Commonwealth Edison
1400 Opus Place
Downers Grove, Illinois 60515

December 6, 1993

Mr. John Zwolinski
Assistant Director for Projects
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: LaSalle County Nuclear Power Station Unit 1
Request for NRR Enforcement Discretion Regarding Facility Operating License NPF-11, Appendix A, Technical Specification 4.0.3
NRC Docket No. 50-373

Dear Mr. Zwolinski:

This letter documents the results of the teleconference held on December 4, 1993, between Commonwealth Edison (CECo) and the NRC Staff, in which Commonwealth Edison requested a Notice of Enforcement Discretion from Technical Specification 4.0.3 for LaSalle County Unit 1.

At 1715 on December 4, 1993, LaSalle County Unit 1 entered Technical Specification 3.4.2. Action a., due to the inoperability of two (2) Safety/Relief Valves. These two (2) Safety/Relief valves were conservatively determined to be inoperable per Technical Specification 4.0.3 due to the time period since verification of the lift settings for valves 1B21-F013B and 1B21-F013J. Technical Specification 4.0.3 states "Failure to perform a Surveillance Requirement within the specified time interval shall constitute a failure to meet the OPERABILITY requirements for a Limited Condition for Operation." Technical Specification 4.0.5 provides surveillance requirements for inservice inspection and testing of ASME Code Class 1, 2, and 3 components per ASME Section XI of the Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a(g). The reactor coolant system Safety/Relief Valves are included in the ASME Code components to be tested. The surveillance intervals for the Safety/Relief Valves are based upon ASME Section XI Table IWV-3510-1. Safety/Relief Valves 1B21-F013B and 1B21-F013J exceeded the specified surveillance interval requirements (every 5 years) and were conservatively determined to be inoperable.

Because it is not possible to perform the required testing with Unit 1 in operation and the reactor coolant system pressurized, CECo requested that Unit 1 be allowed to continue to operate until the March 1994 Refueling Outage (L1R06) or until the next Cold Shutdown, whichever occurs first. A Notice of Enforcement Discretion was verbally approved by NRR at 2025 CST on December 4, 1993.

The basis of the request is provided in Attachment 1 and includes:

- The Technical Specification that will be violated;

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- The circumstances surrounding the condition, including the need for prompt action;
- The safety basis for the request that enforcement discretion be exercised, including an evaluation of the safety significance and potential consequences of the proposed course of action;
- Any proposed compensatory measure(s);
- The justification for the duration of the request;
- The basis for the conclusion that the request will not have a potential adverse impact on the public health and safety and that a significant safety hazard is not involved;
- The basis for the conclusion that the request will not involve adverse consequences to the environment.

If during duration of this Notice of Enforcement Discretion additional Safety/Relief valves are made or found to be inoperable for any reason, CECo will follow the Action requirements of the Technical Specifications. CECo requests that this Notice of Enforcement Discretion be in effect until an Exigent Technical Specification Amendment is approved. A request for an Exigent Technical Specification Amendment will be submitted on December 10, 1993, for NRC Staff review.

This request for Enforcement Discretion has been reviewed and approved by the LaSalle County On-Site Review Committee, in accordance with LaSalle County Station procedures.

CECo sincerely appreciates the NRC staff's effort and participation in the review of this request. Please direct any questions or comments to Peter Piet, Nuclear Licensing Administrator, at (708) 663-7286.

Very truly yours,

Peter L. Piet
Nuclear Licensing Administrator

Attachment

cc: J. B. Martin, Regional Administrator - RIII
D. Hills, Senior Resident Inspector - LaSalle County
A. Gody, Project Manager - NRR
NRC Document Control Desk

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ATTACHMENT

1. TECHNICAL SPECIFICATION OR LICENSING CONDITION THAT WILL BE VIOLATED

At 1715 on December 4, 1993, LaSalle County Unit 1 entered Technical Specification 3.4.2, Action a., due to the inoperability of two (2) Safety/Relief Valves. These two (2) Safety/Relief valves were conservatively determined to be inoperable per Technical Specification 4.0.3 due to the time period since verification of the lift settings for valves 1B21-F013B and 1B21-F013J. Technical Specification 4.0.3 states "Failure to perform a Surveillance Requirement within the specified time interval shall constitute a failure to meet the OPERABILITY requirements for a Limited Condition for Operation." Technical Specification 4.0.5 provides surveillance requirements for inservice inspection and testing of ASME Code Class 1, 2, and 3 components per ASME Section XI of the Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a(g). The reactor coolant system Safety/Relief Valves are included in the ASME Code components to be tested. The surveillance intervals for the Safety/Relief Valves are based upon ASME Section XI Table IWV-3510-1. Safety/Relief Valves 1B21-F013B and 1B21-F013J exceeded the specified surveillance interval requirements (every 5 years) and were conservatively determined to be inoperable. It is not possible to perform the required testing with Unit 1 in operation and the reactor coolant system pressurized.

Therefore, Commonwealth Edison requests Enforcement Discretion from Technical Specification 4.0.3 for SRVs 1B21-F013B and 1B21-F013J be granted to allow continued unit operation until approval of an exigent Technical Specification amendment. The amendment will request that valves 1B21-F013B and 1B21-F013J be exempted from the requirements of Technical Specification 4.0.3 until the sixth refueling outage or the first Cold Shutdown prior to L1R06, whichever comes first. In addition, the LCO for Technical Specification 3.4.2 will be modified to require all 18 SRVs to be OPERABLE, with the exemption to 4.0.3 for SRVs 1B21-F013B and 1B21-F013J.

LaSalle County Station (LaSalle) Unit 2 has been reviewed and each SRV has been tested within the last 5 years.

2. CIRCUMSTANCES SURROUNDING THE SITUATION

LaSalle's Inservice Testing Program is based on ASME Section XI 1980 Edition, Winter 1980 Addenda. Subsection IWV, Table IWV-3510-1 of Section XI requires "...that at each refueling all valves which have not been tested during the preceding 5 year period shall be tested." LaSalle Station's Engineering Department discovered that two (2) Unit 1 Main Steam Safety/Relief Valves did not meet this requirement coming out of the unit's fifth refuel outage in January of 1993. The two SRVs, 1B21-F013B

and 1B21-F013J, were last setpoint tested during the first Unit 1 refuel outage (L1R01), which ended in October, 1986. Thus, the 2 valves have not been setpoint tested in approximately 7 years. SRVs 1B21-F013B and 1B21-F013J are presently scheduled to be tested during the sixth Unit 1 refuel (L1R06) in March 1994.

During previous cycles, it was LaSalle's interpretation that each SRV required testing once within each fixed five (5) year period. Each SRV was setpoint tested during the first fixed 5 year period, January 1984 to January 1990. (This includes time from the end of the 5 year interval until the next scheduled Refuel.) The end of the third Unit 1 refuel (L1R03) in January 1990 marked the start of the second fixed 5 year period. LaSalle is presently on schedule to have each SRV setpoint tested again by the end of this period. The testing above meets LaSalle's original interpretation.

Recent review of Table IWV-3510-1 has rendered a different interpretation. This interpretation concludes that when coming out of a refueling outage, each installed SRV must have been tested within the previous 5 year span, irregardless of the particular fixed 5 year period for the unit. Because the test sequencing was different within the two fixed 5 year periods, the most recent setpoint test of SRVs 1B21-F013B and 1B21-F013J do not fall within the 5 year envelope preceding the fifth Unit 1 refuel (L1R05).

3. <u>EVALUATION OF SAFETY SIGNIFICANCE AND CONSEQUENCES</u>

The possible effects of the safety setpoint drift are:

1) Setpoint drift HIGH (delayed valve opening in the safety mode).

2) Setpoint drift LOW (early safety mode opening).

EVALUATION OF CASE 1) SETPOINT DRIFT HIGH:

The bounding transients for LaSalle Station were reviewed by General Electric. The review showed that the limiting MCPR transient (Turbine Load Reject without Bypass, or LRNBP) is not affected because the minimum MCPR is reached prior to reactor pressure reaching the lowest SRV Relief Setpoint (1076 psig). The affected SRVs are in the two highest actuation groups and therefore cannot impact the results of this event. The other (non limiting) MCPR events did not require re-evaluation for the potential of becoming limiting for the following reasons:

1. The non-limiting pressurization events (Turbine Trip without Bypass, Feedwater Controller Failure without Bypass) add reactivity (and lost MCPR margin) by the same physical mechanism as the LRNBP. Therefore, these events will also experience the minimum MCPR prior to the first SRV relief setpoint, and the possible Safety Mode setpoint drift

cannot affect the outcome of the event analyses.

2. Non-pressurization MCPR events (Loss of Feedwater Heaters, and Rod Withdrawal error) do not result in loss of the turbine pressure regulator, and do not involve SRV operation. Therefore SRV setpoint drift cannot affect the outcome of the event analyses.

EOC RPT with one SRV Out-of-Service (OOS) is analyzed for both LaSalle Units. This analysis assumes the lowest pressure SRV does not actuate, and its response is only a function of relief mode operation. The effect of safety mode setpoint drift does not affect the relief mode of operation.

The limiting pressurization transient for vessel over pressurization is the MSIV closure with flux scram (which assumes MSIV limit switch scram fails). For this event, the peak reactor pressure of 1266 psig is only slightly impacted (by less than 10 psig), because the remaining relief capacity is adequate. The ASME pressure limit of 1375 psig for the RPV bottom head (1325 psig steam dome pressure) is not exceeded.

EVALUATION OF CASE 2) SETPOINT DRIFT LOW:

For the second affect (setpoint drift low), the possible concerns are:

1) Drift to below operating pressure and initiation of stuck open SRV accident, and

2) Complication of an event which utilizes safety mode.

The historical data for Safety Mode setpoint drift was reviewed. LaSalle has 18 Crosby Safety/Relief Valves, Style HB-65-BP and Size 6xRx10, installed on each unit. In addition to these 36 valves, LaSalle also has 9 spare valves of the same style and size. The valves have safety set pressures of 1150 psig, 1175 psig, 1185 psig, 1195 psig, or 1205 psig. Of 76 test points from the Unit 1 and Unit 2 SRV populations, there have been a total of 19 failures to meet the setpoint criteria of +1% / -3% of lift pressure. The data is evenly distributed between the units and evenly distributed between positive drift and negative drift data. There also does not appear to be a correlation to age or life of plant for either the frequency or magnitude of setpoint drifts. Therefore, no indications of a systematic drift factor are present, and the performance of the two subject valves is expected to reflect the random variations experienced in the population so far.

To check for the potential for an exaggerated drift due to the extended surveillance interval, a possible drift of twice the worst observed setpoint drift was postulated. This would be approximately 8% of setpoint, and for the SRV 1B21-

FO13B, a possible drift down from 1195 (nominal setpoint) to 1099 psig might occur. SRV 1B21-FO13J has a higher setpoint and greater margin for downward setpoint drift. This is above the unit operating pressure (nominally 1005 psig), and above the lowest pressure relief setpoint (1076) psig. Therefore, a decreasing setpoint condition that induces the Stuck Open SRV Accident is not considered credible.

The relief mode was exercised during the Unit 1 scram dated September 14, 1993. During the scram, reactor pressure reached approximately 1075 psig. Neither SRV 1B21-FO13B or 1B21-FO13J experienced safety mode actuations at this pressure. This verifies that their safety mode setpoint did not drift below this pressure, and supports the expectation that a stuck open SRV accident is not credible (due to the concern of safety setpoint drift).

SRV 1B21-FO13B was manually opened twice during that event and SRV 1B21-FO13J was manually opened once. Both operations were satisfactory, indicating that operation and capability of SRV 1B21-FO13B and 1B21-FO13J are not impaired by any other factors.

Four SRVs exhibited indication of actuator air leaks during this event. None of these problems included safety mode anomalies or failures, and neither SRV 1B21-FO13B or 1B21-FO13J exhibited problems.

The potential for complication or changes to the progression of events that would not reach pressures as high as the Safety Mode setpoints of SRV 1B21-FO13B or 1B21-FO13J is minimal. The primary event of interest is the MSIV closure with flux scram, which is the only event analyzed each cycle that reaches pressures above the lowest Safety setpoint group (1146 psig). MSIV closure with flux scram opens all SRVs, so early actuation of 1B21-FO13B or 1B21-FO13J would lessen the severity of the pressurization event. If (downward) setpoint drift occurred into the region of relief valve operations (the first group to be impacted would be the highest relief group at 1116 psig), the SRVs would have been expected to be open in the relief mode already (the relief mode setpoint for SRV 1B21-FO13B is 1106 psig, and for SRV 1B21-FO13J is 1116 psig).

No other parameters (i.e., reactivity, cooldown rates, etc) are expected to be affected by setpoint drift in the downward direction.

4. <u>COMPENSATORY ACTIONS</u>

The condition of concern affects only the safety function setpoints of SRVs 1B21-FO13B and 1B21-FO13J, and does not involve either their relief mode operation, indications, or possible system actuation effects.

The following Compensatory Actions will be placed in effect:

1) Licensed Operators will be trained, prior to assuming the shift beginning December 6, 1993, that for these 2 SRVs, safety mode actuations are potentially affected by setpoint drift.

2) Caution cards will be placed on the control panel, by December 6, 1993, to reinforce the information that safety actuation pressures should be verified during appropriate conditions.

5. JUSTIFICATION FOR THE DURATION OF THE REQUEST:

The surveillance intervals are provided to periodically verify, to the extent possible, that the surveilled component will perform its desired safety function when required. Typically these surveillance tests verify that the component is indeed performing or capable of performing its required safety function. The failure to perform a surveillance on a component does not, in itself, make the equipment unable to perform its function. In this specific case the surveillance requirement is to verify the calibration (safety valve function lift setting) prior to startup from a refueling outage where the last setpoint test exceeds 5 years. The startup from the last outage in which the two safety/relief valves were tested occurred in October 1986, approximately 7 years ago. The next opportunity to perform this test, which requires the reactor to be in cold shutdown, would be the spring 1994 refueling outage, L1R06, presently scheduled for March 1994 or the first cold shutdown prior to L1R06, whichever comes first. The maximum total length of time since the last setpoint test, therefore, will be approximately 7.5 years.

6. EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATION

Commonwealth Edison has evaluated the proposed Technical Specification Amendment and determined that it does not represent a significant hazards consideration. Based on the criteria for defining a significant hazards consideration established in 10 CFR 50.92, operation of LaSalle County Station Unit 1 in accordance with the proposed amendment will not:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated because:

   a. There is no affect on accident initiators so there is no change in probability of an accident. The probability of a failed open Safety/Relief Valve (SRV) is not affected based on observed performance of setpoint drift.

   b. There is no effect or minimal affect on the consequences of analyzed accidents based on an evaluation that the highest reactor vessel pressure that will occur is still less than the Safety Limit of

1325 psig steam dome pressure, for the bounding vessel pressurization event. This evaluation assumed that both SRVs 1B21-FO13B and 1B21-FO13J fail to open.

2) Create the possibility of a new or different kind of accident from any accident previously evaluated because:

The SRVs are not being used in any other mode than original design. The only affect is from the safety mode setpoint drift. This issue does not involve any plant modifications or changes to operating procedures. Therefore, this issue does not create the possibility of a new or different kind of accident from any previously evaluated accident.

3) Involve a significant reduction in the margin of safety because:

The review of previous sensitivity analyses for peak accident pressure indicates that extension of the surveillance intervals for SRV 1B21-FO13B and 1B21-FO13J can not result in exceeding the Safety Limit reactor pressure of 1325 psig steam dome pressure. Therefore, this issue does not involve a significant reduction in the margin of safety.

Guidance has been provided in "Final Procedures and Standards on No Significant Hazards Considerations," Final Rule, 51 FR 7744, for the application of standards to license change requests for determination of the existence of significant hazards considerations. This document provides examples of amendments which are and are not considered likely to involve significant hazards considerations. These proposed amendments most closely fit the example of a change which may either result in some increase to the probability or consequences of a previously analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the applicable Standard Review Plan.

This proposed amendment does not involve a significant relaxation of the criteria used to establish safety limits, a significant relaxation of the bases for the limiting safety system settings or a significant relaxation of the bases for the limiting conditions for operations. Therefore, based on the guidance provided in the Federal Register and the criteria established in 10 CFR 50.92(c), the proposed change does not constitute a significant hazards consideration.

7. ENVIRONMENTAL ASSESSMENT

LaSalle County Station has evaluated the proposed enforcement discretion against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.20. It has been determined that the proposed changes meet the criteria for a categorical exclusion as provided under 10 CFR 51.22(c)(9). This conclusion has been determined because the changes requested do not pose significant hazards considerations or do not involve a significant increase in the amounts, and no significant changes in the types, of any effluents that may be released off-site. Additionally, this request does not involve a significant increase in individual or cumulative occupational radiation exposure.

8. APPROVAL BY ON-SITE REVIEW

The request has been approved by LaSalle County Senior Station Management and On-Site Review (OSR) in accordance with Station procedures.