Commonwealth Edison
1400 Opus Place
Downers Grove, Illinois 60515

November 12, 1993

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 Units 1 and 2
Request for EMERGENCY TECHNICAL SPECIFICATION AMENDMENT
Facility Operation Licenses NPF-72 and NPF-77
Technical Specification Section 4.4.5.0
NRC Docket Nos. 50-456 and 50-457

Reference: S. Berg Letter to J. Zwolinski dated November 10, 1993, transmitting Notice of Enforcement Discretion Pertaining to Braidwood Unit 1 Steam Generator Outage

Dear Dr. Murley:

Pursuant to 10CFR50.91(a)(5), Commonwealth Edison Company (CECo) proposes to amend Appendix A, Technical Specifications of Facility Operating Licenses NPF-72 and NPF-77, and requests that the Nuclear Regulatory Commission (NRC) grant an EMERGENCY amendment to Technical Specification Section 4.4.5.0 , "Steam Generator Surveillance Requirements." The amendment is needed by 1700 (CST) on November 19, 1993. Consistent with NRC guidance, a request for an NRR Notice of Enforcement Discretion for the period until this amendment can be granted was provided in the reference letter.

Technical Specification 4.4.5.0 states that,"Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5."

The proposed amendment would add a footnote to section 4.4.5.0 which addresses the October 24, 1993, Unit 1 unplanned outage which was needed to identified and repair a tube leak on the 1C Steam Generator. The steam generator was determined to be OPERABLE following completion of the inspection plan which was detailed in the reference letter. Additionally, the footnote states that the steam generator shall be demonstrated OPERABLE in accordance with Specification 4.4.5.0 prior to the initial resumption of plant operation following the Unit 1 Cycle 4 Refueling Outage.

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The attached safety analysis shows that this proposal will have minimal impact on safety because data from the inspection of the 1C Steam Generator indicated that the failed tube was an isolated event. Additionally, the data also indicated the last steam generator tube inservice inspection, performed during A1R03, still provided sufficient assurance that Unit 1 steam generators could be safely operated until the next scheduled steam generator tube inservice inspection scheduled to be performed during the next Unit 1 refueling outage.

The need for this Emergency change could not be avoided because surveillance requirements associated with Technical Specifications 3.4.5 were not written to address the Steam Generator Tube Leak which was identified on October 23, 1993. This event resulted in an elective shutdown for tube leakage less than the Technical Specification limit. Because these surveillance requirements were considered inappropriate for this situation, Braidwood developed the inspection plan described in the reference letter.

The situation was not created by a failure to make a timely application of the Technical Specification Amendment because prior to the October 23,1993 event, CECo was unaware that a condition could exist which would question the applicability of the surveillance requirements associated with Technical Specification 3.4.5.

In support of this request, the following information is attached:

Attachment A:	Detailed Description Of The Proposed Changes
Attachment B:	Revised Technical Specification Pages
Attachment C:	Evaluation of Significant Hazards Considerations
Attachment D:	Environmental Assessment

Pursuant to 10CFR50.91(b)(1) a copy of this request has been forwarded to the designated State of Illinois Official.

To the best of my knowledge and belief, the statements contained in this document are true and correct. In some respects these statements are not based on my personal knowledge, but on information furnished by other CECo employees, contractor employees, and/or consultants. Such information has been reviewed in accordance with company practice, and I believe it to be reliable.

Please address any comments or questions regarding this matter to this office.

Respectfully,

Denise M Saccomando

Denise M. Saccomando
Nuclear Licensing Administrator

Sworn & Subscribed before me on this 12th day of November 1993.

Felicia Edwards

"OFFICIAL SEAL"
Felicia Edwards
Notary Public, State of Illinois
My Commission Expires 08/03/96

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Attachments

cc: R. R. Assa, Braidwood Project Manager - NRR
S. G. Dupont, SRI - Braidwood
B. Clayton, Branch Chief - Region III
Office of Nuclear Facility Safety - IDNS

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ATTACHMENT A

DETAILED DESCRIPTION OF THE PROPOSED CHANGE

Description of the Current Requirements:

Technical Specification Surveillance Requirement (TSSR) 4.4.5.0 requires that "[e]ach steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5." TSSRs 4.4.5.1, 4.4.5.2, 4.4.5.3, 4.4.5.4, and 4.4.5.5 delineate the required augmentation of the inservice inspection program.

Bases for the Current Requirements:

The Surveillance Requirements for the inspection of the steam generator tubes ensure that the structural integrity of this portion of the Reactor Coolant System (RCS) will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice testing of steam generator tubing is essential in order to maintain surveillance conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

Description of the Need for Amending the Current Requirements:

On October 23, 1993, at 0545 hours, Area Radiation Monitors (ARM) in the vicinity of loop C main steam line reached their "alert" setpoint at Braidwood Station Unit 1. Since this is a possible indication of primary-to-secondary leakage, the Chemistry department was directed to first determine the presence of Iodine in the secondary system and secondly to calculate the primary to secondary leak rate. Chemistry analysis confirmed the presence of Iodine in the secondary system and subsequently reported a leak rate in the range of 280 - 309 gallons per day (gpd). Although this is less than the Technical Specification limit of 500 gpd, Braidwood operating management had established an administrative limit of 300 gpd. Thus, a shutdown of the unit was directed. This shutdown was completed on October 24, 1993.

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When the decision was made at 1700 hours to shutdown Unit 1 to effect repairs for the tube leak in the 1C Steam Generator, the reactor-to-secondary leakage had been calculated in the range of 280-309 gallons per day (gpd). This leakage rate, although greater than the 300 gpd administrative limit imposed by operating management set earlier in the day, was well within the allowed 500 gpd reactor-to-secondary leakage limit of Specification 3.4.6.2.c. Therefore, the assumptions of the Updated Final Safety Analysis (UFSAR), Chapter 15, Safety Analyses, remain valid and bounding.

Braidwood formed a Technical Review Team, comprised of various site and corporate technical experts, whose purpose was to determine the appropriate course of action in determining and dispositioning the leak.

Following shutdown and cooldown, the secondary side of the 1C Steam Generator was filled with water to a level above the top of the U-bend and the primary side manways were removed. Drip type leakage was observed. With approximately 100 pounds per square inch gauge (psig) of nitrogen overpressure applied to the secondary side, Tube 49/76 (Row 49, Column 76) was found to be leaking in the cold leg side. Nitrogen pressure was increased to approximately 600 psig in order to identify other leaking tubes. None were found. Secondary side water level was slowly lowered and the leak stopped at about 54-55% wide range steam generator level which indicated the leak was in the U-bend region.

Tube 49/76 was then eddy current tested using bobbin coil and a 100% through-wall indication was found in the freespan region about 10 inches above the fourth anti-vibration bar (AVB). Subsequently, a rotating pancake coil (RPC) eddy current inspection, a much more accurate but time consuming method, was performed to fully characterize the flaw. The flaw was determined to be a longitudinal crack approximately 1.3 inches long with a 5/8 inch breach that appears to be superimposed on a much smaller "ridge" approximately 18 inches long. A ridge indicated on the RPC plot is indicative of a scratch or a deposit on the outside of the tube.

Previous cycle bobbin coil eddy current test data was reviewed for a possible missed indication in the area of the flaw; none was found.

Once the flaw was identified the Technical Review Team developed a comprehensive and conservative inspection program. During the development of the inspection program that would be used to identify and repair the leaking tube, the Technical Review Team considered the applicability of Technical Specification Surveillance Requirements (TSSRs) 4.4.5.0 through 4.4.5.5. Based on the circumstances, an unplanned outage to repair a leaking tube in the 1C Steam Generator, the Technical Review Team determined that this inspection was not a scheduled inservice inspection in accordance with Specifications 4.4.5.3.a and 4.4.5.3.b. Furthermore, with the total reactor-to-secondary leakage in the 1C Steam Generator below the limit of 500 gpd in one steam generator as specified in Specification 3.4.6.2.c, the Technical Review Team determined that this inspection was also not an unscheduled inservice inspection in accordance with Specification 4.4.5.3.c since conditions 1 through 4 did not exist. Based upon these determinations, the Technical Review Team decided that TSSRs 4.4.5.0 through 4.4.5.5 did not apply to the existing condition.

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Since it had been determined that the TSSRs did not apply to existing plant conditions, the Technical Review Team developed their inspection plan with the following considerations in mind:

a. a complete inservice inspection of the 1C Steam Generator tubes was not required,

b. the 1C Steam Generator tube inspection results would not be categorized in accordance with the criteria listed on Technical Specification Page 3/4 4-14,

c. since the 1C Steam Generator tube inspection results would not be categorized, it would be unnecessary to perform any additional actions that would have been required by Table 4.4-2, and

d. the reporting of this 1C Steam Generator tube inspection would be deferred and incorporated into the reports required to be submitted following the next scheduled steam generator tube inservice inspection to be performed during A1RO4.

The program which was developed included the following actions:

a. conduct RPC eddy current testing of the tubes surrounding Tube 49/76 between the third AVB and the top support plate in the cold leg. This action was necessary to resolve the possibility of physical damage in the vicinity of the leak,

b. perform a 100% full length bobbin coil eddy current inspection on all tubes in the 1C Steam Generator. This was performed to identify other freespan indications that may be precursors to the flaw observed in Tube 49/76,

c. if other freespan indications similar in nature to the failed tube were identified, a new degradation mechanism could be indicated, and consideration would be given to inspections in the other three steam generators,

d. if no freespan indications were identified, then the failure of Tube 49/76 would be considered an isolated event, and

e. it was fully expected to find other eddy current indications (particularly at the support plates on the hot leg side) which under planned inservice inspection (ISI) conditions would be dispositioned by Commonwealth Edison Company (CECo) guidelines. For the purposes of the current situation, the Technical Review Team decided that the appropriate and prudent response to these indications would consist of:

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1) plug all clear indications of greater than or equal to 40% through-wall degradation consistent with the Technical Specification Plugging Limit of Specification 4.4.5.4, and

2) plug any distorted indications which showed abnormal growth when compared to data from previous outages.

The RPC eddy current testing of the tubes in the vicinity of the leak showed no indication similar in nature to the flaw.

The bobbin coil eddy current testing of 100% of the available tubes in the 1C Steam Generator identified 17 freespan indications. Each indication was further evaluated by RPC eddy current testing. It was determined that none of these indications were similar in nature to the flaw in Tube 49/76. Based on the results of these inspections, it was determined that the flaw in Tube 49/76 was an isolated incident. There was no reason to believe that a flaw similar in nature to that in Tube 49/76 would be identified if the other steam generators were inspected.

At the hot leg tube support plates, 116 tubes displayed indications of greater than 40% through-wall degradation, and no distorted indications displayed abnormal growth rates

A total of 117 tubes were ordered plugged. This work was completed on November 6, 1993.

Historically, the 1C Steam Generator has exhibited the greatest amount of tube degradation. Since the additional tube degradation of the 1C Steam Generator identified was consistent with the expected degradation since the last steam generator tube inservice inspection, it was decided that it was not necessary to accelerate the scheduled steam generator tube inservice inspections for the other steam generators at this time to monitor tube degradation. The Technical Review Team determined that the last steam generator tube inservice inspections, performed during the previous Unit 1 Cycle 3 Refuel Outage (A1RO3), still provided sufficient assurance that Unit 1 steam generators could be safely operated until the next scheduled steam generator tube inservice inspection scheduled to be performed during A1RO4, currently scheduled for March 5, 1994.

During discussions with the Nuclear Regulatory Commission (NRC) Staff regarding this event, it became apparent that the NRC Staff had differing views as to the applicability of TSSRs. Mr. John Zwolinski, Assistant Director for Projects - Region III, Office of Nuclear Reactor Regulation (NRR), verbally granted enforcement discretion from TSSR 4.4.5.0, portions of TSSR 4.4.5.2 and TSSR 4.4.5.5. This relief was documented in a request for an NRR Notice of Enforcement Discretion (NOED) by Letter # SVP/93-063, S. M. Berg, Jr. (CECo) to J. Zwolinski (NRC), dated November 10, 1993.

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Description of the Proposed Amendment:

The proposed amendment would add a footnote to TSSR 4.4.5.0 to reference the inspection program used to demonstrate the OPERABILITY of the 1C Steam Generator following the unplanned outage (A1F26) which began October 24, 1993, to repair a tube leak which was less than the reactor-to-secondary leakage limit of Specification 3.4.6.2c for one steam generator. The result of this inspection program will satisfy OPERABILITY requirements until the next scheduled steam generator tube inservice inspection to be performed during the Unit 1 Cycle 4 Refuel Outage (A1RO4) currently scheduled to begin March 5, 1993.

Bases for the Proposed Amendment:

The basis for this proposed amendment is to incorporate into the Braidwood Technical Specifications the relief granted verbally on November 5, 1993, by Mr. John Zwolinski, Assistant Director for Projects - Region III, Office of Nuclear Reactor Regulation (NRR). This relief was documented in a request for an NRR Notice of Enforcement Discretion (NOED) by Letter # SVP/93-063, S. M. Berg, Jr. (CECo) to J. Zwolinski (NRC), dated November 10, 1993.

Schedular Requirements:

It is requested that this proposed amendment be approved no later than November 19, 1993, in order to incorporate the relief granted verbally on November 5, 1993.