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November 19, 1990

Dr. Thomas E. Murley, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attn: USNRC Document Control Desk

Subject: Byron Station Unit 2
Cycle 3 Reload
NRC Docket No. 50-455

References: See Attachment 3

Dear Dr. Murley:

Byron Unit 2 is completing a refueling outage that began September 2, 1990 following its second cycle of operation. Byron Unit 2 Cycle 2 attained a final cycle burnup of approximately 17,614 MWD/MTU. Cycle 3 is expected to commence in late November, 1990. This letter is to summarize Commonwealth Edison Company's (CECo) plans and evaluations regarding the Byron Unit 2 Cycle 3 reload core, and to provide the Cycle 3 Core Operating Limits Report per Generic letter 88-16 and Technical Specification 6.9.1.9.

Attachment 1 describes the Byron 2 Cycle 3 reload and CECo's reload safety evaluation process. The Byron 2 Cycle 3 reload design and safety evaluation process included a core redesign effort which included the effects of limited fuel assembly reconstitution, the discharge of an assembly damaged during reconstitution and its seven symmetrical partners, and the removal of selected upper core plate guide pins. All aspects of the redesign have been reviewed against the safety analysis parameters of record and it has been determined that all safety parameters remain valid for Cycle 3. The Byron 2 Cycle 3 reload review was performed in accordance with the provisions of 10CFR50.59 and no unreviewed safety issues were identified.

Attachment 2 provides the Core Operating Limits Report for Cycle 3 pursuant to Technical Specification 6.9.1.9. CECo and its vendor (Westinghouse) apply NRC approved reload design methodologies described in Reference-1. Commonwealth Edison requested approval to perform the neutronic portion of the PWR reload designs using the methods described in Reference 2, and the NRC approved this request in Reference 3. Specifically, the Byron Unit 2 Cycle 3 reload design and core operating limits were generated by Commonwealth Edison using NRC approved methodologies. This included all nuclear design inputs to the safety parameter verification process and resulting Reload Safety Evaluation. Some operational data have been generated by Commonwealth Edison using approved Westinghouse codes for which Edison

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specific application is currently under NRC review (Reference 4). This operational data includes tables, curves, and computer code constants used in Estimated Critical Predictions and Technical Specification surveillances. As discussed in Reference 4, NRR has concurred with this limited application because: a) the codes have been NRC approved (References 5 and 6), b) they are not used in reload design, as safety analysis input, or to generate core limits for Byron 2 Cycle 3, and c) the codes will improve the Estimated Critical Predictions and similar operational calculations.

With respect to the removal of selected guide pins from the upper core plate, the modification has been performed per 10CFR50.59. Although information has previously been provided and discussed with the Senior Resident Inspector and NRR Project Manager, additional discussions and/or transmittals can be provided should additional information be needed by your Staff or the Region III Staff.

Please direct any questions regarding this notification to this office.

Very truly yours,

Allen R. Checca

A.R. Checca
Nuclear Licensing Administrator

cc: T. Boyce - Project Manager, NRR
A. B. Davis - Regional Administrator, RIII
NRC Resident Inspector - Byron

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ATTACHMENT 1

Byron 2 Cycle 3 Reload Description

The Byron Unit 2 Cycle 3 core is a "Low Leakage" design. Commonwealth Edison has successfully developed and used similar "Low Leakage" designs at its Braidwood, Byron and Zion units. During the Cycle 2/3 refueling, eighty-four (84) VANTAGE 5 fuel assemblies will be inserted into the core. The Byron Unit 2 core will then contain a combination of fresh Westinghouse 17x17 VANTAGE 5 assemblies and previously irradiated 17x17 Optimized Fuel Assemblies (OFA's), as described in the Reference 7, licensing submittal on the VANTAGE 5 fuel transition. The NRC approved the use of VANTAGE 5 at Byron Unit 2 for Cycle 3 and thereafter, under the provisions of 10CFR50.90 in Reference 8. Reference 7 fully justified the compatibility of Westinghouse OFA and VANTAGE 5 assemblies in a reload core, and verified 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. The IFBA rods contain fuel pellets with enriched B-10 coating. Both WABAs and IFBA fuel rods have been used extensively by Commonwealth Edison.

The reload VANTAGE 5 fuel assemblies will incorporate Westinghouse standardized fuel pellets, reconstitutable top nozzles (RTN), extended burnup design features, and snag resistant Intermediate Flow Mixers (IFM) and grids. Similar features have been successfully utilized previously in Commonwealth Edison's Byron and Braidwood Units. The reload fuel assemblies will also 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 may potentially enter the active fuel region. This feature is currently in operation at Byron Unit 1 and Braidwood Unit 2.

The Byron Unit-2 Cycle-3 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 operating characteristics will be equivalent or 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 been performed to demonstrate that the results of the postulated events are within allowable limits.

The reload design reflects the reconstitution of three fuel assemblies (T21J, T70K, and T74K) the discharge of a damaged fuel assembly (T77K), and the Upper Internals modification which removed fuel assembly guide pins at selected core locations. Fuel assembly T77K, and its seven symmetric assemblies, were discharged when assembly T77K was damaged during Byron 2 reconstitution. Since the original design and Reload Safety Evaluation did not reflect these changes, a comprehensive redesign and reexamination of safety limits has been performed. Both the redesign and the resulting detailed reverification of safety parameters (conducted jointly between Westinghouse

and Commonwealth Edison) considered the possible effects on local core power distribution that may result from the removal of fuel guide pins: B-7S, B-8S, B-9S, D-7N, D-9N, D-10N, E-7N, and E-9N. Please note that although the redesign efforts assumed that 8 guide pins would be deleted, only 6 pins were physically removed (guide pins at B-7S and E-7N were successfully re-straightened). This had no impact on the analysis.

The reload fuel's nuclear design has been evaluated generically in Reference-7. As OFA and VANTAGE 5 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 VANTAGE 5 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 process per the methodolgy described in References 1 and 2.

CECo has also analyzed the potential increased peaking resulting from theoretical displacements of fuel assemblies that may result from the deletion of selected upper internal fuel guide pins. Specifically, using the maximum possible relative displacements as determined by the Westinghouse Commercial Fuel Division, CECo performed unit assembly calculations using the 2D (TORTISE) basic design code. 2D is the current NRC approved (Reference 3) Edison nuclear design methodology. The shift in assembly power due to the assumed increased water gap, and resulting impacts on FNDH and other Safety Parameter Interaction List (SPIL) parameters, were calculated for each affected assembly. CECo has determined that all neutronic reload parameters remain within the previously established cycle safety and transient SPIL limits. This conclusion considers the combined impacts of the redesigned loading pattern, assembly reconstitution, and guide pin deletion, and includes but is not limited to SPIL items for non-LOCA and LOCA considerations.

The radial nuclear heat flux hot channel limits, Technical Specification 3/4.2.2 (Heat Flux Hot Channel Factor), as presented in the Core Operating Limits Report (COLR), have been generated to conservatively consider the increased local peaking that is theoretically possible in assemblies at locations where fuel guide pins have been removed. Commonwealth Edison will consider the impact of the Upper Internals modification on Technical Specification 3/4.2.3 (Nuclear Enthalpy Rise Hot Channel Factor) by applying additional conservatisms to the measured FNDH during the associated surveillance.

The thermal-hydraulic design for the Cycle 3 reload core has not significantly changed from the reviewed and accepted previous cycle design. Tests and analyses have confirmed that the VANTAGE 5 assemblies are hydraulically compatible with the OFA assemblies reloaded as Regions 2, 3 and 4.

The FNDH limits of less than 1.55 for OFA assemblies and less than 1.65 for VANTAGE 5 assemblies ensure that the DNB ratio, of the limiting 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 (SPIL/RSE review) is a verification to ensure that the previously reviewed and approved accident analyses are not adversely impacted by the cycle specific reload core design. Commonwealth Edison's Byron 2 Cycle 3 Reload Safety Evaluation applied both the LOCA and non-LOCA safety analyses presented in Reference 7, and relied on previously reviewed and accepted analyses reported in the 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 7. Commonwealth Edison verified that accident analyses presented in the UFSAR, as modified by the analyses described in Reference 7, were not affected by the reload core characteristics. The review considered the impacts of the redesign, including reconstitution and the fuel guide pin modification. No impacts were identified; therefore no further analysis or additional changes to the Technical Specifications besides those previously approved in Reference 2 are required to ensure safe operation during Cycle 3. Westinghouse did identify some minor impact to the LOCA analysis resulting from potential changes in the fuel assembly to assembly configuration due to the Upper Internals Guide Pin modification. The impact of the altered fuel assembly configurations on the final Peak Clad Temperature (PCT); however, was negligible and all of the 10CFR50.46 criteria continue to be satisfied. Westinghouse also evaluated the removed guide pin's mechanical and structural considerations (including seismic and LOCA forces). CECo reviewed Westinghouse's conclusions as part of the modification process and found the conclusions to be acceptable.

Finally, verification of the Byron Unit 2 Cycle 3 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; and

6. Startup power distribution measurements using the incore flux mapping system.

In summary:

1. There is no increase in the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report;

2. No additional accident or malfunction of a different type than any evaluated previously in the safety analysis reported has been created; and

3. There has been no reduction in the margin of safety as defined in the basis for any Byron Unit 2 Technical Specification.

Therefore, as there are no unreviewed safety questions associated with this reload, no additional prior NRC review and approval of the reload core analyses and/or amendments to the Unit 2 operating license are required as a result of the cycle specific reload design for Cycle 3.

ATTACHMENT 2

Byron Unit 2 Cycle 3 Redesign
Operating Limit Report - Fxy Portion

This Radial Peaking Factor Limit Report is provided in accordance with Paragraph 6.9.1.9 of the Byron Unit 2 Nuclear Plant Technical Specifications.

The Fxy limits for RATED THERMAL POWER within specified core planes for Cycle 3 shall be:

a. For the lower core region from greater than or equal to 0% to less than or equal to 50%:

   1. F_{xy}^{RTP} less than or equal to 2.006 for all core planes containing bank "D" control rods.

   2. F_{xy}^{RTP} less than or equal to 1.738 for all unrodded core planes.

b. For the upper core region from greater than 50% to less than or equal to 100%:

   1. F_{xy}^{RTP} less than or equal to 1.848 for all core planes containing bank "D" control rods, and

   2. F_{xy}^{RTP} less than or equal to 1.748 for all unrodded core planes

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

$$F_Q(z) \leq \left[\frac{2.50}{P}\right] [K(z)] \quad \text{for } P > 0.5 \text{ and,}$$

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

assuming the most limiting axial power distributions expected to result from 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 $[F_Q^T \cdot P_{Rel}]$ vs. Axial Core Height.

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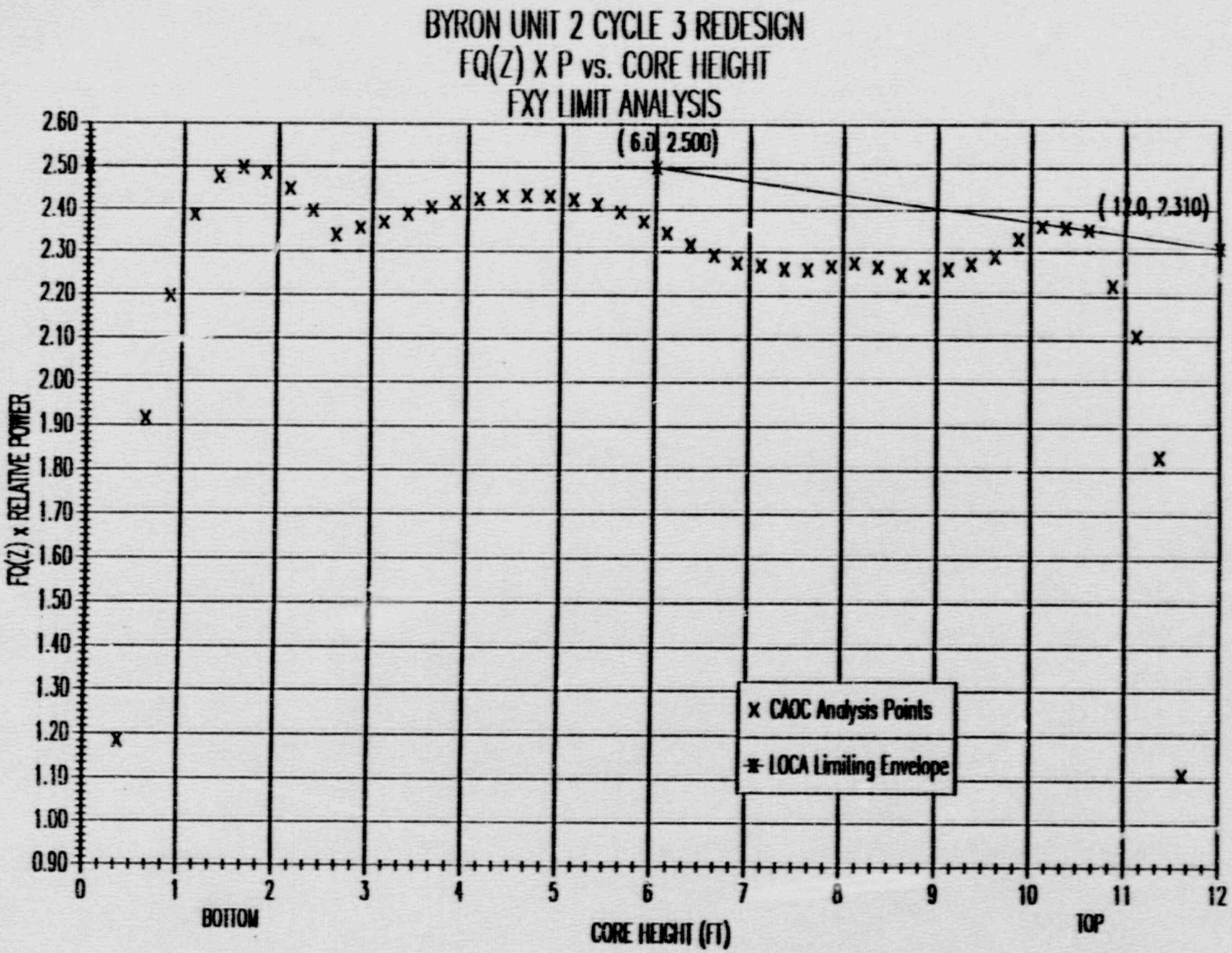
BYRON UNIT 2 CYCLE 3 REDESIGN
FQ(Z) X P vs. CORE HEIGHT
FXY LIMIT ANALYSIS
(6.0, 2.500)
(12.0, 2.310)
FQ(Z) x RELATIVE POWER
2.60
2.50
2.40
2.30
2.20
2.10
2.00
1.90
1.80
1.70
1.60
1.50
1.40
1.30
1.20
1.10
1.00
0.90
0
1
2
3
4
5
6
7
8
9
10
11
12
BOTTOM
CORE HEIGHT (FT)
TOP
x CAOC Analysis Points
LOCA Limiting Envelope

FIGURE 1

ATTACHMENT 3

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. CECo/NFS topical report NFSR-0081, "Benchmark of PWR Nuclear Design Methods Using the PHOENIX-P and ANC Codes," (dated July 1990) submitted for NRC review by letter from J. Silady to T. E. Murley dated July 13,

5. Westinghouse topical report WCAP-11596-P-A, "Qualification of the PHOENIX/ANC Nuclear Design System for Pressurized Water Reactor Cores," dated June 1988.

6. Westinghouse topical report WCAP-10965-P-A, "ANC: Westinghouse Advanced Nodal Computer Code," dated September 1986.

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

8. NRC Letter from L. N. Olshan to T. E. Kovach, "Amendment No. 36, Use of VANTAGE 5 Fuel," dated January 31, 1990.

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