Commonwealth Edison
1400 Opus Place
Downers Grove, Illinois 60515

February 16, 1990

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

Attn: Document Control Desk

Subject: Braidwood Station Units 1 and 2
Supplement to Application for Amendment
to Facility Operating Licenses NPF-72 and NPF-77
NRC Docket Nos. 50-456 and 50-457

Reference: a) October 19, 1989 S.C. Hunsader letter
to T.E. Murley

b) January 9, 1990 S.C. Hunsader letter
to T.E. Murley

Dear Dr. Murley:

In reference (a) pursuant to 10 CFR 50.90, Commonwealth Edison proposed to amend Appendix A, Technical Specifications of Facility Operating Licenses NPF-72 and NPF-77. The proposed amendment requests the use of VANTAGE 5 fuel or the combination of VANTAGE 5 fuel and the present Optimized Fuel Assembly core at Braidwood Station Units 1 and 2. Reference (b) provided additional information to supplement the No Significant Hazards evaluation.

The purpose of this letter is to provide corrections to information previously submitted.

Attachment A provides a revised page 24 of Attachment 4 to reference (a) to correct the maximum local Zr-H_2O reaction percent from 2.14% to 3.26%.

Included in Attachment B is Section 6.0 which is being provided for completeness. Similar information had been previously provided and is included in Attachment 2 to reference (a).

Attachment C provides a revised Section 7.0 that corrects the identification numbers for the reference documents, listed. Also documents that were previously listed, but are not applicable to the information in the report text, have been deleted.

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Included in Attachment D is a revised marked-up Technical Specification page 2-8, that changes "Unit 1, Cycle 4" to "Unit 1, Cycle 3", and "Unit 2, Cycle 3" to "Unit 2, Cycle 2", both of which are the correct power cycles for Braidwood Station.

Commonwealth Edison is notifying the State of Illinois of this supplement to the application for the amendment by transmitting a copy of this letter and its attachments to the designated State Official.

Please direct any questions regarding this submittal to this office.

Very truly yours,

S. C. Hunsader

S. C. Hunsader
Nuclear Licensing Administrator

/lmw:0487T:5

Attachments A, B, C & D

cc: Braidwood Resident Inspector
S.P. Sands - NRR
Office of Nuclear Reactor Safety - IDNS
S. Sun - NRR

Attachment A

The maximum calculated local metal-water was 3.26% which is well below the embrittlement limit of 17% specified in 10CFR50.46. The total core wide metal-water reactions is less than 0.3% for all breaks, as compared with the 1% criterion of 10CFR50.46 and in all cases the cladding temperature transient was terminated at a time when the core geometry was still amenable to cooling. As a result, the core temperature will continue to drop and the ability to remove decay heat generated in the fuel for an extended period of time will be provided.

These results provide assurance that operation with VANTAGE 5 fuel and with the RCS hot leg temperature in the range of 600 to 619.3°F can be accomplished within the requirements of 10CFR50.46 and Appendix K to 10CFR50.46.

Small Break Results

This section presents the results of a spectrum of small break sizes analyzed for the Byron/Braidwood Stations. As noted previously, the calculated peak clad temperature resulting from a small break LOCA is less than that calculated for a large break. Based on the results of LOCA sensitivity studies (Reference 14 and 21) the limiting small break was found to be less than a 10-inch diameter rupture of the RCS cold leg. The worst breaks size (small break) is a 3-inch diameter break in the cold leg. This limiting break size was also analyzed for the reduced RCS operating temperatures to show that the reduced temperature results in a less severe transient. The time sequence of events and the results for all the breaks analyzed is shown in Tables 15.6-1 and 15.6-4.

During the earlier part of the small break transient, the effect of the break flow is not strong enough to overcome the flow maintained by the reactor coolant pumps through the core as they are coasting down following reactor trip. Therefore, upward flow through the core is maintained. The resultant heat transfer cools the fuel rods and cladding to very near the coolant temperature as long as the core remains covered by a two-phase mixture. This effect is evident in the accompanying figures.

Attachment B

6.0 SUMMARY OF TECHNICAL SPECIFICATION CHANGES

Table 6.1 presents a list of the Technical Specification changes and justification for the changes. The changes noted in Table 6.1 are given in the proposed Technical Specification page mark ups (see Appendix A of this report). Included in Appendix A are separate mark ups for the Byron and Braidwood Stations Units 1 and 2.

TABLE 6.1

Summary and Justification for Byron and Braidwood Stations Units 1 and 2 Technical Specification Changes for VANTAGE 5 Fuel

PAGE	SECTION	DESCRIPTION OF CHANGE	JUSTIFICATION
2-8	Table 2.2-1	Revised the F(ΔI) offset wings and gains with cycle specific identification.	These changes are due to the VANTAGE 5 fuel design.
B 2-1 B 2-2	2.1.1 Basis	Added DNB correlations and design and Safety Analysis DNBR limits for the VANTAGE 5 fuel. Added new $F^N_{\Delta H}$ values.	These changes reflect the DNB correlations and the values for $F^N_{\Delta H}$ for the VANTAGE 5 and OFA fuel.
3/4 1-4 3/4 1-5	3.1.1.3	BOL deleted from MTC LCO and Surveillance 4.1.1.3. modified to compare BOL MTC with predicted MTC with burnup and develop rod withdrawal limits to keep MTC negative. Added "Provisions of Specification 3.0.4 are not applicable." to the Action Statement.	These changes reflect increasing MTC with burnup before decreasing toward EOL for VANTAGE 5 core and to allow entry into Modes 1 and 2, if the requirements of the Action Statements are met.
3/4 1-19	3.1.3.4	Revised the rod drop time to ≤ 2.7 seconds and added cycle specific identification.	This change is the result of an increase in the core hydraulic resistance due to the VANTAGE 5 fuel design.

TABLE 6.1 (continued)

PAGE	SECTION	DESCRIPTION OF CHANGE	JUSTIFICATION
3/4 2-4	3.2.2	Added new F_Q limit and cycle specific identification.	This change reflects the value for F_Q assumed in the safety analysis for the VANTAGE 5 fuel design.
3/4 2-5	3.2.2	Replaced Figure 3.2-2 with 2 segment curve.	This curve is consistent with the VANTAGE 5 analysis.
3/4 2-7	3.2.2	In 4.2.2.2.f.3 add "(except VANTAGE 5 fuel assembly IFM grids)".	The VANTAGE 5 fuel assembly IFM grids wil not significantly distort the indicated flux during the F_{xy} surveillance.
3/4 2-8	3.2.3	Revised the $F^N_{\Delta H}$ limits.	These changes reflect the values for $F^N_{\Delta H}$ assumed in the safety analyses for VANTAGE 5 and OFA fuel.
B 3/4 1-2	3/4 1.1.3 Basis	Reworded Surveillance justification paragraph.	This change reflects increasing MTC with burnup before decreasing toward EOL for VANTAGE 5 core.
B 3/4 2-1	3/4.2 Basis	Revised basis discussion of DNB.	These changes reflect the new DNB correlations used for the VANTAGE 5 and OFA fuel.

TABLE 6.1 (continued)

PAGE	SECTION	DESCRIPTION OF CHANGE	JUSTIFICATION
B 3/4 2-1	3/4.2.1 Basis	Changed the axial peaking factor multiplier to F_Q limit.	This change reflects the value for F_Q assumed in the safety analyses for either OFA or VANTAGE 5 fuel design.
B 3/4 2-4	3/4.2.2 3/4.2.3	Revised basis discussion for rod bow penalty.	This change reflects the new DNB correlations used for VANTAGE 5 and OFA fuel.
B 3/4 2-5	3/4.2.2 3/4.2.3	Revised basis discussion of $F^N_{\Delta H}$ limits.	Revised $F^N_{\Delta H}$ limits to include VANTAGE 5 fuel design.

Attachment C

7.0 REFERENCES

1. Davidson, S. L., Iorii, J. A., "Reference Core Report - 17x17 Optimized Fuel Assembly," WCAP-9500-A, May 1982.

2. Davidson, S. L., Kramer, W. R. (Eds.) "Reference Core Report VANTAGE 5 Fuel Assembly," WCAP-10444-P-A, September 1985.

3. Davidson, S. L. (Ed.), et al., "Westinghouse Reload Safety Evaluation Methodology," WCAP-9272-P-A, July 1985.

4. Miller, J. V., "Improved Analytical Models Used in Westinghouse Fuel Rod Design Computations," WCAP-8720 (Proprietary), October 1976.

5. Weiner, R. A., et al., "Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations," WCAP-10851-P-A, August 1988.

6. Davidson, S. L. (Ed.) et al., "Extended Burnup Evaluation of Westinghouse Fuel," WCAP-10125-P-A, December 1985.

7. Letter from W. J. Johnson (W) to M. W. Hodges (NRC), "VANTAGE 5 Bottom Nozzle," NS-NRC-88-3366, September 1988.

8. Skaritka, J., (Ed.), "Fuel Rod Bow Evaluation," WCAP-8691, Revision 1 (Proprietary), July 1979.

9. Davidson, S. L., Iorii, J. A. (Eds.) "Verification Testing and Analyses of the 17x17 Optimized Fuel Assembly," WCAP-9401-P-A, August 1981.

10. Skaritka, J., et al., "Westinghouse Wet Annular Burnable Absorber Evaluation Report," WCAP-10021-P-A, Revision 1, October 1983.

11. Chelemer, H., Boman, L. H., Sharp, D. R., "Improved Thermal Design Procedure," WCAP-8567-P-A, February 1989.

12. Letter from NRC to Westinghouse from Stolz to Eicheldinger, SER on WCAP-7956, 8054, 8567 and 8762 dated April 1978.

13. Motley, F. E., et al., "Westinghouse Correlation WRB-1 for Predicting Critical Heat Flux in Rod Bundles with Mixing Vane grids," WCAP-8762-P-A and WCAP-8763-A, July 1984.

14. Letter from E. P. Rahe (W) to Miller (NRC) dated March 19, 1982, NS-EPR-2573, WCAP-9500 and WCAPS 9401/9402 NRC SER Mixed Core Compatibility Items.

15. Letter from C. O. Thomas (NRC) to Rahe (W) - Supplemental Acceptance No. 2 for Referencing Topical Report WCAP-9500, January 1983.

16. Letter from W. J. Johnson (W) to M. W. Hodges (NRC), NS-NRC-87-3268, "VANTAGE 5 DNB Transition Core Effects," October 2, 1987.

17. Letter from M. W. Hodges (NRC) to W. J. Johnson (W), NRC SER on VANTAGE 5 Transition Core Effects, dated February 24, 1988.

18. Schueren, P., McAtee, K. R., "Extension of Methodology for Calculating Transition Core DNBR Penalities," WCAP-11837, May 1988.

19. Butler, J. C. and D. S. Love, "Steamline Break Mass/Energy Releases for Equipment Environmental Qualification Outside Containment," WCAP-10961-P (Proprietary) and WCAP-11184 (Non-Proprietary), October 1985.

20. DiTommaso, S. D. et al., "Byron/Braidwood T_{Hot} Reduction Final Licensing Report," WCAP-11386, Rev. 2 (Proprietary) and WCAP-11387, Rev. 2 (Non-Proprietary), November 1987.

21. Kabadi, J. N. et al., "The 1981 Version of the Westinghouse ECCS, Evaluation Model Using the BASH Code," WCAP-10266 P-A, Revision 2, with Addenda, (Proprietary), March 1987.

22. Besspiata, J. J. et al., "The 1981 Version of the Westinghouse ECCS, Evaluation Model Using the BASH Code, Power Shape Sensitivity Studies," WCAP-10266-P-A, Revision 2, Addendum 1, (Proprietary), December 15, 1987.

23. Lee, N., et al., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10054-P-A (Proprietary) and WCAP-10081-A (Non-Proprietary), August 1985.

24. Young, M., et al., "BART-1A: A Computer Code for the Best Estimate Analyzed Reflood Transients," WCAP-9561-P-A, 1984.

25. Meyer, P.E., "NOTRUMP, A Nodal Transient Small Break and General Network Code," WCAP-10079-P-A (Proprietary) and WCAP-10080-P-A (Non-Proprietary) August 1985