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

MARKED UP PAGES FOR PROPOSED CHANGES

<u>REVISED PAGES</u>

<u>NPF-11</u>

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6-25b

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

2.1.1 THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.4.

1.12 1.11

THERMAL POWER, High Pressure and High Flow

2.1.2 The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.08 with two recirculation loop operation and shall not be less than 1.09 with single recirculation loop operation with the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION: 1.12 1.11

With MCPR less than 1.08 with two recirculation loop operation or less than 1.09 with single recirculation loop operation and the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.4.

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1325 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, and 4.

ACTION:

With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above 1325 psig, be in at least HOT SHUTDOWN with reactor coolant system pressure less than or equal to 1325 psig within 2 hours and comply with the requirements of Specification 6.4.

2.1 SAFETY LIMITS

1.12 1.11

BASES

2.0 The fuel cladding, reactor pressure vessel and primary system piping are the principal barriers to the release of radioactive materials to the environs. Safety Limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that the MCPR is not less than 1.07. MCPR greater than 1.07 for two recirculation loop operation and 1.08 for single recirculation loop operation represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the Limiting Safety System Settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with a margin to the conditions which would produce onset of transition boiling, MCPR of 1.0. These conditions represent a significant departure from the condition intended by design for planned operation.

2.1.1 THERMAL POWER, Low Pressure or Low Flow

The use of the GEXL correlation is not valid for all critical power calculations at pressures below 785 psig or core flows less than 10% of rated flow. Therefore, the fuel cladding integrity Safety Limit is established by other means. This is done by establishing a limiting condition on core THERMAL POWER with the following basis. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a bundle flow of 28×10^3 lbs/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be greater than 28×10^3 lbs/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressure below 785 psig is conservative.

Core Operating Limits Report (Continued)

(16) Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors, XN-NF-79-71(P)(A), Revision 2 Supplements 1, 2, and 3, Exxon Nuclear Company, March 1986.

(17) Generic Mechanical Design Criteria for BWR Fuel Designs, ANF-89-98(P)(A), Revision 1 and Revision 1 Supplement 1, Advanced Nuclear Fuels Corporation, May 1995.

(18) NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," (latest approved revision).

(19) Commonwealth Edison Topical Report NFSR-0085, "Benchmark of BWR Nuclear Design Methods," (latest approved revision).

(20) Commonwealth Edison Topical Report NFSR-0085, Supplement 1, "Benchmark of BWR Nuclear Design Methods - Quad Cities Gamma Scan Comparisons," (latest approved revision).

(21) Commonwealth Edison Topical Report NFSR-0085, Supplement 2, "Benchmark of BWR Nuclear Design Methods - Neutronic Licensing Analyses," (latest approved revision).

(22) Commonwealth Edison Topical Report NFSR-0091, "Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods," Revision 0, Supplements 1 and 2, December 1991, March 1992, and May 1992, respectively; SER letter dated March 22, 1993.

(23) BWR Jet Pump Model Revision for RELAX, ANF-91-048(P)(A), Supplement 1 and Supplement 2, Siemens Power Corporation, October 1997.

(24) ANFB Critical Power Correlation Application for Coresident Fuel, EMF-1125(P)(A), Supplement 1, Appendix C, Siemens Power Corporation, August 1997.

(25) ANFB Critical Power Correlation Determination of ATRIUM-9B Additive Constant Uncertainties, ANF-1125(P)(A), Supplement 1, Appendix E, Siemens Power Corporation, September 1998.

(26) RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model, EMF-85-74(P), Supplement 1 (P)(A) and Supplement 2 (P)(A), Siemens Power Corporation, February 1998.

ATTACHMENT C
Proposed Change to Technical Specifications
for LaSalle County Station Unit 1
INFORMATION SUPPORTING NO SIGNIFICANT HAZARDS FINDING

ComEd has evaluated this proposed amendment and has determined that it involves no 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.

ComEd proposes to change the Technical Specifications (TS) of Facility Operating License NPF-11 for LaSalle County Station Unit 1. The proposed changes are to:

1. Section 2.1, Safety Limits, to reflect a change to the LaSalle Unit 1 Minimum Critical Power Ratio (MCPR) Safety Limit.

2. Section 6.6.A.6.b to add an NRC-approved Siemens Power Corporation (SPC) methodology list of Topical Reports for the Core Operating Limits Report.

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

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

The probability of an evaluated accident is derived from the probabilities of the individual precursors to that accident. The consequences of an evaluated accident are determined by the operability of plant systems designed to mitigate those consequences. Limits have been established consistent with NRC-approved methods to ensure that fuel performance during normal, transient, and accident conditions is acceptable. These changes do not affect the operability of plant systems, nor do they compromise any fuel performance limits.

Changing the MCPR Safety Limit for LaSalle Unit 1 will not increase the probability or the consequences of an accident previously evaluated. This change implements the MCPR Safety Limit resulting from the SPC ANFB critical power correlation methodology using the approved ATRIUM-9B additive constant uncertainty. For each cycle, cycle specific MCPR Safety Limit calculations will be performed, consistent with SPC's approved methodology, to confirm the appropriateness of the MCPR Safety Limit. Additionally, operational MCPR limits will be applied that will ensure the MCPR Safety Limit is not violated during all modes of operation and anticipated operational occurrences. The MCPR Safety Limit ensures that less than 0.1% of the rods in the core are expected to experience boiling transition. Therefore the probability or consequences of an accident will not increase.

Adding EMF-85-74, Revision 0, Supplement 1 (P)(A) and Supplement 2 (P)(A) to Section 6 does not increase the probability or consequences of an accident previously evaluated. The NRC-approved burnup extension for RODEX2A applications has been demonstrated to meet all applicable design criteria. Therefore adding this methodology to Technical Specification Section 6 does not increase the probability or consequences of an accident previously evaluated.

Therefore, this proposed amendment does not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

Creation of the possibility of a new or different kind of accident would require the creation of one or more new precursors of that accident. New accident precursors may be created by modifications to the plant configuration, including changes in allowable modes of operation. This Technical Specification submittal does not involve any modifications to the plant configuration or allowable modes of operation. No new precursors of an accident are created and no new or different kinds of accidents are created. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

Changing the MCPR Safety Limit does not create the possibility of a new accident from any accident previously evaluated. This change does not alter or add any new equipment or change modes of operation. The MCPR Safety Limit is established to ensure that 99.9% of the rods avoid boiling transition.

The MCPR Safety Limit is changing for LaSalle Unit 1 to support Cycle 9 operation. This change does not introduce any physical changes to the plant, alter the processes used to operate the plant, or change allowable modes of operation. Therefore, no new accidents are created that are different from any accident previously evaluated.

The addition of RODEX2A (EMF-85-74, Revision 0, Supplement 1 (P)(A) and Supplement 2 (P)(A)) does not create the possibility of a new accident from an accident previously evaluated. This change does not alter or add any new equipment or change modes of operation. This change does not introduce any physical changes to the plant, alter the processes used to operate the plant, or change allowable modes of operation. Therefore, no new accidents are created that are different from any accident previously evaluated.

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

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

Changing the MCPR Safety Limit for LaSalle Unit 1 wili not involve any reduction in margin of safety. The MCPR Safety Limit provides a margin of safety by ensuring that less than 0.1% of the rods are calculated to be in boiling transition. The proposed Technical Specification amendment request reflects the MCPR Safety Limit results from evaluations by SPC using NRC-approved methodology.

The revised MCPR Safety Limit will ensure the same level of fuel protection. Additionally, operational limits will be established based on the proposed MCPR Safety Limit to ensure that the MCPR Safety Limit is not violated during all modes of operation including anticipated operation occurrences. This will ensure that the fuel design safety criterion of more than 99.9% of the fuel rods avoiding transition boiling during normal operation as well as during an anticipated operational occurrence is met.

The addition of EMF-85-74, Revision 0, Supplement 1 (P)(A) and Supplement 2 (P)(A) to Section 6 does not decrease the margin of safety. The burnup limit extension for RODEX2A applications has been reviewed and approved by the NRC. The data supporting the burnup extension demonstrates that all applicable design criteria are met. Therefore, since the burnup extension is acceptable and within the design criteria, using the approved burnup extension will not affect the margin of safety.

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

Therefore, based upon the above evaluation, ComEd has concluded that these changes involve no significant hazards consideration.

<u>ATTACHMENT D</u>

Proposed Change to Technical Specifications
for LaSalle County Station Unit 1
INFORMATION SUPPORTING AN ENVIRONMENTAL ASSESSMENT

ComEd has evaluated this proposed operating license amendment request against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. ComEd has determined that this proposed license amendment request meets 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 amendment meets the following specific criteria:

(i) the amendment involves no significant hazards consideration.

As demonstrated in Attachment C, this proposed amendment does not involve any 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.

As documented in Attachment C, there will be no change in the types or significant increase in the amounts of any effluents released offsite.

(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 this change.