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June 21, 1989

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

Subject: LaSalle County Station Units 1 and 2
Application for Amendment to Facility
Operating Licenses NPF-11 and NPF-18
Revision of Pressure - Temperature Curves
NRC Docket Nos. 50-373 and 50-374

References: F.J. Miraglia letter to All Licensees of
Operating Reactors dated June 12, 1988,
regarding Generic Letter 88-11.

Dear Dr. Murley:

Pursuant to 10 CFR 50, Commonwealth Edison is hereby applying for an amendment to Facility Operating Licenses NPF-11 and NPF-18, Appendix A - Technical Specifications. The purpose of this amendment is to meet the requirements of Regulatory Guide 1.99, Revision 2, dated May, 1988, "Radiation Embrittlement of Reactor Vessel Materials" by updating the pressure-temperature curves in the Technical Specifications and the associated information that goes with them.

Attachment A contains background information and justification for the proposed changes. Attachment B contains the proposed change to the Technical Specifications. The proposed change has been reviewed and approved by both On-Site and Off-Site Review in accordance with Commonwealth Edison Company procedures. This amendment request has been evaluated in accordance with 10 CFR 50.92(c) and it was determined that no significant hazards consideration exists. That evaluation is documented in Attachment C.

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

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Please direct any questions you may have regarding this matter to this office.

Very truly yours,

Wayne E Morgan

W.E. Morgan
Nuclear Licensing Administrator

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Attachments A: Background and Information
B: Proposed Technical Specification Change
C: Evaluation of Significant Hazards Consideration

cc: Senior Resident Inspector, LaSalle
P.C. Shemanski - Project Manager, NRR
Office of Nuclear Facility Safety - IDNS
Regional Administrator - Region III

SUBSCRIBED AND SWORN to before me this 21st day of June, 1989

Lelia F. Mayo
Notary Public

"OFFICIAL SEAL"
LELIA F. MAYO
Notary Public, State of Illinois
My Commission Expires 4/25/93

ATTACHMENT A

TECHNICAL SPECIFICATION CHANGE REQUEST

LASALLE COUNTY STATION UNITS 1 AND 2

BACKGROUND:

The NRC has indicated (per Generic Letter 88-11) that operating reactor license holders, such as LaSalle County Station, should use the methods described in Revision 2 to Regulatory Guide 1.99 to predict the effect of the neutron radiation on reactor vessel materials as required by Paragraph V.A of 10 CFR Part 50, Appendix G, unless they can justify the use of different methods. The use of the Revision 2 methodology results in a modification of the pressure-temperature limits contained in the Technical Specifications. This modification is needed to be in compliance with the requirements of Section V of 10 CFR Part 50, Appendix G.

LaSalle Station, in cooperation with Edison's Engineering Department and Architect Engineer (A/E) commissioned General Electric Company to prepare a report, in accordance with these guidelines. Using this report, Commonwealth Edison is submitting a Technical Specification and Updated Final Safety Analysis Report (UFSAR) change.

DISCUSSION:

Part of the Effort to assure reactor vessel integrity involves evaluation of the fracture toughness of the vessels ferritic materials. The key values which characterize a material's fracture toughness are the reference temperature of nil-ductility transition (RT_{NDT}) and the upper shelf energy (USE). These are defined in 10 CFR 50, Appendix G. Appendix G of the ASME Boiler and Pressure Vessel Code, Section III contains methods used to establish the pressure-temperature (P-T) operation limits which must be met to assure vessel ductility. These limits are calculated taking into account irradiation effects per Regulatory Guide (RG) 1.99.

The P-T limits in the current LaSalle Technical Specifications are based on the methods in RG 1.99, Revision 1. The NRC has completed the process of issuing RG 1.99, Revision 2, so the P-T limits were recalculated by General Electric Company including these new irradiation predictions.

The P-T curves developed by General Electric Company address the updated requirements of 10 CFR 50, Appendix G, which was revised in 1983. Geometric discontinuities and highly stressed regions, such as the feedwater nozzles and the closure flanges, are evaluated separately from the core beltline region, as before. The operating limits developed, consider the most

limiting condition of the discontinuity regions and the beltline region (including irradiation) to bound all operating conditions. The new P-T curves were developed accounting for 16 effective full power years (EFPY) and 32 EFPY of irradiation.

The approach followed in developing the updated P-T curve limits was to first collect the data on initial RT_{NDT} values and chemistry from the UFSAR and from the General Electric Company records. Next the fluences for the 16 EFPY and 32 EFPY of operation were determined from flux wire dosimetry test results. Then the limiting beltline materials were determined using RG 1.99, Revision 2 irradiation effect predictions. Finally, the P-T limits curves were drawn, considering the most limiting condition for the beltline materials, non-beltline components and closure flange region.

In addition, it was required that the 200°F limit for COLD SHUTDODWN be raised to 212°F for the purpose of performing hydrostatic or leak testing or heat-up by non-nuclear means. The 200°F limit may be raised because at LaSalle, the boiling point of water is 212°F, thus assuring that the reactor coolant will not boil during the test. In addition, since all control rods will be inserted, no potential for criticality exists.

ATTACHMENT B

PROPOSED CHANGES TO THE TECHNICAL SPECIFICATIONS

FOR OPERATING LICENSES NPF-11 AND NPF-18

REVISED PAGES:

NPF-11	NPF-18
3/4 4-16	3/4 4-17
3/4 4-17	3/4 4-18
3/4 4-18	3/4 4-19
3/4 4-18a*	3/4 4-19a*
B 3/4 4-4	B 3/4 4-4
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B 3/4 4-7	B 3/4 4-7

* NEW PAGE

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