Commonwealth Edison
1400 Opus Place
Downers Grove, Illinois 60515

May 15, 1991

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 Unit 1
Cycle 3 Reload
NRC Docket No. 50-456

REFERENCES:

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

2. CECo submittal, J.A. Silady to T.E. Murley dated July 13, 1990; titled "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; 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".

5. NRC SER on CECo's Neutronics Topical (Ref. 4) dated December 13, 1983.

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6. Westinghouse Letter 91CB*-G-0036, March 19, 1991 J. W. Swogger to W. F. Naughton; "Grid Damage to Fuel Assemblies D07S and C22S"

7. NRC Letter from S.P. Sands to T.E. Kovach, Amendment No. 23; "Use of VANTAGE 5 Fuel", dated April 19, 1990.

Dear Dr. Murley:

Braidwood Unit 1 has completed its second cycle of operation and is conducting a refueling outage that began March 1, 1991. Braidwood Unit 1, Cycle 2 attained a final cycle burnup of approximately 13,052 MWD/MTU. Cycle 3 is expected to commence on May 7, 1991. This letter is to advise you of Commonwealth Edison Company's (CECo) plans regarding the Braidwood Unit 1, Cycle 3 reload core.

Attachment 1 describes the core reload including CECo's safety evaluation of the core redesign due to fuel assembly reconstitution and replacement of fuel assemblies: D29T (significant grid damage) and symmetric assemblies D22T, D19T, and D20T. All aspects of the redesign have been reviewed and compared to the original Cycle 3 safety parameters. It has been determined that the Cycle 3 safety parameters remain valid for the redesigned core configuration. The Braidwood Unit 1, Cycle 3 reload review was performed in accordance with the provisions of 10CFR50.59 and no unreviewed safety questions exist as a result of the redesign. 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 developed by Westinghouse as described in Reference 1. Commonwealth Edison requested approval to perform the neutronic portion of the reload design using the methods and codes described in References 2 and 4 and the NRC approved these requests in References 3 and 5, respectively. Specifically, the Braidwood Unit 1, Cycle 3 reload design, including the development of the core operating limits, was generated by Commonwealth Edison using the NRC approved methodologies.

Please direct any questions regarding this notification to this office.

Very truly yours,

Allen R. Checca

Allen R. Checca
Nuclear Licensing Administrator

cc: P.M. Pulsifer - NRR Project Manager
A.B. Davis - Region III Administrator
S. DuPont - Senior NRC Resident Inspector - Braidwood

ZNLD/948/4

ATTACHMENT 1

Braidwood Unit 1, Cycle 3 Reload Description

The Braidwood Unit 1, 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 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. 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). Based on this review, the safety evaluations will be performed by the Commonwealth Edison On-Site and Off-Site Reviews pursuant to the requirements of 10CFR50.59(a) and 10CFR50.59(b).

The Braidwood Unit 1, 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, sixty-four (64) VANTAGE 5 fuel assemblies will be inserted into the core. The Braidwood Unit 1 core will then contain a combination of fresh Westinghouse 17x17 VANTAGE 5 assemblies and previously irradiated 17x17 Optimized Fuel Assemblies (OFA's). The NRC approved the use of VANTAGE 5 at Braidwood Unit 1 for Cycle 3 and thereafter, under the provisions of 10CFR50.90 in Reference 7. Braidwood/Byron UFSAR 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 previously 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. 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 may potentially enter the active fuel region. This feature is currently in operation at Commonwealth Edison's Byron and Braidwood Units.

The Braidwood Unit 1, Cycle 3 core has been redesigned and evaluated using NRC licensed and approved methods. Commonwealth Edison requested approval to perform the neutronic portion of the PWR reload design using the methods described in Reference 2, and the NRC approved this request (Reference 3). Specifically, the Braidwood Unit 1, Cycle 3 reload design, including the development of the core operating limits, was generated and verified by Commonwealth Edison using NRC approved methodology.

The reload redesign also reflects the reconstitution of fuel assemblies D73U, and the discharge of damaged fuel assembly D29T. Fuel assembly D29T, and its three symmetric assemblies, were discharged when Grid #1 of assembly D29T was discovered to have been damaged during the prior core reload. Two adjacent assemblies, D07S and C22S, also sustained some damage to a grid strap. The damage to D07S's grid strap and C22S's grid strap were evaluated by Westinghouse in Reference 6. The evaluation determined that the damage was minor in nature and allowed both assemblies to be used as is in the reload design. As the original design efforts and Reload Safety Evaluation did not reflect these changes, a comprehensive redesign and reexamination of safety limits has been performed.

The reload fuel's nuclear design is evaluated generically in the UFSAR. 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 (for the original Braidwood Unit 1, Cycle 3 design) were evaluated in detail in the CECo/Westinghouse reload safety evaluation process. The redesigned core was then evaluated in detail through a second CECo/Westinghouse reload safety evaluation process. The second meeting was performed to ensure that the redesigned core met the same safety analysis limits and assumptions.

The thermal-hydraulic design for the Cycle 3 reload core has not significantly changed from that of the previously reviewed and accepted initial cycle design. Tests and analysis have confirmed that the VANTAGE 5 assemblies are hydraulically compatible with the OFA assemblies reloaded as Regions 3, 4A, 4B, and 4C. The FNDH limits of less than 1.55 for OFA assemblies and 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 being applied.

Westinghouse determined through the Reload Safety Evaluation (RSE) process that the Braidwood Unit 1 non-LOCA and LOCA analyses described in the Braidwood/Byron UFSAR were not adversely impacted by the original Braidwood Unit 1, Cycle 3 design. Westinghouse and CECo have subsequently reconfirmed that the impact of the redesign, and the reconstitution, on the original Reload Safety Evaluation (RSE) conclusions were negligible.

Westinghouse determined through the Reload Safety Evaluation (RSE) process that the Braidwood Unit 1 non-LOCA and LOCA analyses described in the Braidwood/Byron UFSAR were not adversely impacted by the original Braidwood Unit 1

The Braidwood Unit 1, 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 Braidwood 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.

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 Braidwood Unit 1, Cycle 3 Reload Safety Evaluation applied both the LOCA and non-LOCA safety analyses presented in Braidwood/Byron UFSAR, 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 Braidwood UFSAR, and in Reference 1. Commonwealth Edison verified that accident analyses presented in the UFSAR, as modified by the analyses described in Reference 1, were not affected by the reload core characteristics. All aspects of the redesign have been reviewed against the original Cycle 3 safety parameters, and it has been determined that the Cycle 3 safety limits previously established remains valid for the redesign.

The reload safety evaluation demonstrated that no additional Technical Specification changes, beyond those previously described, are required for operation of Braidwood Unit 1 during Cycle 3. Commonwealth Edison believes that both the On-Site and Off-site reviews will conclude that no unreviewed safety questions, as defined by 10CFR50.59, are involved with this reload. More specifically with this reload:

a) The redesign does not involve an increase in the probability of occurrence of an accident previously evaluated in the safety analysis report. The revised Braidwood Unit 1, Cycle 3 design has been verified to satisfy accident analysis limits and assumptions. The design and construction of the reload core and its fuel assemblies is to the same standards as the previously installed equipment. As such, the probability of accidents applicable to all licensed MODES remains unchanged.

b) The redesign does not involve an increase in the consequences of an accident previously evaluated in the safety analysis report. The Braidwood Unit 1, Cycle 3 reload parameters have been verified to be less limiting than than the bounding values assumed in the safety analyses. As such, the consequence of accidents previously analyzed remain unchanged.

c) The redesign does not involve an increase in the probability of a malfunction of equipment important to safety which has been previously analyzed in the safety analysis report. The reload core does not impact the design or operation of any active plant equipment.

d) The redesign does not involve an increase in the consequences of a malfunction of equipment important to safety which has been previously analyzed in the safety analysis report. The reload core does not impact the design or operation of any active plant equipment.

e) The redesign does not create the possibility of a new or different kind of accident from any accident previously evaluated. This is based on the fact that the reload core's structural, thermal-hydraulic, and nuclear characteristics are not significantly different from previously installed equipment and that the method and manner of plant operation is unchanged.

f) The redesign does not create the possibility of a new or different type of a malfunction of equipment important to safety which has been previously analyzed in the safety analysis report. This is based on the fact that the redesign does not change any core or fuel design requirement and does not affect any other plant system or component.

g) The redesign does not involve a significant reduction in a margin of safety. The Braidwood Unit 1, Cycle 3 reload core has been shown to remain within present safety analysis limits and therefore margins to safety limits have not been impacted. The reference safety analysis of record is documented in the VANTAGE 5 Reload Transition Safety Report which has been approved under the provisions of 10CFR50.59, and in Chapter 4 and 15 of the Byron/Braidwood UFSAR.

Therefore, based upon the assumption that the On-Site and Off-site Reviews will find no unresolved safety questions, no additional prior NRC review and approval of the reload core analyses and application for amendment to the Braidwood Unit 1 operating license is required as a result of the cycle specific reload design for Cycle 3.

Finally, verification of the Braidwood Unit 1, 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;

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

7. In addition, per the requirements of the NRC SER (Reference 3), Braidwood will provide a report to the NRC detailing the results of the startup physics tests.

ATTACHMENT 2

Braidwood Unit 1 Cycle 3
Operating Limit Report - Fxy Portion (Revised 3/21/91)

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

The Fxy limits for RATED THERMAL POWER within s ˙ri˙ ed 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 2.304 for all core planes containing bank "D" control rods, and

   2. F_{xy}^{RTP} less than or equal to 1.755 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 2.304 for all core planes containing bank "D" control rods, and

   2. F_{xy}^{RTP} less than or equal to 1.934 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:

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

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

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

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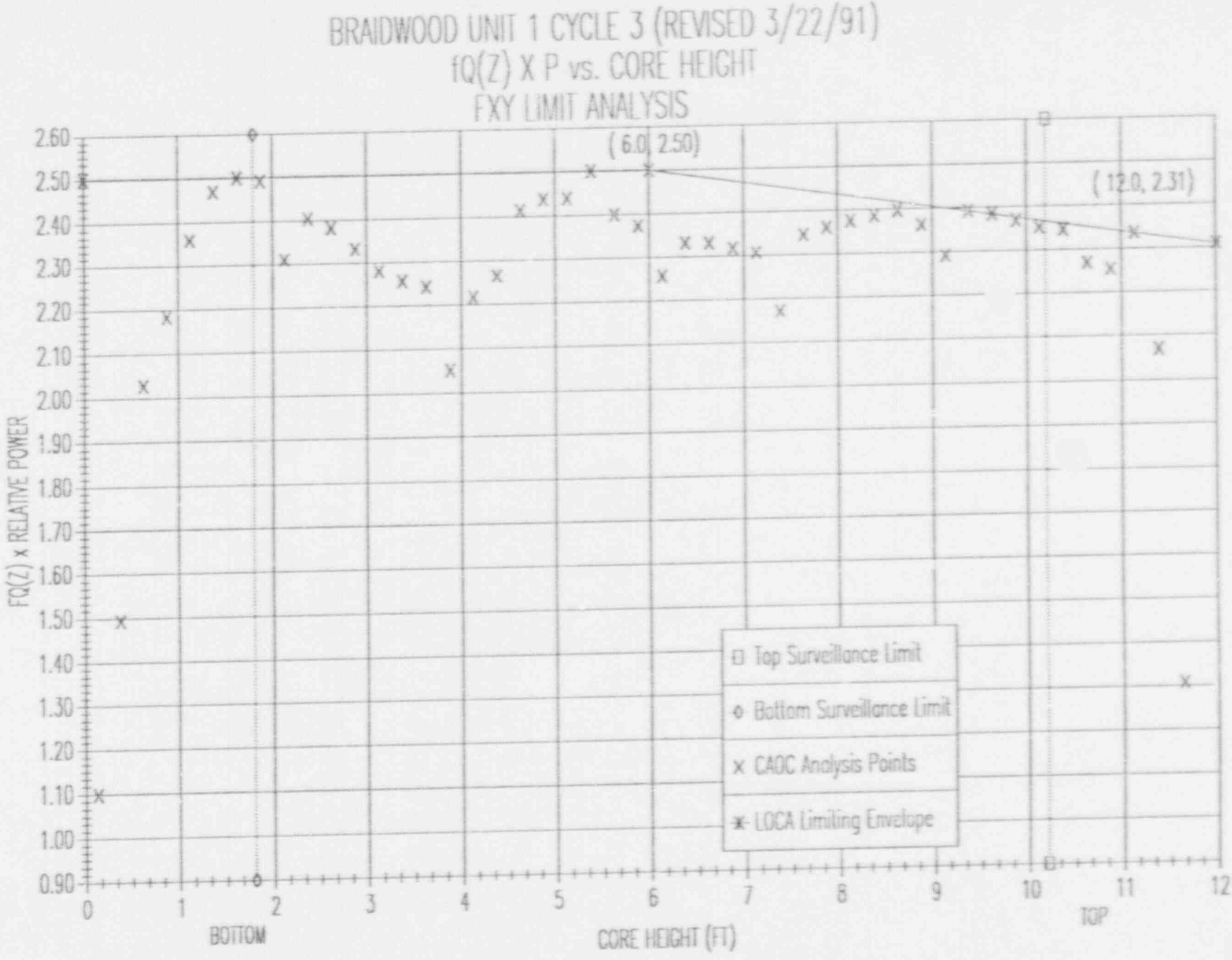
BRAIDWOOD UNIT 1 CYCLE 3 (REVISED 3/22/91)
fQ(Z) X P vs. CORE HEIGHT
FXY LIMIT ANALYSIS
(6.0, 2.50)
(12.0, 2.31)
FQ(Z) x RELATIVE POWER
CORE HEIGHT (FT)
BOTTOM
TOP
Top Surveillance Limit
Bottom Surveillance Limit
CAOC Analysis Points
LOCA Limiting Envelope