January 8, 1998

Mr. Oliver D. Kingsley, President
Nuclear Generation Group
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 500
Downers Grove, IL 60515

SUBJECT: ENVIRONMENTAL ASSESSMENT - BYRON STATION, UNITS 1 AND 2, AND BRAIDWOOD STATION, UNITS 1 AND 2 (TAC NOS. M98344, M98345, M98346 AND M98347)

Dear Mr. Kingsley:

Enclosed is a copy of the Environmental Assessment and Finding of No Significant Impact related to your application for exemption dated April 3, 1997, as supplemented by letter dated June 19, 1997. The proposed exemption would permit the use of the 1996 Addenda to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), Section XI, Appendix G, for the development of pressure-temperature limits and low-temperature overpressure protection (LTOP) system setpoints.

The assessment is being forwarded to the Office of the Federal Register for publication.

Sincerely,

ORIGINAL SIGNED BY:

George F. Dick, Jr., Senior Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket Nos. STN 50-454, STN 50-455
STN 50-456 and STN 50-457

Enclosure: Environmental Assessment

cc w/encl: See next page

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OP-1

January 8, 1998

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Commonwealth Edison Company
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Docket Nos. STN 50-454, STN 50-455
STN 50-456 and STN 50-457

Enclosure: Environmental Assessment

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

January 8, 1998

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Executive Towers West III
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Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket Nos. STN 50-454, STN 50-455
STN 50-456 and STN 50-457

Enclosure: Environmental Assessment

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UNITED STATES NUCLEAR REGULATORY COMMISSION

COMMONWEALTH EDISON COMPANY

DOCKET NOS. STN 50-454, STN 50-455, STN 50-456 AND STN 50-457

BYRON STATION, UNITS 1 AND 2, AND BRAIDWOOD STATION, UNITS 1 AND 2

ENVIRONMENTAL ASSESSMENT AND FINDING OF NO SIGNIFICANT IMPACT

The U.S. Nuclear Regulatory Commission (the Commission) is considering issuance of an exemption from certain requirements of its regulations to Facility Operating License Nos. NPF-37, NPF-66, NPF-72 and NPF-77, issued to Commonwealth Edison Company (the licensee), for operation of Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, located in Ogle County and Will County, Illinois, respectively.

ENVIRONMENTAL ASSESSMENT

Identification of the Proposed Action:

The proposed action would permit the licensee to use the 1996 Addenda to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), Section XI, Appendix G, to determine the reactor vessel pressure-temperature (P-T) limits and the low-temperature overpressure protection (LTOP) system setpoints. By application dated April 3, 1997, as supplemented by letter dated June 19, 1997, the licensee requested an exemption from certain requirements of 10 CFR Part 50.60, "Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operation." The exemption would allow application of an alternate methodology to determine the P-T limits and LTOP system setpoints for Byron, Units 1 and 2, and Braidwood, Units 1 and 2. The proposed alternate methodology is consistent with guidelines developed by the ASME Working Group on Operating Plant Criteria to define pressure limits during LTOP events that avoid certain unnecessary operational restrictions, provide adequate margins against failure of the reactor pressure vessel, and reduce the potential for unnecessary activation of pressure relieving devices used for LTOP. These guidelines have been incorporated into the 1996 Addenda to the ASME Code, Section XI, Appendix G. However, 10 CFR 50.55a, "Codes and Standards," has not been updated to reflect the acceptability of the 1996 Addenda to the ASME Code.

The Need for the Proposed Action:

Pursuant to 10 CFR 50.60, all lightwater nuclear power reactors must meet the fracture toughness requirements for the reactor coolant pressure boundary as set forth in 10 CFR Part 50, Appendix G. 10 CFR Part 50, Appendix G, defines P-T limits during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests to which the pressure boundary may be subjected over its service lifetime, and specifies that these P-T limits must be at least as conservative as the limits obtained by following the methods of analysis and the margins of safety of the ASME Code, Section XI, Appendix G. 10 CFR 50.55a requires that any reference to ASME Code, Section XI, in 10 CFR Part 50 refers to addenda through the 1988 Addenda and editions through the 1989 Edition of the Code, unless otherwise noted. 10 CFR 50.60(b) specifies that alternatives to the described requirements in 10 CFR Part 50, Appendix G, may be used when an exemption is granted by the Commission under 10 CFR 50.12.

To prevent transients that would produce excursions exceeding the P-T limits while the reactor is operating at low temperatures, the licensee installed the LTOP system. The LTOP system includes pressure relieving devices called power-operated relief valves (PORVs) that are set to open at reduced pressure when reactor pressure and temperature are reduced. The

PORVs prevent the pressure in the reactor vessel from exceeding the P-T limits. However, to prevent the PORVs from lifting as a result of normal operating pressure surges, some margin is needed between the normal operating pressure and the PORV setpoint. In addition, when instrument uncertainty is considered, the operating window between the PORV setpoint and the minimum pressure required for reactor coolant pump seals is small and presents difficulties for plant operation.

To prevent pressure from exceeding the P-T limits, the PORVs would be set to open at a pressure very close to the normal pressure inside the reactor. With the PORV setpoint close to the normal operating pressure, minor pressure perturbations that typically occur in the reactor could cause the PORVs to open. This is undesirable from the safety perspective because after every PORV opening there is some concern that the PORV may not reclose. A stuck open PORV would continue to discharge primary coolant and reduce reactor pressure until the discharge pathway was closed by operator action.

The licensee requested use of the 1996 Addenda to the ASME Code, Section XI, Appendix G. These addenda to the Code would permit a slightly higher pressure inside the reactor and a slightly higher PORV setpoint during low-temperature, shutdown conditions. This would reduce the likelihood for inadvertent opening of the PORVs.

Appendix G of the ASME Code requires that the P-T limits be calculated: (a) using a safety factor of two on the principal membrane (pressure) stresses, (b) assuming a flow at the surface with a depth of one quarter (1/4) of the vessel wall thickness and a length of six (6) times its depth, and (c) using a conservative fracture toughness curve that is based on the lower bound of static, dynamic, and crack arrest fracture toughness tests on material similar to the Byron/Braidwood reactor vessel material.

For determining the P-T limits, ComEd proposed to use the safety margins based on the 1996 Addenda to the ASME Code in lieu of the 1989 Edition. When compared to the 1989 Edition of the ASME Code, the 1996 Addenda permits the use of a lower stress intensity factor for determining the applied stress intensity due to pressure and thermal stresses. This results in a slight reduction in the applied stress intensity and a corresponding shift in the allowable pressure at a given temperature in the non-conservative direction; however, this difference is minor when compared to the explicit conservatism incorporated into the Code, and the changes in the stress intensity factor are supported by the work performed by J. A. Keeney and T. L. Dickson at Oak Ridge National Laboratory (ORNL) for the NRC, and others.

1996 Addenda to the ASME Code require that the system pressure is maintained below the P-T limits during normal operation, but allows the pressure that may occur during LTOP events to exceed the P-T limits, provided acceptable margins are maintained during these events. This approach protects the pressure vessel from LTOP events, and maintains the P-T limits applicable for normal heatup and cooldown in accordance with 10 CFR Part 50, Appendix G, and Sections III and XI of the ASME Code.

In determining the PORV setpoint for LTOP events, the licensee proposed to use the safety margins of the 1996 Addenda to the ASME Code, Section XI, Appendix G. This alternate methodology allows determination of the setpoint for LTOP events such that the maximum pressure in the vessel will not exceed 110 percent of the P-T limits that are developed using the 1996 Addenda to the ASME Code, Section XI, Appendix G, methodologies described above. This results in a safety factor of 1.8 on the principal membrane stresses. All other factors, including the assumed flaw size and fracture toughness, remain the same. Although this methodology would reduce the safety factor on the principal membrane stresses, use of the proposed criteria will provide adequate margins of safety for the reactor vessel during LTOP events.

Use of the 1996 Addenda to the ASME Code, Section XI, Appendix G, safety margins will reduce operational challenges during low temperature, low pressure operations. In terms of overall safety, the safety benefits derived from simplified operations and the reduced potential for undesirable opening of the PORVs will more than offset the reduction of the principal membrane stress safety factor that may occur during LTOP events. Reduced operational challenges will reduce the potential for undesirable impacts to the environment.

It should be noted that the provision to set the PORV setpoint such that it protects 110 percent of the P-T limits is already part of the Byron and Braidwood licensing basis. This provision was approved in the exemption to 10 CFR 50.60 granted to Byron on November 29, 1996, and to Braidwood on July 13, 1995, and December 12, 1997, for Units 1 and 2, respectively, to allow the use of ASME Code Case N-514. Therefore, while it represents a change from the 1989 Edition of the ASME Code, it is not a change to the licensing basis for these facilities.

Environmental Impacts of the Proposed Action:

The Commission has completed its review of the proposed action and concludes that the proposed action involves features located entirely within the protected areas as defined in 10 CFR Part 20.

The proposed action will not result in an increase in the probability or consequences of accidents or result in a change in occupational or offsite dose. Therefore, there are no radiological impacts associated with the proposed action.

The proposed action will not result in a change in nonradiological plant effluent and will have no other nonradiological environmental impact.

Accordingly, the Commission concludes that there are no environmental impacts associated with this action.

Alternatives to the Proposed Action:

Since the Commission has concluded there is no measurable environmental impact associated with the proposed action, any alternatives with equal or greater environmental impact need not be evaluated. As an alternative to the proposed action, the staff considered denial of the proposed action. Denial of the application would result in no change in current environmental impacts. The environmental impacts of the proposed action and the alternative action are similar.

Alternative Use of Resources:

This action does not involve the use of any resources not previously considered in the Final Environmental Statement for the Byron Station or the Braidwood Station.

Agencies and Persons Consulted:

In accordance with its stated policy, on January 9, 1998, the staff consulted with the Illinois State official, Frank Niziolek of the Illinois Department of Nuclear Safety, regarding the environmental impact of the proposed action. The State official had no comments.

FINDING OF NO SIGNIFICANT IMPACT

Based upon the environmental assessment, the Commission concludes that the proposed action will not have a significant effect on the quality of the human environment. Accordingly, the Commission has determined not to prepare an environmental impact statement for the proposed action.

For further details with respect to the proposed action, see the licensee's letter dated April 3, 1997, as supplemented by letter dated June 19, 1997, which are available for public inspection at the Commission's Public Document Room, The Gelman Building, 2120 L Street, NW., Washington, DC, and at the Local Public Document Room located: For Byron, the Byron Public Library District, 109 N. Franklin, P.O. Box 434, Byron, Illinois 61010; for Braidwood, the Wilmington Public Library, 201 S. Kankakee Street, Wilmington, Illinois 60481.

Dated at Rockville, Maryland, this 9th day of January 1998.

FOR THE NUCLEAR REGULATORY COMMISSION

George F. Dick Jr

George F. Dick, Jr., Senior Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation