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Downers Grove, Illinois 60515

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Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

ATTN: Document Control Desk

SUBJECT: Byron Nuclear Power Station Unit 1
Cycle 7 Operating Limits Report
NRC Docket No. STN 50-454

REFERENCES: See Attachment 3

Commonwealth Edison Company (ComEd), Byron Nuclear Power Station (Byron), Unit 1 completed its sixth cycle of operation on September 8, 1994. Byron Unit 1 Cycle 6 attained a final cycle burnup of approximately 19,424 megawatt-days per metric ton of uranium (MWD/MTU). Byron Unit 1 is expected to return to service for Cycle 7 on November 2, 1994. This letter summarizes ComEd's safety evaluation for the Byron Unit 1 Cycle 7 reload core.

Attachment 1 describes the core reload design including a summary of ComEd's safety evaluation, performed in accordance with the provisions of Title 10, Code of Federal Regulations, Part 50, Section 59 (10 CFR 50.59), as there are no unreviewed safety issues or additional Technical Specification changes.

Attachment 2 provides the Operating Limits Report for Cycle 7 pursuant to Technical Specification 6.9.1.9. ComEd and our vendor (Westinghouse) apply Nuclear Regulatory Commission (NRC) approved reload design methodologies developed by Westinghouse as described in Reference 1. Westinghouse performed the neutronic portion of the reload design using the methods and codes described in References 1 and 2. Specifically, the Byron Unit 1 Cycle 7 reload design, including the development of the core operating limits, was generated by Westinghouse using NRC approved methodologies.

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Please direct any questions or comments regarding this submittal to this office.

Very truly yours,

Harold D. Pontious, Jr.
Nuclear Licensing Administrator

Attachment 1 - Byron Unit 1 Cycle 7 Reload Description
Attachment 2 - Byron Unit 1 Cycle 7 Operating Limits Report - Fxy Portion
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cc: G. Dick, Byron Project Manager - NRR
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H. Peterson, Senior Resident Inspector - Byron
Office of Nuclear Facility Safety - IDNS

ATTACHMENT 1

Byron Unit 1 Cycle 7 Reload Description

The Byron Unit 1 Cycle 7 reload core was designed to perform under current nominal design parameters, Technical Specifications and related bases, and current Technical Specification set points 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 Braidwood/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 Byron Unit 1 Cycle 7 reload design utilizes "Low Leakage" fuel management techniques. ComEd has successfully developed and operated similar "Low Leakage" reload designs at Byron as well as at our Braidwood and Zion Stations.

During the Cycle 6 refuel outage, ninety-two (92) VANTAGE 5 fuel assemblies have been replaced with ninety-two (92) new VANTAGE 5 fuel assemblies. One of the 92 fuel assemblies (J01E) was originally to be used for Cycle 6 but was damaged by interaction with the upper core plate pins. It was subsequently remanufactured for use in Cycle 7. The other 91 fuel assemblies were originally manufactured with alternate mid-grids rotated 90 degrees. To eliminate an adverse Departure from Nucleate Boiling (DNB) affect due to the mid-grid rotation, the 91 fuel assemblies were returned to the manufacturing plant and were remanufactured without the mid-grid rotation. J01E was not returned for grid reorientation since it was never manufactured with the rotation of alternate mid-grids.

The Byron Unit 1 core contains a full core of Westinghouse 17x17 VANTAGE 5 assemblies (92 new, 88 once-burned, and 13 twice-burned assemblies. The NRC approved the use of VANTAGE 5 fuel at Byron Station under the provisions of 10 CFR 50.90 in Reference 4. The Braidwood/Byron UFSAR describes the compatibility of Westinghouse VANTAGE 5 fuel 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 an enriched B-10 coating. Both WABAs and IFBAs have successfully been used in the past in ComEd reload designs.

The reload VANTAGE 5 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 successfully utilized previously at Byron and Braidwood. In addition, mechanical features which have not been previously used at Byron Unit 1 have been incorporated as part of the VANTAGE 5 design for the Region 9 fuel assemblies. These changes include: (1) an extended burnup bottom grid, (2) a keyless/cuspless top nozzle and modified hold down spring design to eliminate to potential for hold down spring hang-up, (3) standardized alignment of the fuel rods (a manufacturing simplification), (4) a cast composite bottom nozzle (a manufacturing process change), (5) WABA absorber axial repositioning to align fuel and absorbers, (6) a protective zirconium dioxide coating over the bottom six inches of the fuel rod to increase the level of debris defense, (7) a protective grid and modified fuel rod end plugs, and (8) a variable pitch fuel rod plenum spring to provide additional space for fission gas release.

Following the shutdown of Byron Unit 1 Cycle 6 elevated Antimony levels were detected in chemistry samples of the reactor coolant system coolant. The increase in Antimony levels was attributed to a leaking secondary source rod or rodlets. Therefore, two new unirradiated secondary sources containing 6 source rodlets each were installed in place of the old secondary sources which contained 4 source rodlets each.

The safety evaluation also addresses the removal of two fuel assembly guide pins seated on the underside of the upper core plate. During the Cycle 5 refuel outage, these pins were damaged and subsequently removed. It was concluded for subsequent operation that the removal of the fuel assembly guide pins did not represent an unreviewed safety question as defined in 10 CFR 50.59.

To correct for excessive fuel assembly gaps which were identified during the core alignment verification, two fuel assemblies G29E and H49F in core locations N02 and M02 respectively were rotated 180 degrees from the orientation originally specified in the core loading pattern. Evaluations of the revised core configuration concluded that this rotation did not result in an unreviewed safety question as described in 10 CFR 50.59.

For Byron Unit 1 Cycle 7 operation, the effect of increased steam generator tube plugging (SGTP) to either the 15% SGTP level or the Thermal Design Flow (TDF) limit on the fuel design was incorporated. The licensing basis amendment for the increased SGTP and the decreased TDF limit included the licensing of a Positive Moderator Temperature Coefficient (PMTC) Technical Specification, increased boron concentrations in the Refueling Water Storage Tank (RWST) and accumulators, and use of Revised Thermal Design Procedure (RTDP) methodology. The NRC approved all but the positive moderator temperature coefficient (MTC) portion of the amendment in Reference 5. The positive MTC portion of the amendment request remains under NRC review. This reload safety evaluation addresses a positive MTC, however, the Cycle 7 reload was designed to meet a negative MTC throughout the cycle. All the neutronic inputs to the safety analysis remain valid for Byron Unit 1 Cycle 7 without the implementation of a positive MTC.

The Byron Unit 1 Cycle 7 core has been designed and evaluated using NRC licensed and approved methods. Specifically, the Byron Unit 1 Cycle 7 reload design, including the development of the core operating limits, were generated by Westinghouse using NRC approved methodology as described in References 1 and 2. The reload fuel's nuclear design is evaluated generically in the UFSAR. The loading pattern dependent parameters were evaluated in detail in the ComEd/Westinghouse Reload Safety Evaluation/Safety Evaluation Parameter Interaction List (RSE/SPIL) process.

A 100°F Peak Clad Temperature (PCT) penalty will remain in place for Cycle 7 to address the possibility that the chopped cosine is not the most limiting power shape for the large break Loss-of-Coolant-Accident (LOCA) analysis. This PCT penalty will be removed upon the approval of WCAP-12909-P, "Power Shape Sensitivity Methodology."

An investigation into the effects of low initial Rod Internal Pressure (RIP) for IFBA rods on the large and small break LOCA analyses has been completed. It has been demonstrated that RIP uncertainty is not a significant PCT effect for initial cold pressures from 200 to 275 psig on the large break LOCA. Further, special IFBA analyses for small break LOCA, regardless of initial backfill pressure, are not required. Therefore, operation of Byron Unit 1 Cycle 7 with 200 psig backfill IFBA fuel meets the requirements of 10 CFR 50.46.

The effect of Economic Generation Control (EGC), including an increase in the temperature dead-band uncertainty on the Byron Unit 1 Chapter 15 UFSAR transients, including those evaluated for reduced TDF, increased SGTP, PMTC, and the use of RTDP as appropriate, has been evaluated. The results of the evaluation indicated that EGC operation is acceptable in that all applicable safety criteria have been met. Specific evaluations for large break and small break LOCA resulted in peak clad temperature increases of +4°F and +5°F for the large break and small break LOCA, respectively. All 10 CFR 50.46 acceptance criteria continue to be met. It should be noted that if the dead-band is increased from 1.5°F to 4°F, the peak LOCA containment pressure would be affected. Westinghouse has determined that a 0.2 psi "penalty" would be added to the previous peak LOCA containment pressure of 44.4 psig. A net peak containment pressure of 44.6 psig would then account for EGC operation with an increased rod control dead-band. Note that there is no peak LOCA containment pressure penalty if EGC is used without an increase in the rod control dead-band.

The use of RTDP methodology allows the station to gain margin to trip under normal operating conditions for the Overtemperature Delta T trip setpoint constants. The recalculations also required revision of the channel's maximum trip setpoint deviation from the computed trip setpoint for both Overtemperature Delta T and Overpower Delta T. Non-LOCA transients were analyzed using the RTDP uncertainties and verified that adequate margin exists to the new Departure from Nucleate Boiling Ratio (DNBR) design limits.

ComEd has determined that all neutronic reload parameters remain within the previously established RSE/SPIL limits. These include, but are not limited to, Safety Parameters for UFSAR non-LOCA and LOCA transients.

The thermal-hydraulic design for the Cycle 7 reload core has not significantly changed from that of the previously reviewed and accepted cycle design. The Nuclear Enthalpy Rise Hot Channel Factor (FNDH) limits of less than 1.65 for VANTAGE 5 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 (WRB-2) being applied. The W-3 DNBR Correlation continues to be used for conditions which are outside the WRB-2 Correlation (e.g., steam line break).

Summary

ComEd's 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. ComEd's Byron Unit 1 Cycle 7 Reload Safety Evaluation relied on previously reviewed and accepted analyses reported in the UFSAR, fuel technology reports, the VANTAGE 5 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. The Operation of the Byron Unit 1 Cycle 7 has been analyzed in accordance with NRC approved methodologies and satisfies safety analysis limits. 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 7 reload core design will be performed per the standard reload startup physics tests (ANSI/ANS 19.6.1). 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 measurements using the incore flux mapping system.

In summary, ComEd's use of VANTAGE 5 fuel and Westinghouse's use of advanced neutronics methods have been previously approved by the NRC (References 4 and 2). Therefore, no additional NRC review and approval of the reload core analyses or application for amendment to the Byron Unit 1 operating license is required as a result of the specific reload design for Cycle 7.

ATTACHMENT 2

BYRON UNIT 1 CYCLE 7

OPERATING LIMITS REPORT - Fxy PORTION

This Radial Peaking Factor Limit Report is provided in accordance with Specification 6.9.1.9 of the Byron Station Technical Specifications.

The Fxy limits for RATED THERMAL POWER within specified core planes for Cycle 7 shall be (values include missing support pin penalty):

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:

$$Fxy^{RTP} \leq 1.950$$

2) For all unrodded core planes:

$$Fxy^{RTP} \leq 1.695$$

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:

$$Fxy^{RTP} \leq 1.890$$

2) For all unrodded core planes:

$$Fxy^{RTP} \leq 1.711$$

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

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

$$Fq(z) \leq \left[5.00 \right] [K(z)] \text{ for } P \leq 0.5 \text{ and}$$

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 and the Emergency Core Cooling Systems (ECCS) acceptance criteria of 10 CFR 50.46 are met.

See the attached figure for the plot of $[F_q^T * P_{Rel}]$ vs. Axial Core Height.

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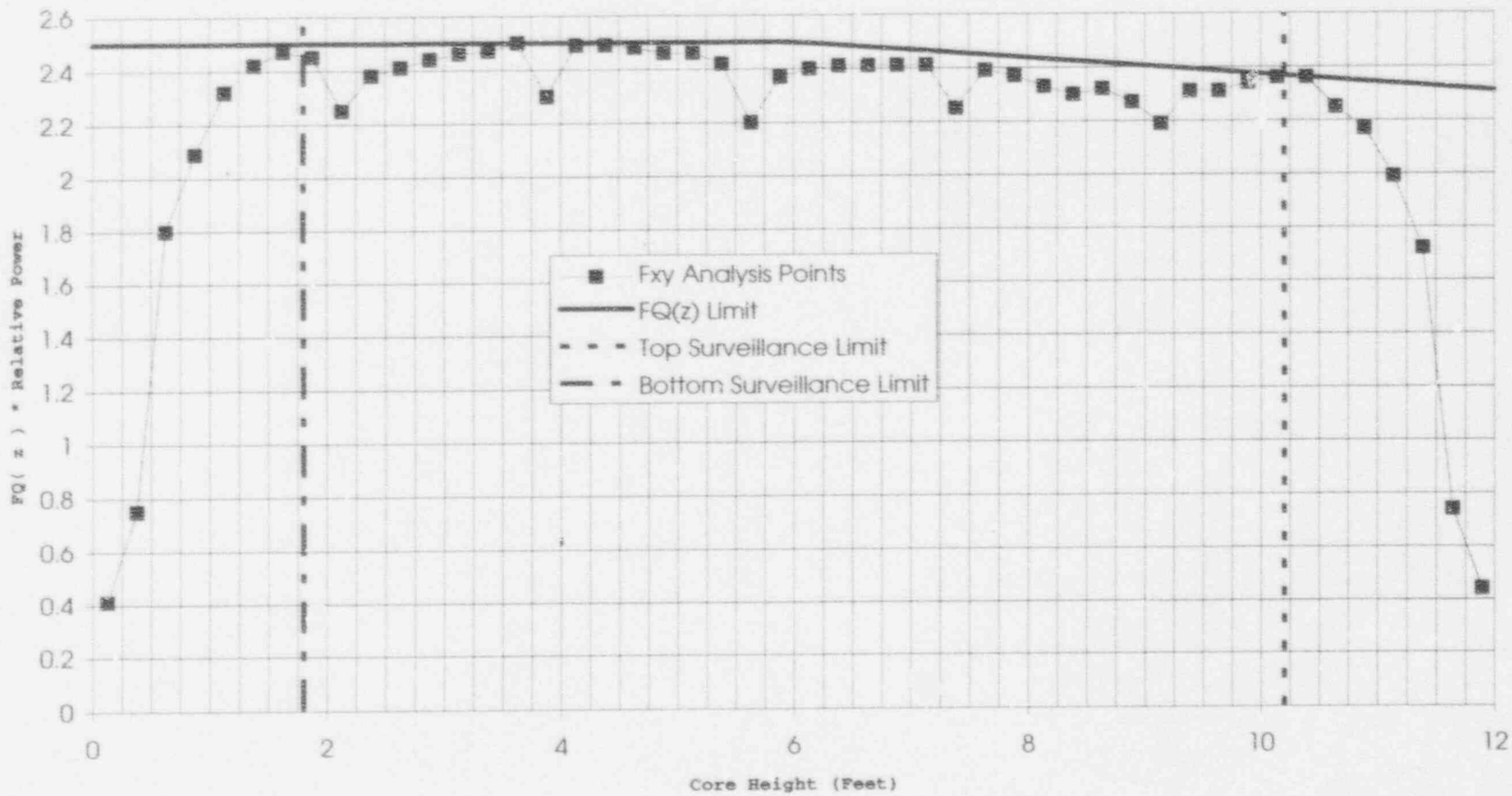

ATTACHMENT 3

References

1. Westinghouse WCAP-9273-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.

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

3. Letter from R. A. Chrzanowski (ComEd) to Dr. T. Murley (NRC), "Byron Station Units 1 and 2 Application for Amendment to Facility Operating Licenses NPF-37 and NPF-66," dated July 31, 1989.

4. Letter from L. N. Olshan (NRC) to T. J. Kovach (ComEd), "Amendment No 36 Use of VANTAGE 5 Fuel," dated January 31, 1990.

5. Letter from G. F. Dick (NRC) to D. L. Farrar (ComEd), "Issuance of Amendments (TAC NOS. M90575, M90576, M90577, and M90578)," dated October 21, 1994.