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May 12, 2010

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

LaSalle County Station, Units 1 and 2
Facility Operating License Nos. NPF-11 and NPF-18
NRC Docket Nos. 50-373 and 50-374

Subject: Additional Information Supporting Request for License Amendment Regarding Measurement Uncertainty Recapture Power Uprate

References:

1. Letter from M. D. Jesse (Exelon Generation Company, LLC) to U. S. NRC, "Request for License Amendment Regarding Measurement Uncertainty Recapture Power Uprate," dated January 27, 2010
2. Letter from C. S. Goodwin (U. S. NRC) to C. G. Pardee (Exelon Generation Company, LLC), "LaSalle County Station, Units 1 and 2 – Request for Additional Information Related to Request for License Amendment Regarding Measurement Uncertainty Recapture Power Uprate," dated April 15, 2010

In Reference 1, Exelon Generation Company, LLC (EGC) requested an amendment to Facility Operating License Nos. NPF-11 and NPF-18 for LaSalle County Station (LSCS), Units 1 and 2, respectively. Specifically, the proposed changes revise the Operating License and Technical Specifications to implement an increase in rated thermal power of approximately 1.65%. In Reference 2, the NRC requested additional information to support review of the proposed changes. In response to this request, EGC is providing the attached information.

Additionally, EGC is providing a correction to the information provided in Reference 1. The correction addresses a minor inconsistency between information contained in two attachments to Reference 1.

EGC has reviewed the information supporting a finding of no significant hazards consideration and the environmental consideration provided to the NRC in Reference 1. The additional information provided in this submittal does not affect the bases for concluding that the proposed license amendment does not involve a significant hazards consideration. In addition, the additional information provided in this submittal does not affect the bases for concluding that neither an environmental impact statement nor an environmental assessment needs to be prepared in connection with the proposed amendment.

There are no regulatory commitments contained in this letter.

Should you have any questions concerning this letter, please contact Mr. Joseph A. Bauer at (630) 657-3376.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 12th day of May 2010.

Respectfully,

Michael D. Jesse
Manager, Licensing – Power Uprate

Attachments: Response to Request for Additional Information

cc: NRC Regional Administrator, Region III
NRC Senior Resident Inspector – LaSalle County Station
Illinois Emergency Management Agency – Division of Nuclear Safety

FIRE PROTECTION BRANCH

NRC Request 1

The NRC staff notes that Enclosure 6 to the GEH Nuclear Energy Safety Analysis report for LSCS, Units 1 and 2 Thermal Power Optimization, NEDO-33485P, Section 6.7, "Fire Protection," states that *"…There is no change in the physical plant configuration and the potential for minor changes to combustible loading as result of TPO uprate…"* The NRC staff requests that the licensee summarize any changes to the combustible loading, however minor, and discuss the impact of these changes on the plant's compliance with the fire protection program licensing basis, 10 CFR 50.48, or applicable portions of 10 CFR 50, Appendix R.

Response

The following information from the LaSalle County Station (LSCS) combustible loading calculations provides assurance that the impact of the thermal power optimization (TPO) uprate on the fire protection program is inconsequential, since the total combustible loading in the affected areas remains well below the administrative fire load limits.

- Unit 1 Leading Edge Flow Meter (LEFM)

  For Zone 4F3 (Auxiliary Building (AB) / elevation 710′ 6″), installation of the LEFM added 17.82 BTU/ft^2 of combustible load. The current total fire loading in this area is 127,416 BTU/ft^2. This addition has an insignificant impact on the fire zone loading.

  For Zone 5C11 (Turbine Building (TB) / elevation 710′ 6″), installation of the LEFM added 8.19 BTU/ft^2 of combustible load. The current total fire loading in this area is 35,032 BTU/ft^2. This addition has an insignificant impact on the fire zone loading.

- Unit 2 LEFM

  For Zone 4F3 (AB / elevation 710′ 6″), when the LEFM is installed, it will add 10.69 BTU/ft^2 of combustible load. The current total fire loading in this area is 127,416 BTU/ft^2. This addition will have an insignificant impact on the fire zone loading.

  For Zone 5C11 (TB / elevation 710′ 6″), when the LEFM is installed, it will add 8.84 BTU/ft^2 of combustible load. The current total fire loading in this area is 35,032 BTU/ft^2. This addition will have an insignificant impact on the fire zone loading.

NRC Request 2

Some plants credit aspects of their fire protection system for other than fire protection activities, e.g., utilizing the fire water pumps and water supply as backup cooling or inventory for non-primary reactor systems. If LSCS credits its fire protection system in this way, please identify the specific situations and discuss to what extent, if any, the MUR power uprate affects these "non-fire-protection" aspects of the plant fire protection system. Please verify if LSCS does not take such credit.

Response

There are no design basis accidents or transients (other than fire) that credit the use of the fire protection (FP) system at LSCS.

LSCS credits the following uses of the FP system for beyond design basis events. As described below, the TPO uprate has no impact on the ability of the FP system to respond to these events.

- LaSalle General Abnormal (LGA) Procedure LGA-FP-01, "Alternate Vessel Injection Using the Fire Protection System," provides guidance to line up and inject water from the FP system to the reactor pressure vessel (RPV). The LGAs and the LaSalle Severe Accident Management Guidelines provide for the use of several alternate methods to inject water to control RPV level if the design basis sources of water are unavailable. Scenarios requiring this alternate makeup to the RPV are unaffected by the TPO uprate, since they are based on initiating conditions consistent with design basis accidents that have been analyzed at bounding power levels, as described in Reference 1, Attachment 6, Section 9.2, Design Basis Accidents."

- LaSalle Operating Abnormal (LOA) Procedure LOA-FC-101(201), "Unit 1(2) Fuel Pool Cooling System / Reactor Cavity Level Abnormal," provides guidance to line up the FP system to fill the spent fuel pool and/or reactor cavity when reactor building access is restricted due to radiation levels, fires, and similar conditions. The beyond design basis scenarios in which make up from the FP system would be required are unaffected by the TPO uprate since the fire protection system makeup capacity significantly exceeds the capacity of the makeup sources credited in the design basis. The capacity of these design basis sources was determined to be acceptable for TPO conditions.

- Use of the FP system is credited in certain security event scenarios. TPO does not affect these scenarios, beyond the events discussed above.

REACTOR SYSTEMS BRANCH

NRC Request 1

Describe the effects the flow straightener has on the flow profile observed by the ultrasonic flowmeter.

Response

The effects of flow straighteners on the flow profile are discussed in detail in the Cameron International Corporation Engineering Report, ER-790, "An Evaluation of the Impact of 55 Tube Permutit Flow Conditioners on the Meter Factor of an LEFM CheckPlus," Revision 1, dated March 2010. Cameron International Corporation previously submitted this report to the NRC in a letter dated March, 19 2010. The following information is taken from the report.

> Parametric calibration tests with five different models of flow straighteners and two different LEFM CheckPlus flow elements indicate that the random uncertainty attributable to the orientation of the flow straightener relative to the LEFM is ± 0.20% (at a 95% confidence level). An additional random uncertainty of ± 0.15% (also at a 95% confidence level) is required to bound the uncertainty in the meter factor of an LEFM CheckPlus flow element when that flow element is calibrated in a test laboratory without a flow straightener upstream; then subsequently installed in the feedwater system 8.5 to 11 diameters downstream of a flow straightener. These two uncertainty terms are independent and are therefore combined as the root sum squares, giving an aggregate "modeling sensitivity" uncertainty attributable to the flow straightener in a single flow measurement of ± 0.25%. In installations where total feedwater flow is determined by summing the indications of two or more LEFMs, (where each LEFM is downstream of a flow straightener), the uncertainty in total feedwater flow attributable to these uncertainty terms should be determined by combining them as the root sum square, since the uncertainty terms are not systematically related. Therefore, for the LSCS installation in which each LEFM measures roughly half of the total flow, the aggregate uncertainty due to the flow straightener is ± 0.25% ÷ $2^{1/2}$ or ± 0.18%.

This modeling sensitivity uncertainty is used as one of the inputs in determining the bounding feedwater mass flow rate uncertainty.

For a complete discussion of the effects of flow straighteners on the LEFM CheckPlus, please refer to ER-790, noted above.

<u>NRC Request 2</u>

What does the effect of a 2.5 R/hr radiation field at full power have on the expected lifetime of transducers?

<u>Response</u>

The transducers in the LEFMs are PZT-5A piezoceramic material. This material has been exposed to gamma irradiation on the order of 10^7 roentgens. The piezoelectric properties were measured before and after exposure. The test results show negligible losses of material properties at these elevated exposure levels. Therefore, at an exposure rate of 2.5 R/hr, the expected life time of the transducers would not be impacted during the operating life of the plant.

<u>MECHANICAL AND CIVIL ENGINEERING BRANCH</u>

<u>NRC Request 1</u>

Section 3.4 of Attachment 8 of the licensee's submittal summarizes the review performed to determine whether the LSCS, Units 1 and 2, reactor vessel internals and safety-related main steam (MS) and feedwater (FW) piping would be able to withstand the effects of increased flow-induced vibration (FIV), due to the higher flow conditions present at the proposed Thermal Power Optimization (TPO or measurement uncertainty recapture (MUR)) power uprate conditions. With regards to the possible effects of increased FIV at the TPO conditions, tabulated under Section 3.4 of Attachment 8, please address the following:

(a) On page 3-19 of Attachment 8, it is stated that the expected TPO uprate vibrations were compared against the General Electric Hitachi (GEH) acceptance criteria. Please confirm that the GEH acceptance criteria are those "…established vibration acceptance limits" cited on page 3-18 of Attachment 8. Additionally, please provide the references which provide the regulatory acceptance bases of the acceptance criterion.

(b) On page 3-19 of Attachment 8, it is stated that the MS and FW piping vibration levels will remain within acceptable limits at TPO conditions, due to acceptable performance during FIV testing during initial plant startup and acceptable operating experience at current licensed thermal power (CLTP). In concert with the response to RAI 1(a) above, please expand on the MS and FW piping FIV acceptance criterion cited here within and provide the references which provide the regulatory acceptance bases of the acceptance criterion.

(c) Please provide justification for citing the acceptance of vibration levels at the requested TPO levels based on the operating experience at CLTP, when it is stated that piping vibrations are expected to be roughly 4 percent higher than the current vibration levels. If this justification includes extrapolations, baseline vibration data, or other analytical methods, please provide the references which provide the regulatory acceptance bases of the referenced methodologies.

Response

(a) It is confirmed that GEH acceptance criteria for reactor internals referenced on page 3-19 of Attachment 8 are the established vibration acceptance criteria limits cited on page 3-18 of Attachment 8. GEH allowable criteria for sustained vibration for the life of the plant are based on fatigue curves in the GE BWR Handbook of Plant Material Properties. The peak vibration stress amplitude acceptance criterion is 10,000 psi.

The GEH limiting value of 10,000 psi is more conservative than the value of 13,600 psi for 10^{11} cycles specified by the ASME design code Section III, Appendix I, Figure I-9.2.2 for the same material.

The GEH acceptance criteria for reactor internals are more conservative than the ASME code, which has been approved by the NRC.

(b) See response to part (a) above. The GEH vibration acceptance criteria for MS and FW piping is the same as that used for reactor internals.

(c) The steady state FIV levels at the requested TPO conditions were based on extrapolating the measured vibration levels during startup and operating experience. The vibrations are assumed to increase in proportion to the fluid density (ρ) and fluid velocity squared (V) squared, or ρV^2. Therefore, the piping vibrations at TPO conditions are expected to be roughly four percent higher than current vibrations levels based on the extrapolations of the flow rates. There is still more than 70% margin to the vibration limits after extrapolation.

INSTRUMENTATION AND CONTROLS BRANCH

NRC Request 1

RIS 2002-03, Attachment I, Section I, Item F asks that licensees provide information related to calibration and maintenance procedures that affect the power calorimetric calculation. Please provide information related to how LSCS will control the hardware and software configuration of the Cameron Leading Edge Flow Meter (LEFM) CheckPlus equipment.

Response

After installation, the LEFM CheckPlus system software configuration will be maintained using existing procedures and processes. The plant computer software configuration is maintained in accordance with the Exelon Nuclear change control process, which includes verification and validation of changes to software configuration. Configuration of the hardware associated with the LEFM CheckPlus system and the calorimetric process instrumentation will be maintained in accordance with Exelon Nuclear configuration control processes.

NRC Request 2

A 72-hour Allowable Outage Time (AOT) has been requested for the LSCS units to remain above the Current Licensed Thermal Power (i.e., 3,489 MWt) up to the requested uprated power (i.e., 3,546 MWt) in the event that the Cameron LEFM CheckPlus is declared non-operational. In support of this request, please provide information on the following:

(a) Please provide a description of what level of Cameron LEFM CheckPlus degradation or system alert would render the Cameron LEFM CheckPlus to be declared non-operational at LSCS.

(b) The license amendment request stated that the feedwater flow nozzle measurements would be used in lieu of the non-operational Cameron LEFM CheckPlus to remain above the Current Licensed Thermal Power during the AOT. Are the feedwater flow nozzles routinely calibrated to the Cameron LEFM CheckPlus measurement? If so, how often is the calibration performed? If not, what measures are taken at LaSalle County Station to ensure that the feedwater flow nozzle measurements are sufficiently accurate during the AOT to justify operation at the uprated power level?

(c) If the power level is below the Current Licensed Thermal Power at the time the Cameron LEFM CheckPlus is declared non-operational or if the power level drops below the Current Licensed Thermal Power during the AOT, will power be raised above the Current Licensed Thermal Power prior to the Cameron LEFM CheckPlus becoming operational? If so, please provide justification.

(d) Has there been any recent evidence of feedwater nozzle fouling at either LSCS unit?

(e) Are there any plant-specific feedwater flow nozzle measurement drift data for the LSCS units? If so, is this data consistent with the measurement drift errors cited from the ER-80P Topical Report?

Response

(a) The LEFM system performs on-line self-diagnostics to verify system operation is within design basis uncertainty limits. Any out-of-specification condition will result in a self-diagnostic alarm condition, either for “alert” status (i.e., increased flow measurement uncertainty) or “failure” status. In either of these cases, the LEFM will be considered non-operational and the proposed Technical Requirements Manual limiting condition for operation (TLCO) Required Actions will be applied. Additionally, if the communication link between the LEFM system and the plant computer is failed, (i.e., LEFM CPU Link A and B Failed), the LEFM will be considered non-operational and the proposed TLCO Required Actions will be applied.

(b) The ratio between the existing feedwater venturi flow measurement and the LEFM system flow measurement will be continuously monitored. LSCS does not calibrate the feedwater flow venturi to the LEFM CheckPlus measurement; however, when the LEFM becomes inoperable, a correction factor based on this ratio will be applied to the feedwater venturi flow measurement. This will ensure accuracy of the core thermal power calculation while relying on the feedwater flow venturi input during the AOT.

(c) If core thermal power is less than (or becomes less than) 3489 MWt (i.e., the Current Licensed Thermal Power) with the LEFM non-operational, power will not be raised above 3489 MWt until the LEFM is made operational. This is controlled by the LSCS Technical Requirements Manual (TRM), Section 3.0, TRM Limiting Condition for Operation (TLCO) 3.0.d, which prohibits entering a mode or condition specified in the applicability when a TLCO is not met, except when the associated actions permit operation in that condition for an unlimited period of time. The Required Actions only permit operation for 72 hours. The applicability for proposed LEFM TLCO 3.3.q states that the TLCO applies to power levels greater than 3489 MWt. Thus, TLCO 3.0.d would prohibit raising power to greater than 3489 MWt with the LEFM non-operational.

(d) Fouling has been observed in the Unit 1 feedwater flow venturis. Inspection during refueling outage L1R12 indicated that a likely cause of the fouling was due to corrosion product deposition on the venturi throat. It is suspected that the corrosion products were a result of degradation of the 13A, 13B and 13C feedwater heaters. The three feedwater heaters in question were replaced in February 2008, and there has been no evidence of an increase or decrease in the degree of fouling of the Unit 1 venturis since that time.

An evaluation of the Unit 2 venturis behavior was also conducted from 2000 through 2006. The data indicated the possibility of a minor amount of fouling that could cause a decline in indicated feedwater flow of no more than approximately 0.5% of rated flow. There has been no evidence of a decrease or increase in the degree of fouling of the Unit 2 venturis since that time.

Because the degree of fouling of the Unit 1 and Unit 2 venturis has been stable for a considerable period of time, a defouling event during the 72-hour AOT is considered unlikely. As discussed in Reference 5, Section 3.2.4, "Disposition of NRC Criteria for Use of LEFM Topical Reports," significant sudden defouling would be detected by a change in secondary plant parameters.

(e) The feedwater flow venturi measurement is made using Rosemount 1151 differential pressure transmitters. The feedwater flow measurement is a non-safety related application; therefore, no instrument loop drift data is maintained. However, instrument loop drift data has been analyzed for the 1151 series of Rosemount differential pressure transmitters that are used in safety-related applications at LSCS. This drift data supports a total uncertainty due to drift of 1.6% for a 24-month cycle, which is comparable to the values cited from the ER-80P topical report.

NRC Request 3

Section 3.4.4 of Attachment 1 of the License Amendment Request notes that the setpoint methodology is described in NES-EIC-20.04, Revision 5. It was noted that the NRC had previously reviewed and approved the setpoint methodology in the document. However, the NRC approval contained in ML011130202 was based upon Revision 3 of the document. Please provide a summary of the changes from Revision 3 to Revision 5 of NES-EIC-20.04.

Response

Revisions 4 and 5 to NES-EIC-20.04, "Analysis of Instrument Channel Setpoint Error and Instrument Loop Accuracy," did not make changes to the setpoint methodology. Revision 4 consisted of administrative changes and clarifications. Revision 5 updated references, made editorial changes, and incorporated consideration of potential effects from high static pressure process measurement for Rosemount flow transmitters.

ACCIDENT DOSE BRANCH

NRC Request 1

Please clarify which analyses of record were assessed to make the determination that post-accident dose consequences, following acceptance of a 1.65 percent MUR (TPO) power uprate, would be bounded by those dose consequence that are currently calculated. Also, please describe the methodology used in those analyses.

Response

The following UFSAR Chapter 15 accidents with radiological dose consequences were reviewed for TPO conditions:

UFSAR Chapter 15 Accidents with Radiological Dose Consequences

Accident/Transient	UFSAR Section	Assumed Reactor Power Level (MWt)	Assumed Reactor Power Level (% of CLTP)	Bounding (Yes/No)	NRC Approval
Loss of Coolant Accident (LOCA) Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary	15.6.5.5	3910	112	Yes	*Reference 1, Section 2.3* *Also discussed in Reference 2, Section 9.2.1*
Rod Drop Accident	15.4.9.5	3559	102	Yes	*Reference 2, Section 9.2.3*
Instrument Line Break	15.6.2.5	N/A	N/A[a]	Yes	*Reference 3, Section 15.3.7* *Also discussed in Reference 2, Section 9.2.5*
Steam System Pipe Break Outside the Secondary Containment	15.6.4.5	N/A	N/A[b]	Yes	*Reference 3, Section 15.3.3* *Also discussed in Reference 2, Section 9.2.2*
Feedwater Line Break	15.6.6.5	3559	102	Yes	Reference 2, Section 9.2.6
Main Condenser Gas Treatment System Failure	15.7.1.1.5	N/A	N/A[c]	Yes	*Reference 2, Section 9.2.7*
Radioactive Liquid Waste System Leak	15.7.3.5	N/A	N/A[d]	Yes	*Reference 3, Section 15.3.9* *Also discussed in Reference 2, Section 9.2.8*
Fuel Handling Accident	15.7.4.5	3631[e]	104	Yes	*Reference 2, Section 9.2.3*
Spent Fuel Cask Drop Accident	15.7.5.5	3489[f]	100	No	*Reference 2, Section 9.2.9*

With the exception of the Spent Fuel Cask Drop accident, all of these analyses that are affected by core thermal power were performed with an analytical value for core thermal power of 3559 MWt (i.e., 102% of CLTP), or higher. As such, this power level envelopes the proposed TPO power level with appropriate allowance for calorimetric uncertainty.

The following accident dose consequences from the table above do not depend on core thermal power or are not limiting accidents:

(a) Instrument Line Break – This accident does not involve fuel failure. The source of radionuclide release is the reactor coolant. The principal radionuclides are iodines and noble gases. The doses are not dependent on core power because they are calculated using the design basis reactor coolant source term, which is not changed under TPO conditions. Furthermore, the amount of liquid released will not change because the reactor operating reactor pressure is unchanged under TPO conditions.

(b) Steam Line Break Outside Containment – This accident does not involve fuel failure. The source of radionuclide release is the reactor coolant. The principal radionuclides are iodines and noble gases. Since the operating reactor pressure is unchanged at TPO conditions, the quantity of released steam and radioactivity remains bounding. Therefore, dose consequences for this postulated event will remain bounding under TPO conditions.

(c) Main Condenser Gas Treatment System Failure – The UFSAR states that the design basis analysis for this event does not depend on core power. Because the stretch power uprate previously used the design basis analysis as the approach for the analysis of this accident type, the design basis analysis is also used as the basis for the analysis here (i.e., for TPO conditions). Consequently, the change in the core power level does not impact dose consequences for this postulated event.

(d) Radioactive Liquid Waste System Leak – This accident does not involve fuel failure. The source of radionuclide release is liquid radwaste which is not significantly dependent on thermal power level.

(e) Fuel Handling Accident - Prior to the previous five percent power uprate, the analysis power level for this event was 3458 MWt (i.e., 1.04 times the original licensed power level of 3323 MWt). The five percent power uprate evaluated this event at 3631 MWt (i.e., 1.05 times 3458 MWt).

(f) Spent Fuel Cask Drop – This accident is not a bounding accident for accident dose consequences. This accident is currently assessed at an analytically assumed core thermal power of 3489 MW_t. The current dose consequences for this accident range from 'negligible' to $<10^{-5}$ rem thyroid, exclusion area boundary (EAB) and low population zone (LPZ), and $<10^{-5}$ rem whole body, EAB and LPZ. For TPO conditions, these consequences were assumed to scale with core power (i.e., an additional factor of 1.02). With this scaling, the Spent Fuel Cask Drop accident will continue to be several orders of magnitude below regulatory limits.

TPO Impact on the Previously Submitted Alternate Source Term (AST) Analysis

As described above, the current licensing basis for limiting offsite dose consequences will continue to be bounding for operations under the proposed power uprate.

Note that the evaluations described above are based on the current analyses of record and are not based on AST methodology. In Reference 4, LSCS submitted a license amendment request utilizing analyses based on AST methodology for the radiological consequences associated with the LOCA and Fuel Handling Accident. These AST analyses are also bounding (with allowance for calorimetric uncertainty) for TPO conditions. Thus, approval of the Reference 4 license amendment request will not alter the conclusion that the radiological consequences of these accidents are bounded for the TPO uprate. The AST-based radiological consequences of the LOCA and Fuel Handling Accident will become the station licensing and design basis when accepted by the NRC.

CORRECTION TO PREVIOUSLY SUBMITTED INFORMATION

Reference 5, Attachment 1, Section 3.4.5, "Grid Studies," discusses the results of a system stability analysis performed by PJM Interconnection (PJM) to assess the impact of the uprate on the rotor angle stability of generating plants in the Commonwealth Edison (ComEd) and neighboring control areas. In discussing the results of that analysis, Section 3.4.5 stated the following:

> The analysis conclusions are as follows:
>
> 1. All of the primary-clearing scenarios were found to be stable.
> 2. All of the primary-clearing scenarios with maintenance outages considered were found to be stable.
> 3. Of the twenty breaker failure scenarios studied with fault detector logic, three are unstable. The study provided remediation measures for these three scenarios, involving modifications to two 345 kV breakers in the LaSalle switchyard. EGC will ensure that any modifications required by PJM are completed prior to uprate implementation. Further details regarding this study are provided in Attachment 12.

EGC is correcting the information stated in the conclusion number 3 above. The correction is shown below with the change underlined.

> 1. All of the primary-clearing scenarios were found to be stable.
> 2. All of the primary-clearing scenarios with maintenance outages considered were found to be stable.
> 3. Of the twenty breaker failure scenarios studied with fault detector logic, three are unstable. The study provided remediation measures for these three scenarios, involving modifications to ~~two~~ <u>three</u> 345 kV breakers in the LaSalle switchyard. EGC will ensure that any modifications required by PJM are completed prior to uprate implementation. Further details regarding this study are provided in Attachment 12.

The corrected information is consistent with the conclusions provided in Reference 1, Attachment 12, "PJM Generator Transient Stability Study for LaSalle Station."

REFERENCES

1. Letter from U. S. NRC to O. D. Kingsley (Commonwealth Edison Company), "Issuance of Amendments," dated May 13, 1996

2. Letter from U. S. NRC to O. D. Kingsley, (Commonwealth Edison Company), "LaSalle - Issuance of Amendments Regarding Power Uprate," dated May 9, 2000

3. "Safety Evaluation Report Related to the Operation of LaSalle County Station Units 1 and 2," NUREG-0519, dated March 1981

4. Letter from P. R. Simpson (EGC) to U. S. NRC, "Additional Information Regarding Request for License Amendment Regarding Application of Alternative Source Term," dated October 23, 2008

5. Letter from M. D. Jesse (Exelon Generation Company, LLC) to U.S. NRC, "Request for License Amendment Regarding Measurement Uncertainty Recapture Power Uprate," dated January 27, 2010