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April 08, 1993

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

Attn: Document Control Desk

SUBJECT: Byron Station Unit 1
Cycle 6 Reload
NRC Docket No. 50-454

REFERENCES: See Attachment 3

Dear Dr. Murley:

Byron Unit 1 has completed its fifth cycle of operation and is conducting a refueling outage that began February 5, 1993. Byron Unit 1 Cycle 5 attained a final cycle burnup of approximately 17,143 MWD/MTU. Cycle 6 is expected to commence on April 15, 1993. This letter is to summarize Commonwealth Edison Company's (CECo) safety evaluation regarding the Byron Unit 1 Cycle 6 reload core.

Attachment 1 describes the core reload redesign including a summary of CECo's safety evaluation, performed in accordance with the provisions of 10CFR50.59. The evaluation identified no unreviewed safety issues. This reload did, however, require one preparatory Technical Specification change (Reference 8) which has been approved by the NRC (Reference 9).

Attachment 2 provides the Operating Limits Report for Cycle 6 pursuant to Technical Specification 6.9.1.9. CECo applies NRC approved reload design methodologies developed by Westinghouse as described in Reference 1. Commonwealth Edison performed the neutronic portion of the reload design using the methods and codes described in References 2 and 4 which were approved in References 3 and 5, respectively. In summary, the Byron Unit 1 Cycle 6 reload design, including the development of the core operating limits, was generated by Commonwealth Edison using NRC approved methodologies.

As expected from Cycle 5 coolant activity trending, fuel inspections during the current outage identified the need to replace one fuel rod in a once-burned assembly with a stainless steel filler rod. The impact of reconstitution on the reload design has also been evaluated with standard Westinghouse methods. No unreviewed safety questions or Technical Specifications changes were identified, the results of this reconstitution evaluation were previously provided to the NRC under separate cover (Reference 10) for informational purposes.

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During the reinstallation of the reactor vessel upper internals package, three fuel assemblies, G22E, G17E and H17E, were damaged as were several fuel assembly guide pins. A redesign of the Byron Unit 1 Cycle 6 core scheduled these assemblies, and six symmetric fuel assemblies, for discharge. A revised core loading pattern was approved using NRC approved reload design methodolog.. In addition, two upper core plate fuel assembly guide pins ("S"-pins) were removed from locations P-9S and R-9S. The effects of these changes on the reload safety evaluation have been analyzed/evaluated by CECo and Westinghouse. These changes do not affect the conclusions of the original reload safety evaluation as summarized in Attachment 1.

Please direct any questions regarding this notification to this office.

Very truly yours,

Joseph A. Bauer

Joseph A. Bauer
Nuclear Licensing Administrator

cc: J. B. Hickman - Byron Project Manager, NRR
A. B. Davis - Regional Administrator, Region III
H. Peterson - Senior Resident Inspector - Byron

ATTACHMENT 1

Byron Unit 1 Cycle 6 Reload Description

The Byron Unit 1 Cycle 6 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 Braidwood/Byron Updated Final Safety Analysis Report (UFSAR) which could potentially be affected by fuel reload, re-analyses or re-evaluations have demonstrated that the results of the postulated events are within allowable limits.

The Byron Unit 1 Cycle 6 core is a "Low Leakage" design. Previously, Commonwealth Edison has successfully developed and operated similar "Low Leakage" designs at Byron as well as at our Braidwood and Zion stations.

During the Cycle 5/6 refueling outage, eighty-eight (88) Westinghouse VANTAGE5 fuel assemblies have been inserted into the core. The Byron Unit 1 core contains a full complement of fresh and previously irradiated 17x17 VANTAGE5 fuel assemblies. The NRC approved the use of VANTAGE5 for Byron Unit 1 for Cycle 3 operations and thereafter, under the provisions of 10CFR50.90 (Reference 6). The Braidwood/Byron UFSAR describes the Westinghouse VANTAGE5 fuel assemblies in a reload core, and documents the compatibility of control rods and reactor internal 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 an enriched B-10 coating. Both WABAs and IFBA burnable absorbers have previously been used by Commonwealth Edison.

The reload VANTAGE5 fuel assemblies will incorporate Westinghouse's standardized fuel pellets, reconstitutable top nozzles (RTN), extended burnup design features, modified Debris Filter Bottom Nozzle (DFBN), and snag resistant Intermediate Flow Mixer (IFM) grids. Similar features have been previously utilized successfully in Commonwealth Edison's other PWRs as well as at other domestic and overseas units.

The Byron Unit 1 Cycle 6 core has been designed and evaluated using NRC licensed and approved methods. Commonwealth Edison requested approval to perform the neutronic portion of PWR reload design using the methods described in Reference 2. The NRC has approved this request as noted in Reference 3. Specifically, the Byron Unit 1 Cycle 6 reload design, including the development of the core operating limits, were generated and verified by Commonwealth Edison using NRC approved methodologies.

The reload fuel's nuclear design is evaluated generically in the UFSAR. The VANTAGE5 fuel uses the same pellet and fuel rod diameters as previously approved in past cycles. Although this is the first full core of VANTAGE5 fuel, the nuclear characteristics of VANTAGE5 fuel are within the range normally seen from cycle to cycle as most reactivity parameters are insensitive to fuel type. The loading pattern dependent parameters were evaluated in detail in the CECo/Westinghouse reload safety evaluation process.

CECo has also analyzed the potential increased peaking which could result from theoretical displacements of fuel assemblies which are at or adjacent to the two fuel assembly locations which contain the removed upper internal fuel assembly guide pins. Specifically, using the worst case possible relative fuel assembly displacements, CECo performed unit assembly calculations using the NRC approved 2D basic design code (similar to Westinghouse's TORTISE code). The shift in assembly power due to the assumed increased water gap, and resulting impacts on FNDH and other key safety parameters, were calculated for each affected assembly. CECo has determined that all neutronic reload parameters remain within the previously established Safety Parameter Interaction List (SPIL) limits. This conclusion considers the combined impacts of the redesigned loading pattern, fuel assembly reconstitution, and guide pin removal.

The radial peaking factor (Fxy) limits, addressed by Technical Specification 4.2.2.2, are presented in the Operating Limits Report. These limits 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. CECo 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 conservatism to the measured FNDH during surveillance.

Since the Byron Unit 1 Cycle 6 core will contain a full complement of VANTAGE5 assemblies, the LOCA PCT transition penalty of 50°F will be removed and credited. Additionally, because of the removal of the two fuel assembly guide pins, P-9S and R-9S, from the upper internals package, a 5°F PCT penalty will be assigned and accounted for in the Byron Station LOCA PCT summary sheet. Any reporting requirements, as stated in 10CFR50.46, will be adhered to.

CECo's reload safety evaluation process (RSE/SPIL 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. CECo's Byron Unit 1 Cycle 6 Reload Safety Evaluation relied on previously reviewed and accepted analyses reported in the UFSAR, fuel technology reports, the VANTAGE5 Reload Transition Safety Report (RTSR), 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. CECo has determined that all neutronic reload parameters remain within the previously established and recently revised reload safety and transient SPIL limits.

A potential issue regarding the Boron Dilution Mitigation System (BDMS) was identified which presented the following concerns:

1. The Inverse Count Rate Ratio (ICRR) data, received from another utility, indicated that previous data used in the design of the BDMS was not bounding; and

2. The BDMS methodology was no longer conservative with respect to the flux doubling (2Φ) setpoint, which included no instrument uncertainties, when defining an equivalent "trip setpoint" as presented in the Technical Specifications.

For Byron Unit 1 Cycle 6, the original SPIL limit for Boron Dilution at Cold Shutdown was not met. The SPIL limit was re-analyzed using an increased Shutdown Margin (SDM), from 1.0% $\Delta k/k$ to 1.3% $\Delta k/k$, requiring a Technical Specification amendment (Reference 9). The submitted BDMS Technical Specification change was approved and incorporated for Byron Unit 1 Cycle 6 and subsequent cycles.

The thermal-hydraulic design for the Cycle 6 reload core has not significantly changed from that of the previously reviewed and accepted cycle design. The FNDH limit of less than 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.

The Operation of Byron Unit 1 Cycle 6 has been analyzed in accordance with NRC approved methodologies and satisfies all safety analysis limits (having taken into consideration the revised input assumptions for the Boron Dilution analysis). The margin of safety, as defined in the bases of the Technical Specifications, is not impacted or reduced.

Finally, verification of the Byron Unit 1 Cycle 6 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, CECo's use of VANTAGE5 fuel and use of advanced neutronics methods (as described in References 7 and 2, respectively) have been previously approved by the NRC (References 6 and 3 respectively). The effects of the single fuel assembly reconstitution, the redesigned loading pattern due to the replacement of three fuel assemblies, and the removal of two fuel assembly guide pins from the upper internals package have been addressed and CECo has concluded that there is no impact on the 10CFR50.59 reload safety evaluation. Specifically, the reload and associated changes do not involve any unreviewed safety questions or additional Technical Specification changes.

ATTACHMENT 2

Byron Unit 1 Cycle 6
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 1 Nuclear Plant Technical Specifications.

The Fxy limits for RATED THERMAL POWER within specified core planes for Cycle 6 shall be:

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

1. For all core planes containing bank "D" control rods:

$$F_{xy}^{RTP} \leq 2.210$$

2. For all unrodded core planes:

$$F_{xy}^{RTP} \leq 1.770$$

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

1. For all core planes containing bank "D" control rods:

$$F_{xy}^{RTP} \leq 2.210$$

2. For all unrodded core planes:

$$F_{xy}^{RTP} \leq 1.830$$

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:

$F_Q(z) \leq [2.50/P][K(z)]$ for $P > 0.5$ and

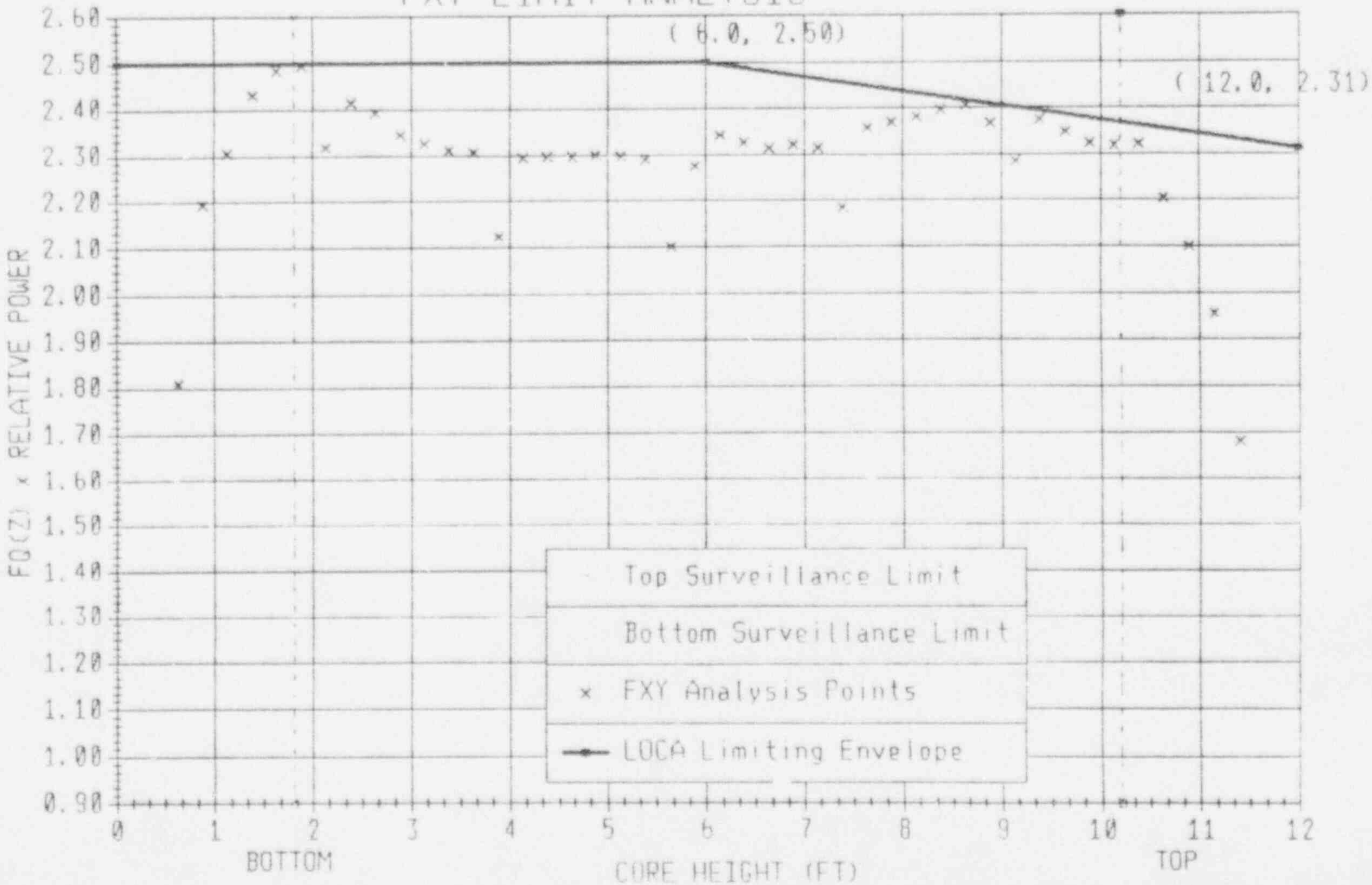
$F_Q(z) \leq [5.00][K(z)]$ for $P > 0.5$

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 Fxy 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 the Attached Figure for the plot of $[F_Q^T * P_{Rel}]$ vs. Axial Core Height.

BYRON UNIT 1 CYCLE 6
FQ(Z) X P vs. CORE HEIGHT
FXY LIMIT ANALYSIS

REV. 1



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ATTACHMENT 3

References

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

2) CECo submittal, J.A. Silaoy to T.E. Murley dated July 13, 1990; entitled "Commonwealth Edison Company Topical Report on Benchmark of PWR Nuclear Design Methods Using The Phoenix-P and ANC Computer Codes", NRC Docket Nos. 50-295/304, 50-454/455, and 50-456/457.

3) NRC SER on CECo's Neutronics Topical (Ref. 2) dated March 11, 1991.

4) CECo submittal, F.G. Lentine to H.R. Denton dated July 27, 1983; entitled "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.

5) NRC SER on CECo's Neutronics Topical (Ref. 5) dated December 13, 1983.

6) NRC Letter from L. N. Olshan (NRC) to T.J. Kovach (CECo), Amendment No. 36, "Use of VANTAGE5 Fuel", dated January 31, 1990.

7) CECo submittal, S.C. Hunsader to T.E. Murley "Braidwood Stations Unit 1 and 2 Application to Facility Operating License NPF-72 and NPF-77", dated October 14, 1989.

8) Letter from T. Simpkin (CECo) to T. E. Murley (NRC), "Boron Dilution Protection System License Amendment Request," dated July 28, 1992.

9) Letter from J. B. Hickman (NRC) to T. J. Kovach (CECo), "Issuance of Amendments, Braidwood Technical Specification Amendment 40 and Byron Technical Specification Amendment 51, to reflect changes to the current Boron Dilution Analyses", dated October 5, 1992.

10) Letter from J. A. Bauer (CECo) to T. E. Murley (NRC), "Byron Station Unit 1 Cycle 6 Fuel Assembly Reconstitution", dated March 16, 1993.