UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-454

BYRON STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 77
License No. NPF-37

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated September 1, 1995, (which superseded the prior amendment requests dated February 13, and July 7, 1995) and as supplemented on January 28, February 7, February 13, March 15, March 20 (two letters), April 3, April 12, April 21, May 25, June 19, June 20, June 30, July 21 (two letters), July 28, July 31, August 14 (two letter), August 25 (two letters), September 1 (two letters), September 2, September 4, September 8, September 15, September 19, September 20, September 22, October 3, October 7, October 11 (two letters), October 13 (three letters), October 23 and October 26, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-37 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 77 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

M. D. Lynch, Senior Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 9, 1995

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-455

BYRON STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 77
License No. NPF-66

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated September 1, 1995, (which superseded the prior amendment requests dated February 13, and July 7, 1995) and as supplemented on January 28, February 7, February 13, March 15, March 20 (two letters), April 3, April 12, April 21, May 25, June 19, June 20, June 30, July 21 (two letters), July 28, July 31, August 14 (two letter), August 25 (two letters), September 1 (two letters), September 2, September 4, September 8, September 15, September 19, September 20, September 22, October 3, October 7, October 11 (two letters), October 13 (three letters), October 23 and October 26, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-66 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A (NUREG-1113), as revised through Amendment No. 77 and revised by Attachment 2 to NPF-66, and the Environmental Protection Plan contained in Appendix B, both of which were attached to License No. NPF-37, dated February 14, 1985, are hereby incorporated into this license. Attachment 2 contains a revision to Appendix A which is hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

M. D. Lynch, Senior Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 9, 1995

ATTACHMENT TO LICENSE AMENDMENT NOS. 77 AND 77

FACILITY OPERATING LICENSE NOS. NPF-37 AND NPF-66

DOCKET NOS. STN 50-454 AND STN 50-455

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change. The page indicated by an asterisk is provided for convenience only.

Remove Pages	Insert Pages
3/4 4-14	3/4 4-14
--	3/4 4-14a
3/4 4-16	3/4 4-16
3/4 4-17	3/4 4-17
3/4 4-17a	3/4 4-17a
3/4 4-17b	3/4 4-17b
--	3/4 4-17c
--	3/4 4-17d
3/4 4-27	3/4 4-27
3/4 4-28	3/4 4-28
3/4 4-29	3/4 4-29
3/4 4-30	3/4 4-30
3/4 4-31	3/4 4-31
*3/4 4-32	*3/4 4-32
B 3/4 4-3a	B 3/4 4-3a
--	B 3/4 4-3b

SURVEILLANCE REQUIREMENTS (Continued)

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

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

3) At least 3 percent 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.

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.

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.

d. For Unit 1, through Cycle 8, implementation of the steam generator tube/tube support plate repair criteria requires a 100 percent bobbin coil inspection for hot-leg and cold-leg tube support plate intersections down to the lowest cold-leg tube support plate with known outside diameter stress corrosion cracking (ODSCC) indications. The determination of the lowest cold-leg tube support plate intersections having ODSCC indications shall be based on the performance of at least a 20 percent random sampling of tubes inspected over their full length.

SURVEILLANCE REQUIREMENTS (Continued)

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

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 steamline 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 percent 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 percent 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 percent 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 8, 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;

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

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.

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

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

f) If an unscheduled mid-cycle inspection is performed, the following mid-cycle repair limits apply instead of the limits identified in 4.4.5.4.a.11.a, 4.4.5.4.a.11.b and 4.4.5.4.a.11.d for outside diameter stress corrosion cracking indications occurring in the steam generator cold-legs. For outside diameter stress corrosion cracking indications occurring in the steam generator hot-legs, the limits in 4.4.5.4.a.11.a and 4.4.5.4.a.11.c apply. The mid-cycle repair limits are determined from the following equations:

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

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

Where:

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

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

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

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

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.

SURVEILLANCE REQUIREMENTS (Continued)

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

4.4.5.5 Reports

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged or repaired in each steam generator shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2;

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following the completion of the inspection. This Special Report shall include:

   1) Number and extent of tubes inspected,

   2) Location and percent of wall-thickness penetration for each indication of an imperfection, and

   3) Identification of tubes plugged or repaired.

c. Results of steam generator tube inspections which fall into Category C-3 shall be reported in a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days and prior to resumption of plant operation. This report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

d. For implementation of the voltage based repair criteria to tube support plate intersections for Unit 1 through Cycle 8, notify the staff prior to returning the steam generators to service should any of the following conditions arise:

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

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

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

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

SURVEILLANCE REQUIREMENTS (Continued)

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

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

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.

3/4.4.8 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

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

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

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

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

ACTION:

MODES 1, 2 and 3*:

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

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

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

**For Unit 1, 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.

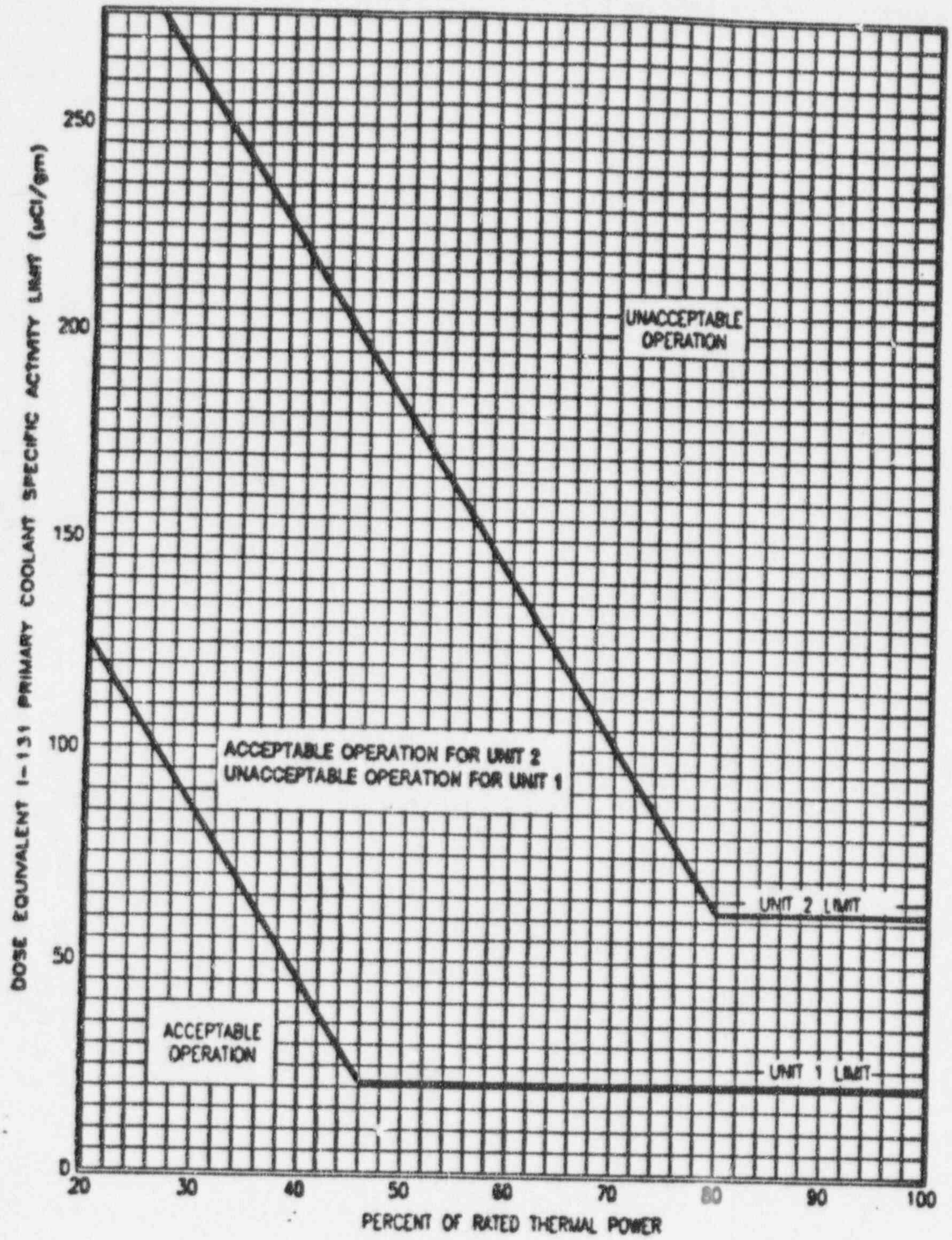

FIGURE 3.4-1
DOSE EQUIVALENT I-131 REACTOR COOLANT SPECIFIC ACTIVITY LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE REACTOR COOLANT SPECIFIC ACTIVITY >1µCi/GRAM DOSE EQUIVALENT I-131*

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

TABLE 4.4-4

REACTOR COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

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

TABLE 4.4-4 (Continued)

TABLE NOTATIONS

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

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

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

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

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

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2a and 3.4-3a for Unit 1 (Figures 3.4-2b and 3.4-3b for Unit 2) during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup of 100°F in any 1-hour period,

b. A maximum cooldown of 100°F in any 1-hour period, and

c. A maximum temperature change of less than or equal to 10°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, as required by 10 CFR Part 50, Appendix H, in accordance with the schedule in Table 4.4-5. The results of these examinations shall be used to update Figures 3.4-2, 3.4-3, and 3.4-4.

3/4.4.5 STEAM GENERATORS (Continued)

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

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

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

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

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

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

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

3/4.4.5 STEAM GENERATORS (Continued)

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

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

The maximum site allowable primary-to-secondary leakage limit for end-of-cycle main steamline break conditions includes the accident leakage from IPC in addition to the accident leakage from F^* on the faulted steam generator and the operational leakage limit of Specification 3.4.6.2.c. The operational leakage limit of Specification 3.4.6.2.c in each of the three remaining intact steam generators shall include the operational leakage from F^*.

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.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-456

BRAIDWOOD STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 69
License No. NPF-72

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated September 1, 1995, (which superseded the prior amendment requests dated February 13, and July 7, 1995) and as supplemented on January 28, February 7, February 13, March 15, March 20 (two letters), April 3, April 12, April 21, May 25, June 19, June 20, June 30, July 21 (two letters), July 28, July 31, August 14 (two letter), August 25 (two letters), September 1 (two letters), September 2, September 4, September 8, September 15, September 19, September 20, September 22, October 3, October 7, October 11 (two letters), October 13 (three letters), October 23 and October 26, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-72 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 69 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

M. D. Lynch

M. D. Lynch, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: November 9, 1995

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-457

BRAIDWOOD STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 69
License No. NPF-77

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated September 1, 1995, (which superseded the prior amendment requests dated February 13, and July 7, 1995) and as supplemented on January 28, February 7, February 13, March 15, March 20 (two letters), April 3, April 12, April 21, May 25, June 19, June 20, June 30, July 21 (two letters), July 28, July 31, August 14 (two letter), August 25 (two letters), September 1 (two letters), September 2, September 4, September 8, September 15, September 19, September 20, September 22, October 3, October 7, October 11 (two letters), October 13 (three letters), October 23 and October 26, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in te attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-77 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 69 and the Environmental Protection Plan contained in Appendix B, both of which were attached to License No. NPF-72, dated July 2, 1987, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date if its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

M. D. Lynch, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 9, 1995

ATTACHMENT TO LICENSE AMENDMENT NOS. 69 AND 69

FACILITY OPERATING LICENSE NOS. NPF-72 AND NPF-77

DOCKET NOS. STN 50-456 AND STN 50-457

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

Remove Pages	Insert Pages
3/4 4-14	3/4 4-14
3/4 4-14a	3/4 4-14a
3/4 4-16	3/4 4-16
3/4 4-17	3/4 4-17
3/4 4-17a	3/4 4-17a
3/4 4-17b	3/4 4-17b
--	3/4 4-17c
--	3/4 4-17d
3/4 4-27	3/4 4-27
3/4 4-28	3/4 4-28
3/4 4-29	3/4 4-29
3/4 4-30	3/4 4-30
3/4 4-31	3/4 4-31
B 3/4 4-3a	B 3/4 4-3a
--	B 3/4 4-3b

SURVEILLANCE REQUIREMENTS (Continued)

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

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

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

4) A tube inspection (pursuant to Specification 4.4.5.4a.8) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

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.

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.

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.

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

SURVEILLANCE REQUIREMENTS (Continued)

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.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.4 Acceptance Criteria

a. As used in this specification:

1) Imperfection means an exception to the dimensions, finish or contour of a tube or sleeve from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube or sleeve wall thickness, if detectable, may be considered as imperfections;

2) Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube or sleeve;

3) Degraded Tube means a tube or sleeve containing unrepaired imperfections greater than or equal to 20% of the nominal tube or sleeve wall thickness caused by degradation;

4) % Degradation means the percentage of the tube or sleeve wall thickness affected or removed by degradation;

5) Defect means an imperfection of such severity that it exceeds the plugging or repair limit. A tube or sleeve containing an unrepaired defect is defective;

6) Plugging or Repair Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging or repaired by sleeving in the affected area. The plugging or repair limit imperfection depth is equal to 40% of the nominal wall thickness. For Unit 1, this definition does not apply to defects in the tubesheet that meet the criteria for an F* tube. For Unit 1 Cycle 6, this definition does not apply to 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;

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

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.

11) For Unit 1 Cycle 6, the Tube Support Plate Plugging Limit is used for the disposition of an alloy 600 steam generator tube for continued service that is experiencing predominantly axially oriented outside diameter stress corrosion cracking confined within the thickness of the tube support plates. At tube support plate intersections, the plugging (repair) limit is based on maintaining steam generator tube serviceability as described below:

a. Steam generator tubes, with degradation attributed to outside diameter stress corrosion cracking within the bounds of the cold-leg tube support plate with bobbin voltages less than or equal to the lower voltage repair limit [Note 1] will be allowed to remain in service. Steam generator tubes, with degradation attributed to outside diameter stress corrosion cracking within the bounds of the hot-leg tube support plate with bobbin voltages less than or equal to 3.0 volts will be allowed to remain in service.

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

SURVEILLANCE REQUIREMENTS (Continued)

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

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

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

f. If an unscheduled mid-cycle inspection is performed, the following mid-cycle repair limits apply instead of the limits identified in 4.4.5.4.a.11.a, 4.4.5.4.a.11.b and 4.4.5.4.a.11.d for outside diameter stress corrosion cracking indications occurring in the steam generator cold-legs. For outside diameter stress corrosion cracking indications occurring in the steam generator hot-legs, the limits in 4.4.5.4.a.11.a and 4.4.5.4.a.11.c apply. The mid-cycle repair limits are determined from the following equations:

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

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

SURVEILLANCE REQUIREMENTS (Continued)

Where:

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

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

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

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

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.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.5 Reports

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged or repaired in each steam generator shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2;

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following the completion of the inspection. This Special Report shall include:

1) Number and extent of tubes inspected,

2) Location and percent of wall-thickness penetration for each indication of an imperfection, and

3) Identification of tubes plugged or repaired.

c. Results of steam generator tube inspections which fall into Category C-3 shall be reported in a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days and prior to resumption of plant operation. This report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

d. For implementation of the voltage based repair criteria to tube support plate intersections for Unit 1 Cycle 6, notify the staff prior to returning the steam generators to service should any of the following conditions arise:

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

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

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

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

SURVEILLANCE REQUIREMENTS (Continued)

5. If the calculated conditional burst probability based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds 1 x 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.

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.

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.

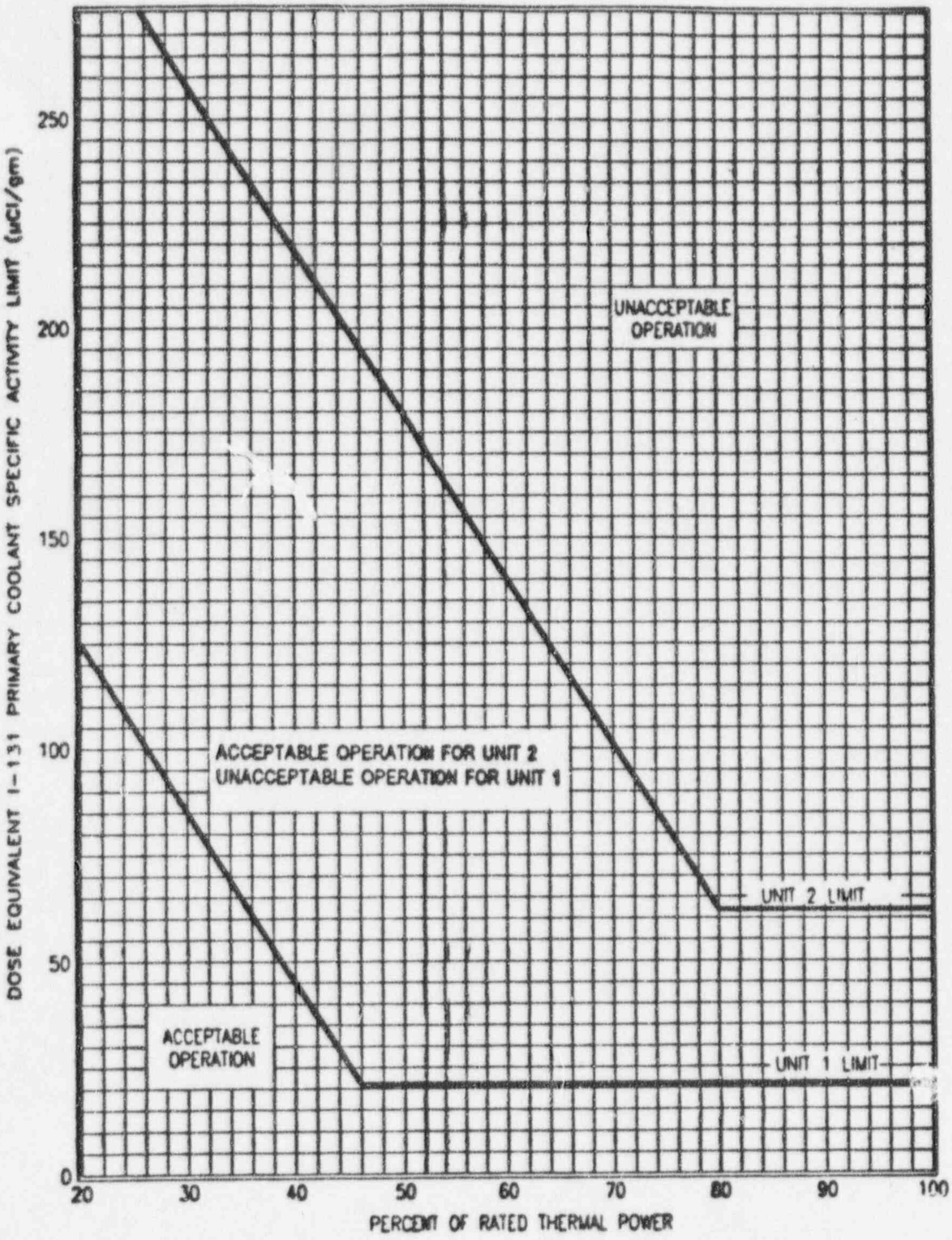

FIGURE 3.4-1
DOSE EQUIVALENT I-131 REACTOR COOLANT SPECIFIC ACTIVITY LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE REACTOR COOLANT SPECIFIC ACTIVITY >1µCI/GRAM DOSE EQUIVALENT I-131*

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

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/Ē μCi/gram of gross radioactivity, and	1#, 2#, 3#, 4#, 5#
	b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15% of the RATED THERMAL POWER within a 1-hour period.	1, 2, 3

TABLE NOTATIONS

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

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

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

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

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

3/4.4.5 STEAM GENERATORS (continued)

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

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

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

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

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

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

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

3/4.4.5 STEAM GENERATORS (continued)

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

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

The maximum site allowable primary-to-secondary leakage limit for end-of-cycle main steamline break conditions includes the accident leakage from IPC in addition to the accident leakage from F* on the faulted steam generator and the operational leakage limit of Specification 3.4.6.2.c. The operational leakage limit of Specification 3.4.6.2.c in each of the three remaining intact steam generators shall include the operational leakage from F*.

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.