ATTACHMENT B-1

MARKED UP PAGES FOR
PROPOSED CHANGES TO APPENDIX A
TECHNICAL SPECIFICATIONS OF
FACILITY OPERATING LICENSES
NPF-37 AND NPF-66

BYRON STATION UNITS 1 & 2
REVISED PAGES

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

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.

1) All tubes that previously had detectable tube wall penetrations greater than 20 percent 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) 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,

Westinghouse Model D4 steam generators

4) 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, and

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.

d. For Unit 1, through Cycle 8, 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.

For Westinghouse Model D4 and D5 steam generators,

e. A random sample of at least 20% of the total number of laser welded sleeves and at least 20% of the total number of TIG welded sleeves installed shall be inspected for axial and circumferential indications at the end of each cycle. In the event that an imperfection exceeding the repair limit is detected, an additional 20% of the unsampled sleeves shall be inspected, and if an imperfection exceeding the repair limit is detected in the second sample, all remaining sleeves shall be inspected. These inservice inspections will include the entire sleeve, the tube at the heat treated area, and the tube to sleeve joints. The inservice inspection for the sleeves is required on all types of sleeves installed in the Byron and Braidwood Steam Generators to demonstrate acceptable structural integrity.

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 or initial operation following a steam generator replacement. 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 for the tubing and laser welded sleeves is equal to 40% of the nominal wall thickness. The plugging limit imperfection depth for TIG welded sleeves is equal to 32% 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;

Replace with Insert A

For Unit 1, through Cycle 8, this definition does not apply to tube support plate intersections for which the voltage-based plugging criteria are being applied. Refer to 4.4.5.4.a.11 for the repair limit applicable to these intersections;

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;

INSERT A

The plugging or repair limit imperfection depth for tubing is equal to 40% of the nominal wall thickness. For Westinghouse Model D4 and D5 steam generators, the plugging or repair limit imperfection depth for laser welded sleeves is equal to 40% of the nominal sleeve wall thickness, and for TIG welded sleeves is equal to 32% of the nominal sleeve wall thickness. For Westinghouse Model D4 steam generators, this definition does not apply to defects in the tubesheet that meet the criteria for an F* tube;

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

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 (for Westinghouse Model D4 or D5 steam generators) will be performed by the following processes:

a) Laser welded sleeving as described in a Westinghouse Technical Report currently approved by the NRC, subject to the limitations and restrictions as noted by the NRC staff, or

b) TIG welded sleeving as described in ABB Combustion Engineering Inc. Technical Reports: Licensing Report CEN-621-P, Revision 00, "Commonwealth Edison Byron and Braidwood Unit 1 and 2 Steam Generators Tube Repair Using Leak Tight Sleeves, FINAL REPORT," April 1995, and Licensing Report CEN-627-P, Revision 00-P, "Verification of the Installation Process and Operating Performance of the ABB CENO Steam Generator Tube Sleeve for Use at Commonwealth Edison Byron and Braidwood Units 1 and 2," January 1996, subject to the limitations and restrictions as noted by the NRC Staff.

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.

11) For Unit 1 through Cycle 8, 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 outer 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, with degradation attributed to outside diameter stress corrosion cracking within the bounds of the cold-leg 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. Steam generator tubes, with degradation attributed to outside diameter stress corrosion cracking within the bounds of the hot-leg tube support plate with bobbin voltages less than or equal to 3.0 volts will be allowed to remain in service.

SURVEILLANCE REQUIREMENTS (Continued)

b) Steam generator tubes with degradation attributed to outside diameter stress corrosion cracking within the bounds of the cold-leg 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.d below.

c) Steam generator tubes with degradation attributed to outside diameter stress corrosion cracking within the bounds of the hot-leg tube support plate with a bobbin voltage greater than 3.0 volts will be repaired or plugged.

d) Steam generator tubes, with indications of potential degradation attributed to outside diameter stress corrosion cracking within the bounds of the cold-leg 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 within the bounds of the cold-leg tube support plate with a bobbin voltage greater than the upper voltage repair limit [Note 2] will be plugged or repaired.

e) 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.

f) 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.d for outside diameter stress corrosion cracking indications occurring in the steam generator cold-legs. For outside diameter stress corrosion cracking indications occurring in the steam generator hot-legs, the limits in 4.4.5.4.a.11.a and 4.4.5.4.a.11.c apply. 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 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, 4.4.5.4.a.11.c and 4.4.5.4.a.11.d.

Note 1: The lower voltage repair limit is 1.0 volt for indications of outside diameter stress corrosion cracking occurring at cold-leg tube support plate intersections.

Note 2: The upper voltage repair limit for indications of outside diameter stress corrosion cracking occurring at cold-leg tube support plate intersections is calculated according to the methodology in Generic Letter 95-05 as supplemented.

of a Westinghouse Model D4 steam generator

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.

Westinghouse Model D4

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.

SURVEILLANCE REQUIREMENTS (Continued)

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.

d. For implementation of the voltage based repair criteria to tube support plate intersections for Unit 1 through Cycle 8, 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 steamline 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.

SURVEILLANCE REQUIREMENTS (Continued)

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.

6) Following a steam generator internals inspection, if indications detrimental to the integrity of the load path necessary to support the 3.0 volt IPC are found, notify the NRC and provide an assessment of the safety significance of the occurrence.

Westinghouse Model D4 steam generators'

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.

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 mininum 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.

$S = 3 \frac{N}{n} \%$ 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

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 mircoCurie 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, through Cycle 8 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:

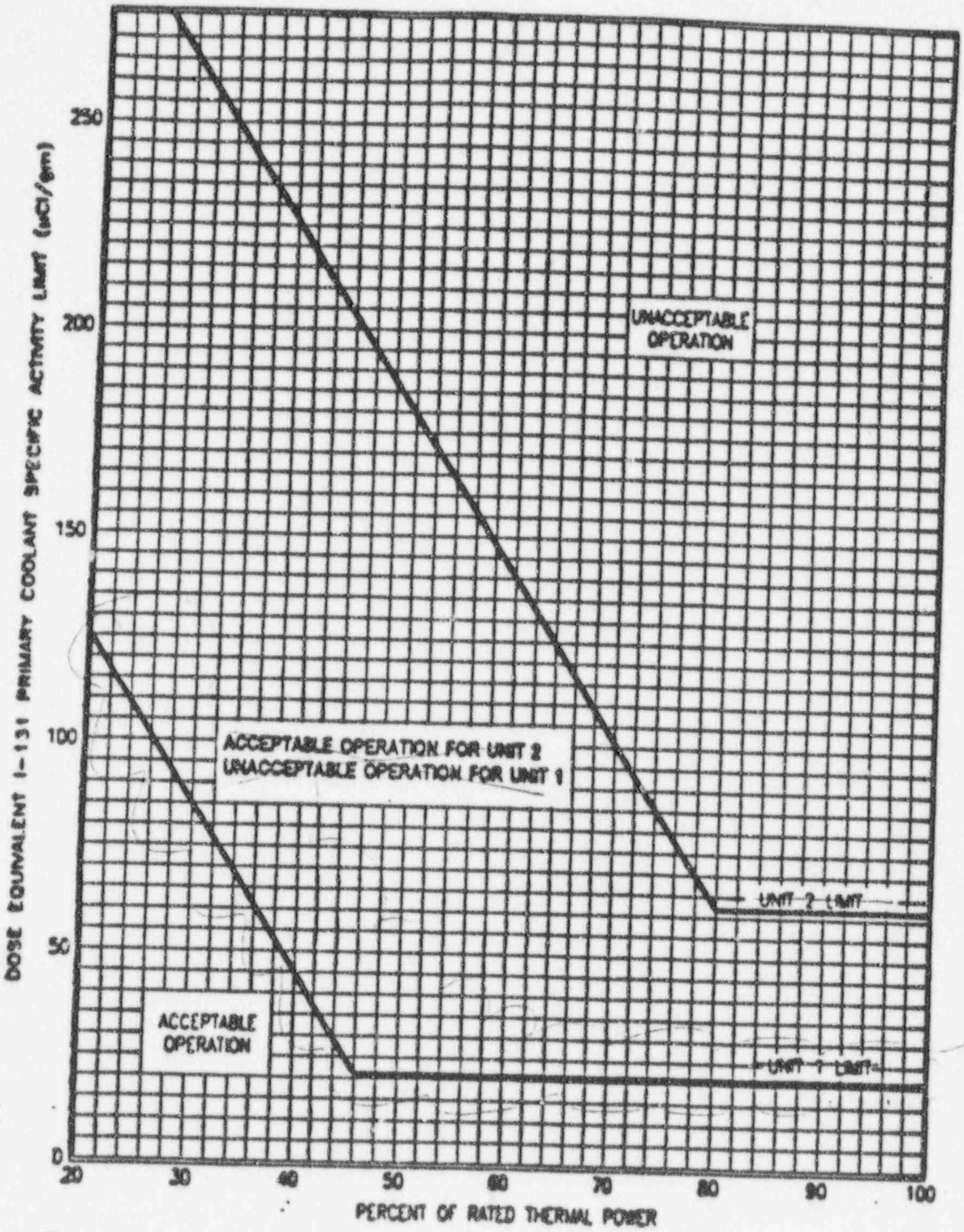
With the specific activity of the reactor coolant greater than 1 microCurie per gram DOSE EQUIVALENT I-131* or greater than $100/\bar{E}$ microCuries per gram, perform the sampling and analysis requirements of Item 4.a) of Table 4.4-4 until the specific activity of the reactor coolant is restored to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

through Cycle 8

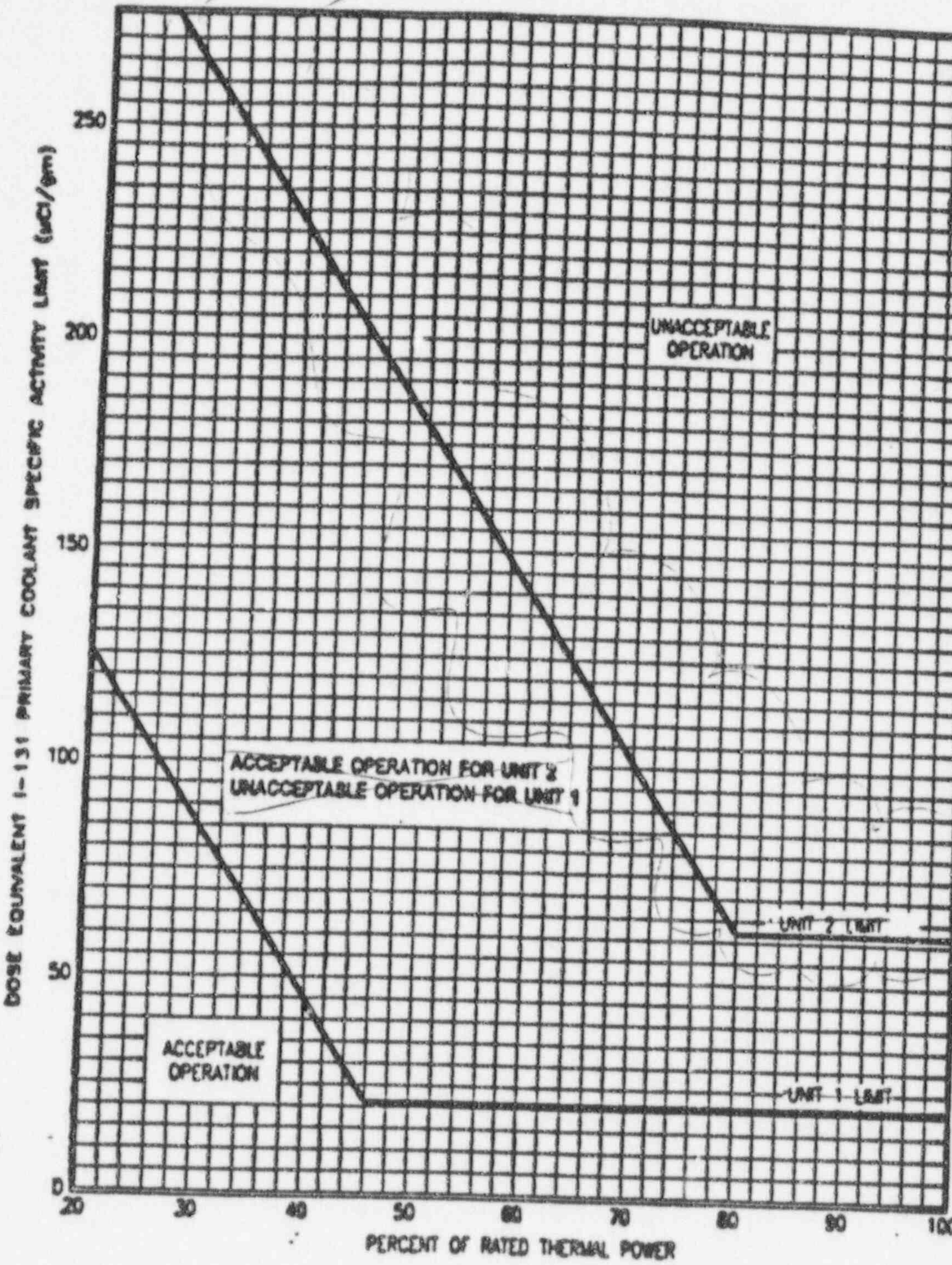
*For Unit 1, reactor coolant DOSE EQUIVALENT I-131 will be limited to 0.35 microCuries per gram.



UNIT 1 (AFTER CYCLE 8) AND UNIT 2

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~~



UNIT 1 THROUGH CYCLE 8

FIGURE 3.4-~~1~~ 2
DOSE EQUIVALENT I-131 REACTOR COOLANT SPECIFIC ACTIVITY
LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE REACTOR
COOLANT SPECIFIC ACTIVITY >~~1~~ 0.35 µCi/GRAM DOSE EQUIVALENT I-131~~*~~

~~*For Unit 1, Reactor Coolant Specific Activity >0.35 µCi/Gram DOSE EQUIVALENT I-131~~

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 percent 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 percent confidence level. The latest available data may be used for pure beta-emitting radionuclides.

***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 radioiodines, 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 percent confidence level.

****For Unit 1, through Cycle 8 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 ABB Combustion Engineering, Inc. Technical Reports (which are applicable for Westinghouse Model D4 and D5 steam generators only).

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 (Westinghouse Model D4) "F" tubes. (Acceptable plugging criteria for Westinghouse Model D4 and D5 sleeved tubes are: 1) a) laser welded sleeved tube must be plugged if a through wall penetration is detected in the sleeve that is equal to or greater than 40% of the nominal sleeve thickness (and, 2) TIG welded sleeved tubes must be plugged if a through wall penetration is detected in the sleeve that is equal to or greater than 32% of the nominal sleeve thickness. The 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 ABB Combustion Engineering, Inc. Technical Reports. Steam Generator tube and sleeve inspections have demonstrated the capability to reliably detect degradation 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)

The voltage-based repair limits for Unit 1 in Surveillance Requirement (SR) 4.4.5 implement the guidance in Generic Letter 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking" (Generic Letter 95-05) for Westinghouse-designed steam generators (SGs) with the exception of the specific voltage limit. Generic Letter 95-05 discusses a 1.0 volt Alternate Plugging Criteria (APC) that can be applied to more than one cycle of operation. Byron SR 4.4.5 implements a 3.0 volt hot-leg Interim Plugging Criteria (IPC) and a 1.0 volt cold-leg IPC for the Unit 1 SGs per WCAP-14273, "Technical Support for Alternative Plugging Criteria with Tube Expansion at Tube Support Plate Intersections for Braidwood-1 and Byron-1 Model D-4 Steam Generators" for a specified operating cycle.

The voltage-based repair limits of SR 4.4.5 are applicable only to Westinghouse-designed SGs with outside diameter stress corrosion cracking (ODSCC) located at the tube-to-tube support plate intersections. The voltage-based repair limits are not applicable to other forms of SG tube degradation nor are they applicable to ODSCC that occurs at other locations within the SG. Additionally, the repair criteria apply only to indications where the degradation mechanism is dominantly axial ODSCC with no significant cracks extending outside the thickness of the support plate. Refer to Generic Letter 95-05 for additional description of the degradation morphology.

Application of the 3.0 volt hot-leg IPC requires verification of the integrity of load path necessary to support this IPC in accordance with the Byron/Braidwood Steam Generator Internals Inspection Plan.

Implementation of SR 4.4.5 requires a derivation of the voltage structural limit from the burst versus voltage empirical correlation and then the subsequent derivation of the voltage repair limit from the structural limit (which is then implemented by this surveillance).

The voltage structural limit is the voltage from the burst pressure/bobbin voltage correlation, at the 95-percent prediction interval curve reduced to account for the lower 95/95-percent tolerance bound for tubing material properties at 650°F (i.e., the 95-percent LTL curve). The voltage structural limit must be adjusted downward to account for potential flaw growth during an operating interval and to account for NDE uncertainty. The upper voltage repair limit for cold-leg indications at the tube support plate; V_{URL}, is determined from the structural voltage limit by applying the following equation:

$$V_{URL}=V_{SL}-V_{Gr}-V_{NDE}$$

where V_{Gr} represents the allowance for flaw growth between inspections and V_{NDE} represents the allowance for potential sources of error in the measurement of the bobbin coil voltage. Further discussion of the assumptions necessary to determine the voltage repair limit is contained in Generic Letter 95-05.

3/4.4.5 STEAM GENERATORS (Continued)

The mid-cycle equation in SR 4.4.5.4.a.11.f should only be used during unplanned inspections in which eddy current data is acquired for indications at the cold-leg tube support plates. The voltage repair limit for indications at the hot-leg tube support plates remains at 3.0 volts during unplanned inspections.

SR 4.4.5.5 implements several reporting requirements recommended by Generic Letter 95-05 for situations which the NRC wants to be notified prior to returning the SGs to service. For the purposes of this reporting requirement, leakage and conditional burst probability can be calculated based on the as-found voltage distribution rather than the projected end-of-cycle voltage distribution (refer to Generic Letter 95-05 for more information) when it is not practical to complete these calculations using the projected end-of-cycle voltage distributions prior to returning the SGs to service. Note that if leakage and conditional burst probability were calculated using the measured end-of-cycle voltage distribution for the purposes of addressing Generic Letter 95-05 sections 6.a.1 and 6.a.3 reporting criteria, then the results of the projected end-of-cycle voltage distribution should be provided per Generic Letter 95-05 section 6.b(c) criteria.

The maximum site allowable primary-to-secondary leakage limit for end-of-cycle main steamline break conditions includes the accident leakage from IPC 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*.

(Westinghouse Model D4 steam generators)

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.

ATTACHMENT B-2

MARKED UP PAGES FOR
PROPOSED CHANGES TO APPENDIX A
TECHNICAL SPECIFICATIONS OF
FACILITY OPERATING LICENSES
NPF-72 AND NPF-77

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*NOTE: THESE PAGES HAVE NO CHANGES BUT ARE INCLUDED FOR CONTINUITY

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

SECTION PAGE

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.

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) 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,

Westinghouse Model D4 steam generators

4) 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, and

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.

d. For Unit 1 Cycle 6, 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.

For Westinghouse Model D4 and D5 steam generators

e. A random sample of at least 20% of the total number of laser welded sleeves and at least 20% of the total number of TIG welded sleeves installed shall be inspected for axial and circumferential indications at the end of each cycle. In the event that an imperfection exceeding the repair limit is detected, an additional 20% of the unsampled sleeves shall be inspected, and if an imperfection exceeding the repair limit is detected in the second sample, all remaining sleeves shall be inspected. These inservice inspections will include the entire sleeve, the tube at the heat treated area, and the tube to sleeve joints. The inservice inspection for the sleeves is required on all types of sleeves installed in the Byron and Braidwood Steam Generators to demonstrate acceptable structural integrity.

SURVEILLANCE REQUIREMENTS (Continued)

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.

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~~ on initial operation following a steam generator replacement. 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 for the tubing and laser welded sleeves is equal to 40% of the nominal wall thickness. The plugging limit imperfection depth for TIG welded sleeves is equal to 32% 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 6, this definition does not apply to the 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;

Replace with Insert A

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 A

The plugging or repair limit imperfection depth for tubing is equal to 40% of the nominal wall thickness. For Westinghouse Model D4 and D5 steam generators, the plugging or repair limit imperfection depth for laser welded sleeves is equal to 40% of the nominal sleeve wall thickness, and for TIG welded sleeves is equal to 32% of the nominal sleeve wall thickness. For Westinghouse Model D4 steam generators, this definition does not apply to defects in the tubesheet that meet the criteria for an F* tube;

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 (for Westinghouse Model D4 or D5 Steam Generators) will be performed by the following processes:

a) Laser welded sleeving as described in a Westinghouse Technical Report currently approved by the NRC, subject to the limitations and restrictions as noted by the NRC staff, or

b) TIG welded sleeving as described in ABB Combustion Engineering Inc. Technical Reports: Licensing Report CEN-621-P, Revision 00, "Commonwealth Edison Byron and Braidwood Unit 1 and 2 Steam Generators Tube Repair Using Leak Tight Sleeves, FINAL REPORT," April 1995, and Licensing Report CEN-627-P, Revision 00-P, "Verification of the Installation Process and Operating Performance of the ABB CENO Steam Generator Tube Sleeve for Use at Commonwealth Edison Byron and Braidwood Units 1 and 2," January 1996, subject to the limitations and restrictions as noted by the NRC Staff.

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.

11) For Unit 1 Cycle 6, 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, with degradation attributed to outside diameter stress corrosion cracking within the bounds of the cold-leg 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. Steam generator tubes, with degradation attributed to outside diameter stress corrosion cracking within the bounds of the hot-leg tube support plate with bobbin voltages less than or equal to 3.0 volts will be allowed to remain in service.

b. Steam generator tubes with degradation attributed to outside diameter stress corrosion cracking within the bounds of the cold-leg 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.d below.

SURVEILLANCE REQUIREMENTS (Continued)

c. Steam generator tubes with degradation attributed to outside diameter stress corrosion cracking within the bounds of the hot-leg tube support plate with a bobbin voltage greater than 3.0 volts will be repaired or plugged.

d. Steam generator tubes, with indications of potential degradation attributed to outside diameter stress corrosion cracking within the bounds of the cold-leg 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 within the bounds of the cold-leg tube support plate with a bobbin voltage greater than the upper voltage repair limit [Note 2] will be plugged or repaired.

e. 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.

f. 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.d for outside diameter stress corrosion cracking indications occurring in the steam generator cold-legs. For outside diameter stress corrosion cracking indications occurring in the steam generator hot-legs, the limits in 4.4.5.4.a.11.a and 4.4.5.4.a.11.c apply. The mid-cycle repair limits are determined from the following equations:

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

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

SURVEILLANCE REQUIREMENTS (Continued)

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 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, 4.4.5.4.a.11.c and 4.4.5.4.a.11.d.

Note 1: The lower voltage repair limit is 1.0 volt for indications of outside diameter stress corrosion cracking occurring at cold-leg tube support plate intersections.

Note 2: The upper voltage repair limit for indications of outside diameter stress corrosion cracking occurring at cold-leg tube support plate intersections is calculated according to the methodology in Generic Letter 95-05 as supplemented.

of a Westinghouse Model D4 steam generator

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.

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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.

SURVEILLANCE REQUIREMENTS (Continued)

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.

d. For implementation of the voltage based repair criteria to tube support plate intersections for Unit 1 Cycle 6, 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.

6. Following a steam generator internals inspection, if indications detrimental to the integrity of the load path necessary to support the 3.0 volt IPC are found, notify the NRC and provide an assessment of the safety significance of the occurrence.

Westinghouse Model D4 steam generators'

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.

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 mininum 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.

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

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 through Cycle 7, 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:

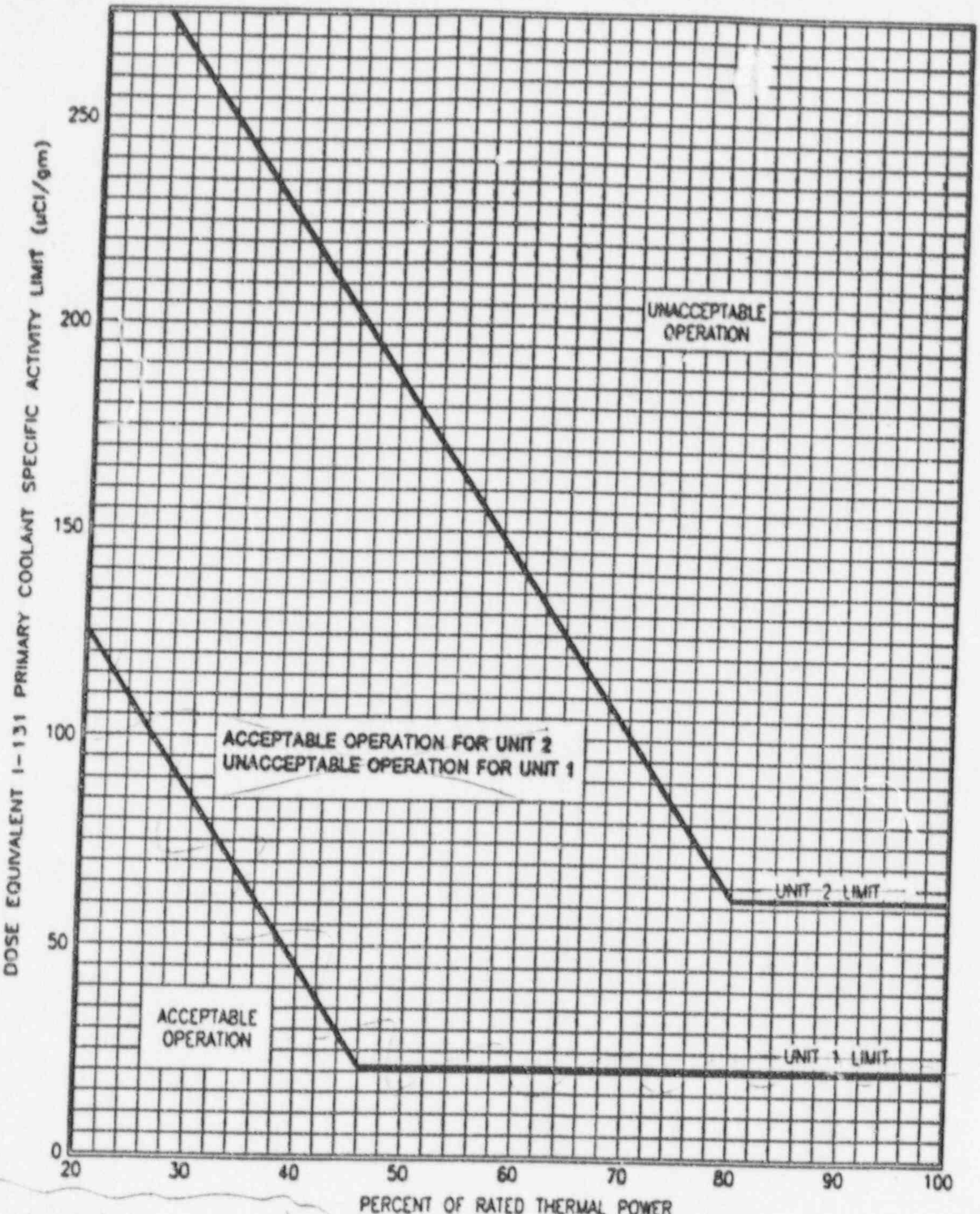
With the specific activity of the reactor coolant greater than 1 microCurie per gram DOSE EQUIVALENT I-131* or greater than 100/Ē microCuries per gram, perform the sampling and analysis requirements of Item 4.a) of Table 4.4-4 until the specific activity of the reactor coolant is restored to within its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

through Cycle 7

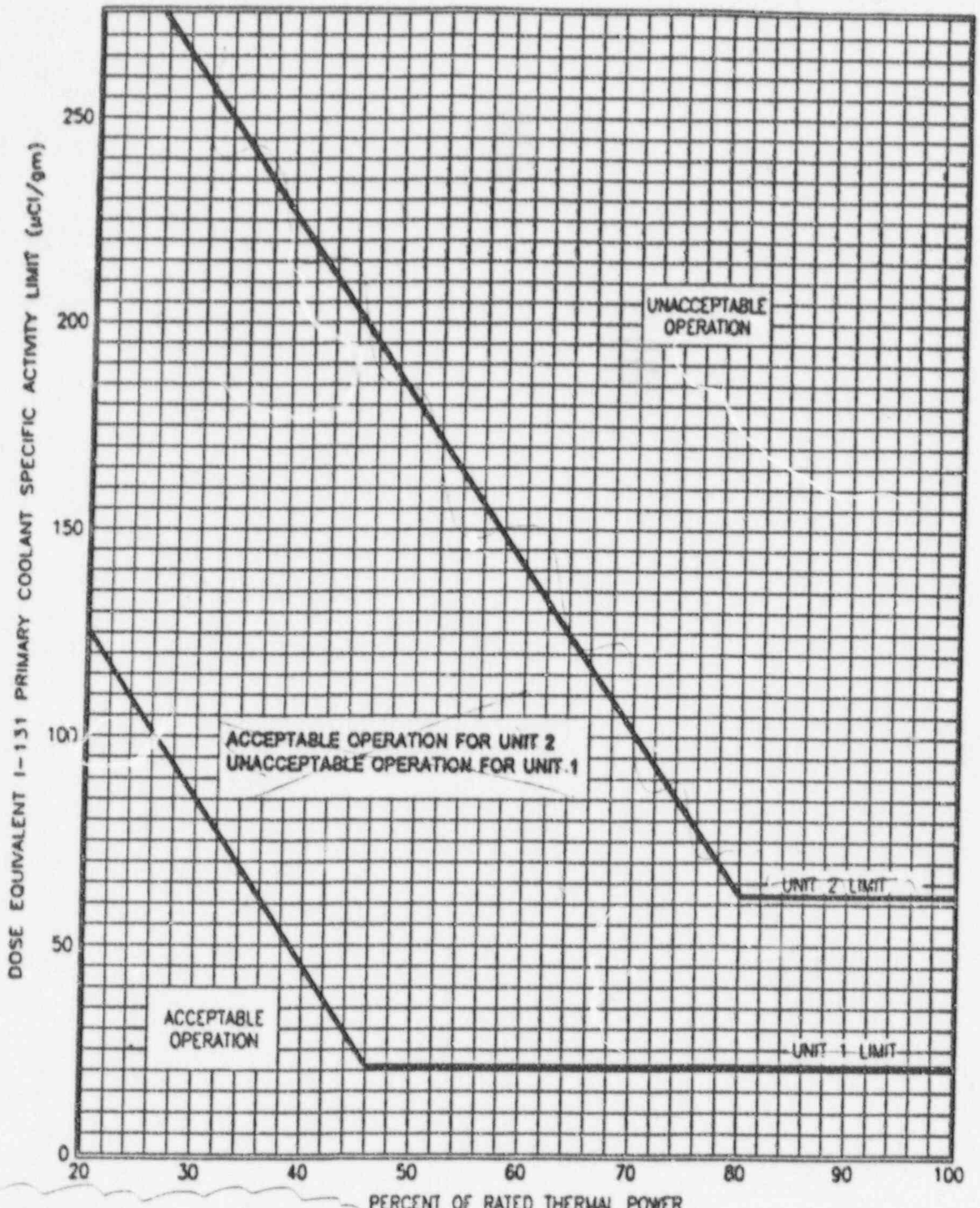
*For Unit 1, reactor coolant DOSE EQUIVALENT I-131 will be limited to 0.35 microCuries per gram.



Unit 1 (After Cycle 7) and Unit 2

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~~



Unit 1 Through Cycle 7

FIGURE 3.4-~~1~~ 2

DOSE EQUIVALENT I-131 REACTOR COOLANT SPECIFIC ACTIVITY
LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE REACTOR
COOLANT SPECIFIC ACTIVITY >~~1~~ 0.35 µCI/GRAM DOSE EQUIVALENT I-131*

*For Unit 1, Reactor Coolant Specific Activity >0.35 µCi/Gram DOSE EQUIVALENT I-131

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 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 radionuclides.

*** 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 radioiodines, 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 through Cycle 7, 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 ABB Combustion Engineering, Inc. Technical Reports (which are applicable to Westinghouse Model D4 and D5 ...).

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. A (Westinghouse Model D4) laser welded sleeved tube must be plugged if a through wall penetration is detected in the sleeve that is equal to or greater than 40% of the nominal sleeve thickness, and a TIG welded sleeved tubes must be plugged if a through wall penetration is detected in the sleeve that is equal to or greater than 32% of the nominal sleeve thickness. The 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 ABB Combustion Engineering, Inc. Technical Reports. Steam Generator tube and sleeve inspections have demonstrated the capability to reliably detect degradation 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.

(Acceptable plugging criteria for Westinghouse Model D4 and D5 sleeved tubes are ...)

3/4.4.5 STEAM GENERATORS (continued)

The voltage-based repair limits for Unit 1 in Surveillance Requirement (SR) 4.4.5 implement the guidance in Generic Letter 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking" (Generic Letter 95-05) for Westinghouse-designed steam generators (SGs) with the exception of the specific voltage limit. Generic Letter 95-05 discusses a 1.0 volt Alternate Plugging Criteria (APC) that can be applied to more than one cycle of operation. Braidwood SR 4.4.5 implements a 3.0 volt hot-leg Interim Plugging Criteria (IPC) and a 1.0 volt cold-leg IPC for the Unit 1 SGs per WCAP-14273, "Technical Support for Alternative Plugging Criteria with Tube Expansion at Tube Support Plate Intersections for Braidwood-1 and Byron-1 Model D-4 Steam Generators" for a specified operating cycle.

The voltage-based repair limits of SR 4.4.5 are applicable only to Westinghouse-designed SGs with outside diameter stress corrosion cracking (ODSCC) located at the tube-to-tube support plate intersections. The voltage-based repair limits are not applicable to other forms of SG tube degradation nor are they applicable to ODSCC that occurs at other locations within the SG. Additionally, the repair criteria apply only to indications where the degradation mechanism is dominantly axial ODSCC with no significant cracks extending outside the thickness of the support plate. Refer to Generic Letter 95-05 for additional description of the degradation morphology.

Application of the 3.0 volt hot-leg IPC requires verification of the integrity of the load path necessary to support this IPC in accordance with the Byron/Braidwood Steam Generator Internals Inspection Plan.

Implementation of SR 4.4.5 requires a derivation of the voltage structural limit from the burst versus voltage empirical correlation and then the subsequent derivation of the voltage repair limit from the structural limit (which is then implemented by this surveillance).

The voltage structural limit is the voltage from the burst pressure/bobbin voltage correlation, at the 95-percent prediction interval curve reduced to account for the lower 95/95-percent tolerance bound for tubing material properties at 650°F (i.e., the 95-percent LTL curve). The voltage structural limit must be adjusted downward to account for potential flaw growth during an operating interval and to account for NDE uncertainty. The upper voltage repair limit for cold-leg indications at the tube support plate; V_{URL}, is determined from the structural voltage limit by applying the following equation:

$$V_{URL} = V_{SL} - V_{Gr} - V_{NDE}$$

where V_{Gr} represents the allowance for flaw growth between inspections and V_{NDE} represents the allowance for potential sources of error in the measurement of the bobbin coil voltage. Further discussion of the assumptions necessary to determine the voltage repair limit is contained in Generic Letter 95-05.

3/4.4.5 STEAM GENERATORS (continued)

The mid-cycle equation in SR 4.4.5.4.a.11.f should only be used during unplanned inspections in which eddy current data is acquired for indications at the cold-leg tube support plates. The voltage repair limit for indications at the hot-leg tube support plates remains at 3.0 volts during unplanned inspections.

SR 4.4.5.5. implements several reporting requirements recommended by Generic Letter 95-05 for situations which the NRC wants to be notified prior to returning the SGs to service. For the purposes of this reporting requirement, leakage and conditional burst probability can be calculated based on the as-found voltage distribution rather than the projected end-of-cycle voltage distribution (refer to Generic Letter 95-05 for more information) when it is not practical to complete these calculations using the projected end-of-cycle voltage distributions prior to returning the SGs to service. Note that if leakage and conditional burst probability were calculated using the measured end-of-cycle voltage distribution for the purposes of addressing Generic Letter 95-05 sections 6.a.1 and 6.a.3 reporting criteria, then the results of the projected end-of-cycle voltage distribution should be provided per Generic Letter 95-05 section 6.b(c) criteria.

The maximum site allowable primary-to-secondary leakage limit for end-of-cycle main steamline break conditions includes the accident leakage from IPC 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*.

(Westinghouse Model D4 steam generators)

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.

OPERATIONAL LEAKAGE (Continued)

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady-State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride, and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady-State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting concinued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady-State Limits.

The Surveillance Requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the reactor coolant ensure that the resulting 2-hour doses at the SITE BOUNDARY will not exceed an appropriately small fraction of 10 CFR Part 100 dose guideline values following a steam generator tube rupture accident in conjunction with an assumed steady-state reactor-to-secondary steam generator leakage rate of 1 gpm. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Braidwood Station, such as SITE BOUNDARY location and meteorological conditions, were not considered in this evaluation.

INSERT B

INSERT B

For Unit 1 through Cycle 7, the limitations on the specific activity of the reactor coolant ensure that the resulting 2-hour off-site doses will not exceed an appropriately small fraction of the 10 CFR Part 100 dose guideline values following a Main Steam Line Break accident in conjunction with an assumed steady-state primary-to-secondary steam generator leakage rate of 150 gpd from each unfaulted steam generator and maximum site allowable primary-to-secondary leakage from the faulted steam generator.

ATTACHMENT C

EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATIONS FOR PROPOSED CHANGES TO APPENDIX A TECHNICAL SPECIFICATIONS OF FACILITY OPERATING LICENSES NPF-37, NPF-66, NPF-72, NPF-77

Commonwealth Edison (ComEd) has evaluated this proposed amendment and determined that it involves no significant hazards considerations. According to Title 10 Code of Federal Regulations Section 50 Subsection 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 does 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

Commonwealth Edison Company proposes to amend Technical Specifications (TS) 3/4.4.5, "Steam Generators," and TS 3/4.4.8, "Reactor Coolant System Specific Activity," for Byron Nuclear Power Station Units 1 and 2 (Byron) and Braidwood Nuclear Power Station Units 1 and 2 (Braidwood) to support steam generator (SG) replacement. ComEd will be replacing the original Unit 1 Westinghouse D4 steam generators (OSGs) at Byron and Braidwood with Babcock and Wilcox International (BWI) steam generators. The replacements are scheduled to occur at the end of Cycle 8 for Byron Unit 1 and the end of Cycle 7 for Braidwood Unit 1.

The changes proposed to TS 3/4.4.5 and associated Bases will: 1) permit application of Interim Plugging Criteria (IPC) to Unit 1 OSGs only through Cycle 8 for Byron and Cycle 6 for Braidwood, 2) permit application of F* to Westinghouse Model D4 SGs only, and 3) limit tube repair methods by sleeving to Westinghouse Model D4 and D5 steam generators. Note that a previous request was submitted to extend application of IPC through Cycle 9 for Byron and through Cycle 7 for Braidwood. As a result of steam generator replacement, Byron would only require IPC through Cycle 8.

The changes proposed to TS 3/4.4.8 will raise the Unit 1 Reactor Coolant System (RCS) Dose Equivalent (DE) I-131 limit from 0.35 microCuries per gram (μCi/gm) to 1.0 μCi/gm after completion of Cycle 8 for Byron and Cycle 7 for Braidwood.

With the implementation of this license amendment request, the Braidwood and Byron Unit 1 SGs will continue to retain adequate structural and leakage integrity in accordance with the requirements of GDC 14, 15, 30, 31, and 32 of Title 10, Code of Federal Regulations Part 50 (10CFR50) Appendix A.

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.

Due to design differences between the replacement Steam Generators (RSGs) and OSGs, the analyses supporting the application of the F* and voltage-based repair criteria do not apply to the RSGs. Also, the analyses supporting sleeving repair by the Westinghouse laser welded or Combustion Engineering Tungsten Inert Gas (TIG) welded sleeving methodologies do not apply to the RSGs due to the design differences. The RSG and OSG tube bundle configurations are similar, however, the RSG tubes are smaller in diameter, constructed of Inconel Alloy 690 instead of Alloy 600, and supported by stainless steel lattice grids instead of the drilled carbon steel plates used in the OSGs. The RSG tubes are hydraulically expanded into the tube sheet during initial assembly. The RSG upper tube bundle shape consists of tubes with continuous, smooth, long radius bends.

The structural analysis demonstrates that the tube integrity is maintained for a Main Steamline Break (MSLB) occurring during normal full power operation. The structural evaluation of the tubing for faulted conditions was performed in accordance with the ASME Boiler and Pressure Vessel Code Section III requirements. The tube material selection and size exceed the strength requirements of the existing steam generators. Comparison of the Alloy 690 tube material used in the RSGs with the Alloy 600 tube material in the OSGs show that the RSG material strength characteristics are as good as or better than those of the existing design. A comparison of the stress margins of the RSG and OSG show that the stress margin in the RSG tubes exceed the stress margin in the OSG tubes.

RSG portions of the reactor coolant pressure boundary are designed to permit periodic inspection and testing of important areas and features to assess structural and leak-tight integrity. ASME Section XI, provides the depth of an allowable outside diameter (O.D.) flaw for tubes in service. The RSG has tubing fabricated from SB-163 material (Inconel Alloy 690) which is examined by eddy current methods to the requirements of ASME Section III, NB-2550. The tubing has a radius to thickness (*r/t*) ratio less than 8.70. In accordance with ASME Section XI, for tubing having an *r/t* ratio of less than 8.70, the depth of an allowable O.D. flaw shall not exceed 40% of the nominal tube wall thickness.

The potential for tube rupture is not increased from the OSGs as demonstrated in the qualification analysis and testing for the RSGs. The program for periodic inservice inspection of the steam generators monitors the integrity of the SG tubing to ensure that there is sufficient time to take proper and timely corrective action if any tube degradation is detected. Therefore, installation of the RSGs will not increase the probability of the occurrence of primary-to-secondary leakage or a steam generator tube rupture (SGTR) during normal or accident conditions.

The design basis doses calculated for postulated accidents involving degradation of SG tubes, such as SGTR and MSLB accidents, as presented in UFSAR Chapter 15 accident analysis have been evaluated and are decreased by installation of the RSGs and restoration of the RCS activity limit to 1.0 μCi/gm. The decrease in offsite dose is primarily due to the smaller RSG tube diameter and less primary-to-secondary transfer during the event. The dose calculations are performed consistent with NUREG-0800, "Standard Review Plan" and ensure site boundary doses are within a small fraction of the Title 10 Code of Federal Regulations Part100 (10CFR100) requirements. Therefore, the change does not involve a significant increase in the consequences of an accident previously evaluated.

Limiting the applicability of TS provisions to a specific cycle or SG type are administrative changes in that they provide clarification consistent with current analyses and do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

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

Restricting application of IPC, F*, and sleeving methodologies to the OSGs and reinstating an RCS activity limit of 1.0 μCi/gm upon installation of the RSGs will not introduce significant or adverse changes to the plant design basis that could lead to a new or different kind of accident being created. The RSG tubing meets the requirements of General Design Criteria (GDC) 14, 15, 30, 31, and 32 of 10CFR50, Appendix A. The RSG tubing has been designed and evaluated consistent with ASME Code Section III criteria and the inspection criteria for the RSGs is consistent with ASME Code Section XI criteria. The RSGs have thermally treated Inconel Alloy 690 tubes which are hydraulically expanded into the tube sheet during initial assembly. Alloy 690 is more resistant to stress corrosion cracking (SCC) than Alloy 600 which is used in the OSG tubing. Overall tube bundle structural and leakage integrity is maintained at a level consistent with or better than the originally supplied tubing during all plant conditions.

ComEd will continue to apply the TS maximum primary-to-secondary leakage limit of 150 gpd (0.1 gpm) through any one SG at Byron and Braidwood to help preclude the potential for excessive leakage during all plant conditions. The EPRI recommended 150 gpd limit provides for leakage detection and plant shutdown in the event of an unexpected tube leak and precludes the potential for excessive leakage or tube burst in the event of a Main Steam Line Break or under Loss of Coolant Accident conditions.

Limiting the applicability of TS provisions to a specific cycle or SG type are administrative changes in that they provide clarification consistent with current analyses.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Restricting application of IPC, F*, and sleeving methodologies to the OSGs for which the supporting analyses apply, does not involve a reduction in a margin of safety. The RSG tubing has been shown to retain adequate structural and leakage integrity during normal, transient, and postulated accident conditions consistent with GDC 14, 15, 30, 31, and 32 of 10CFR50 Appendix A. The RSG tubing has been designed and evaluated consistent with the margins of safety specified in ASME Code Section III. The proposed program for periodic inservice inspection of the replacement steam generators monitors the integrity of the SG tubing to ensure that there is sufficient time to take proper and timely corrective action if any tube degradation is present. The proposed program is consistent with the Standard Technical Specifications.

The Unit 1 RCS dose equivalent I-131 limit is being raised upon installation of the RSGs to eliminate the compensatory lower limit that was adopted in conjunction with IPC for the existing Westinghouse D4 SGs. With the RCS activity limit returned to the Standard Technical Specification value of 1.0 μCi/gm, the assessment of postulated UFSAR Chapter 15 accidents (including SGTR and MSLB) has concluded that the calculated design basis doses presented in Chapter 15 are not adversely impacted by the RSGs. This ensures that the resulting 2-hour dose rates at the Byron and Braidwood site boundaries will not exceed an appropriately small fraction of 10CFR100 dose guideline values.

Limiting the applicability of TS provisions to a specific cycle or SG type are administrative changes in that they provide clarification consistent with current analyses.

Therefore, it is concluded that this change does not involve a significant reduction in a margin of safety with respect to plant safety as defined in the UFSAR or the Technical Specification.

ATTACHMENT D

ENVIRONMENTAL ASSESSMENT FOR
PROPOSED CHANGES TO APPENDIX A
TECHNICAL SPECIFICATIONS OF
FACILITY OPERATING LICENSES
NPF-37, NPF-66, 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. The proposed license amendment amends Technical Specifications (TS) 3/4.4.5, "Steam Generators," and TS 3/4.4.8, "Reactor Coolant System Specific Activity," for Byron Nuclear Power Station Units 1 and 2 (Byron) and Braidwood Nuclear Power Station Units 1 and 2 (Braidwood) to support steam generator (SG) replacement. The proposed amendment will limit application of the Interim Plugging Criteria to Unit 1 through Cycle 8 for Byron and Cycle 6 for Braidwood. The amendment also limits application of F* criteria to Westinghouse D4 SGs and the current sleeving methodologies to Westinghouse Model D4 and D5 SGs. The proposed amendment will also change the Unit 1 reactor coolant dose equivalent I-131 limit from 0.35 microCuries per gram (µCi/gm) to 1.0 µCi/gm after Cycle 8 for Byron and after Cycle 7 for Braidwood.

2. This proposed license amendment request involves no significant hazards considerations as demonstrated in Attachment C;

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.