Commonwealth Edison
One First National Plaza, Chicago, Illinois
Address Reply to: Post Office Box 767
Chicago, Illinois 60690

February 6, 1985

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Byron Generating Station Units 1 and 2
Braidwood Generating Station Units 1 and 2
FSAR Changes
NRC Docket Nos. 50-454/455 and 50-456/457

Dear Mr. Denton:

This letter provides advance copies of revised pages for the Byron/Braidwood FSAR. These changes are being made to provide more explicit descriptions of the design bases for these plants.

Enclosed is a revised page 8.1-14. It now includes a discussion of conformance to IEEE 420-1973 with regard to the allowed use of cable splices within control switchboards. This change is being made to resolve a concern identified during a recent I&E inspection at Braidwood Station.

Also enclosed are revised pages E.20-1, E.20-1a, and E.82-1 for Appendix E of the Byron/Braidwood FSAR. They now specify that the liquid source term used for shielding evaluation and environmental qualification of Byron and Braidwood does not include noble gases. Fission solids are the dominant contributor for long term doses so this does not significantly alter the radiological impact of postulated accidents.

Also included is a revised page E.21-3. It specifies that backup sampling is not provided for hydrogen because of the large volume required. These changes will be incorporated into the FSAR at the earliest opportunity. Please direct questions regarding these matters to this office.

One signed original and fifteen copies of this letter are provided for NRC review.

Very truly yours,

David H Smith
for T. R. Tramm
Nuclear Licensing Administrator

lm

cc: Byron Resident Inspector
Braidwood Resident Inspector

B021
1/1

The physical identification of safety-related equipment is discussed in Subsection 8.3.1.3.

8.1.15 Shared Emergency and Shutdown Electric systems for Multi-Unit Nuclear Power Plants

The criteria followed in designing the two unit station is that each unit shall operate independently of the other and malfunction of equipment or operator error in one unit will not initiate a malfunction or error in the other unit nor affect the continued operation of the other unit.

8.1.16 Qualification of Class 1E Equipment for Nuclear Power Plants

With regard to environmental qualification of instrumentation, control, and electrical equipment important to safety, the Applicant complies with the intent of IEEE 323-1974. Additional information is provided in Section 3.11.

8.1.17 Availability of Electric Power Sources

During abnormal electric power source configurations, plant operations are limited as described in Subsection 16.3/4.8.

8.1.18 Conformance to IEEE 338-1975 (Periodic Testing of Nuclear Power Generating Station Class 1E Power and Protection System)

Conformance to this standard is addressed in Subsection 8.3.1.2 and 7.1.2.19.

8.1.19 Conformance to IEEE 344-1971 (Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Station)

Conformance to this Standard is addressed in Section 3.10.

8.1.20 Conformance to IEEE 387-1972 (Criteria for Diesel Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations)

Vendor qualification tests, preoperational testing, and periodic testing during normal plant operation conform to those procedures described in this standard, except as noted in Subsections 8.3.1.2, 16.3/4.8, and Chapter 14.0.

8.1.21 Conformance to IEEE 420-1973 (IEEE Trial-Use Guide for Class 1E Control Switchboards for Nuclear Power Generating Stations)

Class 1E control switchboards conform to this standard with the following clarification to Paragraph 4.6.1.2: Splices may be used on individual conductors of field cables within switchboards for the purpose of extending individual conductors to their point of termination.

E.20 PLANT SHIELDING (II.B.2)

POSITION:

A radiation and shielding design review was conducted for Byron/Braidwood Stations (B/B) using the guidance provided in NUREG-0737. Reference 9 describes in detail the source terms used for radiation qualification. Inside the containment, the source term is based on the release of 100% of the noble gases and 50% of the halogens to the containment atmosphere. Systems containing liquid sources are analyzed using a source term consisting of 50% of the core inventory of halogens and 1% of the core inventory of fission solids.

Note that the liquid source term does not contain noble gases as specified in this clarification. This is not done because it is inconsistent with a severe accident, and since the fission solids are the dominant nuclide group for long term doses (one year period), there is no effect on the evaluation of the radiological impact.

Two accident scenarios were considered:

a. Line Break Accident - a loss-of-coolant accident initiated by a major break in a primary coolant pipe; and

b. No Line Break Accident - an accident during which all activity released from the fuel remains in the primary coolant.

The postaccident radiation environment was determined by (1) analyzing each system operating following an accident to establish pathways for fission products out of containment; (2) investigating process streams in the auxiliary building to identify contaminated equipment and associated activity levels; (3) calculating the radiation field due to each source with computer codes ISOSHLD (Ref. 3), QAD (Ref. 4), and G^3 (Ref. 5); and, (4) superimposing the effects of all sources to obtain the maximum expected dose rate throughout the plant. The radiation environment was evaluated 1 hour, 1 day, and 1 week following the reactor shutdown that precedes fuel failure.

The potentially contaminated systems considered in identifying postaccident radiation sources included (1) the emergency core cooling system, which consists of all or parts of the safety injection, residual heat removal, and chemical and volume control systems; (2) the containment spray and hydrogen recombiner systems, which assist in maintaining containment integrity; (3) the control room, technical support center, and auxiliary building HVAC systems, portions of which are designed to remove airborne fission products; (4) the high radiation sampling system; (5) those parts of the chemical and volume control and the boron recycle systems associated with charging, letdown, and seal water; (6) the shutdown cooling portion of the residual heat removal (RHR) system; and, (7) those parts of the gaseous waste processing system and the liquid radwaste system which would normally operate in conjunction with the other systems under consideration. The large amount of equipment in the auxiliary building operating following an accident produces elevated dose rates in many areas. However, since no operator actions other

d. a special cart equipped with a shielding cask to transport the radioactive sample to its destination; and,

e. a ventilation system drawing air out of the sampling panels and discharging into a remote HVAC train.

Due to the high dose associated with an alternate means of sampling for chlorides, no backup sampling capability is in place for such a sample.

AIR SAMPLING SUBSYSTEM

The containment air sampling panel is capable of sampling the primary containment atmosphere. The sample is drawn from the containment through a dedicated penetration. Once the interfacing valves are arranged and the sampling programmer is inititiated, the containment air sampling panel utilizes automatically sequenced sampling to trap the designated sample in a shielded cart. The air sample will then be analyzed onsite. Excessive exposure to the operator is limited by:

a. steel shielding in the containment air sampling panel;

b. concrete shielding above, below, and around the sides of the panel to prevent radiation from scattering around the steel shielding;

c. automatic sampling;

d. special carts each equipped with a shielding cask to transport the radioactive sample to its destination; and,

e. a ventilation system drawing air out of the sampling panels and discharging into a remote HVAC train.

Due to the large volume required for a hydrogen sample, no backup sampling capability is in place for such a sample.

SAMPLING PROGRAM FREQUENCY

Actual frequency of sampling shall be determined by station management, however, as a minimum the first sample can be taken within 1 hour from the time a decision is made to take a sample, continuing with at least one sample per day for the next 7 days and at least one sample per week thereafter. The time interval between taking a sample and receipt by plant management of the results of the analysis is estimated to be less than 2 hours.

E.82 REFERENCES

1. Letter from B. J. Youngblood (NRC) to J.S. Abel (Commonwealth Edison) dated June 11, 1981, regarding NUREG-0737 item I.C.1.

2. Letter from D. G. Eisenhut (NRC) to R. W. Jurgensen (Commonwealth Edison) dated May 28, 1981.

3. R. L. Engel, J. Greenborg, and M. M. Hendrickson, "ISOSHLD -- A Computer Code for General-Purpose Isotope Shielding Analysis," BNWL-236, Pacific Northwest Laboratory, Richland, Washington, June 1966; Supplement 1, March 1967; Supplement 2, April 1969.

4. R. E. Malenfant, "QAD -- A Series of Point-Kernel General-Purpose Shielding Programs," LA-3573, Los Alamos Scientific Laboratory, April 5, 1967.

5. R. E. Malenfant, "G^3 -- A General-Purpose Gamma-Ray Scattering Program," LA-5176, Los Alamos Scientific Laboratory, June 1973.

6. Letter from R. C. Youngdahl (Consumers Power) to H. R. Denton (NRC) dated July 1, 1981.

7. Letter from R. C. Youngdahl (Consumers Power) to D. G. Eisenhut (NRC) dated December 15, 1980.

8. Letter from D. F. Ross, Jr. (NRC) to all pending operating license applicants of nuclear steam supply systems designed by Westinghouse and Combustion Engineering, dated March 10, 1980.

9. "Byron Station Units 1 and 2, Braidwood Station Units 1 and 2, Environmental Qualification of Electrical Equipment, NRC Docket Nos. 50-454, 50-455, 50-456, 50-457," letter from T. R. Tramm, Commonwealth Edison Company to Harold R. Denton, Director, Office of Nuclear Reactor Regulation, USNRC, dated June 17, 1982, with attachments.