

ATTACHMENT C

PROPOSED CHANGES TO APPENDIX A TECHNICAL SPECIFICATIONS OF FACILITY OPERATING LICENSE NPF-72

BRAIDWOOD STATION UNIT 1 REVISED PAGES:

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*THESE PAGES HAVE NO CHANGES BUT ARE INCLUDED FOR CONTINUITY.

REACTOR COOLANT SYSTEM

3/4.4.5 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more steam generators inoperable, restore the inoperable steam generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.

SURVEILLANCE REQUIREMENTS

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

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.5.2 Steam Generator Tube* Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. When applying the expectations of 4.4.5.2.a through 4.4.5.2.c, previous defects or imperfections in the area repaired by the sleeve are not considered an area requiring reinspection. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

- a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas;
- b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

*When referring to a steam generator tube, the sleeve shall be considered a part of the tube if the tube has been repaired per Specification 4.4.5.4.a.10.

SURVEILLANCE REQUIREMENTS (Continued)

- 1) All tubes that previously had detectable tube wall penetrations greater than 20% that have not been plugged or sleeved in the affected area, and all tubes that previously had detectable sleeve wall penetrations that have not been plugged,
 - 2) Tubes in those areas where experience has indicated potential problems,
 - 3) At least 3% of the total number of sleeved tubes in all four steam generators or all of the sleeved tubes in the generator chosen for the inspection program, whichever is less. These inspections will include both the tube and the sleeve, and
 - 4) A tube inspection (pursuant to Specification 4.4.5.4a.8) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.
- c. The tubes selected as the second and third samples (if required by Table 4.4-2) during each inservice inspection may be subjected to a partial tube inspection provided:
- 1) The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found, and
 - 2) The inspections include those portions of the tubes where imperfections were previously found.

The results of each sample inspection shall be classified into one of the following three categories:

<u>Category</u>	<u>Inspection Results</u>
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes or sleeves must exhibit significant (greater than 10% of wall thickness) further wall penetrations to be included in the above percentage calculations.

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- d. For Unit 1 Cycle 5, implementation of the tube support plate interim plugging criteria limit requires a 100% bobbin coil probe inspection for all hot leg tube support plate intersections and all cold leg intersections down to the lowest cold leg tube support plate with outer diameter stress corrosion cracking (ODSCC) indications. An inspection using a rotating pancake coil (RPC) probe is required in order to show OPERABILITY of tubes with flaw-like bobbin coil signal amplitudes greater than 1.0 volt but less than or equal to 2.7 volts. For tubes that will be administratively plugged or repaired, no RPC inspection is required. The RPC results are to be evaluated to establish that the principal indications can be characterized as ODSCC.

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

- a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections, not including the pre-service inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months;
- b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 at 40-month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.5.3a.; the interval may then be extended to a maximum of once per 40 months; and
- c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:
 - 1) Reactor-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2c., or
 - 2) A seismic occurrence greater than the Operating Basis Earthquake, or
 - 3) A Condition IV loss-of-coolant accident requiring actuation of the Engineered Safety Features, or
 - 4) A Condition IV main steam line or feedwater line break.

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.4 Acceptance Criteria

a. As used in this specification:

- 1) Imperfection means an exception to the dimensions, finish or contour of a tube or sleeve from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube or sleeve wall thickness, if detectable, may be considered as imperfections;
- 2) Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube or sleeve;
- 3) Degraded Tube means a tube or sleeve containing unrepaired imperfections greater than or equal to 20% of the nominal tube or sleeve wall thickness caused by degradation;
- 4) % Degradation means the percentage of the tube or sleeve wall thickness affected or removed by degradation;
- 5) Defect means an imperfection of such severity that it exceeds the plugging or repair limit. A tube or sleeve containing an unrepaired defect is defective;
- 6) Plugging or Repair Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging or repaired by sleeving in the affected area. The plugging or repair limit imperfection depth is equal to 40% of the nominal wall thickness. Add Insert B
- 7) Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 4.4.5.3c., above;
- 8) Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg. For a tube that has been repaired by sleeving, the tube inspection shall include the sleeved portion of the tube, and

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For Unit 1 Cycle 5, this definition does not apply to the region of the tube subject to the tube support plate interim plugging criteria limit, i.e., the tube support plate intersections. Specification 4.4.5.4.a.11 describes the repair limit for use within the tube support plate intersection of the tube;

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

9) Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

10) Tube Repair refers to a process that reestablishes tube serviceability. Acceptable tube repairs will be performed by the following processes:

- a) Laser welded sleeving as described by Westinghouse report WCAP-13698, Rev. 1, or
- b) Kinetic welded sleeving as described by Babcock & Wilcox Topical Report BAW-2045PA, Rev. 1.

Tube repair includes the removal of plugs that were previously installed as a corrective or preventative measure. A tube inspection per 4.4.5.4.a.8 is required prior to returning previously plugged tubes to service.

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- b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug or repair in the affected area all tubes exceeding the plugging or repair limit) required by Table 4.4-2.

4.4.5.5 Reports

- a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged or repaired in each steam generator shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2;
- b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following the completion of the inspection. This Special Report shall include:
 - 1) Number and extent of tubes inspected,
 - 2) Location and percent of wall-thickness penetration for each indication of an imperfection, and
 - 3) Identification of tubes plugged or repaired.
- c. Results of steam generator tube inspections which fall into Category C-3 shall be reported in a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days and prior to resumption of plant operation. This report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

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- 11) Tube Support Plate Interim Plugging Criteria Limit is used for the disposition of a steam generator tube for continued service that is experiencing ODSCC confined within the thickness of the tube support plates. For application of the tube support plate interim plugging criteria limit, the tube's disposition for continued service will be based upon standard bobbin coil probe signal amplitude of flaw-like indications. The plant specific guidelines used for all inspections shall be amended, as appropriate, with respect to the voltage/depth parameters specified in Specification 4.4.5.2. Pending incorporation of the voltage verification requirements in ASME standard verifications, an ASME standard calibrated against the laboratory standard will be utilized in Unit 1 steam generator inspections for consistent voltage normalization.
1. A tube can remain in service with a flaw-like bobbin coil signal amplitude of less than or equal to 1.0 volt, regardless of the depth of the tube wall penetration, if, as a result, the projected end of cycle distribution of crack indications is verified to result in total primary to secondary leakage less than 9.4 gpm (includes operational and accident leakage). The basis for determining expected leak rates from the projected crack distribution is provided in Westinghouse letter report NSD-TAP-3069, "Braidwood 1: Technical Support for Cycle 5 S/G Interim Plugging Criteria, Pre-WCAP Release," dated April 21, 1994.
 2. A tube can remain in service with a flaw-like bobbin coil signal amplitude greater than 1.0 volt but less than or equal to 2.7 volts provided an RPC inspection does not detect degradation.
 3. A tube with a flaw-like bobbin coil signal amplitude of greater than 2.7 volts shall be plugged or repaired.

Certain tubes identified in Westinghouse letter report NSD-TAP-3069, "Braidwood 1: Technical Support for Cycle 5 S/G Interim Plugging Criteria, Pre-WCAP Release," dated April 21, 1994, shall be excluded from application of the tube support plate interim plugging criteria limit. It has been determined that these tubes may collapse or deform following a postulated LOCA + SSE.

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d. For Unit 1 Cycle 5, the results of inspection for all tubes in which the tube support plate interim plugging criteria limit has been applied shall be reported to the Commission pursuant to Specification 6.9.2 following completion of the steam generator tube inservice inspection and prior to Cycle 5 operation. The report shall include:

1. Listing of the applicable tubes,
2. Location (applicable intersections per tube) and extent of degradation (voltage), and
3. Projected Steam Line Break (MSLB) Leakage.

TABLE 4.4-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE
INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	Yes
No. of Steam Generators per Unit	Four
First Inservice Inspection	Two
Second & Subsequent Inservice Inspections	One ¹

TABLE NOTATION

1. The inservice inspection may be limited to one steam generator on a rotating schedule encompassing $3 N \%$ of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions. Each of the other two steam generators not inspected during the first inservice inspections shall be inspected during the second and third inspections. The fourth and subsequent inspections shall follow the instructions described above.

TABLE 4.4-2

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S. G.	C-1	None	N.A.	N.A.	N.A.	N.A.
	C-2	Plug or repair defective tubes and inspect additional 2S tubes in this S. G.	C-1	None	N.A.	N.A.
			C-2	Plug or repair defective tubes and inspect additional 4S tubes in this S. G.	C-1	None
					C-2	Plug or repair defective tubes
			C-3	Perform action for C-3 result of first sample	N.A.	Perform action for C-3 result of first sample
	C-3	Perform action for C-3 result of first sample	N.A.	N.A.		
	C-3	Inspect all tubes in this S. G., plug or repair defective tubes and inspect 2S tubes in each other S. G. Notification to NRC pursuant to §50.72 (b)(2) of 10 CFR Part 50	All other S. G.s are C-1	None	N.A.	N.A.
			Some S. G.s C-2 but no additional S. G. are C-3	Perform action for C-2 result of second sample	N.A.	N.A.
Additional S. G. is C-3			Inspect all tubes in each S. G. and plug or repair defective tubes. Notification to NRC pursuant to §50.72(b)(2) of 10 CFR Part 50	N.A.	N.A.	

$S = 3 \frac{N}{n}$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

- a. No PRESSURE BOUNDARY LEAKAGE,
- b. 1 gpm UNIDENTIFIED LEAKAGE,
- c. ~~1 gpm~~ ^{600 gallons per day} total reactor-to-secondary leakage through all steam generators not isolated from the Reactor Coolant System and ~~500~~ ¹⁵⁰ gallons per day through any one steam generator,
- d. 10 gpm IDENTIFIED LEAKAGE from the Reactor Coolant System,
- e. 40 gpm CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 ± 20 psig, and
- f. 1 gpm leakage at a Reactor Coolant System pressure of 2235 ± 20 psig from any Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1.*

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

- a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.
- b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE and leakage from Reactor Coolant System Pressure Isolation Valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.
- c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the above limit, reduce the leakage rate to within limits within 4 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

*Test pressures less than 2235 psig but greater than 350 psig are allowed. Observed leakage shall be adjusted for the actual test pressure up to 2235 psig assuming the leakage to be directly proportional to pressure differential to the one-half power.

REACTOR COOLANT SYSTEM

BASES

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the 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 inspection of steam generator tubing is essential in order to maintain surveillance of the 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.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the Reactor Coolant System and the Secondary Coolant System (reactor-to-secondary leakage ~~500~~ gallons per day per steam generator). Cracks having a reactor-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that reactor-to-secondary leakage of ~~500~~ gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged or repaired by sleeving. The technical bases for sleeving are described in Westinghouse report WCAP-13698 Rev. 1 and Babcock & Wilcox Topical Report BAW-2045PA Rev. 1.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging or sleeving will be required for all tubes with imperfections exceeding the plugging or repair limit of 40% of the tube nominal wall thickness. If a sleeved tube is found to contain a through wall penetration in the sleeve of equal to or greater than 40% of the nominal wall thickness, the tube must be plugged. The 40% plugging limit for the sleeve is derived from Reg. Guide 1.121 analysis and utilizes a 20% allowance for eddy current uncertainty and additional degradation growth. Inservice inspection of sleeves is required to ensure RCS integrity. Sleeve inspection techniques are described in Westinghouse Report WCAP-13698 Rev. 1 and Babcock & Wilcox Topical Report BAW-2045PA Rev. 1. Steam Generator tube and sleeve inspections have demonstrated the capability to reliably detect degradation that has penetrated 20% of the pressure retaining portions of the tube or sleeve wall thickness. Commonwealth Edison will validate the adequacy of any system that is used for periodic inservice inspection of the sleeves and, as deemed appropriate, will upgrade testing methods as better methods are developed and validated for commercial use.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be reported to the Commission pursuant to Specification 6.9.2 prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

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For Unit 1 Cycle 5, tubes experiencing outer diameter stress corrosion cracking within the thickness of the tube support plates will be dispositioned in accordance with Specification 4.4.5.4.a.11.

REACTOR COOLANT SYSTEM

BASES

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS Leakage Detection Systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. These Detection Systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.6.2 OPERATIONAL LEAKAGE

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 gpm. This threshold value is sufficiently low to ensure early detection of additional leakage.

The total steam generator tube leakage limit of ~~1 gpm~~^{600 gpd} for all steam generators not isolated from the RCS ensures that the dosage contribution from the tube leakage will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of either a steam generator tube rupture or steam line break. The ~~1 gpm~~ limit is consistent with the assumptions used in the analysis of these accidents. The ~~500 gpd~~¹⁵⁰ leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

The 10 gpm IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the Leakage Detection Systems.

The CONTROLLED LEAKAGE limitation restricts operation when the total flow supplied to the reactor coolant pump seals exceeds 40 gpm with the modulating valve in the supply line fully open at a nominal RCS pressure of 2235 psig. This limitation ensures that in the event of a LOCA, the Safety Injection flow will not be less than assumed in the safety analyses.

The 1 gpm leakage from any RCS pressure isolation valve is sufficiently low to ensure early detection of possible in-series check valve failure. It is apparent that when pressure isolation is provided by two in-series check valves and when failure of one valve in the pair can go undetected for a substantial length of time, verification of valve integrity is required. Since these valves are important in preventing overpressurization and rupture of the ECCS low pressure piping which could result in a LOCA that bypasses containment, those valves should be tested periodically to ensure low-probability of gross failure.

ATTACHMENT D

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 TSP 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 as well as a main steam line break (MSLB) event.

Tube burst is precluded during normal operating plant conditions since the tube support plates are adjacent to the degraded regions of the tube in the tube to tube support plate crevices.

Under accident conditions, conservatively assuming MSLB leakage is free span leakage, significant margins exist for free span burst considerations for voltage growth in excess of 95% cumulative probability. For the largest rotating pancake coil (RPC) probe confirmed indications left in service, the projected end-of-cycle (EOC) 5 voltage at 95% growth is 2.8 volts compared to the 4.54 volts structural limit for free span burst of 1.43 times steam line break pressure differential. Even at 99% cumulative probability, the voltage growth is bounded by 2.7 volts and the structural limit is satisfied for the 1.0 volt RPC confirmed indications left in service.

The leakage assessment during a MSLB event on the worst SG results in a maximum anticipated leak rate of 3.0 gpm. This is more than a factor of 3 lower than the allowable 9.1 gpm primary-to-secondary leak rate limit with containment bypass during a MSLB.

In addition, the following analyses were done to demonstrate even larger margins:

LIMITED TUBE SUPPORT PLATE DISPLACEMENT

A demonstration of limited TSP displacement was done to further reduce the likelihood of a tube burst to negligible levels. Limited TSP displacement would reduce leakage compared to free span indications.

PROBABILITY OF DETECTION

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 clearly demonstrated the voltage dependence of the POD and argues for a POD of > 0.6 for ODSCC indications larger than 1.0 volt.

RISK EVALUATION OF CORE DAMAGE

As part of CECO's evaluation of the operability of Braidwood Unit 1 Cycle 5, a risk evaluation was completed. The objective of this evaluation was to compare core damage frequency, with containment bypass, with and without the interim plugging criteria applied at Braidwood Unit 1.

The total Braidwood core damage frequency is estimated to be $2.74\text{E}-5$ per reactor year with a total contribution from containment bypass sequences of $2.9\text{E}-8$ 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.

Therefore, as implementation of the 1.0 volt IPC for Braidwood Unit 1 Cycle 5 does not adversely affect steam generator tube integrity and results in acceptable dose consequences, 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.

Implementation of the proposed steam generator tube interim plugging criteria 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 plugging criteria 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 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.

As steam generator tube integrity upon implementation of the 1.0 volt IPC continues to be maintained through inservice inspection and primary-to-secondary leakage monitoring, the possibility of a new or different kind of accident from any previously evaluated is not created.

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

The use of the voltage based bobbin coil probe SG TSP IPC for Braidwood Unit 1 Cycle 5 is demonstrated to maintain steam generator tube integrity commensurate with the criteria of RG 1.121. Upon implementation of the criteria, even under the worst case 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 EOC 5 distribution of crack indications at the TSP elevations are confirmed to result in acceptable primary-to-secondary leakage during all plant conditions and that radiological consequences are not adversely impacted.

In addressing the combined effects of Loss of Coolant Accident (LOCA) coincident with a Safe shutdown Earthquake (SSE) on the SG (as required by GDC 2), it has been determined that tube collapse may occur in the SGs at some plants.

There are two issues associated with SG tube collapse. First, the collapse of SG tubing reduces the RCS flow area through the tubes. The reduction in flow area increases the resistance to flow of steam from the core during a LOCA which, in turn, may potentially increase Peak Clad Temperature (PCT). Second, there is a potential that partial through-wall cracks in tubes could progress to through-wall cracks during tube deformation or collapse.

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 interim tube plugging criteria 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 tube support plate elevations, and RPC inspection requirements for the larger indications left inservice to characterize the principal degradation as ODSCC.

As noted previously, implementation of the SG TSP IPC will decrease the number of tubes which must be repaired. The installation of SG tube plugs reduces the RCS flow margin. Thus, implementation of the SG TSP IPC will maintain the margin of flow that would otherwise be reduced in the event of increased tube plugging. Therefore, there will not be a significant reduction in the margin of safety as a result of the implementation of this proposed license amendment request.

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

ATTACHMENT E

ENVIRONMENTAL ASSESSMENT 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 against the criteria for and identification of licensing and regulatory actions requiring environmental assessment in accordance with Title 10, Code of Federal Regulations, Part 51, Section 21 (10 CFR 51.21). CECo has determined that this proposed license amendment request meets the criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(9). This determination is based upon the following:

1. The proposed licensing action involves the issuance of an amendment to a license for a reactor pursuant to 10 CFR 50 which changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or which changes and inspection or a surveillance requirement. This proposed license amendment request changes the surveillance requirements for the Braidwood Unit 1 steam generator (SG) tube inservice inspection program and reduces the allowable reactor-to-secondary leakage from 1 gallon per minute 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;
2. this proposed license amendment request involves no significant hazards considerations;
3. there is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite; and
4. there is no significant increase in individual or cumulative occupational radiation exposure.

Therefore, pursuant to 10 CFR 51.22(b), neither an environmental impact statement nor an environmental assessment is necessary for this proposed license amendment request.

ATTACHMENT F

Westinghouse Letter Report NSD-TAP-3069,
"Braidwood 1: Technical Support for Cycle 5 S/G Interim Plugging
Criteria, Pre-WCAP Release,"
dated April 21, 1994