Commonwealth Edison
72 West Adams Street, Chicago, Illinois
Address Reply to: Post Office Box 767
Chicago, Illinois 60690 - 0767

September 22, 1989

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

Subject: Braidwood Station Unit 1
Cycle 2 Reload
NRC Docket No. 50-456

References: (1) Westinghouse WCAP-10021-P-A, dated October 1983; "Westinghouse Wet Annual Burnable Absorber Evaluation Report"

(2) CECo submittal, K.A. Ainger to U.S. NRC, "Byron/Braidwood Thot Reduction LOCA Analysis", dated December 4, 1987.

(3) 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". NL-83-0041.

(4) NRC SER on CECo's Neutronics Topical (Ref. 3) dated December 13, 1983

Dear Dr. Murley:

Braidwood Unit 1 is completing its first cycle of operations and will be conducting a refueling outage beginning on September 2, 1989. Braidwood Unit 1, Cycle 2 is expected to commence the week of November 13, 1989. The purpose of this letter is to advise the NRC Staff of Commonwealth Edison's (Edison) plans regarding the Cycle 2 reload core and to provide the Cycle 2 Core Operating Limits Report. Braidwood Unit 1 Cycle 1 is projected to attain a final cycle burnup of approximately 17,600 MWD/MTU.

A001
1/1

Attachment A describes the reload and Edison's in-progress review in accordance with 10 CFR 50.59.

Attachment B provides the Core Operating Limits Report for Cycle 2 pursuant to Technical Specification 6.9.1.9.

Please direct any questions regarding this submittal to this office.

S.C. Hunsader

S.C. Hunsader
Nuclear Licensing Administrator

/scl:0267T:3-4

cc: S.P. Sands
A.B. Davis
J. Hinds
Resident Inspector-Braidwood

ATTACHMENT A

BRAIDWOOD STATION UNIT 1

CYCLE 2 RELOAD DESCRIPTION

The Braidwood Unit 1, Cycle 2 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 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. Edison performed a detailed review with Westinghouse on the bases, including all the postulated incidents considered in the UFSAR, of the Reload Safety Evaluation (RSE).

The Braidwood Unit 1 Cycle 2 core will be a Low Leakage design. Edison has successfully developed and used similar Low Leakage designs at its Byron and Zion units. The Braidwood Unit 1 core will continue to be refueled with Westinghouse's 17X17 Optimized Fuel Assemblies (OFA's). Wet Annular Burnable Absorbers (WABAs) will be used as the burnable poison. WABAs were approved for use in Reference 1, and have been used extensively by Commonwealth Edison at Zion and Byron stations. The reload fuel mechanical, nuclear, and thermal-hydraulic design for the Cycle 2 reload core has not significantly changed from that of the previously reviewed and accepted initial cycle design. The reload fuel assemblies will incorporate Westinghouse standardized fuel pellets, reconstitutable top nozzles (RTN), extended burnup design features and snag resistant grids. Similar features have been successfully utilized by Commonwealth Edison at Byron Station Units 1 and 2. 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 inhibiting debris from entering the active fuel region.

The current FNDH limit of less than 1.55 ensures that the DNB ratio of the limiting power rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being applied. In addition, based upon the performance of an eighteen case FAC analysis, a total peaking factor (Fq) of 2.306 is the maximum which could occur for the full range of power distributions, including load follow maneuvers allowable under Constant Axial Offset Control (CAOC). Therefore, additional surveillance of Fq(z) is not required in Cycle 2. The Cycle 2 radial peaking factor (Fxy) limits are described in the Core Operating Limits Report. (See Attachment B.)

The Braidwood 1 Cycle 2 reload safety evaluation applied the LOCA analysis presented in Reference 2. The analysis was performed using a Fq limit of 2.40 and a FNDH of 1.62 for the LOCA events only. It should be noted that CECo will not take credit for the increased peaking limits at Braidwood 1 during Cycle 2. The reference 2 LOCA analysis also incorporated a two lines segment K(z) curve instead of the existing three line segment curve. Edison will continue to apply Fxy surveillance limits that reflect the existing 3 line segment curve until approval of the two line segment curve is received from the NRC.

The RCCA Bank Withdrawal from Subcritical event was reanalyzed with a reduced differential rod worth in order to generate additional margin for the W-3 correlation. The WRB-1 correlation had been used in the past to evaluate DNBR over the entire length of the fuel assembly; however, it was determined that the WRB-1 correlation was inappropriate for use in the span between the lower non-mixing vane grid an the first mixing vane grid. Therefore, the W-3 correlation was used since it does not credit the mixing vane grids. The WRB-1 correlation remains applicable for the rest of the fuel assembly. On this basis the UFSAR conclusions were shown to remain valid for Cycle 2 operation.

It should be noted that Edison currently plans to operate Cycle 2 at a reduced Thot of 608°F in order to minimize potential primary water stress corrosion cracking in the steam generator tubes. Analysis demonstrating that Edison can operate the Byron and Braidwood units at Thot temperatures as low as 600°F, WCAP-11387 (Reference 2), has been submitted to the NRC for information only. The nominal Tavg, at Rated Thermal Power, used in the OTDT and OPDT setpoint equations will be rescaled to the reduced Thot nominal average temperature. This change is required to provide effective reactor protection at the Cycle 2 nominal operating conditions. Reduced Thot operation has been successfully demonstrated at Byron Station Units 1 and 2.

The reload safety evaluation 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 paramters affecting the postulated accident analyses reported in the Braidwood UFSAR. For those incidents whose consequences could potentially be affected by the reload core characteristics, the incidents were reanalyzed. Edison verified that the reanalyses were performed in accordance with the Westinghouse reload safety evaluation methodology, as outlined in the March 1978 Westinghouse Topical Report entitled "Westinghouse Reload Safety Evaluation Methodology" (WCAP-9272-P-A), and were consistent with the PWR neutronic methods currently employed by Commonwealth Edison, as qualified in the Reference 3 topical report and related NRC SER (Reference 4). Edison also verifies that the results of these reanalyses were within previously reviewed and accepted limits.

The reload safety evaluation demonstrates that reload related Technical Specification changes are not required for operation of Braidwood Unit 1 during Cycle 2. Edison's On-Site and Off-Site reviews have concluded that no unreviewed safety questions, as defined by 10 CFR 50.59, are involved with this reload. More specifically with this reload:

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 Braidwood Unit 1 Technical Specification.

Accordingly no prior NRC review and approval of the reload core analyses or an application for amendment to the Braidwood Unit 1 operating license will be required as a result of the cycle specific design for Cycle 2.

Finally, verification of the reload core design will be performed per the standard reload startup physics tests. These tests include, but are not limited:

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.

SCH/scl:0267T:6

ATTACHMENT B

BRAIDWOOD STATION UNIT 1

CYCLE 2

CORE OPERATING LIMITS REPORT

SCH/scl:0267T:8

Braidwood Unit 1 Cycle 2
Operating Limit Report - Fxy Portion

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 F_{xy} limits for RATED THERMAL POWER within specified core planes for Cycle 2 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 1.856 for all core planes containing bank "D" control rods, and

2. F_{xy}^{RTP} less than or equal to 1.597 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 1.728 for all core planes containing bank "D" control rods, and

1. F_{xy}^{RTP} less than or equal to 1.641 for all unrodded core planes.

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

$F_Q(z) \leq [\frac{2.32}{P}]\ [K(z)]$ for $P > 0.5$ and,

$F_Q(z) \leq [4.64]\ [K(z)]$ 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.

Figure 1
Maximum FQT*Power versus Axial Height
During Normal Core OPeration