Commonwealth Edison
1400 Opus Place
Downers Grove, Illinois 60515

December 10, 1993

Dr. Thomas E. Murley, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attn: Document Control Desk

SUBJECT: LaSalle County Nuclear Power Station Unit 1
Request for Exigent Technical Specification Amendment, to Appendix A, Technical Specification 3.4.2, "Safety Relief Valves" of Facility Operating License NPF-11
NRC Docket 50-373

REFERENCE: Letter dated December 6, 1993 from P.L. Piet to J. Zwolinski, LaSalle County Nuclear Power Station Unit 1, Request for Enforcement Discretion Regarding Facility Operating License NPF-11, Appendix A, Technical Specification 4.0.3.

Dear Dr. Murley:

In the above Reference, Commonwealth Edison (CECo) requested a Notice of Enforcement Discretion (NOED) from Unit 1 Technical Specification 4.0.3, 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. As stated in the Reference, the NOED was verbally approved at 2025 CST on December 4, 1993. Also stated in the reference was a CECo commitment that an Exigent Technical Specification Amendment would be transmitted to NRR no later than December 10, 1993. This document fulfills that commitment. The NOED will remain valid until approval of the Exigent Technical Specification Amendment.

Pursuant to 10 CFR 50.91(a)(6), Commonwealth Edison Company (CECo) proposes to amend Appendix A, Technical Specifications, of Facility Operating Licenses NPF-11 and requests that the Nuclear Regulatory Commission (NRC) grant an exigent amendment to Technical Specification 3.4.2, "Safety Relief Valves". This exigent Technical Specification Amendment will add a footnote to exempt Safety/Relief Valves 1B21-F013B and 1B21-F013J from the requirements of Technical Specification 4.0.3 (with respect to the safety valve function lift setting setpoint test frequency specified in ASME Code Section XI, Table IWV-3510-1) until Unit 1 enters Cold Shutdown at the end of Unit 1 cycle six or the next cold shutdown, 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.

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An exigent change is needed and could not be avoided due to the short Technical Specification timeclock that resulted from the discovery of the two valves lift settings not being verified. If the NOED would not have been granted, LaSalle Unit 1 would have been required to be in Hot Shutdown whithin 12 hours and in Cold Shutdown within the following 24 hours. Therefore, this condition was not created by the failure to make a timely application for a Technical Specification Ammendment. Approval of this exigent change will allow Unit 1 to continue power operation. This provides a net safety benefit by not requiring the unit to shutdown and undergo unnecessary thermocycles on plant equipment and any associated challenges to safety systems.

This proposed exigent amendment request is subdivided as follows:

1. Attachment A gives a description and safety analysis of the proposed changes in this amendment.

2. Attachment B includes a summary of the proposed changes and the marked-up Technical Specifications pages for LaSalle Unit 1 with the requested changes indicated.

3. Attachment C describes CECo's evaluation performed in accordance with 10 CFR 50.92 (c), which confirms that no significant hazard consideration is involved.

4. Attachment D provides an Environmental Assessment Applicability Review

This proposed amendment has been reviewed and approved by CECo On-Site and Off-Site Review in accordance with Commonwealth Edison procedures.

To the best of my knowledge and belief, the statements contained above are true and correct. In some respect these statements are not based on my personal knowledge, but from obtained information furnished by other Commonwealth Edison employees, contractor employees, and consultants. Such information has been reviewed in accordance with company practice, and I believe it to be reliable.

Commonwealth Edison is notifying the State of Illinois of this application for amendment by transmitting a copy of this letter and its attachments to the designated state official.

Please direct any questions you may have concerning this submittal to this office.

Very truly yours,

Gary G Benes

Gary G. Benes
Nuclear Licensing
Administrator

Subscribed and Sworn to before me
on this 10 day of
DECEMBER, 1993.

"OFFICIAL SEAL"
James L. Garling
Notary Public, State of Illinois
My Commission Expires 4/8/97

James L Garling
Notary Public

Attachments:

A. Description and Safety Analysis of the Proposed Changes
B. Marked-Up Technical Specification Pages
C. Evaluation of Significant Hazards Considerations
D. Environmental Assessment Applicability Review

cc: J. B. Martin, Regional Administrator - RIII
D. L. Hills, Senior Resident Inspector - LSCS
A. T. Gody Jr., Project Manager, NRR
Office of Nuclear Facility Safety - IDNS

ATTACHMENT A

DESCRIPTION AND SAFETY ANALYSIS OF THE PROPOSED CHANGES

Description of the Proposed Change

Technical Specification Exigent Amendment Request for Safety/Relief Valves (SRVs) 1B21-F013B and 1B21-F013J to be exempted from the requirements of Technical Specification 4.0.3 until the sixth refueling outage (L1R06) 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. The change allows SRVs 1B21-F013B and 1B21-F013J to remain operable with their setpoint verification tests not performed within the required surveillance interval per ASME Section XI, IWV-3510-1.

Description of the Current Operating License/Technical Specification Requirement

Technical Specification 4.0.5 and 4.0.5.a are as follows:

"4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2, & 3 components shall be as Follows:

a. "Inservice inspection of ASME Code Class 1, 2, & 3 components and inservice testing of ASME Code Class 1, 2, & 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i)."

Section XI of the ASME Boiler and Pressure Vessel Code, Table IWV-3510-1, "Category C: Safety and Relief Valves Testing Schedule," Note (1) is as follows:

"(1) N_1, N_2, N_3, etc., are the numbers of months from startup to first refueling, second refueling, third refueling, etc. When N is a number larger than 60, all valves which have not been tested during the preceding 5 year period shall be tested. The following period shall then be considered to be the same as "startup to first refueling" for purposes of determining test frequency, with the added requirement that at each refueling all valves which have not been tested during the preceding 5 year period shall be tested. ..."

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Technical Specification 3.4.2 and action a are as follows:

3.4.2 The safety valve function of 17 of the below listed 18 reactor coolant system safety/relief valves shall be OPERABLE with the specified code safety valve function lift setting*#; all installed valves shall be closed with OPERABLE position indication.

a. With the safety valve function of one or more of the above required safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

Bases for the Current Requirement

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. 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 Limiting Condition for Operation." Technical Specification 3.4.2 is the Limiting Condition for Operation applicable to the SRVs.

The Limiting Condition for Operation currently requires 17 of the 18 SRVs installed to be OPERABLE. The safety valve function of the safety/relief valves operates to prevent the reactor coolant system from being pressurized above the Safety Limit of 1325 psig, reactor steam dome pressure, in accordance with the ASME Code. Analysis has shown that with the safety function of one of the eighteen safety/relief valves inoperable, the reactor pressure is limited to within ASME III allowable values for the worst case upset transient. Therefore, operation with any 17 SRVs capable of opening is allowable, although all installed SRVs must be closed and have position indication to ensure that integrity of the primary coolant boundary is known to exist at all times. Demonstration of the safety/relief valve lift settings will occur only during shutdown and will be performed in accordance with the provisions of Specification 4.0.5. The worst case upset transient conservatively assumes that one of the two lowest set pressure SRVs fails to open, thus providing justification for any one SRV to be inoperable.

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Description of the Need for Amending the Technical Specification and Basis for Exigency

The specified code safety valve lift setting has not been verified since the Unit 1 first refuel outage, ending October 1986 for SRVs 1B21-F013B and 1B21-F013J. The 5 year period ended in October 1991, which was prior to the fifth refuel on LaSalle Unit 1. Therefore, the valves have exceeded the ASME Code specified frequency. (The code allows extending the interval of testing for a given SRV to the next scheduled refuel outage, but 1B21-F013B and 1B21-F013J were not tested during the last completed refueling outage, which was the first outage after the 5 year limit was reached.) Per Technical Specification 4.0.3, SRVs 1B21-F013B and 1B21-F013J were declared inoperable at 5:15 p.m. CST, December 4, 1993 due to their surveillance frequency being exceeded. Per Technical Specification 3.4.2, Action a, LaSalle Unit 1 was required to be in Hot Shutdown within 12 hours and in Cold Shutdown within the following 24 hours. Based on the Notice of Enforcement Discretion, that was granted (see cover letter reference), SRVs 1B21-F013B and 1B21-F013J will remain OPERABLE until Unit 1 is shut down for L1R06 or the next time Unit 1 is in Cold Shutdown, whichever comes first. During this period of time, the LCO must be modified to require all 18 SRVs to be OPERABLE until the lift setting is verified for SRVs 1B21-F013B and 1B21-F013J. This Technical Specification amendment request is required to comply with the Notice of Enforcement Discretion. The Enforcement Discretion granted is valid until this Technical Specification is granted or denied. Without the Notice of Enforcement Discretion, LaSalle Unit 1 would be required to begin shutdown per Action a of specification 3.4.2. The determination that the 2 SRVS installed in Unit 1 were not tested within the frequency required by ASME section XI, Table IWV-3510-1 was not made until December 4, 1993 at 5:15 p.m. CST. Therefore, the need for a Technical Specification amendment was determined too late to process either an emergency or an exigent request. This amendment request is being submitted as exigent in compliance with the Notice of Enforcement Discretion.

Description of the Amended Technical Specification Requirement

The Limiting Condition for Operation (LCO) for Technical Specification 3/4.4.2, Safety/Relief Valves, is being modified by new footnote ##. The note is being tied to the number 17, because the number of SRVs required to be operable during the remainder of this cycle or the first Cold Shutdown of Unit 1 is being changed to 18. The changes are included in the footnote to indicate the temporary nature of this change.

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Technical Specification 3.4.2 is proposed to be as follows:

3.4.2 The safety valve function of 17## of the below listed 18 reactor coolant system safety/relief valves shall be OPERABLE with the specified code safety valve function lift setting*#; all installed valves shall be closed with OPERABLE position indication.

> ##until Unit 1 enters Cold Shutdown at the end of Unit 1 cycle six or the next Cold Shutdown, whichever comes first, Limiting Condition for Operation 3.4.2 is modified as follows:
>
> 1. The number of SRVs required to be OPERABLE is changed from 17 to 18 of the 18 SRVs installed.
>
> 2. The provisions of specification 4.0.3 are not applicable to SRVs 1B21-F013B and 1B21-F013J with respect to the safety valve function lift setting setpoint test frequency specified in ASME Code Section XI, Table IWV-3510-1.

Bases for the Amended Technical Specification Request

By not performing the safety valve function lift test, the possibility of the setpoint drifting may occur.

The possible effects of the safety setpoint drift are as follows:

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

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

3) Complete failure to open

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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

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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 evaluate for the potential for an exaggerated drift due to the extended surveillance interval, the pressure margin to decreasing setpoint drift was calculated by two methods: The first was to take two times the largest drift from the as-found test data (4% of setpoint). This would be approximately 8% of setpoint, and for the SRV 1B21-F013B, a possible drift down from 1195 (nominal setpoint) to 1099 psig. The second calculation included all setpoint drift data, and calculated the pressure assuming a downward drift of 3*sigma, where sigma is the sample standard deviation of all setpoint data taken for both LaSalle Units since plant startup. This calculation resulted in a low pressure of 1133 psig. Both of these evaluations show considerable margin to normal operating pressures (1005 psig), and confirm that downward setpoint drift is highly unlikely to cause an operational transient. SRV 1B21-F013J has a higher setpoint and greater margin for downward setpoint drift than does SRV 1B21-F013B.

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-F013B or 1B21-F013J 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-F013B was manually opened twice during that event and SRV 1B21-F013J was manually opened once. Both operations were satisfactory, indicating that operation and capability of SRV 1B21-F013B and 1B21-F013J are not impaired by any other factors.

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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-F013B or 1B21-F013J 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-F013B or 1B21-F013J 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-F013B or 1B21-F013J 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-F013B is 1106 psig, and for SRV 1B21-F013J is 1116 psig).

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

The condition of concern affects only the safety function setpoints of SRVs 1B21-F013B and 1B21-F013J, and does not involve either their relief mode operation, indications, or possible system actuation effects.

Evaluation of Case 3) SRVs B and J Fail to Open:

The peak vessel pressure for the MSIV closure with flux scram was evaluated from parametric values determined in the General Electric Standard Application for Reloads (GESTAR), in the evaluation performed for CECo by General Electric on 4 December [illegible] In this document (GESTAR) the peak pressure as a function of # inoperable [illegible] closed or flanged off) SRVs was evaluated. In the letter report provided to CECo, the impact on peak vessel pressure was given as less than 10 psig above the LaSalle 1 Cycle 6 analysis value of 1266 psig.

Because the transient evaluation was done assuming that neither SRV B or J would open, this case is therefore specifically analyzed to be acceptable.

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The following Compensatory Actions are in effect:

1) Licensed Operators are being 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 have been placed on the control panel to reinforce the information that safety actuation pressures should be verified during appropriate conditions.

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.

Schedule

This Technical Specification amendment request is being submitted as an exigent change as committed in the Notice of Enforcement Discretion.