Commonwealth Edison
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Address Reply to: Post Office Box 767
Chicago, Illinois 60690

October 11, 1984

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20005

SUBJECT: Byron Generating Stations Units 1 and 2,
Braidwood Generating Stations Units 1 and 2
Technical Specifications
NRC Docket Nos. 50-454, -455, -456, and 457

Reference: (a) July 19, 1984 letter from B.J. Youngblood to D.L. Farrar
(b) July 26, 1984 letter from E.D. Swartz to H.R. Denton

Dear Mr. Denton:

This letter provides supplemental responses to NRC questions regarding the proposed technical specifications for Byron Station. NRC review of these responses is needed to resolve the remaining concerns of the Reactor Systems Branch reviewers.

In reference (b), Commonwealth Edison's responses were provided to the NRC's questions transmitted in reference (a). Questions 1, 4, 10, and 11 raised issues which appear to be generic to Westinghouse PWR's, so we committed to resolve those issues through the efforts of the Westinghouse Owners' Group (WOG). After review of those responses with the NRC staff, we recognize the need for a committment to a plant-specific resolution of these issues in the event that the WOG efforts do not satisfactorily address the NRC's concerns. We propose to do just that. If the WOG is unable to resolve the generic technical specification issues relating to operability of pressurizer relief valves, main steam isolation valves (in mode 4), power range neutron monitor flux trips (in modes 3 and 4), and the residual heat removal

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pump (in mode 4), then Commonwealth Edison will undertake to resolve these issues specifically for the Byron and Braidwood plants. This committment is also made for one of the other technical specification issues, operability of the containment pressure high-1 instrumentation in mode 4.

Enclosed with this letter are the supplemental responses to NRC questions 1, 4, 5b, 10, and 11 and the additional containment pressure instrumentation topic. Please address further questions regarding this matter to this office.

Very truly yours,

T. R. Tramm

Thomas R. Tramm
Nuclear Licensing Administrator

cc: NRC Senior Resident Inspector - Byron
Lenney Olshan

QUESTION 1:

Relief Valves (Section 3.4.4, page 3/4 4-10)

It is the staff's understanding that your steam generator tube rupture analysis presented in Chapter 15 of your FSAR relied on the availability and operability of the pressurizer power operated relief valves (PORVs) and the steam generator atmospheric dump valves (ADVs) for depressurization and cooldown in order to limit offsite doses to within 10 CFR 100 guideline values. Similarly, your cooldown evaluation in FSAR Section 5.4.7 performed to show compliance with BTP RSB 5-1 relied on the availability and operability of the PORVs and ADVs to provide the necessary depressurization and cooldown functions. Your proposed technical specifications however, appear to be inconsistent with your FSAR assumptions in that they allow the PORVs to be taken out of service for an indefinite period of time and, on the other hand, they do not contain an operability requirement for the steam generator ADVS. Please demonstrate how you comply with the requirements of 10 CFR 50.36 regarding how your technical specifications for the PORVs were derived from the FSAR safety analyses. Specifically, we believe it is necessary to show that the steam generator tube rupture criteria and the RSB 5-1 criteria can be met assuming inoperable PORVs and ADVs consistent with your proposed technical specifications. Otherwise, you should demonstrate that your technical specification is consistent with the FSAR analyses.

RESPONSE:

Concerning the Pressurizer PORVs, revised action statements for Technical Specification 3.7.7 have been made and incorporated into the "Final Draft" of the Technical Specifications.

Commonwealth Edison recognizes the need to address this issue which will be presented to the Westinghouse Owners Group for resolution on a generic basis. At the conclusion of this review, Commonwealth Edison will incorporate the results/recommendations of the Owners Group as applicable to Byron Station. In the event that the Owners Group elects not to address this issue generically, Commonwealth Edison will have a review performed and incorporate the findings of this review as appropriate.

QUESTION 4:

Plant Systems, Main Steam Isolation Valves 3.7.1.5 (page 3/4 7-9)

The Technical Specifications do not require manual isolation capability for the Main Steam Isolation valves in mode 4 (below a RCS temperature of 350°F).

Justify that in the event of a steam generator tube rupture in mode 4 that the offsite dose consequences calculated in the FSAR would not be exceeded.

RESPONSE:

Commonwealth Edison recognizes the need to address this issue which will be presented to the Westinghouse Owners Group for resolution on a generic basis. At the conclusion of this review, Commonwealth Edison will incorporate the results/recommendations of the Owners Group as applicable to Byron Station. In the event that the Owners Group elects not to address this issue generically, Commonwealth Edison will have a review performed and incorporate the findings of this review as appropriate.

QUESTION 5b:

Reactor Coolant System, Pressurizer 3.4.3 (page 3/4 4-9)

The Technical Specifications limit the pressurizer level to less than 92% for operation in modes 1, 2 and 3 and impose no limits for operation below mode 3. Justify that the recommendations of Branch Technical Positions RSB 5-1 (cold shutdown) and RSB 5-2 (TOP) can be met within the above limits in view of the following considerations.

b. A pressurizer vapor space corresponding to an indicated water level of 25% is required to permit boration to cold shutdown without letdown. (Q212.154 P.7)

RESPONSE 5b):

In the scenario described in Q212.154 the plant is assumed to be operating at power when the reactor is tripped and brought to zero load hot standby conditions. At zero percent power, pressurizer level is automatically maintained by the pressurizer level control system at 25% in accordance with the pressurizer level program. If the pressurizer level deviates from the program level by ± 5%, then an annunciator alarms making the operator aware of the pressurizer level deviation so corrective action can be taken. This can include establishing normal or excess letdown, closing the charging flow path valve or stopping the charging pumps. If the operator were manually controlling pressurizer level, he would strive to duplicate the pressurizer level program (i.e. maintain 25% level at zero percent power).

In accordance with Byron Station's Plant Shutdown and Cooldown procedure BGP 100-5, the RCS will be borated to cold shutdown, zenon-free boron concentration before the cooldown is initiated which is started in Mode 3. The cooldown can be initiated while boration is in progress if adequate Shutdown Margin is available from xenon for the duration of the boration operation. Upon reaching Hot Standby, the Shutdown Margin will be verified and calculated once per 24 hours while in Modes 3, 4 or 5 per Technical Specifications 3.1.1.1 and 3.1.1.2. Also in Hot Standby the boron concentration required for Cold Shutdown will be calculated and an evaluation of the plant conditions and the availability of equipment and systems that can be used in the shutdown will be made.

As mentioned above, at no load condition the pressurizer level will be maintained at 25% so that pressurizer level can be increased to 95% of span to provide sufficient boron to compensate for xenon decay at Hot Standby. To allow some margin for the pressurizer level not being exactly 25% the pressurizer level could be taken to 100% of span without a concern for taking the pressurizer solid. There is a steam volume in the pressurizer above the upper level tap when pressurizer level indication is exactly 100%. In addition, as the cooldown is started, the pressurizer level will shrink. This also allows additional margin if the level is not exactly 25% of span. In accordance with Byron Station procedures for a loss of normal letdown the operator will be taking actions to restore normal letdown or to establish excess letdown as soon as possible.

QUESTION 10:

Table 3.3-1, Reactor Trip Instrumentation (page 3/4 3-2)

For rod withdrawal accident at subcritical conditions, staff is under the impression that reactor trip is initiated by the power range neutron flux trip. However, the power range neutron flux trip needs only to be operable in modes 1 and 2 according to the Technical Specifications. Please explain this apparent discrepancy. If your explanation takes credit for either the intermediate range or source range trips, then the setpoint methodology will have to be amended to reflect this.

RESPONSE:

Commonwealth Edison recognizes the need to address this issue which will be presented to the Westinghouse Owners Group for resolution on a generic basis. At the conclusion of this review, Commonwealth Edison will incorporate the results/recommendations of the Owners Group as applicable to Byron Station. In the event that the Owners Group elects not to address this issue generically, Commonwealth Edison will have a review performed and incorporate the findings of this review as appropriate.

QUESTION 11:

Reactor Coolant System Hot Shutdown 3.4.1.3 (page 3/4 4-3)

Technical Specification 3.4.1.3 permits operation in mode 4 with one RHR loop in operation. Justify that the consequences of an inadvertent control rod withdrawal event with one RHR loop in operation in mode 4 would be bounded by the FSAR analysis which assumes two reactor coolant pumps in operation in mode 2. In your evaluation consider the effect of non uniform flow distribution through the core on minimum DNBR.

RESPONSE:

Commonwealth Edison recognizes the need to address this issue which will be presented to the Westinghouse Owners Group for resolution on a generic basis. At the conclusion of this review, Commonwealth Edison will incorporate the results/recommendations of the Owners Group as applicable to Byron Station. In the event that the Owners Group elects not to address this issue generically, Commonwealth Edison will have a review performed and incorporate the findings of this review as appropriate.

Additional Topic: Containment Pressure High-1 in Mode 4
(Table 3.3.3, item 1C, page 3/4 3-15)

Concerning the issue of adding Mode 4 to the Applicable Modes column for Containment Pressure-High-1, Commonwealth Edison recognizes the need to address this issue which will be presented to the Westinghouse Owners Group for resolution on a generic basis. At the conclusion of this review, Commonwealth Edison will incorporate the results/recommendations of the Owners Group as applicable to Byron Station. In the event that the Owners Group elects not to address this issue generically, Commonwealth Edison will have a review performed and incorporate the findings of this review as appropriate.

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