Commonwealth Edison
1400 Opus Place
Downers Grove, Illinois 60515

November 5, 1992

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

SUBJECT: Cycle 4 Reload Braidwood Station Unit 1
NRC Docket No. 50-456

REFERENCES: See Attachment C

Dear Dr. Murley:

Braidwood Unit 1 has completed its third cycle of operation and has completed a refueling outage that began September 5, 1992. Braidwood Unit 1 Cycle 3 attained a final cycle burnup of approximately 17,200 MWD/MTU. Cycle 4 commenced on November 3, 1 92. This letter is to summarize Commonwealth Edison Company's (CECo) plans and evaluation regarding the Braidwood Unit 1 Cycle 4 reload core.

Attachment A describes the core reload including a summary of CECo's safety evaluation, performed in accordance with the provisions of 10CFR50.59 as there are no unreviewed safety issues or additional Technical Specification changes. This reload required a Technical Specification change which was submitted (Reference 8) and recently approved by the NRC (Reference 9).

Attachment B provides the Core Operating Limits Report (COLR) for Cycle 4 pursuant to Technical Specification 6.9.1.9. CECo and our vendor (Westinghouse) apply NRC approved reload design methodologies developed by Westinghouse as described in Reference 1. Commonwealth Edison performed the neutronic portion of the reload design using the methods and codes described in References 2 & 4 as approved in References 3 & 5, respectively. Specifically, the Braidwood Unit 1 Cycle 4 reload design, including the development of the core operating limits, was generated by Commonwealth Edison using the NRC approved methodologies.

100041

ZNLD/2278/1

Please direct any questions regarding this notification to this office.

Respectfully

Terrence W. Simpkin

T. W. Simpkin
Nuclear Licensing Administrator

Attachments

cc: R. Elliot - Project Manager, NRR
A. B. Davis - Region III Administrator
S. DuPont - NRC Resident Inspector - Braidwood

ATTACHMENT A

Braidwood Unit 1 Cycle 4 Reload Description

The Braidwood Unit 1, Cycle 4 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 re-evaluations have demonstrated that the results of the postulated events are within allowable limits.

The only exception to this was the initial results of the Boron Dilution Analysis as discussed below.

The Braidwood Unit 1 Cycle 4 core is a "Low Leakage" design. Previously, Commonwealth Edison has successfully developed and operated similar "Low Leakage" designs at Braidwood as well as our Byron and Zion stations.

During the Cycle 3/4 refueling outage, eighty-four (84) VANTAGE 5 fuel assemblies were inserted into the core. The Braidwood Unit 1 core now contains a combination of fresh and previously irradiated 17x17 VANTAGE 5 fuel assemblies (84 new and 64 once-burned), and 45 twice-burned 17x17 Optimized Fuel Assemblies (OFA's). The NRC approved the use of VANTAGE 5 at Braidwood Unit 1 for Cycle 3 operations and thereafter, under the provisions of 10CFR50.90 (Reference 6). The Braidwood/Byron UFSAR describes the compatibility of Westinghouse OFA and 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 IFBA burnable absorbers have successfully been used previously by Commonwealth Edison.

The reload VANTAGE 5 fuel assemblies 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 Mixers (IFM) grids. Similar features have been successfully utilized previously in Commonwealth Edison's Byron and Braidwood Units.

A full core of Enhanced Performance Rod Cluster Control Assemblies (EP-RCCAs) with silver-indium-cadmium (Ag-In-Cd) absorber material is being utilized with Cycle 4.

The Braidwood Unit 1 Cycle 4 core has been designed 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 has approved this request (Reference 3). Specifically, the Braidwood Unit 1 Cycle 4 reload design, including the development of the core operating limits, were generated and verified by Commonwealth Edison using NRC approved methodology.

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 were evaluated in detail in the CECo/Westinghouse reload safety evaluation process.

Commonwealth Edison has determined that all neutronic reload parameters remain within the previously established and recently revised reload safety and transient Safety Parameter Interaction List (SPIL) limits. These include, but are not limited to, SPIL items for UFSAR non-LOCA and LOCA transients and the impact of the resolution of the Westinghouse Potential Issue (PI) which addresses Boron Dilution Analysis for Braidwood/Byron (B/B) due to CVCS Malfunctions in Modes 3, 4, and 5. The Bw1C4 reload value did not meet the SPIL limit and reanalysis was required. Additionally, a potential issue regarding the Boron Dilution Protection System (BDPS) was identified which presented the following concerns:

1) The Inverse Count Rate Ratio (ICRR) data received from another utility indicated that previous data used in the design of the BDPS, was not bounding.

2) The methodology is no longer conservative with respect to the flux doubling (2) setpoint, which includes no instrument uncertainties, in defining an equivalent "trip setpoint" as presented in the Technical Specifications.

In addition, for Braidwood Unit 1 Cycle 4, the SPIL limit for Boron Dilution at Cold Shutdown was not met. The SPIL limit was re-analyzed using an increased Shutdown Margin (SDM), from 1.0% $\Delta k/k$ to 1.3% $\Delta k/k$, with a Technical Specification amendment change being required. The revised Boron Dilution safety analysis performed for the CVCS malfunction events reflects the design and Operational requirements of the BDPS. The Limiting Condition for Operation requirements for BDPS to be operable are incorporated in the Technical Specification amendment mentioned above (Reference 9). This re-analysis also considered the BDPS issues. The BDPS will continue to provide the same level of protection during a boron dilution event.

The thermal-hydraulic design for the Cycle 4 reload core has not significantly changed from that of the previously reviewed and accepted cycle design. 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.

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. CECo's Braidwood Unit 1 Cycle 4 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 Braidwood UFSAR. The Operation of the Braidwood Unit 1 Cycle 4 has been analyzed in accordance with NRC approved methodologies and satisfies safety analysis limits (having taken into consideration the revised input assumptions for the Boron Dilution analysis). The margin of safety, as defined in the bases of the Technical Specifications, is not impacted or reduced. The BDPS Technical Specification change was approved and incorporated for the Braidwood Unit 1 Cycle 4 and following cycles (Reference 9).

Finally, verification of the Braidwood Unit 1 Cycle 4 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.

In summary, CECo's use of VANTAGE 5 fuel and use of advanced neutronics methods (as described in References 7 and 2, respectively) have been previously approved by the NRC (References 6 and 3 respectively).