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 leas 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.

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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.

Insert A

6 ~~5~~) For Unit 1, tubes which remain in service due to the application of the F^* criteria will be inspected, in the tubesheet region, during all future outages.

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.

Insert B

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.~~

e. A random sample of at least 20 percent of the total number of sleeves shall be inspected for axial and circumferential indications at the end of each cycle. In the event that an imperfection of 40 percent or greater depth is detected, an additional 20 percent of the unsampled sleeves shall be inspected, and if an imperfection of 40 percent or greater depth is detected in the second sample, all remaining sleeves shall be inspected. These inservice inspections will include the entire sleeve and the tube at

Insert A
(4.4.5.2.b)

5) For Unit 1, indications left in service as a result of application of the tube support plate voltage-based repair criteria shall be inspected by bobbin coil probe during all future refueling outages.

Insert B
(4.4.5.2)

d. For Unit 1, implementation of the steam generator tube/tube support plate repair criteria requires a 100-percent bobbin coil inspection for hot-leg and cold-leg tube support plate intersections down to the lowest cold-leg tube support plate with known outside diameter stress corrosion cracking (ODSCC) indications. The determination of the lowest cold-leg tube support plate intersections having ODSCC indications shall be based on the performance of at least a 20-percent random sampling of tubes inspected over their full length.

SURVEILLANCE REQUIREMENTS (Continued)

the heat treated area. The inservice inspection for the sleeves is required until the corrosion resistance for the laser welded or kinetically welded joints in tubes that bound the material parameters of the tubes installed in the steam generators has been demonstrated acceptable. If conformance with the acceptable criteria of Specification 4.4.5.4 for tube structural integrity is not confirmed, the tubes containing the sleeves in question shall be removed from service.

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

Category	Inspection Results
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.

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 preservice 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.

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. For Unit 1, this definition does not apply to defects in the tubesheet that meet the criteria for an F* tube~~,~~. ~~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;~~

Replace with Insert C

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

Insert C
(4.4.5.4.a.6)

For Unit 1, this definition does not apply to tube support plate intersections for which the voltage-based repair criteria are being applied. Refer to 4.4.5.4.a.11 for the repair limit applicable to these intersections;

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.

Replace with Insert D

~~11) Tube Support Plate Interim Plugging Criteria Limit for Unit 1 Cycle 5 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 consistent with the eddy current guidelines in Appendix A of WCAP-13854 as appropriate to accommodate the additional information needed to evaluate tube support plate signals with respect to the voltage parameters as 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, provided Item 3 below is satisfied.~~

~~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 and provided Item 3 below is satisfied.~~

3. The projected end of cycle distribution of crack indications is verified to result in total primary to secondary leakage less than 9.1 gpm (includes operational and accident leakage). The basis for determining expected leak rates from the projected crack distribution is provided in WCAP-14046, "Braidwood Unit 1 Technical Support for Cycle 5 Steam Generator Interim Plugging Criteria" dated May 1994.

4. 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 WCAP-14046, "Braidwood Unit 1 Technical Support for Cycle 5 Steam Generator Interim Plugging Criteria," dated May 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.

12) F^* Distance is the distance into the tubesheet from the secondary face of the tubesheet or the top of the last hardroll, whichever is further into the tubesheet, that has been determined to be 1.7 inches.

13) F^* Tube is a Unit 1 steam generator tube with degradation below the F^* distance and has no indications of degradation (i.e., no indication of cracking) within the F^* distance. Defects contained in an F^* tube are not dependant on flaw geometry.

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.

Insert D
(4.4.5.4.a)

11) For Unit 1, the Tube Support Plate Plugging Limit is used for the disposition of an alloy 600 steam generator tube for continued service that is experiencing predominantly axially oriented outside diameter stress corrosion cracking confined within the thickness of the tube support plates. At tube support plate intersections, the plugging (repair) limit is based on maintaining steam generator tube serviceability as described below:

a. Steam generator tubes, whose degradation is attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with bobbin voltages less than or equal to the lower voltage repair limit [Note 1], will be allowed to remain in service.

b. Steam generator tubes whose degradation is attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than the lower voltage repair limit [Note 1], will be repaired or plugged, except as noted in 4.4.5.4.a.11.c below.

c. Steam generator tubes, with indications of potential degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than the lower voltage repair limit [Note 1] but less than or equal to the upper voltage repair limit [Note 2], may remain in service if a rotating pancake coil inspection does not detect degradation. Steam generator tubes, with indication of outside diameter stress corrosion cracking degradation with a bobbin voltage greater than the upper voltage repair limit [Note 2] will be plugged or repaired.

d. Certain intersections as identified in WCAP-14046, Section 4.7, will be excluded from application of the voltage-based repair criteria as it is determined that these intersections may collapse or deform following a postulated LOCA + SSE event.

Insert D
(continued)

e. If an unscheduled mid-cycle inspection is performed, the following mid-cycle repair limits apply instead of the limits identified in 4.4.5.4.a.11.a, 4.4.5.4.a.11.b, and 4.4.5.4.a.11.c. The mid-cycle repair limits are determined from the following equations:

$$V_{MURL} = \frac{V_{SL}}{1.0 + NDE + Gr\left(\frac{CL - \Delta t}{CL}\right)}$$

$$V_{MLRL} = V_{MURL} - (V_{URL} - V_{LRL})\left(\frac{CL - \Delta t}{CL}\right)$$

Where:

V_{URL}	=	upper voltage repair limit
V_{LRL}	=	lower voltage repair limit
V_{MURL}	=	mid-cycle upper voltage repair limit based on time into cycle
V_{MLRL}	=	mid-cycle lower voltage repair limit based on V_{MURL} and time into cycle
Δt	=	length of time since last scheduled inspection during which V_{URL} and V_{LRL} were implemented.
CL	=	cycle length (the time between two scheduled steam generator inspections)
V_{SL}	=	structural limit voltage
Gr	=	average growth rate per cycle length
NDE	=	95-percent cumulative probability allowance for nondestructive examination uncertainty (i.e., a value of 20-percent has been approved by the NRC)

Implementation of these mid-cycle repair limits should follow the same approach as in TS 4.4.5.4.a.11.a, 4.4.5.4.a.11.b, and 4.4.5.4.a.11.c.

Insert D
(continued)

Note 1: The lower voltage repair limit is 1.0 volt for 3/4-inch diameter tubing.

Note 2: The upper voltage repair limit is calculated according to the methodology in Generic Letter 95-05, "Voltage Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking," as supplemented.

SURVEILLANCE REQUIREMENTS (Continued)

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.

Replace with Insert E

~~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.~~

e. The results of inspections of F* Tubes shall be reported to the Commission prior to the resumption of plant operation. The report shall include:

1) Identification of F* Tubes, and

2) Location and size of the degradation.

Insert E
(4.4.5.5)

d. For implementation of the voltage based repair criteria to tube support plate intersections for Unit 1, notify the staff prior to returning the steam generators to service should any of the following conditions arise:

1. If estimated leakage based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds the leak limit (determined from the licensing basis dose calculation for the postulated main steam line break) for the next operating cycle.

2. If circumferential crack-like indications are detected at the tube support plate intersections.

3. If indications are identified that extend beyond the confines of the tube support plate.

4. If indications are identified at the tube support plate elevations that are attributable to primary water stress corrosion cracking.

5. If the calculated conditional burst probability based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds 1×10^{-2}, notify the NRC and provide an assessment of the safety significance of the occurrence.

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
			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.

N Where N is the number of steam generators in the unit, and n is the number of steam

3/4.4.8 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.8 The specific activity of the reactor coolant shall be limited to:

a. Less than or equal to 1 microCurie per gram DOSE EQUIVALENT I-131, and**

b. Less than or equal to $100/\bar{E}$ microCuries per gram of gross radioactivity.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1, 2 and 3*:

a. With the specific activity of the reactor coolant greater than 1 microCurie per gram DOSE EQUIVALENT I-131** for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours; and

b. With the specific activity of the reactor coolant greater than $100/\bar{E}$ microCuries per gram, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours.

*With T_{avg} greater than or equal to 500°F.

**For Unit 1 ~~Cycle 5,~~ reactor coolant DOSE EQUIVALENT I-131 will be limited to 0.35 microCuries per gram.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

MODES 1, 2, 3, 4, and 5:

With the specific activity of the reactor coolant greater than 1 microCurie per gram DOSE EQUIVALENT I-131* or greater than 100/Ē microCuries per gram, perform the sampling and analysis requirements of Item 4.a) of Table 4.4-4 until the specific activity of the reactor coolant is restored to within its limits.

SURVEILLANCE REQUIREMENTS

4.4.8 The specific activity of the reactor coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.

* For Unit 1, reactor coolant DOSE EQUIVALENT I-131 will be limited to 0.35 microCuries per gram.

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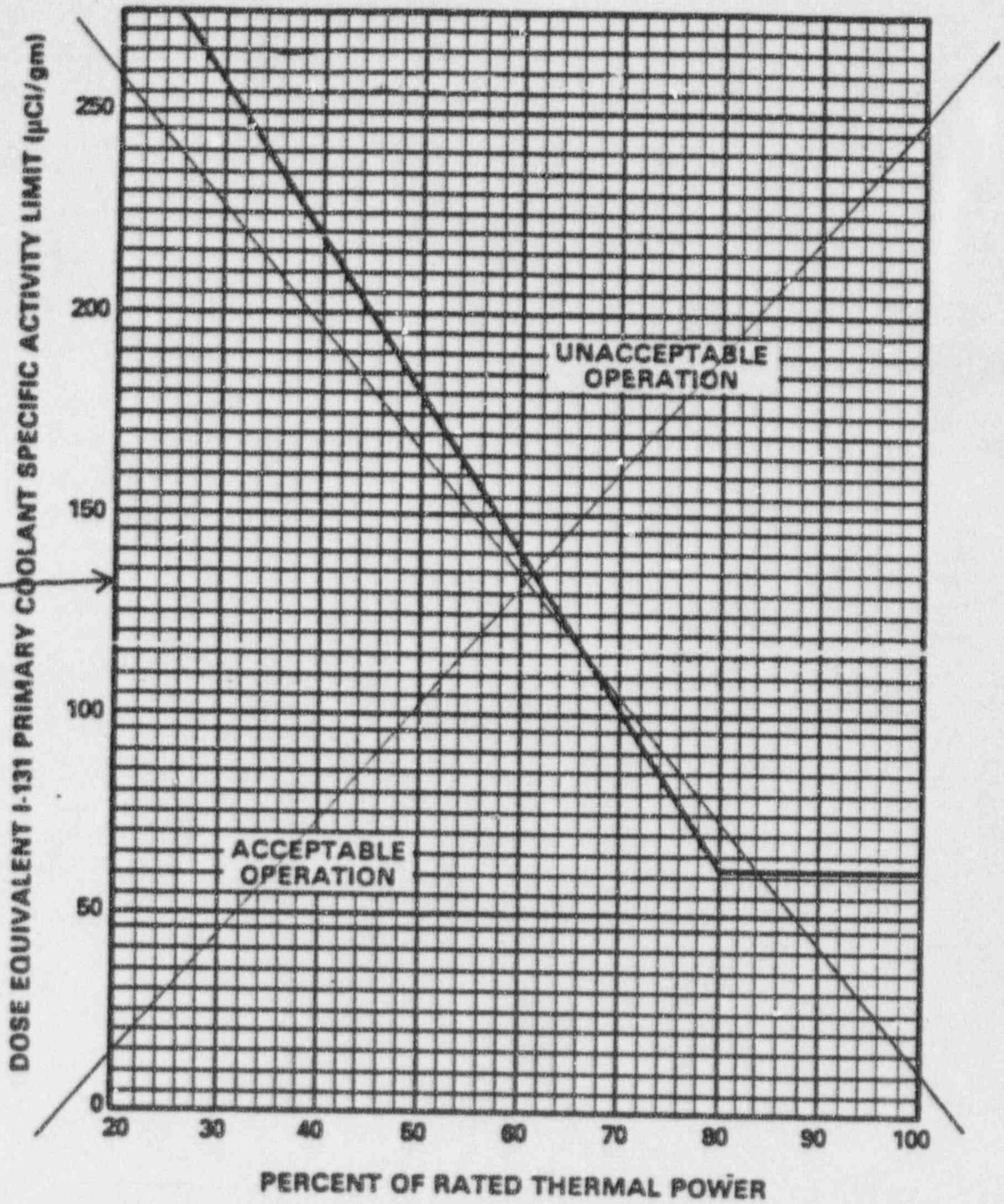

FIGURE 3.4-1

DOSE EQUIVALENT I-131 REACTOR COOLANT SPECIFIC ACTIVITY LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE REACTOR COOLANT SPECIFIC ACTIVITY > 1µCi/GRAM DOSE EQUIVALENT I-131*

* For Unit 1, REACTOR COOLANT SPECIFIC ACTIVITY > 0.35 µCi/GRAM DOSE EQUIVALENT I-131

AMENDMENT No.

Insert G
(Figure 3.4-1)

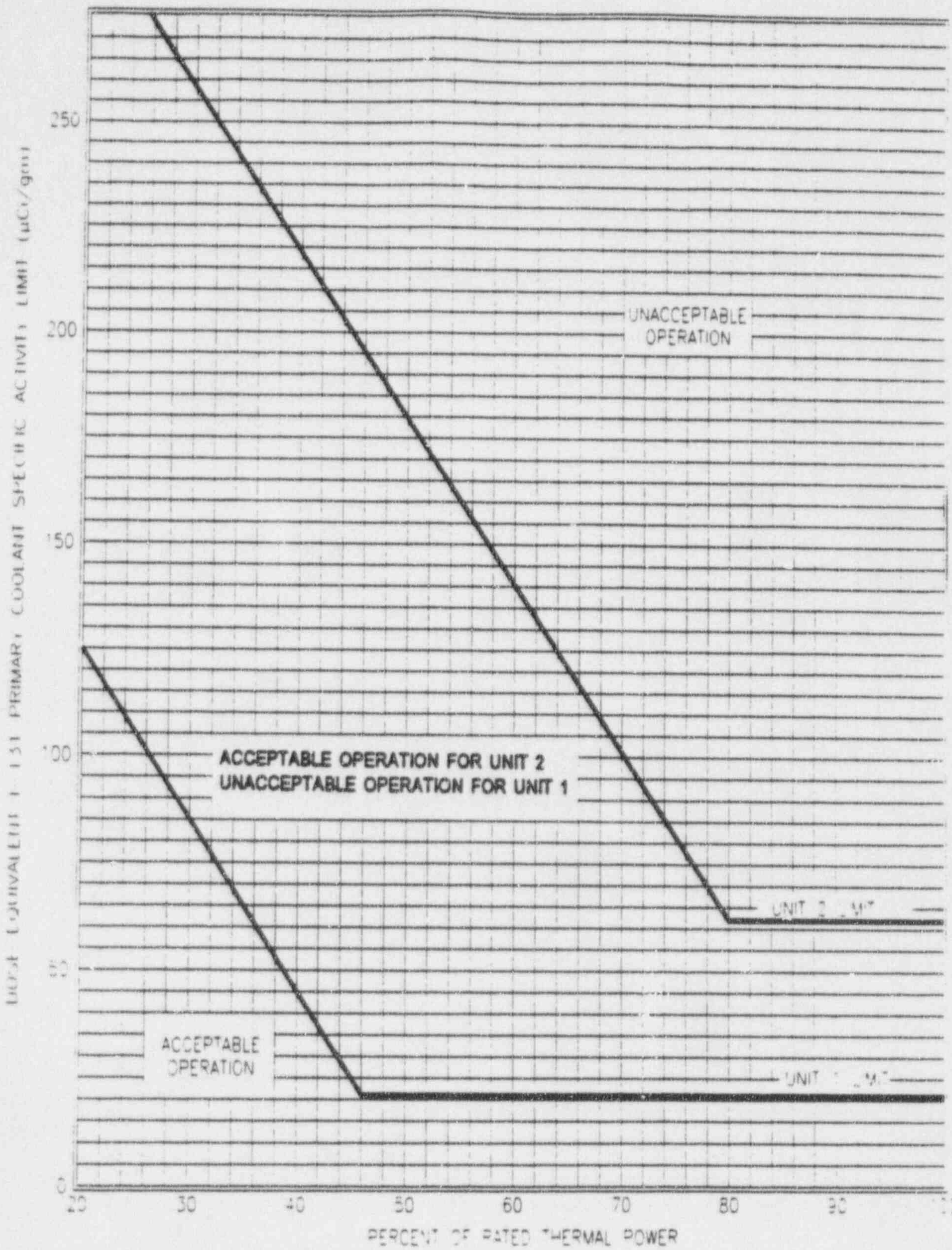

TABLE 4.4-4

REACTOR COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. Gross Radioactivity Determination**	At least once per 72 hours	1, 2, 3, 4
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	Once per 14 days	1
3. Radiochemical for $\bar{E}$ Determination***	Once per 6 months*	1
4. Isotopic Analysis for Iodine Including I-131, I-133, and I-135	a) Once per 4 hours, whenever the specific activity exceeds 1 µCi/gram DOSE EQUIVALENT I-131**** or 100/$\bar{E}$ µCi/gram of gross radioactivity, and	1#, 2#, 3#, 4#, 5#
	b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15% of the RATED THERMAL POWER within a 1-hour period.	1, 2, 3

TABLE 4.4-4 (Continued)

TABLE NOTATIONS

#Until the specific activity of the Reactor Coolant System is restored within its limits.

*Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.

**A gross radioactivity analysis shall consist of the quantitative measurement of the total specific activity of the reactor coolant except for radionuclides with half-lives less than 10 minutes and all radioiodines. The total specific activity shall be the sum of the degassed beta-gamma activity and the total of all identified gaseous activities in the sample within 2 hours after the sample is taken and extrapolated back to when the sample was taken. Determination of the contributors to the gross specific activity shall be based upon those energy peaks identifiable with a 95% confidence level. The latest available data may be used for pure beta-emitting radio-nuclides.

***A radiochemical analysis for $\bar{E}$ shall consist of the quantitative measurement of the specific activity for each radionuclide, except for radionuclides with half-lives less than 10 minutes and all radio-iodines, which is identified in the reactor coolant. The specific activities for these individual radionuclides shall be used in the determination of $\bar{E}$ for the reactor coolant sample. Determination of the contributors to $\bar{E}$ shall be based upon these energy peaks identifiable with a 95% confidence level.

**** For Unit 1, reactor coolant DOSE EQUIVALENT I-131 will be limited to 0.35 microCuries per gram.

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 = 150 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 150 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown, mainsteam lines, or the steam jet air ejectors. 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 the current Westinghouse or Babcock & Wilcox Nuclear Technologies Technical Reports.

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, excluding defects that meet the criteria for F* tubes. 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 the current Westinghouse or Babcock & Wilcox Nuclear Technologies Technical Reports. 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.

3/4.4.5 STEAM GENERATORS (continued)

~~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. The leakage limit, 9.1 gpm,~~ includes the accident leakage from ~~IPC,~~ APC in addition to the accident leakage from F^* on the faulted steam generator and the operational leakage limit of Specification 3.4.6.2.c. The operational leakage limit of Specification 3.4.6.2.c in each of the three remaining intact steam generators shall include the operational leakage from F^*.

For Unit 1, plugging or repair is not required for tubes with degradation within the tubesheet area which fall under the alternate tube plugging criteria defined as F^*. The F^* Criteria is based on "Babcock & Wilcox Nuclear Technologies (BWNT) Topical Report BAW-10196 P."

F^* tubes meet the structural integrity requirements with appropriate margins for safety as specified in Regulatory Guide 1.121 and the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB and Division I Appendices, for normal operating and faulted conditions.

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.

The maximum site allowable primary-to-secondary leakage limit for End of Cycle Main Steamline Break conditions

Insert F

ATTACHMENT D

EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATIONS FOR PROPOSED CHANGES TO APPENDIX A TECHNICAL SPECIFICATIONS OF FACILITY OPERATING LICENSES NPF-72 AND NPF-77

Commonwealth Edison Company (ComEd) 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.

A. INTRODUCTION

ComEd proposes to amend the following Braidwood Technical Specifications:

Specification 3/4.4.5 REACTOR COOLANT SYSTEM-STEAM GENERATORS

The changes proposed to TS 3.4.5 renew the voltage-based Alternate Plugging Criteria (APC) limit of 1.0 volt, which was previously approved for Braidwood Unit 1, Cycle 5. Braidwood's probability of tube burst limit is decreased from $2.5x10^{-2}$ to $1.0x10^{-2}$ consistent with Generic Letter 95-05, "Voltage Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking".

Technical Specification Bases Section 3/4.4.5, STEAM GENERATORS (SG) will be modified to reflect these changes.

Specification 3/4.4.8 REACTOR COOLANT SYSTEM-SPECIFIC ACTIVITY

This amendment request also proposes to modify the footnote to TS 3.4.8.a which limits Unit 1 Cycle 5 RCS DOSE EQUIVALENT Iodine 131 (DE I-131) to 0.35 µCi/gm. The footnote will be changed such that the 0.35 µCi/gm DE I-131 limit is not cycle specific.

In the most recent Braidwood Unit 1 Steam Generator Mid-Cycle Outage (A1M05), conducted in the ~~spring of~~ 1995, a SG tube inservice inspection was performed in accordance with the current TSSR 4.4.5.0. The results of this inspection identified a total of 3935 bobbin coil indications at the tube support plate locations. Using a rotating pancake coil to confirm these indications and a 1 volt interim plugging criteria (IPC), 874 flaws were identified due to ODSCC at the TSPs in 815 SG tubes. The 815 tubes were removed from service by plugging. This increased the overall plugging total for Braidwood Unit 1 to 1678 tubes or 9.2% of the tubes. Of the 1678 tubes plugged to date, 1587 were plugged due to ODSCC at the tube support plate locations.

For the upcoming Braidwood Unit 1 Refueling Outage (A1R05), the prediction on the number of pluggable indications using the previous TSSR 4.4.5 acceptance criteria (40-percent through-wall) is approximately 2600 tubes. This would result in 4278 tubes (23.4-percent) being plugged or repaired. With the approval to use the APC as proposed, the predicted number of tubes requiring removal from service by plugging or repair by sleeving would be reduced to 975. This represents a savings of approximately $7.5M in plugging and sleeving repair costs alone. In addition, APC implementation saves a minimum of 20 days in critical path outage time and eliminates the associated replacement power costs. Also, permitting these tubes to remain in service maximizes RCS flow and heat transfer area availability and minimizes RCS loop asymmetries and loss of rated thermal power.

B. NO SIGNIFICANT HAZARDS ANALYSIS

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

Consistent with Regulatory Guide (RG) 1.121, "Basis for Plugging Degraded PWR Steam Generator Tubes," Revision 0, August 1976, the traditional depth-based criteria for SG tube repair implicitly ensures that tubes accepted for continued service will retain adequate structural and leakage integrity during normal operating, transient, and postulated accident conditions. It is recognized that defects in tubes permitted to remain in service, especially cracks, occasionally grow entirely through-wall and develop small leaks. Limits on allowable primary-to-secondary leakage established in Technical Specifications ensure timely plant shutdown before the structural and leakage integrity of the affected tube is challenged.

The proposed license amendment request to implement voltage amplitude SG tube support plate APC for Braidwood Unit 1 meets the requirements of RG 1.121. The APC methodology demonstrates that tube leakage is acceptably low and tube burst is a highly improbable event during either normal operation or the most limiting accident condition, a postulated main steam line break (MSLB) event.

During transients, the tube support plate (TSP) is conservatively assumed to displace due to the ~~thermal~~-hydraulic loads associated with the transient. This may partially expose a crack which is within the boundary of the TSP during normal operations to free span conditions. Burst is therefore conservatively evaluated assuming the crack is fully exposed to free span conditions. The structural eddy current bobbin coil voltage limit for free-span burst is 4.75 volts. This limit takes into consideration a 1.43 safety factor applied to the steam line break differential pressure that is consistent with RG 1.121 requirements. With additional considerations for growth rate assumptions and an upper 95% confidence estimate on voltage variability, the maximum voltage indication that could remain in service is given by the upper voltage repair limit equation in Generic Letter 95-05. For added conservatism, the allowable indication voltage is further reduced in the proposed amendment to a 1.0 volt confirmed ODSCC indication limit. All indications greater than 1.0 volt will be subject to an RPC examination. Tubes with RPC confirmed outside diameter stress corrosion cracking (ODSCC) indications will be plugged or sleeved. Any ODSCC indications between 1.0 volt and the upper voltage repair limit which are not confirmed as ODSCC will be allowed to remain in service since these indications are not as likely to affect tube structural integrity or leakage integrity over the next operating cycle as the indications that are detectable by both bobbin and rotating pancake coil (RPC) inspections.

The eddy current inspection process has been enhanced to address RG 1.83, "Inservice Inspection of PWR Steam Generator Tubes," Revision 1, July 1975, considerations as well as the EPRI SG Inspection Guidelines. Enhancements in accordance with Generic Letter 95-05 are in place to increase detection of ODSCC indications and to ensure reliable, consistent acquisition and analysis of data. Based on the conservative selection of the voltage criteria and the increased ability to identify ODSCC, the probability of tube failure during an accident is also not significantly increased due to application of requested APC.

Modification of the Braidwood Specifications for conformance with Generic Letter 95-05 requirements does not impact any accidents previously evaluated. The decrease in the allowed burst probability from 2.5×10^{-2} to 1.0×10^{-2} is conservative.

Calculations conducted for Braidwood have shown that the resulting 2-hour doses at the site boundaries will not currently exceed an appropriately small fraction of 10 CFR 100 dose guideline values in conjunction with the predicted MSLB leakage calculated in accordance with this submittal and a DE I-131 level of 1.0 µCi/gm. The site allowable leakage calculated using a DE I-131 level of 1.0 µCi/gm is 9.4 gallons per minute (gpm). This leakage includes accident leakage and the allowed 0.1 gpm primary-to-secondary leakage of the 3 unfaulted SGs per TS 3.4.6.2.c. However, in order to provide a defense in depth approach to application of this requested APC

and to envelope any future increases in MSLB leakage due to tube degradation, Braidwood ~~is~~ ~~low~~ering the RCS DE I-131 levels to 0.35 µCi/gm for all future cycles until SG replacement. The site allowable leak rate calculated using 0.35 µCi/gm DE I-131 is 26.8 gpm. This leakage also includes accident leakage and the allowed 0.1 gpm primary-to-secondary leakage of the 3 unfaulted SGs per TS 3.4.6.2.c. Lowering the limit to 0.35 µCi/gm DE I-131 is conservative and will not increase the probability or consequences of any accidents previously evaluated.

Renewal of the 1.0 volt IPC for Braidwood Unit 1 does not adversely affect steam generator tube integrity and results in acceptable dose consequences. Therefore, the proposed license amendment request does not result in any significant 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.**

Renewal of the proposed SG tube APC for Braidwood Unit 1 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 since industry experience indicates that ODSCC originating within the tube support plate does not extend significantly beyond the thickness of the support plate. This criteria only applies to ODSCC contained within the region of the tube bounded by the tube support plate. Therefore, neither a single or multiple tube rupture event would be expected in a steam generator in which APC has been applied.

In addressing the combined effects of Loss of Coolant Accident (LOCA) coincident with a Safe Shutdown Earthquake (SSE) on the SG (as required by General Design Criteria 2), it has been determined that tube collapse of select tubes may occur in the SGs at some plants, including Braidwood Unit 1. 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, that demonstrate the potential for tube collapse during a LOCA + SSE event. Because of this potential, these tubes have been excluded from application of the voltage-based SG TSP APC.

ComEd has implemented a maximum primary to secondary leakage limit of 150 gallons per day (gpd) through any one SG at Braidwood to help preclude the potential for excessive leakage during all plant conditions. The 150 gpd limit provides for leakage detection and plant shutdown in the event of an unexpected single crack leak associated with the longest permissible free span crack length. The 150 gpd limit provides adequate leakage detection and plant shutdown criteria in the event an unexpected single crack results 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 a conservative limit to prompt plant shutdown prior t reaching critical crack lengths under MSLB conditions.

Calculations conducted for Braidwood have shown that the resulting 2-hour doses at the site boundaries will not currently exceed an appropriately small fraction of 10 CFR 100 dose guideline values in conjunction with the predicted MSLB leakage calculated in accordance with this submittal and a DE I-131 level of 1.0 µCi/gm. The site allowable leakage calculated using a DE I-131 level of 1.0 µCi/gm is 9.4 gpm. This leakage includes accident leakage and the allowed 0.1 gpm primary-to-secondary leakage of the 3 unfaulted SGs per TS 3.4.6.2.c. However, in order to provide a defense in depth approach to application of this requested APC and to envelope any future increases in MSLB leakage due to tube degradation, Braidwood is lowering the RCS DE I-131 levels to 0.35 µCi/gm for all future cycles until SG replacement. The site allowable leak rate calculated using 0.35 µCi/gm DE I-131 is 26.8 gpm. This leakage also includes accident leakage and the allowed 0.1 gpm primary-to-secondary leakage of the 3 unfaulted SGs per TS 3.4.6.2.c. Lowering the Braidwood Unit 1 RCS DE I-131 concentration limit to the 0.35 µCi/gm is conservative and will not introduce any changes to the design basis for Braidwood Station.

Modification of the Braidwood Specifications for conformance with Generic Letter 95-05 requirements will not alter the plant design basis. The decrease in the allowed burst probability from 2.5×10^{-2} to 1.0×10^{-2} is conservative.

Upon renewal of the 1.0 volt APC for Braidwood Unit 1, steam generator tube integrity continues to be maintained through inservice inspection and primary-to-secondary leakage monitoring. Therefore, 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 APC for Braidwood Unit 1 will maintain steam generator tube integrity commensurate with the criteria of RG 1.121 as discussed above. 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 distribution of crack indications at the TSP elevations results in acceptable primary-to-secondary leakage during all plant conditions and radiological consequences are not adversely impacted by the application of APC.

The installation of SG tube plugs and sleeves reduces the RCS flow margin. As noted previously, renewal of the SG TSP APC will decrease the number of tubes which must be repaired by plugging or sleeving. Thus, renewal of APC will retain additional flow margin that would otherwise be reduced due to increased tube plugging. Therefore, no significant reduction in the margin of safety will occur as a result of this proposed license amendment request.

Although not relied upon to prove adequacy of the proposed amendment request, the following analyses demonstrate that significant conservatisms exist in the methods and justifications described above:

LIMITED TUBE SUPPORT PLATE DISPLACEMENT

An analysis was performed to verify the extent of limited TSP displacement during accident conditions (MSLB). Application of minimum TSP displacement assumptions provides conservatism and reduces the likelihood of a tube burst to negligible levels. Consideration of limited TSP displacement would also reduce potential MSLB leakage when compared to the leakage calculated assuming free span indications.

PROBABILITY OF DETECTION

The Electric Power Research Institute (EPRI) Performance Demonstration Program analyzed the performance of approximately 20 eddy current data analysts evaluating data from a unit with 3/4" inside diameter and 0.043" wall thickness tubes. The results of this analysis clearly show that the detectability of larger voltage indications is increased which lends creditability for application of a POD of > 0.6 for ODSCC indications larger than 1.0 volt.

RISK EVALUATION OF CORE DAMAGE

As part of ComEd's evaluation of the operability of Braidwood Unit 1, a risk evaluation was completed. The objective of this evaluation was to compare core damage frequency under containment bypass conditions, with and without the APC applied at Braidwood Unit 1. The total Braidwood core damage frequency is estimated to be 3.09E-5 per reactor year with a total contribution from containment bypass sequences of 3.72E-8 per reactor year according to the results of the current individual plant evaluation (IPE). Operation with the requested APC resulted in an insignificant increase in core damage frequency resulting from MSLB with containment bypass conditions.

Calculations conducted for Braidwood have shown that the resulting 2-hour doses at the site boundaries will not currently exceed an appropriately small fraction of 10 CFR 100 dose guideline values in conjunction with the predicted MSLB leakage calculated in accordance with this submittal and a DE I-131 level of 1.0 µCi/gm. The site allowable leakage calculated using a DE I-131 level of 1.0 µCi/gm is 9.4 gpm. This leakage includes accident leakage and the allowed 0.1 gpm primary-to-secondary leakage of the 3 unfaulted SGs per TS 3.4.6.2.c. However, in order to provide a defense in depth approach to application of this requested APC and to envelope any future increases in MSLB leakage due to tube degradation, Braidwood is lowering the RCS DE I-131 levels to 0.35 µCi/gm for all future cycles until SG replacement. The site allowable leak rate calculated using 0.35 µCi/gm DE I-131 is 26.8 gpm. This leakage also includes accident leakage and the allowed 0.1 gpm primary-to-secondary leakage of the 3 unfaulted SGs per TS 3.4.6.2.c. Lowering the Braidwood Unit 1 RCS DE I-131 concentration limit to the 0.35 µCi/gm is conservative and will not introduce any changes to the design basis for Braidwood Station. Thus this change is in conformance with Braidwood's current TS and does not involve a reduction in a margin of safety.

Modification of the Braidwood Specifications for conformance with Generic Letter 95-05 requirements will not reduce any safety margins. The decrease in the allowed burst probability from 2.5×10^{-2} to 1.0×10^{-2} is conservative.

Therefore, based on the evaluation above, ComEd 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 LICENSES NPF-72 AND NPF-77

Commonwealth Edison Company (ComEd) has evaluated this proposed license amendment request against the criteria for 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). ComEd 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 an inspection or a surveillance requirement. This proposed license amendment request changes Braidwood Technical Specification (TS) 3.4.5, "Steam Generators," the bases for TS 3.4.5, and Braidwood TS 3.4.8, "Specific Activity."

   The changes proposed to TS 3.4.5 renew the voltage-based Alternate Plugging Criteria (APC) limit of 1.0 volt, which was previously approved for Braidwood Unit 1, Cycle 5. The request decreases Braidwood's probability of tube burst limit from $2.5x10^{-2}$ to $1.0x10^{-2}$ and makes various format changes which are consistent with Generic Letter 95-05, "Voltage Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking".

   Technical Specification Bases Section 3/4.4.5 will also be modified to reflect these changes.

   Technical Specification 3.4.8.a, which limits RCS DOSE EQUIVALENT I-131 to 0.35 µCi/gm for Unit 1 Cycle 5, is being changed such that this limit is no longer cycle specific.

   Additional changes are proposed to make the Braidwood TS more consistent with Generic Letter 95-05.

2. This proposed license amendment request involves no significant hazards considerati~~ons;~~

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.