur concurrently with the earthquake. In the RMIEP study, a load event tree was depicted based on the earthquake level, actuation of any one of several types of SRV discharges, and pipe breaks. The hydrodynamic loads identified for LaSalle County Station were: Safety/relief valve discharge loads, LOCA-induced loads, jet forces, pool swell, condensation-oscillation (CO), and chugging. It was found that only vertical accelerations induced by CO loads and SRV actuation could both occur concurrently with the earthquake and could be of large enough magnitude to warrant further attention. An elaborate load combination methodology was used in the RMIEP study for this evaluation. However, hydrodynamic loads which may be experienced in BWRs during an earthquake are not significant at LaSalle. The average hydrodynamic load (averaged over all responses) was 0.04 g.

Plant Logic Models

The classes of seismic initiating events which were considered for LaSalle are essentially the same as those in the internal event analysis. The frequencies of the conditional seismic initiators and the analogous internal initiating events are listed below:

Initiating Event	Conditional Seismic Initiators	Internal Events Initiators (per year)
Reactor Vessel Rupture	1.0E-9	< 3E-7
Large LOCA	1.2E-4	1E-4
Medium LOCA	4.5E-3	3E-4
Small LOCA	1E-1	3E-2
Loss-of-Offsite Power	9E-1	9.6E-2

The fault tree models of the safety systems included both random and seismically induced failures. However, only seismically induced failures are included in the definition of the initiating events. Thus, the calculated CDF is a full measure of, and only of, the seismic increment to risk.

The event trees for the seismic events are taken directly from the RMIEP internal event trees, with two simplifying modifications. First, the event trees, whose systems are

dependent upon offsite power, were eliminated since a loss-of-offsite power was assumed during a seismic event. The second simplification is the elimination of the suppression pool cooling and containment spray systems from the Large and Medium LOCA event trees and the venting system from all of the event trees since the RMIEP seismic study was concerned only with core damage, and not beyond.

The safety system fault trees of the RMIEP internal events were modified to include seismic failure modes. Any event in the fault tree which could be the result of either a random failure or a seismically induced failure was modified by adding OR-gates with two basic event inputs. After the event trees and fault trees were developed, a detailed database providing the basic events, associated response fragility, and random failure data was generated to feed into the SEISIM code to yield the CDFs for all six levels of earthquake.

Results and Conclusions

The seismically induced CDF was estimated to be 6E-7/year for LaSalle Unit-2. The most significant observation is that the risk contribution is seen to monotonically decrease as one moves from the lowest earthquake (Level 1) to the highest level (Level 6). This means that the greatest seismic risk is contributed by the smaller earthquakes. A large portion of the risk (90%) fell into the lower three earthquake levels. By contrast, in the Cook Station IPEEE, a pressurized water reactor (PWR), it was found that the greatest contribution to risk came from earthquakes having ground accelerations in the range of 0.26 to 0.75 g. The dominance of the seismic risk by the low level earthquakes is basically due to loss-of-offsite power during an earthquake. The conditional CDF increases with increasing earthquake level (e.g., the conditional CDF increases by a factor of 4 from Level 1 to Level 6). However, low level earthquakes are much more frequent than high level earthquakes (e.g., a Level 1 earthquake is 100 times more frequent than a Level 6 earthquake). In other words, the CDF does not increase with earthquake intensity as rapidly as the falloff in frequency of having a higher intensity earthquake. This observation is also significant in terms of the hazard curve used. As mentioned above, the hazard curve used in the RMIEP study may not be conservative for ground acceleration greater than 0.3 g as compared to the latest LLNL hazard curve. However, the underestimation is not significant since the contributions of CDF from the high level earthquakes are not significant.

It is of interest that the three most dominant accident sequences are the loss-of-offsite power transient sequences, contributing roughly 90% of the seismic CDF.

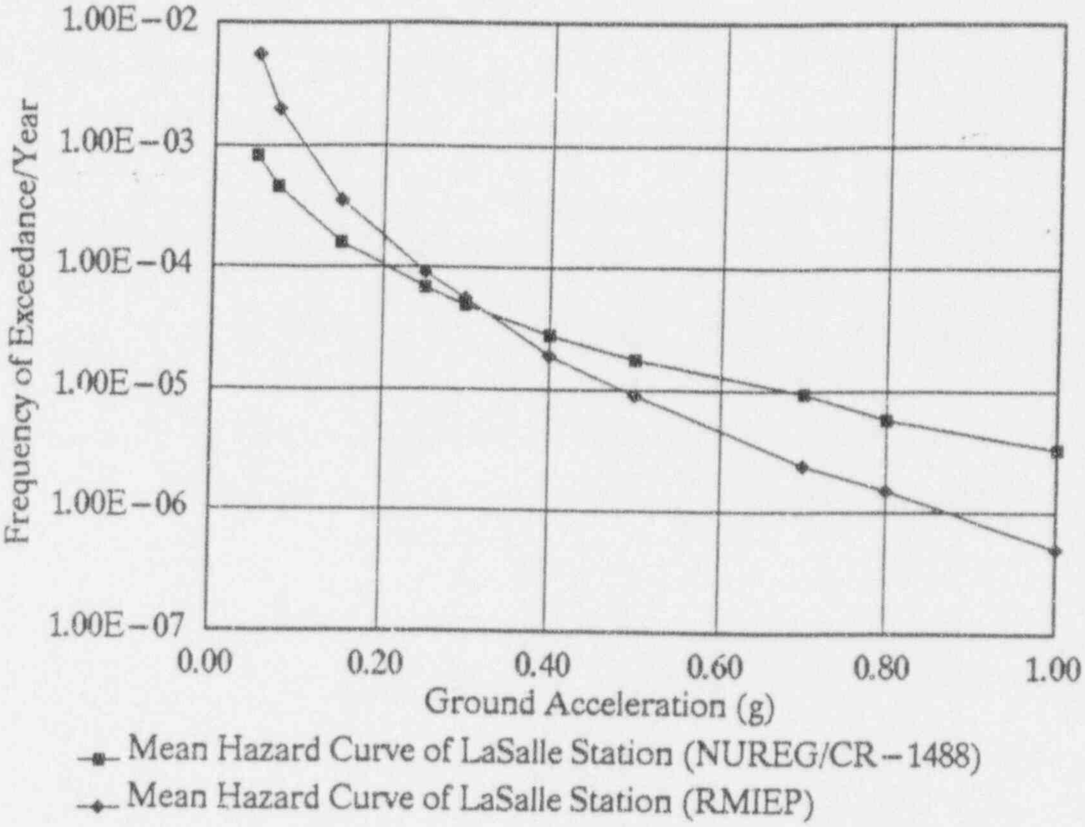

FIGURE 7-1
MEAN HAZARD CURVES OF LaSalle County Station

7.2 Fire IPEEE

Methodology Overview

The RMIEP work employs a detailed initial screening analysis to demonstrate that most plant areas are of no risk significance from a fire contribution standpoint. This screening involves:

- The identification of fire areas of potential interest;
- The initial quantification of the fire initiating event frequencies for each plant area;
- The determination of key accident sequences to be quantified;
- The mapping of all cables (and routing) to components on the fault trees for the key sequences;
- The solution of the fault trees and random failures;
- The screening of the accident sequence cutsets based on fire barriers, random failures and operator actions, and;
- The final screening of cutsets based on absolute frequency.

A final screening was conducted on all remaining fire areas which involved the actual modelling of fire propagation using "COMPBRN", a probabilistic assessment of fire barrier integrity, a fire recovery analysis, and a final probabilistic quantification.

Fire Initiators and Response Analyses

Initiators

The RMIEP analyses employed a very extensive list of initiating events for the fire analyses. Appendix C to Vol.9 of NUREG/CR-4832 delineates this list which is drawn from industry wide fire experience. The actual employment of these numerous initiators in the frequency determination for various LaSalle plant areas is consistent with past PRA fire analysis practice. Very few of the "initiating events" cited in the RMIEP work did, or even could, lead to any significant fire having any significant likelihood of propagation, even if unchecked. The overall assessment of this issue is that the RMIEP approach is very conservative; the initiating event frequency may have been overestimated (in CECo's judgement) by a factor of 10 or more for some areas.

Response Analyses

The RMIEP work found ten plant areas having substantive fire-related contributions to CDF. These were:

1. Control room
2. Turbine building corridor
3. Cable spreading room
4. Electrical equipment room
5. Auxiliary equipment room (2 areas)

6. Division 2 essential switchgear room
7. Division 1 essential switchgear room
8. Auxiliary building Rad Chem offices
9. BOP cable area north
10. Cable shaft area

The mean total CDF due to fire at LaSalle was found to be 3.21E-5 per year.

The results of the analyses are heavily driven by the interactions of the tails of the distributions for individual sub-tier analyses. This can readily be discerned by examining the distributions and considering the mean versus median values derived for each. In some sense, while mathematically pure, these results tend toward the extreme if viewed on the basis of mean values alone.

The RMIEP work also assigned significant penalties to the control room panel design at LaSalle as a result of the "open top" feature of the panels (a ventilation feature). The analyses performed were quite conservative in that they assumed panel fire internal propagation and smoke/combustion product off-gassing which rendered the control room uninhabitable very quickly.

Results and Conclusions

The RMIEP work shows that fire is a significant contributor to mean total CDF at LaSalle. However, the conservatism inherent in the RMIEP effort appears to have the effect of accentuating the result. The fire analysis in RMIEP may be iudged to be conservative in terms of its representation of the fire contribution to relative risk; due to the magnitude of this conservatism, plant modifications would not be recommended based upon this analysis.

Summary Technical Concerns with Fire IPEEE

- **Fire initiator frequency determination is conservative.**

7.3 Internal Flooding Analysis

Methodology Overview

The first step in the analysis was to define the potential flood sources within the plant and the sets of locations that would be affected by those sources. The plant was broken up into unique locations that were of sufficient detail to reflect all possible flood scenarios. The largest source of water in each location was determined and full rated flow was assumed for a given period of time to determine the flood volume. In this analysis, systems were assumed to operate 30 minutes at rated flow.

Once the flood volumes associated with each source were established, the areas of the plant potentially affected by the flood sources were determined and flood water propagation rules were established. The propagation rules considered the available area

within each location, the potential for water retention within a location, available area for flooding within a location, flow paths out of a location, and the existence of flood barriers.

The next step in the analysis was a subsuming process that was done by looking at floods that started in each location and eliminating all the floods that flooded smaller areas and had smaller amounts of water. (For example, if two flood scenarios started in a location W but one flood scenario flooded locations WXYZ and the other flooded locations WXY, the WXY flood scenario was allowed to be subsumed by the WXYZ flood scenario).

The flooding impact on the systems (fault trees) was then determined. The result of this process was a Boolean screening on the flood scenarios that remained after the subsuming process. Conservative initiating event frequencies were assumed and the fault trees were quantified using a cutoff value of 1E-08 (cutsets below this value were not retained).

Another screening was then performed using realistic human error and recovery probabilities as well as realistic maintenance modeling. As a result the following flood scenarios remained prior to a plant visit:

1. those that propagate throughout the reactor building and eventually reach the pump rooms,

2. those that flood the turbine and/or auxiliary building and potentially propagate to the control room, and

3. those that flood the turbine or auxiliary buildings, or the diesel rooms with water potentially propagating to the switchgear rooms.

As a result of a plant walkdown the only flood scenarios found to be potentially important were those that originated in the reactor building.

Another screening was then performed on these reactor building scenarios for each of the core damage transient sequences. This process was aided by a matrix which listed the systems and the system component locations in the reactor building. This screening resulted in reducing the number of flood scenarios that could actually lead to core damage to only two.

The last step in this study was the quantification of these two reactor building flood scenarios. This quantification included the construction of event trees for the initiation, propagation, and isolation of each of these two scenarios.

Results and Conclusions

The contribution to the core damage frequency from internal flooding at LaSalle Unit 2 nuclear power plant was determined to have a mean frequency of 3.4E-06 per year. This core damage contribution was due to two flood scenarios. The contribution to core

damage frequency for the two scenarios was 2.1E-07 per year and 3.2E-06 per year respectively. All other flood scenarios were eliminated after the detailed location analysis and flood propagation analysis showed that their frequencies were each on the order of 1E-08 per year or lower.

In general, the results of the analysis show that the LaSalle plant was very well designed for minimizing the impact of floods on plant equipment and for reducing the probability of core damage as a result of a flood. In almost all plant areas, the separation of equipment was adequate to allow operation of sufficient equipment to avoid core damage. In most cases, two independent systems were still functional, reducing the frequency of core damage from these scenarios to less than the frequency cutoff for this analysis (1E-08/yr). It should be noted that both of the flood scenarios that were found to be important involved the failure of service water system piping, which lead to main feedwater and condensate system failures.

Several items of concern regarding this internal flooding analysis methodology were identified by the RMIEP. These concerns are briefly described below.

- The flood definition process needs to be improved. In the RMIEP analysis, the largest pipe in each of the systems appearing in the risk model and all the other systems in each room were identified as the flood initiators. It was found that the initiators required more specific definition than provided through this process.

- The flood propagation should be more precisely defined in the initial stages of the analysis. In several situations the overly conservative assumption that a flood would fail all equipment on an intermediate level in the reactor building led to difficulties later in the analyses.

- Subsuming of one flood by a larger flood may mask subtle effects and, if the flood propagation definition has been overly conservative, can result in the expenditure of effort analyzing an incorrect scenario.

CECo will perform an internal flooding analysis of LaSalle that is consistent with those performed in its prior IPEs, during the updating of the LaSalle PRA.

7.4 Other External Events

Methodology Overview

An extensive review of information on the site region and plant design was made to identify all external events to be considered in the RMIEP study. The external events identified are:

> Aircraft impact, Avalanche, Biological events, Coastal Erosion, Drought, External flooding, Extreme winds and Tornadoes, Fog, Forest fire, Frost,Hail, High tide, high lake level or high river stage, High summer temperature, Hurricane, Ice cover, Industrial or military facility accident, Internal flooding, Landslides, Lightning, Low

lake or river water level, low winter temperature, Meteorite, Pipeline accident, Intense precipitation , Release of chemicals in onsite storage, River diversion, Sandstorm, Seiche, Seismic activity, Snow, Soil shrink-swell or consolidation, Storm surge, Transportation accidents, Tsunami, Toxic gas, Turbine generated missiles, Volcanic activity, and Waves.

Aside from seismic, fire and internal flood which have already been included in the detailed external hazards analysis, the following events were identified for a more detailed study after an initial screening process was carried out:

1. Military and industrial facilities accidents
2. Pipeline accidents
3. Release of chemicals in onsite storage
4. Aircraft impact
5. External flooding
6. Transportation accidents
7. Turbine missiles
8. Winds and tornadoes

The top three events in this group were further eliminated based on the analyses and information which is presented in the LaSalle FSAR.

A probabilistic bounding analysis was performed for each of the remaining five events in the above list. For aircraft impact, the median frequency of CDF was calculated as 5E-7/year. The bounding analysis for external flooding showed that the probability of occurrence of the probable maximum precipitation at the site is indeed very low. The bounding analysis for transportation accidents, including toxic chemical release and chemical explosions, showed that these accidents do not significantly contribute to the plant risk. The 95 percent confidence bound on the CDF due to turbine generated missiles is on the order of 1E-7/year. Extreme winds were eliminated after the plant structures were evaluated. The median frequency of CDF due to tornadoes was calculated to be 3E-7/year.

Since the plant system failures and consequence analysis were conservatively neglected for the bounding analyses, the CDF frequencies mentioned above should not be directly compared with the other CDF frequencies reported in the RMIEP study.

Results and Conclusions

Due to the conservatism introduced in the bounding analyses by neglecting the plant system failures and consequence analysis and due to the low CDFs resulting from these bounding analyses, the RMIEP study concluded that none of the external events listed above presented a significant contributor to the plant risk.

8.0 Conclusions

The LaSalle County Station RMIEP project resulted in a very comprehensive PRA. It has provided CECo with a new level of understanding of the plant and its behavior under a variety of potential accident scenarios.

The LaSalle mean core damage frequency due to internal events was determined to be 4.4E-05 per year. Of the total core damage frequency, over 64% is due to one sequence in which a transient is followed by failure of all high and low pressure injection. The next most likely sequence (contributing 15% to CDF) is a transient in which all high pressure and low pressure injection systems (except RCIC) fail. The next three sequences collectively contribute approximately 16% to the total CDF. These sequences are similar and are due to failures of the ability to remove heat from the containment.

The contribution to core damage due to external events is composed of contributions for fire, seismic and for internal flooding. The mean core damage frequency for fire events is 3.21E-05 per year. The mean core damage frequency for internally initiated flood events is 3.39E-06 per year. The mean core damage frequency for seismic events is 7.58E-07 per year.

Commonwealth Edison's RMEIP analysis team has performed the review of the RMIEP study for applicability as CECo's response to the objectives of Generic Letter 88-20 for LaSalle County Station. They have, therefore, been intimately involved in the LaSalle RMIEP IPE/IPEEE process. As a result of this program, CECo personnel have developed a unique understanding of the behavior of the LaSalle plant under accident conditions and of the total plant capabilities to respond to accidents.

Because of the plant configuration cutoff date of 1985 and because of methodology differences, CECo feels that reanalysis would show that the dominant contributors to core damage would change and that sequences that are currently important would be significantly less important or eliminated. Specifically, the contribution from the dominant RMIEP sequences (representing 95% of the internal events core damage frequency) would be significantly reduced by the RCIC "sneak circuit" modification, a more realistic model of the common cause failures of diesel generator cooling, and credit for the ECCS pumps operability under low NPSH conditions.

Although there are several technical concerns noted in this summary document, the principal purpose of the LaSalle County Station RMIEP IPE/IPEEE was to develop an understanding of the severe accident behavior of LaSalle and of the severe accidents postulated by the analysis. It accomplished this purpose. CECo has gained a better understanding of the probability of core damage at the LaSalle County Station as a result of this review. The numerous insights developed during the LaSalle RMIEP IPE/IPEEE process will be provided to the station for disposition. Those insights dealing with accident management will form the basis for future development and implementation of the LaSalle County Station Accident Management program.

APPENDIX A

ACRONYM LIST

Appendix A - Acronym List

AC	Alternating Current
ADS	Automatic Depressurization System
AM	Accident Management
APB	Accident Progression Bin
APET	Accident Progression Event Tree
ATWS	Anticipated Transient Without Scram
BOP	Balance of Plant
BWR	Boiling Water Reactor
BWROG	BWR Owners Group
CCA	Common Cause Analysis
CCI	Core Concrete Interaction
CCDF	Component Cumulative Distribution Function
CDF	Core Damage Frequency
CDS	Condensate System
CECo	Commonwealth Edison Company
CO	Condensation-Oscillation
CRD	Control rod drive
CSCS	Core Standby Cooling System
CSS	Containment Spray System
DC	Direct Current
DCH	Direct Containment Heating
ECCS	Emergency Core Cooling System
EOP(s)	Emergency Operating Procedure(s)
HEP(s)	Human Error Probability(ies)
HPCS	High Pressure Core Spray
HRA	Human Reliability Analysis
IE	Initiating Event
IE-T1	Turbine Trip with Turbine Bypass Available Initiating Event
IE-T2	Turbine Trip with Turbine Bypass Unavailable Initiating Event
IE-T3	Total Main Steam Isolation Valve Closure Initiating Event
IE-T5	Complete Loss of Feedwater Initiating Event
IE-T9A	Loss of 125VDC Bus 2A Initiating Event
IE-T101(2)	Loss of 4160VAC Bus 241(2)Y Initiating Event
IE-LOSP	Loss of Offsite Power Initiating Event
IPE	Individual Plant Examination
IPEEE	Individual Plant Examination (External Events)
LLNL	Lawrence Livermore National Laboratory
LLOCA	Large Loss of Coolant Accident
LOCA(s)	Loss of Coolant Accident(s)
LPCI	Low Pressure Coolant Injection
LPCS	Low Pressure Core Spray
LSCS	LaSalle County Station
MAAP	Modular Accident Analysis Program
MFW	Main Feedwater

MLOCA	Medium Loss of Coolant Accident
MSIV(s)	Main Steam Isolation Valve(s)
NMP2	Nine Mile Point 2
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
PCS	Power Conversion System
PDS	Plant Damage State
PES(s)	Phenomenological Evaluation Summary(ies)
PRA	Probabilistic Risk Assessment
RCIC	Reactor Core Isolation Cooling
RHR	Residual Heat Removal
RMIEP	Risk Methods Integration and Evaluation Program
RPV	Reactor Pressure Vessel
SBO	Station Blackout
SCS	Shutdown Cooling System
SMACS	[Undefined in RMIEP study]
SPC	Suppression Pool Cooling
SRV	Safety Relief Valve
SSI	Soil-Structure Interaction
SSMRP	Seismic Safety Margin Research Program
T100	Transient Sequence Number 100
T18	Transient Sequence Number 18
T20	Transient Sequence Number 20
T22	Transient Sequence Number 22
T62	Transient Sequence Number 62
THERP	Technique For Human Error Rate Prediction
UFSAR	Updated Final Safety Analysis Report
Vent	Containment venting
WNP2	Washington (Public Power Service) Nuclear Plant 2

COMPARISON OF NUREG-1335 REQUIREMENTS TO THE LASALLE COUNTY STATION RMIEP IPE REPORT

NUREG-1335 Item (Section)	LaSalle County IPE Reference
• Executive Summary (Table 2.1)	Executive Summary Summary of LaSalle County RMIEP IPE NUREG-4832 Volume 1
• General Methodology (2.1.1)	NUREG-4832 Volume 3, Part 1
• Information Assembly (2.1.2)	NUREG-4832 Volume 1 Section 1.4
- Plant Layout (other than FSAR info)	NUREG-5305 Volume 1; NUREG-4832 Volume 1
- List of PRA Studies (reviewed)	Summary of LaSalle County RMIEP IPE WNP-2 IPE; Nine Mile Point IPE
- Description of Plant Documentation	NUREG-4832 Volume 5. IPE Reflecting the Plant Documentation of 1985-87
- Description of Walkdown Activity	NUREG-4832 Volume 10 Section 4 (Internal Flooding issues only)
• Accident Sequence Delineation (2.1.3)	NUREG-4832 Volume 4
- List of Initiating Events	NUREG-4832 Volume 4
- Developed or Adapted Event Trees	NUREG-4832 Volume 4
- Separate Event Trees - Special Events	NUREG-4832 Volume 4
- Support System Event Trees	Not Applicable (linked fault tree methodology)
- Accident Sequence Grouping Discussion	NUREG-5305 Volume 1 Section 2.2-2.5
- List or Summary of Bins of Accident Sequence	NUREG-5305 Volume 1 Section 2.5
- Spatial or Phenomenological Dependencies	NUREG-5305 Volume 1 Section 3.2,3.3

COMPARISON OF NUREG-1335 REQUIREMENTS TO THE LASALLE COUNTY STATION RMIEP IPE REPORT (Continued)

NUREG-1335 Item (Section)	LaSalle County IPE Reference
• System Analysis (2.1.4)	NUREG-4832 Volume 6 (Individual System Fault trees)
- Description Simplified Diagram FL & Supp Sys.	NUREG-4832 Volume 6 (Individual System Fault trees)
- Fault Trees Retained (i.e., not in Report)	NUREG-4832 Volume 6 (Individual System Fault trees)
- Dependency Matrix Including Shared Systems	NUREG-4832 Volume 6 (by system)
- Differences Between Subject Plant & Reference Plant	Not Applicable
- Method Used for Unavailability of Hardware & Data	NUREG-4832 Volume 2, Section 2.2.4, Volume 5
• Quantification Process (2.1.5)	NUREG-4832 Volume 3, Part 1
- Types of Common Cause & Data	NUREG-4832 Volume 5
- Internal Flooding Initiators & Quantification Impact	NUREG-4832 Volume 10
- Types of Human Failure Considered	NUREG-4832 Volume 3, Part 1, Section 5
- List of Human Rel. Data & Time Available for O.R.A.	NUREG-4832 Volume 3, Part 1, Section 5
- List of Plant Specific Exp. Items, Method Used & Data	Mostly Generic data used. Selective LaSalle Specific, (NUREG-4832 Volume 5).

COMPARISON OF NUREG-1335 REQUIREMENTS TO THE LASALLE COUNTY STATION RMIEP IPE REPORT (Continued)

NUREG-1335 Item (Section)	LaSalle County IPE Reference
- Method for Accident Sequence Quantification	NUREG-4832 Volume 3, Part 1
• Front-end Results & Screening (2.1.6)	NUREG-4832 Volume 3 Part 1
- Describe How Screening Criteria Used	NUREG-4832 Volume 3 Part 1
> Identify Sequences Meeting More than one of the following criteria	NUREG-4832 Volume 3 Part 1
> Systemic Sequence $>1x10^{-7}$/yr	NUREG-4832 Volume 3 Part 1
> Systemic Sequence in Upper 95%	NUREG-4832 Volume 1 Summary Section 5
> Upper 95% Total Containment Failure Freq.	NUREG-4832 Volume 3 Part 1
> Bypass Frequency $>1x10^{-7}$/yr	NUREG-4832 Volume 6
> Any Sequence Determined to be Important	NUREG-4832 Volume 1
- List of Sequences Selected	Summary Section 5 - Dominant Sequences

COMPARISON OF NUREG-1335 REQUIREMENTS TO THE LASALLE COUNTY STATION RMIEP IPE REPORT (Continued)

NUREG-1335 Item (Section)	LaSalle County IPE Reference
- List of Major Contributor to Those Selected	Summary Section 5 - Dominant Contributors to Dominant Accident Sequences
- Thorough Discussion of Decay Heat Removal	NUREG-5305 Volume 1 Section 3.2
- List of any "Vulnerabilities" (i.e., areas for improvement)	NUREG-4832 Volume 3 Part 1 Section 7.4
- Identification of Sequences that are low due to low human error rates	Not Applicable (Linked Fault Tree Methodology - recovery action used)
- Other USI or GSI Evaluations	NONE
• Back-End - Containment Response (2.2)	NUREG-5305 Volumes 1 and 3
- Intro.	NUREG-5305 Volume 1 Section 1.0
• General Methodology (2.2.1)	NUREG-5305 Volume 1 Section 1.2
- Cognizance & Use of the Issue Papers	Not Applicable
• Plant Data & Description (2.2.2.1)	NUREG-5305 Volume 1 Sections 3.2,3.3
- Identify & Highlight Significant Data	NUREG-5305 Volume 1 Section 3.3
- Geometry	NUREG-5305 Volume 1 Section 3.2
• Plant Models & Methods for Physical Processes (2.2.2.2)	NUREG-5305 Volume 3 Sections 2.0,3.0
- Provide Documentation of Analytical Model	NUREG-5305 Volume 3 Sections 2.0,3.0

COMPARISON OF NUREG-1335 REQUIREMENTS TO THE LASALLE COUNTY STATION RMIEP IPE REPORT (Continued)

NUREG-1335 Item (Section)	LaSalle County IPE Reference
- Reference Those Well Represented in Literatures	NUREG-5305 Volume 3 Section 2.0
• Bins & Plant Damage Status (2.2.2.3)	NUREG-5305 Volume 1 Sections 2.2-2.5
- Justification for Bins on Back-End Similarity	NUREG-5305 Volume 1 Sections 2.6,2.7
- Results of Binning	NUREG-5305 Volume 1 Section 2.5
• Containment Failure Characterization (2.2.2.4)	NUREG-5305 Volume 1 Sections 3.2-3.5
- Comparison or Assessments of Strength	NUREG-5305 Volume 1 Section 3.2; Volume 2 App. B.7
- Failure Mechanisms	NUREG-5305 Volume 1 Sections 3.3-3.5
• Containment Event Trees (2.2.2.5)	NUREG-5305 Volume 1 Section 3.3
- Pathways for Isolation Failures	NUREG-4832 Volume 6 (Individual System Fault trees)
- Signals Required to Isolate Penetration	NUREG-4832 Volume 6 (Individual System Fault trees)
- Potential for Generating the Signals for all Events	NUREG-4832 Volume 6 (Individual System Fault trees)
- Examination of Testing & Maintenance Procedures	NUREG-4832 Volume 5 Section 4.3
- Quantification of each Isolation Failure Mode	NUREG-4832 Volume 6 (Individual System Fault trees quantification)
• Accident Progression & CET Quantification (2.2.2.6)	NUREG-4832 Volume 3

COMPARISON OF NUREG-1335 REQUIREMENTS TO THE LASALLE COUNTY STATION RMIEP IPE REPORT (Continued)

NUREG-1335 Item (Section)	LaSalle County IPE Reference
- Characterize Containment Performance for each CET	NUREG-5305 Volume 1 Sections 3.4,3.5
- Each Predicted Load Should be Adequately Supported	NUREG-5305 Volume 1 Section 3.3
- Use of EOP & Verify Adequate Training	NUREG-4832 Volume 5
- Documentation to Support Availability & Survivability	Survivability in post-accident phase: NUREG-4832 Vol 3 part 1, section 6.
- Description of Data Used in Conditional Probability of Non-Isolation	Not Applicable
- Description of Sequence Assessment for Bypass	NUREG-4832 Volume 6 (Individual System Fault trees quantification)
- Methods for Handling Phenomenological Uncertainty	Phenomenological Evaluation Summaries (brief statement in Summary)
• Radionuclide Release Characterization (2.2.2.7)	NUREG-5305 Volume 1 Section 4.0
- Estimates of Source Term for Sequences Exceeding Criteria	NUREG-5305 Volume 1 Sections 4.1-4.5
- Rationale for Combining Source Term into Categories	NUREG-5305 Volume 1 Section 3.4
- Containment Failure Mode & Timing	NUREG-5305 Volume 1 Section 3.5

COMPARISON OF NUREG-1335 REQUIREMENTS TO THE LASALLE COUNTY STATION RMIEP IPE REPORT (Continued)

NUREG-1335 Item (Section)	LaSalle County IPE Reference
- Accident Management with Regard to Source Term	NONE
- Ranking of Release Categories	NUREG-5305 Volume 1 Section 6.3
• Safety Features & Potential Plant Improvements (2.3)	NUREG-5305 Volume 1 (Summary section)
- Unique/or Important Features	NUREG-5305 Volume 1 (Summary section)
- Worthwhile Strategies for Which Credit was Taken	NUREG-5305 Volume 1 (Section 7.1.1.5)
- Identify Potential Improvements Implemented or Selected for Implementation	NUREG-4832 Volume 3 Part 1 Section 7.4
- Discuss Anticipated Benefits & Downside	NONE
- Tabular form Options Scheduled for Implementation & Respective Timing	NONE
• Utility Team & Internal Review (2.4)	Summary Section 2
- Cognizance	Summary Section 2
- Quality Control	Summary Section 2
• Consideration of External Events (2.5)	Summary Section 7, NUREG-4832 Volumes 7, 8, 9