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June 20, 2003

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D C 20555

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

Response to Request for Additional Information to Support Request for Amendment to Technical Specification 5.5.13, "Primary Containment Leakage Rate Testing Program"

References: (1) Letter from K. R. Jury (EGC) to the NRC, "Request for Amendment to Technical Specification 5.5.13, 'Primary Containment Leakage Rate Testing Program,'" dated October 24, 2002

(2) Letter from W. A. Macon Jr. (NRC) to J. L. Skolds (EGC), "LaSalle County Station, Units 1 and 2 – Request for Additional Information (TAC Nos. MB6574 and MB6575)," dated May 21, 2003

Exelon Generation Company (EGC), LLC, in Reference 1, requested changes to the Technical Specifications (TS) of LaSalle County Station (LSCS) Units 1 and 2. Specifically, the proposed changes would revise TS 5.5.13, "Primary Containment Leakage Rate Testing Program" to reflect a one-time deferral of the primary containment Type A test to no later than June 13, 2009 for Unit 1 and no later than December 7, 2008 for Unit 2. The NRC in Reference 2 requested additional information to support their review of the submittal. Attached to this letter is the requested information.

Additionally, during the preparation of the responses to the questions, it was discovered that an engineering evaluation conducted in 1999 incorrectly concluded that a change to the LSCS Primary Containment Leakage Rate Testing Program could be implemented without prior NRC approval. The change involved the testing/inspection frequency of primary containment isolation valves' packing that is not exposed to test pressure during containment isolation valve testing. LSCS has entered this situation into its Corrective Action Program and currently plans to submit an exemption request to the NRC in July 2003.

Should you have any questions concerning this matter, please contact Mr. T. W. Simpkin at (630) 657-2821.

I declare under penalty of perjury that the foregoing is true and correct.

Respectfully,

Executed on June 20, 2003

T. W. Simpkin

T. W. Simpkin
Manager – Licensing
Mid-West Regional Operating Group

Attachment: Response to Request for Additional Information to Support Request for Amendment to Technical Specification 5.5.13

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – LaSalle County Station
Office of Nuclear Facility Safety – Illinois Department of Nuclear Safety

ATTACHMENT

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION TO SUPPORT REQUEST FOR AMENDMENT TO TECHNICAL SPECIFICATION 5.5.13

Question 1
Section 5.5 of Attachment 2 of the reference provides a summary of the containment inservice inspection. The summary indicates that the first period inspections were performed in accordance with the 1992 Edition and the 1992 Addenda (1992 E&A) of Subsections IWE and IWL of Section XI of the ASME Code and were completed in 2000. The summary also notes that the future inspections will be performed using the 1998 edition of the ASME Code as approved by NRC relief request. The staff requests the licensee to clarify (1) if the 1992 E&A will continue to be used until the end of the current inspection interval or whether a new interval will be established for use of the 1998 Edition of the ASME Code and (2) if the licensee is planning to use the 1998 Edition of the ASME Code, its extent of compliance with the amended rule to 10CFR 50.55a (67 FR 60520) which incorporates by reference the 1998 Edition through the 2000 Addenda of the Code including any modifications and limitations in 10 CFR 50.55a(b)(2).

Response 1
The Containment Inservice Inspection Program at LaSalle County Station (LSCS) was developed in accordance with the 1992 Edition, 1992 Addenda of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI , Subsection IWE and IWL, as modified by NRC final rulemaking to 10CFR 50.55a published in the Federal Register on August 8, 1996. The first containment inspection interval plan was developed based on the 1992 Edition, 1992 Addenda of the ASME code. The NRC in Reference 4 approved the use of the 1998 Edition of the ASME code, as supplemented by Exelon commitments, as an approved alternative to the 1992 Edition, 1992 Addenda of the ASME code for the first containment inspection interval. The NRC approval did not require the initiation of a new interval and the thus the 1992 Edition, 1992 Addenda will be considered the code of record for the first interval. The supplemental LSCS commitments identified in the granted relief are of a similar nature to the modifications and limitations in 10 CFR 50.55a(b)(2) for the 1998 Edition through the 2000 Addenda of the Code. The LSCS commitments and the NRC approval are documented in References 1, 2, 3 and 4.

Question 2
The staff requests the licensee to provide a summary of significant findings (pits in excess of 10% of the nominal liner thickness) in the submerged areas of the wet-well.

Response 2
No significant findings (i.e., pits in excess of 10% of the nominal liner thickness) have been observed in the submerged areas of Suppression Pool. It has been determined that there is no loss of section or structural integrity.

Question 3
The staff requests the licensee to provide a summary of findings of the examination of the containment concrete performed in accordance with 10CFR 50.55a and Subsection IWL including the acceptance criteria used for concrete and reinforcing bar degradation.

Response 3
During the initial IWL inspections of the Units 1 and 2 concrete containments, various indications were observed, documented and evaluated. All findings were determined to be acceptable and no loss of structural integrity of containment was observed. The following provides a summary of the inspection findings and acceptance criteria.

Unit 1 Inspection Findings

- **Cracks in Concrete and Coatings** – A majority of the cracking observed in the walls were horizontal cracks, with some radial cracking around penetrations. Both crack patterns are normal shrinkage cracks for concrete walls. The cracks reported ranged from 4 inches long to 14 feet long by 0.002 inch to 0.03 inch wide and determined to be acceptable.
- **Staining of Concrete** – Minor staining was observed on the containment walls and floors. The staining was the result of an indeterminate source and determined to be acceptable.
- **Concrete Spalling, Popouts and Voids** – Various concrete spalls and popouts were observed. The spalls ranged from 4 to 8 inches long by 3/8 to 2 inches deep. The popouts ranged from ¼ to 2 inches in diameter by ½ to 2-1/8 inches deep. The spalls and popouts were determined to be acceptable.
- **Coating Deterioration** – Containment coating was found to be in generally good condition. Deterioration was in the form of chipping which appeared to be from external damage, possibly scaffold erection. All deteriorations were determined to be acceptable.

Unit 2 Inspection Findings

- **Cracks in Concrete** – Horizontal cracks were the majority of the cracks observed with some pattern/vertical cracks. These cracks are normal shrinkage cracks for concrete walls. One significant crack (3/4 inch wide) observed in a corbel but does not impact the capacity of the containment or reactor building was recommended for repair to prevent reinforcing steel degradation. The cracks reported ranged from 1-1/2 inch to 20 feet long by 0.015 to 0.030 inch wide and determined to be acceptable.
- **Concrete Spalling, Popouts and Voids -** Various concrete spalls and popouts were observed. The maximum spall dimension observed was ½ inch deep and 3 feet in length. The maximum void dimension observed was 1-1/2 inch deep and 2 inch in diameter. All spalls, popouts and voids were determined to be acceptable.
- **Concrete Coating and Staining** – Containment coating was found to be in generally good condition. Peeling and cracking of coating was observed to be minimal. Staining of the concrete walls due to minor grease leakage from tendon cans was subsequently cleaned. The coating and staining were determined to be acceptable.

Acceptance Criteria

The acceptance criteria used for concrete are in accordance with Subsection IWL-3000 of the ASME code. Additionally as discussed in the NRC safety evaluation in Reference 4, LSCS's general and detailed visual examinations procedures have been developed from the guidelines of ACI 201.1R-92 "Guide For Making a Condition Survey of Concrete in Service," and ACI 349.3R-96, "Evaluation of Existing Nuclear Safety-Related Concrete Structures." These procedures follow a tiered approach for recording of concrete degradation. If the recording criteria are exceeded further review is required by the Responsible Engineer.

The LSCS Responsible Engineer found the condition of the concrete surface acceptable with no evidence of damage or degradation sufficient to warrant further evaluation. Thus, no inspection of any reinforcing bar was performed.

Question 4
Recognizing the hardship associated with examining seals, gaskets, and pressure retaining bolts during each inspection periods, and that the examination will be performed prior to Type B testing as required by Option A of Appendix J, the staff had approved such relief to a number of licensees. However, implementation of Option B of Appendix J allows flexibility in performing Type B testing based on leak rate performance of the penetrations. Because the performance-based testing allows certain leak rate through the penetrations, minor initial degradation of the associated seals, gaskets and bolting can go undetected, and the 10-year examination interval could be too long for the degraded components. Thus, examination of the seals, gaskets and pressure retaining bolting should be scheduled based on their performance (i.e., plant-specific experience, replacement scheduled for resilient seals, etc.) to ensure that, if Type B testing is not performed during the ILRT extension period, the examination schedule will detect degradation of these components. In view of this discussion staff request the licensee to provide a schedule for examination (testing) of these components; especially for equipment hatches and other penetrations with resilient seals.

Response 4
The LSCS scheduling rules as allowed by Option B of Appendix J are as follows.

- The initial test frequency for performing a leak test on seals and gaskets, which are Type B components, is a base interval of 30 months. The interval may be extended to up to 120 months based on acceptable performance. Acceptable performance for extending this interval is established by passing two as-found LLRTs with leakage less than or equal to the established administrative limits and that are at least 24 months apart or a normal refueling interval. Type B components whose test intervals are extended to greater than 60 months are tested on a staggered basis to allow for early detection of common mode failure mechanism.
- If a test result is greater than the administrative limit for the components, the component is restored to a leak rate below the administrative limit and the test interval is re-established at 30 months.

Additionally, any repair or disassembly of a component with a seal, gasket, or bolted connection requires a post-maintenance Appendix J Type B test. The proposed technical specification amendment does not affect the current examination schedule of these components.

Question 5
Based on the write-up in Section 5.6 of Attachment 2 of the submittal, the staff assumes that there are no containment pressure boundary bellows at LaSalle Unit 1 and 2. The staff requests the licensee to confirm this assumption.

Response 5
LSCS does not have any bellows that act as part of the containment boundary.

Question 6
The October 24, 2002 submittal provides risk impacts for a change in test frequency from 1 test in 10 years to 1 test in 15 years. The staff requests the licensee to provide the corresponding risk results (for Δperson-rem, ΔLERF, and ΔCCFP) for a change in test frequency from 3 tests in 10 years to 1 test in 15 years.

Response 6
The increase in Large Early Release Frequency (LERF) from the 3-in-10 year ILRT interval to the 1-in-15 year interval was determined to be 2.98E-08/yr, which is below the NRC Regulatory Guide 1.174 criterion of 1.0E-7/yr for "very small" risk changes. In addition, the dose increase was determined to be 7.73E-02 person-rem/yr, which is 1.5% above the 3-in-10 year value of 5.1039 person-rem/yr. The increase in the containment failure probability (CCFP) was determined to be insignificant (i.e., 84.0% for the 1-in-15 year case versus 83.5% for the 3-in-10 year case).

When the risk assessment is considered using either the 3-in-10 yr. interval or the 1-in-10 yr. interval as the reference point, the conclusion of the assessment does not change; that is, the LSCS ILRT interval extension to 1-in-15 yr. has a minimal impact on public risk.

Question 7
The staff requests the licensee to provide the technical justification for the assumption in the risk analysis that no long-term station blackout scenarios contribute to LERF. If this justification is based on timing arguments, provide a timeline for a representative scenario that includes consideration of the time at which the various emergency action levels are declared, the decision to evacuate is made, and the evacuation is initiated and completed. If this justification is based on source term magnitude, provide the estimated source terms for a representative scenario, and the definition of LERF used for this determination.

Response 7
The LSCS long-term station blackout core damage accidents (i.e., Class IBL) result in non-LERF releases based on release timing and not on release magnitude (i.e., LaSalle IBL core damage accidents have the potential to result in the entire spectrum of release magnitudes,

including High magnitude releases; but, they can not result in early releases). The following discussion focuses on the timing issues of Class IBL scenarios.

Typical of many industry PRAs, the LSCS PRA uses a radionuclide release categorization scheme comprised of two factors: release timing and release magnitude. Three timing categories are used, as follows:

- Early (E) Less than 6 hours
- Intermediate (I) Greater than or equal to 6 hours, but less than 24 hours
- Late (L) Greater than or equal to 24 hours.

The above accident release categories are based upon past experience concerning offsite accident response:

- 0-6 hours is conservatively assumed to include cases in which minimal offsite protective measures have been observed to be performed in non-nuclear accidents.
- 6-24 hours is a time frame in which much of the offsite nuclear plant protective measures can be assured to be accomplished.
- >24 hours are times at which the offsite measures are assumed to be effective.

The timing categories are relative to the declaration of the LaSalle General Emergency Action Level.

The LSCS IBL accident scenarios include only those sequences in which high pressure injection (i.e., Reactor Core Isolation Cooling) is available initially in the accident but subsequently fails. The representative IBL sequence for LSCS is sequence LOOP-17 of the LOOP event tree. Sequence LOOP-17 proceeds as follows.

Event	Time After Plant Trip
- Loss of Offsite Power initiating event	0
- Failure of emergency AC power (EDGs)	0
- Failure of HPCS	0
- RCIC Initiation	~1 min.
- RPV/containment parameters exceed HCTL curve	7 hrs.
- Battery depletion	7 hrs.
- Failure to blowdown (no DC power)	7 hrs.
- Loss of RCIC (all) injection	7 hrs.
- Time for RPV level to drop to TAF	8.8 hrs.
- Time to core damage (1800F)	9.9 hrs.
- Time to energetic containment failure (fastest, but low frequency, release scenario)	~12 hrs.

As can be seen from the above scenario, the LaSalle IBL accident class results in a radionuclide release no earlier than 10 hours after the LOOP initiator. The 12 hour release for the IBL core damage accident makes the conservative assumption that an early energetic containment failure mode (in-vessel corium-steam explosion) occurs at about the time of core melt and dislocation to the lower head (a low probability containment failure mode for the IBL accident).

LaSalle procedure EP-AA-1005 (Recognition Category MG1) directs declaration of a General Emergency (i.e., the emergency classification with associated directives for evacuation) for the following station blackout conditions:

- Loss of power from TR-241 and TR-242,
- Emergency diesel generators fail to supply power to buses 241Y and 242Y, and
- Restoration of power to bus 241Y or 242Y within 4 hours is judged NOT likely.

The loss of offsite and emergency power to buses 241Y and 242Y occurs at t=0 for sequence LOOP-17. The LSCS PRA assumes that the determination that AC power is not likely to be restored in the 4-hour time frame is made at approximately 1 hour into the accident. As such, a General Emergency is declared at 1 hour into the event. The evacuation process would be initiated within minutes after the declaration, because LaSalle procedure EP-AA-111 states that the local authorities must be notified within 15 minutes after the General Emergency declaration. It is likely that the evacuation process would be completed within 4 hours based on site specific evacuation studies for weather and times of day variations. The earliest possible release for the IBL scenario occurs at approximately 12 hours (i.e., approximately 7 hours after evacuation is expected to be completed). Therefore, the IBL core damage accident is not an Early release.

Question 8
The staff requests the licensee to provide an assessment of the impact on risk results (Δperson-rem, ΔLERF, and ΔCCFP) if long-term station blackouts were not removed from the residual core damage frequency when determining the Category 3a and 3b frequencies.

Response 8
Including long-term station blackout (SBO) scenarios in the Electric Power Research Institute (EPRI) Category 3a and 3b frequency calculations would not be typical or consistent with the Nuclear Energy Institute (NEI) ILRT risk assessment methodology, but those evaluations are performed here in response to this question. The results are shown in Table 8-1.

The increase in LERF from the 3-in-10 year ILRT interval to the 1-in-15 year interval was determined to be 4.05E-08/yr, which remains below the NRC Regulatory Guide 1.174 criterion of 1.0E-7/yr for "very small" risk change. In addition, the dose increase was determined to be 1.05E-01 person-rem/yr, which is 2% above the 3-in-10 year value of 5.1108 person-rem/yr. The increase in the containment failure probability (CCFP) was determined to be insignificant (84.2% for the 1-in-15 year case versus 83.5% for the 1-in-3 year case). Note that these sensitivity case values do not include the concealed containment flaw sensitivity of RAI #9; however, if that issue were incorporated into this sensitivity case the results would still remain reflective of a "very small" risk change.

Including long term SBO scenarios in the EPRI Category 3a and 3b frequencies does not change the conclusion of the risk assessment; that is, the LSCS ILRT interval extension to 1-in-15 yr. has a minimal impact on plant risk.

Question 9

Inspections of some reinforced and steel containments (e.g., North Anna, Brunswick, D. C. Cook, and Oyster Creek) have indicated degradation from the uninspectable (embedded) side of the steel shell and liner of primary containments. The staff requests the licensee to describe the uninspectable areas of the LaSalle containments, and the programs used to monitor their condition. Provide a quantitative assessment of the impact on LERF due to age-related degradation in these areas, in support of the requested ILRT interval extension from 10 to 15 years.

Response 9

The liner sections are completely welded together and anchored into the concrete. There is no air space between the liner and the concrete structure. The corrosion / oxidation effects associated with water being in contact with the carbon steel liner and the concrete reinforcing bars are minimized due to the lack of available oxygen between the concrete and the liner. Furthermore, the liner is intended to be a membrane and constitute a leak-proof boundary for the drywell. The liner is nominally ¼" thick and has been oversized to serve as formwork for concrete pouring during plant construction.

A separate risk assessment has been performed regarding the potential for containment leakage due to age-related degradation in non-inspectable areas and the impact of this potential issue on the LaSalle ILRT interval risk assessment results. This analysis was performed using the same approach used by other industry plants (e.g., Calvert Cliffs) to respond to similar NRC RAIs. The results of this analysis are that the increase in LERF due to extending the ILRT interval from 3-in-10 years to 1-in-15 years is 3.29E-8/yr, of which 3.09E-9/yr is due to corrosion. This value is well below the threshold for "very small" changes in risk. Additionally, a series of parametric sensitivity studies regarding the potential age related corrosion effects on the steel liner indicate that even with very conservative assumptions, the conclusions from the original analysis would not change.

Key assumptions in this assessment were as follows.

- A half failure is assumed for basemat concealed liner corrosion due to the lack of identified failures. Assuming 0.5 failures when 0 failures have occurred is a typical PRA approach.
- The two corrosion events used to estimate the liner flaw probability in the Calvert Cliffs analysis are assumed to be applicable to the LaSalle containment analysis. These events, one at North Anna Unit 2 and one at Brunswick Unit 2, were initiated from the non-visible (backside) portion of the containment liner.
- The estimated historical flaw probability is calculated using a 5.5 year data period to reflect the years since September 1996 when 10 CFR 50.55a started requiring visual inspection. Additional success data were not used to limit the aging impact of this corrosion issue, even though inspections were being performed prior to this date and there is no evidence that additional corrosion issues were identified.

- The corrosion-induced steel liner flaw likelihood is assumed to double every five years. This is based solely on judgment and is included in this analysis to address the increased likelihood of corrosion as the steel liner ages.

- In the Calvert Cliffs analysis, the likelihood of the containment atmosphere reaching the outside atmosphere given that a liner flaw exists was estimated (based on an assessment of the containment fragility curve versus the ILRT test pressure) as 1.1% for the containment walls and dome region and 0.11% (10% less) for the basemat. For LaSalle the containment failure probabilities are conservatively assumed to be 10% for the drywell and wetwell outer walls, and since the basemat for the LaSalle Mark II containment is in the suppression pool, it is judged that failure of this area would not lead to LERF. In any event, a 1% probability is assigned as a conservatism.
- A 5% visual inspection detection failure likelihood given the flaw is visible and a 5% likelihood of a non-detectable flaw is used. Therefore, a total undetected flaw probability of 10% is assumed in the base case analysis. Again, this is considered conservative since essentially 100% of the LaSalle containment interior surface is visible, whereas only 85% of the interior wall surface was estimated as being visible at Calvert Cliffs. Additionally, it should be noted that to date, all liner corrosion events have been detected through visual inspection and repaired.
- All non-detectable containment failures are assumed to result in early releases. This approach avoids a detailed analysis of containment failure timing and operator recovery actions.

References

1. Letter from R. M. Krich (ComEd) to NRC, "Request for Inservice Inspection Program relief Regarding Containment Inspections By Approved Alternate Means," dated May 8, 2000.
2. Letter from R. M. Krich (ComEd) to NRC, "Response to Request for Additional Information Concerning Inservice Inspection Program Relief Regarding Containment Inspections By Approved Alternate Means," dated August 18, 2000.
3. Letter from R. M. Krich (ComEd) to NRC, "Supplemental Response to Request for Additional Information Concerning Inservice Inspection Program Relief Regarding Containment Inspections By Approved Alternate Means," dated August 30, 2000.
4. Letter from A. J. Mendiola (NRC) to O. D. Kingsley (ComEd), "Byron, Dresden and LaSalle – Evaluation of Relief Requests: Use of 1998 Edition of Subsections IWE and IWL of the ASME Code for Containment Inspection (TAC Nos. MA8933, MA8934, MA8935, MA8936, MA8937 and MA8938)," dated September 18, 2000.

Table 8-1

QUANTITATIVE RESULTS AS A FUNCTION OF ILRT INTERVAL
- Sensitivity Case for RAI #8, Include Long-Term SBO Contributions in Category 3a and 3b Frequencies -

EPRI Category	Dose (Person-Rem Within 50 miles)	Baseline (3-per-10 year ILRT)		Current (1-per-10 year ILRT)		Proposed (1-per-15 year ILRT)	
		Accident Frequency (per year)	Population Dose Rate (Person-Rem/Year Within 50 miles)	Accident Frequency (per year)	Population Dose Rate (Person-Rem/Year Within 50 miles)	Accident Frequency (per year)	Population Dose Rate (Person-Rem/Year Within 50 miles)
1	2.09E+04	**8.31E-07**	**1.74E-02**	**5.72E-07**	**1.20E-02**	**3.86E-07**	**8.08E-03**
2	1.24E+06	3.22E-09	3.98E-03	3.22E-09	3.98E-03	3.22E-09	3.98E-03
3a	2.09E+05	**1.01E-07**	**2.12E-02**	**3.37E-07**	**7.05E-02**	**5.06E-07**	**1.06E-01**
3b	7.32E+05	**1.01E-08**	**7.40E-03**	**3.37E-08**	**2.47E-02**	**5.06E-08**	**3.70E-02**
4	n/a	n/a	n/a	n/a	n/a	n/a	N/a
5	n/a	n/a	n/a	n/a	n/a	n/a	N/a
6	n/a	n/a	n/a	n/a	n/a	n/a	n/a
7a	1.07E+06	1.21E-06	1.30E+00	1.21E-06	1.30E+00	1.21E-06	1.30E+00
7b	1.05E+06	2.52E-06	2.65E+00	2.52E-06	2.65E+00	2.52E-06	2.65E+00
7c	1.05E+06	4.63E-07	4.88E-01	4.63E-07	4.88E-01	4.63E-07	4.88E-01
7d	1.24E+06	1.97E-07	2.43E-01	1.97E-07	2.43E-01	1.97E-07	2.43E-01
7e	1.14E+06	2.54E-07	2.89E-01	2.54E-07	2.89E-01	2.54E-07	2.89E-01
8	1.24E+06	7.12E-08	8.79E-02	7.12E-08	8.79E-02	7.12E-08	8.79E-02
TOTALS:		5.66E-06	**5.1108**	5.66E-06	**5.1720**	5.66E-06	**5.2157**
Increase in Dose Rate (1)							**4.05E-08**
Increase in LERF (2)							**0.1049**
Increase in $CCFP_{\%}$ (3)							**0.7**

(1) The Increase in Dose Rate (person-rem/year) is the delta between the 3-in-10 yr case and the 1-in-15 yr case.
(2) The Increase in LERF is the delta between the 3-in-10 yr case and the 1-in-15 yr case, and is calculated by subtracting the EPRI Category 3b frequencies.
(3) The Increase in CCFP% (units in percentage points) is the delta between the 3-in-10 yr case and the 1-in-15 yr case. The CCFP% is calculated as:

$$CCFP_{\%} = [1 - ((\text{Category 1 Frequency} + \text{Category 3a Frequency}) / \text{CDF})] \times 100$$