Commonwealth Edison
1400 Opus Place
Downers Grove, Illinois 60515

February 26, 1990

Mr. Thomas E. Murley, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Byron Station Unit 1
Cycle 4 Reload Core
NRC Docket No. 50-454

References:

1. Westinghouse WCAP-9272-P-A, dated October 1985; "Westinghouse Reload Safety Evaluation Methodology", (originally issued March 1978).

2. CECo submittal, F. G. Lentine to H. R. Denton dated July 27, 1983; titled "Zion Stations Units 1 and 2, Byron Station Units 1 and 2, Braidwood Station Units 1 and 2, Commonwealth Edison Company Topical, Report on Benchmark of PWR Nuclear Design Methods, NRC Docket Nos. 50-295/304, 50-454/455, and 50-456/457.

3. NRC SER on CECo's Neutronics Topical (Ref. 2) dated December 13, 1983.

4. Westinghouse WCAP-11596-P-A, dated June 1988; "Qualification of the *PHONEIX/ANC Nuclear Design System for Pressurized Water Reactor Cores"

5. CECo submittal, R.A. Chrzanowski to T.E. Murley, "Byron Station Units 1 and 2 Application for Amendment to Facility Operating Licenses NPF-37 and NPF-66," dated July 31, 1989.

Dear Mr. Murley:

Byron Unit 1 has completed its third cycle of operation and is conducting a refueling outage that began January 7, 1990. Byron Unit 1 Cycle 3 attained a final cycle burnup of approximately 16,200 MWD/MTU. Cycle 4 is expected to commence early in March 1990. This letter provides you with two reports pertaining the Byron Unit 1 Cycle 4 reload core. Attachment 1 describes the reload core and summarizes Edison's review performed in accordance with 10CFR50.59. Attachment 2 provides the Core Operating Limits Report for Cycle 4 pursuant to Technical Specification 6.9.1.9.

Add: NRR Chatterton 1 1

A001
41

Commonwealth Edison and its vendor (Westinghouse) apply NRC approved reload design methodology developed by Westinghouse as described in Reference 1. Commonwealth Edison requested approval to perform the neutronic portion of the reload design in Reference 2, and the NRC approved this request (Reference 3). However, Commonwealth Edison elected to not exercise the option of performing the Byron Unit 1 Cycle 4 neutronic design in-house because of the transition to the Westinghouse VANTAGE5 fuel. The Byron Unit 1 Cycle 4 neutronic design, including development of the core operating limits, was generated by Westinghouse using the NRC approved methodology of Reference 1 and the NRC approved neutronic code package as described in Reference 4.

Please address any questions regarding this submittal to this office.

Very truly yours,

Terence K. Schuster

T. K. Schuster
Nuclear Licensing Administrator
Byron Station

cc: P. C. Shemanski
A. B. Davis
B. Clayton
Resident Inspector-Byron

TKS:wj/0708T/2

ATTACHMENT 1

Byron 1 Cycle 4 Reload Description

The Byron Unit 1, Cycle 4 reload core was designed to perform under current nominal design parameters, Technical Specifications and related bases, and current Technical Specification setpoints such that:

1. Core characteristics will be less limiting than those previously reviewed and accepted; or

2. For those postulated incidents analyzed and reported in the Updated Byron Final Safety Analysis Report (UFSAR) which could potentially be affected by fuel reload, reanalyses or reevaluations have demonstrated that the results of the postulated events are within allowable limits. The reanalysis, described in Reference 5, was reviewed and approved by the NRC. Commonwealth Edison Company performed a detailed review with Westinghouse on the bases, including all the postulated incidents considered in the UFSAR, of the Reload Safety Evaluation (RSE).

The Byron Unit 1 Cycle 4 core will be a "Low Leakage" design. Commonwealth Edison has successfully developed and used similar "Low Leakage" designs at its Byron and Zion Units. During the Cycle 3/4 refueling, eighty-eight (88) VANTAGE5 fuel assemblies will be inserted into the core. The Byron Unit 1 core will then contain a combination of fresh Westinghouse VANTAGE5 fuel assemblies and Westinghouse's 17x17 Optimized Fuel Assemblies (OFA's), as described in the Reference 5 Amendment submittal. Reference 5 requested approval for the transition to VANTAGE5 fuel and associated proposed changes to the Byron Technical Specifications. The submittal fully justified the compatibility of Westinghouse OFA and VANTAGE5 assemblies in a reload core, and verifies compatibility with control rods and reactor internals interfaces. A mixture of Integral Fuel Burnable Absorber (IFBA) rods and Wet Annular Burnable Absorbers (WABAs) will be used as the burnable poison. WABAs have been used extensively by Commonwealth Edison. A description and evaluation of IFBA rods is presented in Reference 5.

The reload fuel assemblies will incorporate Westinghouse standardized fuel pellets, reconstitutable top nozzles (RTN), extended burnup design features, and snag resistant grids. Similar features have been successfully utilized in Commonwealth Edison's Byron and Braidwood Units. Additionally, the reload fuel assemblies will incorporate the Debris Filter Bottom Nozzle (DFBN). The DFBN, hydraulically and structurally equivalent to the nozzle used on the existing fuel assemblies, is expected to improve fuel performance by reducing the size of any debris that enters the active fuel region. This feature is currently in operation in Braidwood Unit 1 Cycle 2. The significant new mechanical features of the VANTAGE5 design are the Intermediate Flow Mixer (IFM) grids and the Axial Blankets. Structural evaluations of these fuel features provided in Reference 5 verify that the VANTAGE5 assembly design is structurally acceptable. In addition, Westinghouse's Enhanced Performance Rod Cluster Control Assemblies (EP-RCCAs), which contain a Silver-Indium-Cadmium (Ag-In-Cd) absorber instead of hafnium, will be utilized in Cycle 4.

The relo. iuel's nuclear design has been evaluated in Reference 5. As OFA and VANTAGE5 fuel have the same pellet and fuel rod diameters, most reactivity parameters are insensitive to fuel type. Changes in nuclear characteristics due to the transition from OFA to VANTAGE5 fuel are within the range normally seen from cycle to cycle due to fuel management effects. The loading pattern dependent parameters were evaluated in detail in the CECo/Westinghouse reload safety evaluation described below. In addition, based upon the performance of an eighteen case FAC analysis, a total peaking factor (Fq) of less than 2.50 is the maximum which could occur for the full range of power distributions, including load follow maneuvers, allowable under Constant Axial Offset Control (CAOC). Therefore, additional surveillance of Fq(z) is not required in Cycle 4. The Cycle 4 radial peaking factor (Fxy) limits are described in the attached "Core Operating Limits" report.

The thermal-hydraulic design for the Cycle 4 reload core has not significantly changed from that of the previously reviewed and accepted initial cycle design. Tests and analysis have confirmed that the VANTAGE5 assemblies are hydraulically compatible with the OFA assemblies reloaded as Regions 4 and 5. The approved VANTAGE5 amendment FNDH limits of less than 1.55 for OFA assemblies and 1.65 for VANTAGE5 assemblies ensures that the DNB ratio of the limiting power rod during Condition I and Condition II events is greater than or equal to the DNBR limit of the DNBR correlation being applied.

Commonwealth Edison's reload safety evaluation process is a verification that previously reviewed and approved accident analyses are not adversely impacted by the cycle specific reload core design. Commonwealth Edison's Byron Unit 1 Cycle 4 reload safety evaluation applied both the LOCA and non-LOCA safety analyses presented in Reference 5 and relied on previously reviewed and accepted analysis reported in UFSAR, fuel technology reports, and previous reload safety evaluation reports. A detailed review of the core characteristics was performed to determine those parameters affecting the postulated accident analyses reported in the Byron UFSAR, and in Reference 5. Commonwealth Edison verifies that accident analyses presented in the UFSAR, as modified by the analyses described in Reference 5, were not affected by the reload core characteristics. Commonwealth Edison has previously verified that the results of the Reference 5 reanalyses were within previously reviewed and accepted limits.

The reload safety evaluation demonstrated that no additional Technical Specification changes, beyond those previously submitted for NRC approval in Reference 5, are required for operation of Byron Unit 1 during Cycle 4. Commonwealth Edison has reviewed the cycle-specific licensing analysis results and concludes that no unreviewed safety questions exist for this reload, as defined by 10CFR50.59. Therefore, no additional prior NRC review and approval of the reload core analyses or application for amendment to the Byron Unit 1 operating license beyond that requested in Reference 5, is required as a result of the cycle-specific reload design for Cycle 4.

Finally, verification of the reload core design will be performed per the standard reload startup physics tests. These tests include, but are not limited to:

1. A physical inventory of the fuel in the reactor by serial number and location prior to the replacement of the reactor head;
2. Control rod drive tests and drop times;
3. Critical boron concentration measurements;
4. Control bank worth measurements using the rod swap technique;
5. Moderator temperature coefficient measurements;
6. Startup power distribution measurement using the incore flux mapping system.

TKS:wj/C708T/5

Attachment 2

Byron Unit 1 Cycle 4
Limit Report - Fxy Portion

This Radial Peaking Factor Limit is provided in accordance with Paragraph 6.9.1.9 of the Byron Unit 1 Nuclear Plant Technical Specifications.

The F_{xy} limits for RATED THERMAL POWER within the specified core planes for Cycle 4 shall be:

1. F_{xy}^{RTP} less than or equal to 1.98 for all core planes containing bank "D" control rods, and

2. F_{xy}^{RTP} less than or equal to 1.75 for all unrodded core planes.

These Fxy(z) limits were used to confirm that the heat flux hot channel factor FQ(z) will be limited to the Technical Specification values of:

$$F_Q(z) \leq [\frac{2.501}{P}] \ [K(z)] \text{ for } P > 0.5 \text{ and,}$$

$$F_Q(z) \leq [5.00] \ [K(z)] \text{ for } P \leq 0.5$$

assuming the most limiting axial power distributions expected to result from the insertion and removal of Control Banks C and D during operation, including the accompanying variations in the axial xenon and power distributions as described in the "Power Distribution Control and Load Following Procedures", WCAP-8403, September, 1974. Therefore, these F_{xy} limits provide assurance that the initial conditions assumed in the LOCA analysis are met, along with the ECCS acceptance criteria of 10 CFR 50.46.

See Figure 1 for a plot of $[F_Q^T . P_{Rel}]$ versus Axial Core Height.

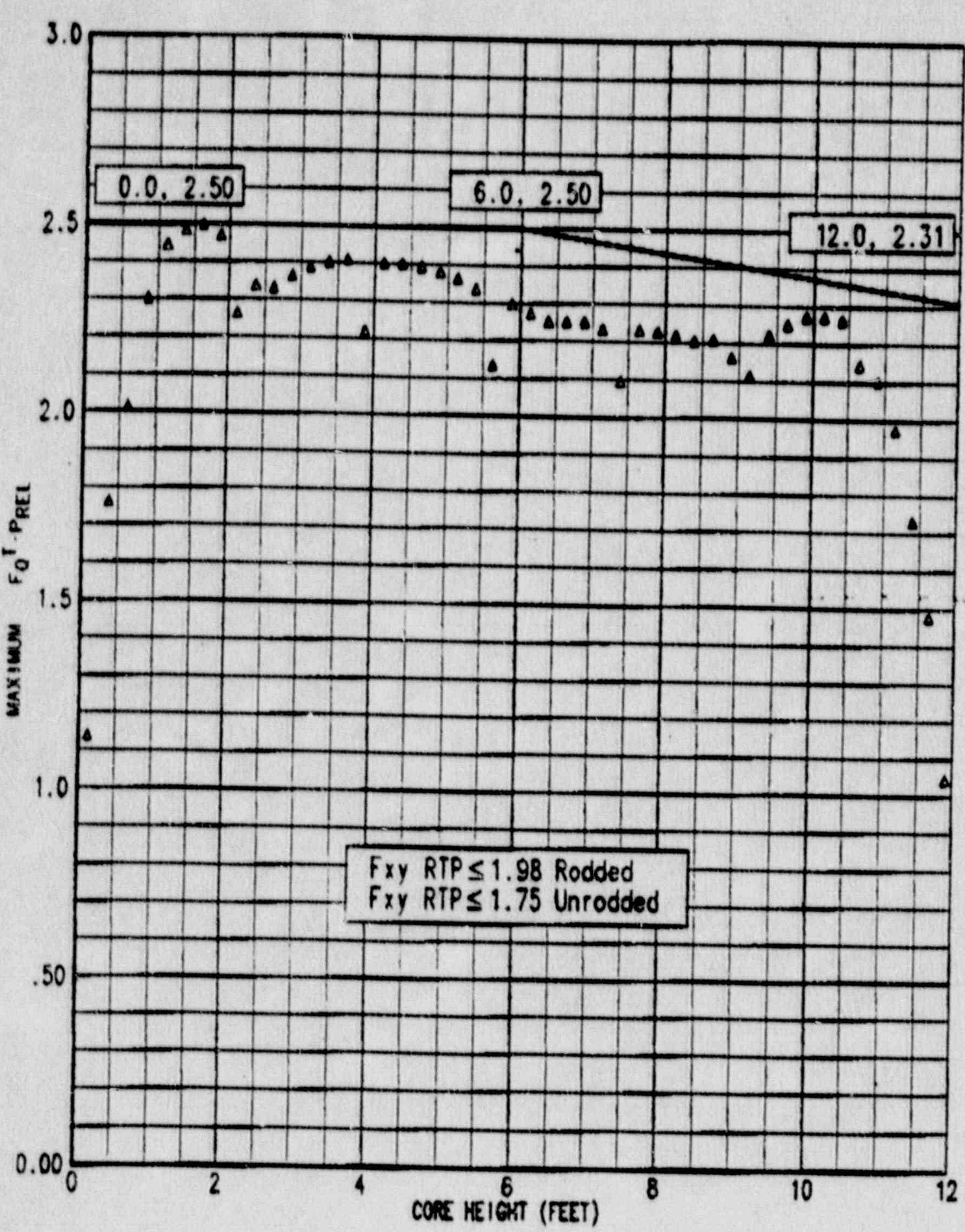

FIGURE 1

MAXIMUM FQT X POWER VERSUS AXIAL HEIGHT DURING NORMAL CORE OPERATION

BYRON UNIT 1 CYCLE 4