Commonwealth Edison Company
1400 Opus Place
Downers Grove, IL 60515-5701

January 30, 1997

ComEd

United States Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Document Control Desk

Subject: Byron Nuclear Power Station, Units 1 & 2
Facility Operating Licenses NPF-37 & NPF-66
NRC Docket No. 50-454 and 50-455

Braidwood Nuclear Power Station, Units 1 & 2
Facility Operating Licenses NPF-72 & NPF-77
NRC Docket No. 50-456 and 50-457

"Primary Containment" and "Reactor Coolant System Volume"

Pursuant to Title 10, Code of Federal Regulations, Part 50, Section 90 (10CFR 50.90), Commonwealth Edison Company (ComEd) proposes to amend Appendix A, Technical Specifications, for Facility Operating Licenses NPF-37 and NPF-66 for Byron Nuclear Power Station, Units 1 & 2 (Byron), and Facility Operating Licenses NPF-72 and NPF-77 for Braidwood Nuclear Power Station, Units 1 and 2 (Braidwood).

Please note that although the proposed Technical Specifications amendment is applicable to Byron and Braidwood Unit 1 only, this license amendment request is being docketed to reflect Byron Units 1 and 2 and Braidwood Units 1 and 2 due to common Technical Specification pages being used for both units.

ComEd proposes to revise Technical Specification (TS) 1.0, "Definitions," 3/4.6.1, "Primary Containment" and associated Bases and 5.4.2, "Reactor Coolant System Volume" for Byron and Braidwood to support steam generator replacement. ComEd will be replacing the original Westinghouse D4 steam generators (OSGs) at Byron and Braidwood with Babcock & Wilcox International (BWI) steam generators. The replacement steam generators (RSGs) increase the Reactor Coolant System (RCS) volume which results in a higher calculated peak containment pressure (P_a) value

070021

A001 1/1

k\nla\byrbwd\stmgens\parcs1.doc

The RCS volume and P_a for Unit 2 will remain unchanged. Additionally, several editorial changes are being made to improve clarity and consistency of the TS.

This package consists of the following:

Attachment A Description and Safety Analysis of Proposed Changes to Appendix A

Attachment B Proposed Changes to the Technical Specification Pages for Byron and Braidwood Stations

Attachment C Evaluation of No Significant Hazards

Attachment D Environmental Assessment

Please note the affected improved Technical Specifications (ITS) pages will be prepared and submitted at a later date showing the proposed changes for Byron and Braidwood.

The proposed changes in this license amendment have been reviewed and approved by both On-Site and Off-Site review in accordance with ComEd procedures.

ComEd is notifying the State of Illinois of our application for this license amendment request by transmitting a copy of this letter and its attachment to the designated State Official.

The Byron Unit 1 Steam Generator Replacement Outage (SGRO) is scheduled during the eighth refuel outage (B1R08). The Braidwood Unit 1 SGRO is scheduled during the seventh refuel outage (A1R07). ComEd respectively requests the NRC Staff review and approve this license amendment request no later than November 3, 1997, to support the current outage schedule for the lead station, Byron Unit 1.

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

Please address any comments or questions regarding this matter to Marcia Lesniak, Nuclear Licensing Administrator at (630) 663-6484.

Sincerely,

OFFICIAL SEAL
JACQUELINE T EVANS
NOTARY PUBLIC, STATE OF ILLINOIS
MY COMMISSION EXPIRES: 12/15/97

John B Hosmer
John B. Hosmer
Vice President

Signed before me on this 30th day of January, 1997 by

Jacqueline T. Evans
Notary Public

Attachment A: Description and Safety Analysis of the Proposed Changes

Attachment B-1: Proposed Changes to Appendix A, Technical Specification, for the Byron Nuclear Power Plant, Units 1 & 2

Attachment B-2: Proposed Changes to Appendix A, Technical Specification, for the Braidwood Nuclear Power Plant, Units 1 & 2

Attachment C: Evaluation of Significant Hazards

Attachment D: Environmental Assessment

cc: A. B. Beach, Regional Administrator - RIII
G. F. Dick, Jr., Byron/Braidwood Project Manager - NRR
S. D. Burgess, Senior Resident Inspector - Byron
C. J. Phillips, Senior Resident Inspector - Braidwood
Office of Nuclear Safety - IDNS

ATTACHMENT A

DESCRIPTION AND SAFETY ANALYSIS OF
PROPOSED CHANGES TO APPENDIX A
TECHNICAL SPECIFICATIONS OF
FACILITY OPERATING LICENSES
NPF-37, NPF-66, NPF-72, AND NPF-77

A. DESCRIPTION OF THE PROPOSED CHANGE

Commonwealth Edison (ComEd) proposes to revise Technical Specifications (TS) 1.0, "Definitions," 3/4.6.1, "Primary Containment" and associated Bases, and 5.4.2, "Reactor Coolant System Volume," for Byron Nuclear Power Station (Byron) and Braidwood Nuclear Power Station (Braidwood) to support steam generator replacement. ComEd will be replacing the original Westinghouse D4 steam generators (OSGs) at Byron and Braidwood with Babcock and Wilcox International (BWI) steam generators. The replacement steam generators (RSGs) increase the Reactor Coolant System (RCS) volume which results in a higher calculated peak containment pressure (P_a) value. The RCS volume and P_a for Unit 2 will remain unchanged. Additionally, several editorial changes are being made to improve clarity and consistency of the TS.

These proposed changes are discussed in detail in Section E of this attachment. Affected TS pages showing the proposed changes are included in Attachments B-1 and B-2 for Byron and Braidwood, respectively. Affected Improved Technical Specifications (ITS) pages will be prepared and submitted at a later date showing the proposed changes for Byron and Braidwood.

B. DESCRIPTION OF THE CURRENT REQUIREMENT

Definition 1.20a specifies that P_a is the maximum calculated primary containment pressure related to the Byron and Braidwood design basis loss-of-coolant accident (LOCA). The value of P_a is specified as 44.4 psig.

Technical Specification Surveillance Requirement (TSSR) 4.6.1.3 requires leakage testing of containment airlocks to assure that the overall air lock leakage will not become excessive due to seal damage. Specific leakage rates (at specified test pressures) are provided in both a percentage of the maximum allowable primary containment leak rate, L_a, and equivalent numerical value in standard cubic feet per hour (SCFH).

TSSR 4.6.1.7.3 requires that at least once per 6 months, the containment inboard and outboard 48-inch containment purge supply and exhaust isolation valves with resilient material seals be demonstrated to have a leakage rate no greater than the specified limit. This leakage rate must be

less than 0.05 of the maximum allowable primary containment leakage rate, L_a, when pressurized to at least the maximum calculated primary containment pressure, P_a (44.4 psig).

TSSR 4.6.1.7.4 requires that at least once per 3 months, each 8-inch containment purge supply and exhaust isolation valve with resilient material seals be demonstrated to have a leakage rate less then 0.01 L_a when pressurized to at least the maximum calculated primary containment pressure, P_a (44.4 psig).

Bases 3/4.6.1.4 describes the bases for limitations on the containment internal pressure limit. The maximum pressure limit ensures that the maximum calculated primary containment pressure following a design basis LOCA does not exceed the containment design pressure of 50 psig.

Bases 3/4.6.1.6 describes the bases for surveillance requirements on the structural integrity of the containment. These surveillance requirements assure the containment will withstand the maximum pressure of 44.4 psig following a design basis LOCA.

TS 5.4.2 indicates 12,257 cubic feet for the total water and steam volume of the Reactor Coolant System at a nominal T_{avg} of 588.4 °F for each unit. This information is provided as part of the "Design Features" section of the Byron and Braidwood Technical Specifications and does not represent a limiting condition for operation.

C. BASES FOR THE CURRENT REQUIREMENT

The 44.4 psig P_a value provided in Definition 1.20a represents the maximum calculated primary containment pressure with the original Westinghouse model D4 and D5 steam generators. This value is calculated using NRC approved analysis codes and modeling techniques. The maximum calculated pressure is less than the containment design pressure of 50 psig.

TSSRs 4.6.1.3.a, 4.6.1.3.d, 4.6.1.7.3, and 4.6.1.7.4 are in place to assure containment leakage rates are controlled to an acceptable limit such that the dose at the Exclusion Area Boundary is held to a small fraction of the Title 10 Code of Federal Regulations Part 100 (10CFR100) limits.

According to Bases 3/4.6.1.4 and 3/4.6.1.6, the containment operating restrictions and surveillance requirements assure the integrity of the containment building is maintained following a design basis LOCA. The containment is not required to be a complete and independent safeguard against a LOCA by itself, but functions to contain any fission products released while the emergency core cooling system cools the reactor core. Containment building integrity is essential to assure the dose at the Exclusion Area Boundary is held to a small fraction of the 10CFR100 limits. Maximum calculated primary containment pressure, P_a, is calculated to demonstrate that the peak containment pressure is held to less than the design pressure, and is used as an input for the containment leakage rate testing.

TS 5.4.2 is a statement of the volume of the reactor coolant system with the plant in its original configuration, which includes the Westinghouse Model D4 or D5 steam generators.

D. NEED FOR REVISION OF THE REQUIREMENT

Each of the RSGs has a larger primary side volume than the OSGs. TS 5.4.2 provides information on the total RCS volume and will require a revision to reflect the volume increase associated with the Steam Generator Replacement Project.

As a result of the RCS volume increase, the mass and energy release during the blowdown phase of the large break LOCA is increased. Additionally, the heat transfer rate of the RSGs is greater than the OSGs, and the RSGs will operate at a slightly higher pressure than that for the OSGs. Consequently, the steam enthalpy exiting the break during the reflood period, with the RSG, will be greater than that for the OSG. This results in an increase in the containment building peak pressure, P_a. The specific increase was identified in the Containment LOCA Analysis performed by Framatome Technologies, Inc., in support of the Steam Generator Replacement Project. This analysis identified an increase in P_a from the current value of 44.4 psig to a value of 47.8 psig associated with the RSGs.

The revised P_a value affects the pressure used for containment leak rate testing, but does not affect plant operation. TSSR 4.6.1.3 contains specific allowable leakage rates (at specified test pressures) for use in containment air lock leakage surveillance testing. Text which provides specific values of these allowable leakage rates are provided in terms of both a percentage of the maximum allowable primary containment leak rate, L_a, and its equivalent numerical value in SCFH. L_a is a function of the maximum calculated primary containment pressure, P_a, and, therefore, will change as a result of the proposed change in the Unit 1 P_a value. The applied acceptance criteria values of 0.0024 L_a and 0.01 L_a for these tests remain unchanged.

E. DESCRIPTION OF THE REVISED REQUIREMENT

Definition 1.20a, which is P_a, will be revised to distinguish the revised Unit 1 P_a value from the Unit 2 P_a value. The current value in parentheses will be replaced with the following:

"(44.4 psig*, 47.8 psig**)"

The following footnotes associated with the value change will be added to the affected Definitions page for Byron:

"*Applicable to Unit 1 through cycle 8 and to Unit 2.
**Not applicable to Unit 2. Applicable to Unit 1 after cycle 8."

The following footnotes will be added to the affected Definitions page for Braidwood:

> "*Applicable to Unit 1 through cycle 7 and to Unit 2.
> **Not applicable to Unit 2. Applicable to Unit 1 after cycle 7."

The specific numerical values of P_a and percentages of L_a will be deleted from the following TSSRs:

> TSSR 4.6.1.3.a(1)
> TSSR 4.6.1.3.a(2)
> TSSR 4.6.1.3.d
> TSSR 4.6.1.7.3
> TSSR 4.6.1.7.4

For Bases 3/4.6.1.4 and 3/4.6.1.6, the specific numerical values for P_a will be deleted. The words "44.4 psig" will be replaced with the words "defined as P_a" in the first sentence in the second paragraph of 3/4.6.1.4. All other references to the specific value of 44.4 psig will be replaced with "P_a" in this change.

Technical Specification Design Features section 5.4.2 will be revised to account for the additional 1,251 ft^3 of RCS volume associated with the RSGs. The change will add the following statement for Byron:

> "An additional 1,251 ft^3 of volume is added to the Unit 1 total RCS volume as a result of the four replacement steam generators installed after Cycle 8."

The change will add the following statement for Braidwood:

> "An additional 1,251 ft^3 of volume is added to the Unit 1 total RCS volume as a result of the four replacement steam generators installed after Cycle 7."

F. BASES FOR THE REVISED REQUIREMENT

The UFSAR identifies two events that are analyzed to determine the peak containment building pressure: the large break LOCA (LBLOCA) and the main steam line break (MSLB). Analysis has determined that the LBLOCA is the bounding analysis for peak containment pressure. Critical parameters for the LBLOCA are initial primary mass and the initial internal energy in the primary system.

The primary side volume of the RSGs is greater than the OSGs. This increases the mass and energy release during the blowdown phase of the LBLOCA. Additionally, the heat transfer rate of the RSGs is greater than the OSGs, and the RSGs will operate at a slightly higher pressure than

that for the OSGs. Consequently, the steam enthalpy exiting the break during the reflood period, with the RSG, will be greater than that for the OSG. This results in an increase in the containment building peak pressure.

The determination of the impact of the BWI RSGs on the design basis LOCA peak containment pressure was performed per the methodology of NRC approved Topical Report BAW-10095A, "CONTEMPT - Computer Program for Predicting Containment Pressure-Temperature Response to a Loss-of-Coolant Accident." The evaluation of the effect of the RSGs on the containment building response was determined in terms of an increase to the containment pressure and temperature to be applied to the existing analysis of record using the OSGs. A base calculation was performed utilizing the mass and energy releases from the current analysis for the OSGs. The calculation was then performed using the mass and energy releases associated with the RSGs. All other initial conditions and assumptions were maintained. The two calculations were compared to determine the incremental change in the peak containment pressure and temperature caused by use of the RSGs. Applying this difference to the peak pressure with the OSGs yields a peak containment pressure following a LBLOCA of 47.8 psig. This is the basis for the revised maximum calculated primary containment pressure, P_a, in Definition 1.20a.

The changes proposed to TSSRs 4.6.1.3 and 4.6.1.7 are administrative in nature. In each case, a redundant numerical value is presented along with specified requirements that are either already provided in the "Definitions" section of the TS or are predetermined as a percentage of a specified requirement that is already provided in the "Definitions" section. In no case will the requirement be altered other than by the need to address unit specific values. This can be done once in the "Definitions" section of the Technical Specifications, therefore, simplifying the presentation of the requirements within the TSSRs. This approach is consistent with overall use of definitions throughout the Technical Specifications for values associated with containment leak rate testing values and consistent with the Improved Standard Technical Specification format.

Likewise, the Bases sections 3/4.6.1.4 and 3/4.6.1.6 include specific references to the current value of P_a as 44.4 psig. Removal of these specific values and replacement with the defined value, P_a, is consistent with the use of definitions and provides a means to address the unit specific differences associated with P_a.

The revised TS Design Features, 5.4.2, accounts for the Unit 1 RSGs addition of a total of 1,251 cubic feet of water volume to the RCS. The affects of the increase in total RCS volume were evaluated for all UFSAR, Chapter 15 accidents. The only impact on the Technical Specification resulted from the increased mass and energy release following a LBLOCA event. The impact was that the increased RCS volume was a contributor to the increase in the maximum calculated primary containment pressure, P_a, value. Therefore, the basis for acceptability of the revised RCS volume is addressed by the basis of the P_a increase presented above.

G. IMPACT OF THE PROPOSED CHANGE

The proposed changes to the value of P_a do not require any physical changes to the plant systems, structures, or components. The need for the change is, however, a direct result of the physical changeout of the OSGs and the subsequent increase in the primary RCS volume associated with the RSGs. No instrument setpoint or actuation signals are affected by this change. The requirements of GDC-50, "Containment Design Basis;" GDC-52, "Capability for Containment Leak Rate Testing;" GDC-53, "Provisions for Containment Testing and Inspection;" and GDC-16, "Containment Design" are unaffected by this change.

The Unit 1 plant leakage testing procedures that incorporate values of P_a and L_a will be revised. This change (increased P_a from 44.4 to 47.8 psig for Unit 1 only) will directly impact surveillance requirements by requiring that containment leakage be demonstrated to be less than the maximum allowable primary containment leakage when pressurized to the higher containment pressure than was previously utilized for the OSG plant configuration. The surveillance requirements are in place to assure containment leakage rates are maintained at an acceptable limit such that the dosage at the exclusion area boundary is held to a small fraction of the 10CFR100 limits.

The offsite dose calculation for the LBLOCA is unaffected by the proposed change. The license basis offsite dose calculation is in accordance with NRC Reg Guide 1.4 "Assumptions Used for Evaluating The Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors." This Regulatory Guide states, in part, "...a number of appropriately conservative assumptions, based on engineering judgment and on applicable experimental results from safety research programs conducted by the AEC." These conservatisms include (but are not limited to) the following assumptions;

- Twenty five percent of the equilibrium radioactive full power iodine inventory is immediately available for leakage from the primary containment.
- 100 % of the equilibrium full power radioactive noble gas inventory is immediately available for leakage from the primary containment.
- The primary containment should be assumed to leak at the (maximum) leak rate specified in the technical specifications for the first 24 hours and at 50% of this value for the remaining 29 days of the accident duration.

The design basis leakage corresponding to a peak containment pressure of 50 psig utilized in the design basis accident analysis is 0.10% per day of the containment free air mass. Therefore, the offsite dose calculation was performed with a leakage of .1 % per day for day one and .05 % per day for days two through 30. Isotopic inventories are unaffected by the increase in reactor coolant volume. Thus, the offsite dose, as conservatively calculated by the NRC Regulatory Guide 1.4, is unaffected by the increase in the peak containment pressure.

In addition to evaluating the effect of the RSGs on peak containment pressure, the effect of the increased mass and energy release associated with the RSGs on the equipment qualification temperature inside containment was evaluated. The analysis performed showed that the containment building vapor temperature remains within the equipment qualification envelope at all times following a LBLOCA or MSLB with the RSGs. Therefore, there is no impact on operations, procedures, or materials as a result of the temperature effects associated with the RSGs.

H. SCHEDULE REQUIREMENTS

The Byron Unit 1 Steam Generator Replacement Outage (SGRO) is scheduled during the eighth refuel outage (B1R08). The Braidwood Unit 1 SGRO is scheduled during the seventh refuel outage (A1R07). Approval of this change is requested by November 3, 1997 to support the current outage schedule for the lead station which is Byron Unit 1.