LaSalle County Nuclear Power Station

Individual Plant Examination Insight Support Report for NUREG-1150 Plants

NRC JCN W6187, Subtask 6-1

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Acronyms

ADS	Automatic depressurization system
BWR	Boiling water reactor
CDF	Core damage frequency
CDS	Condensate system
CECo	Commonwealth Edison Company
CPI	Containment performance improvement
CRD	Control rod drive
CS	Containment spray
DFWS	Diesel-driven fire water system
DHR	Decay heat removal
ECCS	Emergency core cooling system
GSIs	Generic safety issues
HEPs	Human error probabilities
HPCS	High pressure core spray
HRA	Human reliability analysis
IPE	Individual plant examination
LOCAs	Loss-of-coolant accidents
LPCI	Low-pressure coolant injection
LPCS	Low-pressure core spray
MFW	Main feedwater
NPSH	Net positive suction head
NRC	Nuclear Regulatory Commission
PDSG	Plant damage state groups
PDSs	Plant damage states
PRUEP	Phenomenology and Risk Uncertainty Evaluation Program
RCIC	Reactor core isolation cooling
RHR	Residual heat removal
RMIEP	Risk Methods Integration and Evaluation Program
RPV	Reactor pressure vessel
rx	reactor
SDC	Shutdown cooling
SNL	Sandia National Laboratories
SPC	Suppression pool cooling
SRVs	Safety relief valves
USIs	Unresolved safety issues

E. EXECUTIVE SUMMARY

This report describes the Sandia National Laboratories (SNL) review of the LaSalle County Nuclear Power Station Individual Plant Examination (IPE) submittal. Because the IPE submittal being reviewed is for a plant that has already undergone a Level 1 and Level 2/3 analysis by the Nuclear Regulatory Commission (NRC) in the Risk Methods Integration and Evaluation Program (RMIEP) and in the Phenomenology and Risk Uncertainty Evaluation Program (PRUEP), both of which have received technical review, and because the submittal *is* the results from the RMIEP and PRUEP studies, this review concentrated on (1) "the utility's certification that the PRA meets the intent of the generic letter, in particular with respect to utility staff involvement," and (2) certification that the PRA "reflects the current plant design and operation" [p. 8 GL 88-20]. This review also considers the technical concerns identified during the utility review of the RMIEP (Level 1) and PRUEP (Level 2/3) results for applicability to the current design and operation of the LaSalle County Nuclear Power Station. With these objectives in mind, the reader should understand that the material presented in this review will differ from other Technical Evaluation Reports performed for other plant submittals.

The purpose of this report is to summarize SNL's review of the LaSalle IPE submittal, including the technical concerns identified by the utility as it reviewed the RMIEP and PRUEP results for applicability to the current design and operation of the LaSalle County Nuclear Power Station, and to present selected results from the IPE submittal (i.e., the RMIEP and PRUEP analyses). This summarization is based on information contained in the IPE submittal [IPE Submittal] and the detailed documentation of the RMIEP (front-end) [NUREG/CR-4832] and PRUEP (back-end) [NUREG/CR-5305] analyses of the LaSalle Unit 2 Nuclear Power Plant.

E.1 Plant Characterization

The LaSalle Unit 2 nuclear power plant is a General Electric boiling water reactor (BWR) 5 rated at 3293 MWt and 1078 MWe which is housed in a Mark II containment. It is owned and operated by the Commonwealth Edison Company (CECo) and is located in LaSalle County, Illinois, about 55 miles southwest of Chicago.

The primary containment is a post-tensioned reinforced concrete structure with a steel liner. It is enclosed by a reinforced concrete reactor building which forms the secondary containment. During power operation, the primary containment, which has a design pressure of 45 psig, is inerted with nitrogen [pp. 1-2 through 1-6 of NUREG/CR-4832, Vol.1].

Important Design Characteristics

High-pressure injection is provided by the following four systems:

- High-pressure core spray (HPCS),
- Reactor core isolation cooling (RCIC),
- Main feedwater (MFW), and
- Control rod drive (CRD).

To use the low-pressure injection systems, reactor vessel pressure must be reduced. This can be accomplished by 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),
- Low-pressure core spray (LPCS),
- Low-pressure coolant injection (LPCI), and
- Diesel-driven firewater system (DFWS).

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 (SDC)—can be used for nonloss-of-coolant accidents (LOCAs). In addition, the containment can also be vented through the containment vent and purge system. Venting can be from either the drywell or the suppression chamber using either a 2-inch valve or a 26-inch valve. The vent pipe ties into the standby gas treatment system (SGTS), which releases the gases to the stack. The vent pipe is attached via an 18-inch pipe to the SGTS with a rubber boot 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, which is divided into two regions: the upper cavity and the lower cavity. The upper cavity volume is large enough to hold all of the debris that would be released should the vessel fail.

E.2 Licensee IPE Process

The submittal states that the IPE "is the result of a detailed review of the NRC's Risk Methods Integration Evaluation Program (RMIEP) ... analysis" [p. ES-1 of submittal].[1]

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[1]Generally, the IPE submittal uses RMIEP to refer to both the Level 1 (RMIEP) and Level 2/3 (PRUEP) analyses performed by the NRC. In this report, whenever possible, a distinction between the two analyses will be made so that the reader will clearly understand from which analysis information was taken.

Thus, ultimately the methodology used in the IPE is the methodology used by RMIEP—the small event tree, large fault tree methodology. Furthermore, the submittal states that "the objectives of Generic Letter 88-20 have been accomplished for both internal and external events through this review process" [p. ES-1 of submittal].

The review process used by Commonwealth Edison to address the main objectives of Generic Letter 83-20 is as follows [pp. 5, 6 of submittal]:

- Gain an appreciation of the behavior of the plant under severe accident conditions by:
  - Reviewing the physical layout of the plant,
  - Reviewing the procedures in use at the plant,
  - Examining other BWR/5, Mark II IPE submittals,
  - Assessing each of the severe accident phenomena listed in NUREG-1335 [NUREG-1335] for applicability to LaSalle,
  - Making limited use of the Modular Accident Analysis Program (MAAP) code to obtain a best-estimate characterization of accident sequence progressions, and
  - Reviewing the RMIEP representations of LaSalle severe accident behavior.
- Gain a more quantitative understanding of the overall probabilities of core damage and release of fission products by:
  - Reviewing and analyzing the dominant sequences (top 95% of core damage frequency) and key basic events from the internal events analysis of RMIEP. This was accomplished by:
    - Identifying and developing a functional understanding and description of the dominant sequences,
    - Identifying and developing a description of the propagation of the accident sequence by identifying the initiating events and subsequent system failures,
    - Analyzing the dominant cut sets for each sequence by identifying the key component failures, and
    - Understanding the common cause and operator action treatment.

- Identify and evaluate observations/insights regarding the station configuration or practices which may affect the risk profile of the plant by examining:

  - The RMIEP study,

  - Plant information in conjunction with the RMIEP review,

  - Specific analyses of similar plants (e.g., Nine Mile Point 2 and WNP2), and

  - Previous Commonwealth Edison IPEs.

- Identify and document technical issues which will be addressed in a future update of the LaSalle IPE.

E.3 IPE Analysis

E.3.1 Front-End Analysis

Section E.3.1.1 summarizes the technical concerns resulting from the CECo review of the RMIEP front-end analysis as presented in Section 2.2 of this report. Section E.3.1.2 summarizes the results from the front-end analysis.

E.3.1.1 Technical Concerns

Technical concerns resulting from the CECo review of the RMIEP front-end analysis are summarized in this section. A more complete description of the concerns along with any SNL review comment pertaining to the concerns is provided in Section 2.2 of this report.

Concerns identified during the CECo review include:

- Common cause analysis not specific for LaSalle

- Beta factor common cause analysis is too conservative; the analysis should be more realistic

- Main feedwater as a viable injection source in large and medium LOCAs is expected to be nonconservative

- Emergency core cooling system (ECCS) pumps unavailable due to low net posititve suction head (NPSH) expected to be conservative

- ECCS pumps unavailable due to containment failure too conservative

- Assumption that rubber boot fails whenever venting occurs

- Accident-mitigative systems would most probably be unavailable due to the resultant (steam) environment

- Plant-specific data should be used instead of generic data

- Plant configuration used in the RMIEP analysis has changed due to plant modifications

- Dual unit initiating events and unit-to-unit system differences outside RMIEP scope

E.3.1.2 Results

The estimate of the overall core damage frequency (CDF) from the RMIEP analysis for internal events (excluding internal floods) is as follows:

point estimate	3.11E-5/yr
mean	4.41E-5/yr
5th percentile	2.05E-6/yr
median	1.64E-5/yr
95th percentile	1.39E-4/yr

The estimate of the CDF from the RMIEP analysis for internal floods is given as follows:

point estimate	3.23E-6/yr
mean	3.39E-6/yr
5th percentile	9.62E-8/yr
median	1.13E-6/yr
95th percentile	3.23E-6/yr

The dominant initiating events contributing to the CDF point estimate from the RMIEP analysis are as follows:

Loss of offsite power	74.3%
Loss of division 1 4160 VAC bus	8.1%
Transient with turbine bypass	5.8%
Loss of division 1 125 VDC bus	5.2%
Transient with total loss of feedwater	2.9%

Dominant hardware failures contributing to the CDF from the RMIEP analysis are as follows:

Hardware Failures

Common cause diesel generator cooling failure
Diesel generator fails to start
Relay failures
Equipment survivability given harsh environment
Breaker failures
Containment failure results in leakage to reactor building

The dominant accident classes contributing to the CDF point estimate from the RMIEP analysis are as follows:

T100	- Transients with failure of all high- and low-pressure systems	64.1%
T62	- Transients with failure of all high-pressure systems except RCIC, failure of heat removal, and failure of low-pressure systems	14.6%
T18	- Transient with HPCS and one train of CRD working, heat removal fails, venting fails, containment pressure increases till a leak occurs, location of leak determines environment to which equipment is subjected, and equipment fails to survive harsh environment	11.1%
T20	- Transient with HPCS and one train of CRD working, heat removal fails, venting fails, containment pressure increases till rupture occurs, location of rupture determines environment to which equipment is subjected, and equipment fails to survive harsh environment	2.9%
T22	- Transient with HPCS working, heat removal fails, venting fails, containment pressure increases till a leak occurs, location of leak determines environment to which equipment is subjected, and equipment fails to survive harsh environment	2.5%

The design characteristics from the RMIEP analysis important to the CDF are as follows:

RCIC "sneak circuit" [2] - could cause RCIC to be unavailable under certain plant conditions.

Rubber boot in vent path - failure results in harsh environment, affecting ability of the components to function properly.

E.3.2 Human Reliability Analysis

Section E.3.2.1 summarizes the technical concerns resulting from the CECo review of the RMIEP human reliability analysis (HRA) as presented in Section 2.3 of this report. Section E.3.2.2 summarizes the results from the HRA analysis.

E.3.2.1 Technical Concerns

Technical concerns resulting from the CECo review of the RMIEP analysis are summarized in this section. A more complete description of the concerns along with any SNL review comment pertaining to the concerns is provided in Section 2.3 of this report.

Concerns identified during the CECo review include:

- RMIEP HRA results expected to be nonconservative
- Plant model should only consider proceduralized actions
- HRA should reflect plant procedures in effect at the LaSalle County Station

E.3.2.2 Results

Important operator actions/errors identified in the RMIEP analysis include the following:

Failure to restore offsite power in 1 hour
Operator fails to reopen RCIC F063 valve
Failure to repair diesel generator failure in 1 hour
Failure to restore offsite power in 10 hours
Failure to repair diesel generator failure in 2 hours
Failure to restore offsite power in 8 hours

Human-performance-related enhancements resulting from the RMIEP analysis include the following:

Changes have been made to LaSalle Procedure LOA-AP-07 "Loss of Auxiliary Electrical Power," which identifies the sequence of events resulting in

[2] See page 26 of the IPE submittal for a more complete description of the RCIC "sneak circuit."

nonrecoverable isolation of RCIC. In addition, operator training during every training cycle targets this concern.

E.3.3 Back-End Analysis

Section E.3.3.1 summarizes the technical concerns resulting from the CECo review of the PRUEP Level 2 analysis as presented in Section 2.4 of this report. Section E.3.3.2 summarizes the results from the Level 2 analysis.

E.3.3.1 Technical Concerns

Technical concerns resulting from the CECo review of the PRUEP analysis are summarized in this section. A more complete description of the concerns along with any SNL review comment pertaining to the concerns is provided in Section 2.4 of this report.

Concerns identified during the CECo review include:

- Containment failure pressure too high
- Failure of ECCS caused by severe environments in the reactor building is overly conservative
- Containment failure due to phenomena associated with reactor pressure vessel (RPV) too high
- Alpha mode (steam explosion) failure probability too high
- No credit given for fission product retention in the drywell

E.3.3.1 Results

For its Level 2 analysis, the IPE incorporated the Level 2 analysis performed in the PRUEP study [NUREG/CR-5305, Vol.s 1,2,3]. In the PRUEP study, cut sets from the Level 1 sequences were grouped into 30 plant damage states (PDSs). Of these 30 PDSs, 22 represented accidents initiated from internal events and internal floods, with the remaining 8 for seismic and fire-induced accidents. Conditional mean containment failure probability, given core damage, is as follows:

Containment Failure Location[3]

Drywell	22%
Wetwell	17%
Vent	49%
Bypass	0%
Intact	12%

Containment Failure Times[3]

Early	13%
Early Vent	5%
Intermediate	17%
Late	9%
Late Vent	44%
Bypass	0%
Intact	12%

E.4 Generic Issues and Containment Performance Improvements

As part of the IPE submittal, a table[4] documenting a comparison between the NUREG-1335 guidelines and the LaSalle County Station RMIEP IPE Report was provided. Page 4 of this table states that a thorough discussion of decay heat removal (DHR) can be found in Section 3.2 of NUREG/CR-5305, Volume 1. The following material has been extracted from the discussions on containment venting and containment heat removal.

For long-term containment heat removal accidents and ATWS sequences, decay heat can be removed by venting the containment. The use of a rubber boot connecting the vent pipe to the standby gas treatment system results in steam being released into the reactor building rather than being directed to the stack when the containment is vented. This high-temperature steam creates a severe environment for components located in the reactor building. This severe environment can affect the ability of the systems to perform their functions. The degree of this impact will depend on the environment produced and the qualification of the equipment subjected to the severe environment. Containment venting requires both divisions of ac power.

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[3]Includes accidents initiated by traditional internal events and internal floods. Failure times are defined as follows: early - prior to or during core damage, intermediate - around the time of vessel failure, and late - after vessel failure.

[4]Hereafter the table is referred to as the NUREG-1335/LaSalle IPE Comparison Table.

Heat can also be removed from the containment by the RHR system during four modes of operation. These are

- Suppression pool cooling,
- Containment spray,
- Shutdown cooling, and
- Low-pressure coolant injection.

Each RHR train (there are two) can function as long as it has ac power and its heat exchanger is being cooled.

The above discussions describe major dependencies within the decay heat removal systems. However, the IPE submittal does not contain any specific criteria for resolution of the DHR vulnerability issue.

Page 4 of the NUREG-1335/LaSalle IPE Comparison Table also states that no other unresolved safety issues (USIs) or generic safety issues (GSIs) were evaluated.

Based on a telephone conversation between the NRC and the utility on December 12, 1995, the following information is provided as it relates to the CPI issues:

(1) Alternate water supply for drywell spray/vessel injection

Provision for and procedures exist to use the firewater system as an alternate water supply for vessel injection. Use of the firewater system as an alternate supply for drywell spray is under consideration. Such use would require a spool piece which is not available at this time.

(2) Enhanced reactor pressure vessel depressurization system reliability

The EOPs provide several different means of vessel depressurization. Examples include 1) turbine bypass valves, 2) turbine-driven reactor feedwater pump, and 3) the RCIC steam line.

(3) Emergency procedures and training

Revision 4 of the BWR Owners Group Emergency Procedure Guidelines have been implemented.

(4) Containment heat removal—Hardened vent

A hardened vent path does exist. It comes off of the normal containment and goes to the reactor building return air riser to two blowout panels—one of which is located at the roof of the auxiliary building, the other at the steam tunnel to the turbine building.

E.5 Vulnerabilities and Plant Improvements

Page 4 of the NUREG-1335/LaSalle IPE Comparison Table described in Section E.4 of this report states that Section 7.4 of NUREG/CR-4832, Vol. 3, Part 1 lists vulnerabilities. The following vulnerabilities were identified from an examination of the section:

- A sneak circuit in the RCIC isolation logic that results in the RCIC steam line inboard isolation valve closing when offsite ac power is lost and the appropriate diesel generator starts.
- RCIC room temperature isolation logic, in cases where train A ac power has failed but train B ac power is available, isolates if no other emergency core cooling system is working.
- Venting using current procedures results in severe environments in the reactor building.

No definition of vulnerability was given in the referenced IPE submittal section.

Page 7 of the NUREG-1335/LaSalle IPE Comparison Table states that Section 7.4 of NUREG/CR-4832, Vol. 3, Part 1 identifies potential improvements implemented or selected for implementation. The following potential improvements were identified from an examination of this section:

- Eliminate the sneak circuit in the RCIC isolation logic that results in the RCIC steam line inboard isolation valve closing when offsite ac power is lost and the appropriate diesel generator starts.
- Change the RCIC room temperature isolation logic so that, in cases where train A ac power has failed but train B ac power is available, RCIC does not isolate if no other emergency core cooling system is working.
- Change the venting procedure so that venting does not result in severe environments in the reactor building.

The only improvement the submittal makes reference to is the sneak circuit. The improvement implemented by the utility is one of procedure modification and operator training, rather than elimination of the sneak circuit by hardware modifications.

In addition, the IPE submittal states that 137 IPE insights were identified during the IPE [p. 35 of submittal] and that 81 accident management insights were identified by a review of the accident management insights generated during the Dresden and Quad Cities IPE effort [pp. 35, 36 of submittal]. However, the submittal does not provide a list of these insights or their disposition. Thus, there is no way to tell if improvements were identified from these insights.

E.6 Observations

The basis of the submittal is a review of the NRC-sponsored RMIEP and PRUEP analyses by utility personnel. As part of this review, 218 LaSalle-related IPE insights were identified [p. ES-3 of submittal]. These insights were not provided as part of the submittal nor were they discussed. Thus, there is no way to judge the importance of these insights.

No detailed discussions were provided by the submittal regarding vulnerabilities or other concerns such as DHR. However, the submittal did address what was classified as technical concerns. These are discussed in detail in Sections 2.2, 2.3, and 2.4 of this report.

The RMIEP/PRUEP analyses are based on plant information as of the mid-1980s. The utility states that it plans to update these NRC studies. This is encouraged as it will provide the utility with an up-to-date representation of LaSalle, both from a hardware and operations point of view.

The submittal documentation made extensive reference to the RMIEP and PRUEP documentation. Little effort was made to extract the appropriate RMIEP/PRUEP information and integrate it into the submittal.

1. INTRODUCTION

1.1 Review Process

This report describes the Sandia National Laboratories (SNL) review of the LaSalle County Nuclear Power Station Individual Plant Examination (IPE) submittal. Because the IPE submittal being reviewed is for a plant that has already undergone a Level 1 and Level 2/3 analysis by the Nuclear Regulatory Commission (NRC) in the Risk Methods Integration and Evaluation Program (RMIEP) and in the Phenomenology and Risk Uncertainty Evaluation Program (PRUEP), both of which have received technical review, and because the submittal *is* the results from the RMIEP and PRUEP studies, this review concentrated on (1) "the utility's certification that the PRA meets the intent of the generic letter, in particular with respect to utility staff involvement," and (2) certification that the PRA "reflects the current plant design and operation" [p. 8 GL 88-20]. This review also considers the technical concerns identified during the utility review of the RMIEP (Level 1) and PRUEP (Level 2/3) results for applicability to the current design and operation of the LaSalle County Nuclear Power Station. With these objectives in mind, the reader should understand that the material presented in this review will differ from other Technical Evaluation Reports performed for other plant submittals.

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1.2 Plant Characterization

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

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

Important Design Characteristics

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.

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 meltthrough 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.

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2. TECHNICAL REVIEW

2.1 Licensee IPE Process

The following three sections describe the process used by the licensee with respect to: completeness and methodology; multiunit effects and as-built, as-operated status; and licensee participation and peer review.

2.1.1 Completeness and Methodology

The submittal states that the IPE "is the result of a detailed review of the NRC's Risk Methods Integration Evaluation Program (RMIEP) ... analysis" [p. ES-1 of submittal].[5] Thus, ultimately the methodology used in the IPE is the methodology used by RMIEP and PRUEP. Furthermore, the submittal states that "the objectives of Generic Letter 88-20 have been accomplished for both internal and external events through this review process" [p. ES-1 of submittal].

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- Gain an appreciation of the behavior of the plant under severe accident conditions by:
  - Reviewing the physical layout of the plant,
  - Reviewing the procedures in use at the plant,
  - Examining other BWR/5, Mark II IPE submittals,
  - Assessing each of the severe accident phenomena listed in NUREG-1335 [NUREG-1335] for applicability to LaSalle,
  - Making limited use of the Modular Accident Analysis Program (MAAP) code to obtain a best-estimate characterization of accident sequence progressions, and
  - Reviewing the RMIEP representations of LaSalle severe accident behavior.

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[5] In the submittal, reference to RMIEP can be either to the RMIEP study or to the PRUEP Study. In this report reference to the appropriate study is made.

- Gain a more quantitative understanding of the overall probabilities of core damage and release of fission products by:

  – Reviewing and analyzing the dominant sequences (top 95% of core damage frequency) and key basic events from the internal events analysis of RMIEP. This was accomplished by:

    - Identifying and developing a functional understanding and description of the dominant sequences,

    - Identifying and developing a description of the propagation of the accident sequence by identifying the initiating events and subsequent system failures,

    - Analyzing the dominant cut sets for each sequence by identifying the key component failures, and

    - Understanding the common cause and operator action treatment.

- Identify and evaluate observations/insights regarding the station configuration or practices which may affect the risk profile of the plant by examining:

  – The RMIEP study,

  – Plant information in conjunction with the RMIEP review,

  – Specific analyses of similar plants (e.g., Nine Mile Point 2 and WNP2), and

  – Previous Commonwealth Edison IPEs.

- Identify and document technical issues which will be addressed in a future update of the LaSalle IPE.

2.1.2 Multiunit Effects and As-Built, As-Operated Status

While RMIEP only examined Unit 2 at the LaSalle County Station, events that occurred at Unit 1 were considered if the events could affect the operation of Unit 2 (e.g., loss of offsite power). However, RMIEP/PRUEP did not quantify the risk associated with the occurrence of the event at Unit 1 since such quantification was beyond the work scope of the projects. Thus, at least one dual-unit initiator exists—dual-unit loss of offsite power—which should be examined. In the update to the IPE, to be performed at a later

date, "a detailed examination for unit-to-unit system differences and other events that could potentially be simultaneous initiators in both units" will be conducted [pp. 15, 16 of submittal].

The models used in the RMIEP analysis represent the plant as it existed in 1985. Since then, plant modifications have occurred. The "impact of these modifications on the RMIEP plant model has not been quantified." However a "top level review of these modifications was performed ... and no modifications which would have a significant, adverse impact on the LaSalle risk profile were identified." The impact of these modifications will be assessed in the updated IPE [p. 15 of submittal].

2.1.3 Licensee Participation and Peer Review

As stated in Section 2.1.1, the submittal is the result of a review of the RMIEP analysis. This review was performed by an organization created by Commonwealth Edison, making use of its internal personnel resources. Collectively, the personnel assigned to the review project "have extensive experience in plant operations and systems engineering, as well as probabilistic risk assessment (PRA) experience." Commonwealth Edison personnel "performed the basic modeling review and analysis, as well as the Level II review and analysis using a CECo-specific version of the Modular Accident Analysis Program (MAAP) code" [p. 1 of submittal].

The submittal also states that "no separate 'independent review' of the LaSalle County Station IPE ... was performed under CECo auspices" [p. 1 of submittal].

2.2 Utility-Identified Front-End Technical Concerns

As stated in the submittal [p. 12] "Analysis of the RMIEP dominant accident sequences revealed that there are several technical issues which significantly impact the results of the top contributors to core damage." These technical concerns are listed below; where appropriate, an SNL response or discussion of the concern is also provided.

Concern: **Common cause analysis not specific for LaSalle**

The submittal states that a "generic common cause failure database was used in the development of the common cause factors for the RMIEP analysis...." The submittal further states that "more recent information is available, for example the June 1992 EPRI report entitled 'A Database of Common Cause Events for Risk and Reliability Evaluations,' (EPRI TR-100382)." The submittal goes on to say that "In the future update of the LaSalle analysis, and consistent with the other CECo IPE analyses, a LaSalle-specific screening will be conducted and LaSalle specific common cause factors will be developed" [pp. 13, 14 of submittal].

Response: As for the first staterr , it is true. However, generally speaking, plant-specific data are not sufficient to estimate all types of common cause failures, generally because common cause failures are more rare than individual-component failures. For the second statement, while use of more up-to-date information is encouraged, the methodology to be used to arrive at these factors specific to LaSalle is not presented; thus no additional comments can be made.

Concern: **Beta factor common cause analysis is too conservative; the analysis should be more realistic**
The submittal states that "The common cause factors currently used in the RMIEP study are recognized to be too conservative." The basis for this statement is a letter to the NRC dated August 7, 1991 the subject of which was "Summary of Meeting with Commonwealth Edison to Discuss the Results of the NRC Sponsored Probabilistic Risk Assessment of LaSalle Station, Unit 2." The submittal further states that "It is expected that re-analysis, consistent with the other CECo IPEs, would reduce common cause factors for some key components" [p. 13 of submittal].

Response: Given that the letter was not available for review and the submittal does not describe the analysis process to be used in the estimation of common cause factors, no additional comments can be made.

Concern: **Main feedwater (MFW) as a viable injection source in large and medium LOCAs expected to be nonconservative**

Response: Use of the MFW system in conjunction with the condensate transfer system as an injection source for large and medium LOCAs was considered a success in the RMIEP analysis [p. 2-24 of NUREG/CR-4832 Vol. 4]. If the future update to the existing IPE indicates that operation of the MFW system is not successful in preventing core damage, then it should be removed from the appropriate LOCA trees.

Concern: **ECCS pumps unavailable due to low NPSH expected to be too conservative**
The submittal states that the RMIEP analysis "assumes that the ... pumps fail when the containment fails, presumably also due to loss of net positive suction head..." [p. 14 of submittal].

Response: RMIEP did not always assume that containment failure always caused failure of the ECCS pumps, or for that matter any of the components necessary for operation of the injection system. This issue was treated probabilistically. That is, probabilities were developed that represented the likelihood that the high steam and temperatures in the reactor building following containment failure or venting would fail electric motors, switching elements, and electrical terminals that were required for successful operation of the ECCS systems. Hence, there was a certain probability that the systems would fail; however, there was also a certain probability that the systems would not be adversely affected by the severe environment. This issue was analyzed using expert judgment techniques and was quantified during the NUREG-1150 study [NUREG/CR-4550, Vol. 2]. Thus, the actual failure mechanism for the pumps is due to a harsh environment, not loss of NPSH.

Concern: **ECCS pumps unavailable due to containment failure too Conservative**
The submittal identifies a concern that RMIEP assumed the ECCS pumps fail when the containment fails [p. 14 of submittal].

Response: RMIEP did not always assume that containment failure always caused failure of the ECCS pumps, or for that matter any of the components necessary for operation of the injection system. This issue was treated probabilistically. That is, probabilities were developed that represented the likelihood that the high steam and temperatures in the reactor building following containment failure or venting would cause electric motors, switching elements, and electrical terminals that were required for successful operation of the ECCS systems to fail. Hence, there was a certain probability that the systems would fail. However, there was also a certain probability that the systems would not be adversely affected by the severe environment and would continue to operate, thereby preventing core damage. This issue was analyzed using expert judgment techniques and was quantified during the NUREG-1150 study [NUREG/CR-4550, Vol. 2].

Concern: **Assumption that boot failure occurs whenever venting implemented**
The submittal accurately states that RMIEP assumed that "the rubber boot connecting the standby gas treatment ... system to the containment vents will fail if the 24" vent valves are opened." It further states that MAAP analyses performed for other IPE

analyses, i.e., Dresden and Quad Cities, indicate that the 2-inch vent lines are expected to prevent the pressure from increasing to the point where the 24-inch lines would be opened [p. 15 of submittal].

Response: The RMIEP analysis indicates that operators are directed to vent the containment by opening 2-inch vent lines when the containment pressure reaches 60 psig. It states that these two 2-inch lines cannot remove sufficient energy to prevent further pressurization and that the operators will be directed to vent using the 24-inch lines. Finally, it states that these two 24-inch lines connect via a common 18-inch line to the standby gas treatment system which contains a short section of ductwork and a rubber boot, both of which are virtually certain to fail if a 24-inch line is opened. Given the introduction of a relatively high-pressure environment to the ductwork and rubber boot, the assumed failure is not unwarranted.

Since there is a difference in the predicted venting results (prevention or nonprevention of pressurization increase when using the 2-inch vent lines), depending on which analysis is examined, it appears to be prudent for an analysis to clearly identify and document any uncertainties associated with the claim that the 2-inch lines can or cannot prevent pressurization to the point requiring the 24-inch lines to be opened.

Concern: **Accident-mitigative systems would most probably be unavailable due to the resultant environment**
The IPE submittal states that, given the location of the venting rupture, the ECCS "pumps would not be as likely to fail as a result of the environment as is indicated in the RMIEP analysis" [p. 15 of submittal].

Response: Again, this issue was treated probabilistically. That is, probabilities were developed that represented the likelihood that the high steam and temperatures in the reactor building following containment venting, as given by MELCOR calculations, would cause the failure of electric motors, switching elements, and electrical terminals that were required for successful operation of systems. Hence, there was a certain probability that the systems would fail. However, there was also a certain probability that the systems would not be adversely affected by the severe environment and would continue to operate, thereby preventing core damage. This issue was

analyzed using expert judgment techniques and was quantified during the NUREG-1150 study [NUREG/CR-4550, Vol. 2].

If additional information on the survivability of equipment and/or more detailed environmental assessment calculations are available, this information should be used and documented in any future IPE update.

Concern: **Plant-specific data should be used instead of generic data**
The IPE accurately states that "Generic data ... was used extensively ... for both component unavailabilities and for initiating event frequencies" in the RMIEP analysis. It also acknowledges that at the time the RMIEP analysis was performed very little plant-specific data were available.

Response: Use of plant-specific data during any future IPE update is encouraged.

Concern: **Plant configuration used in the RMIEP analysis has changed due to plant modification**
The submittal states that plant modifications have occurred since construction of the RMIEP system models was completed in 1985, and the impact of these changes has not been quantified [p. 15 of submittal]. A top-level impact review was performed without identifying any modifications which would have a significant, adverse impact on the LaSalle risk profile.

Response: The detailed assessment of the impact of these changes in any update to the LaSalle IPE is encouraged.

Concern: **Dual-unit initiating events and unit-to-unit system differences outside RMIEP scope**

Response: Identification and assessment of dual-unit initiating events, which was not part of the RMIEP analysis, is encouraged. Also, identifying unit-to-unit system differences should provide for a more complete understanding of the LaSalle County Station's response to accident sequences and is therefore encouraged.

2.3 Utility-Identified Human Reliability Analysis Technical Concerns

Technical concerns associated with the RMIEP human reliability analysis (HRA) are listed below; where appropriate, an SNL response or discussion of the concern is also provided.

Concern: **RMIEP HRA Results expected to be nonconservative**

The IPE states that a comparison of human actions found to be important in the Dresden and Quad Cities IPEs was made with similar actions in the RMIEP analysis, with the RMIEP human error probabilities (HEPs) generally being lower. Differences in the methods used to incorporate the use of plant procedures was sited as one possible reason for the differences in the HEP. The IPE also states that, "Recent critical reviews of LaSalle performance reinforce that the HEPs used at LaSalle should be higher than comparable HEPs at other CECo BWRs."—Dresden and Quad Cities [p. 13 of submittal].

Response: If additional information pertinent to the estimation of HEPs is available, it should be used regardless of the HEP estimation technique used.

Concern: **Plant model should only consider proceduralized actions**

Response: While it is recognized that considering only proceduralized actions is the most defensible position, it is unclear why this is called out as a concern with regard to the RMIEP analysis. The human actions considered in the RMIEP analysis [p. 5-11 of NUREG/CR-4832, Vol. 3, Part 1] are actions that would normally be expected to be called out in procedures. If this list contains nonproceduralized actions, then removal of such actions would be warranted if only proceduralized actions are to be considered.

Concern: **HRA should reflect plant procedures in effect at the LaSalleCounty station**

The RMIEP HRA analysis process did not model *all operator actions in the LaSalle EOPs as implemented during the progression of an accident*. The submittal states that techniques that allow this level of detailed modeling will be used in the update to the LaSalle IPE.

Response: This is encouraged.

2.4 Utility-Identified Back-End Technical Concerns

The Level 2 study performed as part of the PRUEP study [NUREG/CR-5305, Vols. 1,2,3] was incorporated into the LaSalle IPE. The submittal states, however, that the PRUEP study "uses a fundamentally different approach than Level II analyses performed and submitted in other CECo IPE studies." The submittal goes on to state

that "CECo BWR IPEs to-date have relied on bounding deterministic analyses to conclude whether or not a particular severe accident phenomenon will challenge containment integrity" and "...the bounding approach is favored and will be used in the update of the LaSalle PRA." In addition to the philosophical differences in approach to the Level 2 analysis, the submittal identified several specific technical concerns regarding the PRUEP Level 2 results. These technical concerns are listed below; where appropriate, an SNL response or discussion of the concern is also provided.

Concern: **Containment failure pressure**
The submittal states that "The mean containment failure pressure used in the RMIEP study (195 psig) appears to be too high and there is no accounting for containment strength degradation at high temperatures." The LaSalle IPE suggests, based on LaSalle-specific studies performed for CECo, that a failure pressure of 147 psig is more reasonable.

Response: The containment failure distribution used in the RMIEP/PRUEP studies was quantified using expert judgment techniques in conjunction with the structural expert elicitations that were performed as part of the NUREG-1150 study [NUREG/CR-5305, Vol. 2, Part 1]. The opinions of three different independent experts were elicited on this parameter. As part of this quantification process, the experts also provided distributions for containment failure at elevated temperatures (i.e., >500 °F). However, results from MELCOR calculations that were performed in support of the RMIEP/PRUEP studies [NUREG/CR-5305, Vol. 3] indicated that, within the time frame of interest, high drywell and wetwell structural temperatures (i.e., 800 °F to 1200 °F) were not a concern and, hence, the containment failure distributions that represented containment strength degradation caused by elevated temperatures were not used in the RMIEP/PRUEP studies. If new information indicates that structures are expected to be at elevated temperatures, then the use of containment failure pressure distributions that reflect containment performance at elevated temperatures is encouraged.

Concern: **Failure of ECCS caused by severe environments in the reactor building**
The submittal states that, "It is overly conservative to assume, as the RMIEP study does, that containment failure during the ATWS and LOCA sequences always causes a sufficiently harsh reactor building environment to fail all ECCS pumps."

Response: The RMIEP/PRUEP studies did not always assume that containment failure during the ATWS and LOCA sequences always caused failure of the ECCS pumps. This issue was treated probabilistically. That is, probabilities were developed that represented the likelihood that the high steam and temperatures in the reactor building following containment failure or venting would fail electric motors, switching elements, and electrical terminals that were required for successful operation of the ECCS systems. Hence, there was a certain probability that the systems would fail. However, there was also a certain probability that the systems would not be adversely affected by the severe environment. This issue was analyzed using expert judgment techniques and was quantified during the NUREG-1150 study [NUREG/CR-4550, Vol. 2].

Concern: **Containment failure due to phenomena associated with RPV failure**
The submittal states that the "probability of containment failure due to phenomena associated with RPV failure is too high." The submittal states that "bounding assessments performed for CECo BWRs suggest that pressure loads associated with ex-vessel steam explosions, direct containment heating or vessel blowdown will not challenge LaSalle containment integrity unless a very high containment pressure exists prior to vessel failure."

Response: Given that the calculations that were used to support these conclusions were not provided in the submittal nor were they discussed in any detail, no additional comments can be made.

Concern: **Alpha mode failures**
The submittal states that "the α-mode (steam explosion) mean failure probability is too high." The submittal justifies this conclusion by stating that, "It has been concluded that the slumping of molten debris into the RPV lower plenum could not result in sufficient energy release to threaten the vessel integrity and, hence would not lead directly to containment failure. Likewise, evaluations of both the steam generation rate and shock waves induced by ex-vessel steam explosions show that these would not be of sufficient magnitude to threaten the containment integrity."

Response: Given that the calculations that were used to support these conclusions were not provided in the submittal nor were they

discussed in any detail, no additional comments can be made. The α-mode event included in the PRUEP study was based on the probability distributions used in the NUREG-1150 studies.

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

Response: The source term analysis that was performed as part of the PRUEP study does take credit for fission product retention in the drywell regardless of the operation of the containment sprays. These phenomena are addressed through the parameters FCONV and FCONC in the parametric source term approach used in the study.

2.5 DHR, Other GSI/USIs, and CPI

2.5.1 Evaluation of Decay Heat Removal

As part of the IPE submittal, a table[6] documenting a comparison between the NUREG-1335 guidelines and the LaSalle County Station RMIEP IPE Report was provided. Page 4 of this table states that a thorough discussion of decay heat removal can be found in Section 3.2 of NUREG/CR-5305, Volume 1. The following has been extracted from the discussions on containment venting and containment heat removal.

For long-term containment heat removal accidents and ATWS sequences, decay heat can be removed by venting the containment. The use of a rubber boot connecting the vent pipe to the standby gas treatment system results in steam being released into the reactor building rather than being directed to the stack when the containment is vented. This high-temperature steam creates a severe environment for components located in the reactor building. This severe environment can affect the ability of the systems to perform their functions. The degree of this impact will depend on the environment produced and the qualification of the equipment subjected to the severe environment. Containment venting requires both divisions of ac power.

Heat can also be removed from the containment by the residual heat removal (RHR) system. The RHR system can be used to remove heat during four modes of operation. These are

[6]Hereafter the table is referred to as the NUREG-1335/LaSalle IPE Comparison Table.

Suppression pool cooling:	Water from the suppression pool is pumped through a heat exchanger that is cooled by water from the ultimate heat sink and then back to the suppression pool.
Containment spray:	The process is the same as in suppression pool cooling except that the cooled water is sprayed into the drywell and then drains back to the suppression pool.
Shutdown cooling:	For accidents that are not LOCAs, water can be taken directly from one of the primary system recirculation loops, passed through the heat exchanger, and then injected back into the vessel.
Low-pressure coolant injection:	Water from the suppression pool is passed through the heat exchanger, injected into the vessel, then flows back into the suppression pool via a LOCA or by boiloff through the SRVs.

Each train of RHR (there are two) can function as long as it has ac power and its heat exchanger is being cooled.

The above discussions describe the different modes of DHR and the typical dependencies within the DHR systems. The RMIEP analysis did not identify any unusual dependencies or weaknesses in the DHR function; therefore, we conclude that there are no DHR vulnerability issues even though the IPE submittal does not conatin any specific criteria for resolution of the DHR vulnerability issue.

2.5.2 Other GSI/USIs Addressed in the Submittal

Page 4 of the NUREG-1335/LaSalle IPE Comparison Table described in Section 2.5.1 of this report states that no other unresolved safety issues (USIs) or generic safety issues (GSIs) were evaluated.

2.5.3 Responses to CPI Program Recommendations

Based on a telephone conversation between the NRC and the utility on December 12, 1995, the following information is provided:

(1) Alternate water supply for drywell spray/vessel injection

Provision for and procedures exist to use the firewater system as an alternate water supply for vessel injection. Use of the firewater system as an alternate supply for drywell spray is under consideration. Such use would require a spool piece which is not available at this time.

(2) Enhanced reactor pressure vessel depressurization system reliability

The EOPs provide several different means of vessel depressurization. Examples include 1) turbine bypass valves, 2) turbine-driven reactor feedwater pump, and 3) the RCIC steam line.

(3) Emergency procedures and training

Revision 4 of the BWR Owners Group Emergency Procedure Guidelines have been implemented.

(4) Containment heat removal—Hardened vent

A hardened vent path does exist. It comes off of the normal containment and goes to the reactor building return air riser to two blowout panels—one of which is located at the roof of the auxiliary building, the other at the steam tunnel to the turbine building.

2.6 Vulnerabilities and Plant Improvements

2.6.1 Vulnerabilities

Page 4 of the NUREG-1335/LaSalle IPE Comparison Table states that Section 7.4 of NUREG/CR-4832, Vol. 3, Part 1 lists vulnerabilities. The following vulnerabilities were identified from an examination of the section:

- A sneak circuit in the RCIC isolation logic that results in the RCIC steam line inboard isolation valve closing when offsite ac power is lost and the appropriate diesel generator starts.
- RCIC room temperature isolation logic, in cases where train A ac power has failed but train B ac power is available, isolates if no other emergency core cooling system is working.
- Venting using current procedures results in severe environments in the reactor building.

No definition of vulnerability was given in the section.

2.6.2 Proposed Improvements and Modifications

Page 7 of the NUREG-1335/LaSalle IPE Comparison Table states that Section 7.4 of NUREG/CR-4832, Vol. 3, Part 1 identifies potential improvements implemented or selected for implementation. The following potential improvements were identified from an examination of this section:

- Eliminate the sneak circuit in the RCIC isolation logic that results in the RCIC steam line inboard isolation valve closing when offsite ac power is lost and the appropriate diesel generator starts.

- Change the RCIC room temperature isolation logic so that, in cases where train A ac power has failed but train B ac power is available, RCIC does not isolate if no other emergency core cooling system is working.

- Change the venting procedure so that venting does not result in severe environments in the reactor building.

The only improvement the submittal makes reference to is the sneak circuit. The improvement implemented by the utility is one of procedure modification and operator training, rather than elimination of the sneak circuit by hardware modifications.

In addition, the IPE submittal states that 137 IPE insights were identified during the IPE [p. 35 of submittal] and that 81 accident management insights were identified by a review of the accident management insights generated during the Dresden and Quad Cities IPE effort [pp. 35, 36 of submittal]. However, the submittal does not provide a list of these insights or their disposition. Thus, there is no way to tell if improvements were identified from these insights.

2.6.3 IPE Insights

The IPE submittal states that "observations regarding the station configuration or practices which may affect the risk profile of the plant have been gathered for evaluation" [p. 8 of submittal]. Furthermore, the submittal states that LaSalle County Station insights were obtained from:

- Applicable insights from Dresden and Quad Cities insights,
- The RMIEP analysis, specifically:
  - NUREG/CR-4832 Volume 1, Sections 3 and 4
    Volume 3, Part 1, Section 7
    Volume 4, and

- NUREG/CR-5305 Volume 1, Sections 6 and 7,

- A review of insights from Nine Mile point and WNP2,

- Analyses performed for LaSalle County Station using the CECo-specific version of the MAAP code, and

- Insights identified by all PRA engineers working on the LaSalle IPE/IPEEE effort [pp. 8, 9 of submittal].

The IPE submittal states that 137 IPE insights were identified during the IPE [p. 35 of submittal] and that 81 accident management insights were identified by a review of the accident management insights generated during the Dresden and Quad Cities IPE effort [pp. 35, 36 of submittal]. The submittal does not provide a list of these insights or their disposition.

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3. CONTRACTOR OBSERVATIONS AND CONCLUSIONS

The basis of the submittal is a review of the NRC-sponsored RMIEP and PRUEP analyses by utility personnel. As part of this review, 218 LaSalle-related IPE insights were identified [p. ES-3 of submittal]. These insights were not provided as part of the submittal nor were they discussed. Thus, there is no way to judge the importance of these insights.

No detailed discussions were provided by the submittal regarding vulnerabilities or other concerns such as DHR. However, the submittal did address what was classified as technical concerns. These have been discussed in Sections 2.2, 2.3, and 2.4 of this report.

The RMIEP/PRUEP analyses are based on plant information as of the mid-1980s. The utility states that it plans to update these NRC studies. This is encouraged as it will provide the utility with an up-to-date representation of LaSalle, both from a hardware and operations point of view.

The submittal documentation made extensive reference to the RMIEP and PRUEP documentation. Little effort was made to extract the appropriate RMIEP/PRUEP information and integrate it into the submittal.

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4. DATA SUMMARY SHEET

NOTE: All information was extracted from NUREG/CR-4832 Vols. 1, 3, 5, and 10 and NUREG/CR-5305 Vol. 1, except for the modification which comes from the submittal.

- Total mean core damage frequency (CDF), excluding internal floods: 4.41E-5/yr
- Total mean CDF for internal floods: 3.39E-6/yr
- Initiating events contributing to the total point estimate CDF are

Initiator	Contribution
Loss of offsite power	74.3%
Loss of division 1 4160 VAC bus	8.1%
Transient with turbine bypass	5.8%
Loss of division 1 125 VDC bus	5.2%
Transient with total loss of feedwater	2.9%

- Classes of accident sequences contributing to the total point estimate CDF are

Sequence	Contribution
Transients with failure of all high- and low-pressure systems	64.1%
Transients with failure of all high-pressure systems except RCIC, failure of heat removal, and failure of low-pressure systems	14.6%
Transient with HPCS and one train of CRD working, heat removal fails, venting fails, containment pressure increases till a leak occurs, location of leak determines environment to which equipment is subjected, and equipment fails to survive harsh environment	11.1%
Transient with HPCS and one train of CRD working, heat removal fails, venting fails, containment pressure increases till rupture occurs, location of rupture determines environment to which equipment is subjected, and equipment fails to survive harsh environment	2.9%
Transient with HPCS working, heat removal fails, venting fails, containment pressure increases till a leak occurs, location of leak determines environment to which equipment is subjected, and equipment fails to survive harsh environment	2.5%

- Major operator actions to prevent core damage or containment failure:

- Restore offsite power in 1 hour
- Reopen RCIC F063 valve
- Repair diesel generator failure in 1 hour
- Restore offsite power in 10 hours
- Repair diesel generator failure in 2 hours
- Restore offsite power in 8 hours
- Vent containment

- Conditional mean containment failure probability given core damage:

Containment Failure Locations[1]

Drywell	22%
Wetwell	17%
Vent	49%
Bypass	0%
Intact	12%

Containment Failure Times[1 and 2]

Early	13%
Early vent	5%
Intermediate	17%
Late	9%
Late vent	44%
Bypass	0%
Intact	12%

[1] Includes accidents initiated by traditional internal events and internal floods

[2] Failure times are defined as follows: early - prior to or during core damage, intermediate - around the time of vessel failure, and late - after vessel failure.

- Significant PRA findings:

Design or operational features having the most significant impact in reducing the CDF are

- Common cause diesel generator cooling failure
- Diesel generator fails to start
- Relay failures

- Equipment survivability given harsh environment
- Breaker failures
- Containment failure results in leakage to reactor building

Systems or actions whose failure would have the most significant impact in increasing the CDF are

- Circuit breaker from 4160 V ac emergency bus B to 480 V ac buses 236X and 236Y

- Reactor scram

- Random failure component of the common cause failure of the core standby cooling system pump

- Various electric power circuit breaker failures or maintenance unavailabilities

- Potential improvements under evaluation (or made):

- Changes have been made to LaSalle Procedure LOA-AP-07, "Loss of Auxiliary Electrical Power," which identify the sequence of events resulting in nonrecoverable isolation of RCIC. In addition, operator training during every training cycle targets this concern

- Important plant hardware and plant characteristics:

  - RCIC "sneak circuit" could cause RCIC to be unavailable under certain plant conditions.

  - Reactor pedestal arrangement includes an upper and lower pedestal cavity. Failure of the floor that separates the two cavities can result in ex-vessel releases bypassing the suppression pool.

  - Certain pedestal drain pipes are isolated outside containment

  - Containment venting through SGTS can result in a vent failure that leads to a severe environment in the reactor building which can potentially threaten ECCS systems.

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REFERENCES

[IPE Submittal] LaSalle County Station Individual Plant Examination Submittal Report, April 28, 1994.

[NUREG/CR-4832] A. C. Payne, Jr., "Analysis of the LaSalle Unit 2 Nuclear Power Plant: Risk Methods Integration and Evaluation Program (RMIEP) Summary, NUREG/CR-4832, SAND92-0537.

[NUREG/CR-5305] T. D. Brown et al., "Integrated Risk Assessment for the LaSalle Unit 2 Nuclear Power Plant: Phenomenology and Risk Uncertainty Evaluation Program (PRUEP)," NUREG/CR-5305, SAND90-2765.

[NUREG-1335] U.S. Nuclear Regulatory Commission, "Individual Plant Examination: Submittal Guidance," NUREG-1335.