ENCLOSURE 1

LASALLE COUNTY NUCLEAR POWER STATION, UNITS 1 AND 2
INDIVIDUAL PLANT EXAMINATION
STAFF EVALUATION REPORT

I. INTRODUCTION

On April 28, 1994 Commonwealth Edison Company (ComEd or the licensee) submitted the results of the Individual Plant Examination (IPE) for LaSalle County Nuclear Power Station Units 1 & 2 in response to Generic Letter (GL) 88-20 and associated supplements. LaSalle Unit 2 had already undergone a Level 1 and a Level 2/3 analysis by the NRC under the Risk Methods Integration and Evaluation Program (RMIEP) (NUREG/CR-4832) and the Phenomenology and Risk Uncertainty Evaluation Program (PRUEP) (NUREG/CR-5305), respectively. Because both of those analyses had received a detailed technical review and because the IPE submittal for LaSalle Units 1 and 2 is the result of a detail review by the licensee of the RMIEP and PRUEP studies, the staff modified its "Step 1" review procedure. Due to the information available, requests for additional information were not necessary.

The staff's review concentrated on (1) the reasonableness of the results given the design, operation, and history of LaSalle Units 1 and 2 and (2) the completeness of the information. In particular, the staff's review focused on the utility's staff involvement; the incorporation of current plant design, operation, and history; the resolution of Unresolved Safety Issue (USI) A-45 "Shutdown Decay Heat Removal [DHR] Requirements"; and the licensee's response to the Containment Performance Improvement (CPI) Program. The NRC employed Sandia National Laboratories (SNL) to review the LaSalle IPE submittal. The results of SNL's modified "Step 1" review and the staff's conclusions are discussed in this staff evaluation report (SER). SNL's technical evaluation report (TER) (appendix) is attached.

The licensee raised several "technical concerns" pertaining to the NRC RMIEP analysis in its submittal; ComEd plans to address these concerns as part of the IPE's update. Among the technical concerns raised are that (1) the RMIEP "beta factor common cause analysis process is too conservative" and (2) the RMIEP human reliability analysis results "appear to be non-conservative." However, the staff has also identified several weaknesses in the licensee's method for both common cause and human reliability analysis in other ComEd IPE submittals reviewed thus far (i.e., Zion, Dresden, and Quad Cities). Therefore, the licensee should consider the staff's technical concerns regarding these two issues (documented in the SERs for the Zion, Dresden, and Quad Cities IPEs) in the LaSalle IPE update, as appropriate.

In accordance with GL 88-20, the licensee proposed to resolve USI A-45. The licensee did not propose resolution of other specific USIs or generic safety issues (GSIs) as part of the LaSalle IPE.

II. EVALUATION

LaSalle Units 1 and 2 are BWR 5 reactors with Mark II containments. In the submittal the licensee states that LaSalle IPE "is the result of a detailed review of the NRC's Risk Methods Integration Program (RMIEP) (NUREG/CR-4832) analysis," and that "it is ComEd's position that the objectives of the Generic Letter 88-20 have been accomplished for both internal and external events through this review process."

To achieve the objectives of GL 88-20, the licensee reviewed the plant physical layout and procedures; examined the IPE submittals for other BWR 5 reactors with Mark II containments; performed limited modular accident analysis program (MAAP) calculations; reviewed and analyzed the RMIEP dominant sequences (top 95% of core damage frequency (CDF)) and key basic events; developed observations and insights regarding plant configuration or practices that may affect the risk; and identified and documented technical issues that it will address in the update of the LaSalle IPE. In the submittal the licensee stated that the utility personnel assigned to the review project "have extensive experience in plant operations and systems engineering, as well as probabilistic risk assessment (PRA) experience." And that ComEd personnel "performed the basic modeling review and analysis, as well as the Level II review and analysis using a ComEd-specific version of the MAAP code." The staff concludes that the objective of extensive participation of utility personnel in the IPE process has been met.

For the IPE's Level 1 analysis, the licensee reviewed the results of the NRC's RMIEP study. In the RMIEP analysis the staff estimated a CDF of 5E-5/reactor-year from internally initiated events including a contribution of 3E-6/reactor-year from internal floods. Loss of offsite power contributes 74%; loss of one AC Division 8%; transient with turbine bypass 6%; loss of one dc division 5%; and transient with loss of feedwater 3%. Of the remaining initiating events each contributes less than 3% of the total CDF. The important systems and equipment contributors to the estimated CDF are diesel generator (DG) cooling failure due to common cause, DG failure to start, relay failures, equipment unsurvivability under harsh environment, breaker failures, and containment failure resulting in leakage in the reactor building. The licensee appears to have examined the significant initiating events and dominant sequences identified in the RMIEP analysis.

Also, the licensee appears to have examined the RMIEP HRA where potential failures in human-system interactions and human related recovery failures were quantified and documented. The following operator actions were identified in RMIEP as important to CDF: restore offsite power in 1 hour, reopen reactor core isolation cooling (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, and vent containment.

For the IPE's Level 2 analysis, the licensee reviewed the results of the NRC's PRUEP study. Under PRUEP the staff evaluated and quantified the results of severe accident progression through the use of a containment event tree and considered uncertainties in containment response through the use of sensitivity analyses. The probabilities of containment failure (assuming core damage) are the following: [in terms of containment failure locations] vent 49%, drywell 22%; wetwell 17%, intact 12%, bypass 0%, [in terms of containment failure times] early 13%, early vent 5%, intermediate 17%, late 9%, late vent 44%, bypass 0%, intact 12%. [Failure times indicated are defined as follows: early - before or during core damage, intermediate - about the time of vessel failure, and late - after vessel failure.] These probabilities include accidents initiated by traditional internal events and internal floods. The licensee's response to containment performance improvement program recommendations is consistent with the intent of Generic Letter 88-20 and associated Supplement 3.

Because the licensee did not identify any vulnerabilities from its review RMIEP and PRUEP analyses of LaSalle Unit 2 and because these studies were ongoing at the time of the issuance of GL 88-20 (published in 1992), were in-depth NRC studies, and had received a detailed peer review, the staff concludes that the objective of the GL 88-20 that IPEs reflect "current" plant design, operation, and history, has been met. In addition, the staff notes that the licensee is committed to update the IPE to appropriately reflect plant design, and operation. Specifically, the licensee is committed to "perform a detailed unit-to-unit system difference and other events that could potentially be simultaneous initiators at both units" and to use plant specific data whenever available "to properly characterize plant behavior."

Regarding the resolution of USI A-45, the licensee states in the submittal that a thorough discussion of DHR could be found in Section 3.2 of NUREG/CR-5305, Volume 1. Since NUREG/CR-5305 did not identify any unusual features or weaknesses regarding LaSalle's DHR function, the staff considers that USI A-45 has been resolved for LaSalle Units 1 and 2.

The licensee did not define what constitutes a vulnerability. However, in the table entitled, "Comparison of NUREG-1335 Requirements to the LaSalle Station RMIEP Report," it is stated that Section 7.4 of NUREG/CR-4832, Vol 3, Part 1 lists "vulnerabilities (i.e., areas for improvement)." From an examination of this section, the following areas for improvement were identified in NUREG/CR-4832:

(1) A sneak circuit in the reactor core isolation cooling (RCIC) isolation logic results in the closing of the RCIC steam line inboard isolation valve when offsite ac power is lost and the appropriate diesel generator starts.

(2) 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.

(3) Venting using current procedures results in severe environments in the reactor building.

The following potential improvements were identified in Section 7.4 of NUREG/CR-4832, Vol 3, with regards to these areas:

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

(2) 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, this isolation logic does not isolate if no other emergency core cooling system is working.

(3) Change the venting procedure so that venting does not result in severe environments in the reactor building.

Instead of eliminating the sneak circuit, the licensee modified procedure LOA-AP-07, "Loss of Auxiliary Electrical Power," and improved operator training on this procedure. The submittal did not contain a discussion on the other potential improvements identified under RMIEP. However, it did state that 137 insights were identified during the IPE and that 81 accident management insights were identified by a review of the accident management insights generated during the Dresden and Quad Cities IPEs (i.e., a total of 218 insights). The submittal did not contain a list of those insights or a discussion regarding their disposition.

III. CONCLUSION

On the basis of the above findings, the staff notes that the IPE results are reasonable. Furthermore, the licensee is committed to update the IPE in order to appropriately reflect current plant design, operation, and history. As a result, the staff concludes that the licensee's IPE process is capable of identifying the most likely severe accidents and severe accident vulnerabilities and, therefore, the LaSalle Units 1 and 2 IPE meets the intent of GL 88-20.

It should be noted that the staff's review primarily focused on the licensee's ability to examine LaSalle Units 1 and 2 for severe accident vulnerabilities. Although the staff may have explored certain aspects of the IPE in more detail than others, the review is not intended to validate the accuracy of the licensee's detailed findings (or quantification estimates) that stemmed from the examination. Therefore, this SER does not constitute NRC approval or endorsement of any IPE material for purposes other than those associated with meeting the intent of GL 88-20.

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ENCLOSURE 2

LASALLE COUNTY NUCLEAR POWER STATION, UNITS 1 AND 2
INDIVIDUAL PLANT EXAMINATION
TECHNICAL EVALUATION REPORT