50-454

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

February 3, 1998

Mr. Oliver D. Kingsley, President
Nuclear Generation Group
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 500
Downers Grove, IL 60515

SUBJECT: ISSUANCE OF AMENDMENTS (TAC NOS. M98070, M98071, M98072 AND M98073)

Dear Mr. Kingsley:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 101 to Facility Operating License No. NPF-37 and Amendment No. 101 to Facility Operating License No. NPF-66 for the Byron Station, Unit Nos. 1 and 2, respectively, and Amendment No. 92 to Facility Operating License No. NPF-72 and Amendment No. 92 to Facility Operating License No. NPF-77 for the Braidwood Station, Unit Nos. 1 and 2, respectively. The amendments are in response to your application dated February 28, 1997. Information related to the proposed restoration of the primary coolant dose equivalent iodine-131 (DEI) to their original licensing basis had been previously submitted in Commonwealth Edison Company's (ComEd) letter dated November 13, 1996, which was supplemented in subsequent letters dated March 20, June 24, August 19 and November 3, 1997.

The amendments revise the technical specifications (TS) to reflect the forthcoming replacement of the original steam generators (OSG) in Byron, Unit 1, and Braidwood, Unit 1, which are Westinghouse Model D4 steam generators (SG), with the replacement steam generators (RSG) which are Babcock and Wilcox, International (BWI) SG. The TS revisions were proposed because of design differences between the OSG and the RSG. The present TS for the OSG reflect the incorporation of certain repair criteria added by prior license amendments; these prior TS revisions were required because of certain forms of SG tube degradation in the OSG. The present revisions to the TS remove the interim plugging criteria (IPC) related to outer diameter stress corrosion cracking (ODSCC) in the OSG as well as the F* alternative repair criteria and two separate SG tube sleeving methodologies which are not needed for the RSG.

Finally, the RSG are not subject to the relatively large end-of-cycle (EOC) SG tube leakage rates predicted in accordance with either the methodology in Generic Letter (GL) 95-05 for estimating EOC leakage or the leakage methodology proposed in your letter dated October 15, 1997, for Braidwood, Unit 1, as modified in your presentation to the staff on December 11, 1997. Subsequently, your letter dated January 14, 1998, transmitted a technical report describing your latest approach to voltage-dependent growth rates for SG tubes subject to ODSCC. Accordingly, your proposal to restore the primary coolant DEI concentration from 0.35 to 1.00 microcuries per gram for both Byron, Unit 1, and Braidwood, Unit 1, is also approved, but is to be implemented only in the first and subsequent operating cycles following installation of the RSG. The DEI in Byron, Unit 1, had been administratively reduced from 0.35 to 0.20 microcuries per gram prior to EOC-8 after an operability determination. The present DEI in

DF01

Braidwood, Unit 1, has been administratively reduced in two steps from 0.35 to 0.10 and subsequently to 0.05 microcuries per gram after two separate operability assessments.

The RSG for Byron, Unit 1, are presently being installed during the refueling outage which began on November 7, 1997. The RSG will be installed in Braidwood, Unit 1, during the fall 1998 refueling outage.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Orig. signed by:

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

Docket Nos. STN 50-454, STN 50-455,
STN 50-456 and STN 50-457

Enclosures:
1. Amendment No. 101 to NPF-37
2. Amendment No. 101 to NPF-66
3. Amendment No. 92 to NPF-72
4. Amendment No. 92 to NPF-77
5. Safety Evaluation

cc w/encl: see next page

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G. Hill (8), T5C3	ACRS, T2E26	M. Jordan, RIII	C. Miller
E. Sullivan	J. Hayes	W. Beckner, O13H15	

*see previous page for concurrence

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M. David Lynch, Senior Project Manager
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Docket Nos. STN 50-454, STN 50-455,
STN 50-456 and STN 50-457

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cc w/encl: see next page

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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. 101
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 February 28, 1997, as supplemented on November 13, 1996, and March 20, June 24, August 19 and November 3, 1997, 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. 101 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 and shall be implemented in the first operating cycle after installation of the Babcock and Wilcox, International replacement steam generators.

FOR THE NUCLEAR REGULATORY COMMISSION

M. David 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: February 3, 1998

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. 101
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 February 28, 1997, as supplemented on November 13, 1996, and March 20, June 24, August 19 and November 3, 1997, 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. 101 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. David 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: February 3, 1998

ATTACHMENT TO LICENSE AMENDMENT NOS. 101 AND 101

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.

Remove Pages	Insert Pages
VIII	VIII
3/4 4-14	3/4 4-14
3/4 4-15	3/4 4-15
3/4 4-16	3/4 4-16
3/4 4-17	3/4 4-17
3/4 4-17b	3/4 4-17b
3/4 4-17d	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 4-29a
3/4 4-31	3/4 4-31
B 3/4 4-3	B 3/4 4-3
B 3/4 4-3b	B 3/4 4-3b

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

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

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

4) For Westinghouse Model D4 steam generators, indications left in service as a result of application of the tube support plate voltage-based repair criteria shall be inspected by bobbin coil probe during all future refueling outages, and

5) For Westinghouse Model D4 steam generators, 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.

e. For Westinghouse Model D4 and D5 steam generators, a random sample of at least 20% of the total number of laser welded sleeves and at least 20% of the total number of TIG welded sleeves installed shall be inspected for axial and circumferential indications at the end of each cycle. In the event that an imperfection exceeding the repair limit is detected, an additional 20% of the unsampled sleeves shall be inspected, and if an imperfection exceeding the repair limit is detected in the second sample, all remaining sleeves shall be inspected. These inservice inspections will include the entire sleeve, the tube at the heat treated area, and the tube to sleeve joints. The inservice inspection for the sleeves is required on all types of sleeves installed in the Byron and Braidwood Steam Generators to demonstrate acceptable structural integrity.

SURVEILLANCE REQUIREMENTS (Continued)

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

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

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

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

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality or initial operation following a steam generator replacement. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months;

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 at 40-month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.5.3a.; the interval may then be extended to a maximum of once per 40 months; and

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:

   1) Reactor-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2c., or

2) A seismic occurrence greater than the Operating Basis Earthquake, or

3) A Condition IV loss-of-coolant accident requiring actuation of the Engineered Safety Features, or

4) A Condition IV main steam line or feedwater line break.

4.4.5.4 Acceptance Criteria

a. As used in this specification:

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

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

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

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

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

6) Plugging or Repair Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging or repaired by sleeving in the affected area. The plugging or repair limit imperfection depth for tubing is equal to 40% of the nominal wall thickness. For Westinghouse Model D4 and D5 steam generators, the plugging or repair limit imperfection depth for laser welded sleeves is equal to 40% of the nominal sleeve wall thickness, and for TIG welded sleeves is equal to 32% of the nominal sleeve wall thickness. For Westinghouse Model D4 steam generators, this definition does not apply to defects in the tubesheet that meet the criteria for an F* tube;

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

7) Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 4.4.5.3c., above;

8) Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg. For a tube that has been repaired by sleeving, the tube inspection shall include the sleeved portion of the tube, and

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

10) Tube Repair refers to a process that reestablishes tube serviceability. Acceptable tube repairs for Westinghouse Model D4 or D5 steam generators will be performed by the following processes:

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

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

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

11) For Unit 1 through Cycle 8, the Tube Support Plate Plugging Limit is used for the disposition of an alloy 600 steam generator tube for continued service that is experiencing predominantly axially oriented outer diameter stress corrosion cracking confined within the thickness of the tube support plates. At tube support plate intersections, the plugging (repair) limit is based on maintaining steam generator tube serviceability as described below:

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

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

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

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 of a Westinghouse Model D4 steam generator 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 Westinghouse Model D4 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)

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 Westinghouse Model D4 steam generators' 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 through Cycle 8, reactor coolant DOSE EQUIVALENT I-131 will be limited to 0.35 microCuries per gram.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

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

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

SURVEILLANCE REQUIREMENTS

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

*For Unit 1 through Cycle 8, reactor coolant DOSE EQUIVALENT I-131 will be limited to 0.35 microCuries per gram.

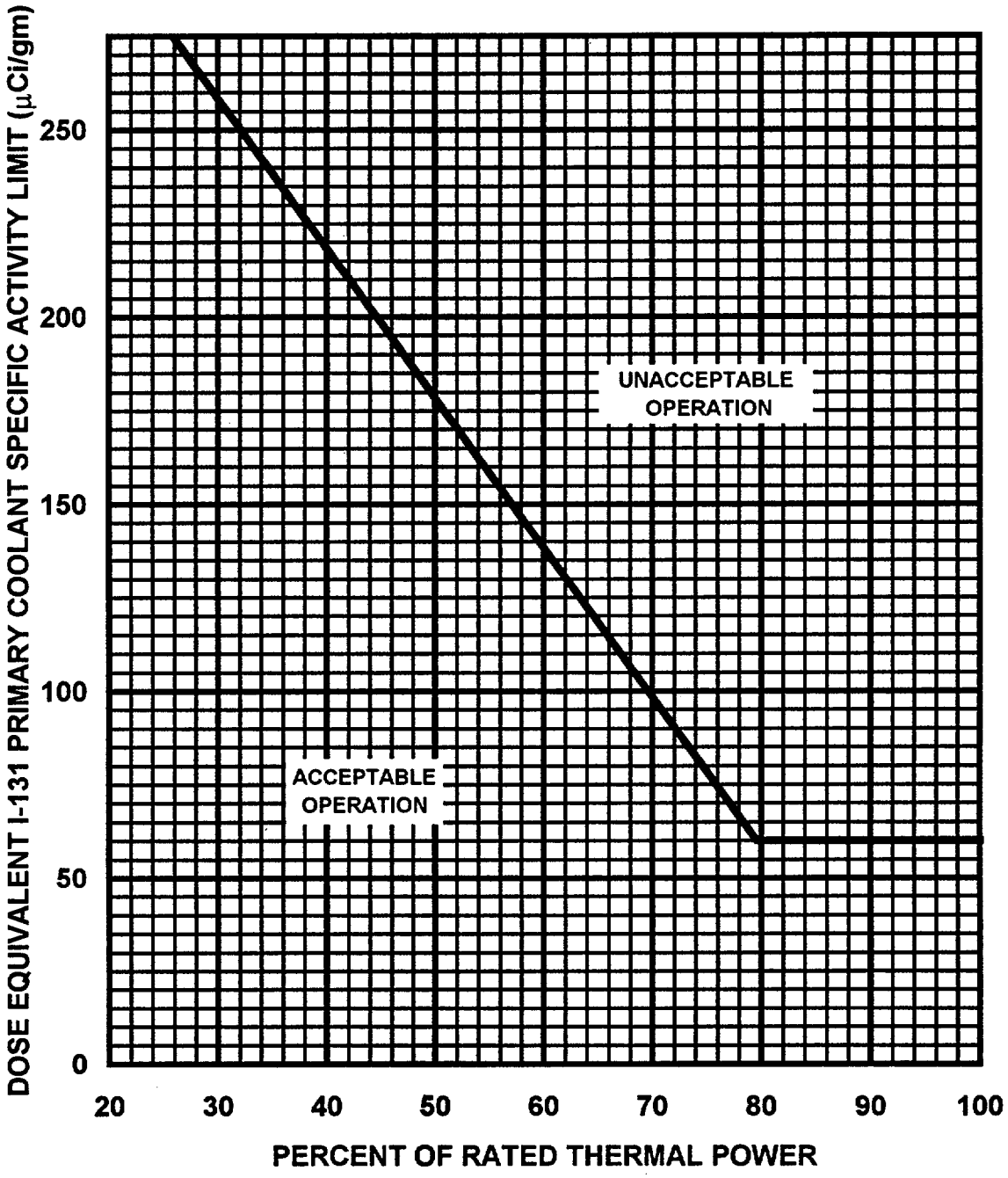

FIGURE 3.4-1

FOR UNIT 1 (AFTER CYCLE 8) AND UNIT 2
DOSE EQUIVALENT I-131 REACTOR COOLANT SPECIFIC ACTIVITY LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE REACTOR COOLANT SPECIFIC ACTIVITY >1 μCI/GRAM DOSE EQUIVALENT I-131

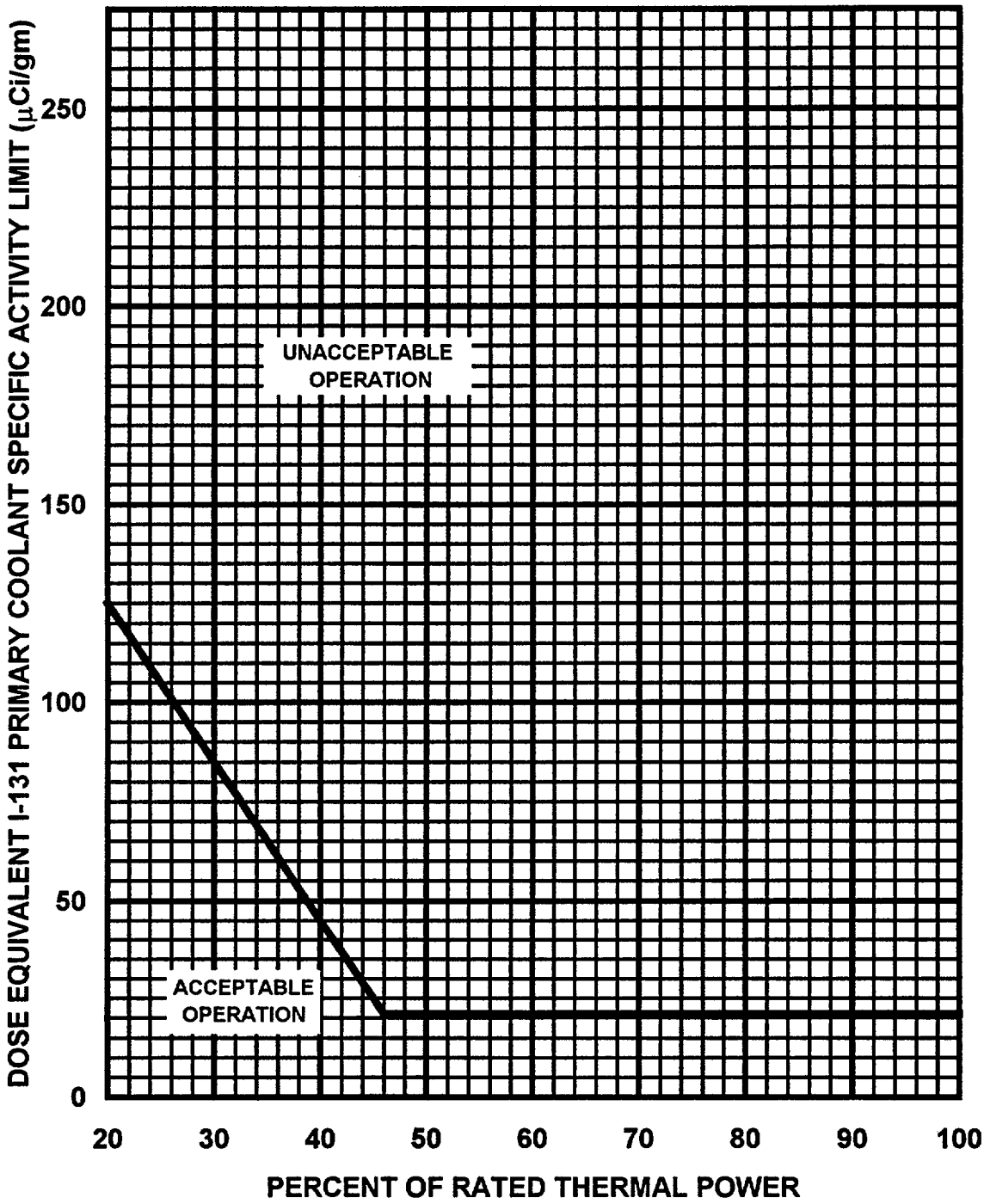

FIGURE 3.4-2

UNIT 1 THROUGH CYCLE 8
DOSE EQUIVALENT I-131 REACTOR COOLANT SPECIFIC ACTIVITY LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE REACTOR COOLANT SPECIFIC ACTIVITY >0.35 µCi/GRAM DOSE EQUIVALENT I-131

TABLE NOTATIONS

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

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

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

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

**** For Unit 1 through Cycle 8, reactor coolant DOSE EQUIVALENT I-131 will be limited to 0.35 microCuries per gram.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the Reactor Coolant System and the Secondary Coolant System (reactor-to-secondary leakage = 150 gallons per day per steam generator). Cracks having a reactor-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that reactor-to-secondary leakage of 150 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown, mainsteam lines, or the steam jet air ejectors. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged or repaired by sleeving. The technical bases for sleeving are described in the current Westinghouse or ABB Combustion Engineering, Inc. Technical Reports which are applicable for Westinghouse Model D4 and D5 steam generators only.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging or sleeving will be required for all tubes with imperfections exceeding the plugging or repair limit of 40% of the tube nominal wall thickness, excluding defects that meet the criteria for Westinghouse Model D4 F* tubes. Acceptable plugging criteria for Westinghouse Model D4 and D5 sleeved tubes are: 1) a laser welded sleeved tube must be plugged if a through wall penetration is detected in the sleeve that is equal to or greater than 40% of the nominal sleeve thickness, and 2) TIG welded sleeved tubes must be plugged if a through wall penetration is detected in the sleeve that is equal to or greater than 32% of the nominal sleeve thickness. The plugging limit for the sleeve is derived from Reg. Guide 1.121 analysis and utilizes a 20% allowance for eddy current uncertainty and additional degradation growth. Inservice inspection of sleeves is required to ensure RCS integrity. Sleeve inspection techniques are described in the current Westinghouse or ABB Combustion Engineering, Inc. Technical Reports. Steam Generator tube and sleeve inspections have demonstrated the capability to reliably detect degradation of the pressure retaining portions of the tube or sleeve wall thickness. Commonwealth Edison will validate the adequacy of any system that is used for periodic inservice inspection of the sleeves and, as deemed appropriate, will upgrade testing methods as better methods are developed and validated for commercial use.

3/4.4.5 STEAM GENERATORS (Continued)

The 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 Westinghouse Model D4 steam generators, 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 and Wilcox Nuclear Technologies (BWNT) Topical Report BAW-10196P."

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. 92
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 February 28, 1997, as supplemented on November 13, 1996, and March 20, June 24, August 19 and November 3, 1997, 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. 92 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 and shall be implemented in the first operating cycle after installation of the Babcock and Wilcox, International replacement steam generators.

FOR THE NUCLEAR REGULATORY COMMISSION

M. David 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: February 3, 1998

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. 92
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 February 28, 1997, as supplemented on November 13, 1996, and March 20, June 24, August 19 and November 3, 1997, 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-77 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 92 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. David 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: February 3, 1998

ATTACHMENT TO LICENSE AMENDMENT NOS. 92 AND 92

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
VIII	VIII
3/4 4-14	3/4 4-14
3/4 4-15	3/4 4-15
3/4 4-16	3/4 4-16
3/4 4-17	3/4 4-17
3/4 4-17c	3/4 4-17c
3/4 4-17d	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 4-29a
3/4 4-31	3/4 4-31
B 3/4 4-3	B 3/4 4-3
B 3/4 4-3b	B 3/4 4-3b
B 3/4 4-5	B 3/4 4-5

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) 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,

4) For Westinghouse Model D4 steam generators, indications left in service as a result of application of the tube support plate voltage-based repair criteria shall be inspected by bobbin coil probe during all future refueling outages, and

5) For Westinghouse Model D4 steam generators, 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 7, 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. For Westinghouse Model D4 and D5 steam generators, a random sample of at least 20% of the total number of laser welded sleeves and at least 20% of the total number of TIG welded sleeves installed shall be inspected for axial and circumferential indications at the end of each cycle. In the event that an imperfection exceeding the repair limit is detected, an additional 20% of the unsampled sleeves shall be inspected, and if an imperfection exceeding the repair limit is detected in the second sample, all remaining sleeves shall be inspected. These inservice inspections will include the entire sleeve, the tube at the heat treated area, and the tube to sleeve joints. The inservice inspection for the sleeves is required on all types of sleeves installed in the Byron and Braidwood Steam Generators to demonstrate acceptable structural integrity.

SURVEILLANCE REQUIREMENTS (Continued)

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

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality or initial operation following a steam generator replacement. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months;

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 at 40-month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.5.3a.; the interval may then be extended to a maximum of once per 40 months; and

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:

   1) Reactor-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2c., or

   2) A seismic occurrence greater than the Operating Basis Earthquake, or

   3) A Condition IV loss-of-coolant accident requiring actuation of the Engineered Safety Features, or

   4) A Condition IV main steam line or feedwater line break.

4.4.5.4 Acceptance Criteria

a. As used in this specification:

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

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

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

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

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

6) Plugging or Repair Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging or repaired by sleeving in the affected area. The plugging or repair limit imperfection depth for tubing is equal to 40% of the nominal wall thickness. For Westinghouse Model D4 and D5 steam generators, the plugging or repair limit imperfection depth for laser welded sleeves is equal to 40% of the nominal sleeve wall thickness, and for TIG welded sleeves is equal to 32% of the nominal sleeve wall thickness. For Westinghouse Model D4 steam generators, this definition does not apply to defects in the tubesheet that meet the criteria for an F* tube. For Unit 1 Cycle 7, this definition does not apply to the tube support plate intersections for which the voltage-based repair criteria are being applied. Refer to 4.4.5.4.a.13 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 for Westinghouse Model D4 or D5 steam generators will be performed by the following processes:

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

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

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

11) Locked-Tube Model Intersection means all steam generator hot-leg tube-to-tube support plate intersections which have been analyzed to experience a tube support plate displacement less than 0.1 inches during accident conditions, excluding the following:

    a) All tube-to-tube support plate intersections where IPC can not be applied per Generic Letter 95-05;

    b) All Flow Distribution Baffle intersections;

    c) All steam generator tube intersections adjacent to an intersection that contains a corrosion induced dent greater than 0.065 inches; and

    d) All tube-to-tube support plate intersections that will be displaced more than 0.1 inches during accident conditions due to failure of the steam generator internal structures.

Note 2: The upper voltage repair limit for indications of outside diameter stress corrosion cracking occurring in the Free-Span Model intersections is calculated according to the methodology in Generic Letter 95-05 as supplemented.

14) F* Distance is the distance into the tubesheet of a Westinghouse Model D4 steam generator 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.

15) F* Tube is a Westinghouse Model D4 steam generator tube with degradation below the F* distance and has no indications of degradation (i.e., no indication of cracking) within the F* distance. Defects contained in an F* tube are not dependant on flaw geometry.

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

4.4.5.5 Reports

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

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

1) Number and extent of tubes inspected,

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

3) Identification of tubes plugged or repaired.

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 7, notify the staff prior to returning the steam generators to service should any of the following conditions arise:

SURVEILLANCE REQUIREMENTS (Continued)

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.

5) If cracking is observed in the tube support plates.

6) If any tube which previously passed a 0.610 inch diameter bobbin coil eddy current probe currently fails to pass a 0.610 inch diameter bobbin coil eddy current probe.

7) 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.

8) 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 Westinghouse Model D4 steam generators 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 through Cycle 7, reactor coolant DOSE EQUIVALENT I-131 will be limited to 0.35 microCuries per gram.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

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

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

SURVEILLANCE REQUIREMENTS

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

*For Unit 1 through Cycle 7, reactor coolant DOSE EQUIVALENT I-131 will be limited to 0.35 microCuries per gram.

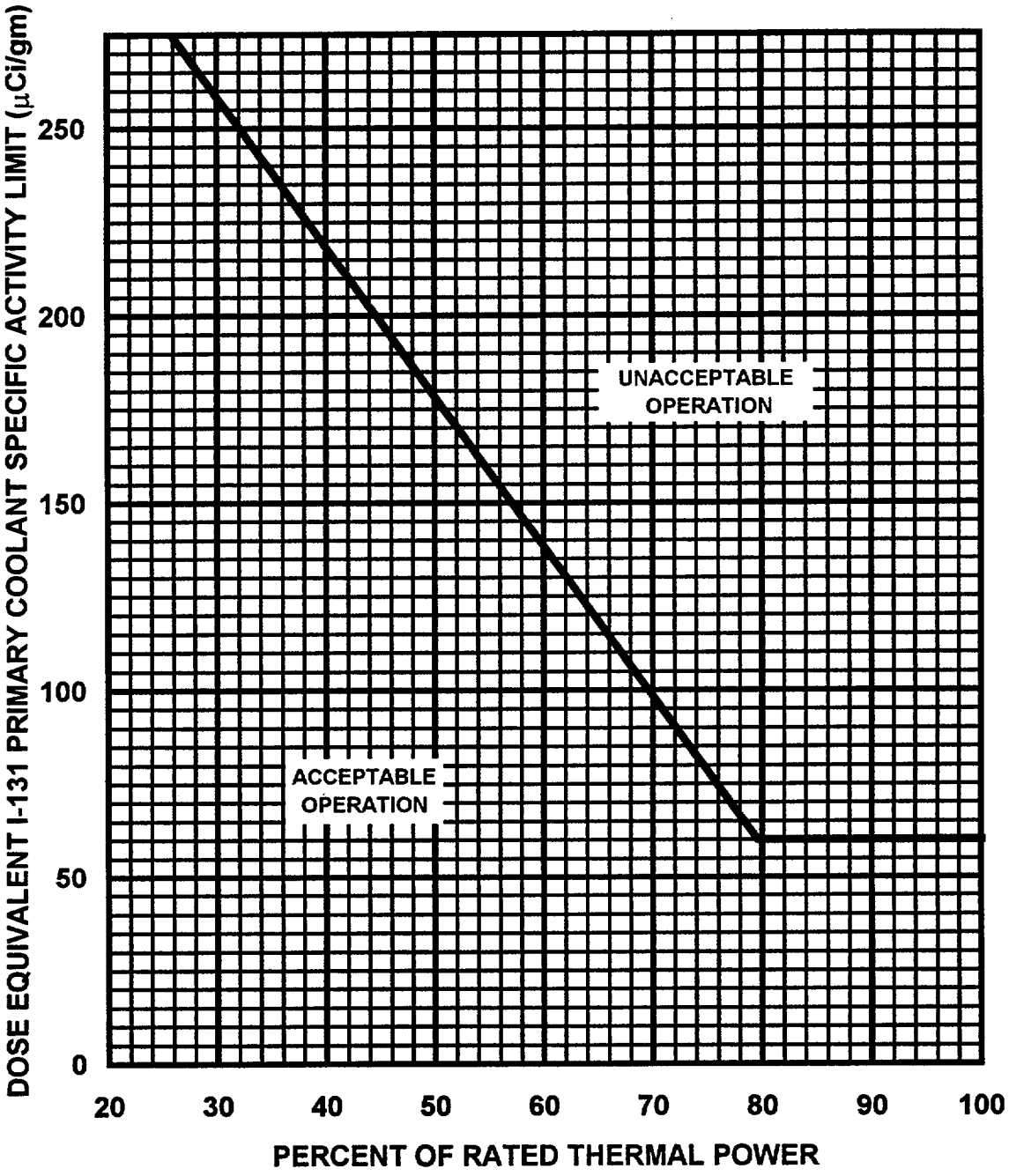

FIGURE 3.4-1

FOR UNIT 1 (AFTER CYCLE 7) AND UNIT 2
DOSE EQUIVALENT I-131 REACTOR COOLANT SPECIFIC ACTIVITY LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE REACTOR COOLANT SPECIFIC ACTIVITY >1 μCI/GRAM DOSE EQUIVALENT I-131

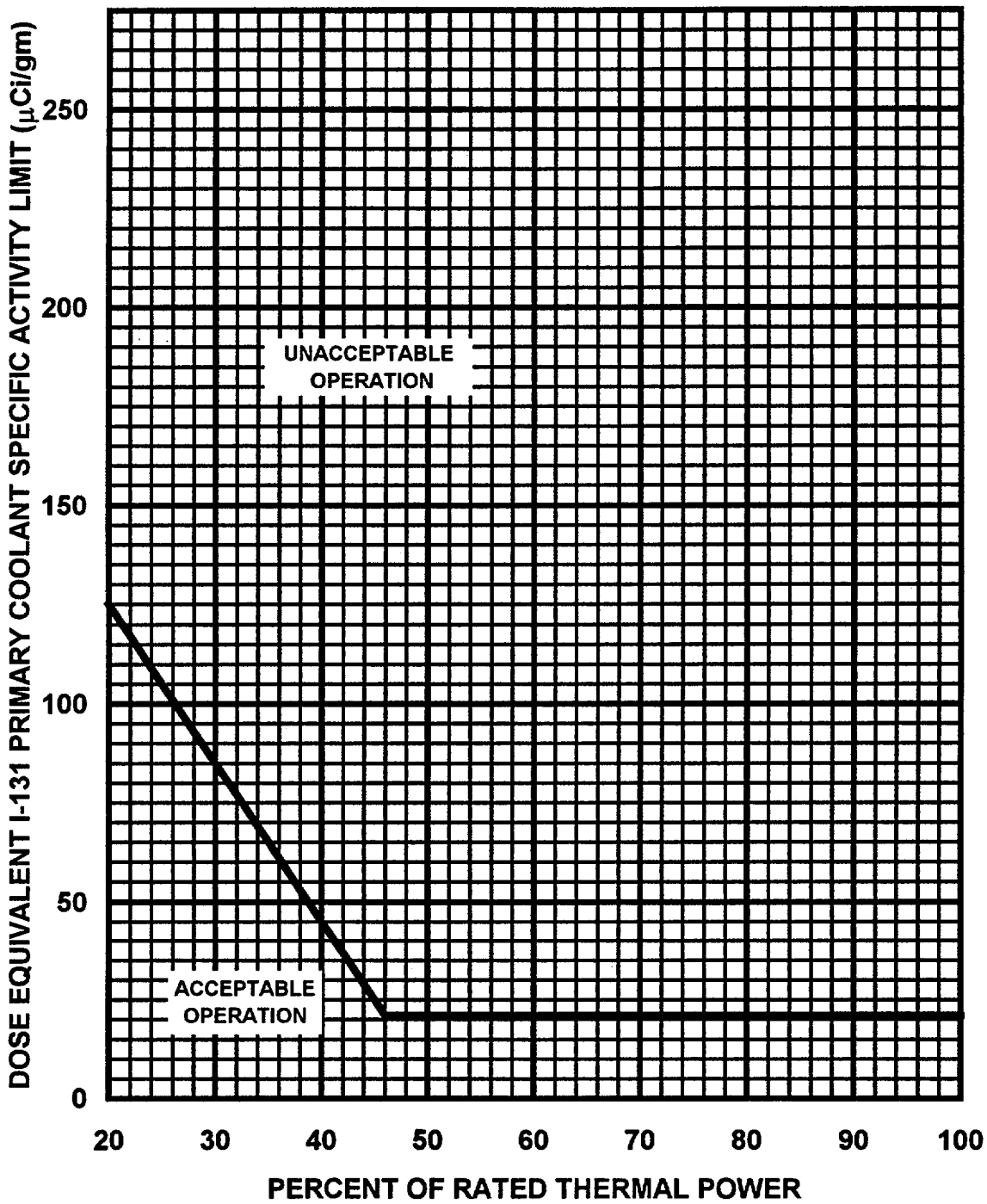

FIGURE 3.4-2

UNIT 1 THROUGH CYCLE 7
DOSE EQUIVALENT I-131 REACTOR COOLANT SPECIFIC ACTIVITY LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE REACTOR COOLANT SPECIFIC ACTIVITY >0.35 µCi/GRAM DOSE EQUIVALENT I-131

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 E for the reactor coolant sample. Determination of the contributors to E shall be based upon these energy peaks identifiable with a 95% confidence level.

**** For Unit 1 through Cycle 7, reactor coolant DOSE EQUIVALENT I-131 will be limited to 0.35 microCuries per gram.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the Reactor Coolant System and the Secondary Coolant System (reactor-to-secondary leakage = 150 gallons per day per steam generator). Cracks having a reactor-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that reactor-to-secondary leakage of 150 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown, mainsteam lines, or the steam jet air ejectors. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged or repaired by sleeving. The technical bases for sleeving are described in the current Westinghouse or ABB Combustion Engineering, Inc. Technical Reports which are applicable to Westinghouse Model D4 and D5 steam generators only.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging or sleeving will be required for all tubes with imperfections exceeding the plugging or repair limit of 40% of the tube nominal wall thickness, excluding defects that meet the criteria for Westinghouse Model D4 F* tubes. Acceptable plugging criteria for Westinghouse Model D4 and D5 sleeved tubes are: 1) a laser welded sleeved tube must be plugged if a through wall penetration is detected in the sleeve that is equal to or greater than 40% of the nominal sleeve thickness, and 2) TIG welded sleeved tubes must be plugged if a through wall penetration is detected in the sleeve that is equal to or greater than 32% of the nominal sleeve thickness. The plugging limit for the sleeve is derived from Reg. Guide 1.121 analysis and utilizes a 20% allowance for eddy current uncertainty and additional degradation growth. Inservice inspection of sleeves is required to ensure RCS integrity. Sleeve inspection techniques are described in the current Westinghouse or ABB Combustion Engineering, Inc. Technical Reports. Steam Generator tube and sleeve inspections have demonstrated the capability to reliably detect degradation of the pressure retaining portions of the tube or sleeve wall thickness. Commonwealth Edison will validate the adequacy of any system that is used for periodic inservice inspection of the sleeves and, as deemed appropriate, will upgrade testing methods as better methods are developed and validated for commercial use.

3/4.4.5 STEAM GENERATORS (Continued)

The mid-cycle equation in SR 4.4.5.4.a.13.f should only be used during unplanned inspections in which eddy current data is acquired for indications at the Free-Span Model Intersections. The voltage repair limit for indications at the Locked-Tube Model Intersections 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 Westinghouse Model D4 steam generators, plugging or repair is not required for tubes with degradation within the tubesheet area which fall under the alternate tube plugging criteria defined as F^*. The F^* Criteria is based on "Babcock & Wilcox Nuclear Technologies (BWNT) Topical Report BAW-10196 P."

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

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

OPERATIONAL LEAKAGE (Continued)

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

3/4.4.7 CHEMISTRY

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

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

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the reactor coolant ensure that the resulting 2-hour doses at the SITE BOUNDARY will not exceed an appropriately small fraction of 10 CFR Part 100 dose guideline values following a steam generator tube rupture accident in conjunction with an assumed steady-state reactor-to-secondary steam generator leakage rate of 1 gpm. For Unit 1 through Cycle 7, the limitations on the specific activity of the reactor coolant ensure that the resulting 2-hour off-site doses will not exceed an appropriately small fraction of the 10 CFR Part 100 dose guideline values following a Main Steam Line Break accident in conjunction with an assumed steady-state primary-to-secondary steam generator leakage rate of 150 gpd from each unfaulted steam generator and maximum site allowable primary-to-secondary leakage from the faulted steam generator. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Braidwood Station, such as SITE BOUNDARY location and meteorological conditions, were not considered in this evaluation.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 101 TO FACILITY OPERATING LICENSE NO. NPF-37,

AMENDMENT NO. 101 TO FACILITY OPERATING LICENSE NO. NPF-66,

AMENDMENT NO. 92 TO FACILITY OPERATING LICENSE NO. NPF-72,

AND AMENDMENT NO. 92 TO FACILITY OPERATING LICENSE NO. NPF-77

COMMONWEALTH EDISON COMPANY

BYRON STATION, UNIT NOS. 1 AND 2

BRAIDWOOD STATION, UNIT NOS. 1 AND 2

DOCKET NOS. STN 50-454, STN 50-455, STN 50-456 AND STN 50-457

1.0 INTRODUCTION

By letter dated February 28, 1997, as supplemented by letters dated November 13, 1996, and March 20, June 24, August 19 and November 3, 1997, the Commonwealth Edison Company (ComEd, the licensee) requested license amendments to revise the technical specifications (TS) for the Byron Station, Units 1 and 2, and for the Braidwood Station, Units 1 and 2. The intent of these requests was to eliminate certain steam generator (SG) tube repair criteria as well as certain inspection and reporting requirements for the SG tubes. These license amendment requests were necessitated by the planned replacement of the original steam generators (OSG) in Byron, Unit 1, and Braidwood, Unit 1, which are Westinghouse Model D4 SG, with the replacement steam generators (RSG) which are Babcock & Wilcox, International (BWI) SG. The proposed TS revisions reflect the significant design differences between the OSG and the RSG. The SG tube repair criteria and the associated surveillance and reporting requirements are identified as the interim plugging criteria (IPC). These repair criteria were required to address a form of SG tube degradation in the OSG known as outer diameter stress corrosion cracking (ODSCC). The November 13, 1996, and March 20, June 24, August 19 and November 3, 1997, submittal provided additional information that did not change the initial proposed no significant hazards consideration determination.

Additionally, the F* alternative repair criteria (ARC) for SG tube flaws within the OSG tubesheet as well as two separate OSG tube repair sleeving methodologies are no longer needed in the Byron, Unit 1, and Braidwood, Unit 1, TS due to the design differences between the OSG and the RSG.

The IPC were added to the Byron, Unit 1, and Braidwood, Unit 1, TS in license amendments issued on November 9, 1995. These criteria and associated surveillance and reporting requirements were not added to the Byron, Unit 2, and Braidwood, Unit 2, TS since both these units have Westinghouse Model D5 SG which have not experienced any significant ODSCC SG tube degradation. While the pending license amendment requests only affect the Byron, Unit 1,

and Braidwood, Unit 1, TS, the requested license amendments apply to Unit 2 of both stations so as to maintain the continuity of the license amendment numbering system for both the Byron and Braidwood Stations.

Finally, the licensee, in the subject license amendment requests, proposed to restore the limiting TS value for the long-term dose equivalent iodine-131 (DEI) concentration in the primary coolant to its original licensing basis of 1.0 microcurie per gram (μCi/gm) from the present TS limiting values of 0.35 μCi/gm for both Byron, Unit 1, and Braidwood, Unit 1. Similarly, the licensee has proposed to restore the short-term (i.e., less than 48 hours) TS limit to its original licensing basis of 60 μCi/gm from the present TS value of 20 μCi/gm. Both the forthcoming installation of the RSG and the proposed restoration of the DEI to their original licensing bases required the staff to reevaluate the radiation dose exposures which could result from postulated design basis accidents (DBA). For example, the RSG have a larger volume of water and metal than the OSG which affect the evaluation of radiation doses attributable to DBAs due to the difference in the release of steam during transient or accident conditions.

Both the Byron, Unit 1, and the Braidwood, Unit 1, TS revisions are to be implemented in the first and subsequent operating cycles following installation of the RSG. For Byron, Unit 1, the RSG installation started in the refueling outage initiated in November 1997. Accordingly, the revisions to the Byron, Unit 1, TS are now effective and will be implemented in the forthcoming fuel cycle (Cycle 9) starting in early 1998. In a similar fashion, the revisions to the Braidwood, Unit 1, TS are effective now and will be implemented after the RSG installation in Braidwood, Unit 1, which is scheduled for the fall 1998 refueling outage (Cycle 8).

2.0 EVALUATION

2.1 Discussion

The NRC staff reviewed four separate DBA whose onsite and offsite radiological consequences would be altered as a result of the installation of the RSG in both Byron, Unit 1, and Braidwood, Unit 1. Though both stations are nearly identical, there are several differences between the two sites such as the local meteorology which affects the values of the atmospheric dispersion factors. Further, the unfiltered infiltration rates for the control rooms are also different between each station. Accordingly, though such features as the RSG, the reactor core design and containment are identical for each station, the staff analyzed each site separately in recognition of these site specific differences.

The results of these station specific analyses are presented in separate tables in this safety evaluation. Most of the attached tables, however, are applicable to both stations due to the nearly identical physical plant characteristics. The staff's conclusions in this Safety Evaluation (SE) are also presented separately for each station since the conclusions are slightly different for each station.

The removal of the IPC for ODSCC flaws from the Byron, Unit 1, and the Braidwood, Unit 1, TS required no independent staff analysis since the RSG do not have the same type of SG tube support structures as the OSG and, therefore, the SG tube repair criteria for the ODSCC flaws which occurred in the OSG, are not applicable to the RSG. Accordingly, the RSG will not be subject to the relatively large end of cycle SG tube leakage which could occur under postulated

accident conditions. Similarly, the removal of the SG tube ARC for flaws occurring within the OSG tubesheets (i.e., the F* repair criteria) also required no independent staff analysis for the same reason as cited above. This is also true for the proposed removal of the two separate sleeving methodologies presently in the Byron, Unit 1, and the Braidwood, Unit 1, TS. The removal of these various SG tube repair criteria and their associated surveillance and reporting requirements, are considered to be administrative in nature since they are not required to ensure the safe operation of the RSG due to the significant design differences between the OSG and the RSG.

2.2 Evaluation of the Radiological Consequences

2.2.1 Byron, Unit 1, and Braidwood, Unit 1, Radiological Analysis of Postulated DBAs

The licensee performed various reanalyses of the Byron, Unit 1, and Braidwood, Unit 1, for the postulated accident conditions analyzed in Chapter 15 of the Final Safety Analysis Report (FSAR). These reanalyses were required due to the change in the design of the Unit 1 SGs of both stations from the Westinghouse Model D4 steam generators to the Babcock & Wilcox steam generators and because the licensee has proposed an amendment to restore both the maximum instantaneous and the 48 hour TS values for DEI to the original licensing basis levels.

The change in SG design also affects the manner in which the reactor responds in the event of certain accidents. This change in response can impact the radiological releases, thereby affecting offsite and onsite radiological doses.

On November 13, 1996, the licensee submitted its revised steam generator tube rupture (SGTR) analysis. In letters dated February 11, May 20, July 18, and October 3, 1997, the staff transmitted to the licensee requests for additional information (RAI). In letters dated March 20, June 24, August 19, and November 3, 1997, the licensee provided responses to these RAIs. In these responses, the licensee identified four postulated accidents whose consequences would be altered as a result of the change in SG design from the OSG to the RSG. These are:

1. Main Steamline Break
2. Steam Generator Tube Rupture
3. Locked Rotor
4. Rod Ejection

The licensee indicated that all other postulated accidents would be unaffected by the change in the SG design.

The licensee calculated the offsite doses for the postulated main steamline break (MSLB) and the SGTR accidents. The licensee calculated doses which met the acceptance criteria of Standard Review Plan (SRP) Sections 15.1.5 and 15.6.3. The licensee indicated that the control room operator radiation exposure doses from a postulated loss-of-coolant accident (LOCA) would bound the operator's doses from the MSLB and SGTR accidents. The licensee did not calculate any doses for the postulated locked rotor or rod ejection accidents even though the quantity of steam released during the course of these accidents increased due to the change in SG design and the quantity of melted fuel increased for the rod ejection accident. The licensee also provided the change in steaming rates as a result of the change in steam generator design.

The following sections provide the results of the staff's independent assessment of the licensee's reanalysis of the FSAR Chapter 15 accidents affected by both the SG design change and the increase in the maximum permissible TS values of the primary coolant DEI.

2.2.2 Postulated Main Steamline Break Accident for Byron, Unit 1, and Braidwood, Unit 1

The staff performed a confirmatory evaluation of the consequences of an MSLB accident outside containment. Two cases were analyzed. In the first case, a pre-existing spike was assumed to have occurred prior to the event. During the event, primary-to-secondary leakage was assumed to occur at the TS maximum allowable rate of 150 gallons per day (gpd) per SG. For this case, the reactor coolant iodine specific activities were assumed to be at the TS Figure 3.4-1 full power DEI limit of 60 μCi/gm. The secondary coolant iodine specific activity was assumed to be at the secondary coolant specific activity equilibrium value of 0.1 μCi/gm.

The second case assumed that the postulated accident itself initiates a concurrent iodine spike. The reactor coolant activity level was assumed to be at the 48 hour TS value of 1.0 μCi/gm DEI. Secondary coolant activity was assumed to be at the TS limit of 0.1 μCi/gm DEI. It is assumed that when the iodine spike occurs, it results in a release of iodine from the fuel gap to the reactor coolant at a rate in Ci/unit time which is 500 times the normal iodine release rate necessary to maintain the reactor coolant at 1.0 μCi/gm (i.e., a spiking factor of 500). The licensee indicated that the MSLB event did not result in failed fuel.

For both analyses, it was assumed that a 150 gpd primary-to-secondary tube leak occurred in the faulted SG until it was isolated. For the intact SG, it was assumed that primary-to-secondary leakage occurred at a rate of 150 gpd/SG for the duration of the accident. Concurrent with the MSLB, it was also assumed that there would be a loss of offsite power. Consequently, the main condenser was unavailable for steam dump. Heat removal from the reactor core would have to occur, therefore, by discharge through the code safety valves. This was anticipated to occur for 40 hours after which no further steam or radioactivity release would occur because the reactor heat removal (RHR) system would be sufficient to maintain core cooling.

Table 1 presents the details of the staff's assumptions for both the Byron and Braidwood Stations. The results of the staff's assessment are presented in Tables 7, 9 and 11 for Byron, Unit 1, and Tables 8, 10 and 12 for Braidwood, Unit 1. Tables 7 and 9 present the offsite doses and Table 11 presents the control room doses for Byron, Unit 1. Similarly, the offsite doses for Braidwood, Unit 1, are in Tables 8 and 10 while the control room doses are in Table 12. For this postulated accident, the staff did not perform an assessment of the whole body dose associated with the release of noble gases because the thyroid dose is so limiting with respect to compliance with General Design Criterion (GDC) 19 of Appendix A to 10 CFR Part 50 and the guideline values in 10 CFR Part 100. The doses were found to be within a small fraction of Part 100 guidelines for the accident-initiated spike case, within Part 100 for the pre-existing spike case, and less than GDC 19 requirements for the control room in both cases for both Byron, Unit 1, and Braidwood, Unit 1.

2.2.3 Postulated Steam Generator Tube Rupture Accident for Byron, Unit 1, and Braidwood, Unit 1

The staff performed a confirmatory evaluation of the consequences of a postulated SGTR accident. As with the MSLB accident, two cases were analyzed. One case involved a pre-existing iodine spike and the other case, an accident-initiated spike. For the pre-existing spike case, the reactor coolant iodine activity level of the DEI was assumed to be at the full power level of 60 μCi/gm of the DEI in TS Figure 3.4-1. The secondary coolant iodine specific activity was based the TS normal operation limit of 0.1 μCi/gm DEI.

The accident-initiated spike case assumed the SGTR event itself initiated a concurrent iodine spike. The reactor coolant was assumed to be at the TS 48 hour reactor coolant activity level of the DEI of 1.0 μCi/gm. The secondary system activity was assumed to be at the TS limit of 0.1 μCi/gm DEI. The SGTR is assumed to initiate an iodine spike which results in a release of iodine from the fuel gap to the reactor coolant at a rate in Ci/unit time which is 500 times the normal iodine release rate necessary to maintain primary coolant at 1.0 μCi/gm. The licensee indicated that the postulated SGTR accident did not result in any failed fuel.

For both analyses, it was assumed that a primary-to-secondary leak occurred in the intact SGs at a rate of 150 gpd for the duration of the accident. It was also assumed for these two analyses that offsite power was lost and the main condenser was unavailable as a heat sink to remove decay heat from the reactor. After 40 hours following this event, the analysis indicated that reactor core heat had been sufficiently removed so that the unit could be placed on the RHR, and no further steam release or activity release was assumed to occur. For completeness, the staff incorporated the contribution from the intact SG for this period even though the dose contribution would be small from this source. The licensee had excluded this contribution from its assessment.

Table 4 presents the assumptions used by the staff in its assessment of the Unit 1 RSG for both the Byron and Braidwood Stations. Some of the assumptions presented for the MSLB are also relevant to the SGTR. Therefore, Table 1 should be referred to as a source of information for the postulated SGTR accident also.

The potential consequences of a postulated SGTR accident are presented in Tables 7, 9 and 11 for Byron, Unit 1, and in Tables 8, 10 and 12 for Braidwood, Unit 1. For this accident, the staff did not perform an assessment of the whole body dose associated with the release of noble gases because the thyroid dose is so limiting with respect to compliance with GDC 19 and 10 CFR Part 100. The doses were found to be within a small fraction of Part 100 guidelines for the accident-initiated spike case, within Part 100 for the pre-existing spike case for offsite exposures, and less than the requirements in GDC 19 for the control room in both cases for both Byron, Unit 1, and Braidwood, Unit 1.

2.2.4 Postulated Locked Rotor Accident for Byron, Unit 1, and Braidwood, Unit 1

The staff performed a confirmatory assessment of the consequences of a postulated reactor coolant pump locked rotor event and subsequent leakage of steam from the secondary system due to the leakage of primary coolant to the secondary system. Leakage from the primary side to the secondary side was assumed to exist prior to the accident. Initial conditions assumed a

contribution to releases from the initial reactor coolant activity levels at the proposed TS values and contribution of activity from five percent of the gap inventory being released to the primary coolant.

Table 5 presents the assumptions used by the staff in their assessment of the consequences of a locked rotor accident. The staff's assessment of the potential consequences of a postulated locked rotor accident are presented in Tables 7, 9 and 11 for Byron, Unit 1, and in Tables 8, 10 and 12 for Braidwood, Unit 1. The doses were found to be less than the requirements in GDC 19 and within a small fraction of the guideline values in 10 CFR Part 100 for offsite exposures for both Byron, Unit 1, and Braidwood, Unit 1.

2.2.5 Postulated Rod Ejection Accident for Byron, Unit 1, and Braidwood, Unit 1

The staff performed a confirmatory assessment of the consequences of a postulated rod ejection accident. It was assumed that the reactor was operating with equilibrium activity levels in the primary and secondary systems based upon a failed fuel rate of one percent and a primary-to-secondary leak rate of 150 gpd/SG. The staff's analyses assumed that the release pathway to the environment would be either via containment leakage or via primary-to-secondary leakage. The first pathway assumes that the postulated rod ejection results in the release of activity from the fuel gap and that a certain amount of fuel melting occurs. This activity is released from the fuel to the primary coolant. The primary coolant is then released to containment where the activity leaks from the containment at a rate consistent with the TS maximum allowable containment leak rate for the first 24 hours and at half that leak rate after 24 hours. The second pathway assumes that the activity from the fuel gap and the melted fuel is transported to the primary coolant. The primary coolant is then assumed to leak to the secondary side at the TS limit for the SG (i.e., a leak rate of 150 gpd per SG). Activity is released from the secondary side via the relief valves since it is assumed that there is a loss of offsite power concurrent with the rod ejection. The licensee indicated that in the event of a rod ejection, gap activity would be released from ten percent of the fuel and that five percent of the fuel would undergo melting.

Table 6 presents the assumptions used by the staff in its assessment. The results of the staff's assessment are presented in Tables 7, 9 and 11 for Byron, Unit 1, and in Tables 8, 10 and 12 for Braidwood, Unit 1 . The doses were found to be less than the requirements of GDC 19 and well within Part 100 guidelines for offsite exposures for Byron, Unit 1.

The radiation exposure doses for Braidwood, Unit 1, were found to be less than the requirements of GDC 19 and slightly greater than "well within the guideline values in 10 CFR Part 100" as stated in Appendix A of SRP Section 15.4.8 for offsite exposures resulting from the containment leakage pathway. Although the doses exceeded the guidelines of Appendix A, the staff concluded that this departure from the criteria in SRP Section 15.4.8 is acceptable because:

1. The doses are below the guideline values of 10 CFR Part 100;

2. The consequences were calculated on the basis of fuel cladding failures and fuel melting that are predicted by conservative models; and

3. The SRP acceptance criterion itself is conservative.

2.2.6 Conclusions for Byron, Unit 1

The staff has assessed those accidents for which the change from the Westinghouse Model D4 (i.e., the OSG) to the BWI SG (i.e., the RSG) would impact the offsite and control room operator doses. As a result of this assessment, the staff has concluded that, for those accidents which are affected by the installation of the RSG and the restoration of the TS values of the short-term and steady state DEI to their original licensing bases, the doses would not exceed the dose guidelines presently contained in the Standard Review Plans for the postulated rod ejection, MSLB, SGTR and locked rotor accidents and that in no case would the appropriate fraction of 10 CFR Part 100 be exceeded offsite nor would the GDC 19 of 10 CFR Part 50, Appendix A, requirement limiting operator dose exposures, be exceeded. Therefore, the staff finds that the proposed replacement of the Westinghouse Model D4 SG with the B&W SG as well as the change to the maximum permissible reactor coolant activity levels of the DEI in TS Section 3.4.8.a and TS Figure 3.4-1, are acceptable from an evaluation of the radiological consequences. Specifically, these changes restore the long-term (i.e., greater than 48 hours) DEI to 1.0 μCi/gm and the short-term DEI to 60 μCi/gm.

2.2.7 Conclusions for Braidwood, Unit 1

The staff has assessed those accidents for which the change from the Westinghouse Model D4 SG to the BWI SG would affect the offsite and control room operator doses. As a result of this assessment, the staff has concluded that, for those accidents which are affected by the change to the BWI SG, the doses would not exceed the dose guidelines presently contained in the Standard Review Plans for the MSLB, SGTR and locked rotor accidents and that in no case would the appropriate fraction of the radiation exposure guidelines in 10 CFR Part 100 be exceeded nor would the requirements for limiting radiation exposure to control room operators in GDC 19 of 10 CFR Part 50, Appendix A, be exceeded. For the rod ejection accident, it was determined that the offsite doses exceeded the guidelines of SRP Section 15.4.8. However, this was considered acceptable because the assumptions with respect to fuel melting and fuel cladding failure were considered conservative and the offsite consequences were below the guideline values of 10 CFR Part 100, The control room operator doses from a rod ejection are within the requirements of GDC 19. Therefore, the staff finds that the proposed replacement of the Westinghouse Model D4 SG with the BWI SG as well as the change to the maximum permissible reactor coolant activity levels of the DEI in TS Section 3.4.8.a and TS Figure 3.4-1, are acceptable from a radiological standpoint.

2.2.8 Changes to the Technical Specifications

The changes to the TS of the Byron and Braidwood Stations are identical because the SG tube repair criteria being removed from the TS were identical in Unit 1, in both stations. Some of these changes are primarily administrative in nature while others were reviewed for their effect on public health and safety (such as onsite and offsite radiation exposure doses which could occur under postulated accident conditions).

The Table of Contents for both stations are revised to reflect the restoration of the primary coolant's maximum permissible short-term DEI to its original licensing basis of 60 μCi/gm. This is reflected in the addition of a revised TS Figure 3.4-1 which increases the short-term DEI to

60 μCi/gm for Unit 1 of both stations which are applicable in the operating cycles after installation of the BWI RSG. This change is primarily administrative.

Sections 4.4.5.2 in the TS of both stations are being revised to indicate that the voltage-based SG tube repair criteria added in License Amendment Nos. 77 for the Byron Station and License Amendment Nos. 69 for the Braidwood Station, are only applicable for the OSG which are being replaced by the BWI RSG. The voltage-based repair criteria are not applicable to the RSG since the RSG do not have the same type of SG tube support structures as the OSG and, therefore, the SG tube repair criteria for the ODSCC flaws which occurred in the OSG, are not applicable to the RSG.

Sections 4.4.5.3 "Inspection Frequencies," in the TS of both stations are being revised to reflect the installation of the RSG. Specifically, the proposed revision requires as an added feature, an inspection no later than 24 calendar months after initial operation following a SG replacement. This revision is acceptable in that it applies a requirement on the inspection frequency of the first inservice inspection of the RSG which is identical to that for the OSG.

Section 4.4.5.4, "Acceptance Criteria," in the TS of both stations is being revised to redefine the plugging or repair limit and the sleeving SG tube repair process itself, when implementing either laser welded or tungsten inert gas (TIG) welded sleeves, as applicable only to Westinghouse Model D4 SG (i.e., Unit 1 of the Byron and Braidwood Stations) and Westinghouse Model D5 SG (i.e., Unit 2 of the Byron and Braidwood Stations). This revision does not affect in any way the existing definition of the plugging or repair limit for Model D4 or Model D5 SG. Accordingly, the net affect of this revision is to make this definition and the sleeve repair process not applicable to the RSG. This is acceptable in that the RSG do not require SG tube sleeve repairs.

These TS Sections are also being revised to indicate that the repair criteria for SG tube flaws within the OSG tubesheet (i.e., the F* SG tubes) are only applicable to the OSG and not the RSG. This is also acceptable for the same reason cited above.

Section 4.4.5.5, "Reports," in the TS of both stations is being revised to indicate that the results of the F* tube inspections need be submitted to the Commission only for the OSG. This is acceptable in that the revision does not eliminate any existing reporting requirements for the OSG and this revision will not require the licensee to report F* indications in the RSG since the RSG are not subject to the same forms of SG tube degradation in the tubesheet area which have occurred in the OSG.

Section 3.4.8 and 4.4.8 in the TS for both stations are being revised to state that the present maximum permissible primary coolant TS long-term (i.e., greater than 48 hours) value of the DEI concentration of 0.35 μCi/gm is applicable to the OSG in both the Byron and Braidwood Stations only until their replacement by the RSG. After the installation of the RSG in these stations, the long-term DEI TS limits for the RSG are restored to 1.0 μCi/gm for Unit 1 of both stations. Similarly, the short-term maximum permissible DEI in TS Figure 3.4-1 in the TS applicable to the RSG of both stations are revised to restore these values to their original licensing bases. These revisions are acceptable as discussed in Sections 2.2.6 and 2.2.7 of this SE. In summary, the staff concluded in these sections that the behavior of the RSG under transient and postulated accident conditions with the short-term and long-term maximum permissible DEI limits restored to their original licensing bases, did not adversely affect public health and safety.

Finally, the bases sections in the TS of both stations are revised to reflect the forthcoming installation of the RSG and the restoration of the TS limits for the DEI to their original licensing bases.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Illinois State official was notified of the proposed issuance of the amendments. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (62 FR 66134). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Attachments: As Stated

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Date: February 3, 1998

TABLE 1

Assumptions for the Postulated Main Steamline Break Accident

BWI Replacement Steam Generators

Byron and Braidwood Stations, Unit 1

Iodine Partition Factor	
Faulted SG	1.0
Intact SGs	0.1
Steam and H_2O Releases from Faulted SG	
0-120 minutes	1.18E5
Steam Release from Intact SGs (lbs)	
0-2 hours	4.10E5
2-8 hours	9.49E5
8-16 hours	6.77E5
16-24 hours	5.81E5
24-32 hours	5.22E5
32-40 hours	4.82E5
Primary to Secondary Leak Rate (gpd/SG)	150
Time to Isolate Faulted SG (min)	120
Flashing Fraction	Variable with respect to time. Provided in ComEd letter dated 11/3/97.
Scrubbing Fraction	0
Primary Bypass Fraction for Intact SGs	0
Duration of Plant Cooldown (hrs)	40

TABLE 1 (continued)

Assumptions for the Postulated Main Steamline Break Accident

BWI Replacement Steam Generators

Byron and Braidwood Stations, Unit 1

Primary coolant concentration of 60 μCi/gm of dose equivalent ^{131}I.

Pre-existing Spike Value (μCi/gm)

^{131}I = 46.2
^{132}I = 51.7
^{133}I = 73.9
^{134}I = 11.1
^{135}I = 40.6

Volume and Mass of primary coolant and secondary coolant.

Primary Coolant Volume (ft^3)	12,062 @586.2 °F
Primary Coolant Temperature (°F)	586.2
Mass of Primary Coolant (lbs)	538,361
Primary Coolant Pressure (psia)	2,293
Pressurizer Temperature (°F)	657
Pressurizer Pressure (psia)	2,293
Pressurizer Volume (ft^3)	1,150
Secondary Coolant Steam Volume (ft^3)	2,780
Secondary Coolant Liquid Volume (ft^3)	2,423
Secondary Coolant Steam Mass/SG (lbs)	5,571
Secondary Coolant Liquid Mass/SG (lbs)	105,224
Secondary Coolant Steam Temperature (°F)	523
Secondary Coolant Feedwater Temperature (°F)	440

TS limits for DE ^{131}I in the primary and secondary coolant.

Primary Coolant DE ^{131}I concentration (μCi/gm)	
Maximum Instantaneous Value	60
48 Hour Value	1.0
Secondary Coolant DE ^{131}I concentration (μCi/gm)	0.1

TABLE 1 (continued)

Assumptions for the Postulated Main Steamline Break Accident

BWI Replacement Steam Generators

Byron and Braidwood Stations, Unit 1

TS value for the primary to secondary leak rate.

Primary to secondary leak rate, any SG (gpd)	150
Primary to secondary leak rate, total (gpd)	600

Maximum primary to secondary leak rate to the faulted and intact SGs.

Faulted SG (gpm)	150
Intact SGs (gpm/SG)	150

Letdown Flow Rate (gpm)	75

Equilibrium Release Rate from Fuel for 1 μCi/gm of Dose Equivalent ^{131}I

	Ci/day
^{131}I =	2,040
^{132}I =	5,300
^{133}I =	5,330
^{134}I =	7,370
^{135}I =	5,300

Control Room

Free Volume (ft^3)	4.05E5
Filtered Recirculation Flow (cfm)	4.45E4
Recirculation Efficiency for all forms of Iodine (%)	90
Makeup Filter Efficiency for all forms of Iodine (%)	99
Makeup Air Filtration Rate (cfm)	5400

TABLE 2

Assumptions for the Postulated Main Steamline Break Accident

BWI Replacement Steam Generators

Byron Station, Unit 1

Control Room	
Unfiltered Air Infiltration Rate (cfm)	89
Occupancy Factors	
0-1 day	1.0
1-4 days	0.6
Atmospheric Dispersion Factors (sec/m^3)	
Control Room	
0-8 hours	4.05E-3
8-24 hours	1.9E-3
1-4 days	5.7E-4
4-30 days	3.8E-4
EAB	6.8E-4
LPZ	
0-8 hours	2.3E-5
8-24 hours	1.5E-5
1-4 days	6.4E-6
4-30 days	1.4E-6
Spiking Factor for Accident-Initiated Spike	500

TABLE 3

Assumptions for the Postulated Main Steamline Break Accident

BWI Replacement Steam Generators

Braidwood Station, Unit 1

Control Room	
Unfiltered Air Infiltration Rate (cfm)	25
Occupancy Factors	
0-1 day	1.0
1-4 days	0.6
Atmospheric Dispersion Factors (sec/m^3)	
Control Room	
0-8 hours	6.2E-3
8-24 hours	3.2E-3
1-4 days	8.4E-4
4-30 days	1.4E-4
EAB	7.7E-4
LPZ	
0-8 hours	7.9E-5
8-24 hours	5.2E-5
1-4 days	2.1E-5
4-30 days	5.6E-6
Spiking Factor for Accident-Initiated Spike	500

TABLE 4

Assumptions for Postulated Steam Generator Tube Rupture (SGTR) Accident

BWI Replacement Steam Generators

Byron and Braidwood Stations, Unit 1

Iodine Partition Factor	0.01
Steam Release from Defective SG	
0-2 hours(lbs)	9.55E4
>2 hours (lbs)	0
Steam Release from Intact SGs (lbs)	
0-2 hours	1.61E5
2-40 hours	assumed to be at the same rate as for a postulated main steamline break
Reactor Coolant Released to Faulted SG (lbs)	1.41E5
Primary to Secondary Leak Rate (gpd/SG)	150
Time to Isolate Faulted SG (sec)	3300

TABLE 5

Assumptions for the Postulated Locked Rotor Accident

BWI Replacement Steam Generators

Byron and Braidwood Stations, Unit 1

Core Thermal Power Level (MWt)	3565
Duration of Plant Cooldown by Secondary System (hr)	40
Gap Fraction:	
Iodines	0.12
^{85}Kr	0.30
All others	0.10
Cladding Failure (%)	5
Primary to Secondary Leak Rate (gpd/SG)	150
Iodine Partition Factor in SG	0.01
Steam Released from 4 SGs (lbs)	
0-2 hours	5.65E5
2-8 hours	1.09E6
8-16 hours	8.03E5
16-24 hours	7.07E5
24-32 hours	6.48E5
32-40 hours	6.08E5
Primary Coolant Dose Equivalent ^{131}I Level (μCi/gm)	60

TABLE 6

Assumptions for the Postulated Rod Ejection Accident

BWI Replacement Steam Generators

Byron and Braidwood Stations, Unit 1

Core Thermal Power (MWt)	3565
Fuel Cladding Defects (%)	10
Primary to Secondary Leak Rate (gpd/SG)	150
Melted Fuel (% of core)	5
Activity released to reactor coolant from melted fuel and available for release (%)	
Primary to Secondary Release Pathway	100 for noble gases 50 for iodides
Containment Pathway	100 for noble gases 25 for iodides
Iodine Partition Factor	
From Relief Valves	1.0
During Steaming	0.01
Containment	
Volume (ft^3)	2.76E6
Leak Rate (%/day)	0.1 for t = 0-1 day 0.05 for t > 1 day
Steam Dump from Relief Valves (lbs)	1.16E5
Duration of Steam Dump from Relief Valves (sec)	500
Time between Accident and Equalization of Primary to Secondary System Pressure (sec)	3250
Steaming Rate	Refer to Table 1 for values

TABLE 7

Thyroid Doses from Postulated Accidents (Rem)

BWI Replacement Steam Generators

Byron Station, Unit 1

	Accident	EAB	LPZ
1.	Main Steamline Break		
	Coincident Spike *	2.4	3.3
	Pre-existing Spike**	3.4	0.55
2.	Steam Generator Tube Rupture		
	Coincident Spike *	4.9	0.23
	Pre-existing Spike**	24.7	0.85
3.	Locked Rotor *	0.60	0.54
4.	Rod Ejection ***		
	Primary to Secondary Pathway	6.1	0.50
	Containment Pathway	67	26

*Acceptance Criterion is 30 rem

**Acceptance Criterion is 300 rem

***Acceptance Criterion is 75 rem

TABLE 8

Thyroid Doses from Postulated Accidents (Rem)

BWI Replacement Steam Generators

Braidwood Station, Unit 1

	Accident	EAB	LPZ
1.	Main Steamline Break		
	Coincident Spike *	2.7	11
	Pre-existing Spike**	3.8	1.9
2.	Steam Generator Tube Rupture		
	Coincident Spike *	5.5	0.79
	Pre-existing Spike**	28	2.9
3.	Locked Rotor *	0.70	2.1
4.	Rod Ejection ***		
	Primary to Secondary Pathway	6.9	1.7
	Containment Pathway	76	104

* Acceptance Criterion is 30 rem

** Acceptance Criterion is 300 rem

*** Acceptance Criterion is 75 rem

TABLE 9

Whole Body Doses from Postulated Accidents (Rem)

BWI Replacement Steam Generators

Byron Station, Unit 1

	Accident	EAB	LPZ
1.	Main Steamline Break		
	Coincident Spike *	<1	<1
	Pre-existing Spike**	<1	<1
2.	Steam Generator Tube Rupture		
	Coincident Spike *	<1	<1
	Pre-existing Spike**	<1	<1
3.	Locked Rotor *	0.18	0.069
4.	Rod Ejection ***		
	Primary to Secondary Pathway	0.27	0.009
	Containment Pathway	0.39	0.039

* Acceptance Criterion is 2.5 rem

** Acceptance Criterion is 25 rem

*** Acceptance Criterion is 6.25 rem

TABLE 10

Whole Body Doses from Postulated Accidents (Rem)

BWI Replacement Steam Generators

Braidwood Station, Unit 1

	Accident	EAB	LPZ
1.	Main Steamline Break		
	Coincident Spike *	<1	<1
	Pre-existing Spike**	<1	<1
2.	Steam Generator Tube Rupture		
	Coincident Spike *	<1	<1
	Pre-existing Spike**	<1	<1
3.	Locked Rotor *	0.20	0.27
4.	Rod Ejection ***		
	Primary to Secondary Pathway	0.30	0.031
	Containment Pathway	0.44	0.14

* Acceptance Criterion is 2.5 rem

** Acceptance Criterion is 25 rem

*** Acceptance Criterion is 6.25 rem

TABLE 11

Control Room Doses from Postulated Accidents (Rem)

BWI Replacement Steam Generators

Byron Station, Unit 1

	Accident	Thyroid	Whole Body
1.	Main Steamline Break		
	Coincident Spike	2.3	<1
	Pre-existing Spike	0.68	<1
2.	Steam Generator Tube Rupture		
	Coincident Spike	0.15	<1
	Pre-existing Spike	0.55	<1
3.	Locked Rotor	0.43	0.76
4.	Rod Ejection		
	Primary to Secondary Pathway	0.31	0.11
	Containment Pathway	17	0.45

* Acceptance criteria are 5 rem whole body and 30 rem thyroid.

TABLE 12

Control Room Doses from Postulated Accidents (Rem)

BWI Replacement Steam Generators

Braidwood Station, Unit 1

	Accident	Thyroid	Whole Body
1.	Main Steamline Break		
	Coincident Spike	1.8	<1
	Pre-existing Spike	0.29	<1
2.	Steam Generator Tube Rupture		
	Coincident Spike	0.11	<1
	Pre-existing Spike	0.41	<1
3.	Locked Rotor	0.34	1.2
4.	Rod Ejection		
	Primary to Secondary Pathway	0.24	0.16
	Containment Pathway	10	0.69

* Acceptance criteria are 5 rem whole body and 30 rem thyroid.