ATTACHMENT C-1

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

BRAIDWOOD STATION UNITS 1 & 2
REVISED PAGES:

3/4 4-13*
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*NOTE: THESE PAGES HAVE NO CHANGES BUT ARE INCLUDED FOR CONTINUITY.

k:nla:bybwd:ipc3aup3.wpf:45

3/4.4.5 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

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

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas;

b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

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

SURVEILLANCE REQUIREMENTS (Continued)

1) All tubes that previously had detectable tube wall penetrations greater than 20% that have not been plugged or sleeved in the affected area, and all tubes that previously had detectable sleeve wall penetrations that have not been plugged,

2) Tubes in those areas where experience has indicated potential problems,

3) At least 3% of the total number of sleeved tubes in all four steam generators or all of the sleeved tubes in the generator chosen for the inspection program, whichever is less. These inspections will include both the tube and the sleeve, and

Insert A

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.

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.

replace with 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)

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.

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.

SURVEILLANCE REQUIREMENTS (Continued)

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

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections, not including the 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;

Replace with Insert C

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;~~

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 Cycle 6, 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 in a Westinghouse Technical Report currently approved by the NRC, subject to the limitations and restrictions as noted by the NRC staff, or

   b) Kinetic welded sleeving as described in a Babcock & Wilcox Nuclear Technologies Technical Report currently aproved by the NRC, 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.

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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) <u>F^* Distance</u> 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) <u>F^* Tube</u> 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 <u>Reports</u>

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.

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(4.4.5.4.a.11)

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.

    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

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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

INSERT D
(continued)

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.

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.

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

~~1. Listing of the applicable tubes,~~

~~2. Location (applicable intersections per tube) and extent of degradation (voltage), and~~

~~3. Projected Steam Line Break (MSLB) Leakage.~~

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)

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.

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** 1

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. 1

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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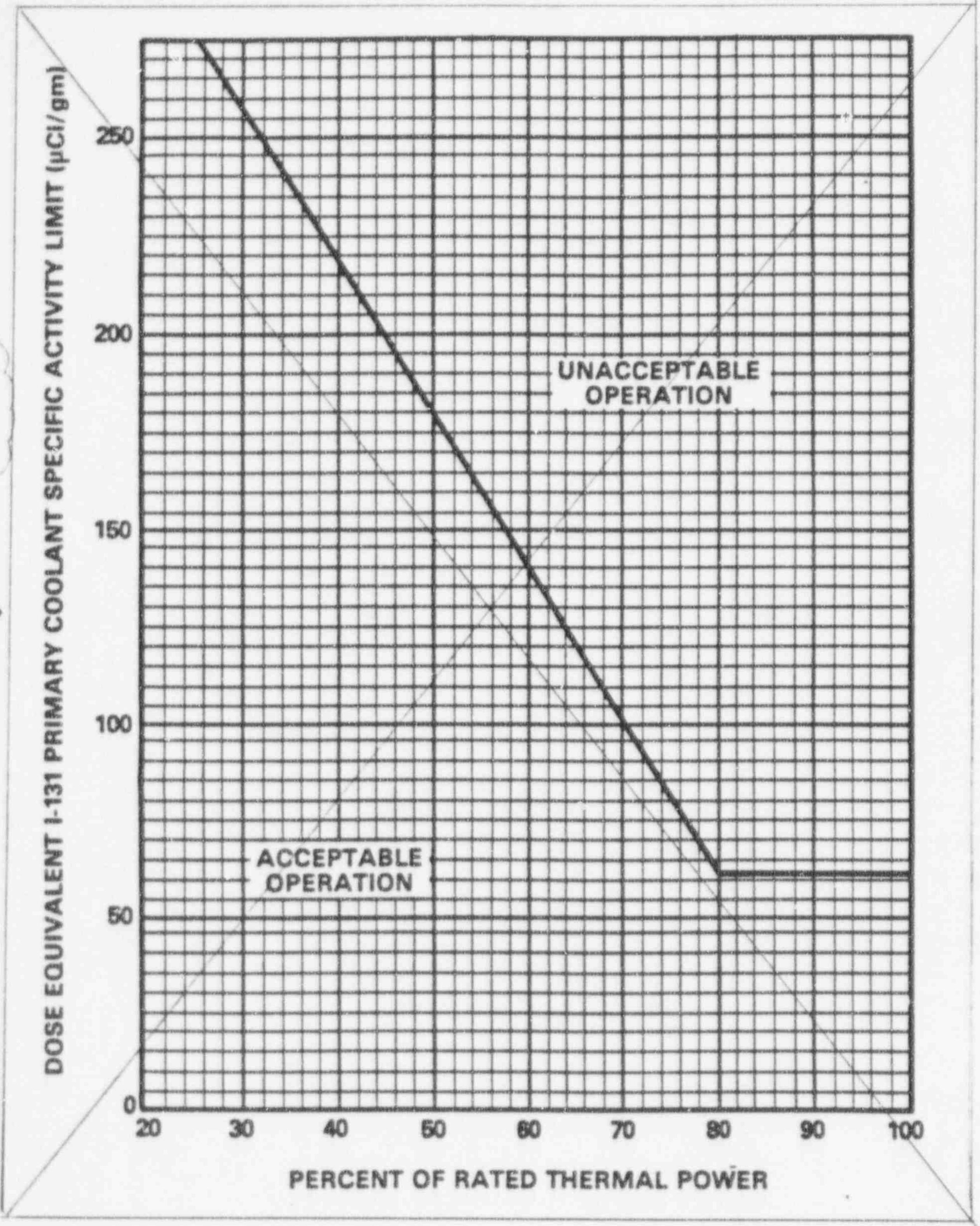

FIGURE 3.4-1

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

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(Figure 3.4-1)

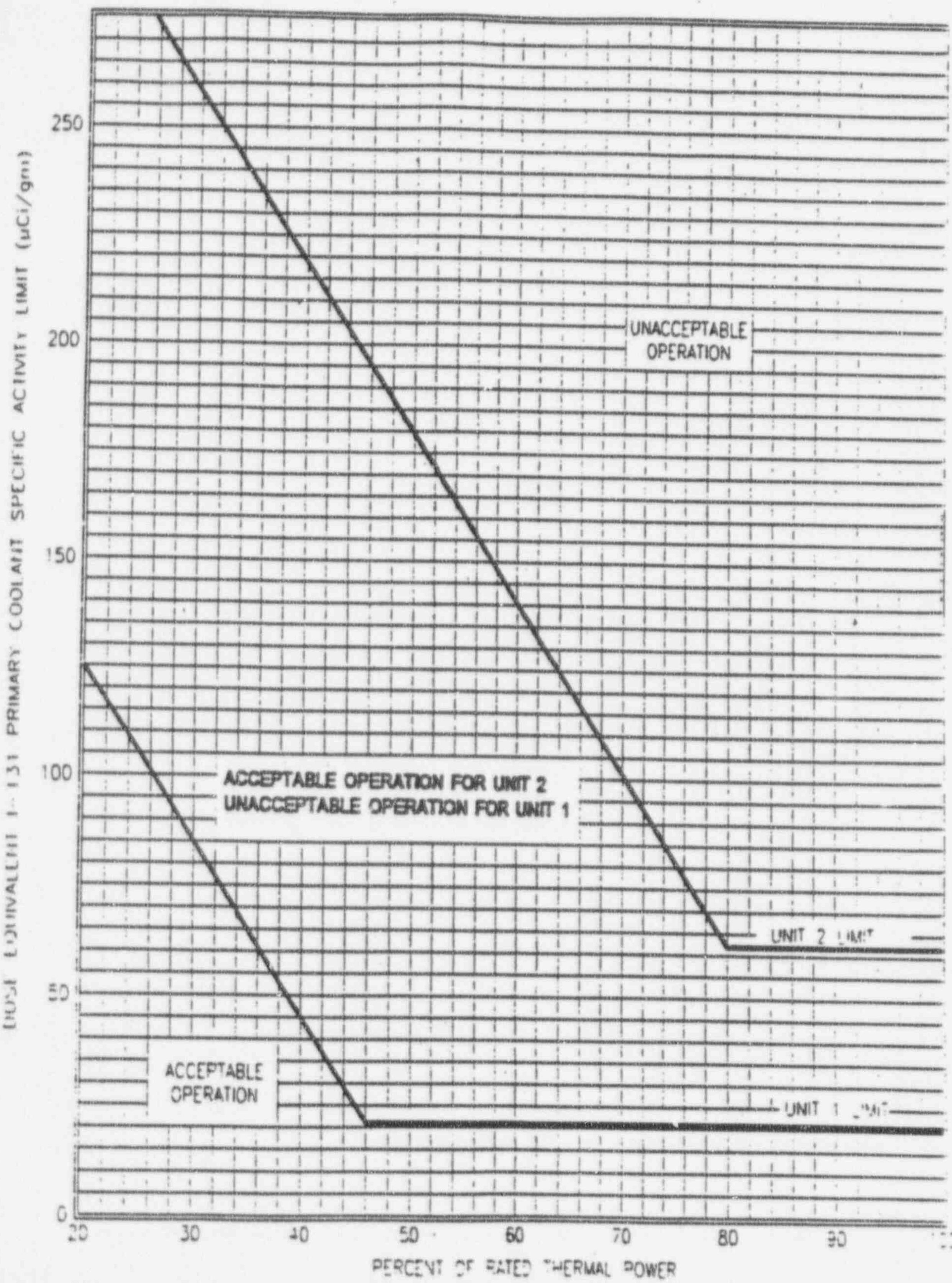

TABLE 4.4-4

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)

Insert 6

~~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 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 Steam Line Break Conditions

INSERT G

(Bases 3/4.4.5)

The voltage-based repair limits for Unit 1 in Surveillance Requirement (SR) 4.4.5 implement t' e 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).

INSERT G
(continued)

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.

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.

ATTACHMENT C-2

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

3/4 4-13*
3/4 4-14
3/4 4-15*
3/4 4-16
3/4 4-17
3/4 4-17a
3/4 4-17b
3/4 4-27
3/4 4-28
3/4 4-29
3/4 4-30
3/4 4-31
B 3/4 4-3*
B 3/4 4-3a

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

INSERTS A

Not Used

INSERT B

Not Used

INSERT C

Not Used

3/4.4.5 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

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

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas;

b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

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

SURVEILLANCE REQUIREMENTS (Continued)

1) All tubes that previously had detectable tube wall penetrations gre.ter 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.

indications

voltage-based repair

5) For Unit 1, ~~tubes~~ left in service as a result of application of the tube support plate ~~plugging~~ criteria shall be inspected by bobbin coil probe during all future outages.

refueling

6) 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.

steam generator tube/

through &

percent

repair

and cold-leg

lowest cold leg

d. For Unit 1, Cycle ~~7~~ 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. The determination of the tube support plate intersections having ODSCC indications shall be based on the performance of at least a 20% random sampling of tubes inspected over their full length.

percent

known outside

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 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 through Cycle ~~7~~ 8, this definition does not apply to tube support plate intersections for which the voltage-based ~~plugging~~ repair 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;

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

SURVEILLANCE REQUIREMENTS (Continued)

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

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

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

   b) Kinetic welded sleeving as described in a Babcock & Wilcox Nuclear Technologies Technical Report currently approved by the NRC, 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.

Replace with Insert D

~~11) For Unit 1 Cycle 7, the Tube Support Plate Interim Plugging Criteria Limit is used for the disposition of a steam generator tube for continued service that is experiencing outer diameter stress corrosion cracking confined within the thickness of the tube support plates. At tube support plate intersections, the repair limit is based on maintaining steam generator tube serviceability as described below:~~

   ~~a) Degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with bobbin voltage less than or equal to 1.0 volt will be allowed to remain in service.~~

   ~~b) Degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with bobbin voltage greater than 1.0 volt will be repaired or plugged except as noted in 4.4.5.4.a.11)c) below.~~

   ~~c) 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 1.0 volt but less than or equal to 2.7 volts may remain in service if a rotating pancake coil inspection does not detect degradation. Indications of outside diameter stress corrosion cracking degradation with bobbin voltage greater than 2.7 volts 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.

e) If, as a result of leakage due to a mechanism other than ODSCC at the tube support plate intersection, or some other cause, an unscheduled mid-cycle inspection is performed, the following repair criteria apply instead of 4.4.5.4.11)c). If bobbin voltage is within expected limits, the indication can remain in service. The expected bobbin voltage limits are determined from the following equation:

$$V < \frac{\frac{\Delta t}{CL}(V_{SL} - V_{BOC}) + V_{BOC}}{1 + (0.2)(\frac{\Delta t}{CL})}$$

where:
V = measured voltage
V_{BOC} = voltage at BOC
Δt = time period of operation to unscheduled outage
CL = cycle length (full operating cycle length where operating cycle is the time between two scheduled steam generator inspections)
V_{SL} = 4.5 volts

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

INSERT D

(4.4.5.4.a.11)

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 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.

    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

INSERT D
(continued)

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

INSERT D
(continued)

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.

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 Unit 1 Cycle 7, implementation of the voltage-based repair criteria to tube support plate intersections, reports to the Staff shall be made as follows:

Replace with Insert E

1) Notify the Staff prior to returning the steam generators to service should any of the following conditions arise:

a) If estimated leakage based on the actual measured end-of-cycle voltage distribution would have exceeded the leak limit (for postulated main steam line break utilizing licensing basis assumptions) during the previous operation cycle.

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

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

d) If the calculated conditional burst probability exceeds 1×10^{-2}, notify the NRC and provide an assessment of the safety significance of the occurrence.

2) The final results of the inspection and the tube integrity evaluation shall be reported to the Staff pursuant to Specification 6.9.2 within 90 days following restart.

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)

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 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.

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 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, 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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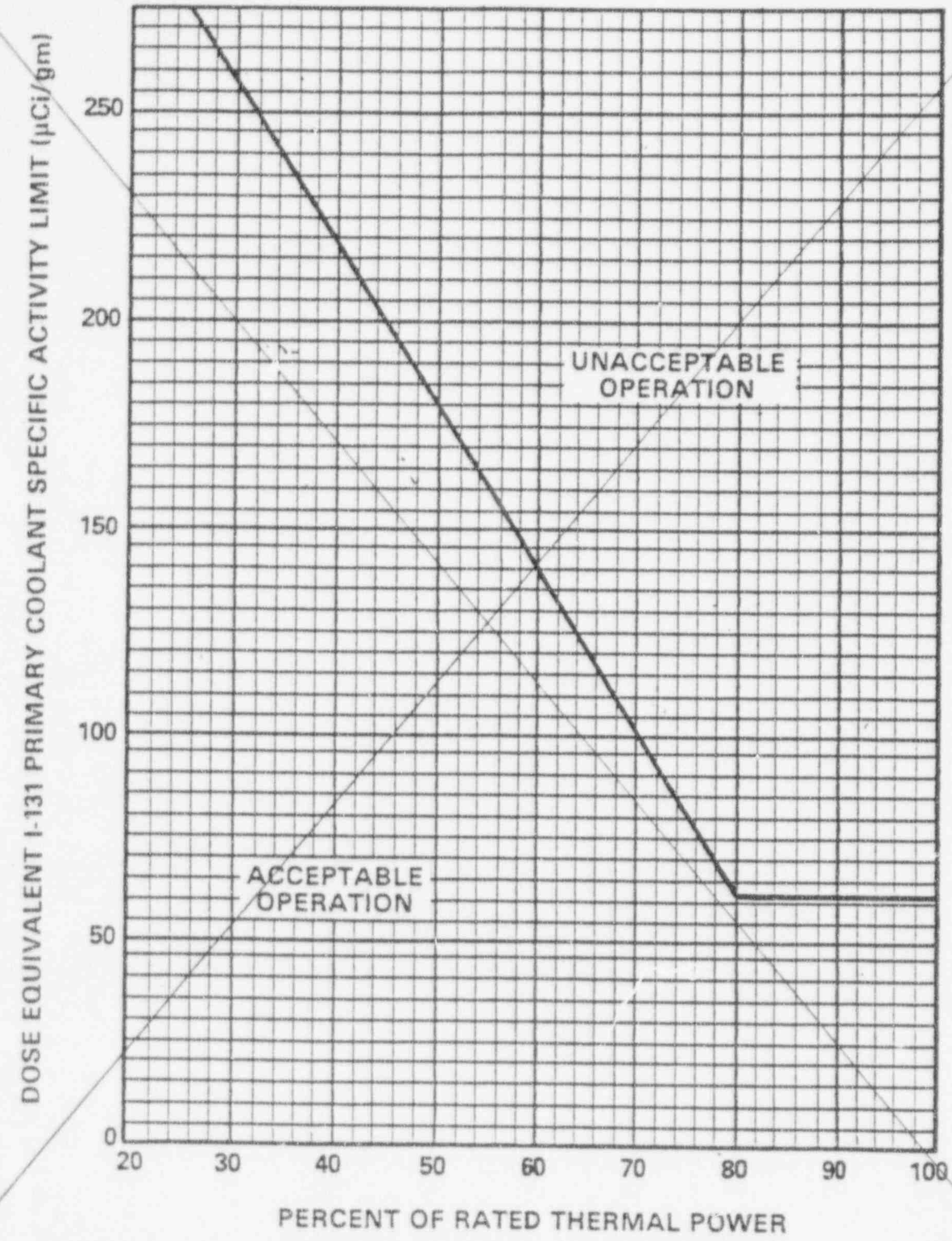

Figure 3.4-1 Dose Equivalent I-131 Reactor Coolant Specific Activity Limit Versus Percent of Rated Thermal Power with the Reactor Coolant Specific Activity > 1µCi/Gram Dose Equivalent I-131*

* For Unit 1, Reactor Coolant Specific Activity 0.35 µCi/Gram Dose Equivalent I-131

INSERT F

(Figure 3.4-1)

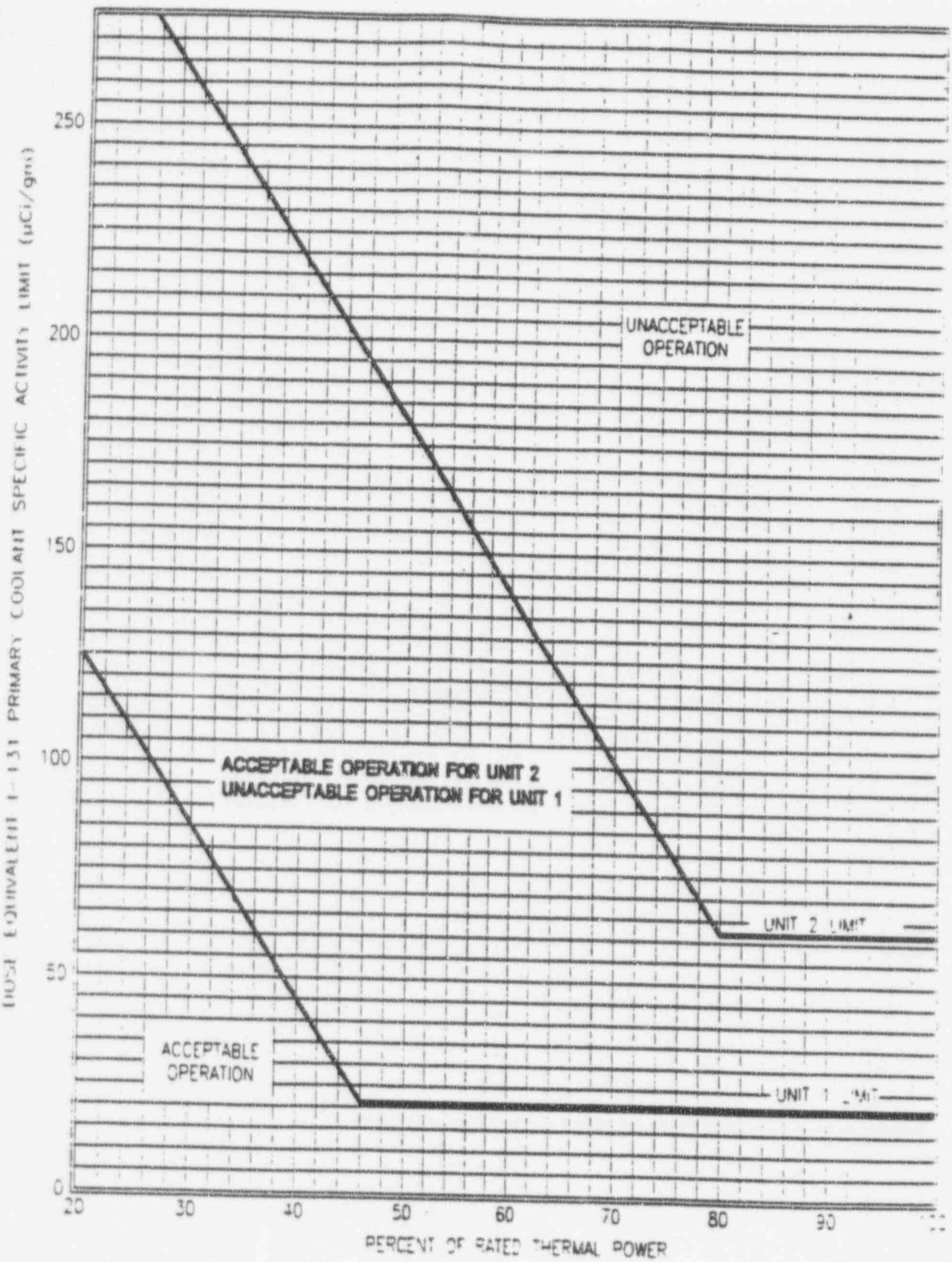

TABLE 4.4-4

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 xxxx 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.

xxxx 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 ejecters. 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)

INSERT (G) ~~For Unit 1 Cycle 7, 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 operating period may be adjusted to less than the full operating cycle to meet the~~ The maximum site allowable primary-to-secondary leakage limit for End of Cycle Main Steam Line Break conditions~~. The leakage limit, 12.8 gpm,~~ 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^*.

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.

INSERT G

(Bases 3/4.4.5)

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).

INSERT G
(continued)

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 650^0F (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.

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.

ATTACHMENT D

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

ComEd proposes to amend Byron and Braidwood Technical Specification (TS) 3.4.5, "Steam Generators," the bases for TS 3.4.5, and TS 3.4.8, "Specific Activity."

The changes proposed to TS 3.4.5 will implement an increased voltage, bobbin coil probe, Steam Generator (SG) Tube Support Plate (TSP) Interim Plugging Criteria (IPC) limit for Outside Diameter Stress Corrosion Cracking (ODSCC)indications at the hot-leg TSP intersections for Braidwood Unit 1 Cycle 6 and Byron Unit 1 through Cycle 8.

The changes proposed to TS 3.4.8 involve reducing Reactor Coolant System (RCS) dose equivalent Iodine-131 (I-131) for Unit 1 at both Byron and Braidwood.

For Braidwood and Byron, additional changes are proposed to make the TS more consistent with the Model TS contained in Generic Letter 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking," August 3, 1995 (Generic Letter 95-05).

For Byron Unit 1 and Braidwood Unit 1, ComEd is requesting implementation of an IPC for ODSCC indications at hot-leg TSP intersections. This IPC will increase the current plugging criteria voltage for ODSCC occurring at hot-leg TSP intersections up to a maximum of 3.0 volts. Selected SG tubes will be expanded above and below the hot-leg TSP to limit TSP movement during a Main Steam Line Break (MSLB) to reduce tube burst probabilities to negligible levels.

For Byron, the equation for determining voltage acceptance criteria for an unscheduled outage is revised for conformance to Generic Letter 95-05. For Braidwood, the equation for mid-cycle unscheduled outage voltage acceptance criteria is added to the specification for conformance with Generic Letter 95-05.

Braidwood's probability of tube burst limit is decreased from $2.5x10^{-2}$ to $1.0x10^{-2}$ consistent with Generic Letter 95-05.

Byron and Braidwood will be adding footnotes to TS 3.4.8 and revising Figure 3.4-1 to reduce the Unit 1 dose equivalent I-131 limit from 1.0 microCurie per gram (µCi/gm) to 0.35 µCi/gm.

Finally, bases changes are being made to Braidwood and Byron TS in order to accurately reflect the changes made to the individual specifications.

During the processing of this amendment request, ComEd evaluated the experience of several domestic and foreign units with steam generator tube support plate degradation, eddy current signal distortions, and component misalignment. The degradation experienced by these other units include missing or damaged tube support plate sections and tube support plate cracking. The mechanisms suspected to cause the tube support plate degradations, signal distortions, and component misalignment have been evaluated for significance and impact on application of 3.0 volt hot-leg IPC repair limits at Byron Unit 1 and Braidwood Unit 1. As discussed in the No Significant Hazards Analysis below, these experiences are not applicable to Byron Unit 1 and Braidwood Unit 1.

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.

The previously evaluated accidents of interest are steam generator tube burst and main steam line break. Their potential impact on public health and safety due to the change in SG tube plugging criteria proposed in this amendment request is very low as discussed below. Tube burst related to the types of cracks under consideration is precluded during normal operating plant conditions since the tube support plates are adjacent to the degraded regions of the tube in the tube to tube support piate crevices.

During accident conditions, i.e., MSLB, the tubes and TSP may move relative to each other, which can expose a crack length portion to freespan conditions. Testing has shown that the burst pressure correlates to the crack length that is exposed to the freespan, regardless of the length that is still contained within the TSP bounds.

Therefore, a more appropriate methodology has been established for addressing leakage and burst considerations that is based on limiting potential TSP displacements during postulated MSLB events, thus reducing the freespan exposed crack length to minimal levels. The tube expansion process to be employed in conjunction with this TS change is designed to provide postulated TSP displacements that result in negligible tube burst probabilities due to the minimal freespan exposed crack lengths.

Thermal hydraulic modeling was used to determine TSP loading during MSLB conditions. A safety factor was conservatively applied to these loads to envelope the collective uncertainties in the analyses. Various operating conditions were evaluated and the most limiting operating condition was used in the analyses. Additional models were used to verify the thermal hydraulic results.

Assessment of the tube burst probability was based on a conservative assumption that all hot-leg TSP intersections (32,046) contained throughwall cracks equal to the postulated displacement and that the crack lengths were located within the boundaries of the TSP. Alternatively, it was assumed that all hot-leg TSP intersections contained throughwall cracks with length equal to the thickness of the TSP. The postulated TSP motion was conservatively assumed to be uniform and equal to the maximum displacement caiculated.

The total burst probability for all 32,046 throughwall indications given a uniform MSLB TSP displacement of 0.31" is calculated to be $1x10^{-5}$. This is a factor of 1000 less than the Generic Letter 95-05 burst probability limit of $1x10^{-2}$. Therefore, the functional design criteria for tube expansion is to limit the TSP motion to 0.31" or less. However, the design goal for tube expansion limits the TSP MSLB motion to less than 0.1", which results in a total tube burst probability of $1x10^{-10}$ for all 32,046 postulated throughwall indications. Additional tubes will be expanded to provide redundancy to the required expansions.

The structural limit for the hot-leg SG tube repair criteria with tube expansion is based on axial tensile loading requirements to preclude axial tensile severing of the tube. Axially oriented ODSCC does not significantly impact the axial tensile loading of the tube, therefore, the more limiting degradation mode with respect to affecting the tube structural limit at TSPs is cellular corrosion. Tensile tests that measure the force required to sever a tube with cellular corrosion and uncorroded cross sectional areas are used to establish the lower bound structural limit. Based upon these tests, a lower bound 95% confidence level structural voltage limit of 37 volts was established for cellular corrosion. This limit meets the Regulatory Guide (RG) 1.121, "Basis for Plugging Steam Generator Tubes," structural requirements based upon the normal operating pressure differential with a safety factor of 3.0 applied. Due to the limited database supporting this value, the structural limit was conservatively reduced to 20 volts. Accounting for voltage growth and Non-Destructive Examination (NDE) uncertainty, the full IPC upper limit exceeds 10 volts. However, for added conservatism a single voltage repair limit for hot-leg indications is specified in this request. All hot-leg indications with bobbin coil probe voltages greater than the hot-leg voltage repair limit will be plugged or repaired.

The freespan tube burst probability must be calculated for the cold-leg TSP indications to be within the requirements of Generic Letter 95-05. The freespan structural voltage limit is calculated using correlations from the database described in Generic Letter 95-05, with the inclusion of the recent Byron and Braidwood tube pull results. This structural limit is 4.75 volts. The lower voltage repair limit for cold-leg indications continues to be 1.0 volt. The upper voltage repair limit for cold-leg indications will be calculated in accordance with Generic Letter 95-05. Since flow distribution baffle indications are to be repaired to the 40% depth criteria, no leakage or burst analyses are required for these indications.

Per Generic Letter 95-05, MSLB leak rate and tube burst probability analyses are required prior to returning to power and are to be included in a report to the Nuclear Regulatory Commission (NRC) within 90 days of restart. If allowable limits on leak rates and burst probability are exceeded, the results are to be reported to the NRC and a safety assessment of the significance of the results is to be performed prior to returning the steam generators to service.

A postulated MSLB outside of containment but upstream of the Main Steam Isolation Valve (MSIV) represents the most limiting radiological condition relative to the IPC. The ODSCC voltage distribution at the TSP intersections are projected to the end of the cycle and MSLB leakage is calculated.

A site specific calculation has determined the allowable MSLB leakage limit for Byron Unit 1 and Braidwood Unit 1. These limits use the recommended dose equivalent Iodine-131 transient spiking values consistent with NUREG-0800, "Standard Review Plan" and ensure site boundary doses are within a small fraction of the 10 CFR 100 requirements. The projected MSLB leakage rate calculation methodology described in WCAP-14046, "Braidwood Unit 1 Technical Support for Cycle 5 Steam Generator Interim Plugging Criteria," and WCAP 14277, "SLB Leak Rate and Tube Burst Probability Analysis Methods for ODSCC at TSP Intersections," will be used to calculate end-of-cycle (EOC) leakage. This method includes a Probability Of Detection (POD) value of 0.6 for all voltage amplitude ranges and uses the accepted leak rate versus bobbin voltage correlation methodology (full Monte Carlo) for calculating leak rate, as described in Generic Letter 95-05. The database used for the leak and burst correlations is consistent with that described in Generic Letter 95-05 with the inclusion of the Byron Unit 1 and Braidwood Unit 1 tube pull results. The EOC voltage distribution is developed from the POD adjusted beginning-of-cycle (BOC) voltage distributions and uses Monte Carlo techniques to account for variances in growth and uncertainty.

The Electric Power Research Institute (EPRI) leak rate correlation has been used. It is based on free span indications that have burst pressures above the MSLB pressure differential. There is a low but finite probability that indications may burst at a pressure less than MSLB pressure. With limited TSP motion due to tube expansion, the tube is constrained by the TSP and tube burst is precluded. However, the flanks of the crack open up to contact the Inside Diameter (ID) of the TSP hole and result in a primary-to-secondary leak rate potentially exceeding that obtained from the EPRI correlation. This phenomenon is known as an Indication Restricted from Burst (IRB) condition.

ComEd has performed laboratory testing to determine the bounding leak rate obtainable in an IRB condition. The bounding leak rate value was then applied in a leak rate calculation methodology that accounts for the MSLB leak rate contribution from IRB indications to the total MSLB leak rate calculated as described above. Results indicate that the IRB contribution to the total leak rate value is negligible, however, ComEd will conservatively add a leakage contribution due to IRBs in addition to the leakage calculated in accordance with Generic Letter 95-05. When this is done, the dose at the site boundary resulting from the predicted leakage is shown to be a small fraction (less than 10%) of 10 CFR 100 limits.

Modification of the Byron and Braidwood Specifications for conformance with Generic Letter 95-05 requirements is primarily administrative and does not significantly increase the probability of any accidents previously evaluated. For Braidwood, the changes decrease the allowed burst probability from 2.5×10^{-2} to 1.0×10^{-2}. This change is in the conservative direction. Byron Station has previously incorporated this requirement.

In addition, defense in depth is provided by lowering the Unit 1 RCS dose equivalent I-131 limit from 1.0 μCi/gm to 0.35 μCi/gm. Based on current predictions of MSLB leakage at the time of SG replacement, the lower RCS dose equivalent I-131 limit also ensures that the resulting 2-hour dose rates at the Braidwood and Byron site boundaries will not exceed an appropriately small fraction of 10 CFR 100 dose guideline values.

For these reasons, an increase in the IPC voltage repair limit to a maximum of 3.0 volts for the hot-leg support plate intersections does not adversely affect steam generator tube integrity and results in acceptable dose consequences. By effectively eliminating tube burst at hot-leg TSP intersections, the likelihood of a tube rupture is substantially reduced and the probability of occurrence of an accident previously evaluated is reduced.

This conclusion is not affected by recent foreign and domestic plant SG experiences. As the following evaluation shows, these experiences are not relevant to Byron and Braidwood. A foreign unit detected eddy current signal detections in one area of the top tube support plate during a 1995 inspection. The steam generators had been chemically cleaned in 1992. Visual inspection showed that a small section of the top support plate had broken free and was resting next to the steam generator tube bundle wrapper. The support plate showed indications of metal loss. The chemical cleaning process used by the foreign unit was developed by the utility and differs significantly from the modified EPRI/SGOG process performed at Byron Unit 1 in 1994.

The foreign process, coupled with specific application of the process, resulted in tube support plate corrosion of up to 250 mils compared to a maximum of 2.16 mils (11 mils maximum allowed) measured at Byron. During the Byron eddy current inspection performed after the chemical cleaning, no distortion of the tube support plate signals was reported. Therefore, these differences in cleaning processes imply that this foreign experience is irrelevant to the effects of the chemical cleaning process on the TSPs at Byron.

A number of units have experienced TSP cracking associated with severe tube denting due to TSP corrosion at the tube to TSP crevice. WCAP 14273, Section 12.4, shows that a diametral reduction of 65 mils is required to develop stress levels above yield in the TSP ligaments at dented intersections. The bobbin voltage associated with a 1 mil radial dent is 20 to 25 volts.

Although, Byron Unit 1 and Braidwood Unit 1 have not seen corrosion induced denting, an appropriately sized bobbin probe will be used as a go/no-go gauge to assess hot-leg dents, if they occur in the future. If a tube has a dent at a hot-leg intersection that fails to pass the go/no-go test probe, cold-leg repair criteria will be applied to the affected tube and the adjacent tubes. In this way, any indications at these locations will be treated as free-span indications for the purposes of burst and leakage evaluation, which is bounded by the existing 1.0 volt IPC analysis. IPC repair limits will not be applied to tubes with dents > 5.0 volts since they could mask a 1.0 volt signal. Tubes with corrosion-induced dents > 5.0 volts and those tubes adjacent to such a tube will not be selected for tube expansion to preclude adverse effects of the failure of such a tube on limiting TSP displacement. Therefore, the denting experience at other plants is not relevant to Byron and Braidwood.

A foreign utility's steam generators have experienced cracking at the top tube support plate. The cause of the cracking appears to be the configuration of the single anti-rotation device, connected between the steam generator shell and wrapper, and the wrapper internals. The single anti-rotation device carries the full load associated with wrapper to shell motion. This rotational load is believed to be transferred to the TSP via the wrapper internals. The Byron/Braidwood Unit 1 steam generator design (D-4) uses three anti-rotation devices to spread the rotational load. The D-4 wrapper internals are configured such that this load is not directly transmitted to the TSP.

No top support plate cracking has been detected at Byron Unit 1 or Braidwood Unit 1 and very few (<1%) of the indications seen at Byron and Braidwood to date have been at the top TSP elevation.

Nevertheless, an analysis was performed to assess the impact of cracking of the top support plate. The results show an increase in top support plate deflection for a very limited number of tubes to greater than the 0.10" limit used in the 3.0 volt IPC analysis. The deflections of the lower support plates also increase, but remain within the 0.10" limit. Thus, hot-leg indications in a cracked top TSP continue to be bounded by the existing analysis. ComEd will develop an inspection plan for the SG internals to identify if indications detrimental to the load path exist. If the inspection determines that indications detrimental to the integrity of the load path necessary to support the 3 volt IPC are found, the results are to be reported to the NRC and a safety assessment of the significance of the results is to be performed prior to returning the steam generators to service.

A domestic utility reported several distorted TSP signals over the past three refueling outage tube inspections. It was determined that these signals were associated with the TSP geometry in an area where an access cover is welded into the TSP. These signal distortions are not attributed to TSP cracking or degradation. Since the distorted signals were due to TSP geometry which did not indicate or result in a defect of the TSP, there is no increase in the probability or consequences of an accident previously evaluated due to Byron Unit 1 and Braidwood Unit 1 steam generator TSP geometries which may result in distorted eddy current signals.

One foreign unit observed a dislocation of the tube bundle wrapper when they were unable to pass sludge lancing equipment through a handhole in the wrapper. The dislocation appears to be a result of improper attachment of the wrapper to the support structure. Steam generator sludge lance operations have been successfully performed on Byron Unit 1 and Braidwood Unit 1 which indicates that no problem with wrapper attachment exists. The foreign unit's wrapper support design is significantly different than that used on Byron Unit 1 and Braidwood Unit 1. Therefore, a similar wrapper dislocation will not occur and the foreign experience is not applicable to Byron and Braidwood.

Therefore, the proposed amendment does not result in any significant increase in the probability or consequences of an accident previously evaluated within the Byron Unit 1 and Braidwood Unit 1 Updated Final Safety Analysis Report (UFSAR).

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

Implementation of the proposed steam generator tube plugging criteria with tube expansion 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 of the region of the tube support plate elevations as ODSCC does not extend beyond the thickness of the tube support plates. Neither a single nor multiple tube rupture event would be expected in a steam generator in which the plugging criteria has been applied.

The tube burst assessment involves a Monte Carlo simulation of the site specific voltage distribution to generate a total burst probability that includes the summation of the probabilities of 1 tube bursting, 2 tubes bursting, etc. For the hot-leg TSP intersections, the maximum total probability of burst, by design, is estimated to be $1x10^{-10}$ with all tube expansions functional.

Accounting for the unlikely event of expansion failures, a sufficient number of redundant expansions exist to ensure that the burst probability remains below $1x10^{-5}$. This includes the conservative assumption that all 32,046 hot-leg TSP intersections contain throughwall indications. This level of burst probability is considered to be negligible when compared to the Generic Letter 95-05 limit of $1x10^{-2}$.

In addressing the combined effects of Loss Of Coolant Accident (LOCA) + Safe Shutdown Earthquake (SSE) on the SG as required by General Design Criteria (GDC) 2, it has been determined that tube collapse may occur in the steam generators at some plants. The tube support plates may become deformed as a result of lateral loads at the wedge supports located at the periphery of the plate due to the combined effects of the LOCA rarefaction wave and SSE loadings. The resulting pressure differential on the deformed tubes may cause some of the tubes to collapse. 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 throughwall cracks in tubes could progress to throughwall cracks during tube deformation or collapse. The tubes subject to collapse have been identified via a plant specific analysis and excluded from application of the voltage-based criteria. This analysis is included in revision 3 to WCAP-14046 which was submitted to the NRC June 19, 1995.

ComEd will continue to apply a maximum primary-to-secondary leakage limit of 150 gallons per day (gpd) through any one SG at Byron and Braidwood to help preclude the potential for excessive leakage during all plant conditions. The RG 1.121 criterion for establishing operational leakage limits that require plant shutdown are based on detecting a free span crack prior to resulting in primary-to-secondary operational leakage which could potentially develop into a tube rupture during faulted 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.

Tube burst is precluded during normal operation due to the proximity of the TSP to the tube and during a postulated MSLB event with tube expansion. The 150 gpd limit provides a conservative limit for plant shutdown prior to reaching critical crack lengths should significant crack extension unexpectedly occur outside the thickness of the TSP.

Lowering the Unit 1 RCS dose equivalent I-131 limit from 1.0 µCi/gm to 0.35 µCi/gm is conservative and provides a defense in depth approach to implementation of this IPC.

Based on current predictions of MSLB leakage at the time of SG replacement, the lower RCS dose equivalent I-131 limit also ensures that the resulting 2-hour dose rates at the Braidwood and Byron site boundaries will not exceed an appropriately small fraction of 10 CFR 100 dose guideline values.

Modification of the Byron and Braidwood Specifications for conformance with Generic Letter 95-05 requirements is primarily administrative and will not alter the plant design basis. For Braidwood, the decrease in the allowed burst probability from 2.5×10^{-2} to 1.0×10^{-2} is conservative. Byron Station has previously incorporated this requirement.

With implementation of an increased IPC voltage repair limit (up to a maximum of 3.0 volts) using tube expansion for the hot-leg support plate intersections, steam generator tube integrity continues to be maintained through inservice inspection, tube repair and primary-to-secondary leakage monitoring. By effectively eliminating tube burst at hot-leg TSP intersections, the potential for multiple tube ruptures is essentially eliminated. Therefore, the possibility of a new or different kind of accident from any previously evaluated is not created.

ComEd has evaluated industry experiences with TSP degradation, eddy current signal distortions, and component misalignment. Eddy current signal distortions due to TSP geometry are not indicative of TSP degradation and do not result in any kind of accident.

The component misalignment experienced by one unit is not applicable to Byron Unit 1 or Braidwood Unit 1 and, thus, will not result in any kind of accident. Specific limitations, as discussed above, will be applied to indications at hot-leg intersections which contain dents. These limitations ensure that integrity of the SG tubes is maintained consistent with current analyses should tube denting or TSP cracking occur. Application of the 3.0 volt hot-leg IPC to Byron Unit 1 and Braidwood Unit 1, with the limitations specified, will not result in 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.

The use of the voltage-based, bobbin coil, tube support plate elevation plugging criteria with tube expansion at Byron Unit 1 and Braidwood Unit 1 is demonstrated to maintain steam generator tube integrity commensurate with the criteria of RG 1.121. RG 1.121 describes a method acceptable to the NRC staff for meeting GDC 14, 15, 31, and 32 by reducing the probability or the consequences of steam generator tube rupture.

This is accomplished by determining an eddy current inspection voltage value which represents a limit for leaving a SG tube in service. Tubes with ODSCC voltage indications beyond this limiting value must be removed from service by plugging or repaired by sleeving. Upon implementation of an increased IPC voltage repair limit (up to a maximum of 3.0 volts) for the hot-leg, even under the worst case conditions, the occurrence of ODSCC at the tube support plate elevations has been evaluated and shown not to present a credible potential for a steam generator tube rupture event during normal or faulted plant conditions. The End Of Cycle (EOC) distribution of crack indications at the tube support plate elevations will be confirmed to result in acceptable primary-to-secondary leakage during all plant conditions such that radiological consequences are not adversely impacted.

Addressing RG 1.83 considerations, implementation of the increased hot-leg tube support plate intersection bobbin coil voltage-based repair criteria is supplemented by enhanced eddy current inspection guidelines to provide consistency in voltage normalization and a 100% eddy current inspection sample size at the affected tube support plate elevations.

For the leak and burst assessments, the population of indications in the voltage distribution is dependant on the POD function. The purpose of the POD function is to account for indications that may not be identified by the data analyst.

In implementing this proposed IPC, ComEd will use the conservative Generic Letter 95-05 POD value of 0.6 for all voltage amplitude ranges.

Lowering the Unit 1 RCS dose equivalent I-131 limit from 1.0 μCi/gm to 0.35 μCi/gm is conservative and provides a defense in depth approach to implementation of this IPC. Based on current predictions of MSLB leakage at the time of SG replacement, the lower RCS dose equivalent I-131 limit also ensures that the resulting 2-hour dose rates at the Braidwood and Byron site boundaries will not exceed an appropriately small fraction of 10 CFR 100 dose guideline values.

Modification of the Byron and Braidwood Specifications for conformance with the Generic Letter 95-05 requirements is primarily administrative and will not reduce any safety margins. For Braidwood, the decrease in the allowed burst probability from 2.5×10^{-2} to 1.0×10^{-2} is conservative. Byron Station has previously incorporated this requirement.

Implementation of the tube support plate elevation repair limits will decrease the number of tubes which must be repaired. The installation of steam generator tube plugs or sleeves reduces the RCS flow margin. Thus, implementation of the interim plugging criteria will maintain the margin of flow that would otherwise be reduced in the event of increased tube plugging.

As discussed previously, ComEd has evaluated industry experiences with TSP degradation, eddy current signal distortions, and component misalignment. Eddy current signal distortions at tube support plates will be evaluated to attempt determination of the cause of the distortion. A signal distortion alone will not result in reduction in the margin of safety. The foreign unit that experienced the component misalignment was of a significantly different design than the Byron Unit 1 and Braidwood Unit 1 steam generators. Analysis of the design differences shows that component misalignment of that type is not applicable to Byron Unit 1 or Braidwood Unit 1 and, thus, will not result in a reduction in the margin of safety.

Specific limitations, as discussed previously, will be applied to indications at hot-leg intersections which contain dents. These limitations conservatively treat indications as freespan to ensure that integrity of the SG tubes is maintained consistent with current analyses should tube denting or TSP cracking occur. Also, tubes with large dents (> 5.0 volts) and tubes adjacent to these dented tubes will not be used for tube expansion to ensure success of tube support plate motion limitation under accident conditions. Application of the 3.0 volt hot-leg IPC to Byron Unit 1 and Braidwood Unit 1, with the limitations specified, will not result in a reduction in a margin of safety.

Thus, the implementation of this amendment does not result in a significant reduction in a margin of safety.

Therefore, based on the above evaluation, ComEd has concluded that these changes involve no significant hazards considerations.

ATTACHMENT E

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. This proposed license amendment request changes Byron and Braidwood Technical Specification (TS) 3.4.5, "Steam Generators," the bases for TS 3.4.5, and TS 3.4.8, "Specific Activity."

   The changes proposed to TS 3.4.5 will implement a 3.0 volt bobbin coil probe, Steam Generator (SG) Tube Support Plate (TSP) Interim Plugging Criteria (IPC) limit for Outside Diameter Stress Corrosion Cracking (ODSCC)indications at the hot-leg TSP intersections. A 1.0 volt IPC will be applied to ODSCC indications at the cold-leg TSP intersections.

   The changes proposed to TS 3.4.8 involve reducing Reactor Coolant System (RCS) dose equivalent Iodine-131 (I-131) for Byron Unit 1 and Braidwood Unit 1.

   Additional changes are proposed to make the Byron Unit 1 and Braidwood Unit 1 TS more consistent with the Model TS contained 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).

2. This proposed license amendment request involves no significant hazards considerations;

3. There is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite; and

4. There is no significant increase in individual or cumulative occupational radiation exposure.

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

ATTACHMENT F

INDICATION RESTRICTED FROM BURST LEAK RATE CALCULATION METHODOLOGY

Commonwealth Edison and the Electric Power Research Institute (EPRI) have conducted a program to evaluate the impact of outside diameter stress corrosion cracking (ODSCC) tube indications in the support plate crevice area, which are unable to burst, regardless of crack geometry, because of tube support plate constraint. These indications are known as indications restricted from burst (IRBs). The objective of the program was to define a bounding leak rate for such indications and to use that bounding leak rate in the appropriate leak rate calculation under main steam line break (MSLB) conditions for an end-of-cycle (EOC) distribution of indications in the support plate.

The leak rate calculation to be used for the Byron Unit 1 and Braidwood Unit 1 3.0 volt Interim Plugging Criteria (IPC) uses the freespan leakage correlations developed by the Electric Power Research Institute and defined 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). To assure that IRBs are conservatively included in this calculation, a leakage term is substituted for the freespan leakage for those indications which may leak at a higher rate under MSLB conditions. The EOC voltage distribution is predicted using a beginning-of-cycle voltage distribution, a Probability of Detection (POD) of 0.6, and by applying a growth rate, as defined in Generic Letter 95-05. The EOC leak rate is then predicted by calculating the freespan leakage for the EOC distribution indications that are not IRBs. The IRB leakage is determined by the probability of burst correlation in Generic Letter 95-05, and, for those indications predicted to burst over the entire range of voltages, a bounding IRB leak rate is added.

Calculations completed by ComEd indicate that the impact of IRBs on the EOC leak rate calculation is not significant. However, ComEd has chosen to use the freespan leak rate plus IRB leak rate methodology to more directly include IRBs in the calculation.

Further information concerning this leak rate model is contained in ComEd's response to the Nuclear Regulatory Commission's Request For Additional Information question 39. This response is documented in the July 21, 1995 letter from D. Saccomando to the Office of Nuclear Reactor Regulation "Response to Request for Additional Information Pertaining to the Application for Amendment to Facility Operating License: Byron Nuclear Power Station, Unit 1 and 2 NPF-37/66; NRC Docket Nos. 50-454/455, Braidwood Nuclear Power Station Unit 1 and 2 NPF-72/77; NRC Docket Nos. 50-456/457."

To address TSP cracking issues identified in other plants, and SG MSLB load path integrity issues, ComEd will develop a SG internals inspection plan. This inspection plan will be provided to the NRC in a separate document. Results of this inspection will be reported to the NRC in accordance with the proposed reporting requirements section of this amendment request.