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

50-454

June 22, 1995

Mr. D. L. Farrar
Manager, Nuclear Regulatory Services
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 500
Downers Grove, IL 60515

SUBJECT: ISSUANCE OF AMENDMENTS (TAC NOS. M89523, M89524, M89525, AND M89526)

Dear Mr. Farrar:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 72 to Facility Operating License No. NPF-37 and Amendment No. 72 to Facility Operating License No. NPF-66 for the Byron Station, Unit Nos. 1 and 2, respectively, and Amendment No. 63 to Facility Operating License No. NPF-72 and Amendment No. 63 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 May 20, 1994, as revised on February 2, 1995, and supplemented on December 2, 1994, and March 14, 1995.

The amendments revise the Technical Specifications (TSs) as they apply to Byron, Unit 1, and Braidwood, Unit 1, to incorporate an alternative repair criteria for defects found in the portion of the expanded steam generator tubes within the tubesheet. Under the amendments, steam generator tubes with degradation in excess of the current plugging limits could remain in service without repair provided the degradation exists below a specified distance (F* (F-star)) from the secondary face of the tubesheet or the top of the last hardroll, whichever is further into the tubesheet.

Although the specific changes apply to Byron, Unit 1, and Braidwood, Unit 1, the licenses for Byron, Unit 2, and Braidwood, Unit 2, are also being amended because both Byron and Braidwood have common TSs.

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,

Original signed by:

George F. Dick, Jr., 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. 72 to NPF-37
2. Amendment No. 72 to NPF-66
3. Amendment No. 63 to NPF-72
4. Amendment No. 63 to NPF-77
5. Safety Evaluation

cc w/encl: see next page

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DOCUMENT NAME: BRAID-BY\BB89523.AMD

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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. 72
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 May 20, 1994, as revised on February 2, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 72 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

George F. Dick, Jr.

George F. Dick, 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: June 22, 1995

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-455

BYRON STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 72
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 May 20, 1994, as revised on February 2, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter 1;

   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. 72 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

George F. Dick Jr

George F. Dick, 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: June 22, 1995

ATTACHMENT TO LICENSE AMENDMENT NOS. 72 AND 72

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
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 4-17a	3/4 4-17a
3/4 4-17b	3/4 4-17b
B 3/4 4-3	B 3/4 4-3
B 3/4 4-3a	B 3/4 4-3a

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

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

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

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

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

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

c. The tubes selected as the second and third samples (if required by Table 4.4-2) during each inservice inspection may be subjected to a partial tube inspection provided:

1) The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found, and

2) The inspections include those portions of the tubes where imperfections were previously found.

d. For Unit 1, Cycle 7 implementation of the tube support plate interim plugging criteria limit requires a 100% bobbin coil probe inspection for all hot leg tube support plate intersections and all cold leg intersections down to the lowest cold leg tube support plate with outer diameter stress corrosion cracking (ODSCC) indications. The determination of the tube support plate intersections having ODSCC indications shall be based on the performance of at least a 20% random sampling of tubes inspected over their full length.

e. A random sample of at least 20 percent of the total number of sleeves shall be inspected for axial and circumferential indications at the end of each cycle. In the event that an imperfection of 40 percent or greater depth is detected, an additional 20 percent of the unsampled sleeves shall be inspected, and if an imperfection of 40 percent or greater depth is detected in the second sample, all remaining sleeves shall be inspected. These inservice inspections will include the entire sleeve and the tube at the heat treated area. The inservice inspection for the sleeves is required until the corrosion resistance for the laser welded or kinetically welded joints in tubes that bound the material

SURVEILLANCE REQUIREMENTS (Continued)

parameters of the tubes installed in the steam generators has been demonstrated acceptable. If conformance with the acceptable criteria of Specification 4.4.5.4 for tube structural integrity is not confirmed, the tubes containing the sleeves in question shall be removed from service.

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

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

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

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

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

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

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

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

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

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

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

4.4.5.4 Acceptance Criteria

a. As used in this specification:

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

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

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

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

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

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

For Unit 1 Cycle 7, 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

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

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

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

where:

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

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

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

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

4.4.5.5 Reports

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

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

1) Number and extent of tubes inspected,

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

3) Identification of tubes plugged or repaired.

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

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

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

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

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

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

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

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

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

1) Identification of F^* Tubes, and

2) Location and size of the degradation.

3/4.4.5 STEAM GENERATORS

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

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

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

3/4.4.5 STEAM GENERATORS (Continued)

For Unit 1 Cycle 7, tubes experiencing outer diameter stress corrosion cracking within the thickness of the tube support plates will be dispositioned in accordance with Specification 4.4.5.4.a.11. The operating period may be adjusted to less than the full operating cycle to meet the maximum site allowable primary-to-secondary leakage limit for End of Cycle Main Steam Line Break conditions. The leakage limit, 12.8 gpm, includes the accident leakage from IPC in addition to the accident leakage from F^* on the faulted steam generator and the operational leakage limit of Specification 3.4.6.2.c. The operational leakage limit of Specification 3.4.6.2.c in each of the three remaining intact steam generators shall include the operational leakage from F^*.

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

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

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

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. 63
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 May 20, 1994, as revised on February 2, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 63 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Ramin R. Assa, 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: June 22, 1995

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-457

BRAIDWOOD STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 63
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 May 20, 1994, as revised on February 2, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter 1;

   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. 63 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

Ramin R. Assa

Ramin R. Assa, 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: June 22, 1995

ATTACHMENT TO LICENSE AMENDMENT NOS. 63 AND 63

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

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

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

Remove Pages	Insert Pages
3/4 4-14	3/4 4-14
3/4 4-14a	3/4 4-14a
3/4 4-16	3/4 4-16
3/4 4-17a	3/4 4-17a
3/4 4-17b (Unit 1)	3/4 4-17b
B 3/4 4-3	B 3/4 4-3
B 3/4 4-3a	B 3/4 4-3a

SURVEILLANCE REQUIREMENTS (Continued)

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

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

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

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

5) For Unit 1, 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 5, implementation of the tube support plate interim plugging criteria limit requires a 100% bobbin coil probe inspection for all hot leg tube support plate intersections and all cold leg intersections down to the lowest cold leg tube support plate with outer diameter stress corrosion cracking (ODSCC) indications. An inspection using a rotating pancake coil (RPC) probe is required in order to show OPERABILITY of tubes with flaw-like bobbin coil signal amplitudes greater than 1.0 volt but less than or equal to 2.7 volts. For tubes that will be administratively plugged or repaired, no RPC inspection is required. The RPC results are to be evaluated to establish that the principal indications can be characterized as ODSCC.

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

SURVEILLANCE REQUIREMENTS (Continued)

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

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

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

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

4.4.5.4 Acceptance Criteria

a. As used in this specification:

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

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

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

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

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

6) Plugging or Repair Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging or repaired by sleeving in the affected area. The plugging or repair limit imperfection depth is equal to 40% of the nominal wall thickness. For Unit 1, this definition does not apply to defects in the tubesheet that meet the criteria for an F* tube; for Unit 1 Cycle 5, this definition does not apply to the region of the tube subject to the tube support plate interim plugging criteria limit, i.e., the tube support plate intersections. Specification 4.4.5.4.a.11 describes the repair limit for use within the tube support plate intersection of the tube;

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

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

4. A tube with a flaw-like bobbin coil signal amplitude of greater than 2.7 volts shall be plugged or repaired.

Certain tubes identified in WCAP-14046, "Braidwood Unit 1 Technical Support for Cycle 5 Steam Generator Interim Plugging Criteria," dated May 1994, shall be excluded from application of the tube support plate interim plugging criteria limit. It has been determined that these tubes may collapse or deform following a postulated LOCA + SSE.

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

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

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

4.4.5.5 <u>Reports</u>

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

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

1) Number and extent of tubes inspected,

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

3) Identification of tubes plugged or repaired.

SURVEILLANCE REQUIREMENTS (Continued)

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

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

   1. Listing of the applicable tubes,

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

   3. Projected Steam Line Break (MSLB) Leakage.

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

   1) Identification of F^* Tubes, and

   2) Location and size of the degradation.

3/4.4.5 STEAM GENERATORS

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

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

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

3/4.4.5 STEAM GENERATORS (continued)

For Unit 1 Cycle 5, tubes experiencing outer diameter stress corrosion cracking within the thickness of the tube support plates will be dispositioned in accordance with Specification 4.4.5.4.a.11. The leakage limit, 9.1 gpm, includes the accident leakage from IPC in addition to the accident leakage from F^* on the faulted steam generator and the operational leakage limit of Specification 3.4.6.2.c. The operational leakage limit of Specification 3.4.6.2.c in each of the three remaining intact steam generators shall include the operational leakage from F^*.

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

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

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 72 TO FACILITY OPERATING LICENSE NO. NPF-37,

AMENDMENT NO. 72 TO FACILITY OPERATING LICENSE NO. NPF-66,

AMENDMENT NO. 63 TO FACILITY OPERATING LICENSE NO. NPF-72,

AND AMENDMENT NO. 63 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 May 20, 1994, Commonwealth Edison Company (ComEd, the licensee) submitted a request to amend the Technical Specifications (TSs) as they apply to Byron Station, Unit 1, and Braidwood Station, Unit 1. The application was revised by letter dated February 2, 1995. The amendment requests proposed an alternative repair criteria for defects found in the steam generator tube expansion region within the tubesheet. Under the requests, steam generator tubes with degradation in excess of the current plugging limits could remain in service without repair provided the indications exist below a specified distance (F* (F-star)), from the secondary face of the tubesheet or the top of last hardroll, whichever is further into the tubesheet. To support the licensee's request, Babcock & Wilcox Nuclear Technologies (BWNT) completed a qualification test program to demonstrate that the proposed F* distance satisfies the necessary structural and leakage integrity requirements of Appendix A to 10 CFR Part 50 and the plant TSs. The results of the test program (W-D4 F* Qualification Report) were included with the May 20, 1994, request in both the proprietary version (BAW 10196P) and non proprietary version (BAW 10196).

Surveillance requirements within plant TSs require a periodic inspection of steam generator tubes for the detection of potential degradation (i.e., cracks, dents, corrosion, etc.) which could diminish the structural margins and leakage integrity of the tubes. Detection of tube degradation in excess of the TS limits requires a repair or removal of the tube from service. The licensee has proposed a revised repair criteria that would allow steam generator tube defects to remain in place without repair provided the defects reside a specified distance below the secondary face of the tubesheet. This distance is called F*. Degradation identified in a steam generator tube below the F* length would be allowed to remain in service without repair. This is based on the results of testing which determined the minimum interference fit engagement length necessary to retain steam generator tubes within the tubesheet.

A staff request for additional information (RAI) was sent to the licensee on November 3, 1994. The licensee responded by letter dated December 2, 1994. A second RAI was sent on February 22, 1995, and the licensee responded by letter dated March 14, 1995. The forgoing responses by the licensee provided clarifying information within the scope of the license amendment applications and did not affect the proposed no significant hazards consideration determination. The staff has reviewed the information supplied by the licensee and completed an evaluation of the licensee's request to amend the Byron, Unit 1, and Braidwood, Unit 1, TSs to include the F* criteria.

Byron, Unit 1, and Braidwood, Unit 1, were both constructed with Westinghouse D4 steam generators. Because the steam generators are of the same design and the accident loads imposed on the tubes are similar, this safety evaluation addresses the proposed F* amendment for both units.

2.0 BACKGROUND

Steam generator tubes comprise a significant portion of the reactor coolant pressure boundary. Maintenance of this barrier is provided by the integrity of the steam generator tube wall and the tube-to-tubesheet connection. The connection between the tube and tubesheet is an interference fit made by roll expanding the tube into a bore through the tubesheet. The tubes were originally installed by first expanding them into the tubesheet followed by seal welding at the primary face. Step rolls were then performed to fully expand the tube within the tubesheet. The inelastically deformed steam generator tube is held in place by the elastic springback of the tubesheet. Undegraded, the tube-to-tubesheet joint provides sufficient strength to maintain adequate structural and pressure boundary (leakage) integrity.

General Design Criteria (GDC) 14, "Reactor Coolant Pressure Boundary," and GDC 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," of Appendix A to 10 CFR Part 50 state the requirements applicable to maintaining adequate structural and leakage integrity for steam generator tubes. Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," describes an acceptable method for establishing the limiting safe conditions of tube degradation of steam generator tubing.

In order to demonstrate adequate structural margin for degraded steam generator tubes, the bases for the proposed steam generator tube repair criterion must address the limiting conditions during normal operation, anticipated operational occurrences, and postulated accident conditions. The margin of failure under normal operating conditions as recommended in RG 1.121 should not be less than three at any tube location. Subsection NB-3225 of Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) specifies the margins of safety under postulated accident conditions.

Structural loads imposed on the steam generator tube-to-tubesheet connections primarily result from the differential pressure between the primary and secondary sides of the tubes. The peak postulated loading occurs during a steam line break due to a lowering of the secondary side pressure. However, normal operating loads, cyclic joint loading from major plant transients (i.e., startup/shutdown), and potential thermal expansion loads can also be significant. The analysis (BAW-10196P) supporting the licensee's proposed amendment to the Byron, Unit 1, and Braidwood, Unit 1, TSs addressed the limiting conditions necessary to maintain adequate integrity of the tube-to-tubesheet interference fit. Specifically, the tube must not experience excessive displacement relative to the tubesheet.

The elastic preload between the tube and tubesheet not only prevents pullout of the tube from the tubesheet, but also provides a leaktight barrier minimizing the potential for primary to secondary coolant leakage. With sufficient length of hardroll, the tube-to-tubesheet connection will not allow any leakage under normal and faulted conditions. Steam generator tube through-wall degradation within the roll expanded joint decreases the path length necessary for primary to secondary leakage. The licensee's proposed amendment would permit tubes with such degradation to remain in service provided there exists a sufficient length of undegraded hardroll. Therefore, an acceptable F* distance must be such that leakage integrity is not jeopardized during all analyzed conditions. Leakage through steam generator tubes is limited by plant TSs. The acceptance criteria for the qualification test program established a maximum primary-to-secondary leak rate of 1 gpm through any one steam generator. The maximum tolerable leakage rate through a single F* tube was determined assuming all F* tubes in the affected generator were leaking at the maximum rate.

3.0 EVALUATION

3.1 Testing to Determine F*

Babcock & Wilcox Nuclear Technologies completed a test program to determine the F* distance. Two failure criteria were considered for testing -- the ultimate pullout load of the tube from the tubesheet and primary-to-secondary leakage requirements. The following describes the methodology used in the test program.

3.1.1 Fabrication of Test Specimens

Mockup blocks were fabricated to simulate the actual installed conditions for the tubes within the Byron, Unit 1, and Braidwood, Unit 1, steam generators. Lengths of steam generator tubing were roll expanded into holes drilled through the mockup blocks. Several peripheral tubes were roll expanded in the test block to simulate additional constraint by surrounding tubes. The interior tubes were used for testing. In order to simulate tube wall degradation, tubes were severed at a certain distance below the upper face of the mockup block. Such a configuration is representative of a 360 degree through-wall crack present in the tube.

Several mockup blocks were fabricated for testing. After the tubes were expanded into the blocks and the hardroll length was verified by nondestructive evaluation methods, some of the test blocks were thermally soaked to simulate the effects of actual steam generator service temperature. Heating the test block could theoretically lead to thermal stress relaxation in the roll expansion joint.

To account for potential factors which might affect the calculated F* length, several variables were changed within the test matrix. For example, tubing with both high and low yield strengths were tested. In addition, tubesheet bore surface roughness, as well as tubesheet bore diameter were varied in the test matrix. The results from the tests revealed the effects from these variables.

In the qualification test program the effects of boric acid corrosion were considered. If primary coolant penetrated through the steam generator tube wall and came in contact with the carbon steel tubesheet the potential exists for initiating stress corrosion cracking in the tubesheet. Based on previous studies, the likelihood of developing significant corrosion of the tubesheet bore due to boric acid corrosion is low.

3.1.2 Testing for F* Determination

To determine the necessary roll expansion joint engagement length, BWNT completed a series of mechanical tests on the simulated steam generator tubes. The qualification testing used a combination of both internal pressure and axial loading to simulate the applied loads on the steam generator tubes to determine the F* distance. Under service conditions, the differential pressure acting over the cross section of the tube induces an axial force tending to force the tube out of the tubesheet. This axial load is counterbalanced by the frictional force between the tube and tubesheet from the roll expanded interference fit. The extent of radial interference between the tube and tubesheet increases at operating conditions due to differential thermal expansion forces. This increases the strength of the joint.

The primary-to-secondary differential pressure acts to both increase and decrease the tube-to-tubesheet interference fit. The two competing factors involve the outward expansion of the tube in the radial direction and the tubesheet bowing effect. The primary-to-secondary differential pressure slightly expands the tube in the radial direction strengthening the joint. However, the higher primary side pressure tends to bow the tubesheet in the upward direction. The tubesheet bowing effect results in the dilation of the tubesheet bores decreasing the tube-to-tubesheet joint strength. The net radial force from these phenomena affects the frictional force between the tube and tubesheet resisting pullout. The reduction in tube-to-tubesheet loading from bowing was accounted for analytically in the testing.

Three different mechanical tests were conducted to determine F*; a locked tube test, pressure cycling, and an ultimate load test. All tests were conducted at ambient temperatures. The locked tube test simulated the loading applied to a steam generator tube during cooldown of the plant assuming the tube was locked at a tube support plate location. The unequal coefficients of thermal expansion of the tube wrapper and the tube would lead to an applied tensile load on the tube. For the pressure cycling test, several tubes were subjected to pressure cycling between low and normal operating pressures. Motion of the tube was monitored during the cyclic loading. Finally, tubes were subjected to an ultimate load test. Tubes were internally pressurized and subjected to an increasing axial tensile load until failure. Failure was defined as a relative movement of a specified distance between the tube and tubesheet.

As part of the test program to provide the basis for the proposed F* length, steam generator tubes were subject to leak rate testing. Tubes were internally pressurized to simulate differential pressures during normal operating and faulted conditions. The acceptance criteria for these tests specified an allowable leakage limit. Tube displacements were also monitored during the tests.

3.1.3 F* Test Results

Based on the results of the leakage rate and mechanical testing, the licensee determined a nominal engagement length necessary to ensure adequate margins of safety. Accounting for limited sample size, statistical scatter in the data, and NDE inspection error, this value was increased appropriately. The licensee has proposed that steam generator tubes with degradation in the roll expanded portion of the tube can remain in service if all degradation lies below the F* distance. The F* distance is equal to the 1.7 inches and is measured down from the secondary face of the tubesheet or the top of last hardroll, whichever is further into the tubesheet.

3.2 Evaluation of Proposed Technical Specification Changes

The licensee proposed a revision to the applicable Byron, Unit 1, and Braidwood, Unit 1, TSs to implement the F* criterion. The following summarizes the proposed changes:

1. The TSs define the F* distance, that has been determined to be 1.7 inches, as the distance into the tubesheet from the secondary face of the tubesheet or the top of the last hardroll, whichever is further into the tubesheet.

2. The revised TSs include a definition of an F* tube, which is a steam generator tube with degradation below the F* distance and has no indications of degradation within the F* distance.

3. In addition to the minimum sample size for steam generator tube inspection, all F* tubes in the tubesheet region will be inspected during all future outages.

In support of the proposed amendment to the Byron, Unit 1, TSs and the Braidwood, Unit 1, TSs, tests were completed to determine an acceptable F* distance. The testing utilized specimens which reflect the actual tube-to-tubesheet joint configuration within the plant steam generators. Unknown variables, which could potentially affect the calculated F* distance were taken into consideration in developing the test matrix. Applied loads for structural assessment and leakage rate testing were specified in accordance with staff recommendations in RG 1.121 and the ASME Code. The licensee's proposed changes to the Byron, Unit 1, and Braidwood, Unit 1, TSs are consistent with the conclusions from the test program to determine F*.

To ensure continued integrity of F* tubes, the licensee has incorporated a requirement into the plant TSs to reinspect F* tubes during all future outages. The continued inspection of F* tubes during each examination will minimize the potential for the existence of degradation within the F* distance of all F* tubes.

The staff has reviewed the TS change related to the implementation of the F* criterion proposed by the licensee in their submittals dated May 20, 1994, December 2, 1994, February 2, 1995, and March 14, 1995. Based on information provided in these submittals, the staff finds the licensee's proposed changes acceptable.

4.0 SUMMARY

The licensee submitted a proposed amendment to the applicable TSs for Byron Station, Unit 1, and Braidwood Station, Unit 1. The proposed changes would permit steam generator tubes to remain in service with degradation in excess of the current plugging limit provided the degradation exists below the F* distance. The proposed F* distance is 1.7 inches measured down into the tubesheet from the secondary face of the tubesheet or the top of the last hardroll, whichever is further into the tubesheet. The licensee evaluated a worst-case flaw present at the F* distance and concluded that the proposed changes are consistent with the NRC guidelines for developing steam generator tube plugging criteria.

The staff has reviewed the licensee's submittals and concludes that the proposed TS changes applicable to Byron, Unit 1, and Braidwood, Unit 1, on steam generator tube surveillance requirements are acceptable.

5.0 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.

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with regard to the installation or use of a facility component located within the restricted area 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 (59 FR 34659 and 60 FR 16184). 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.

7.0 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.

Principal Contributor: P. Rush

Date: June 22, 1995