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ComEd

July 25, 1997

U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Document Control Desk

Subject: Byron Nuclear Power Station Unit 2
Facility Operating License NPF-66
NRC Docket No. 50-455

Braidwood Nuclear Power Station Unit 2
Facility Operating License NPF –77
NRC Docket Nos. 50-457

Request for Exemption to 10 CFR 50, Appendix J

References:
1) Letter from H.J. Miller (NRC) to C. Reed (Commonwealth Edison) transmitting Byron Inspection Reports 50-454/91022 and 50-455/91022, dated October 22, 1991.

2) Letter from H.J. Miller (NRC) to C. Reed (Commonwealth Edison) transmitting Braidwood Inspection Reports 50-456/91023 and 50-457/91021, dated January 31, 1992.

Pursuant to 10 CFR 50.12, Commonwealth Edison Company (ComEd), is requesting an exemption from 10 CFR 50, Appendix J that would allow implementing the performance-based option (Option B) for Byron Unit 2 and Braidwood Unit 2. Regulatory Guide 1.163, September 1995 provides guidance concerning a performance-based leakage program (10 CFR 50, Appendix J, Option B). The regulatory guide endorses Nuclear Energy Institute (NEI) document NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J." Section 9.2.3 of the NEI document allows Type A testing to be performed at least once per 10 years based on acceptable performance history, which is defined as completion of two consecutive periodic Type A tests. ComEd believes that the unique circumstances surrounding the classification of the tests conducted during the second refueling outage on each unit are such that application of the requirement to complete two consecutive successful tests prior to transitioning to a ten-year frequency is not necessary to meet the underlying purpose of the rule.

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For Byron Unit 2, the first Type A test following commencement of power operation was conducted in September 1990. During this test, a steam generator manway leaked. The leakage was isolated by closing the Main Steam Isolation Valve and pressurizing the secondary side of the steam generator to approximately one psig below the containment test pressure. The test was successfully completed after this leakage path was isolated. However, the test was conservatively classified as a failure in Reference 2 due to the circumstances surrounding the isolation of the leaking steam generator. The next Type A test was successfully completed in September 1993.

For Braidwood Unit 2, the first Type A test following commencement of power operation was conducted in November 1991. During this test, a steam generator manway leaked. The leakage was isolated by pressurizing the secondary side of the steam generator to approximately one psig below the containment test pressure. The test was successfully completed after this leakage path was isolated. However, the test was conservatively classified as a failure in Reference 3 due to the circumstances surrounding the isolation of the leaking steam generator. The next Type A test was successfully completed in November 1994.

Approval of this request is required prior to September 1, 1997 to provide adequate time to complete the outage scope and schedule for the next Braidwood Unit 2 refueling outage, which is scheduled to begin on September 27, 1997.

The attachment to this letter provides additional details supporting this request.

Please address any comments or questions regarding this matter to Marcia Lesniak, Nuclear Licensing Administrator, at (630) 663-6484.

Sincerely,

Levi Starley

for John B. Hosmer
Engineering Vice President

Attachment

cc: A.B. Beach, Regional Administrator - Region III
C.F. Dick, Jr., Project Manager - NRR
S. D. Burgess, Senior Resident Inspector - Byron
C. Phillips, Senior Resident Inspector - Braidwood
Office of Nuclear Facility Safety - IDNS

BACKGROUND AND JUSTIFICATION FOR EXEMPTION TO 10 CFR 50, APPENDIX J

Background

The first Byron Unit 2 Type A test following the successful pre-operational test was conducted in September 1990. During this test, a steam generator manway leaked. The leakage was isolated by closing the Main Steam Isolation Valve and pressurizing the secondary side of the steam generator to approximately one psig below the containment test pressure. Initially the test was considered successful, since the steam generator pressure was greater than containment pressure expected during post-accident conditions. The manway leak was not considered a post-accident leakage path. However, following consultation with Nuclear Reactor Regulation, Region III informed ComEd that the manway leakage should be treated as valve leakage, and the as-found test was considered a failure. The next Type A test, conducted in September 1993, was completed successfully on both an as-found and a performance basis.

The first Braidwood Unit 2 Type A test following the successful pre-operational test was conducted in November 1991. During this test, a steam generator manway leaked. This leakage was isolated by pressurizing the secondary side of the steam generator to approximately one psig below the containment test pressure. The steam generator pressure was continuously monitored to ensure it was maintained below the containment test pressure. This was done to ensure that air was not introduced into the containment during the test, thus masking other leakage. The test was then successfully completed. However, as detailed in Inspection Report 50-456/91023; 50-457/91021, instrument tolerance for the direct-reading pressure gauge utilized to maintain the steam generator pressure was not considered; therefore, the test was conservatively classified as a failure on both an as-found and performance basis. The second Type A test, conducted in November 1994, was completed successfully on both an as-found and a performance basis.

Justification for Exemption

Commonwealth Edison Company (ComEd), is requesting an exemption from 10 CFR 50, Appendix J that would allow implementing the performance-based option (Option B) for Byron Unit 2 and Braidwood Unit 2. Regulatory Guide 1.163, September 1995 provides specific guidance concerning a performance-based leakage program (10 CFR 50, Appendix J, Option B). The regulatory guide endorses Nuclear Energy Institute (NEI) document NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J." Section 9.2.3 of the NEI document allows Type A testing to be performed at least once per 10 years based on acceptable performance history, which is defined as completion of two consecutive periodic Type A tests.

10 CFR 50.12(a)(2)(ii) permits an exemption to be granted if there are special circumstances where application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule. ComEd believes that the application of the regulation in this case is not necessary to achieve the underlying purpose of the rule.

The underlying purpose of the testing requirements of 10 CFR 50, Appendix J is to assure that (a) leakage through the primary reactor containment and systems and components penetrating containment shall not exceed allowable leakage rate values as specified in the Technical Specifications or associated Bases and (b) periodic surveillance of the reactor containment penetrations and isolation valves is performed so that proper maintenance and repairs are made during the service life of the containment.

The steamlines leaving the containment are classified as General Design Criterion 57 penetrations and are, therefore, not subject to Type C local leak rate testing. This is because the secondary side of the steam generators is not considered part of the reactor coolant pressure boundary nor is it connected directly to the containment atmosphere.

The secondary manways of the steam generators are periodically removed during refueling outages to support such maintenance activities as inspections and sludge lancing. The procedures and methodology utilized to reinstall the steam generator secondary manways following these necessary maintenance activities provide high confidence that a leaktight barrier is established. At the end of each refueling outage, a walkdown of components inside containment is performed as the unit approaches the normal operating parameters of 557 °F and 2235 psig in the Reactor Coolant System. Under these conditions, the secondary side of the steam generators is pressurized to greater than 1000 psig. This walkdown identifies any manway leakage prior to commencement of power operation. Leakage would be repaired prior to returning the unit to service.

In addition, during operation, containment parameters are routinely monitored. Any leakage from secondary manways would condense and be transported to the Reactor Building floor drain sump via the floor drain system. This sump is equipped with a weir plate which is capable of detecting a one gallon per minute increase in sump input within one hour. This ensures that any unexpected input is promptly identified and assessed. This assessment would identify if leakage from the secondary side of the steam generators has developed.

For a leaking steam generator secondary side penetration to become a viable leakage path outside of containment, the steam generator secondary side must depressurize and remain below the containment pressure during an accident, and, at the same time, radioactive elements must be introduced to the containment. There are no design basis accidents or transients that produce such a combination of conditions. For the large break Loss-of-Coolant Accident (LOCA) that results in the largest release of fission products to the reactor containment, the peak containment pressure is approximately 45 psig. The steam generator secondary side pressure would have to be at or below this pressure for a viable release path to be established. The containment pressure increase is quickly mitigated and controlled by the action of the Containment Spray and Reactor Containment Fan Cooler systems. Therefore, the period of time that the containment is at a significantly elevated pressure is limited.

To decrease the steam generator pressure to less than 50 psig, the steam generators would have to be cooled from their normal operating temperature of greater than 540 °F to 281 °F. This is not considered credible because the main steamlines are isolated when the containment pressure exceeds the trip setpoint of 8.2 psi. The steam generator blowdown and main feedwater pathways are isolated upon initiation of the safety injection and resulting containment Phase A isolation signals when the containment pressure exceeds the trip setpoint of 3.4 psi. The only input to the steam generators is provided by the Auxiliary Feedwater system. This input is manually controlled to maintain the steam generators at the desired water level. Since no removal pathways would be in use, the amount of auxiliary feedwater is very limited, so significant cooldown of the secondary coolant would not occur until the containment pressure was restored to normal parameters. With the steam generator heat removal pathways isolated, the only remaining energy removal pathways are via the steam generator power-operated relief valves and safety valves. A spurious failure of a power operated relief valve can be quickly isolated via the manual isolation valve. ComEd has previously demonstrated that a stuck open power-operated relief valve can be isolated within 20 minutes of identification. This was demonstrated in support of the steam generator tube rupture analysis for Byron and Braidwood. The failure of the spring-loaded safety valves in the open position is a separately analyzed transient. The simultaneous occurrence of two analyzed transients is not considered credible.

Therefore, the leaking steam generator manway does not represent a viable containment leakage pathway that would result in doses in excess of those already analyzed.

<u>Summary</u>

ILRT testing is resource and schedule intensive. The costs associated with each test are estimated at $1.9 million. Therefore, performance of the tests on a frequency greater than that necessary to provide confidence in the integrity of the primary containment is undesirable. The preceding discussion demonstrates that the underlying purpose of the rule can be met without the performance of an ILRT during the next refueling outage for each unit, and that the health and safety of the public will continue to be protected. Additionally, the requested relief is authorized by law and no inconsistency with the common defense and security has been identified. As such, ComEd is requesting that the requested exemption request be approved.

Approval of this request is required prior to September 1, 1997 to provide adequate time to finalize the outage scope and schedule for Braidwood Unit 2 refueling outage, which is scheduled to commence on September 27, 1997.