Commonwealth Edison
Braidwood Nuclear Power Station
Route #1, Box 84
Braceville, Illinois 60407
Telephone 815/458-2801

#2 5/9

April 30, 1994

Mr. William Russell, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attn: Document Control Desk

Subject: Braidwood Station Unit 1
Supplemental Information to the Request for Emergency Technical Specification Amendment
Facility Operating License: NPF-72
Technical Specification 3/4.4.5
NRC Docket No 50-456

Reference:

1) D. Saccomando letter to W. Russell dated April 25, 1994, transmitting request for Emergency Technical Specification Amendment for Specification 3/4.4.5

2) Teleconference dated April 29, 1994, between the Nuclear Regulatory Commission and Commonwealth Edison Company (CECo)

3) NRC telecopy to CECo dated April 29, 1994, transmitting Questions in Response to Braidwood April 25, 1994 Submittal

4) Draft NUREG-1477, "Voltage-Based Interim Plugging for Steam Generator Tubes-Task Group Report," June 1, 1993

Dear Mr. Russell:

Reference 1 transmitted CECo's request to process an Emergency Technical Specification Amendment to Specification 3/4.4.5 for Braidwood Unit 1. The proposed amendment modifies the Technical Specification to incorporate a 1.0 volt steam generator tube interim plugging criteria (IPC) for Cycle 5.

Through numerous teleconferences with the Nuclear Regulatory Commission (NRC) Staff, both prior to and after submittal of the Emergency Amendment request, Braidwood became aware that the NRC Staff's review of the supporting analysis for the proposed Technical Specification would not be able to be expedited in a timely manner. The enclosed attachments provides information which should facilitate the approval of this amendment request on an interim basis until the Staff can complete a comprehensive review of the amendment request.

ACD1 1/1

Attachment A addresses the use of Draft NUREG-1477, "Voltage-Based Interim Plugging for Steam Generator Tubes-Task Group Report" to determine steam line break leak rate analysis. This analysis shows that Braidwood Unit 1 can operate for approximately 3.4 Effective Full Power Months if the allowable reactor coolant Dose Equivalent Iodine-131 concentration is reduced to 0.35 microCuries/gram.

Attachment B contains Braidwood's revised Significant Hazards Consideration which incorporates the Draft NUREG-1477 steam line break leak rate analysis.

Attachment C provides Braidwood's response to question #3 which was transmitted in Reference 3. This attachment contains a discussion of Braidwood's steam generator tube leakage action plan.

Braidwood acknowledges that the proposed revision to the Technical Specification pages which were transmitted with Reference 1 will need to reflect the conditions addressed in this supplement. We propose the following footnote to Technical Specification 4.4.5.0 for your consideration.

> "For Unit 1 Cycle 5, the steam generators will be considered OPERABLE for the first _____ Effective Full Power Days of operation. During that time, reactor coolant DOSE EQUIVALENT I-131 will be limited to 0.35 microCuries per gram."

Braidwood recognizes that approval of the previously submitted IPC amendment for Unit 1 has been the subject of considerable discussion, and is aware that full resolution of this amendment request may not be immediately eminent. However, we do believe based upon our conversations with the Staff, the information provided in this supplement should help reconcile the approval of the amendment for a limited amount of time, thus facilitating the start-up of Unit 1. Further we believe that once the Staff has had adequate time to review our previously submitted package they will approve the IPC amendment for the remainder of Unit 1 Cycle 5 operation. CECo appreciates the Staff's efforts in reviewing and approving this amendment request in an expeditious manner.

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

Respectfully,

Denise M. Saccomando
Nuclear Licensing Administrator

Attachments

cc: R. Assa, Braidwood Project Manager-NRR
S. Dupont, Senior Resident Inspector-Braidwood
J. Martin, Regional Administrator-Region III
Office of Nuclear Facility Safety-IDNS

ATTACHMENT A

MAIN STEAM LINE BREAK LEAK RATE ANALYSIS

Through numerous teleconferences with the Nuclear Regulatory Commission (NRC) Staff, Braidwood recognized that the Staff would not be able to approve the emergency amendment request for IPC within an expeditious time frame. To facilitate restart of Unit 1 it was agreed that Braidwood would apply portions of Draft NUREG-1477, "Voltage-Based Interim Plugging Criteria for Steam Generator Tubes-Task Group Report," to determine the appropriate amount of time that Unit 1 could operate without exceeding the calculated primary-to-secondary leakage during main steam line break (MSLB) conditions.

Calculation of the leakage is based upon:

Use of NRC leak rate correlation database,
- This data base included data that CECo previously dispositioned as outlier data (V.C. Summer tube R28C41 with a leak rate of 2496 liters per hour).

Use of log-logistic method of analysis, and
- In discussions between CECo and the Staff during the Reference teleconference, it was mutually agreed that it was appropriate for CECo to apply the log-logistic analysis method.

Use of a Probability of Detection of 0.6.
- POD that is recommended in the Draft NUREG-1477.

Based upon the above inputs, Braidwood calculated both beginning-of-cycle (BOC) and end-of-cycle (EOC) MSLB leak rates. The most limiting steam generators were determined to be "A" and "D".

SG "A" BOC	SG "A" EOC	SG "D" BOC	SG "D" EOC
18.9 gpm	47.2 gpm	14.8 gpm	33.3 gpm

The current Technical Specification Dose Equivalent (DE) Iodine-131 (I-131) limit for reactor coolant activity is 1.0 microCurie/gram. As previously stated in Reference 1, this activity limits the allowable primary-to-secondary leakage during a MSLB in the faulted steam generator to 9.1 gpm. This value ensures Titile 10, Code of Federal Regulations, Part 100 (10 CFR 100) limits are not exceeded. Braidwood determined that a reduction in the allowable DE I-131 activity to 0.35 microCuries/gram would raise the allowable leakage rate to 26 gpm. Braidwood concluded that reduction in the allowable DE I-131 activity to 0.35 microCuries/gram is appropriate. This value will be controlled administratively and referenced in the Technical Specification.

Using the most limiting BOC leakrate value (SG "A"), a fuel cycle length of 1.15 Effective Full Power Years, and a linear estimate of leakage increase throughout the cycle, Braidwood concluded that Unit 1 "A" steam generator could be operated for approximately 3.4 Effective Full Power Months before exceeding the 26 gpm limit.

ATTACHMENT B

EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATIONS FOR PROPOSED CHANGES TO APPENDIX A TECHNICAL SPECIFICATIONS OF FACILITY OPERATING LICENSE NPF-72

Commonwealth Edison Company (CECo) has evaluated this proposed license amendment request and determined that it involves no significant hazards considerations. According to Title 10, Code of Federal Regulations, Part 50, Section 92, Paragraph c [10 CFR 50.92(c)], a proposed amendment to an operating license involves no significant hazards considerations if operation of the facility in accordance with the proposed amendment would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated; or

2. Create the possibility of a new or different kind of accident from any accident previously evaluated; or

3. Involve a significant reduction in a margin of safety.

During the Braidwood Unit 1 Cycle 4 Refuel Outage (A1R04) which began March 4, 1994, a steam generator (SG) tube inservice inspection was performed in accordance with Technical Specification Surveillance Requirement (TSSR) 4.4.5.0. The results of this inspection indicated that under the current technical specification acceptance criteria a total of 1423 SG tubes, of which 1390 are due to outside diameter stress corrosion cracking (ODSCC) at the tube support plates (TSPs), would have to be removed from service by plugging or repaired by sleeving. Additionally, the distribution of these SG tubes would cause a large disparity in the number of tubes removed from service between SGs "B" and "C." This disparity between SGs "B" and "C" would probably cause a noticeable reactor coolant system (RCS) flow imbalance and result in potential RCS loop power asymmetries. Plugging of all tubes would require re-analysis since SG "C" would exceed the currently analyzed plugging limit. Sleeving of even the minimum number of tubes necessary in SG "C" to conform with the current analysis would greatly increase the cost of SG repair and result in a significant extension of the outage critical path. This option would also limit the unit to approximately 90% of rated thermal power.

CECo proposes to amend the following Braidwood Technical Specifications:

Specification 3.4.5	REACTOR COOLANT SYSTEM-STEAM GENERATORS
Specification 3.4.6.2	REACTOR COOLANT SYSTEM-OPERATIONAL LEAKAGE

This proposed license amendment request will modify Specification 3.4.5 to allow an eddy current bobbin coil probe voltage based SG TSP interim plugging criteria (IPC) to be applied for Braidwood Unit 1 Cycle 5.

This proposed license amendment request will also modify Specification 3.4.6.2 to reduce the allowable reactor-to-secondary leakage from 1 gallon per minute (gpm) total through all SGs and 500 gallons per day (gpd) through any one SG to 600 gpd total through all SGs and 150 gpd through any one SG.

Technical Specification Bases Sections 3/4.4.5, STEAM GENERATORS, and 3/4.4.6.2, OPERATIONAL LEAKAGE, will also be modified to reflect these changes, respectively.

With the implementation of this proposed license amendment request the Braidwood Unit 1 SGs will still satisfy the requirements of Regulatory Guide (RG) 1.121, "Basis for Plugging Degraded PWR Steam Generator Tubes," Revision 0, August 1976. 964 SG tubes will remain in service that would have otherwise been removed from service by plugging or repaired by sleeving due to ODSCC at the TSPs. This represents an approximate $2.91M cost savings in SG repairs alone. This will also minimize the RCS loop asymmetries and allow the unit to return to power operation at approximately rated thermal power. Additionally, implementation of this proposed license amendment request represents the avoidance of a minimum 18 day critical path outage extension, and the associated replacement power costs. CECo believes this to be the quickest way to return Braidwood Unit 1 to power operation prior to the commencement of CECo's peak load season.

1. **The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.**

The proposed license amendment request to implement SG IPC for Braidwood Unit 1 Cycle 5 meets the requirements of RG 1.121 by demonstrating that tube leakage is acceptably low and tube burst is a highly improbable event during normal operation or a main steam line break (MSLB) event over the limited time period of this license amendment.

Under accident conditions, conservatively assuming MSLB, significant margins exist for free span burst considerations for voltage growth in excess of 95% cumulative probability. For the largest confirmed indications left in service, the projected voltage at 95% growth is 2.6 volts at the end-of-cycle (EOC) 5, compared to the 4.54 volts structural limit for a free span burst pressure of 1.43 times steam line break pressure differential. Even at 99% cumulative probability, the observed voltage growth is bounded by 2.7 volts and the structural limit is satisfied for the 1.0 volt rotating pancake coil (RPC) confirmed indications left in service.

In addition, the following analyses were done to provide assurance that there are additional margins provided by the following considerations:

- A demonstration of limited TSP displacement was done which reduces the likelihood of a tube burst. Limited TSP displacement would also reduce leakage compared to free span indications.

- The Electric Power Research Institute (EPRI) Performance Demonstration Program analyzed the performance of some 20 eddy current data analysts evaluating data from a unit with 3/4" inside diameter and 0.049" wall thickness tubes. This data demonstrated the voltage dependence of the probability of detection (POD) and argues for a POD of greater than 0.6 for ODSCC indications larger than 1.0 volt.

- CECo's risk evaluation of the operability of Braidwood Unit 1 Cycle 5 compared core damage frequency, with containment bypass, with and without the IPC applied at Braidwood Unit 1. The total Braidwood core damage frequency is estimated to be 2.74E-5 per reactor year with a total contribution from containment bypass sequences of 2.9E-8 per reactor year in the current individual plant evaluation (IPE). Operation with the requested IPC resulted in an insignificant increase in the MSLB with containment bypass sequence frequency.

- The analyses presented above applied to a full cycle of operation. Because plant operation approved by the proposed amendment would be for a significantly shorter period, the probability of an accident is much less than that calculated for a full cycle.

To support the restart and limited operation of Braidwood Unit 1, RCS Dose Equivalent (DE) Iodine 131 (I-131) will be limited to 0.35 microCuries per gram.

Therefore, since implementation of the 1.0 volt IPC for a limited time period for Braidwood Unit 1 Cycle 5 does not adversely affect SG tube integrity and results in acceptable dose consequences for a worst case postulated accident, the proposed license amendment request does not result in any increase in the probability or consequences of an accident previously evaluated within the Braidwood Updated Final Safety Analysis Report.

2. **The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.**

The proposed SG tube IPC does not introduce any significant changes to the plant design basis. Use of the criteria does not provide a mechanism which could result in an accident outside the tube support plate elevations; no ODSCC is occurring outside the thickness of the tube support plates. Neither a single or multiple tube rupture event would be expected in a steam generator in which the IPC has been applied.

CECo will implement a maximum leakage rate limit of 150 gpd through any one SG to help preclude the potential for excessive leakage during all plant conditions. The RG 1.121 criterion for establishing operational leakage rate limits that require plant shutdown are based upon leak-before-break considerations to detect a free span crack before potential tube rupture during faulted plant conditions. The 150 gpd limit will provide for leakage detection and plant shutdown in the event of the occurrence of an unexpected single crack resulting in leakage that is associated with the longest permissible free span crack length. Since tube burst due to ODSCC is precluded during normal operation due to the proximity of the TSP to the tube and the potential exists for the crevice to become uncovered during MSLB conditions, the leakage from the maximum permissible crack must preclude tube burst at MSLB conditions. Thus, the 150 gpd limit provides for plant shutdown prior to reaching critical crack lengths for MSLB conditions.

3. **The proposed change does not involve a significant reduction in a margin of safety.**

Upon implementation of the RG 1.121 criteria, even under the worst case postulated accident conditions, the occurrence of ODSCC at the TSP elevations is not expected to lead to a steam generator tube rupture event during normal or faulted plant conditions. The distribution of crack indications at the TSP elevations are confirmed to result in acceptable primary-to-secondary leakage during all plant conditions for the limited time period of this license amendment and that radiological consequences are not adversely impacted.

Loss of Coolant Accident (LOCA) coincident with (+) a Safe Shutdown Earthquake (SSE) on the SG (as required by GDC 2), may cause a tube collapse in the SGs at some plants. A number of tubes have been identified, in the "wedge" locations of the SG TSPs, to represent a potential for tube collapse during a LOCA + SSE event. These tubes have been excluded from application of the voltage based SG TSP IPC.

Addressing RG 1.83, "Inservice Inspection of PWR Steam Generator Tubes," Revision 1, July 1975, considerations, implementation of the bobbin coil probe voltage based IPC of 1.0 volt is supplemented by: enhanced eddy current inspection guidelines to provide consistency in voltage normalization, a 100% eddy current inspection sample size at the TSP elevations, and RPC inspection requirements for the larger indications left in service to characterize the principal degradation as ODSCC.

Implementation of the SG TSP IPC will decrease the number of tubes which must be repaired. Therefore, the margin of flow that would otherwise be reduced in the event of increased tube plugging would be maintained.

Therefore, based on the evaluation above, CECo has concluded that this proposed license amendment request does not involve a significant hazards consideration.

ATTACHMENT C

RESPONSE TO NRC QUESTION 3
FROM APRIL 29, 1994 TELECOPY

Question 3. The application proposes that the 150 gpd leakage limit be incorporated into TS, and that operator action be administratively required for a leak rate increase in a range of 25 to 100 gpd in 1 hour (with an added condition of leakage above 50 gpd). Explain this apparent change from the leak rate increase limit of 25 gpd per hr instituted in December 1993.

Answer 3. During Braidwood Unit 1 Cycle 4, reactor coolant system (RCS) iodine activity was elevated due to leaking fuel. In October 1993 when the leak occurred on the 1C Steam Generator (SG), Braidwood was able to detect a 25 gpd in 1 hour change in leak rate because of the initial elevated RCS iodine activity levels.

After the 1C SG tube leak event and in consultation with Commonwealth Edison's Byron and Zion Stations, the 50 gpd threshold was added. It was determined that during fuel cycles in which no leaking fuel is present, the RCS iodine activity will be less than that which was present during the 1C SG tube leak event. With less iodine activity, the ability to detect and accurately quantify a primary-to-secondary leak is reduced. Although a leak rate of less than 50 gpd can be measured, the accurate repeatability of this leak rate determination will not be possible due to analytical limitations. Measured leak rate changes of greater than 25 gpd within one hour may be observed when the actual leak rate has not changed by 25 gpd if the leakage is less than 50 gpd. The 50 gpd threshold was based on no leaking fuel and the ability to accurately detect a 25 gpd change within one hour.

The administrative limits regarding changing leak rates are detailed in Braidwood Operating Abnormal Procedure (BWOA) SEC-8, STEAM GENERATOR TUBE LEAK, and are summarized as follows:

If the leak rate is greater than 50 gpd and

1. it increases by greater than or equal to 25 gpd but less than 100 gpd in a one hour period, then the unit will be shut down within five hours; or

2. it increases by greater than or equal to 100 gpd in a one hour period, then the unit will be shut down within four hours.