Commonwealth Edison
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Downers Grove, Illinois 60515

May 3, 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 Unit 2
Cycle 2 Reload
NRC Docket No. 50-457

References:

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

(2) CECo submittal, F.G. Lentine to H.R. Denton Dated July 27, 1983; entitled "Zion Station 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".

(3) NRC Safety Evaluation Report (SER) on CECO's Neutronics Topical (Ref. 1), dated December 13, 1983.

(4) CECo submittal, S. C. Hunsader to T.E. Murley, "Braidwood Station Units 1 and 2 Application to Facility Operating License NPF-72 and NPF-77," dated October 11, 1989, regarding VANTAGE 5 Fuel.

Dear Dr. Murley:

Braidwood Station Unit 2 has completed its first cycle of operation and is conducting a refueling outage that began on March 17, 1990. Braidwood Unit 2 Cycle 2 is expected to commence the week of May 6, 1990. The purpose of this letter is to advise the NRC staff of Commonwealth Edison Company's (CECo) plans regarding the Braidwood Unit 2 Cycle 2 reload core. Braidwood Unit 2 Cycle 1 attained a final cycle burnup of approximately 17,900 MWD/MTU.

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Attachment A describes the core reload and Edison's in-progress review which is being performed in accordance with 10CFR50.59. Attachment B provides the Core Operating Limits Report for Cycle 2, pursuant to Technical Specification 6.9.1.9. Commonwealth Edison applies NRC approved reload design methodology developed by Westinghouse as described in Reference 1. Commonwealth Edison requested approval to perform the neutronic portion of the reload design in Reference 2, and the NRC staff approved this request in Reference 3. The Braidwood 2 Cycle 2 reload design, including development of the core operating limits has, therefore, been generated by Commonwealth Edison using NRC approved methodology.

Please direct any questions regarding this submittal to this office.

Very truly yours,

Steven C Hunsader

Steven C. Hunsader
Nuclear Licensing Administrator

cc: S.P. Sands (NRR)
A.B. Davis (RIII)
W. Shafer (RIII)
Braidwood Resident Inspector

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ATTACHMENT A

BRAIDWOOD STATION UNIT 2 CYCLE 2 RELOAD DESCRIPTION

The Braidwood Unit 2 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 Byron/Braidwood Stations Final Safety Analysis Report (UFSAR) which could potentially be affected by fuel reload, it has been demonstrated that the results of the postulated events are within allowable limits. The reanalysis, described in Reference 4, has been submitted for NRC review and approval. Commonwealth Edison received NRC approval on April 19, 1990 as a part of Amendment 23 to the Braidwood Technical Specification. An identical analysis was approved for Byron Station on January 31, 1989. Commonwealth 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). Based on this review, the Westinghouse RSE, the NRC approval of CECo's Reference 4 submittal, 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 2 Cycle 2 core is a "Low Leakage" design. Commonwealth Edison has successfully developed and used similar "Low Leakage" designs at its Byron and Zion units. During the Cycle 1/2 refueling, eighty-four (84) VANTAGE 5 fuel assemblies are planned to be inserted into the core. The Braidwood Unit 2 core will then contain a combination of fresh Westinghouse VANTAGE 5 assemblies and Westinghouse's 17x17 Optimized Fuel Assemblies (OFA's), as described in Reference 4. Reference 4 requested approval for the transition to VANTAGE 5 fuel and associated proposed changes to the Braidwood Technical Specifications. NRC approval was granted on April 19, 1990. The information provided in the submittal fully justified the compatibility of Westinghouse OFA and VANTAGE 5 assemblies in a reload core, and verifies 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. WABAs have been used extensively by Commonwealth Edison. A description and evaluation of IFBA rods is presented in Reference 4. Commonwealth Edison is currently operating Byron Unit 1 with a similar combination of fuel types and burnable absorbers.

The reload fuel assemblies incorporate Westinghouse standardized fuel pellets, reconstitutable top nozzles (RTN), extended burnup design features, and snag resistant grids. Similar features have been successfully utilized previously in Commonwealth Edison's Byron and Braidwood Units. Additionally, the reload fuel assemblies 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 enters the active fuel region. This feature is currently in operation at both Byron Unit 1 and Braidwood Unit 1. The significant new mechanical features of the VANTAGE 5 design are the Intermediate Flow Mixer (IFM) grids and the Axial Blankets. Structural evaluations of these fuel features provided in Reference 4 verify that the VANTAGE 5 assembly design is structurally acceptable.

The reload fuel's nuclear design has been evaluated in Reference 4. 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 and VANTAGE 5 fuel are within the range normally seen from cycle to cycle due to fuel management effects. The loading pattern dependent parameters were evaluated in detail in the CECo/Westinghouse reload safety evaluation described below. In addition, based upon the results of an eighteen case FAC analysis, a total peaking factor (Fq) of less than 2.50 is the maximum which could occur for the full range of power distributions, including load follow maneuvers, allowable under Constant Axial Offset Control (CAOC). The Cycle 2 radial peaking factor (Fxy) limits are described in the attached "Core Operating Limits Report". (See Attachment B)

The thermal-hydraulic design for the Cycle 2 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 2 and 3. The proposed 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.

Commonwealth Edison's reload safety evaluation process is a verification that previously reviewed and approved accident analyses are not adversely impacted by the cycle specific reload core design. Commonwealth Edison's Braidwood Unit 2 Cycle 2 reload safety evaluation applied both the LOCA and non-LOCA safety analyses presented in Reference 4, 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 Reference 4. Commonwealth Edison verifies that accident analyses presented in the UFSAR, as modified by the analysis described in Reference 4, have not been affected by the reload core characteristics.

The reload safety evaluation demonstrated that no additional Technical Specification changes, beyond those previously submitted for NRC approval in Reference 4, are required for operation of Braidwood Unit 2 during Cycle 2. The conclusion that no unreviewed safety questions exist, as defined by 10CFR50.59, is also being confirmed by the inprogress On-site and Off-Site reviews. 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 2 Technical Specification.

Accordingly, prior NRC review and approval of the reload core analysis and application for amendment to the Braidwood Unit 2 operating license, beyond that requested in Reference 4, is not required as a result of the cycle specific reload 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 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.

Attachment B

Braidwood Unit 2 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 2 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 1.860 for all core planes containing bank "D" control rods, and

2. F_{xy}^{RTP} less than or equal to 1.704 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 1.823 for all core planes containing bank "D" control rods, and

1. F_{xy}^{RTP} less than or equal to 1.704 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:

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

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

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 2 Cycle 2

Figure1

Maximum FQT*Power versus Axial Height During Normal Core Operation