Docket Nos.: STN 50-454
and STN 50-455

JUL 19 1984

Mr. Dennis L. Farrar
Director of Nuclear Licensing
Commonwealth Edison Company
Post Office Box 767
Chicago, Illinois 60690

Dear Mr. Farrar:

Subject: Byron Technical Specifications - Request for Additional Information

Enclosed is a request for additional information which is required to complete the staff's evaluation of the Byron Technical Specifications. In order to support your licensing schedule, provide the requested information within seven days of the date of this letter. These items have previously been discussed with members of the CECo staff. If this schedule cannot be met, please advise the Licensing Project Manager immediately.

For further information or clarification, please contact the Byron Project Manager, Leonard N. Olshan, (301) 492-7070.

Sincerely,

/S/

B. J. Youngblood, Chief
Licensing Branch No. 1
Division of Licensing

Enclosure:
Request for Additional Information

cc: See next page

CONCURRENCES:

DL:LB#1	RSB	AD:RS	SSPB	DL:LB#1	
LNOlshan:es	BSheron	RHouston	CThomas	BJYoungblood	[illegible]
7/19/84	7/ /84	7/ /84	7/19/84	7/19/84	7/ /84

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BYRON
Request for Additional Information by the
Reactor Systems Branch

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 PORV 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 PORV 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.

2. Table 3.3-5, Engineered Safety Features Response Time (page 3/4 3-30)

The high steam generator level trip delay of 2 seconds to close feedwater system valves and trip the turbine in table 15.0-5 is not consistent with the values in T.S. table 3.3-5.

3. Table 3.3-3, Engineered Safety Features Actuation System Instrumentation (page 3/4 3-14)

The Technical Specifications do not require automatic safety injection in the event of a main steam line break outside containment below P-11 (1930 psig RCS pressure). Justify that a postulated steam line break at the end of core life, when the moderator density coefficient is highly negative, would be within the calculated FSAR results for operation below P-11.

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.

5. 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 (LTOP) can be met within the above limits in view of the following considerations.

a. A 450 ft^3 bubble is required to provide a ten minute warning to the operator before the Appendix G limits are reached for low temperature in the reactor system. (SER page 5-4, Q212.153)

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)

6. Reactor Coolant System, Overpressure Protection Systems 3.4.9.3 (page 3/4 4-35)

The Technical Specifications provide for lockout of ECCS pumps below a RCS temperature of 380°F but do not provide for measures to prevent operation of reactor coolant pumps or accumulators at low reactor system temperatures. The staff's conclusion that Byron was adequately protected against low temperature overpressure events was based on the commitment that inadvertent RCP operation or accumulator injection at low RCS temperature would be prevented (SER page 5-4). Justify that the Appendix G limits will not be exceeded from inadvertent RCP or accumulator operation at low temperature.

7. Technical Specifications are not provided for surveillance of the RHR miniflow bypass valves which open at <500 gpm for RHR pump protection and close at >1000 gpm to provide for maximum ECCS flow. Justify that the consequences of LOCA will remain within those documented in the FSAR in the absence of operability surveillance for these valves.

8. Table 3.3-2, Reactor Trip System Instrumentation Response Times (page 3/4 3-7, 8)

Several of the response times listed in Table 3.3-5 are not verifiable by review of Chapter 15. Please provide references for those times not listed in Chapter 15. If specific actuated equipment is not taken credit for in any of the transient analyses, it is permissible to state that in lieu of a reference for the associated response time.

Specifically provide verification for the response times for those operations, other than reactor trip, for

a) containment pressure - high, high-2, high-3
b) pressurizer pressure - low
c) steam line pressure - low
d) RWST level - low 2, coincident with SI
e) undervoltage RCP Bus
f) division 1 ESF Bus undervoltage

g) loss of power

h) steam line pressure negative rate

i) phase "A" isolation

9. Table 3.4-1, Reactor Coolant System Pressure Isolation Valves (page 3.4-21)

The staff notes that the charging system check valves were recently removed from the list of valves for which leak surveillance will be performed. Justify that the low pressure portions of the charging system are adequately protected against full reactor system pressure in the event that all charging flow were lost and that a LOCA outside containment will not occur.

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.

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.

12. Table 3.3-1, Reactor Trip Instrumentation

Item 19, the minimum channels operable for interlock P-10 for Mode 1 conflicts with FSAR Section 7.2.1.1.2. That is, when coming down in power it takes a 3 out of 4 P-10 channels to reinstate the intermediate range high neutron flux trip and the low power range neutron flux trip. Item 19 shows 2 out of 4. Please resolve this inconsistency.

13. Plant Systems, Turbine Cycle Valves, B 3/4.7.1.1 (page B3/4 7-1)

What is the basis for the equation that derives the reduced reactor trip setpoints whenever there are inoperable safety valves? Is there an analysis to support this equation?