suppression system actuations, (2) effects of fire protection system actuation on non-safety related and safety-related equipment, (3) effects of moisture intrusion on non-safety related and safety-related equipment (the portion of this issue related to flooding has been adequately addressed), (4) non-safety-related control system/safety-related system dependencies, (5) seismically-induced fires, (6) seismically induced flooding, and (7) effects of hydrogen line ruptures) were not addressed in the IPEEE submittal. The need for any additional assessment or actions related to the resolution of these individual issues for LaSalle will be addressed by NRC staff separately from the IPEEE program.

On the basis of the review of RMIEP for IPEEE issues, the staff concludes that the licensee's IPEEE process is capable of identifying the most likely severe accidents and severe accident vulnerabilities. The RMIEP and PRUEP studies extensively involved LaSalle personnel. Thus, the licensee has gained a greater understanding of the LaSalle plant than would otherwise have been attained without the PRA. Based on this involvement, the licensee would have (1) developed a greater appreciation of severe accident behavior, (2) a better understanding of the most likely severe accident sequences that could occur at LaSalle, and (3) gained a qualitative understanding of the overall likelihood of core damage and fission product releases. Thus, the three major objectives of the IPEEE program (Supplement 4 to GL 88-20) have been satisfied, even though not all of the external event aspects of the USIs and GSIs have been resolved. Therefore, the staff considers that the intent of Supplement 4 to GL 88-20 has been satisfied for LaSalle.

If you have any comments regarding the enclosed evaluation, please contact me at (301) 415-1322.

Sincerely,

/RA/

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Project Directorate III
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Docket Nos. 50-373 and 50-374

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*Safety evaluation provided by memo dated 11/29/00 incorporated with no significant changes.

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

December 8, 2000

Mr. Oliver D. Kingsley, President
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SUBJECT: LASALLE COUNTY STATION, UNITS 1 AND 2, NRC STAFF EVALUATION OF THE INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS (IPEEE) SUBMITTAL (TAC NOS. M83634 AND M83635)

Dear Mr. Kingsley:

On June 28, 1991, the Nuclear Regulatory Commission (NRC) issued Generic Letter (GL) 88-20, Supplement 4, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - Title 10 CFR50.54(f)," requesting that each licensee perform an IPEEE to identify plant-specific vulnerabilities to severe accidents, and report the results to the NRC together with any licensee determined improvements and corrective actions. By letter dated April 28, 1994, Commonwealth Edison Company (ComEd) provided its response to GL 88-20, Supplement 4, for LaSalle County Station, Units 1 and 2.

The staff performed a screening review which examined the IPEEE results for their "completeness and reasonableness" considering the design and operation of the plant. On the basis of this review and further review by a senior review board (SRB), the staff concluded that the aspects of seismic events, fires and high winds, floods, transportation and other external events (HFO) were adequately addressed. The SRB is comprised of Office of Research and Office of Nuclear Reactor Regulation staff and consultants (Sandia National Laboratories) with probabilistic risk assessment expertise for external events. The staff's review findings are summarized in the enclosed safety evaluation report (SER), and the details of the contractor's findings in the Technical Evaluation Reports appear in Attachments A and B to the SER.

GL 88-20 recognized that many licensees had plant-specific probabilistic risk assessments (PRAs) or similar analyses. The use of existing PRA analyses was encouraged in achieving the objectives of the Individual Plant Examination (IPE) program. LaSalle made use of this provision. As a result, the LaSalle submittal was brief and referred to the information in NUREG/CR-4832, *Analysis of the LaSalle Unit 2 Nuclear Power Plant: Risk Methods Integration and Evaluation Program (RMIEP)*, and NUREG/CR-5305, *Integrated Risk Assessment for the LaSalle Unit 2 Nuclear Power Plant: Phenomenology and Risk Uncertainty Evaluation Program (PRUEP)* which was performed by Sandia National Laboratories (SNL) and sponsored by the United States Nuclear Regulatory Commission (USNRC). The information in these two reports is sufficient to identify potential plant vulnerabilities to severe accidents from external events, but it is incomplete in other areas (i.e., the resolution of related generic safety issues).

LaSalle is a 0.3g focused scope plant (per NUREG-1407). The licensee relied exclusively on the information in RMIEP and PRUEP, which is a full scope level III PRA. The seismic portion of the LaSalle RMIEP (on which the IPEEE was based) was performed by Lawrence Livermore National Laboratories (LLNL) and its subcontractor EQE. Based on the RMIEP study, the total seismic core damage frequency was estimated to be $6x10^{-7}$ per reactor-year (ry).

For the fire evaluation, the licensee relied exclusively on the information in the RMIEP. The fire analysis includes all of the major steps of a fire PRA (circa 1985) and is comparable with the Electric Power Research Institute (EPRI) fire-induced vulnerability evaluation (FIVE) methodology fire compartment interaction analysis (FICA). RMIEP models were used for some portions of the fire analysis. The definition of a fire area was not provided in the submittal. Walkdowns of the plant were conducted to support the RMIEP analysis. The mean fire-induced core damage frequency (CDF) as determined in RMIEP is $3.2x10^{-5}$/ry. The submittal did not revise the probable maximum precipitation with current information as was specified in NUREG-1407. However, RES's contractor, SNL, did perform an assessment using new data to address external flooding. The licensee has eliminated all other external events (i.e., high winds, tornadoes, and others) on the basis of estimated CDF values of less than $1x10^{-6}$/ry. The licensee estimated previously a CDF due to internal events as about $4.7x10^{-5}$/ry, including internal flooding.

The licensee did not explicitly provide a definition of a severe accident vulnerability. However, the licensee stated that the RMIEP results were well within the safety goals (Safety Goal Policy Statement, 51 FR 28044, August 4, 1986).

The licensee identified no plant improvements resulting from the fire, seismic, or HFO analyses in the IPEEE submittal. The RMIEP report identified three issues in the fire area that were "provided to the station for disposition." These were (1) to install tops on the main control room electrical panels, (2) to institute an inspection program for penetration seals at the top of switchgear panels, and (3) "insights dealing with accident management." LaSalle's submittal contains no additional discussion of these specific recommendations.

The submittal does not discuss any unresolved safety or generic issues, but points to the RMIEP documentation. Since the RMIEP is based on the plant as of 1985 (i.e., before Supplement 4 to the GL 88-20 was issued), it does not explicitly discuss Unresolved Safety Issue (USI) USI A-45, *Shutdown Decay Heat Removal Requirements*, and generic safety issues (GSIs) GSI-57, *Effects of Fire Protection System Actuation on Safety-Related Equipment*, GSI-103, *Design for Probable Maximum Precipitation*, and the Sandia Fire Risk Scoping Study (FRSS) issues which were requested in Supplement 4 to GL 88-20 and its associated guidance in NUREG-1407. However, RMIEP does contain some information related to USI A-45, GSI-103, and FRSS. The staff considers the information submitted by the licensee to be sufficient to resolve USI A-45. While GSI-103 was not explicitly discussed, the NRC's contractor, SNL, reviewed meteorological data from nearby areas. This review indicates that the LaSalle design basis maximum precipitation would adequately bound the conditions of concern in GSI-103, except those related to site drainage. Potential flooding from the Illinois River was not discussed. The submittal provided inadequate information related to GSI-57 (related to the issue of spurious or inadvertent actuation of the fire protection system) and to four aspects of the FRSS (the issues of (1) seismic-fire interactions, (2) adequacy of fire

barriers, (3) smoke control and manual fire-fighting effectiveness, and (4) the equipment survival in a fire-induced environment). The submittal contained adequate information for the staff to consider the FRSS issue on fire-induced alternate shutdown/control room panel interaction resolved. The need for any additional assessment or actions related to the resolution of GSI-57, GSI-103, and the FRSS for LaSalle will be addressed by NRC staff separately from the IPEEE program.

The RMIEP and PRUEP analyses were the state-of-the-art PRA technology in 1985 and included some information on issues identified as USIs or GSIs. Due to the licensee's extensive participation in these studies, the licensee (1) developed a greater appreciation for potential severe accident behavior, (2) developed a better understanding of the most likely severe accident sequences, and (3) gained a qualitative and quantitative understanding of the overall likelihood of core damage and fission product releases. Since the submittal did not identify any plant improvements, no changes were necessary to reduce the overall likelihood of core damage and fission product releases. The submittal's lack of adequate information to resolve all aspects of the USIs and GSIs was reviewed by the SRB. The SRB concluded that this lack of adequate information represented, from the IPEEE's perspective, a minor weakness in the submittal related to finding plant vulnerabilities.

The effects of shared components between Units 1 and 2 have not been addressed. (Two components have been identified as being shared, the main control room and a swing diesel generator.) Other IPEEEs that have addressed the effects of shared components have not identified the shared component as representing a dominant risk contributor. In addition, the RMIEP study was only for Unit 2. The licensee did not discuss the applicability of the RMIEP for Unit 1, except to say that the risk profiles would be similar for all BWR5s with Mark II containments. This is considered a weakness in the LaSalle IPEEE.

Although the licensee did not address the effects of plant modifications made since 1985, they stated that a "top level review" indicates that any changes would not have "a significant, adverse impact" on the plant risk profile. The licensee has made numerous references in the submittal to a "future update of the LaSalle PRA analysis." The submittal stated that this updated analysis would have included the plant modifications. Furthermore, the licensee stated that it expects an updated LaSalle PRA to reveal a different set of dominant sequences than those identified by the RMIEP study, albeit at a lower overall total plant risk level. No updated PRA was provided to the staff as part of the IPEEE review. This weakness in the LaSalle IPEEE limits its usefulness for risk-informed activities or assessments.

While the licensee's IPEEE submittal contains no explicit discussion of generic issues, the RMIEP documentation does provide some information that addresses the external event aspects of certain generic safety issues (GSI-147, *Fire-Induced Alternate Shutdown/Control Room Panel Interactions* and GSI-172, *Multiple System Responses Program* (MSRP)). No information was provided to address the external events aspects of GSI-148, *Smoke Control and Manual Fire-Fighting Effectiveness.* The specific information associated with each of these issues is identified and discussed in the enclosed SER. Based on the review of the information contained in the submittal, the staff considers that the licensee's process is capable of identifying potential vulnerabilities associated with GSI-147 and part of the issues in GSI-172. However, GSI-148 and seven aspects of GSI-172 (related to (1) seismically induced fire

suppression system actuations, (2) effects of fire protection system actuation on non-safety related and safety-related equipment, (3) effects of moisture intrusion on non-safety related and safety-related equipment (the portion of this issue related to flooding has been adequately addressed), (4) non-safety-related control system/safety-related system dependencies, (5) seismically-induced fires, (6) seismically induced flooding, and (7) effects of hydrogen line ruptures) were not addressed in the IPEEE submittal. The need for any additional assessment or actions related to the resolution of these individual issues for LaSalle will be addressed by NRC staff separately from the IPEEE program.

On the basis of the review of RMIEP for IPEEE issues, the staff concludes that the licensee's IPEEE process is capable of identifying the most likely severe accidents and severe accident vulnerabilities. The RMIEP and PRUEP studies extensively involved LaSalle personnel. Thus, the licensee has gained a greater understanding of the LaSalle plant than would otherwise have been attained without the PRA. Based on this involvement, the licensee would have (1) developed a greater appreciation of severe accident behavior, (2) a better understanding of the most likely severe accident sequences that could occur at LaSalle, and (3) gained a qualitative understanding of the overall likelihood of core damage and fission product releases. Thus, the three major objectives of the IPEEE program (Supplement 4 to GL 88-20) have been satisfied, even though not all of the external event aspects of the USIs and GSIs have been resolved. Therefore, the staff considers that the intent of Supplement 4 to GL 88-20 has been satisfied for LaSalle.

If you have any comments regarding the enclosed evaluation, please contact me at (301) 415-1322.

Sincerely,

Donna M. Skay

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Docket Nos. 50-373 and 50-374

Enclosure: Safety Evaluation Report

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STAFF EVALUATION REPORT OF INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS (IPEEE) SUBMITTAL ON LASALLE COUNTY POWER STATION, UNITS 1 AND 2

1. INTRODUCTION

On June 28, 1991, the Nuclear Regulatory Commission (NRC) issued Generic Letter (GL) 88-20, Supplement 4 (with NUREG-1407, *Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities*). This GL requested all licensees to (1) perform individual plant examinations of external events to identify plant-specific vulnerabilities to external events that could lead to severe accidents, and (2) report the results to the Commission together with any licensee-determined improvements and corrective actions. In a letter dated April 28, 1994, the licensee, Commonwealth Edison, submitted its response to NRC (referred to in this safety evaluation report (SER) as the submittal).

An IPEEE Senior Review Board (SRB) was established to be independent of the IPEEE review process and to meet on a regular basis. The purposes of the SRB are (1) for the contractor to present the findings and conclusions of its review and the bases for its conclusions, and (2) for the SRB members to provide their perspectives on the contractor's findings and conclusions and to make recommendations based on their technical expertise. In this manner, the SRB provides additional assurance that (1) the scope of the review meets the objectives of the program, and (2) critical issues that have the potential to mask vulnerabilities are not overlooked.

GL 88-20 recognized that many licensees had plant-specific probabilistic risk assessment (PRAs) or similar analyses. The use of existing PRA analyses was encouraged in achieving the objectives of the Individual Plant Examination (IPE) program. LaSalle made use of this provision. As a result, the LaSalle submittal was brief and referred to the information in NUREG/CR-4832, *Analysis of the LaSalle Unit 2 Nuclear Power Plant: Risk Methods Integration and Evaluation Program (RMIEP)*, and NUREG/CR-5305, *Integrated Risk Assessment for the LaSalle Unit 2 Nuclear Power Plant: Phenomenology and Risk Uncertainty Evaluation Program (PRUEP)* which was performed by Sandia National Laboratories (SNL) and sponsored by the USNRC. The information in these two reports is sufficient to identify potential plant vulnerabilities to severe accidents from external events, but it is incomplete in other areas (i.e., the resolution of related generic safety issues).

The staff contracted with SNL to conduct screening reviews of the seismic; fire; and high winds, floods, and other (HFO) external events aspects of the licensee's IPEEE submittal. The LaSalle IPEEE submittal is quite brief, presents only a limited discussion, and exclusively references RMIEP and PRUEP as the basis for their IPEEE. The RMIEP and PRUEP studies were performed by Sandia under contract to NRC and in cooperation with Commonwealth

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Edison. Because the staff's SRB reviewed the Sandia review, it was judged to be acceptable for Sandia to review the LaSalle IPEEE submittal. The staff and contractor review relies extensively, then, on the results of the RMIEP and PRUEP analyses published in 1985. No requests for additional information were sent to the licensee. Based on the results of the review of the RMIEP and PRUEP reports, the staff concluded that aspects of seismic events; fires; and high winds, floods, transportation, and other external events were addressed. The review findings are summarized in the evaluation section below. Details of the contractor's findings are presented in the two technical evaluation reports attached to this staff evaluation report.

Supplement 4 to GL 88-20, and its associated guidance in NUREG-1407, explicitly requested licensee to provide information to address the resolution of Unresolved Safety Issue (USI) A-45, *Shutdown Decay Heat Removal Requirements*, Generic Safety Issues (GSIs)-57, *Effects of Fire Protection System Actuation on Safety-Related Equipment*, GSI-103, *Design for Probable Maximum Precipitation*, and Sandia Fire Risk Scoping Study (FRSS) issues. The licensee submittal does not discuss any unresolved safety or generic issues, but only points to the RMIEP and PRUEP documentation. Since the RMIEP is based on the plant as of 1985 (i.e., before Supplement 4 to the GL 88-20 was issued), it does not explicitly discuss these issues. However, as discussed in Section II of this SER, sufficient relevant information is provided to resolve some of these issues.

2. EVALUATION

LaSalle is a two unit site. Each unit is a 1078 MW_e General Electric Company boiling water reactor (BWR) with a Mark II (inerted) containment. NUREG-1407 categorized LaSalle as a 0.3 g focused scope plant for the IPEEE seismic evaluation.

The licensee has relied exclusively upon the RMIEP and PRUEP analyses for their IPEEE. Part of this program included soil structure interactions and relay chatter. The RMIEP is a full scope Level I PRA. The complementary PRUEP provides the Level II and Level III containment integrity and offsite release analyses, respectively. The RMIEP and PRUEP, although not in conformance with all aspects of the NUREG-1407 guidance, represent the state-of-the-art (circa 1985) in probabilistic analysis. The high winds, transportation, and other external events (HFO) areas were either screened out or computed to have "bounding" core damage frequency (CDF) values less than $1x10^{-6}$/ reactor-year (ry), in conformance with the guidance in NUREG-1407. While GSI-103 was not discussed, RES's contractor, SNL, reviewed meteorological data from nearby areas. This review indicates that the LaSalle design basis maximum precipitation would adequately bound the conditions of concern in GSI-103, except those related to site drainage. Potential flooding from the Illinois River was not discussed.

The RMIEP and PRUEP studies were only for Unit 2. The licensee has provided no discussion as to the applicability of the RMIEP for Unit 1, except to say that the risk profiles would be similar for all BWR5s with Mark II containments. The licensee did not address the potential effects of shared components, e.g., the main control room and the swing diesel generator. This is considered a weakness in the LaSalle IPEEE.

The licensee has made numerous references in the submittal to a "future update of the LaSalle PRA analysis." The submittal stated that this updated analysis would have included the plant modifications since 1985. Furthermore, the licensee states on page 16 of the submittal that

they expect an updated LaSalle PRA to reveal a different set of dominant sequences than those identified by the RMIEP study. No updated PRA was provided to the staff as part of the IPEEE review. This weakness in the LaSalle IPEEE limits its usefulness for risk-informed activities or assessments.

Core Damage Frequency Estimates

RMIEP internal event trees were used to model seismic events. The total seismic CDF was estimated to be $6x10^{-7}$/ry for LaSalle Unit 2. The licensee (RMIEP) estimated the mean fire CDF to be $3.2x10^{-5}$/ry. Potential flooding from the Illinois River was not discussed. The licensee did not estimate a CDF for other external events (i.e., tornadoes, high winds, and others). These were screened out or computed to have "bounding" CDF values less than 10^{-6}/ry. The licensee estimated previously a CDF due to internal events of about $4.7x10^{-5}$/ry, including internal flooding.

Dominant Contributors

The seismic CDF was estimated to be $6x10^{-7}$/ry. The three most dominant seismic accident sequences are the loss of offsite power transient (including station blackout) sequences. The station blackout sequence (TLOSP-01) accounts for 73 percent of the total seismic CDF. The other two loss of offsite power sequences represent 17 percent of the total seismic CDF. The remaining sequences, including loss of offsite power in conjunction with transient-induced loss-of-coolant accident (LOCA), contribute less than 10 percent of the total seismic CDF.

The total fire CDF was estimated to be $3.21x10^{-5}$/ry. The dominant fire areas are the control room (43 percent), the Division 2 switchgear room (27 percent), the Division 1 switchgear room (16 percent), and the Auxiliary Equipment Room (8 percent). The cable spreading room was estimated to contribute $1.6x10^{-7}$/ry (0.5 percent) to the total fire CDF.

LaSalle Units 1 and 2 were licensed for commercial operation in 1982 and 1984, respectively. Therefore, LaSalle was licensed using the 1975 Standard Review Plan (SRP) criteria. The licensee identified 38 HFO events in RMIEP and considered eight for further evaluation. Of these eight, three (military and industrial facilities accidents, pipeline accidents, and release of chemicals in onsite storage) were dismissed by virtue of having been "presented in the LaSalle FSAR" (Final Safety Analysis Report). The submittal stated the mean CDF for aircraft impacts to be $5x10^{-7}$/ry, for turbine-generated missiles to be $1x10^{-7}$/ry, and for tornadoes to be $3x10^{-7}$/ry. RMIEP performed bounding analyses for those remaining HFO events (winds, transportation accidents, and external flooding). External flooding was dismissed based on "the probable maximum precipitation at the site is indeed very low" (Attachment A, TER, Section 1.3.2, page 8). RES's contractor, SNL, reviewed meteorological data from nearby areas. This review indicates that the LaSalle design basis maximum precipitation would adequately bound the conditions of concern in GSI-103, except those related to site drainage. Potential flooding from the Illinois River was not discussed. Therefore, all of the HFO events were judged by the licensee to not be a significant contributor to the total CDF. NUREG-1407 states that if the plant conforms to the 1975 SRP criteria and a walkdown reveals no potential vulnerabilities not included in the original design basis, it is judged that the contribution from that hazard to CDF is less than $1x10^{-6}$/ry and the IPEEE screening criterion is met. Although the licensee performed no walkdown, the plant was licensed to the 1975 SRP criteria. Based on the plant being

licensed to the 1975 SRP criteria, the staff concludes that LaSalle as related to HFO external events is consistent with the intent of Supplement 4 to GL 88-20.

The licensee's IPEEE assessment appears to have examined the significant initiating events and dominant accident sequences.

Containment Performance

The containment structure for LaSalle Unit 2 was analyzed for seismic failure modes. No unique containment failure modes were identified. Relay chatter induced interfacing system LOCA was not explicitly considered. However, relays were explicitly included in the system fault trees and assigned a generic fragility factor. Eight of the 16 cutset basic events which survived the probabilistic screening process were either relay or circuit breaker failure events. Since the failure of these components in eight of the cutsets did not result in unacceptable consequences, i.e., relay failure was acceptable, no plant vulnerability was identified as the result of relay chatter.

The licensee included the containment structures in their initial list of fires areas. There is no further mention of the containment structures. Since the containment is inerted during normal operation, no fire could be sustained inside containment. Thus, no fire-related containment failure modes different from those identified in the internal events analysis were found.

The licensee's containment performance analyses for seismic and internal fire events appeared to have considered important containment performance issues and are consistent with the intent of Supplement 4 to GL 88-20.

Generic Safety Issues

As a part of the IPEEE, a set of unresolved and generic safety issues (USI A-45, GSI-131, GSI-103, GSI-57, and the Sandia FRSS issues) were identified in Supplement 4 to GL 88-20 and its associated guidance in NUREG-1407 as needing to be addressed in the IPEEE. These safety issues were evaluated by the NRC contractor. The results of these evaluations are provided in the attached TER. For those safety issues that were not completely resolved by the contractor, the NRC staff performed additional review in order to arrive at a conclusion. It should be noted that the submittal relied exclusively on the RMIEP and PRUEP documents. Although the licensee's IPEEE submittal does not explicitly address any issues as identified in GL 88-20, Supplement 4, or NUREG-1407, information related to some of these issues was presented in the RMIEP/PRUEP studies. The staff's evaluation of these issues is provided below.

1. USI A-45, *Shutdown Decay Heat Removal (DHR) Requirements*

   The licensee referenced the RMIEP and PRUEP analyses which was a full scope PRA, including Level 2 and Level 3 analyses. As such, consideration was given to potential loss of DHR systems. Components required for long-term decay heat removal were included in the seismic fault trees and event trees. No unique seismic, fire, or HFO vulnerabilities were identified.

The staff finds that the licensee's evaluation is consistent with the guidance provided in Section 6.3.3.1 of NUREG-1407 for resolution of USI A-45, and, therefore, the staff considers this issue resolved.

2. GSI-131, *Potential Seismic Interactions Involving the Movable In-Core Flux Mapping System Used in Westinghouse Plants*

   This is not applicable to LaSalle Units 1 and 2 because LaSalle is a BWR plant.

3. GSI-103, *Design for Probable Maximum Precipitation*

   There is no discussion in the submittal or RMIEP of the updated probable maximum precipitation (PMP), roof ponding issues, or GSI-103. The plant's licensing basis is the 1975 SRP. The guidance in NUREG-1407 states that plants built to the 1975 SRP can screen out external floods as a significant hazard. However, the FSAR identified the design PMP of 32 inches of rainfall in 24 hours as occurring for 48 hours with all drains clogged. RES's contractor, SNL, reviewed data collected from the site, nearby Chicago, and the entire northern half of Illinois which indicates that the PMP has an annual occurrence probability of less than 10^{-8}/ry (95 percent confidence level). Thus, even though neither the submittal nor the RMIEP report discussed enhanced local precipitation rates and roof ponding related to GSI-103, the RES contractor's review of local historical data showed that the risk due to local intense precipitation is negligible. However, no discussion is provided on the plant's ability to withstand external flooding as related to site drainage considerations, i.e., potential flooding from the Illinois River was not discussed. The staff finds that the potential effects of local intense precipitation is consistent with the guidance provided in Section 6.2.2.3 of NUREG-1407, and therefore, the staff considers this aspect of the generic issue resolved. The staff considers the aspect of GSI-103 related to site drainage as unresolved.

4. GSI-57, *Effects of Fire Protection System Actuation on Safety-Related Equipment*

   The licensee did not provide adequate information related to GSI-57 to resolve this issue. A review of RMIEP identified that there are not many areas where there is an automatic fire suppression system. Further, the fire suppression system is confined to the immediate vicinity of the area. The licensee did not provide a definition of an "area," e.g., a room completely surrounded by a three-hour fire barrier. Reviewing the RMIEP report indicates that an "area" is clearly smaller than one room completely surrounded by a three-hour fire barrier as the submittal stated that there are two areas in "auxiliary equipment room." Potential flooding from activation of the fire suppression system was addressed in the internal flooding analysis. However, it is not clear whether the spurious or inadvertent actuation of the fire protection system could adversely affect safety-related components. In particular, the licensee has not provided information to verify that redundant safety-related components would not be adversely affected by the activation of the fire suppression system. The staff considers that licensee has not adequately addressed the issue of spurious or inadvertent actuation of the fire protection system, and therefore, the staff considers this issue unresolved.

5. Fire Risk Scoping Study Issues

The licensee has provided information related to the fire risk scoping study issues (seismic-fire interactions (Page 1-2 of RMIEP); adequacy of fire barriers (Page 3-65 of RMIEP); smoke control and manual fire-fighting effectiveness (Page A-6, Appendix A of RMIEP); equipment survival in a fire-induced environment (Section 1.1 of RMIEP); and fire-induced alternate shutdown/control room panel interaction (Pages 3-100 and 3-101 of RMIEP)). However, only the last issue was satisfactorily addressed, the others are not resolved. For the seismic-fire interactions, the licensee stated that this is "thought to be probabilistically unimportant." As related to the adequacy of fire barriers, RMIEP noted that two areas had "protective wrappings" for some cables. In one case, Thermo-lag was cited by name. However, there is no discussion of how these barriers were treated in the analysis. RMIEP did perform a multi-area screening analysis with a conditional barrier failure probability of 0.1. However, there is no information as to whether this includes normally open doors, water curtain failures, and ventilation damper failures. There is no discussion of the potential for the spread of smoke and heat through opening of access doors. Concerning smoke control and manual fire fighting effectiveness, the RMIEP report said "there is an extremely low probability that, upon arriving at the scene, the Fire Brigade will be unable to locate the fire due to smoke obstruction" and "personnel were not required to suit-up with SCBA (Self-Containment Breathing Apparatus)." As related to the potential for equipment to survive in a fire-induced environment, the RMIEP report did discuss potential flooding concerns and did state that "there are not many areas ... with automatic fire suppression systems." The licensee did not discuss either the potential failure of safety-related equipment nor the potential for failing safety-related equipment of two or more redundant divisions or trains as a result of a fire-induced environment. Based on the results of the IPEEE submittal review, the staff considers that the licensee's process is consistent with the guidance provided in NUREG-1407 for evaluation of fire-induced alternate shutdown/control room panel interactions, and therefore, the staff considers this one FRSS issue resolved. The staff considers that the licensee has not adequately addressed the issues of (1) seismic-fire interactions, (2) adequacy of fire barriers, (3) smoke control and manual fire-fighting effectiveness, and (4) equipment survival in a fire-induced environment. The need for any additional assessment or actions related to the resolution of these four FRSS issues will be addressed by NRC staff separately from the IPEEE program. The staff considers the four issues discussed above to be unresolved.

In addition to those safety issues discussed above that were explicitly requested in GL 88-20, Supplement 4, four generic safety issues were not specifically identified as issues to be resolved under the IPEEE program; thus, they were not explicitly discussed in Supplement 4 to GL 88-20 or NUREG-1407. However, subsequent to the issuance of the GL, the NRC evaluated the scope and the specific information requested in the GL and the associated IPEEE guidance, and concluded that the plant-specific analyses being requested in the IPEEE program could also be used, through a satisfactory IPEEE submittal review, to resolve the external event aspects of these four safety issues. For those generic issues that were not completely resolved by the contractor, the NRC staff performed additional review in order to arrive at a conclusion. The LaSalle IPEEE submittal did not explicitly address any of these issues. However, a review of the RMIEP documentation did provide some information. The staff's evaluation of these issues is provided below.

1. GSI-147, *Fire-Induced Alternate Shutdown/Control Room Panel Interactions*

   A review of the RMIEP documentation indicates that RMIEP has examined fire-induced alternate shutdown/control room interactions for LaSalle (Pages 3-100 and 3-101 of RMIEP). The licensee has reviewed their ability to safely shutdown the plant with a fire in the control room. The licensee concluded that they can safely shutdown the plant from outside of the control room with any fire in any sub-panel in the control room. The licensee has also evaluated the probability of core damage from a fire in the control room to be approximately 1.3×10^{-5}/ry. On the basis that no vulnerability associated with this issue was identified in the IPEEE submittal, the staff considers this issue resolved.

2. GSI-148, *Smoke Control and Manual Fire-Fighting Effectiveness*

   The IPEEE submittal provided no discussion related to smoke control and manual fire-fighting effectiveness. The RMIEP claimed that there is no need for SCBA for manual fire fighting because "the large room areas or openness to other areas preclude buildup of sufficient smoke to obscure the fire location." There was no discussion of the fire brigade response time, knowledge of fire fighting equipment and its proper use, or knowledge of pre-fire plans. Further, there is no discussion on (1) the potential negative effects of smoke on operators during the performance of safe shutdown activities, or (2) the potential for misdirected manual fire suppression efforts that could damage equipment not directly involved in the fire and negatively impact the ability to safely shutdown the plant.

   Based on the results of the IPEEE submittal review, the staff considers that the licensee did not provide information to resolve this issue. The need for any additional assessment or actions related to the resolution of GSI-148 will be addressed by NRC staff separately from the IPEEE program. The staff considers this issue unresolved.

3. GSI-156, *Systematic Evaluation Program (SEP)*

   LaSalle, Units 1 and 2, began commercial operation in December 1982 and October 1984, respectively. The plant was licensed under the 1975 SRP. Therefore, GSI-156, *SEP*, is not applicable to LaSalle.

4. GSI-172, *Multiple System Responses Program (MSRP)*

   A review of RMIEP provided information to address the following external events-related MSRP issues: (1) seismic induced relay chatter (Pages 205 and 243 of RMIEP, Volume 8); (2) effects of flooding on non-safety related and safety-related equipment (Volume 10 of RMIEP) (the aspect of this issue related moisture intrusion was not addressed); (3) seismically induced spatial and functional interactions (Pages 197-202 of RMIEP, Volume 8); (4) the IPEEE-related aspects of common cause failures related to human errors (Page 246 of RMIEP Volume 8) and (5) evaluation of earthquake magnitudes greater than the safe shutdown earthquake. The last issue (item 5) is satisfactorily resolved by the licensee's reference to the plant specific PRA (RMIEP) having been performed and no potential vulnerability (or unusually low plant seismic capacity) associated with the seismic

concerns were identified in the IPEEE submittal. Based on the results of the IPEEE submittal review, the staff considers that the licensee's process is consistent with the guidance provided in NUREG-1407, and therefore, the staff considers these issues resolved.

Seismically induced fire suppression system actuations is discussed on pages 1-2 of RMIEP report, Volume 9, but only to say that during the licensee's initial planning of the project the licensee thought this issue "to be probabilistically unimportant." As related to the potential effects of the fire protection system actuation on safety-related equipment, the licensee did discuss potential flooding concerns and did state that "there are not many areas ... with automatic fire suppression systems." However, the licensee did not discuss either the potential failure of safety-related equipment nor the potential for failing safety-related equipment of two or more redundant divisions or trains. While the effects of flooding on non-safety related and safety-related equipment have been addressed in part by consideration of the failure of piping and tanks, the effects of moisture intrusion on non-safety-related and safety-related equipment from damage to the fire suppression system have not been addressed. The licensee considered a plant walkdown (for the RMIEP assessment) for potential seismic and HFO vulnerabilities to be sufficient to resolve non-safety-related control system/safety-related system dependencies (RMIEP, Volume 8, Pages 197-202). While this may be capable of identifying some seismic II/I issues, it is not likely that such a walkdown could identify potential unrecognized dependencies between control and protection systems. This issue not considered to be adequately resolved. The licensee stated that the issue of seismically-induced fires was "out of the scope of the RMIEP" and the licensee stated that this is "thought to be probabilistically unimportant." This issue is not considered to be adequately resolved. The submittal did not discuss any plant walkdowns to address seismically induced flooding. The effects of hydrogen line ruptures is not explicitly discussed in the IPEEE submittal. In summary, the following issues are not adequately resolved:

1. seismically induced fire suppression system actuations,
2. effects of fire protection system actuation on non-safety related and safety-related equipment,
3. effects of moisture intrusion on non-safety related and safety-related equipment (there was adequate information to address the aspect of this issued related flooding),
4. non-safety-related control system/safety-related system dependencies,
5. seismically-induced fires,
6. seismically induced flooding, and
7. effects of hydrogen line ruptures.

The need for any additional assessment or actions related to the resolution of these issues will be addressed by NRC staff separately from the IPEEE program.

Plant Safety Features, Potential Vulnerabilities, and Improvements

The licensee did not identify any unique plant safety features in the IPEEE submittal or in RMIEP. The licensee did not provide a definition of vulnerability.

No plant improvements were cited in the IPEEE submittal. However, the RMIEP (1985) had noted "several technical concerns" in the seismic area which were to be "provided to the station for disposition." The IPEEE submittal did not discuss what these concerns were or what action was taken on them. The fire analysis identified three potential areas for plant improvement: (1) to put tops on the control room electrical panels to reduce the potential for spread of fire to the overhead cables, (2) to institute a program to inspect the penetration seals at the top of the switchgear panels to minimize the potential that switchgear fires might damage overhead cables and (3) "insights dealing with accident management." No discussion of these specific recommendations is provided in the IPEEE submittal.

The licensee stated that plant changes have been made since the 1985 RMIEP analyses, but none of these unidentified changes have been accounted for in the IPEEE submittal. The licensee has made numerous references in the submittal to a "future update of the LaSalle PRA analysis." The submittal stated that this updated analysis would have included the plant modifications. Furthermore, the licensee states on page 16 of the submittal that they expect an updated LaSalle PRA to reveal a different set of dominant sequences than those identified by the RMIEP study.

3. CONCLUSION

On the basis of the above findings, the staff notes that: (1) the licensee's IPEEE (i.e., RMIEP and PRUEP) is complete with regard to the information requested by Supplement 4 to GL 88-20 (and associated guidance in NUREG-1407) except in the area of the GSIs, and (2) the IPEEE (i.e., RMIEP and PRUEP) results are reasonable given the LaSalle Units 1 and 2 design, operation, and history. Therefore, the staff concludes that the licensee's IPEEE (i.e., RMIEP and PRUEP) process is capable of identifying the most likely severe accidents and severe accident vulnerabilities from external events and therefore, the LaSalle IPEEE (i.e., RMIEP and PRUEP) has met the intent of Supplement 4 to GL 88-20. However, the LaSalle IPEEE (i.e., RMIEP and PRUEP) has not resolved all of the specific generic safety issues as discussed in this SER.

The licensee has made numerous references in the submittal to a "future update of the LaSalle PRA analysis." The submittal stated that this updated analysis would have included the plant modifications. Furthermore, the licensee states on page 16 of the submittal that they expect an updated LaSalle PRA to reveal a different set of dominant sequences than those identified by the RMIEP study. This weakness in the LaSalle IPEEE limits its usefulness for risk-informed activities and assessments.

The effects of shared components between Units 1 and 2 have not been addressed. (Two components have been identified as being shared, the main control room and a swing diesel generator.) Although the licensee did not address the effects of plant modifications made since 1985, they stated that a "top level review" indicates that the change would not have "a significant, adverse impact" on the plant risk profile. Furthermore, the RMIEP study was only for Unit 2. The licensee has provided no insight as to the applicability of the RMIEP for Unit 1, except to say that the risk profiles would be similar for all BWR5s with Mark II containments. This is considered a weakness in the LaSalle IPEEE.

As indicated in Section 2 of this SER, the licensee did not provide sufficient information to resolve the following generic safety issues: one aspect of GSI-57 (related to the issue of spurious or inadvertent actuation of the fire protection system), one aspect of GSI-103 (related to the issue of external flooding and site drainage), four aspects of the FRSS (the issues of (1) seismic-fire interactions, (2) adequacy of fire barriers, (3) smoke control and manual fire-fighting effectiveness, and (4) the effects of fire protection system actuation on safety-related equipment), GSI-148, and seven aspects of GSI-172 (related to (1) seismically induced fire suppression system actuations (2) effects of fire protection system actuation on non-safety related and safety-related equipment, (3) effects of moisture intrusion on non-safety related and safety-related equipment, (4) non-safety-related control system/safety-related system dependencies, (5) seismic/fire interactions, (6) seismically induced flooding, and (7) effects of hydrogen line ruptures) that the licensee did not address in the IPEEE submittal. The need for any additional assessment or actions related to the resolution of these issues for LaSalle will be addressed by NRC staff separately from the IPEEE program.

The RMIEP and PRUEP analyses were the state-of-the-art PRA technology in 1985 and included some information on issues identified as USIs or GSIs. With the licensee's extensive participation in these studies, the licensee (1) developed a greater appreciation for potential severe accident behavior, (2) developed a better understanding of the most likely severe accident sequences, and (3) gained a qualitative and quantitative understanding of the overall likelihood of core damage and fission product releases. Since the submittal did not identify any plant improvements, no changes were necessary to reduce the overall likelihood of core damage and fission product releases. These are the major objectives for the IPEEE program identified in NUREG-1407. The submittal's lack of adequate information to resolve all aspects of the USIs and GSIs was reviewed by the SRB. The SRB concluded that this lack of adequate information represented, from the IPEEE's perspective, a minor weakness in the submittal related to finding plant vulnerabilities. Therefore, the staff can conclude that the licensee has met the intent of GL 88-20, Supplement 4, even though not all of the external event aspects of the USIs and GSIs have been resolved.

It should be noted that the staff focused its review primarily on the licensee's ability to examine LaSalle for severe accident vulnerabilities from external events. Although certain aspects of the IPEEE were explored in more detail than others, the review was not intended to validate the accuracy of the licensee's detailed findings (or quantification estimates) that underlie or stemmed from the examination. Therefore, this SER does not constitute NRC approval or endorsement of any IPEEE material for purposes other than those associated with meeting the intent of Supplement 4 to GL 88-20 and the resolution of specific generic safety issues discussed in this SER.

Attachment: TER related to seismic and high winds, floods, and other external events

Principal Contributor: T. King

Date: December 8, 2000

Attachment A

LASALLE COUNTY POWER STATION INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS (IPEEE) SANDIA NATIONAL LABORATORIES TECHNICAL EVALUATION REPORT RELATED TO SEISMIC AND HIGH WINDS, FLOODS, AND OTHER EXTERNAL EVENTS

Review of the Submittal in Response to U.S. NRC Generic Letter 88-20, Supplement 4: "Individual Plant Examination - External Events"

Seismic and HFO Submittal Screening Review Technical Evaluation Report: LaSalle County Power Station June 12, 2000

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1.0 INTRODUCTION

1.1 Objectives

The performance of an Individual Plant Examination of External Events (IPEEE) was requested of all commercial U.S. nuclear power plants by the U.S. Nuclear Regulatory Commission (USNRC) in Supplement 4 of Generic Letter 88-20 [1]. Additional guidance on the intent and scope of the IPEEE process was provided in NUREG-1407 [2]. The USNRC has established a screening review process to help the USNRC staff determine if the individual licensee submittals have met the intent of the generic letter and the extent to which the assessment addresses certain other specific issues and ongoing programs. This document presents the results of a "Step 0" review of the LaSalle County Nuclear Station (LaSalle) IPEEE seismic and high winds, floods and other (HFO) external events assessments.

LaSalle submitted a combined IPE/IPEEE submittal to the USNRC in April 1994 [3]. The external events portions of that document were previously reviewed by SNL in February 1996 [4]. At that time, the IPEEE review process had not yet been fully established, and no specific review guidance was available. Since that time IPEEE review guidance has been developed [5] and the scope of the reviews has been expanded to include consideration of a specific set of special issues identified in the Generic Safety Issues and Unresolved Safety Issues programs. The objective of this report is to re-consider the LaSalle IPEEE seismic and HFO events assessments within the specific context of the current IPEEE "Step 0" review process.

1.2 Plant Description

LaSalle is a two unit site with Units 1 and 2 sharing a common control room. Each unit is a General Electric (GE) boiling water reactor (BWR) 5 with a design output rating of 3293 MWt and 1078 MWe. The reactor itself is housed within a Mark II containment structure. The plant is owned and operated by Commonwealth Edison Company (CECo) and is located in LaSalle County, Illinois, about 55 miles southwest of Chicago. Construction began in 1973 for Unit 1 and in 1972 for Unit 2. Commercial operation was initiated in 12/82 and 10/84 respectively. The IPEEE study is based on the analysis of Unit 2.

The LaSalle site is a soil site, with approximately 170 ft of alluvial deposits overlying bedrock. The design basis SSE is 0.20g. The original design spectra were specified at the foundation elevation (44 feet below ground level). The plant was not a SEP plant. For the IPEEE, the plant was put in the Focussed Scope category with a target HCLPF of 0.3g.

The primary containment is a post-tensioned reinforced concrete structure with a steel liner. The containment consists of a lower cylindrical portion founded on a base mat and an upper portion that is in the form of the frustum of a cone. The containment is topped by an elliptical steel

dome called the drywell head. The lower portion is called the suppression chamber (or wetwell) and it contains the suppression pool; the upper portion is called the drywell and it houses the reactor pressure vessel. The primary containment is enclosed by a reinforced concrete reactor building which forms the secondary containment. During power operation, the primary containment is inerted with nitrogen. The internal design pressure of the primary containment is 45 psig. The nominal free volumes of the drywell and the suppression chamber are 219,800 ft^3 and 165,100 ft^3, respectively. The nominal volume of the suppression pool is 128,800 ft^3.

Important design characteristics include the following features. High-pressure injection is provided by the following four systems:

- High-pressure core spray (HPCS): HPCS consists of a motor-driven pump with its own dedicated diesel. It draws water from either the condensate storage tank or the suppression pool and sprays coolant into the core.

- Reactor core isolation cooling (RCIC): RCIC consists of a turbine-driven pump which pumps water from the condensate storage tank or the suppression pool to the reactor vessel. The turbine uses steam from the reactor pressure vessel; thus, system operation cannot be ensured after the reactor vessel pressure decreases below a specified point—57 psig. In addition, RCIC isolates when the containment pressure reaches about 15 psig.

- Main feedwater (MFW): MFW takes suction from the condenser hotwell using two turbine-driven pumps and one motor-driven pump and injects the water into the vessel through the main feedwater lines. MFW can also take suction from the condensate storage tank; however flow is limited to a maximum of 1200 gpm. All pumps require offsite power to operate.

- Control rod drive (CRD): CRD can be used to inject several hundred gallons of water per minute into the reactor vessel and is only useful once the decay heat load has decreased, as in a long-term accident, or in conjunction with another system.

To use the low-pressure injection systems, reactor vessel pressure must be reduced. This can be accomplished by the use of the automatic depressurization system (ADS), manual operation of the safety relief valves (SRVs), or by a break in the system that allows system pressure to be reduced below the shutoff head of the low-pressure systems. Low-pressure injection is provided by the following four systems:

- Condensate system (CDS): CDS takes water from the condenser hotwell and pumps it through the feedwater line into the reactor pressure vessel using four motor-driven pumps. CDS can also take suction from the condensate storage tank; however, flow is limited to a maximum of 1200 gpm. CDS requires offsite power to operate.

- Low-pressure core spray (LPCS): LPCS is a single-train system that takes water from the suppression pool and injects it into the reactor pressure vessel via a motor-driven pump. LPCS is powered by train A of emergency power.

- Low pressure coolant injection (LPCI): LPCI is a three-train system that takes water from the suppression pool and injects it into the reactor pressure vessel using three motor-driven pumps. Train A is powered by train A of emergency power, and trains B and C are powered by train B of emergency power.

- Diesel-driven firewater system (DFWS): The DFWS must be manually connected to the MFW injection line before injection into the vessel can occur. Diesel-driven pumps are then used to inject water. These diesel-driven pumps make operation during station blackout possible.

Heat can be removed from the containment by the residual heat removal (RHR) system. Three modes of operation are possible. The first two—suppression pool cooling (SPC) and containment spray (CS)—can be used during any type of accident. The last—shutdown cooling—can be used for non-LOCAs. In addition, the containment can also be vented by use of the containment vent and purge system. Venting can be from either the drywell or the suppression chamber using either 2-inch lines or 26-inch lines. The 26-inch vent lines tie into the standby gas treatment system (SGTS), which releases the gases to the stack. The 26-inch vent lines are attached via an 18-inch pipe to the SGTS with a rubber boot which is assumed to fail when high- pressure steam is released through the vent.

Directly below the reactor pressure vessel is the reactor pedestal cavity. The arrangement of the pedestal cavity has potentially important implications for severe accidents. The cavity is divided into two regions: the upper cavity and the lower cavity. The upper cavity communicates with the drywell airspace while the lower cavity communicates with the wetwell airspace. The upper cavity volume is large enough to hold all of the debris that would be released should the vessel fail and, hence, drywell melt-through scenarios that are a concern in Mark I containments are not a concern in the LaSalle plant. The drywell is provided with drains that direct water that accumulates in the drywell to the upper pedestal cavity sumps. There are additional lines that drain water from the sumps to systems outside the containment. While these sump drains include isolation valves, these valves are isolated outside the containment boundary. A potential containment isolation failure mechanism is failure of the drain line outside the containment from energetic events that occur during a severe accident. An additional concern is that energetic loads that can accompany vessel failure and thermal attack by molten core debris released from a failed vessel can potentially fail the floor that separates the upper cavity from the lower cavity. Failure of the floor establishes a pathway that connects the drywell and wetwell airspace and bypasses the suppression pool.

1.3 Submittal Results of the IPEEE Seismic and HFO Analyses

The estimated core damage frequencies (CDFs) due to seismic and HFO events as reported in the LaSalle IPEEE submittal are taken directly from a study performed in the Risk Methods Integration and Evaluation Program (RMIEP) [6]. RMIEP was a USNRC-sponsored full-scope Level I Probabilistic Risk Assessment (PRA). The RMIEP study was led by Sandia National Laboratories (SNL). The seismic portion of the LaSalle RMIEP study (on which the IPEEE was based) was performed by Lawrence Livermore National Laboratory (LLNL) and its subcontractor EQE, Inc.

1.3.1 IPEEE Seismic Analysis Methodology and Results

Under the Seismic Safety Margin Research Program (SSMRP) sponsored by the USNRC [7], Lawrence Livermore National Laboratory developed a simplified seismic PRA methodology which concentrated on reducing the effort required to determine seismic hazard and to calculate seismic responses for structures, systems, and components important to safety. This approach was applied in the LaSalle seismic IPEEE. The key elements of the LaSalle simplified seismic risk analysis include:

- Development of the seismic hazard at the LaSalle site including the effect of local site conditions.
- 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.
- Development of building and component fragilities (probabilistic failure criteria) for important structures and components.
- Investigation of the effects of hydrodynamic loads on seismic risk.
- Development of the system models (event and fault trees).
- Estimation of the seismically-induced core damage frequency.

The approach used in each of these steps is described below.

Seismic Hazard

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 was shown in Figure 7-1 of the LaSalle seismic submittal. Also shown in Figure 7-1 was the latest LLNL mean hazard curve for the LaSalle County Station. It could 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. (This will be discussed later.) 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: 0.18-0.27g, 0.27-0.36g, 0.36-0.46g, 0.46-0.58g, 0.58-0.73g, and >0.73g for quantification purposes.

Soil-Structure Interaction, Building, and Component Response Analyses

Seismic responses and component fragilities together allow for the calculation of seismically induced failure probabilities. The SMACS methodology of the SSMRP was used in the LaSalle response analysis. SMACS links together seismic input, soil-structure interaction (SSI), structure response, and piping system and component response to produce a probabilistic description of critical structural, piping and component responses. SMACS analyses were performed on the LaSalle structure complex including the effects of soil-structure interaction (SSI).

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 at some BWRs during an earthquake) were found not to be significant at LaSalle. The average hydrodynamic load (averaged over all responses) was 0.04 g.

Plant Logic Models

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

Initiating Event	Conditional Seismic Initiators (per eq)	Internal Events Initiators (per ry)
Reactor Vessel Rupture	1.0E-9	< 3E-7
Large LOCA	1.2E-4	1.0E-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 measure of only the seismic increment to total 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 study were modified to include seismic failure modes. Any event in the fault tree which could result from either random failure or 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 and fragility, and random failure data was generated to feed into the SSMRP-developed SEISIM code to yield the core damage frequency at each of the six levels of earthquake hazard.

Quantification and Results

After screening, a total of six seismic accident sequences survived for final quantification. Each of these was evaluated at the six discrete levels of the seismic hazard curve. The largest contributors are the TLOSP-01 and TLOSP-02 sequences which are loss of offsite power/station blackout sequences. The loss of offsite power is caused by seismically induced failure of

ceramic insulators in the switchyard. The loss of onsite AC power is due to random (common mode) failures of the emergency diesels to start. The remaining sequences (TL1-01, TL1-03, TL2-01, and TL3- 01) are loss of offsite power sequences in conjunction with failure (to re-close) of 1, 2, or 3 safety relief valves. Thus these are transient-induced LOCA sequences. They are, however, only minor contributors. The TLOSP-01 station blackout sequence alone accounts for 73% of the total seismic core damage frequency.

The total seismically induced CDF was estimated to be 6E-7/year for LaSalle Unit-2. The three most dominant accident sequences are the loss-of-offsite power transient sequences, together contributing roughly 90% of the seismic CDF.

The most significant observation (made in the submittal) 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, (i.e., 90% of the risk was due to the lower three earthquake hazard levels). The dominance of the seismic risk by the lower 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 0.3g 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.

1.3.2 IPEEE HFO Analyses Methodology and Results

As part of the HFO analyses, 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 considered were:

> 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, Snow, Soil shrink-swell or consolidation, Storm surge, Transportation accidents, Tsunami, Toxic gas, Turbine generated missiles, Volcanic activity, and Waves.

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 screened out based on information 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 5E7/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.

(As noted in the submittal, since the plant system failures and consequence analysis were conservatively neglected for these 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

All HFO events were either screened out or computed to have "bounding" CDF values less than 1E-6/ry. 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.

2.0 SEISMIC AND HFO ASSESSMENTS EVALUATION

2.1 Scope and Limitations of the IPEEE Seismic and HFO Analyses Review

This LaSalle IPEEE submittal itself is contained in a combined IPE/IPEEE submittal [3]. This submittal itself is quite brief, and presents only a limited discussion of the licensee's review of the underlying PRA study used as the basis for both the IPE and IPEEE. The actual results are taken directly from the Risk Methods Integration and Evaluation Program (RMIEP) [6]. Given that only very limited information is provided in the IPE/IPEEE submittal itself, this review has also considered the documentation available for the RMIEP studies, and in particular, Volumes 7 and 8 of the RMIEP report which document the seismic and HFO analyses.

This review was limited to verifying that the critical elements of an acceptable seismic and HFO analysis have been presented consistent with the objectives of the IPEEE process. An in-depth evaluation of the various inputs, assumptions, and calculations was not performed. The review was performed according to the guidance presented in Reference 5. The technical review results are presented in Section 2.2. Conclusions as to the adequacy of the LaSalle IPE/IPEEE submittal with regard to the seismic and HFO assessments and their use in supporting the resolution of other issues are presented in Section 3.0.

The following subsections provide the technical results of the review of the LaSalle IPEEE seismic and HFO assessments. The review compares the IPEEE assessments against the USNRC guidelines for performing the IPEEE. The review also considers the applicability of the IPEEE analyses to certain other issues. Both areas of potential weakness and strength of the seismic and HFO assessments are highlighted.

2.2 Compliance with NRC IPEEE Guidelines

The USNRC guidelines for performance of the IPEEE derive from two major documents. The first is NUREG-1407, and the second is Supplement 4 to USNRC Generic Letter 88-20. Factors for determining the adequacy of a seismic assessment in comparison to these guidelines, as determined in a Step 0 review, have been outlined in "Guidance for the Performance of Screening Review of Submittals In Response to U. S. NRC Generic Letter 88-20, Supplement 4: 'Individual Plant Examinations - External Events,'" [5]. The following sections discuss the licensee's IPEEE submittal in the context of the specific review objectives set forth in this screening review guidance document and assesses the extent to which the submittal has achieved the stated objectives.

2.2.1 Methodology Documentation

The submittal states that both the IPE and the IPEEE are "the result of a detailed review of the NRC's Risk Methods Integration Evaluation Program (RMIEP) ... analysis" [pg. ES-1 of the submittal].[1] Hence, the LaSalle IPEEE submittal is based largely on the results and findings of the RMIEP external events analysis for LaSalle Unit 2. RMIEP was a USNRC-sponsored risk study. LaSalle participated in the study on a voluntary basis. The seismic analysis was performed by Lawrence Livermore National Laboratory (LLNL) and its subcontractor EQE, Inc. The RMIEP study is a full scope, Level I Probabilistic Risk Assessment (PRA). The complementary Phenomenology and Risk Uncertainty Evaluation Program (PRUEP) provides the Level II/III containment integrity and off-site release analyses.

The licensee has cited the RMIEP study results directly in its IPE/IPEEE submittal. These results are generally cited without major modification. The licensee does take exception to certain aspects of the RMIEP analysis. In the case of the seismic analysis, the licensee IPEEE discussions suggest that the RMIEP seismic hazard curve may not be conservative for ground acceleration greater than 0.3g as compared to the latest LLNL hazard curve. However, this was concluded to not have a significant potential impact on the seismic risk estimates. Hence, no significant technical concerns regarding the RMIEP seismic analysis are cited by the utility. (However, see Section 2.2.3 for reviewer's observations.)

No concerns were cited by the licensee with respect to the RMIEP HFO analyses.

Overall, the documentation of the seismic IPEEE/RMIEP methodology is adequate provided that one considers both the IPEEE submittal itself and the underlying documentation for the RMIEP analyses. The IPEEE submittal itself provides only an abbreviated discussion of the analysis methods and findings. The full RMIEP study is publicly available as NUREG/CR documents. The seismic analyses of building, component and soil-structure interactions performed by EQE, Inc. and the seismic PRA were documented in the RMIEP documentation (Vol. 8 of NUREG/CR-4832). (Although not a requirement for an IPE submittal, this report does not contain sufficient detail to allow reproducing any of the final results.) However, the documentation is considered satisfactory.

The documentation of the HFO analyses is contained entirely in the RMIEP documentation (Vol. 7 of NUREG/CR-4832). The documentation is satisfactory.

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[1] Generally, the licensee submittal uses RMIEP to refer to both the Level 1 (RMIEP) and Level 2/3 (PRUEP) analyses performed by the NRC. A similar approach has been used in this report.

In conclusion, the LaSalle IPEEE submittal, taken in conjunction with the publicly available documentation of the RMIEP study, contains adequate detail regarding the seismic and HFO methods of analysis and study results.

2.2.2 Plant Walk-Downs

As a part of the original RMIEP seismic and HFO analyses, several walk-downs were performed in support of RMIEP assessment. These walk-downs were performed by LLNL staff and their subcontractors, and included licensee staff knowledgeable of both plant structural design and plant systems important to safety. However, the licensee has not cited any supplemental or confirmatory walk-downs as having been performed specifically to support the IPEEE analysis.

In conclusion, the walk-downs performed in conjunction with the original RMIEP analysis complies with the USNRC IPEEE guidance. However, the licensee has not cited that confirmatory walk-downs were performed specifically as a part of the IPEEE process. Given the substantial involvement of licensee personnel in the original RMIEP study, however, this is considered a minor weakness of the IPEEE.

1.1.3 Findings on Seismic Analysis

The LaSalle seismic analysis methodology includes all of the major steps of a state-of-the-art seismic PRA. This includes detailed system fault trees and accident scenario event trees, state-of-the-art soil-structure interaction building and component response analyses and plant-specific building and component fragility calculations. The quantification of the point estimate mean seismic risk by integrating the conditional building and component fragilities over the mean hazard curve is appropriate. The analysis, screening and fragilities development was based on a number of supporting plant walk-downs.

Seismic Hazard Characterization

The methodology used to develop the hazard curves was a combination of the methods and data developed in the NRC-sponsored EUS Seismic Hazard Characterization project [8] and made use of the seismic experts' judgement to supplement data for the site. In the usual application of the EUS Characterization project methodology, only regional corrections for the local site soil column conditions are made. In the LaSalle application, rigorous corrections for the local soil column were made using measured soil column data and 1D vertical wave propagation studies using the SHAKE code. This approach (which had been used in the NRC-sponsored SSMRP seismic PRA of the Zion site) is considered to be state-of-the-art and a significant improvement over the use of regional site correction factors.

In regard to the licensee expressed concern that the seismic hazard curve used in the RMIEP quantification was non-conservative for peak ground accelerations above 0.3g (in comparison to the more recent NRC-developed hazard curves reported in NUREG-1488), it would seem that there has been some apparent misunderstanding. The seismic hazard curve denoted as "mean RMIEP" on Figure 7-1 of the Utility IPEEE submittal is not the mean hazard curve used in the RMIEP quantification but rather the median curve as given on Table 4.2 of Volume 8 of NUREG/CR-4832. The NUREG-1488 hazard curve shown on this figure is the mean hazard curve for the LaSalle site from the tables in NUREG-1488. Thus this figure is comparing a "mean" curve against a "median" curve.

The mean hazard curve from the family of hazard curves actually used in the RMIEP quantification is shown on Fig. 4.6-1 of the RMIEP Summary report (NUREG/CR-4832, Vol. 8) labeled as the "PRA curve". (Unfortunately, it is not labeled as a mean curve - a confusing omission.) However, numerical values of points on the true mean RMIEP hazard curve can be derived from Table 4.2 of Volume 8 of NUREG/CR-4832 (by assuming that the uncertainty distribution of the hazard curve at any acceleration is a log normal distribution as was done in the RMIEP quantification). The true mean RMIEP hazard curve is sketched on a copy of Figure 7-1 of the Utility IPEEE submittal on Figure 1 of this report. Comparison of the true mean RMIEP curve with the mean NUREG-1488 curve (which is correctly plotted in the Utility submittal) shows that the true mean RMIEP hazard curve exceeds the NUREG-1488 curve at all accelerations.

However, it should be noted that there is less of a difference between the true mean RMIEP hazard curve and the NUREG-1488 curve than the figure indicates. The RMIEP curve incorporates the effects of soil amplification at the specific LaSalle site. By contrast, the NUREG-1488 incorporates only a minor correction for soil conditions based on averaged regional conditions. In fact, for direct comparison, the NUREG-1488 curves would have to be further modified for the exact LaSalle soil profile. This would have the effect of raising the NUREG-1488 curve, and decreasing (or eliminating) the difference between the two curves.

In summary, the earthquake exceedance frequencies used in the LaSalle seismic analysis were obtained using an NRC accepted approach and data base that was state-of-the-art at the time of the original study.

Initiating Events

The suite of initiating events considered in the RMIEP seismic analysis for LaSalle is considered appropriate and consistent with the USNRC-sponsored SSMRP methodology.

An error was evidently made in the RMIEP quantification of the conditional probability of LOSP at the various seismic levels. The LOSP event was modeled by failure of the ceramic insulators

in the switchyard, whose fragility was given (Table 8.3 of Volume 8 of NUREG/CR-4832) as

$$M_f = 0.21g \qquad \beta_r = 0.30 \qquad \beta_u = 0.37$$

and the corresponding free-field acceleration at the various seismic levels is given as

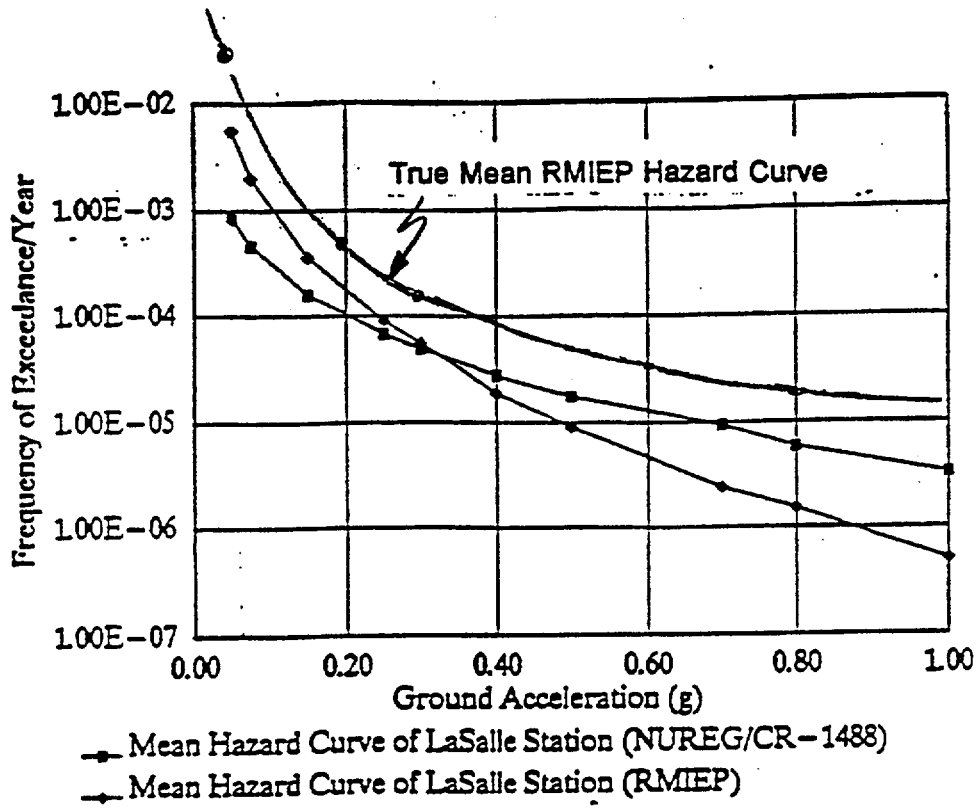

Figure 1 True mean LaSalle seismic hazard curve in comparison with submittal Fig. 7-1

Seismic Level	Median Free Field Acceleration
LL1	0.20g
L1	0.22
L2	0.25
L3	0.28
L4	0.31
L5	0.35
L6	0.40

with corresponding log normal uncertainties of

$$\beta_r = 0.30$$
$$\beta_u = 0.37$$

If these values are used for computing the LOSP conditional probabilities at each seismic level, values greater than those reported in RMIEP (Vol. 2, NUREG/CR-4832, pages C-151 through C- 181) will result. In fact, at the dominant seismic levels (LL1 and L1) the values are about double those reported. Thus the final RMIEP seismic quantification core damage frequencies are about a factor of two too small. However, from a PRA viewpoint this difference is inconsequential.

Building and Component Responses

Accelerations at various floor slabs in the structures, shear loads the critical walls in the structures and moments in the important piping systems were determined by detailed soil structure interaction calculations as embodied in the SSMRP SMACS code approach. The soil structure interaction calculations played important role in the final results inasmuch as the soil layers underlying the LaSalle plant are very soft, and laboratory data obtained for the site showed a very significant degradation of soil stiffness with increasing levels of acceleration. Lumped mass shear beam models of the critical structures (both horizontal and vertical) were used in the analysis. The overall effect of the soft soil layers was a significant reduction in the accelerations and forces experienced by the buildings and components over those used in the original design analyses. Thus the IPEEE re-analysis, using best estimate median-centered soil properties, showed a considerable reduction in the seismic demand upon the components and structures. This resulted in low probabilities of failure for many of the components and all the structures.

The approach taken is considered to be a strong point of the submittal, and represents a state-of-the-art evaluation.

Building and Component Fragilities

Building and component fragilities were developed using the Engineering Factor of Safety approach employed in the SSMRP. For the structures, detailed fragility evaluations were performed for a large number of shear walls and diaphragms, the primary containment, and concrete internals within the containment. Similarly, for the equipment, fragilities were developed for 50 generic categories of equipment (as defined in the SSMRP) and for 33 LaSalle specific components. For each of these items, the fragility was presented in terms of a median capacity and log normal standard deviations representing both random and modeling components of uncertainty. Particular values are given in Tables 7.3-7.6, 8.2 and 8.3 of the RMIEP report (Vol. 8 of NUREG/CR 4832).

In sum, the development of building and component fragilities was performed by state-of-the-art methods and is considered to be a strength of the submittal. The fact that detailed fragilities were developed for essentially all important structures and components on the fault trees gives a high degree of confidence to the final results of the probabilistic screening process used to identify accident sequences for final quantification.

Systems Analysis

The LaSalle seismic analysis was based directly on the internal events event tree and fault tree models, with the addition of seismic failure modes to the components. Screening was performed, and a dual probabilistic culling criterion (absolute and minimum value) was used in solving the event and fault trees for the final accident sequences. This review finds that the systems modeling performed for the LaSalle seismic analysis was fully consistent with accepted practice at the time of the analysis and with the USNRC IPEEE analysis guidance.

As an historical aside, it was noted that the condensate storage tank (CST) had a relatively low median seismic capacity ($M_f = 0.34g$). (The condensate storage tank sees the same free field acceleration as listed for the ceramic insulators above). However, the CST did not show up in the final RMIEP quantification. Early RMIEP fault tree modelling assumed that failure of the CST would directly result in failure of both HPCS and RCIC, which would be very significant. Further, the dominant seismic sequences as evaluated in RMIEP involve RCIC failures due to station blackout over a period of six hours. By contrast, if the CST failed and failed RCIC, the failure would occur in 10-20 minutes. (This could change the accident progression significantly.)

The fault trees were evidently later modified in a way that decreased the importance of the CST and hence the CST did not show up in the final accident sequences quantified in RMIEP. It would seem prudent for the licensee to review the assumptions made in the final RMIEP fault tree models to assure that the lack of significance of the CST in a seismic event is valid and not compromised by any subtle electro-mechanical or human interactions.

Quantification and Uncertainty Analysis

As mentioned earlier, following the screening process, six accident sequences survived for final quantification. The sequences were dominated by sequences involving loss of off-site power and station blackout conditions. The quantification approach (integrating the conditional occurrence probabilities of the accident sequences over the hazard curve) is state-of-the-art. By varying the modeling uncertainties, an uncertainty analysis was performed by standard approaches. The quantification approach and performance of the uncertainty analysis are considered strengths of the submittal.

The final results of the seismic analysis estimate the total plant seismic-induced core damage frequency as 6.0E-7 per year (mean point estimate).

Details of the results of the quantification are given in Section 11 of the RMIEP report (Vol. 8 of NUREG/CR 4832). Various tables provide the conditional probabilities of the accident sequences as a function of earthquake hazard level, unconditional probabilities of the accident sequences as a function of earthquake hazard level and contributions to total risk as a function of initiating event. These tables provide a detailed presentation of the quantification results and allow one to identify dominant accident sequences and component failures. The fact that - as noted by the licensee - the conditional accident sequence frequencies increased very slowly with increasing earthquake levels is discussed, and explained by the fact that, other than the initiator, the dominant equipment failures are due to random events rather than earthquake induced failures. This result can be traced back to the reanalysis of the building and component responses during earthquakes, which resulted in a significant decrease in the earthquake demand upon structures and components.

No specific importance ranking studies were performed. However the tabulated quantification results allows easy identification of the sensitivity and importance of various events on the fault trees to the final core damage frequency.

Summary of Review Findings on the Seismic Analysis

Overall, the seismic risk assessment reported in the LaSalle IPEEE submittal constitutes a detailed, defendable analysis using state-of-the-art approaches and a detailed evaluation of component and structure responses and fragilities. As such, it meets the intent and requirements of a seismic PRA for an IPEEE submittal.

1.1.4 Findings on HFO Analyses

The three step screening methodology recommended by the NRC-sponsored PRA Procedures Guide [9] was used to screen out the bulk of the "other" external events (listed in Section 1.3.2). This approach is well-documented and accepted. Thus the methods used to screen out HFO events are consistent with the current IPEEE approaches.

The bounding analyses performed for the five events remaining after the initial screening, namely,

Aircraft impact
External flooding
Transportation accidents
Turbine missiles
Winds and tornadoes

were state of the art and described in sufficient detail to have confidence in the conclusion that they were insignificant contributors to overall plant risk.

Overall, the HFO portion of the IPEEE is considered to meet the intent and requirements of the IPEEEE program.

2.3 Special Issues

As a part of the IPEEE submittal, the utilities were asked to address a number of issues identified by the USNRC as Unresolved Safety Issues (USI) and Generic Safety Issues (GSI). Specific review guidance on these issues is taken from Reference 5.

2.3.1 Decay Heat Removal (USI A-45)

As discussed in Generic Letter 88-20 [1] and NUREG 1407 [2], USI A-45 (which is associated with the adequacy of decay heat removal at nuclear power plants) is subsumed into the IPE program. A submittal meeting the intent of Generic Letter 88-20, Supplement 4 is assumed to satisfy the requirements of USI A-45. Specifically, the seismic assessment presented in the IPEEE submittal should address the adequacy of long-term decay heat removal in the event of earthquakes.

The Licensee submittal refers to NUREG-5305, Volume 1, Section 3.2 [10] as providing a thorough discussion of decay heat removal. Indeed the RMIEP/PRUEP was a full scope PRA including Level 2 and 3 analyses. As such, consideration was given to potential loss of RHR functions. No unique seismic or HFO vulnerabilities in this regard were noted in either the licensee IPE/IPEEE submittal or in the RMIEP analyses. Components required for long-term decay heat removal were included in the seismic fault trees and event trees, and no vulnerabilities were found. Thus USI A-45 was satisfactorily addressed.

2.3.2 GSI –131

This GSI, which applies primarily to seal table restraint failures in Westinghouse PWR reactors, is not applicable to boiling water reactors.

2.3.3 Seismic/Fire Interactions (MSRP)

The issue of seismic/fire interactions involves primarily two concerns. First is the potential that seismic events might also cause fires internal to the plant, and second is the potential that seismic events might either render inoperable or spuriously actuate fixed fire detection and suppression systems. Following this postulated event, fire suppressant inventories may be depleted and/or

damage to safety-related equipment may be an issue. It had been anticipated that either the seismic or fire IPEEE submittals would provide for some treatment of these issues through a focussed seismic/fire interaction walk-down.

The RMIEP study explicitly states that "the issue of seismic/fire interactions was out of the RMIEP scope, although the issue was considered in the initial planning of the project but thought to be probabilistically unimportant." (Refer to Page 1-2 of RMIEP) No further mention of seismic/fire interactions is provided in either the RMIEP seismic analysis or in the IPE/IPEEE submittal itself.

2.3.4 Effects of Hydrogen Line Ruptures (MSRP)

This issue was not specifically addressed in the seismic or HFO submittals.

2.3.5 Common Cause Failures Related to Human Errors (MSRP)

Both operator errors and maintenance errors were included in the RMIEP system fault trees. In fact, one operator error (associated with manual initiation of the ADS) was in the final 16 basic events which survived the probabilistic screening process. No increases in the human error rates due to seismic induced stress were mentioned. The operator error leading to failure to initiate the ADS system (mentioned above) is, however, a common cause failure related to human error. Thus this issue was addressed to some extent in the submittal. (Refer to Pages 12 & 13 of the IPEEE submittal and Page 246 of RMIEP, Volume 8.)

2.3.6 Seismic Induced Spatial and Functional Interactions (MSRP)

This issue is always addressed as part of the seismic/HFO plant walkdowns. Components which potentially could interact with other components in the case of failure or excessive deformation are identified, and their capacities subsequently evaluated. Consideration of such issues as part of the plant walkdowns was specifically mentioned in the RMIEP report. Thus, even though no walkdown findings are specifically documented in the submittal, this issue is considered by this reviewer to have been satisfactorily addressed. Note, also, this issue is formally subsumed into the USI A-46 program. (Refer to Pages 5 & 6 of the IPEEE submittal and Pages 197-202 of RMIEP, Volume 8.)

2.3.7 Seismic Induced Flooding (MSRP)

Not specifically addressed in the seismic submittal. This issue is normally addressed as part of the seismic plant walkdowns, by identifying liquid containing tanks and pipelines whose failure could result in release of sufficient volume of fluid to damage other (nearby) safety-related

components. (Refer to Pages 5 & 6 of the IPEEE submittal and Pages 197-202 of RMIEP, Volume 8.)

In summary, the issue of seismically induced was probably adequately addressed.

2.3.8 Seismic Induced Relay Chatter (MSRP)

Relays were explicitly included in the system fault trees and assigned a generic fragility (which, in light of today's knowledge, is probably conservative for most of the relays). In fact, eight of the sixteen cutset basic events which survived the probabilistic screening process were either relay or circuit breaker failure events. Thus, for the state of knowledge of that day, relay failures were explicitly included in the seismic analysis and hence this issue is considered to be satisfactorily addressed. (Refer to Pages 205 & 243 of RMIEP, Volume 8.)

2.3.9 Non-safety Related Control System/Safety Related Protection System Dependencies (MSRP)

Multiple failures in non-safety-related control systems may have an adverse impact on safety-related protection systems as a result of potential unrecognized dependencies between control and protection systems. The licensee's IPE process should provide a framework for systematic evaluation of interdependence between safety-related and non-safety related systems and identify potential sources of vulnerabilities. (Refer to Pages 5 & 6 of the IPEEE submittal and Pages 197-202 of RMIEP, Volume 8.)

This issue is considered to be satisfactorily addressed by the plant walkdowns performed as part of the seismic/HFO studies.

2.3.10 Effects of Flooding and/or Moisture Intrusion on Non-Safety- and Safety-Related Equipment (MSRP)

Flooding and water intrusion events can affect safety-related equipment either directly or indirectly through flooding or moisture intrusion of multiple trains of non-safety-related equipment. This type of event can result from external flooding events, tank and pipe ruptures, actuation of the fire suppression system, or backflow through part of the plant drainage system

The potential for external flooding was considered in the HFO analyses, but external flooding was screened out as a negligible risk contributor. There is no explicit discussion of this issue in the LaSalle seismic analysis. The seismic analysis did, of course, consider tanks and piping as part of the walkdown and in the fragility analyses, but evidently did not explicitly consider seismic damage to the fire suppression system. However, the separate internal flood analysis did

consider potential flooding due to failures in the fire water supply system. (Refer to RMIEP Volume 10, entire report.)

In summary, the issue of seismically induced or external source flooding has probably been adequately addressed.

2.3.11 Multi-Unit Risk Implications

The LaSalle analysis is based on the analysis of Unit 2 only. No discussion of potential multi-unit risk implications is provided.

2.3.12 GSI-103, Design for Probable Maximum Precipitation (PMP)

The LaSalle HFO analysis included consideration of local PMP issues as part of their flooding evaluation. The analysis did not discuss GSI-103 and did not discuss roof ponding issues explicitly. However, their evaluation of local PMP issues was sufficient to conclude that local intense precipitation causes no risk at the plant, as explained below. (Refer to Pages 3-75 & 3-76 of RMIEP, Volume 7.)

The FSAR design PMP is 32 inches of rainfall in 24 hours. The impact on the plant was evaluated (for design purposes) assuming the PMP occurred for 48 hours, and all drains were clogged.

By contrast, maximum precipitation at the site observed over a 15 year period (prior to the HFO study) was only 4.5 inches in 24 hours. Records kept for Chicago (about 50 miles away) over a 100 year period showed a maximum of 6.2 inches in 24 hours. Records for the entire northern half of Illinois showed a maximum of 7.6 inches in 24 hours. Using the 100 year data for Chicago, four extreme value type distributions were fit to the data, and all showed that the design PMP of 32 inches of rain in 24 hours (even at the 95% confidence level) had an annual occurrence probability of less than 10^{-8} /yr. Thus, even though the submittal did not discuss enhanced local precipitation rates and roof ponding as discussed in GSI-103, the HFO analysis of local historical data shows that the risk due to local intense precipitation is negligible.

2.4 Containment Performance Issues Unique to Seismic Scenarios

No unique containment failure modes were identified as a part of the seismic analysis. The containment structure for Unit 2 was analyzed for seismic failure modes but these were apparently screened out. Thus this issue is considered to be satisfactorily addressed by the PRA.

2.5 Plant Vulnerabilities and Improvements

The LaSalle seismic risk assessment concludes that there are no seismic or HFO vulnerabilities. The licensee states that "the RMIEP results were well within the safety goals established by the Nuclear Regulatory Commission (NRC)", an apparent reference to the USNRC subsidiary safety goals. Hence, it would appear that the licensee's definition of a vulnerability was based on the USNRC subsidiary safety goals.

No specific plant improvements resulting from the seismic or HFO analyses are cited in the IPE/IPEEE submittal. However, the plant does note that the analysis represents the state of the plant as "frozen" in 1985 and that changes have been made to the plant that are not reflected in the analysis. It also acknowledges that the RMIEP summary documents note "several technical concerns." These were to 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." No discussion of these specific recommendations is provided in the IPE/IPEEE submittal.

3.0 FINDINGS AND RECOMMENDATIONS

The LaSalle IPEEE seismic and HFO analyses are based on the USNRC-sponsored RMIEP study. The seismic analysis in RMIEP was a Level 1 full-scope PRA performed using state-of-the-art analysis methods, data, and assumptions. The seismic and HFO analysis processes were clearly capable of identifying potential plant vulnerabilities and were fully consistent with the USNRC guidance for performance of an IPEEE seismic and HFO analysis. The IPEEE submittal concluded that there were no plant vulnerabilities, and no specific plant modifications resulting from either the seismic or HFO analyses were cited.

Some areas of strengths and weaknesses were identified in the analysis. They include:

Strengths:

- The plant systems models used in the seismic analysis were derived directly from the full-scope internal events PRA.

- The available documentation is quite extensive assuming that one includes the underlying RMIEP seismic and HFO analysis NUREG/CR reports (in particular, Volumes 7 and 8).

- The seismic study was based on a complete, probabilistic re-assessment of the building, piping and component responses using the actual site soil column and its properties. This approach is still the state-of-the-art and represents a major effort.

Major Weaknesses:

There were no major weaknesses identified in this review.

Minor Weaknesses:

- The potential contribution of seismic/fire interactions has not been considered explicitly in the fire analysis or in the seismic analysis. The potential contribution of this issue is qualitatively judged in the RMIEP study to be "probabilistically insignificant" but no explicit justification for this conclusion is provided. No supplemental discussion of the issue is provided in the IPEEE submittal itself. This is not considered a major weakness of the study.

- The licensee cites that the risk study reflects the condition of the plant as frozen in 1985. It is cited that plant modifications have been made since that time. These modifications have not been identified, and no attempts were made to update the PRA to incorporate the changes.

- It would appear that the licensee has not performed any confirmatory walk-downs nor risk quantification updates as a part of the IPEEE process.

- The analyses are based on the analysis of Unit 2. There is no explicit discussion of the applicability of the study to Unit 1 (i.e., consideration of unit differences that might impact the risk quantification).

- There was no specific discussion of potential multi-unit risk implications (e.g., assumed reliance on shared equipment or systems and/or the potential impact of multi-unit failures due to the common cause nature of the earthquake demand.

Overall, the author finds that the LaSalle IPEEE seismic and HFO analyses do meet the objectives of the IPEEE process. Given the lack of any identified major weaknesses, no further review or interactions with the licensee are recommended.

4.0 REFERENCES

1. USNRC, "Individual Plant Examination of External Events for Severe Accident Vulnerabilities - 10 CFR §50.54(f)," Generic Letter 88-20, Supplement 4, April 1991.

2. USNRC, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," NUREG-1407, May 1991.

3. *LaSalle County Nuclear Power Station: Individual Plant Examination and Individual Plant Examination (External Events) Submittal Report*, Commonwealth Edison, April 1994.

4. S. Nowlen, M. Bohn, D. Whitehead, "LaSalle County Nuclear Power Station Individual Plant Examination for External Events Insight Support Report," A letter report to the USNRC, Feb. 28, 1996, transmitted under cover letter: Fred T. Harper (SNL), to Mr. Roy Woods (USNRC/RES/PRAB), Feb. 29, 1996, work performed under USNRC JCN W6187, Task 6-2.

5. S. Nowlen, M. Bohn, J. Chen, "Guidance for the Performance of Screening Reviews of Submittals in Response to U.S. NRC Generic Letter 88-20, Supplement 4: 'Individual Plant Examination - External Events,'" Rev. 3, 21 Mar 1997.

6. *Analysis of the LaSalle Unit 2 Nuclear Power Plant: Risk Methods Integration and Evaluation Program (RMIEP)* NUREG/CR-4832, Vols.1-10, March 1993.

7. M. P. Bohn *et al, Application of the SSMRP Methodology to the Seismic Risk at the Zion Nuclear Power Plant*, NUREG/CR-3428, January 1984.

8. P. Sobel, *Revised Livermore Seismic Hazard Estimates for Sixty-Nine Nuclear Power Plant Sites East of the Rocky Mountains,* NUREG-1488, April 1994.

9. ANS and IEEE, *PRA Procedures Guide*, NUREG/CR-2300, Vols. 1 and 2, January 1983.

10. *Integrated Risk Assessment for the LaSalle Unit 2 Nuclear Power Plant: Phenomenology and Risk Uncertainty Evaluation Program (PRUEP)*, Sandia National Laboratories, NUREG/CR-5305, Vol. 1 (of 3), 1992.

Attachment B

LASALLE COUNTY POWER STATION
INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS (IPEEE)
SANDIA NATIONAL LABORATORIES
TECHNICAL EVALUATION REPORT
RELATED TO FIRE EVENTS

Review of the Submittal in Response to U.S. NRC Generic Letter 88-20, Supplement 4: "Individual Plant Examination-External Events"

Fire Submittal Screening Review
Technical Evaluation Report: LaSalle County Power Station
Revision 1 (final): October 18, 1999

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1.0 INTRODUCTION

1.1 Objectives

The performance of an Individual Plant Examination of External Events (IPEEE) was requested of all commercial U.S. nuclear power plants by the U.S. Nuclear Regulatory Commission (USNRC) in Supplement 4 of Generic Letter 88-20 [4]. Additional guidance on the intent and scope of the IPEEE process was provided in NUREG-1407 [5]. The USNRC has established a screening review process to help the USNRC staff determine if the individual licensee submittals have met the intent of the generic letter and the extent to which the assessment addresses certain other specific issues and ongoing programs. This document presents the results of a "Step 0" review of the LaSalle County Nuclear Station (LaSalle) IPEEE fire assessments.

LaSalle submitted a combined IPE/IPEEE submittal to the USNRC in April 1994 [1]. The external events portions of that document were previously reviewed by SNL in February 1996 [2]. At that time, the IPEEE review process had not yet been fully established, and no specific review guidance was available. Since that time IPEEE review guidance has been developed [3] and the scope of the reviews has been expanded to include consideration of a specific set of special issues deriving from the Fire Risk Scoping Study [9], Generic Safety Issues, and Unresolved Safety Issues. The objective of this report is to re-consider the LaSalle IPEEE fire assessment within the specific context of the current IPEEE "Step 0" review process.

1.2 Plant Description

LaSalle is a two unit site (units 1&2). Each unit is a General Electric (GE) boiling water reactor (BWR) 5 with a design output rating of 3293 MWt and 1078 MWe. The reactor itself is housed within a Mark II containment structure. The plant is owned and operated by Commonwealth Edison Company (CECo) and is located in LaSalle County, Illinois, about 55 miles southwest of Chicago. Construction began in 1973 for Unit 1 and in 1972 for Unit 2. Commercial operation was initiated in 12/82 and 10/84 respectively. The IPEEE analysis is based on the analysis of Unit 2.

The primary containment is a post-tensioned reinforced concrete structure with a steel liner. The containment consists of a lower cylindrical portion founded on a base mat and an upper portion that is in the form of the frustum of a cone. The containment is topped by an elliptical steel dome called the drywell head. The lower portion is called the suppression chamber (or wetwell) and it contains the suppression pool; the upper portion is called the drywell and it houses the reactor pressure vessel. The primary containment is enclosed by a reinforced concrete reactor building which forms the secondary containment. During power operation, the primary containment is inerted with nitrogen. The internal design pressure of the primary containment is

45 psig. The nominal free volumes of the drywell and the suppression chamber are 219,800 ft^3 and 165,100 ft^3, respectively. The nominal volume of the suppression pool is 128,800 ft^3.[1]

Important Design Characteristics include the following features. High-pressure injection is provided by the following four systems:

- High-pressure core spray (HPCS): HPCS consists of a motor-driven pump with its own dedicated diesel. It draws water from either the condensate storage tank or the suppression pool and sprays coolant onto the core.

- Reactor core isolation cooling (RCIC): RCIC consists of a turbine-driven pump to pump water from the condensate storage tank or the suppression pool to the reactor vessel. The turbine uses steam from the reactor pressure vessel; thus, system operation cannot be ensured after the reactor vessel pressure decreases below a specified point—57 psig. In addition, RCIC isolates when the containment pressure reaches about 15 psig.

- Main feedwater (MFW): MFW takes suction from the condenser hotwell using two turbine-driven pumps and one motor-driven pump and injects the water into the vessel through the main feedwater lines. MFW can also take suction from the condensate storage tank; however flow is limited to a maximum of 1200 gpm. All pumps require offsite power to operate.

- Control rod drive (CRD): CRD can be used to inject several hundred gallons of water per minute into the reactor vessel and is only useful once the decay heat load has decreased, as in a long-term accident, or in conjunction with another system.

- Shared control room: Units 1 and 2 at LaSalle share a common control room.

To use the low-pressure injection systems, reactor vessel pressure must be reduced. This can be accomplished by the use of the automatic depressurization system (ADS), manual operation of the safety relief valves (SRVs), or by a break in the system that allows system pressure to be reduced below the shutoff head of the low-pressure systems. Low-pressure injection is provided by the following four systems:

- Condensate system (CDS): CDS takes water from the condenser hotwell and pumps it through the feedwater line into the reactor pressure vessel using four motor-driven pumps. CDS can also take suction from the condensate storage tank; however, flow is limited to a maximum of 1200 gpm. CDS requires offsite power to operate.

- Low-pressure core spray (LPCS): LPCS is a single-train system that takes water from the suppression pool and injects it into the reactor pressure vessel via a motor-driven pump. LPCS is powered by train A of emergency power.

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[1] pp. 1-2 through 1-6 of NUREG/CR-4832, Vol.1

- Low pressure coolant injection (LPCI): LPCI is a three-train system that takes water from the suppression pool and injects it into the reactor pressure vessel using three motor-driven pumps. Train A is powered by train A of emergency power, and trains B and C are powered by train B of emergency power.

- Diesel-driven firewater system (DFWS): The DFWS must be manually connected to the MFW injection line before injection into the vessel can occur. Diesel-driven pumps are then used to inject water. These diesel-driven pumps make operation during station blackout possible.

Heat can be removed from the containment by the residual heat removal (RHR) system. Three modes of operation are possible. The first two—suppression pool cooling (SPC) and containment spray (CS)—can be used during any type of accident. The last—shutdown cooling—can be used for non-LOCAs. In addition, the containment can also be vented by use of the containment vent and purge system. Venting can be from either the drywell or the suppression chamber using either 2-inch lines or 26-inch lines. The 26-inch vent lines tie into the standby gas treatment system (SGTS), which releases the gases to the stack. The 26-inch vent lines are attached via an 18-inch pipe to the SGTS with a rubber boot which is assumed to fail when high- pressure steam is released through the vent.

Directly below the reactor pressure vessel is the reactor pedestal cavity. The arrangement of the pedestal cavity has potentially important implications for severe accidents. The cavity is divided into two regions: the upper cavity and the lower cavity. The upper cavity communicates with the drywell airspace while the lower cavity communicates with the wetwell airspace. The upper cavity volume is large enough to hold all of the debris that would be released should the vessel fail and, hence, drywell melt-through scenarios that are a concern in Mark I containments are not a concern in the LaSalle plant. The drywell is provided with drains that direct water that accumulates in the drywell to the upper pedestal cavity sumps. There are additional lines that drain water from the sumps to systems outside the containment. While these sump drains include isolation valves, these valves are isolated outside the containment boundary. A potential containment isolation failure mechanism is failure of the drain line outside the containment from energetic events that occur during a severe accident. An additional concern is that energetic loads that can accompany vessel failure and thermal attack by molten core debris released from a failed vessel can potentially fail the floor that separates the upper cavity from the lower cavity. Failure of the floor establishes a pathway that connects the drywell and wetwell airspace and bypasses the suppression pool.

1.3 Primary Results of the IPEEE Fire Analysis

The estimated core damage frequency (CDF) due to fire at LaSalle as reported in the IPEEE submittal is taken directly from the Risk Methods Integration and Evaluation Program (RMIEP). RMIEP was a USNRC-sponsored full-scope Level I Probabilistic Risk Assessment (PRA). The

RMIEP study was led by Sandia National Laboratories (SNL). The mean fire-induced CDF for LaSalle as determined in RMIEP is 3.21E-5/ry.

The fire analysis initially considered over 160 fire areas, but qualitative and quantitative screening left only ten fire areas for detailed analysis. The results for the ten unscreened fire areas are presented in Table 3.1 on page 3-2 of reference 6.

The single largest contributor is the MCR at 1.39E-5. This value derives from a relatively simplistic analysis of essentially one fire scenario; namely, any MCR fire leading to MCR abandonment. The conditional probability of MCR abandonment given a fire was assumed to be 1-in-10 (0.1). This was based in part on the observations that the LaSalle control panels have open tops with no in-cabinet detection. It was assumed that once established, a fire in the main control panels might grow rapidly and involve the cables above the panels. Manual detection of panel fires was considered unlikely because the control panel access space also acts as a return air plenum for the ventilation system. The subsequent recovery of the plant from the remote shutdown panel (RSP) was explicitly modeled assuming a need to manually recovery the emergency core cooling system (ECCS) in any such fire. The "best estimate" RSP failure probability was estimated as 0.064.

The next two most significant fire areas are the Division 2 switchgear room at 8.51E-6 and the Division 1 switchgear room at 5.15E-6 CDF/ry. This is consistent with past fire risk results in that switchgear areas are often significant fire risk contributors. In these analyses, it was assumed that a switchgear fire would escape through penetrations at the top of the panel. Subsequent fire modeling predicted that even a small fire on top of a panel would propagate up the cable risers (vertical cable air drops) to overhead cable trays. A fourth area of importance was the Auxiliary Equipment Room at 2.62E-6 CDF/ry. The other six contributing areas are all below 1E-6/ry each, including the cable spreading room (CSR) at 1.63E-7/ry.

All of the postulated fire scenarios that contributed to fire risk involve an initial fire source, generally either a panel fire or transient fire source, and subsequent damage to overhead cables. No fire scenarios leading directly to core damage ("singles" or CCDP=1) were identified. All scenarios involved the random or non-fire-induced failure of systems required to mitigate an initiating event.

1.4 Scope and Limitations

This LaSalle IPEEE submittal itself is contained in a combined IPE/IPEEE submittal [1]. This submittal itself is quite brief, and presents only a limited discussion of the licensee's review of the underlying PRA study used as the basis for both the IPE and IPEEE. The actual results are taken directly from the Risk Methods Integration and Evaluation Program (RMIEP) [6]. Given that only very limited information is provided in the IPE/IPEEE submittal itself, this review has also considered the documentation available for the RMIEP studies, and in particular, Volume 9 of the RMIEP report which documents the fire analysis.

This review was limited to verifying that the critical elements of an acceptable fire analysis have been presented consistent with the objectives of the IPEEE process. An in-depth evaluation of the various inputs, assumptions, and calculations was not performed. The review was performed according to the guidance presented in Reference 3. The technical review results are presented in Section 2.0. Conclusions and a recommendation as to the adequacy of the LaSalle IPE/IPEEE submittal with regard to the fire assessment and its use in supporting the resolution of other issues are presented in Section 3.0.

2.0 FIRE ASSESSMENT EVALUATION

The following subsections provide the technical results of the review of the LaSalle IPEEE fire assessment. The review compares the IPEEE fire assessment against the USNRC guidelines for performing the IPEEE. The review also considers the applicability of the IPEEE fire analysis to certain other issues. Both areas of potential weakness and strength of the fire assessment are highlighted.

2.1 Compliance with NRC IPEEE Guidelines

The USNRC guidelines for performance of the IPEEE derive from two major documents. The first is NUREG-1407, and the second is Supplement 4 to USNRC Generic Letter 88-20. Factors for determining the adequacy of a fire assessment in comparison to these guidelines, as determined in a Step 0 review, have been outlined in "Guidance for the Performance of Screening Review of Submittals In Response to U. S. NRC Generic Letter 88-20, Supplement 4: 'Individual Plant Examinations - External Events,'" [3]. The following sections discuss the licensee's IPEEE submittal in the context of the specific review objectives set forth in this screening review guidance document and assesses the extent to which the submittal has achieved the stated objectives.

2.1.1 Methodology Documentation

The submittal states that the both the IPE and the IPEEE are "the result of a detailed review of the NRC's Risk Methods Integration Evaluation Program (RMIEP) ... analysis" [pg. ES-1 of the submittal].[2] Hence, the LaSalle IPEEE submittal is based largely on the results and findings of the RMIEP external events analysis for LaSalle Unit 2. RMIEP was a USNRC-sponsored risk study. LaSalle participated in the study on a voluntary basis. The fire analysis was performed largely by Sandia National Laboratories (SNL). The RMIEP study is a full scope, Level I Probabilistic Risk Assessment (PRA). The complimentary Phenomenology and Risk Uncertainty Evaluation Program (PRUEP) provides the Level II/III containment integrity and off-site release analyses.

The licensee has cited the RMIEP study results directly in its IPE/IPEEE submittal. These results are generally cited without major modification. The licensee does take exception to certain aspects of the RMIEP analysis. In the case of the fire analysis, the licensee cites perceived conservatism in certain fire event frequencies and the analysis of the main control room (MCR) (these points will be discussed further in subsequent sections of this report). However, no individual fire scenarios were actually re-quantified by the licensee. Ultimately, the licensee cites the RMIEP fire analysis CDF results directly as the IPEEE fire analysis results with the notation that the results are perceived to contain substantial conservatism.

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[2] Generally, the licensee submittal uses RMIEP to refer to both the Level 1 (RMIEP) and Level 2/3 (PRUEP) analyses performed by the NRC. A similar approach has been used in this report.

Overall, the documentation of the IPEEE/RMIEP methodology is very complete provided that one considers both the IPEEE submittal itself and the underlying documentation for the RMIEP analyses. The IPEEE submittal itself provides only an abbreviated discussion of the analysis methods and findings. However, the full RMIEP study is publically available as NUREG/CR documents. Hence, full documentation of the licensee IPEEE analysis is readily available.

The fire analysis includes all of the major steps of a state-of-the-art fire PRA of the time (1985-1990 vintage). This includes fire area/compartment identification, mapping of safe shutdown equipment locations, extensive cable tracing, qualitative screening using spatial failure analysis, a inter-compartment interaction analysis equivalent to, or more rigorous than, a typical Fire Induced Vulnerability Evaluation (FIVE) methodology Fire Compartment Interaction Analysis (FCIA) [8], quantitative screening including determining the safe shutdown failure probability for unscreened fire initiators using the PRA models derived from the internal events analysis, a complete set of fire-induced initiating events, fire propagation and damage analyses, uncertainty and sensitivity analyses, and a number of supporting plant walk-downs.

In conclusion, the LaSalle IPEEE submittal, taken in conjunction with the publically available documentation of the RMIEP study, contains more than adequate detail regarding the methods of analysis and study results. The methods applied do comply with the USNRC guidance for performance of the IPEEE fire analysis.

2.1.2 Plant Walk-Downs

As a part of the original RMIEP analysis, several walk-downs were performed in support of the fire assessment. These walk-downs included participation by SNL staff, SNL subcontractors, and licensee staff knowledgeable of both fire protection and plant systems important to fire safety. However, the licensee has not cited any supplemental or confirmatory walk-downs as having been performed specifically to support the IPEEE analysis.

In conclusion, the walk-downs performed in conjunction with the original RMIEP analysis would easily comply with the USNRC IPEEE guidance. However, the licensee has not cited that confirmatory walk-downs were performed specifically as a part of the IPEEE process. Given the substantial involvement of licensee personnel in the original RMIEP study, this is considered, at most, a minor weakness of the IPEEE.

2.1.3 Fire Area Screening

Fire areas for the LaSalle analysis were defined using a definition similar to that given in USNRC Generic Letter 83-33. In this Generic Letter, a fire area is bounded on all sides by three-hour rated barriers. In the LaSalle analysis a slightly more liberal definition was applied in that fire areas were defined based on the existence of any fire rated boundary even if that boundary's rating was less than three hours. The analysis did not, however, rely on the fire zone definitions

cited in the LaSalle *Fire Hazards Analysis*. In particular, fire zones that were connected by openings or by un-sealed or un-rated barriers were combined for screening purposes. In the final post-screening detailed quantification, fires were generally found to have a localized impact near the specific fire location based on the results of fire modeling.

The LaSalle fire analysis included two steps of fire area screening. The first step was qualitative screening based primarily on the results of the system/component mapping. In this step, the analysis considered the potential for both trip initiators and degradation of safe shutdown systems and equipment. If either condition was possible due to fire, the area was retained for further analysis. The second screening step was quantitative and used a screening criterion for single fire areas of 1E-9 CDF/ry (pg. 3-4 of Ref. 6). This step of screening considered, at most, a nominal fire initiation frequency and the CCDP assuming loss of all equipment in a given fire area or combination of fire areas. There is very little information provided as to the basis for screening of each individual fire area. Rather, one is provided with a list of those areas reviewed by walk-down, and those areas retained for detailed analysis. However, the screening criteria and approach are consistent with accepted PRA practice.

The screening also included consideration of room-to-room fire effects. For fire area combinations, a nominal frequency of 1E-4/ry was assumed for challenging fires and a nominal barrier failure probability of 0.1 per barrier was assumed (pg. 3-60 of Ref. 6). Recall that there is a rated fire barrier between any pair of fire areas as defined in the analysis. For example, with a three area combination, two barriers must fail and the nominal probability of fire impacting all three rooms was nominally (1E-4*0.1*0.1) = 1E-6. The CCDP given damage to the equipment in each impacted fire area was computed. A screening criterion of 1E-8 CDF/ry was assumed for screening of multi-room scenarios. Ultimately, all multiple room combinations were screened.

There is a minor discrepancy in the number of surviving fire areas after screening. Page 3-4 of Reference 6 cites that eleven areas survived screening. However, Table 3.11 on page 3-67 cites that there are fourteen "fire areas (rooms) remaining after screening that required evaluation." This discrepancy is resolved in part by the discussions on pages 3-68 through 3-73. In these discussions, a further screening was performed based on a supplemental mapping of equipment and cables into the fire areas of interest.

- It was found that the initial assumptions regarding the potential for loss of equipment were overly conservative for three fire areas (PP, DD, and F). When adjusted, these areas also screened quantitatively.

- The reactor building (area D) was also subsequently screened based on semi-quantitative arguments. That is, the analysis considered the loss of any one level within the reactor building due to fire and bounded the potential CDF contribution. It was determined (qualitatively) that "no level of the reactor building in itself is likely to result in a fire of sufficient magnitude to fail a sufficient number of systems" to be risk significant given

that "at least two alternate injection systems will be available and the random failures would drop the cut set below the truncation limit. The reactor building was therefore eliminated from further consideration."

Given these discussions, it would appear that ten rather than eleven fire areas were ultimately retained for detailed analysis. While the report repeatedly refers to eleven unscreened compartments, the final list of risk contributors contains only ten fire areas, and all other indications are that ten, rather than eleven, fire areas survived screening. For example, on page 3-118 of reference 6, it is noted that "(t)he following eleven areas had core damage contributions of greater than 1E-09 per year." The list that follows identifies only ten fire areas. Hence, it would appear that ten fire areas actually were analyzed in detail.

Ultimately, there is little information provided regarding the actual basis for screening of any given individual fire area. However, this review finds that the screening criteria and process used in the LaSalle fire analysis is fully consistent with the USNRC IPEEE guidance. Areas typically found to be fire risk important were retained and quantified in detail.

2.1.4 Fire Occurrence Frequency

The fire occurrence frequency estimates used in the LaSalle fire analysis were based on the original SNL fire event data base published in 1985 [7]. Indeed, the fire event data base was originally developed specifically in support of the RMIEP program. Fire event frequencies were based on accepted practice for fire PRAs, and those practices have changed only modestly.

However, this is one area in which the licensee has taken some exception to the RMIEP analysis. In particular, the IPE/IPEEE submittal itself [1] states that the approach to quantifying the fire initiating event frequencies used in the RMIEP study "is very conservative" and that "the initiating event frequency may have been overestimated (in CECo's judgement) by a factor of 10 or more for some areas."

The licensee has provided no detail as to which specific fire initiating event frequencies it considers to be conservative, nor which fire scenarios or fire areas are impacted by this concern. Hence, SNL has no basis for assessing the merit of the licensee concern. It is unclear that updated methods and information would significantly alter the fire initiating event frequencies. A partial review of the cited fire frequencies revealed no obvious discrepancies. Ultimately, the licensee has presented the same CDF results as those reported in the RMIEP study and did not attempt to adjust those values to reflect lower fire frequencies.

In summary, the fire event frequencies used in the LaSalle fire analysis were obtained using commonly accepted PRA practice and a data base that was state-of-the-art at the time of the original study. Use of an updated fire event data base would not be expected to substantially alter the fire event frequencies. The licensee has stated the view that the fire frequencies are substantially overstated, but has not specified which values are considered questionable, nor have

any fire scenarios been re-quantified using lower frequency estimates. Hence, there are no apparent discrepancies in the fire event frequencies used, but the licensee considers the values used to be a significant source of conservatism in the analysis.

2.1.5 Fire-Induced Initiating Events and Fire Scenarios

The LaSalle fire analysis was based directly on the internal events models. All fire initiators were modeled using the event tree for transients. This one event modeled all potential plant transients including general transients and special initiators (failures in support systems leading to a plant trip or shutdown simultaneous with the loss of one or more mitigating systems).

The systems model used for the RMIEP study specifically included consideration of spurious equipment actuations (see volume 6, section 1.2 of the RMIEP report [6]).[3] Both single and multiple spurious actuations were considered, in particular, for potential flow diversion paths sufficiently large that system function might be lost. This was in large part done specifically in support of the fire analysis. It would appear that given a prediction of a system failure, the fire quantification has considered the worst-case mode of system failure for any specific path "through" the fault tree without specific consideration of how various spurious actuations might occur. It is not entirely clear whether the spurious opening of the SRV's was considered, however, in that no LOCA scenarios were apparently quantified.

The fire scenarios for any given fire area included fires of at least two different sizes, namely, "large" and "small" fires. A "small" fire was modeled as one gallon of oil in a 2-foot diameter pool (on the order of 165 kW heat output).[4] A "large" fire was modeled as a ten gallons of oil in a 3-foot diameter pool (on the order of 500 kW heat output). Note that these fires are relatively large in comparison to those fire sources postulated in a typical IPEEE fire analysis.

Each size class was assigned a fraction of the overall fire frequency. These fractions were assigned based on the analyst's judgement and on insights derived from a review of USNRC inspection findings relating to transient fuel sources found at various plants. In general, 30% of the fires (best estimate) were assumed to be "large" and 70% were assumed to be "small." For most fire scenarios only large fires caused critical target damage. The exception was certain

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[3]Note that Volume 6 of the RMIEP report was never published, but the last draft version is available in the Public Document Room.

[4]Note that fire intensity is estimated in COMPBRN based on fundamental fuel input parameters and ventilation conditions. The actual estimated fire heat release rate output values are not given in the RMIEP reports. The heat release rate estimates cited here are for information only and are intended to indicate the relative magnitude of the postulated fires. They are based on a correlation from reference [11]. The Babrauskus correlation has a fundamentally similar form to the surface controlled burning model used in COMPBRN.

switchgear fires where small fires did cause damage to overhead cables. (See discussions on pages 3-76, 3-80, and Table 3.20 on page 3-104 of ref. 6.)

In summary, this review finds that the fire modeling performed for the LaSalle analysis was fully consistent with accepted practice at the time of the analysis and with the USNRC IPEEE analysis guidance.

2.1.6 Fire Propagation and Suppression Analysis

Specific fire propagation, damage, and suppression analyses were performed for each fire area surviving screening. Fire modeling utilized a version of the COMPBRN III fire model modified by SNL to correct certain of the concerns identified in the Fire Risk Scoping Study (i.e., those concerns related to errors in implementation of the fire model code and numerical stability). This version of the code was later verified to be fully consistent (from a numerical input/output standpoint) with version IIIe, the EPRI published version of the code which reportedly incorporated the same set of SNL recommended modifications.[5]

The fire-induced damage explicitly considered in the study was limited to thermal damage to exposed cables. The analysis did not explicitly consider suppression-induced or smoke-induced damage although such damage was considered qualitatively as discussed in Sections 2.2.2 and 2.2.4 below.

The cables at LaSalle are generally qualified as low flame spread per the IEEE-383 standard. Some few exceptions were noted relating to a small number of electrical lighting cables. Damage thresholds appropriate to qualified cables were applied (i.e., 350°C).

A value of 500°C was assumed for both piloted and spontaneous ignition of cables. This is now known to be a non-conservative assumption with regard to the piloted ignition threshold.[6] The assumed value was consistent with the state of knowledge at the time of the analysis. Further, in this particular case the reducing the piloted ignition temperature would have little impact. This is because the analysis has already postulated fires of sufficient intensity so that fire propagation was predicted for many scenarios. In those cases where propagation was not predicted, there also was no prediction of fire damage. Given this, the use of a lower ignition temperature would not alter the propagation predictions. Hence, use of a lower piloted ignition temperature would not

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[5]The discussion of the verification of SNL's unique version of the COMPBRN III code as numerically consistent with the EPRI IIIe version is based on first-hand knowledge of the author, S. Nowlen. These results were specifically sought as a part of the later NUREG-1150 studies to ensure consistency of the SNL version with the published EPRI version purported to have incorporated the exact same code modifications. The results did demonstrate full numerical consistency for the two codes when exercised against a set of test problems.

[6] This was raised as USNRC Generic RAI 16 regarding the EPRI *Fire PRA Implementation Guide.*

substantially alter the final risk estimates. (See Table 3.13 on pg. 3-78 of ref. 6 for further information).

Intervention by fire detection and suppression were also credited in the analysis. In general, both manual actions and fixed systems were credited. The analysis of fire detection and suppression involved a detailed analysis of timing factors and fixed system configurations. Fire brigade response times were based on a combination of historical event data, plant walk-downs that included an assessment of anticipated fire brigade response times, expert judgement, and available information from plant fire drills. The brigade response time estimates included minimum, maximum, and average values (interpreted as 5th, 50th, and 95th percentiles of a log-normal distribution respectively). The total time to suppression explicitly included the time to detection, time to response of fire fighters to the local cages where fire fighting equipment was staged within the plant, suit-up time, response to the fire scene, set-up time at the scene, time to assess and search the scene, and the time to actually suppress the fire. These results are presented in detail in Appendix A of reference 6.

Given sufficient time for actuation, fixed fire suppression systems were generally credited using generic reliability values. However, before such credit was allowed, each system's configuration was specifically reviewed by plant walk-down. Code compliance was not specifically assessed, but if any specific installation features that might interfere with performance of the system were noted, then a delay in the response time was assumed. No system deficiencies were identified that substantially impacted risk quantification.

In summary, this review finds that the fire propagation and suppression analysis was fully consistent with accepted PRA practice and with the USNRC IPEEE guidance. The results are explicitly documented, considered all factors important to such an analysis, and included consideration of uncertainty in the quantification. Indeed, the fire detection and suppression analysis was quite detailed and complete, and is therefore considered a strength of the LaSalle analysis.

The only potential point of concern regarding the fire modeling efforts was use of a cable piloted ignition temperature that is non-conservative based on more recent test results. However, because the LaSalle fire modeling results did predict cable fire propagation for most of the risk significant fire scenarios, reduction of the cable piloted ignition temperature would be expected to have, at most, a minor impact on the fire CDF estimates. Hence, this is considered a minor weakness of the study.

2.1.7 Quantification and Uncertainty Analysis

The quantification of fire scenarios is presented for each fire scenario analyzed in detail to a sufficient level as to allow for reproduction of the final risk estimates. For each scenario the fire modeling results are also presented. No discrepancies with regard to quantification of the individual fire scenarios, nor the quantification of overall plant fire CDF were noted.

The LaSalle fire analysis did explicitly consider uncertainty in several key parameters including fire size, fire location, and time to detection and suppression of the fire. The only potential weakness in the uncertainty analysis is that in the fire modeling portions, uncertainty was only accounted for in a select set of the model input parameters. This is considered, at most, a minor weakness. The basis for choosing which parameters were considered uncertain is not established. Overall, the uncertainty analysis portions of the LaSalle fire analysis are considered a strength of the analysis.

The final results of the fire analysis estimate the total plant fire-induced core damage frequency as 3.21E-5 per year (mean). The results for the ten unscreened fire areas are presented in Table 3.1 on page 3-2 of reference 6. The single largest contributor is the MCR at 1.39E-5 followed by the Division 2 switchgear room at 8.51E-6, the Division 1 switchgear room at 5.15E-6, and the Auxiliary Equipment Room at 2.62E-6. The other six contributing areas are all below 1E-6/ry each, including the CSR at 1.63E-7/ry.

2.1.8 Sensitivity and Importance Ranking Studies

The LaSalle fire analysis does provide and explicit sensitivity and importance ranking analysis. Insights gained from this study are listed on page 3-119 through 3-120 of reference 6.

Three events associated with the MCR quantification are repeatedly cited in the importance rankings. These are the fire frequency in the MCR, the conditional probability of MCR abandonment, and failure to shutdown from the remote shutdown panel dominate the risk importance rankings. This is not surprising given the dominance of the MCR fire risk contribution to the overall plant fire CDF, and the relatively simple analysis of the MCR which is dominated by these three factors. Hence, these particular insights are rather obvious.

Also ranked as important to the potential for risk reduction is the fraction assumed for large fires. This can be attributed to the fact that for most of the dominant areas, only large fires were predicted to cause the critical damage. Hence, the risk estimates scale almost directly with the assumed value of the large/small fire "split" fraction. Most of the other importance insights relate to factors that derive from the internal events study including in particular certain non-fire-induced (random) equipment failure probabilities.

2.2 Special Issues

As a part of the IPEEE fire submittal, the utilities were asked to address a number of fire-related issues identified in the Fire Risk Scoping Study (FRSS) [9] and USNRC Unresolved Safety Issues (USI) and Generic Safety Issues (GSI). Specific review guidance on these issues is taken from Reference 7.

It should be noted that the initial LaSalle fire analysis was completed prior to completion of the FRSS. However, Section 1.1 of the RMIEP report [6] does provide a nominal discussion as to

the extent that the identified FRSS issues were addressed in the RMIEP study. Much of the material in the following subsections derives from this discussion.

2.2.1 Decay Heat Removal (USI A-45)

As discussed in Generic Letter 88-20 [7] and NUREG 1407 [8], USI A-45 associated with the adequacy of decay heat removal at nuclear power plants is subsumed into the IPE submittals. A submittal meeting the intent of Generic Letter 88-20, Supplement 4 is assumed to satisfy the requirements of USI A-45. Specifically, the fire assessment presented in the IPEEE submittal should address the adequacy of long-term decay heat removal in the event of fires.

The Licensee submittal refers to NUREG-5305, Volume 1, Section 3.2 [10] as providing a thorough discussion of decay heat removal. Indeed the RMIEP/PRUEP was a full scope PRA including Level 2 and 3 analyses. As such, consideration was given to potential loss of RHR functions. No unique fire vulnerabilities in this regard were noted in either the licensee IPE/IPEEE submittal nor in the RMIEP fire analysis.

2.2.2 Effects of Fire Protection System Actuation on Safety-related Equipment (FRSS, GSI 57, MSRP)

This issue is associated with the concern that traditional fire PRA methods have generally considered only direct thermal damage effects. Other potential damage mechanisms such as smoke and fire suppression damage (either from fixed systems or manual actions) have not been considered. In general, this is an area where the data base on equipment vulnerability is rather sparse. It was anticipated that a typical IPEEE analysis would provide for some treatment of both smoke and suppression-induced damage.

The LaSalle fire analysis considered the potential for damage due to fire suppressant release in only a qualitative manner. In particular, it is stated in Section 1.1 of reference 6 (page 1-2) that "the issue of total environment equipment survival was only indirectly addressed in this analysis." In the context of fire suppressant-induced damage, the following points were cited:

- "no double areas survived the screening analysis" (this is taken as an indication that room to room smoke propagation need not be explicitly addressed given the conservative damage assessments used in the screening process)

- "there are not many areas of the LaSalle plant with automatic fire suppression systems and the fire suppression system actuation is confined to the immediate vicinity of that area" (this, together with the first point, is taken as an indication that suppression system actuation in adjacent areas is unlikely)

- "random faults of the fire suppression systems that could directly affect system operability were included in the internal event fault trees"

- "floods initiated by faulty firewater system operation were included in the flood analysis"

2.2.3 Fire-induced Alternate Shutdown/Control Room Panel Interactions (FRSS, GSI 147)

The issue of fire-induced alternate shutdown/control room panel interactions is associated primarily with the potential that a fire in the main control room or in the cable spreading room might lead to failures in the remote shutdown capability. This issue was raised as a part of the FRSS and was later designated GSI 147 by the USNRC.

This issue has been examined in substantial detail for LaSalle by virtue of both RMIEP and the FRSS itself. In particular, through a cooperative effort between the FRSS and RMIEP, there was considerable assessment of the LaSalle control systems and control functions. Indeed, information available from the then ongoing LaSalle risk analysis was used in the development of insights in the FRSS itself and LaSalle was one of the case studies examined in the FRSS [see pgs. 25-27 of ref. 9]. However, the full results of these assessments were not reflected directly in the final CDF quantification under RMIEP.

As noted above, the RMIEP risk quantification for the LaSalle MCR is based on one fire scenario; namely, any fire leading to MCR abandonment. This analysis did include specific consideration of remote shutdown capability and independence. However, the RMIEP analysis has not included other related information in its quantification of the MCR CDF contribution. The available information includes two areas of additional analysis that did not factor into the CDF quantification presented in the RMIEP analysis.

First, the control and actuation functions for every component included in the system fault trees was traced on electrical drawings to see if they mapped into the MCR. Within the MCR all such functions were traced to individual panels. An importance ranking of the control panels was then performed based on the potential that loss of the panel might reduce either reactor heat removal capability or coolant inventory to the reactor vessel. Two panels were identified as the most critical; namely, the emergency core cooling system (ECCS) panel and the electrical distribution panel. A subsequent analysis considered the conditional core damage probability (CCDP) given that all of the components associated with one of these two panels failed in the most undesirable state. The CCDP estimates were later refined based on a plant walk-down that revealed that the two panels in question were actually subdivided into a total of six sub-panels. The separation between sub-panels was comprised of solid sheets of steel with very limited and well sealed penetrations. Combining this feature with the fact that the MCR is continuously manned led to the assumption that fire fighting would begin quickly, fires would remain localized, and fire damage would likely be limited to one sub-panel. (Note that this is typical of most IPEEE analyses.) Given the revised damage states, only one sub-panel of the electrical distribution panel was found to have a CCDP of greater than 1E-4.

As a second step, the plant control circuits were examined in detail to determine if any faults due to a fire in the MCR might compromise functions at the remote shutdown panel (RSP). Four potential interactions were identified. However, in each case a potential recovery action was also identified. Ultimately, the CDF associated with fire in the electrical distribution panel was independently estimated as 8.1E-6/ry. The study acknowledges that other interactions may exist that would not have been picked up in this particular study.

Ultimately, these results did not find their way directly into the RMIEP fire risk quantification. The MCR fire risk is estimated based on a single fire scenario involving MCR abandonment and failure to shutdown from the RSP (see pg. 3-100,101 of ref. 6). The scenario assumes a nominal fire frequency of 1.85E-3/ry, a conditional probability of abandonment of 0.1, and a RSP failure probability of 0.064 (all best estimate values, uncertainty bounds are given). When handled with proper treatment of the statistical distributions, this resulted in a mean MCR CDF contribution of 1.39E-5/ry.

2.2.4 Smoke Control and Manual Fire Fighting Effectiveness (FRSS, GSI 148)

Smoke control and manual fire fighting effectiveness is associated with the concern that nuclear power plant ventilation systems are known to be poorly configured for smoke removal in the event of a fire, and hence, significant potential exists for the buildup of smoke to hamper the efforts of the manual fire brigade to promptly and effectively suppress fires.

The LaSalle fire analysis explicitly considered and dismissed the potential that smoke might interfere with fire fighting activities. The following conclusion is taken from Appendix A of reference 6 (pg. A-4):

> "There is an extremely low probability that, upon arriving at the scene, the Fire Brigade will be unable to locate the fire due to smoke obscuration. This is due to the fact that, for most of the areas surviving screening, the large room areas or openness to other areas precluded buildup of sufficient smoke to obscure the fire location. This scenario, therefore, was not considered in evaluating the time to manual suppression and the personnel were not required to suit-up with SCBA (Self-Contained Breathing Apparatus)."

Neither the RMIEP analysis nor the IPE/IPEEE submittal discusses the potential that manual fire fighting activities might lead to damage to equipment not directly impacted by the fire itself (collateral damage).

2.2.5 Seismic/Fire Interactions (FRSS, MSRP)

The issue of seismic/fire interactions involves primarily two concerns. First is the potential that seismic events might also cause fires internal to the plant, and second is the potential that seismic events might either render inoperable or spuriously actuate fixed fire detection and suppression systems. Following this postulated event, fire suppressant inventories may be depleted and/or

damage to safety-related equipment may be an issue. It had been anticipated that a typical fire IPEEE submittal would provide for some treatment of these issues through a focused seismic/fire interaction walk-down.

The RMIEP study explicitly states that "(t)he issue of seismic/fire interactions was out of the RMIEP scope, although the issue was considered in the initial planing (sic) of the project but thought to be probabilistically unimportant." (Refer to Page 1-2 of RMIEP, Volume 9.) No further mention of seismic/fire interactions is provide in either the RMIEP fire analysis or in the IPE/IPEEE submittal itself.

2.2.6 Adequacy of Fire Barriers (FRSS)

Barrier reliability and inter-compartment fire effects are related to the potential that fires in one area might impact other adjacent or connected areas through the spread of heat and smoke. In general, it is expected that a utility analysis would provide for some treatment of such potential by considering that (1) manual fire fighting activities might allow for the spread of smoke and heat through the opening of access doors, and (2) that the failure of active fire barrier elements such as normally open doors, water curtains, and ventilation dampers might compromise barrier integrity. Resolution of the fire barrier issue is to verify that fire barriers are properly installed and maintained under a surveillance program.

The LaSalle fire analysis did consider the potential impact of barrier failures on fire risk. In particular, a multi-area screening analysis included consideration of potential barrier failures during a fire. A conditional probability of barrier failure of 0.1 was assumed for each barrier. All area combinations were screened on low CDF contribution.

The RMIEP report does note that two of the areas analyzed in detail had "protective wrappings" provided for some cables. In one case, Thermo-Lag is cited by name (see Table 3.10 on page 3-65 of ref. 6). However, there is no discussion of how these barriers were treated in the analysis.

2.2.7 Effects of Hydrogen Line Ruptures (MSRP)

The use of flammable gases in the plant, including hydrogen, introduces the potential that a rupture of the gas flow lines might lead to the introduction of a serious fire hazard into plant safety areas. It had been anticipated that a typical fire IPEEE analysis would include the consideration of such sources in the analysis.

For the LaSalle fire analysis the generic fire event data base used to develop fire frequencies did include flammable gas fires. However, neither the IPE/IPEEE submittal nor the RMIEP report provides any specific discussion related to the use or routing of flammable gases within the plant. No specific discussion of the treatment of potential flammable gas fires is provided and it would

appear that no specific consideration was given to these sources beyond their inclusion in the generic fire frequency estimates.

2.2.8 Common Cause Failures Related to Human Errors (MSRP)

Common cause failures resulting from human errors include operator acts of omission or commission that could be initiating events or could affect redundant safety-related trains needed to mitigate other initiating events. It had been anticipated that a typical fire IPEEE analysis would include the consideration of such failures in the submittal.

There is no explicit discussion of this issue in the submittal; however, the analysis has, in fact, considered potential common cause factors associated with human errors in the context of the post-fire recovery analysis. The LaSalle risk study included an extensive human reliability analysis (HRA). Further, the internal events HRA models were modified to suit the needs of the fire analysis and to reflect the potential impact of fires on operator actions. A specific analysis of recovery of ECCS from the remote shutdown panel after a MCR fire leading to abandonment was performed.

2.2.9 Non-safety Related Control System/Safety Related Protection System Dependencies (MSRP)

Multiple failures in non-safety-related control systems may have an adverse impact on safety-related protection systems as a result of potential unrecognized dependencies between control and protection systems. The licensee's IPE process should provide a framework for systematic evaluation of interdependence between safety-related and non-safety related systems and identify potential sources of vulnerabilities. It had been anticipated that the fire IPEEE analysis would include the consideration of such dependencies in the submittal.

The LaSalle fire analysis does not provide an explicit discussion of this issue. The event tree used in the fire analysis did include consideration of special initiators, that is, failures in support systems that lead to a plant trip or shutdown simultaneous with the degradation of mitigating systems. The analysis also included an extensive analysis of control systems interactions in the event of a MCR fire. Finally, the system model explicitly considered the potential spurious operation of components due to fire and assumed failures occurred in the most detrimental mode for a given sequence during quantification. The only exception to this is that spurious operation of the SRVs was apparently not considered.

2.2.10 Effects of Flooding and/or Moisture Intrusion on Non-Safety- and Safety-Related Equipment (MSRP)

Flooding and water intrusion events can affect safety-related equipment either directly or indirectly through flooding or moisture intrusion of multiple trains of non-safety-related equipment. This type of event can result from external flooding events, tank and pipe ruptures,

actuations of the fire suppression system, or backflow through part of the plant drainage system. It had been anticipated that the fire IPEEE analysis would include the consideration of such events in the submittal.

There is no explicit discussion of this issue in the context of the LaSalle fire analysis. The analysis did not explicitly consider fire suppression system damage, but assumed that such damage was subsumed in the screening analysis by virtue of the assumption of the loss of all equipment in an area regardless of cause, and by virtue of similar assumptions made in the multi-room screening analysis. A separate flood analysis did consider potential flooding due to failures in the fire water supply system.

2.2.11 Shutdown Systems and Electrical Instrumentation and Control Features (SEP)

The issue of shutdown systems addresses the capacity of plants to ensure reliable shutdown using safety-grade equipment. The issue of electrical instrumentation and control addresses the functional capabilities of electrical instrumentation and control features of systems required for safe shutdown, including support systems. These systems should be designed, fabricated, installed, and tested to quality standards and remain functional following external events. It had been anticipated that the fire IPEEE analysis would include the consideration of this issue in the submittal.

LaSalle is not an SEP plant. There is no explicit discussion of this issue in the submittal.

2.2.12 Multi-Unit Risk Implications

The LaSalle analysis is based on the analysis of Unit 2 only. No discussion of potential multi-unit risk implications is provided.

2.3 Containment Performance Issues Unique to Fire Scenarios

No unique containment failure modes were identified as a part of the fire analysis. The containment structures for both Units 1 and 2 were identified in the initial list of fire areas but are apparently screened. While no explicit discussion of the basis for screening is provided, they were most likely screened because containment is inerted with nitrogen during power operations.

2.4 Plant Vulnerabilities and Improvements

The LaSalle fire assessment concludes that there are no fire vulnerabilities. The licensee states that "(t)he RMIEP results were well within the safety goals established by the Nuclear Regulatory Commission (NRC)", an apparent reference to the USNRC subsidiary safety goals. Hence, it would appear that the licensee's definition of a vulnerability was based on the USNRC subsidiary safety goals.

No specific plant improvements resulting from the fire analysis are cited in the IPE/IPEEE submittal. However, the plant does note that the analysis represents the state of the plant as "frozen" in 1985 and that changes have been made to the plant that are not reflected in the analysis. It also acknowledges that the RMIEP summary documents note "several technical concerns." These were to 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." The fire analysis report did identify two potential areas for plant improvement: (1) to put tops on the MCR electrical panels to reduce the potential for spread of fire to the overhead cables, and (2) to institute a program to inspect the penetration seals at the top of the switchgear panels to minimize the potential that switchgear fires might damage the overhead cables. No discussion of these specific recommendations is provided in the IPE/IPEEE submittal.

3.0 FINDINGS AND RECOMMENDATIONS

The LaSalle IPEEE fire analysis is based on the USNRC-sponsored RMIEP study. The fire analysis in RMIEP was a full-scope fire PRA performed using state-of-the-art analysis methods, data, and assumptions. The fire analysis process was clearly capable of identifying potential plant vulnerabilities and is fully consistent with the USNRC guidance for performance of an IPEEE fire analysis. The IPEEE submittal concluded that there were no plant vulnerabilities, and no specific plant modifications resulting from the fire analysis were cited.

Some areas of strengths and weaknesses were identified in the analysis. They include:

Strengths:

- The plant systems models used in the fire analysis were derived directly from the full-scope internal events PRA.

- Extensive cable tracing was performed for both the Appendix R and credited non-Appendix R systems. The level of cable tracing performed for LaSalle is well in excess of typical practice for the IPEEEs.

- The systems models included consideration of "all potential spurious operations" for all modeled components (apparently excluding the SRVs). This was done, in large part, specifically to support the fire analysis. It would appear that the fire analysis has assumed worse case failure mode, including potential spurious operations, for all fire-damaged systems in the risk quantification. This also apparently extends to random (non-fire induced) failures as well.

- The analysis included a detailed fire detection and suppression analysis. The assessment was performed specific to each fire area and fire scenario analyzed. Further, the assessment included specific consideration of (a) damage and suppression timing factors, (b) uncertainty in the response times for the fire brigade, and (c) walk-down based assessments of the effectiveness of fixed detection and suppression system.

- A substantial uncertainty analysis was included in the assessment for selected key factors including fire size, fire location, fire suppression timing, and certain key fire model input parameters.

- The available documentation is quite extensive assuming that one includes the underlying RMIEP fire risk analysis NUREG/CR report (in particular, Volumes 3, 6, and 9).

- State-of-the-art PRA models available at the time of the analysis (1985-1990) were used to evaluate the conditional safe shutdown failure probability for each unscreened fire area.

- Through a cooperative effort between RMIEP and the FRSS, a considerable assessment of potential Control Systems Interaction issues has been performed for LaSalle. In this regard, the available information for LaSalle is well in excess of that available from a typical IPEEE analysis. In particular, the study assessed the potential impact of multiple spurious actuations in the event of a main control room fire, and the reliability and independence of the remote shutdown panel.

Major Weaknesses:

There were no major weaknesses identified in this review.

Minor Weaknesses:

- The potential contribution of seismic/fire interactions has not been considered explicitly in the fire analysis nor in the seismic analysis. The potential contribution of this issue is qualitatively judged in the RMIEP study to be "probabilistically insignificant" but no explicit justification for this conclusion is provided. No supplemental discussion of the issue is provided in the IPEEE submittal itself. Given the relatively low seismicity of the area surrounding the plant, this is not considered a major weakness of the study.

- The fire modeling analyses utilized a piloted ignition temperature for cables that, while it did reflect the state of knowledge at the time of the RMIEP analysis, is non-conservative in comparison to more recent test results (e.g., 500°C versus 350-400°C). However, because the fire modeling results did predict fire propagation in most of the scenarios that contribute to plant fire risk, the potential impact of reduced cable ignition temperatures would be minimal.

- The licensee cites that the risk study reflects the condition of the plant as frozen in 1985. It is cited that plant modifications have been made since that time. These modifications have not been identified, and no attempts were made to update the PRA to incorporate the changes.

- In the IPE/IPEEE submittal itself, the licensee has taken some exception to the results of the RMIEP fire risk study. In particular, the licensee cites that substantial conservatism exists in the fire event frequency estimates (as much as a factor of 10), and in the analysis of MCR abandonment probabilities. However, no substantiation of these assertions is provided, the licensee does not identify the specific fire scenarios impacted by the fire frequency concerns, and no alternate estimates of the actual fire risk are provided.

- It would appear that the licensee has not performed any confirmatory walk-downs nor risk quantification updates as a part of the IPEEE process.

- While the systems models did include spurious operations, it is not clear whether or not fire analysis considered the possible spurious operation of the SRVs because no LOCA scenarios were specifically modeled.

- The fire analysis is based on the analysis of Unit 2. There is no explicit discussion of the applicability of the study to Unit 1 (i.e., consideration of unit differences that might impact the risk quantification).

- There was no specific discussion of potential multi-unit fire risk implications (e.g., reliance on shared equipment or systems and/or the need to perform a simultaneous multi-unit shutdown due to fires in shared areas or involving shared equipment).

Overall, the author finds that the LaSalle IPEEE fire analysis does meet the objectives of the IPEEE process. Given the lack of any identified major weaknesses, no further review or interactions with the licensee are recommended.

4.0 REFERENCES

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