Docket File

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

January 6, 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. M89674, M89675, M89672, AND M89673)

Dear Mr. Farrar:

The Commission has issued the enclosed Amendment No. 67 to Facility Operating License No. NPF-37 and Amendment No. 67 to Facility Operating License No. NPF-66 for the Byron Station, Unit Nos. 1 and 2, respectively, and Amendment No. 57 to Facility Operating License No. NPF-72 and Amendment No. 57 to Facility Operating License No. NPF-77 for the Braidwood Station, Unit Nos. 1 and 2, respectively. To re-establish parallel amendment numbers for both Braidwood units, Amendment No. 56 for Braidwood Unit 2 will not be used.

On March 4, 1994, the NRC issued Amendment No. 58 to the Byron operating licenses and Amendment No. 46 to the Braidwood operating licenses. Those amendments were issued contingent upon four (4) conditions as specified in NRC's letter of February 22, 1994. Commonwealth Edison Company's (ComEd's) letter of February 24, 1994, committed to meet the four (4) conditions. To assure completion of the commitments, ComEd submitted an application for license amendments on June 3, 1994. These amendments are in response to that request.

The amendments change the Byron and Braidwood operating licenses as well as Technical Specifications 3.4.5.2 and 3.4.6.2 and associated bases to reflect three of the four conditions committed to by ComEd. The nature of the three conditions is to: (1) lower the steam generator primary-to-secondary leakage rate limit, (2) increase the sample size for the inservice inspection of steam generator tube sleeves, and (3) add a condition to the license to conduct additional corrosion testing of the SG tube sleeves.

With regard to the fourth condition, the licensee has indicated that it will incorporate the requirements for post-weld heat treatment into the appropriate sleeving installation procedures. Although this condition is not addressed by the amendments, the licensee's proposal to address the condition is acceptable to the NRC.

DF01

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

Ramin R. Assa, 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, STN 50-457

Enclosures:
1. Amendment No. 67 to NPF-37
2. Amendment No. 67 to NPF-66
3. Amendment No. 57 to NPF-72
4. Amendment No. 57 to NPF-77
5. Safety Evaluation

cc w/encls: see next page

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COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-454

BYRON STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 67
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 June 3, 1994, 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 adding paragraph 2.C.(16)* and by changes to the Technical Specifications as indicated in the attachment to this license amendment. Paragraph 2.C.(2) is amended and 2.C.(16) is added to Facility Operating License No. NPF-37 to read as follows:

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*Pages 6 and 7 are attached, for convenience, for the composite license to reflect this change.

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 67 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.

(16) Steam Generator Sleeving Corrosion Testing

The licensee shall conduct additional corrosion testing to establish the design life for the kinetically or laser welded sleeved tubes in the presence of a crevice. The corrosion testing shall demonstrate the corrosion resistance for the kinetically or laser welded joints in tubes that bound the material parameters in the steam generators. The corrosion testing results shall be reviewed and accepted by the Nuclear Regulatory Commission prior to the Beginning-of-Cycle 9. If conformance with the requirements of the plant Technical Specifications for tube structural integrity is not confirmed, the tubes containing the sleeves in question shall be removed from service.

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

FOR THE NUCLEAR REGULATORY COMMISSION

for

George F. Dick, Jr., Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachments:
1. License pages 6 and 7
2. Changes to the Technical Specifications

Date of Issuance: January 6, 1995

(14) Turbine Missiles (Section 3.5.1.3, SSER #5)

The licensee shall volumetrically inspect all three low pressure turbine rotors by every third refueling outage, until a turbine system maintenance program based on the manufacturer's calculations of missile generation probabilities is approved by the staff.

(15) Operating Staff Experience Requirements (Section 13.1.2.1, SSER #5)

The licensee shall have a licensed senior operator on each shift who has had at least six months of hot operating experience on a similar type plant, including at least six weeks at power levels greater than 20% of full power, and who has had start-up and shutdown experience, except as follows. For those shifts where such an individual is not available on the plant staff, an advisor shall be provided who has had at least four years of power plant experience, including two years of nuclear plant experience, and who has had at least one year of experience on shift as a licensed senior operator at a similar type facility. Use of advisors who were licensed only at the RO level will be evaluated on a case-by-case basis. Advisors shall be trained on plant procedures, technical specifications and plant systems, and shall be examined on theses topics at a level sufficient to assure familiarity with the plant. For each shift, the remainder of the shift crew shall be trained as to the role of the advisors. These advisors shall be retained until the experience levels identified in the first sentence above have been achieved. The NRC shall be notified at least 30 days prior to the date that the licensee proposes to release the advisors from further service.

(16) Steam Generator Sleeving Corrosion Testing

The licensee shall conduct additional corrosion testing to establish the design life for the kinetically or laser welded sleeved tubes in the presence of a crevice. The corrosion testing shall demonstrate the corrosion resistance for the kinetically or laser welded joints in tubes that bound the material parameters in the steam generators. The corrosion testing results shall be reviewed and accepted by the Nuclear Regulatory Commission prior to the Beginning-of-Cycle 9. If conformance with the requirements of the plant Technical Specifications for tube structural integrity is not confirmed, the tubes containing the sleeves in question shall be removed from service.

D. The facility requires exemptions from certain requirements of Appendices A, E and J to 10 CFR Part 50. These include (a) an exemption from the requirements of Paragraph III.D.2(b)(ii) of Appendix J, the testing of containment air locks at times when containment integrity is not required (Section 6.2.6 of the SER), (b) an exemption from GDC-2 of Appendix A, the requirement that structures, systems and components important to safety be designed to withstand the effects of natural phenomena such as earthquakes (Section 3.10 of SSER #5), (c) an exemption from GDC-13 and GDC-17 of Appendix A, the requirement that instrumentation be provided to monitor variables and systems over their anticipated ranges, and the requirement that provisions be included to minimize the probability of losing electric power (Section 9.5.4.1 of SSER #5), (d) an exemption from GDC-19 of Appendix A, the requirement that the control room have adequate radiation protection to permit access and occupancy under accident conditions (Section 6.5.1 of SSER #6), and (e) an exemption from the requirement of Section IV.F of Appendix E that a full participation emergency planning exercise be conducted within one year before issuance of the first operating license for full power and prior to operation above 5% of rated power (Section 13.3 of SSER #6). These exemptions are authorized by law and will not endanger life or property or the

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. 67
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 June 3, 1994, 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 adding paragraph 2.C.(5)* and by changes to the Technical Specifications as indicated in the attachment to this license amendment. Paragraph 2.C.(2) is amended and 2.C.(5) is added to Facility Operating License No. NPF-37 to read as follows:

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*Pages 3 and 3a are attached, for convenience, for the composite license to reflect this change.

(2) Technical Specifications

The Technical Specifications contained in Appendix A (NUREG-1113), as revised through Amendment No. 67 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.

(5) Steam Generator Sleeving Corrosion Testing

The licensee shall conduct additional corrosion testing to establish the design life for the kinetically or laser welded sleeved tubes in the presence of a crevice. The corrosion testing shall demonstrate the corrosion resistance for the kinetically or laser welded joints in tubes that bound the material parameters in the steam generators. The corrosion testing results shall be reviewed and accepted by the Nuclear Regulatory Commission prior to the Beginning-of-Cycle 8. If conformance with the requirements of the plant Technical Specifications for tube structural integrity is not confirmed, the tubes containing the sleeves in question shall be removed from service.

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

FOR THE NUCLEAR REGULATORY COMMISSION

for

George F. Dick, Jr., Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachments:
1. License pages 3 and 3a
2. Changes to the Technical Specifications

Date of Issuance: January 6, 1995

(4) ComEd, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) ComEd, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to posses, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at reactor core power levels not in excess of 3411 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein. The items identified in Attachment 1 to this license shall be completed as specified. Attachment 1 is hereby incorporated into this license.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A (NUREG-1113), as revised through Amendment No.   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) Initial Test Program

Any changes to the Initial Startup Test Program described in Chapter 14 of the FSAR made in accordance with the provisions of 10 CFR 50.59 shall be reported in accordance with 50.59(b) within one month of such change.

(4) Regulatory Guide 1.97, Revision 2 Compliance

The licensee shall submit by March 1, 1987, a preliminary report describing how the requirements of Regulatory Guide 1.97, Revision 2 have been or will be met. The licensee shall submit by September 1, 1987, the final report and a schedule for implementation (assuming the NRC approves the DCRDR by March 1, 1987).

(5) Steam Generator Sleeving Corrosion Testing

The licensee shall conduct additional corrosion testing to establish the design life for the kinetically or laser welded sleeved tubes in the presence of a crevice. The corrosion testing shall demonstrate the corrosion resistance for the kinetically or laser welded joints in tubes that bound the material parameters in the steam generators. The corrosion testing results shall be reviewed and accepted by the Nuclear Regulatory Commission prior to the Beginning-of-Cycle 8. If conformance with the requirements of the plant Technical Specifications for tube structural integrity is not confirmed, the tubes containing the sleeves in question shall be removed from service.

ATTACHMENT TO LICENSE AMENDMENT NOS. 67 AND 67

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. Pages marked with an asterisk are provided for convenience only.

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 4-21	3/4 4-21
3/4 4-22*	3/4 4-22*
B 3/4 4-3	B 3/4 4-3
B 3/4 4-4	B 3/4 4-4

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 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.

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 parameters of the tubes installed in the steam generators has been demonstrated acceptable. If conformance with the acceptable criteria

SURVEILLANCE REQUIREMENTS (Continued)

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

SURVEILLANCE REQUIREMENTS (Continued)

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 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

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 gpm UNIDENTIFIED LEAKAGE,

c. 600 gallons per day total reactor-to-secondary leakage through all steam generators not isolated from the Reactor Coolant System and 150 gallons per day through any one steam generator,

d. 10 gpm IDENTIFIED LEAKAGE from the Reactor Coolant System,

e. 40 gpm CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 ± 20 psig, and

f. 1 gpm leakage at a Reactor Coolant System pressure of 2235 ± 20 psig from any Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1.*

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

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE and leakage from Reactor Coolant System Pressure Isolation Valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the above limit, reduce the leakage rate to within limits within 4 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

*Test pressures less than 2235 psig but greater than 350 psig are allowed. Observed leakage shall be adjusted for the actual test pressure up to 2235 psig assuming the leakage to be directly proportional to pressure differential to the one-half power.

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere gaseous and particulate radioactivity monitor at least once per 12 hours;

b. Monitoring the reactor cavity sump discharge, and the containment floor drain sump discharge and inventory at least once per 12 hours;

c. Measurement of the CONTROLLED LEAKAGE from the reactor coolant pump seals when the Reactor Coolant System pressure is 2235 ± 20 psig at least once per 31 days with the modulating valve fully open. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4;

d. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours; and

e. Monitoring the Reactor Head Flange Leakoff System at least once per 24 hours.

4.4.6.2.2 Each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1 shall be demonstrated OPERABLE by verifying leakage to be within its limit:

a. At least once per 18 months,

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 72 hours or more and if leakage testing has not been performed in the previous 9 months,

c. Prior to returning the valve to service following maintenance, repair or replacement work on the valve, and

d. Within 24 hours following valve actuation due to automatic or manual action or flow through the valve except for valves RH 8701 A and B and RH 8702 A and B.

The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

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. 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.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS Leakage Detection Systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. These Detection Systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.6.2 OPERATIONAL LEAKAGE

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 gpm. This threshold value is sufficiently low to ensure early detection of additional leakage.

The total steam generator tube leakage limit of 600 gpd for all steam generators not isolated from the RCS ensures that the dosage contribution from the tube leakage will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of either a steam generator tube rupture or steam line break. A 1 gpm total steam generator tube leakage limit for all steam generators and a 500 gpd limit per steam generator were the assumptions used in the analysis of these accidents in Chapter 15 of the UFSAR. The assumptions in Chapter 15 remain valid since the leakage limitations implemented for total and individual steam generator leakages are more conservative than those used for the analysis. The 150 gpd leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

The 10 gpm IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the Leakage Detection Systems.

The CONTROLLED LEAKAGE limitation restricts operation when the total flow supplied to the reactor coolant pump seals exceeds 40 gpm with the modulating valve in the supply line fully open at a nominal RCS pressure of 2235 psig. This limitation ensures that in the event of a LOCA, the Safety Injection flow will not be less than assumed in the safety analyses.

The 1 gpm leakage from any RCS pressure isolation valve is sufficiently low to ensure early detection of possible in-series check valve failure. It is apparent that when pressure isolation is provided by two in-series check valves and when failure of one valve in the pair can go undetected for a substantial length of time, verification of valve integrity is required. Since these valves are important in preventing overpressurization and rupture of the ECCS low pressure piping which could result in a LOCA that bypasses containment, those valves should be tested periodically to ensure low-probability of gross failure.

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. 57
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 June 3, 1994, 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 adding paragraph 2.C.(6)* and by changes to the Technical Specifications as indicated in the attachment to this license amendment. Paragraph 2.C.(2) is amended and 2.C.(6) is added to Facility Operating License No. NPF-37 to read as follows:

*Pages 3 and 4 are attached, for convenience, for the composite license to reflect this change.

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 57 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.

(6) Steam Generator Sleeving Corrosion Testing

The licensee shall conduct additional corrosion testing to establish the design life for the kinetically or laser welded sleeved tubes in the presence of a crevice. The corrosion testing shall demonstrate the corrosion resistance for the kinetically or laser welded joints in tubes that bound the material parameters in the steam generators. The corrosion testing results shall be reviewed and accepted by the Nuclear Regulatory Commission prior to the Beginning-of-Cycle 7. If conformance with the requirements of the plant Technical Specifications for tube structural integrity is not confirmed, the tubes containing the sleeves in question shall be removed from service.

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

Attachments:
1. License pages 3 and 4
2. Changes to the Technical Specifications

Date of Issuance: January 6, 1995

(3) ComEd, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) ComEd, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) ComEd, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to posses, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at reactor core power levels is not in excess of 3411 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein and other items identified in Attachment 1 to this license. The items identified in Attachment 1 to this license shall be completed as specified. Attachment 1 is hereby incorporated into this license.

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No.   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) Emergency Planning

In the event that the NRC finds that the lack of progress in completion of the procedures in the Federal Emergency Management Agency's final rule, 44 CFR Part 350, is an indication that a major substantive problem exists in achieving or maintaining an adequate state of emergency preparedness, the provisions of 10 CFR Section 50.54(s)(2) will apply.

(4) Initial Startup Test Program

Any changes to the Initial Test Program described in Section 14 of the FSAR made in accordance with the provisions of 10 CFR 50.59 shall be reported in accordance with 50.59(b) within one month of such change.

(5) Regulatory Guide 1.97, Revision 2 Compliance

The licensee shall submit the final report and a schedule for implementation within six months of NRC approval of the DCRDR.

(6) Steam Generator Sleeving Corrosion Testing

The licensee shall conduct additional corrosion testing to establish the design life for the kinetically or laser welded sleeved tubes in the presence of a crevice. The corrosion testing shall demonstrate the corrosion resistance for the kinetically or laser welded joints in tubes that bound the material parameters in the steam generators. The corrosion testing results shall be reviewed and accepted by the Nuclear Regulatory Commission prior to the Beginning-of-Cycle 7. If conformance with the requirements of the plant Technical Specifications for tube structural integrity is not confirmed, the tubes containing the sleeves in question shall be removed from service.

D. The facility requires an exemption from the requirements of Appendix J to 10 CFR Part 50, Paragraph III.D.2(b)(ii), the testing of containment air locks at times when containment integrity is not required (SER Section 6.2.6). This exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security. This exemption is hereby granted. The special circumstances regarding this exemption are identified in the referenced section of the safety evaluation report and the supplements thereto. This exemption is granted pursuant to 10 CFR 50.12. With this exemption, the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

An exemption was previously granted pursuant to 10 CFR 70.24. The exemption was granted with NRC materials license No. SNM-1938, issued October 8, 1985, and relieved the licensee from the requirement of having a criticality alarm system. Therefore, the licensee is exempted from the criticality alarm system provision of 10 CFR 70.24 so far as this section applies to the storage of fuel assemblies held under this license.

E. The licensee shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report, as supplemented and amended, and as approved in the SER dated November 1983 and its supplements, subject to the following provision:

The licensee may make changes to the approved fire protection program without prior approval of the Commission, only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

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. 57
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 June 3, 1994, 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 adding paragraph 2.C.(5)* and by changes to the Technical Specifications as indicated in the attachment to this license amendment. Paragraph 2.C.(2) is amended and 2.C.(5) is added to Facility Operating License No. NPF-37 to read as follows:

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*Pages 4 and 4a are attached, for convenience, for the composite license to reflect this change.

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 57 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.

(5) Steam Generator Sleeving Corrosion Testing

The licensee shall conduct additional corrosion testing to establish the design life for the kinetically or laser welded sleeved tubes in the presence of a crevice. The corrosion testing shall demonstrate the corrosion resistance for the kinetically or laser welded joints in tubes that bound the material parameters in the steam generators. The corrosion testing results shall be reviewed and accepted by the Nuclear Regulatory Commission prior to the Beginning-of-Cycle 7. If conformance with the requirements of the plant Technical Specifications for tube structural integrity is not confirmed, the tubes containing the sleeves in question shall be removed from service.

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

Attachments:
1. License pages 4 and 4a
2. Changes to the Technical Specifications

Date of Issuance: January 6, 1995

(4) Initial Startup Test Program

Any changes to the Initial Test Program described in Section 14 of the FSAR made in accordance with the provisions of 10 CFR 50.59 shall be reported in accordance with 50.59(b) within one month of such change.

(5) Steam Generator Sleeving Corrosion Testing

The licensee shall conduct additional corrosion testing to establish the design life for the kinetically or laser welded sleeved tubes in the presence of a crevice. The corrosion testing shall demonstrate the corrosion resistance for the kinetically or laser welded joints in tubes that bound the material parameters in the steam generators. The corrosion testing results shall be reviewed and accepted by the Nuclear Regulatory Commission prior to the Beginning-of-Cycle 7. If conformance with the requirements of the plant Technical Specifications for tube structural integrity is not confirmed, the tubes containing the sleeves in question shall be removed from service.

D. The facility requires an exemption from the requirements of Appendix J to 10 CFR Part 50, Paragraph III.D.2(b)(ii), the testing of containment air locks at times when containment integrity is not required (SER Section 6.2.6). This exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security. The staff's environmental assessment was published on May 19, 1988 (53 FR 17995). This exemption was granted in the low power license and is continued for the full power license. The special circumstances regarding this exemption are identified in the referenced section of the Safety Evaluation Report and the supplements thereto. This exemption is granted pursuant to 10 CFR 50.12. With this exemption, the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

An exemption was previously granted pursuant to 10 CFR 70.24. The exemption was granted with NRC materials license No. SNM-1938, issued October 8, 1985, and relieved the licensee from the requirement of having a criticality alarm system. Therefore, the licensee is exempted from the criticality alarm system provision of 10 CFR 70.24 so far as this section applies to the storage of fuel assemblies held under this license.

The facility requires a temporary exemption from the requirements of 10 CFR 50.49(f) and 50.49(j).

Title 10 of the Code of Federal Regulations (10 CFR) Part 50.49(a) states:

Each holder of or each applicant for a license to operate a nuclear power plant shall establish a program for qualifying the electric equipment defined in paragraph (b) in this section.

Section 50.49(f) of 10 CFR 50 states:

Each item of electric equipment important to safety must be qualified by one of the following methods:

1. Testing an identical item of equipment under identical conditions or under similar conditions with a supporting analysis to show that the equipment to be qualified is acceptable.

2. Testing a similar item of equipment with a supporting analysis to show that the equipment to be qualified is acceptable.

ATTACHMENT TO LICENSE AMENDMENT NOS. 57 AND 57

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. Pages marked with an asterisk are provided for convenience only.

Remove Pages	Insert Pages
3/4 4-14	3/4 4-14
3/4 4-14a (Unit 1)	3/4 4-14a
3/4 4-21	3/4 4-21
3/4 4-22*	3/4 4-22*
B 3/4 4-3	B 3/4 4-3
B 3/4 4-3a (Unit 1)	B 3/4 4-3a
B 3/4 4-4	B 3/4 4-4

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.

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 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

SURVEILLANCE REQUIREMENTS (Continued)

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.

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 gpm UNIDENTIFIED LEAKAGE,

c. 600 gallons per day total reactor-to-secondary leakage through all steam generators not isolated from the Reactor Coolant System and 150 gallons per day through any one steam generator,

d. 10 gpm IDENTIFIED LEAKAGE from the Reactor Coolant System,

e. 40 gpm CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 ± 20 psig, and

f. 1 gpm leakage at a Reactor Coolant System pressure of 2235 ± 20 psig from any Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1.*

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

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE and leakage from Reactor Coolant System Pressure Isolation Valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the above limit, reduce the leakage rate to within limits within 4 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

*Test pressures less than 2235 psig but greater than 350 psig are allowed. Observed leakage shall be adjusted for the actual test pressure up to 2235 psig assuming the leakage to be directly proporational to pressure differential to the one-half power.

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere gaseous and particulate radioactivity monitor at least once per 12 hours;

b. Monitoring the reactor cavity sump discharge, and the containment floor drain sump discharge and inventory at least once per 12 hours;

c. Measurement of the CONTROLLED LEAKAGE from the reactor coolant pump seals when the Reactor Coolant System pressure is 2235 ± 20 psig at least once per 31 days with the modulating valve fully open. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4;

d. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours; and

e. Monitoring the Reactor Head Flange Leakoff System at least once per 24 hours.

4.4.6.2.2 Each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1 shall be demonstrated OPERABLE by verifying leakage to be within its limit:

a. At least once per 18 months,

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 72 hours or more and if leakage testing has not been performed in the previous 9 months,

c. Prior to returning the valve to service following maintenance, repair or replacement work on the valve, and

d. Within 24 hours following valve actuation due to automatic or manual action or flow through the valve except for valves RH 8701 A and B and RH 8702 A and B.

The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

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. 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.

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.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS Leakage Detection Systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. These Detection Systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.6.2 OPERATIONAL LEAKAGE

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 gpm. This threshold value is sufficiently low to ensure early detection of additional leakage.

The total steam generator tube leakage limit of 600 gpd for all steam generators not isolated from the RCS ensures that the dosage contribution from the tube leakage will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of either a steam generator tube rupture or steam line break. A 1 gpm total steam generator tube leakage limit for all steam generators and a 500 gpd limit per steam generator were the assumptions used in the analysis of these accidents in Chapter 15 of the UFSAR. The assumptions in Chapter 15 remain valid since the leakage limitations implemented for total and individual steam generator leakages are more conservative than those used for the analysis. The 150 gpd leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

The 10 gpm IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the Leakage Detection Systems.

The CONTROLLED LEAKAGE limitation restricts operation when the total flow supplied to the reactor coolant pump seals exceeds 40 gpm with the modulating valve in the supply line fully open at a nominal RCS pressure of 2235 psig. This limitation ensures that in the event of a LOCA, the Safety Injection flow will not be less than assumed in the safety analyses.

The 1 gpm leakage from any RCS pressure isolation valve is sufficiently low to ensure early detection of possible in-series check valve failure. It is apparent that when pressure isolation is provided by two in-series check valves and when failure of one valve in the pair can go undetected for a substantial length of time, verification of valve integrity is required. Since these valves are important in preventing overpressurization and rupture of the ECCS low pressure piping which could result in a LOCA that bypasses containment, those valves should be tested periodically to ensure low-probability of gross failure.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 67 TO FACILITY OPERATING LICENSE NO. NPF-37,

AMENDMENT NO. 67 TO FACILITY OPERATING LICENSE NO. NPF-66,

AMENDMENT NO. 57 TO FACILITY OPERATING LICENSE NO. NPF-72,

AND AMENDMENT NO. 57 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

In a letter of August 13, 1993, as supplemented on September 15, 1993, September 16, 1993, December 17, 1993, January 19, 1994, February 11, 1994, and February 24, 1994, Commonwealth Edison Company (ComEd, the licensee), submitted an amendment request for Operating Licenses NPF-72 and NPF-77 for Braidwood Station, Units 1 and 2, and NPF-37 and NPF-66 for Byron Station, Units 1 and 2, proposing steam generator (SG) tube sleeving in accordance with the (1) Westinghouse (W), and (2) Babcock and Wilcox Company (B&W) processes. The January 19, 1994, and February 11, 1994, supplements submitted additional and clarifying information specifically for Braidwood and Byron in support of the original request. The December 17, 1993, supplement transmitted a reexamination by W of the sleeve repair/plugging limit using the design criteria in lieu of the repair criteria and determined that the through-wall plugging limit for the four units should be 40-percent and identified the material in the submittal that was pertinent for the review of Byron and Braidwood units. The January 19, 1994, supplement converted data from 7/8- to 3/4-inch diameter tubing and supplied additional corrosion test results suitable for the post-weld heat treatment (PWHT) of laser welds. The February 11, 1994, supplement lists the current status on the installation and inspection of laser-welded sleeves.

The W process consists of a laser welding technique to secure a sleeve inside a SG tube to become the primary barrier within a degraded tube. The technical justification supporting the W process is given in WCAP-13698, Revision 1, "Laser Welded Sleeves for 3/4-Inch Diameter Tube Feedring-Type and Westinghouse Preheater Steam Generators," and is supplemented by the December 17, 1993, and January 19, 1994, submittals. The staff reviewed those portions of WCAP-13698, Revision 1, that pertained to Byron and Braidwood SG.

The B&W process uses a kinetic welding technique to secure the sleeves inside SG tubes. Technical justification supporting the B&W process is given in BAW-2045 PA, Revision 1, "Recirculating Steam Generator Kinetic Sleeve Qualification for 3/4-Inch OD Tubes," and is supplemented by the December 17, 1993, and January 19, 1994, submittals.

The staff issued Amendment No. 58 to the Byron Technical Specifications (TS) and Amendment No. 46 to the Braidwood TSs on March 4, 1994. Included among the amendments was the staff's Safety Evaluation (SE) of the sleeving proposal. The March 4, 1994, amendment granted the proposed sleeving methods contingent upon four conditions (Items 1. - 4.) below:

1. Amend the license to reflect a primary-to-secondary leakage limit of 150 gpd.

2. Amend the Byron and Braidwood operating licenses to reflect an inservice inspection of a minimum of 20 percent of installed sleeves for axial and circumferential indications at the end of each operating cycle. In the event that an imperfection of 40 percent through-wall or greater depth is detected, an additional 20 percent (minimum) of the unsampled sleeves would have to be inspected, and should an imperfection of the unsampled sleeves (be) detected in the second sample, all remaining sleeves would be required to be inspected. The inservice inspection is required until the licensee demonstrates the corrosion resistance for laser-welded and kinetically welded joints in tubes that bound the material properties of the tubes installed in the Byron and Braidwood steam generators. If conformance with the requirements of the plant TSs for tube structural integrity is not confirmed, the tubes containing the sleeves in question shall be removed from service.

3. Add a condition to the license to conduct additional corrosion testing to establish the design life for the kinetically or laser welded sleeved tubes in the presence of a crevice. The testing should: determine the effects that material microstructure, chemistry, and joint crevices will have on primary water stress corrosion cracking (PWSCC) initiation and growth; bound the material conditions that exist in the Byron and Braidwood steam generators; and include the associated stress intensity values.

4. Perform post-weld heat treatment at 1400 °F minimum soak temperature with a 5 minute minimum soak time on freespan kinetically or laser welded joints until additional supporting data becomes available.

The staff required ComEd to submit the required amendments to Byron and Braidwood operating licenses and TSs within 90 days of issuance of Amendment No. 58 to the Byron TSs and Amendment No. 46 to the Braidwood TSs. By letter to the staff on February 24, 1994, ComEd committed to the additional amendments as required by the staff.

2.0 EVALUATION

By letter dated June 3, 1994, ComEd submitted a request to amend Operating Licenses NPF-37 and NPF-66 for Byron Station, Units 1 and 2, and Operating Licenses NPF-72 and NPF-77 for Braidwood Station, Units 1 and 2, and Appendix A, Technical Specifications to these Operating Licenses to amend the Steam Generator Primary-to-Secondary Leakage Limits, and amend the inservice inspection requirements as they pertain to inservice inspections of steam generator tubes sleeved by either kinetically welded or laser welded sleeving methods.

The staff has reviewed the licensee's basis for revising the Byron and Braidwood operating licenses and TSs and has determined that the proposed changes are consistent with the commitments made by ComEd for the four contingencies identified in Section 1.0 of this SE. Item 1 is acceptable because the 150 gpd primary-to-secondary leakage limit is more conservative than the 500 gpd it replaces. The leakage limit ensures that any dosage contributed from a potential SG tube leak will be considerably lower than dosage specified in 10 CFR Part 100. Therefore, the 150 gpd lower limit is more conservative. Item 2 is acceptable because it increases the sample size for the inservice inspection of the sleeves to ensure that the structural integrity of the sleeves and the pressure boundary are maintained. The revised TSs require that the additional inspections continue until the corrosion resistance of the welded joints is demonstrated. This testing will provide additional certainty that the sleeved tubes will not leak. The final revision, Item 3, is also acceptable because the corrosion testing program will establish sleeve design life and corrosion resistance to confirm structural integrity of the SG tubes. The licensee has committed to perform a corrosion testing program and submit the result to NRC for approval. The licensee will remove the sleeved tubes from service if the tube structural integrity is not confirmed. Item 4 is not required to be incorporated into a license condition or into the TSs. Therefore, the licensee will address Item 4 by incorporating the requirements for post-weld heat treatment into the appropriate sleeving installation procedures. This is acceptable to the staff.

The staff has reviewed the licensee's basis for amending the Byron and Braidwood operating licensees and TSs, and has determined that ComEd has met the commitments stated in their February 24, 1994, letter.

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 findings (59 FR 51613). 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.

Principal Contributors: J. Medoff
R. Assa

Date: January 6, 1995

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

Ramin R. Assa, 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, STN 50-457

Enclosures:
1. Amendment No. 67 to NPF-37
2. Amendment No. 67 to NPF-66
3. Amendment No. 57 to NPF-72
4. Amendment No. 57 to NPF-77
5. Safety Evaluation

cc w/encls: see next page

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