Commonwealth Edison Company
LaSalle Generating Station
2601 North 21st Road
Marseilles, IL 61341-9757
Tel 815-357-6761

ComEd

February 21, 2000

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

LaSalle County Station, Unit 2
Facility Operating License No. NPF-18
NRC Docket No. 50-374

Subject: Request for an Exigent Change to Appendix A, Technical Specifications, Surveillance Requirement 4.0.5.f

Reference: Letter from J. A. Benjamin to USNRC dated February 18, 2000, "Request for Notice of Enforcement Discretion Concerning Performance of an Augmented Examination of Weld RH-2005-29"

In accordance with 10 CFR 50.91(a)(6), Commonwealth Edison (ComEd) Company request a change to Appendix A, Technical Specifications (TS), of Facility Operating License NPF-18 on an exigent basis. Specifically, we propose a change to TS Surveillance Requirement (SR) 4.0.5.f to allow the required examination of weld RH-2005-29 to be deferred until the next scheduled refueling outage or December 31, 2000, whichever is earlier.

TS Section 3.4.8," Structural Integrity," requires the structural integrity of American Society of Mechanical Engineers (ASME) Class 1 components to be maintained in accordance with the surveillance requirements of TS Section 4.4.8., "Structural Integrity". TS Section 4.4.8 invokes the surveillance requirements of TS SR 4.0.5. TS SR 4.0.5.f requires that piping susceptible to intergranular stress corrosion cracking (IGSCC), be examined in accordance with the NRC staff positions on schedule, methods, personnel and sample expansion included in NRC Generic Letter 88-01. "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping."

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At 1527 hours Central Standard Time (CST) on February 17, 2000, TS SR 4.0.3 was entered due to a missed TS surveillance requirement allowing 24 hours to perform the surveillance or pursue enforcement discretion. Without enforcement discretion, at 1527 hours CST, on February 18, 2000, LaSalle County Station, Unit 2 would have been required to be in at least STARTUP within the next seven hours, HOT SHUTDOWN within in the following six hours and COLD SHUTDOWN within the subsequent 24 hours, since the action statement of TS Section 3.4.8 could not be complied with due to current plant conditions, in accordance with TS Section 3.0.3. A Notice of Enforcement Discretion (NOED) was requested. At approximately 1130 hours CST on February 18, 2000, the NRC granted approval of the requested NOED. The follow-up written NOED request, referenced above, was submitted by letter dated February 18, 2000.

ComEd requests that this proposed TS change be processed on an exigent basis as specified by the provisions of 10 CFR 50.91(a)(6). 10 CFR 50.91(a)(6)(i)(B)(vi) states that a licensee must state whether the exigency could have been avoided and whether or not the licensee has exerted its best efforts to submit a timely application for an amendment. The need for a license amendment was determined upon the recent review of historical IGSCC weld examination data, including Inductive Heat Stress Improvement (IHSI) data, and Mechanical Stress Improvement (MSIP) records. A clerical error was discovered involving two IGSCC susceptible welds in the Unit 2 "A" Residual Heat Removal (RHR) Shutdown Cooling return piping. The clerical error resulted in the welds never being subjected to any stress improvement process. We had no prior knowledge of this clerical error. Submittal of this change request is consistent with the guidance provided in NRC Administrative Letter 95-05 "Revisions to Staff Guidance for Implementing NRC Policy on Notices of Enforcement Discretion"` Revision 1, for NOEDs that are granted, which also require a license amendment. We have concluded that the circumstances surrounding this request for exigent review were unavoidable and not created by a failure to make a timely application for a license amendment.

This proposed change to TS Section 4.0.5.f is to allow the required examination of weld RH-2005-29 to be deferred until the next scheduled refueling outage or December 31, 2000, whichever is earlier, to prevent undue shutdown of the unit. As we are requesting that this proposed change be processed on an exigent basis, we request approval of this amendment by March 17, 2000.

This proposed change is subdivided as follow:

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

2. Attachment B includes the marked-up TS page with the requested change indicated.

3. Attachment C describes our evaluation performed in accordance with 10 CFR 50.92(c), which provides information supporting a finding of no significant hazards.

4. Attachment D provides information supporting an Environmental Assessment.

This proposed change has been reviewed by The Station Plant Operations Review Committee (PORC)" and approved by the Nuclear Safety Review Board (NSRB) in accordance with the Quality Assurance Program.

ComEd 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.

Should you have any questions concerning this letter, please contact Mr. Frank A. Spangenberg, III, Regulatory Assurance Manager, at (815) 357-6761, extension 2383.

Respectfully,

Jeffrey A. Benjamin
Site Vice President
LaSalle County Station

Attachments:
- A. Description and Safety Analysis for the Proposed Change
- B. Marked-up Pages for the Proposed Change
- C. Information Supporting a Finding of No Significant Hazards Consideration
- D. Information Supporting an Environmental Assessment

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

STATE OF ILLINOIS)

IN THE MATTER OF)

COMMONWEALTH EDISON COMPANY)

LASALLE COUNTY STATION UNIT 2) Docket No. 50-374

Subject: Request for an Exigent Change to Appendix A, Technical Specifications, Surveillance Requirement 4.0.5.f

AFFIDAVIT

I affirm that the content of this transmittal is true and correct to the best of my knowledge, information and belief.

Jeffrey A. Benjamin
Site Vice President
LaSalle County Station

Subscribed and sworn to before me, a Notary Public in and for the State above named, this 21st day of February, 2000.
My Commission expires on November 24, 2001.

" OFFICIAL SEAL "
Joseph V. Sipek
Notary Public, State of Illinois
My Commission Expires 11/24/2001

Notary Public

DESCRIPTION AND SUMMARY SAFETY ANALYSIS FOR PROPOSED CHANGES

A. SUMMARY OF PROPOSED CHANGES

In accordance with 10 CFR 50.91(a)(6), Commonwealth Edison (ComEd) Company proposes a change to Appendix A, Technical Specifications (TS), of Facility Operating License NPF-18. We request that this proposed change be processed on an exigent basis as Unit 2 is currently operating under a Notice of Enforcement Discretion granted by the NRC on February 18, 2000. The proposed change is to TS Surveillance Requirement (SR) 4.0.5.f, to allow the required examination of weld RH-2005-29 to be deferred until the next scheduled refueling outage or December 31, 2000, whichever is earlier.

The proposed change is described in Section E of this Attachment. The marked up TS page is shown in Attachment B.

B. DESCRIPTION OF THE CURRENT REQUIREMENTS

TS Section 3.4.8, "Structural Integrity," requires that the structural integrity of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Class 1, 2 and 3 components be maintained in accordance with SR 4.4.8. TS SR 4.4.8 in turn requires no additional surveillance requirements other than those required by TS SR 4.0.5.

TS Section 4.0.5.f requires that the inservice inspection (ISI) program for piping identified in NRC Generic Letter 88-01 "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping," be performed in accordance with the NRC staff positions on schedule, methods, personnel, and sample expansion included in Generic Letter 88-01 or in accordance with alternate measures approved by the NRC staff.

If the requirements of TS 4.0.5.f can not be met, TS 3.4.8, Action A for Class 1 piping requires restoration of the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System (RCS) temperature more than 50 °F above the minimum temperature required by nil-ductility transition (NDT) considerations. If the Unit is at a temperature in excess of this requirement, the action statement can not be complied with and entry into TS Section 3.0.3 is required.

TS Section 3.0.3 requires that within 1 hour action shall be initiated to place the unit in an operational condition in which the TS requirement does not apply by placing the unit in at least STARTUP within the next 6 hours, HOT SHUTDOWN within the following 6 hours and be in COLD SHUTDOWN within the subsequent 24 hours.

C. BASES FOR THE CURRENT REQUIREMENTS

The inspection program for ASME B&PV Code Class 1, 2, and 3 components provides reasonable assurance that the structural integrity of these components will be maintained at an acceptable level throughout the design life of the component.

D. NEED FOR REVISION OF THE REQUIREMENT

During a review of historical examination data, for welds susceptible to intergranular stress corrosion cracking (IGSCC), including Inductive Heat Stress Improvement (IHSI) data, and Mechanical Stress Improvement (MSIP) records, a clerical error was discovered involving two IGSCC susceptible welds in the Unit 2 "A" Residual Heat Removal (RHR) Shutdown Cooling (SDC) return piping. This piping is ASME B&PV Code Class 1, 12 inch diameter piping and is fabricated from Type 304 austenitic stainless steel. The pipe is welded to a cast austenitic stainless steel valve. The two welds involved are designated RH-2005-28 and RH-2005-29, which connect the upstream piping and downstream elbow to valve 2E12-F090A, "A RHR SDC Return Header Manual Stop Valve." No weld repair activities have been performed on these two welds. The carbon content of valve 2E12-F090A is 0.04%, and the delta ferrite was determined to be greater than 18 FN.

IHSI records of 1983 depicted these two welds at different physical locations than the ISI drawings. Therefore, the stress relief that was documented to have been applied to these two welds was actually applied to welds designated RH-2005-30 and RH-2005-33. This error went undetected, and in 1987 when the remaining balance of the IGSCC weld population was subjected to the MSIP, it was concluded that welds RH-2005-28 and 29 had already been stress relieved in 1983. Consequently, the two welds have never been subjected to any stress improvement process.

Based on having no stress improvement, these two welds should have been categorized as IGSCC Category D, vice Category B, in accordance with NRC Generic Letter 88-01. The examination schedule for Category D welds would have required that the two welds be examined every two refueling cycles beginning with L2R02. Based on the issuance of Generic Letter 88-01 in January of 1988, and the second refueling outage being performed in the fall of 1988; ComEd would have conservatively required the exam completion by the third refueling outage, establishing the every other refueling schedule. Examinations of weld RH-2005-28 were not completed during the 3rd or 5th refueling outages, but the weld was examined during the 7th refueling outage with no indication of cracking noted. Weld RH-2005-29 was examined during the 3rd refueling outage with no indication of cracking noted, but was not re-examined during the 5th or 7th refueling outages as required by ComEd's augmented inspection program that addresses the recommendation of Generic Letter 88-01. The weld inspections conducted on these two welds were done in accordance with ComEd's approved augmented inspection program; in that they were performed using approved procedures, by Electric Power Research Institute (EPRI) qualified individuals.

The schedule of examinations for welds RH-2005-28 and 29 has not been executed in accordance with the Category D schedule specified in the augmented inspection program. This error does not affect the selection process for normal ASME B&PV code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," welds because that selection process does not depend on the application of any stress improvement processes.

Thus, it has been determined that a TS 4.0.5.f required surveillance has been missed. This would necessitate entry into TS Section 3.4.8, Action A, for Class I piping that requires restoration of the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the RCS temperature more than 50 °F above the minimum temperature required by NDT considerations. Since the unit is currently at a temperature in excess of this requirement, the action statement can not be complied with and entry into TS Section 3.0.3 is required.

TS Section 3.0.3 requires that within one hour action shall be initiated to place the unit in an operational condition in which the TS requirement does not apply by placing the unit in at least STARTUP within the next six hours, HOT SHUTDOWN within the following six hours and be in COLD SHUTDOWN within the subsequent 24 hours.

Since TS Section 3.0.3 provides actions to place the unit in a shutdown condition in a time less than 24 hours, TS SR 4.0.3 was applied due to a missed TS surveillance which allows 24 hours to complete the surveillance or to pursue enforcement discretion.

The 24-hour time clock of TS 4.0.3 would have expired on February 18, 2000, at 1527 hours Central Standard Time (CST). At the expiration of this time clock, TS 3.0.3 would have required placing LaSalle County Station, Unit 2 in a COLD SHUTDOWN condition. At approximately 1130 hours CST on February 18, 2000, the NRC granted verbal approval of the NOED. The follow-up NOED written request was submitted on February 18, 2000.

E. DESCRIPTION OF THE PROPOSED CHANGES

The specific proposed change to TS Section 4.0.5.f is to add the following footnote *:

> * "Augmented examination of weld RH-2005-29, per the approved program, will be deferred until the next scheduled refueling outage, L2R08, or December 31, 2000, whichever is earlier."

F. SAFETY ANALYSIS OF THE PROPOSED CHANGES

The integrity of the reactor pressure vessel (RPV) and other components of the primary system of a nuclear plant can be adversely affected by the number of transients that they are subjected to during their lifetime. It is prudent to avoid unnecessary transients provided the health and safety of the public is preserved. ComEd requested an NOED, which the NRC subsequently granted to permit continued operation of LaSalle County Station. Unit 2 while the exigent TS amendment request is being processed.

Weld RH-2005-29 is a valve-to-elbow weld and is required to be considered as Category D in accordance with the augmented inspection program, which follows the recommendations of Generic Letter 88-01. Category D welds are those made of susceptible materials that have not received an IGSCC mitigation treatment. As noted in a report prepared by the Boiling Water Reactor Vessel and Internals Project, (BWRVIP)-75 (EPRI Technical Report (TR-)113932), "BWR Vessel and Internals Project, Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules," industry experience since the issuance of Generic Letter 88-01 for examination of Category D welds has been excellent. The technical basis in BWRVIP-75 for extending weld examination frequencies from every other refueling cycle to once every six years states:

> "In the 33 plants that responded to the survey, there are currently 432 Category D welds that have been examined 1325 times. There are 169 welds currently being effectively treated with Hydrogen Water Chemistry (HWC). There has been one known Category D weld reported to be cracked in the last 8 years. The one known crack was detected at Hope Creek in 1997. The cracking (three pinholes) occurred in a dissimilar weld at a safe-end to nozzle with nickel-based alloy 182. The owner determined that this weld had experienced multiple repairs. .... Other than this weld repair, no other Category D cracking has been reported."

The 432 welds provide data across multiple systems, from different plants, representing a diverse cross section of operating conditions, providing the conclusion that category D welds have behaved and continued to behave without cracking, except as noted for the Hope Creek case. Unlike the Hope Creek weld, RH-2005-29 is not a dissimilar metal weld; it involves only stainless steel base and weld metal and weld RH-2005-29 has not had any documented weld repairs. Hydrogen Water Chemistry, however, has not yet been implemented on Unit 2.

Early industry experience revealed that small-bore, thin-wall pipe would develop IGSCC flaws early in life. This has been attributed to higher weld residual stresses (i.e., near yield stress) in thin-wall pipe. Also, since through-wall weld residual stresses are higher in thin-walled pipe, flaws would be expected to grow faster. This is corroborated by the large number of welds in 4" to 12" piping in the BWR fleet that required weld overlay repairs early in life. The conclusion to be reached is that if weld RH-2005-29 was going to crack, it would have already done so.

A review of the LaSalle County Station Unit 2 reactor water chemistry program reveals that it has been maintained in accordance with procedure NOD-CY-02, "BWR Water Chemistry Control Program." This procedure is based on the guidance given in EPRI Report TR-103515-R1, "BWR Vessel and Internals Project Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules" dated December 1996. Reactor water chemistry was maintained within EPRI guidelines 93.7% of the time since the last inspection of weld RH-2005-29 in 1990. When a non-standard condition developed, the EPRI guidelines were followed to restore chemistry to normal operating parameters. Based on a review of reactor water chemistry for sulfate and chloride data over the past 10 years of operation of LaSalle County Station, Unit 2, the most notable chemistry transient during power operation occurred during January 1996 with a peak sulfate of 74 parts per billion (ppb). The transient was due to a small ingress of resin, and reactor coolant sulfates were reduced to less than 24 ppb in less than 24 hours.

The weld on the opposite side of the valve, RH-2005-28, is also a Category D weld. Its operating environment is similar to that for RH-2005-29. Weld RH-2005-28 was inspected in 1987 and again in 1996 with no recordable indications. It is reasonable to postulate that weld RH-2005-29 has behaved similarly and is therefore not expected to be flawed.

If the foregoing bases should prove to be precluded due to weld-unique conditions (e.g., weld fit-up problem,inside diameter (ID) grinding, etc.) and the weld is actually flawed, safety is still not jeopardized. Austenitic stainless steel is a tough and ductile material and flaw tolerant. Additionally, IGSCC has an irregular crack form. These attributes lead to the conclusion that this piping will leak before it breaks. EPRI Report NP-4991, "Application of the Leak-before-Break Approach to BWR Piping," provides supporting information. A plant-specific critical flaw evaluation was performed to assess the weld using the loads from the LaSalle County Station piping stress reports. The results are bounded by the EPRI Report NP-4991. The conclusion to be drawn is that if weld RH-2005-29 is flawed, and should the flaws propagate through the weld or wall of the pipe, it will create a leak that would be readily detected by RCS leakage detection instrumentation well in advance of the pipe break and the unit could shutdown with no significant impact on safety. In the event of a break at this location, the consequences remain bounded by the current Emergency Core Cooling System (ECCS) – loss of Coolant Accident (LOCA).

Additionally, a risk assessment was performed as part of a defense-in-depth evaluation of weld RH-2005-29. The analysis included the following.

1. Estimate of the nominal pipe rupture frequency of a routinely inspected weld.
2. The conditions at the time of the last inspection in 1990.
3. Estimate of the relative increase in pipe rupture frequency incurred due to the missed inspections.
4. Calculation of the conditional probability of core damage and large early release given a break in the line.
5. Estimate of the risk of continued operation.
6. Evaluation of competing risks associated with plant shutdown to perform inspection.

The risk assessment is based on the latest LaSalle County Station plant-specific Probabilistic Safety Assessment (PSA) for internal events and the EPRI sponsored piping reliability Markov model developed for risk-informed ISI assessment.

The results using these information indicate change in that the risk resulting from continued operation is low:

- Core Damage Frequency (CDF) $\cong 5 \times 10^{-9}$/year, and
- Large Early Release Frequency (LERF) $\cong 7 \times 10^{-11}$/year.

These values are well below the risk increase thresholds considered acceptable for permanent plant changes as delineated in Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," dated July 1998.

The competing risks, associated with alternative actions involving shutting down the reactor for a forced outage, have been evaluated as part of the LaSalle County Station internal events PSA and is estimated to be approximately 2×10^{-7}/manual shutdown, which is factor of approximately 40 times higher than the risk of continued operation with weld RH-2005-29 uninspected until Fall 2000.

The extent of condition has been reviewed in depth and this is the only weld that has not had required inspections performed. In addition, it has been verified that these two welds were the only LaSalle Country Station, Unit 1 or Unit 2, welds defined as susceptible to IGSCC in accordance with the guidance in Generic Letter 88-01 that did not receive stress improvement by either IHSI or MSIP.

G. IMPACT ON PREVIOUS SUBMITTALS

ComEd has reviewed the proposed change regarding impact on any previous submittals, and has determined that there is no impact on any outstanding previous submittals.

H. SCHEDULE REQUIREMENTS

We request that this submittal be approved by March 17, 2000 so that the NOED can be terminated.

ATTACHMENT B
Proposed Change to Technical Specifications for
LaSalle County Station, Unit 2
1 of 1

MARKED-UP TS PAGE FOR PROPOSED CHANGE

REVISED PAGES

APPLICABILITY

SURVEILLANCE REQUIREMENTS (Continued)

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities.

d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

f. The inservice inspection program for piping identified in NRC Generic Letter 88-01 shall be performed in accordance with the NRC staff positions on schedule, methods, personnel, and sample expansion included in Generic Letter 88-01 or in accordance with alternate measures approved by the NRC staff.*

*

* Augmented examination of weld RH-2005-29, per the approved program, will be deferred until the next scheduled refueling outage, L2R08, or December 31, 2000, whichever is earlier.

INFORMATION SUPPORTING A FINDING OF NO SIGNIFICANT HAZARDS CONSIDERATION

ComEd has evaluated the proposed change and determined that it does not involve a significant hazards consideration. According to 10 CFR 50.92(c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

> Involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated;
>
> Create the possibility of a new or different kind of accident from any previously analyzed; or
>
> Involve a significant reduction in a margin of safety.

The amendment request proposes to modify Technical Specifications (TS) Surveillance Requirement (SR) 4.0.5.f. to allow the required examination of weld RH-2005-29 to be deferred until the next scheduled refueling outage or December 31, 2000, whichever is earlier.

The determination that the criteria set forth in 10 CFR 50.92(c) are met for this proposed change is indicated below:

Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change represents a minimal increase in the probability of a pipe break resulting in a Loss of Coolant Accident (LOCA). The proposed change will not impact the source term used in the derivation of the LOCA dose consequences. Therefore the consequences will remain unchanged since the resulting LOCA is bounded by current analysis.

Therefore the proposed change does not involve a significant increase in the probability or consequence of an accident previously evaluated.

Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not introduce a new mode of plant operation and does not involve a physical modification to the plant. The proposed change does not introduce a new failure mode.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Does the change involve a significant reduction in a margin of safety?

Since the LOCA analysis remains unchanged, the fuel integrity margin, as expressed as Peak Cladding Temperature, is not affected. The change does not impact the Reactor Coolant Pressure Boundary Overpressure Analysis; therefore the margin of safety for the Reactor Coolant Pressure Boundary is not affected. The blowdown energy, resulting from a LOCA and the ability of the suppression chamber to maintain the margin of safety of the containment barrier are not affected.

Therefore, the changes do not involve a significant reduction in the margin of safety.

Therefore, based upon the above evaluation, ComEd has concluded that this change involves no significant hazards consideration.

INFORMATION SUPPORTING AN ENVIRONMENTAL ASSESSMENT

ComEd has evaluated the proposed change against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. ComEd has determined that the proposed change meet the criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(9) and as such, has determined that no irreversible consequences exist in accordance with 10 CFR 50.92(b). This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR 50 that changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or that changes an inspection or a surveillance requirement, and the proposed changes meet the following specific criteria:

(i) The proposed changes involve no significant hazards consideration,

The proposed changes do not involve a significant hazards consideration,

(ii) There is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite,

There will be no change in the types or significant increase in the amounts of any effluents released offsite and

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

The proposed changes will not result in changes in the operation or configuration of the facility. There will be no change in the level of controls or methodology used for processing of radioactive effluents or handling of solid radioactive waste, nor will the proposal result in any change in the normal radiation levels within the plant. Therefore, there will be no increase in individual or cumulative occupational radiation exposure resulting from the proposed changes.