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

March 4, 1994

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

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

Dear Mr. Farrar:

SUBJECT: ISSUANCE OF AMENDMENTS (TAC NOS. M87227, M87228, M87229, M87230)

The Commission has issued the enclosed Amendment No. 58 to Facility Operating License No. NPF-37 and Amendment No. 58 to Facility Operating License No. NPF-66 for the Byron Station, Unit Nos. 1 and 2, respectively, and Amendment No. 46 to Facility Operating License No. NPF-72 and Amendment No. 46 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 August 13, 1993, as supplemented by letters dated September 15, 1993, September 16, 1993, December 17, 1993, January 19, 1994, February 11, 1994 and February 24, 1994. Your letter dated February 24, 1994, committed the Commonwealth Edison Company to implement the conditions specified in the attached Safety Evaluation. The supplemental letters provided clarifying information, but did not change the no significant hazards consideration determination.

The amendments revise technical specification 3/4.4.5, "Steam Generators" to allow sleeving of defective steam generator tubes as an alternative to tube plugging. The amendments allow the installation of Westinghouse laser-welded and B&W kinetically-welded sleeves inside steam generators tubes at the tube support plate and tubesheet regions.

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

March 4, 1994

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

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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, Acting Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 58 to NPF-37
2. Amendment No. 58 to NPF-66
3. Amendment No. 46 to NPF-72
4. Amendment No. 46 to NPF-77
5. Safety Evaluation

cc w/enclosures:
See next page

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COPY	YES/NO	YES/NO	YES/NO	YES/NO	YES/NO	YES/NO

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,

Ramin R. Assa, Acting Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 58 to NPF-37
2. Amendment No. 58 to NPF-66
3. Amendment No. 46 to NPF-72
4. Amendment No. 46 to NPF-77
5. Safety Evaluation

cc w/enclosures:
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COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-454

BYRON STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 58
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 August 13, 1993, as supplemented by letters dated September 15, 1993, September 16, 1993, December 17, 1993, January 19, 1994, February 11, 1994, and February 24, 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 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. 58 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

James E. Dyer

James E. Dyer, Director
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 4, 1994

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. 58
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 August 13, 1993, as supplemented by letters dated September 15, 1993, September 16, 1993, December 17, 1993, January 19, 1994, February 11, 1994, and February 24, 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 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. 58 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

James E. Dyer

James E. Dyer, Director
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 4, 1994

ATTACHMENT TO LICENSE AMENDMENT NOS. 58 AND 58

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. Overleaf pages have been provided for convenience and are marked with an asterisk.

Remove Pages	Insert Pages
3/4 4-13	3/4 4-13
3/4 4-14	3/4 4-14
* 3/4 4-15	* 3/4 4-15
3/4 4-16	3/4 4-16
3/4 4-17	3/4 4-17
* 3/4 4-18	* 3/4 4-18
3/4 4-19	3/4 4-19
B 3/4 4-3	B 3/4 4-3

3/4.4.5 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

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

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

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

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

SURVEILLANCE REQUIREMENTS (Continued)

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

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

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

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.

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

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

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

SURVEILLANCE REQUIREMENTS (Continued)

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

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

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

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

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

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

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

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

4.4.5.4 Acceptance Criteria

a. As used in this specification:

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

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

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

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

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

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;

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

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

SURVEILLANCE REQUIREMENTS (Continued)

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

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

   a) Laser welded sleeving as described by Westinghouse report WCAP-13698, Rev. 1, or

   b) Kinetic welded sleeving as described by Babcock & Wilcox Topical Report BAW-2045PA, Rev. 1.

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

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.

TABLE 4.4-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE

INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	Yes
No. of Steam Generators per Unit	Four
First Inservice Inspection	Two
Second & Subsequent Inservice Inspections	One[1]

TABLE NOTATION

1. The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions. Each of the other two steam generators not inspected during the first inservice inspections shall be inspected during the second and third inspections. The fourth and subsequent inspections shall follow the instructions described above.

TABLE 4.4-2

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A mininum of S Tubes per S. G.	C-1	None	N.A.	N.A.	N.A.	N.A.
	C-2	Plug or repair defective tubes and inspect additional 2S tubes in this S. G.	C-1	None	N.A.	N.A.
			C-2	Plug or repair defective tubes and inspect additional 4S tubes in this S. G.	C-1	None
					C-2	Plug or repair defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N.A.	N.A.
	C-3	Inspect all tubes in this S. G., plug or repair defective tubes and inspect 2S tubes in each other S. G. Notification to NRC pursuant to §50.72 (b)(2) of 10 CFR Part 50	All other S. G.s are C-1	None	N.A.	N.A.
			Some S. G.s C-2 but no additional S. G. are C-3	Perform action for C-2 result of second sample	N.A.	N.A.
			Additional S. G. is C-3	Inspect all tubes in each S. G. and plug or repair defective tubes. Notification to NRC pursuant to §50.72(b)(2) of 10 CFR Part 50	N.A	N.A.

$S - 3 \frac{N}{n} \%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection

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 = 500 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 500 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown. 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 Westinghouse report WCAP-13698 Rev. 1 and Babcock & Wilcox Topical Report BAW-2045PA Rev. 1.

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 Westinghouse Report WCAP-13698 Rev. 1 and Babcock & Wilcox Topical Report BAW-2045PA Rev. 1. 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.

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. 46
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 August 13, 1993, as supplemented by letters dated September 15, 1993, September 16, 1993, December 17, 1993, January 19, 1994, February 11, 1994, and February 24, 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 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. 46 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

James E. Dyer

James E. Dyer, Director
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: March 4, 1994

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. 46
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 August 13, 1993, as supplemented by letters dated September 15, 1993, September 16, 1993, December 17, 1993, January 19, 1994, February 11, 1994, and February 24, 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 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. 46 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

James E. Dyer

James E. Dyer, Director
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 4, 1994

ATTACHMENT TO LICENSE AMENDMENT NOS. 46 AND 46

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. Overleaf pages have been provided for convenience and are marked with an asterisk.

Remove Pages	Insert Pages
3/4 4-13	3/4 4-13
3/4 4-14	3/4 4-14
* 3/4 4-15	* 3/4 4-15
3/4 4-16	3/4 4-16
3/4 4-17	3/4 4-17
* 3/4 4-18	* 3/4 4-18
3/4 4-19	3/4 4-19
B 3/4 4-3	B 3/4 4-3

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

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

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

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

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

SURVEILLANCE REQUIREMENTS (Continued)

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

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

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

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.

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

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

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

SURVEILLANCE REQUIREMENTS (Continued)

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

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

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

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

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

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

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

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

4.4.5.4 Acceptance Criteria

a. As used in this specification:

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

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

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

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

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

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;

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

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

SURVEILLANCE REQUIREMENTS (Continued)

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

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

   a) Laser welded sleeving as described by Westinghouse report WCAP-13698, Rev. 1, or

   b) Kinetic welded sleeving as described by Babcock & Wilcox Topical Report BAW-2045PA, Rev. 1.

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

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.

TABLE 4.4-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE

INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	Yes
No. of Steam Generators per Unit	Four
First Inservice Inspection	Two
Second & Subsequent Inservice Inspections	One[1]

TABLE NOTATION

1. The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions. Each of the other two steam generators not inspected during the first inservice inspections shall be inspected during the second and third inspections. The fourth and subsequent inspections shall follow the instructions described above.

TABLE 4.4-2

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A mininum of S Tubes per S. G.	C-1	None	N.A.	N.A.	N.A.	N.A.
	C-2	Plug or repair defective tubes and inspect additional 2S tubes in this S. G.	C-1	None	N.A.	N.A.
			C-2	Plug or repair defective tubes and inspect additional 4S tubes in this S. G.	C-1	None
					C-2	Plug or repair defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N.A.	N.A.
	C-3	Inspect all tubes in this S. G., plug or repair defective tubes and inspect 2S tubes in each other S. G. Notification to NRC pursuant to §50.72 (b)(2) of 10 CFR Part 50	All other S. G.s are C-1	None	N.A.	N.A.
			Some S. G.s C-2 but no additional S. G. are C-3	Perform action for C-2 result of second sample	N.A.	N.A.
			Additional S. G. is C-3	Inspect all tubes in each S. G. and plug or repair defective tubes. Notification to NRC pursuant to §50.72(b)(2) of 10 CFR Part 50	N.A	N.A.

$S = 3\frac{N}{n}\%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection

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 = 500 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 500 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown. 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 Westinghouse report WCAP-13698 Rev. 1 and Babcock & Wilcox Topical Report BAW-2045PA Rev. 1.

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 Westinghouse Report WCAP-13698 Rev. 1 and Babcock & Wilcox Topical Report BAW-2045PA Rev. 1. 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.

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. 58 TO FACILITY OPERATING LICENSE NO. NPF-37,

AMENDMENT NO. 58 TO FACILITY OPERATING LICENSE NO. NPF-66,

AMENDMENT NO. 46 TO FACILITY OPERATING LICENSE NO. NPF-72,

AND AMENDMENT NO. 46 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 August 13, 1993, as supplemented by letters dated September 15, 1993, September 16, 1993, December 17, 1993, January 19, 1994, February 11, 1994, and February 24, 1994, Commonwealth Edison Company (CECo, or the licensee) submitted a request for an amendment to the licenses for Braidwood Station (Units 1 and 2) and Byron Station (Units 1 and 2). The proposed amendment would modify the Technical Specifications (TS) for Byron and Braidwood by including the option to repair tubes using Westinghouse and B&W sleeving process and by including sleeves in the SG tubes sample selection and inspection criteria.

The supplements provided additional and clarifying information, specifically for Braidwood and Byron in support of the original request and did not change the original no significant hazards consideration determination. The December 17, 1993, supplement transmitted a reexamination by Westinghouse 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. It also 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-inch to 3/4-inch diameter tubing and supplied additional corrosion test results suitable for the postweld heat treatment (PWHT) of laser welds. The February 11, 1994, supplement listed the current status on the installation and inspection of laser-welded sleeves. By letter dated February 24, 1994, CECo committed to implementing the conditions specified in this safety evaluation.

The Westinghouse process consists of a laser welding technique to secure a sleeve inside a SG tube. The sleeve then becomes the primary pressure boundary within a degraded tube. The technical justification supporting the

Westinghouse process is given in WCAP-13698, Revision 1, "Laser Welded Sleeves for 3/4-Inch Diameter Tube Feeding-Type and Westinghouse Preheater Steam Generators," and is supplemented by the December 17, 1993, and January 19, 1994, submittal. The staff reviewed those portions of WCAP-13698, Revision 1, that pertained to Byron and Braidwood (Units 1 and 2) SGs. The Babcock and Wilcox (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 following evaluations are for the Westinghouse and B&W processes:

2.0 EVALUATION

2.1 BACKGROUND

Tubes in the SG of an operating pressurized water reactor (PWR) can be degraded by such corrosion phenomena as wastage, pitting, intergranular attack, stress corrosion cracking, and crevice corrosion, as well as other phenomena such as denting and vibration wear. Tubes that become excessively degraded reduce the integrity of the primary-to-secondary pressure boundary and must be removed from service or repaired. Steam generator tubes have degraded in various locations, including the top of the tubesheet, either within the section of tube adjacent to the tubesheet or in the tube adjacent to the tube support plates. Tube degradation at tube support plate intersections has generally been confined to the thickness of the tube support plate. Degradation is more prevalent in Inconel Alloy 600 mill annealed (MA) tubing than the same tubing which has been given an additional thermal treatment (TT). Installation criteria for sleeving require that the tube-to-sleeve joint be located at a specified minimum distance from the degraded area.

Historically, SG tubes that have degraded below a calculated minimum wall-thickness value, termed the "plugging limit," were plugged at both the inlet and outlet ends of the tube. Installing plugs in SG tubes decreases the heat transfer surface area available for reactor core cooling. Alternatively, SG tubes experiencing localized degradation can be fitted with sleeves that cover the degraded area to reestablish the integrity of the primary-to-secondary pressure boundary. The sleeves are expanded, sealed inside the tubes, heat treated, and inspected to provide an acceptable leak-resistant load-carrying path. Installing sleeves inside the SG tubes extends the useful life of the tubes, and only slightly reduces the heat transfer capability and primary flow through the sleeved tubes, as compared to unsleeved tubes.

Westinghouse has installed 914 laser-welded sleeves in the 7/8-inch diameter tubes of recirculation SGs in nuclear power plants; 54 were installed in Europe and 860 were installed in the United States. All laser-welded sleeves were given a postweld heat treatment (PWHT). The sleeves are of three basic designs: full-length tubesheet sleeve (FLTS), elevated tubesheet sleeve (ETS), and tube support plate sleeve (TSPS). The FLTS spans from the end of the tube, at the bottom surface of the tubesheet, to a point above the secondary-side surface of the tubesheet. The ETS spans from a non-degraded location within the tube, approximately 14 inches up from the bottom surface of the tubesheet, to a point above the secondary-side surface of the tubesheet. The TSPS is centered approximately on the tube support plate intersection with the tube.

Tubesheet sleeves are secured by first hydraulically expanding the upper and lower portions of the sleeve. The hydraulic expansion brings the sleeve into contact with the parent tube to optimize weld performance and minimize tube deformation. A continuous circumferential laser weld is applied in the area of the hydraulically expanded region of the upper joint and stress relieved with a PWHT. This weld structurally supports the sleeve and at the same time forms a seal. At the lower hydraulically expanded joint, an additional mechanical roll expansion is performed on the FLTS and ETS. The lower joint is a hybrid expansion joint (HEJ) which provides structural integrity under all plant conditions. Because it is a mechanical seal, the HEJ has not been considered leak-tight, although the test data for Alloy 690 TT HEJ indicates that the joint will be leak-tight at operating and accident temperatures and pressures. An optional continuous circumferential laser weld without PWHT may be applied to the HEJ to provide additional leak-tightness. The HEJ has been previously reviewed by the staff and is implemented in the majority of the 26,000 sleeves Westinghouse has installed in the field.

The TSPS is first hydraulically expanded in place at the upper and lower joint areas. Within the expanded areas, a continuous circumferential laser weld is applied. The laser uses a system that delivers a laser beam from outside the containment to the SG by means of fiber optic cable. For welds in the freespan areas, the licensee has committed to apply a PWHT for stress relief. The weld provides the leakage integrity and satisfies the structural requirements with regard to the structural integrity acceptance criteria which include inherent safety factors within the ASME Code criteria.

A structural analysis of the sleeve and sleeve joints using bounding temperature and pressure differences for the Byron and Braidwood SGs was performed. Corrosion testing, mechanical testing and leak testing of prototypic sleeve specimens was also performed. These evaluations and test programs are summarized in the referenced documents.

2.1.1 MATERIAL

The tubes in Byron and Braidwood (Unit 1) are made from ASME SB-163, Alloy 600 mill-annealed (MA) material. The tubes in Byron and Braidwood (Unit 2) are made from ASME SB-163, Alloy 600 TT material. The outside diameter of the tube is 3/4-inch. The sleeves designed for the 3/4-inch tubes are made from ASME SB-163, Alloy 690 TT material.

2.2 LICENSING BASIS

In accordance with 10 CFR 50.55a, the NRC requires components that are part of the primary pressure boundary to be built to the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III (Code). In Section 5.2.1.1 of the Standard Review Plan, "Compliance with the Codes and Standards Rules, 10 CFR 50.55a," the staff has outlined the standards it uses for the evaluation. Any modification, repair, or replacement of these components must also meet the requirements of Section XI of the ASME Code (1989 Edition), Section III (1989 Edition), Section IX (the latest edition) and be reconciled with the design Code of record. The design of the sleeves is predicated on the requirements of Section III, Sub-article NB-3200, for design analysis, and Sub-article NB-3300 for wall-thickness calculations. The ASME Code provides criteria for evaluating stress levels in tubes for design, normal operating, and postulated accident conditions. The margin of safety is provided, in part, by the inherent safety factors in the criteria and requirements of the ASME Code.

Section IX of the ASME Code, Subsection QW, and Section III, including Code Case N-395, "Laser Welding," define the applicable essential variables for the welding procedure specification and welding procedure qualification test. Section XI, Paragraph IWB-4334, of the ASME Code defines the extent of examination requirements for installation of laser welded sleeves.

Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," addresses tubes with defects. The criteria of RG 1.121 are extended to the laser welded sleeve in order to determine the level of degradation at which the sleeve must be removed from service by plugging. ASME Code design-allowable strength values were used for evaluating the plugging limit. By utilizing the requirements contained in the ASME Code and Regulatory Guide 1.121 to define the acceptance criteria for sleeve design, the sleeve will meet the requirements of General Design Criterion (GDC) 15, "Reactor Coolant System Design," and GDC 31, "Fracture Prevention of Reactor Coolant Pressure Boundary."

Regulatory Guide 1.83, "Inspection of Pressurizer Water Reactor Steam Generator Tubes," and the appropriate Byron and Braidwood TS form the basis for determining the inservice inspection requirements of the sleeve.

Total plant allowable primary-to-secondary leakage rates, derived from the requirements of 10 CFR Part 100, are determined on a plant-specific basis. Offsite doses during either a main steamline break or tube rupture event are not to exceed a small fraction of the 10 CFR Part 100 limits and are based on the appropriate Byron and Braidwood TSs.

The effects of sleeving, as evaluated by the licensee, do not increase the dose consequence estimates established by the utility specific Updated Final Safety Analysis Report (UFSAR). Test results indicate that sleeving will not contribute to offsite dose consequences for a postulated steam line break event. The consequences of a steam generator tube rupture are unaffected by tube sleeving.

2.3 WESTINGHOUSE, WCAP-13698 REVISION 1, "LASER WELDED SLEEVES FOR 3/4 INCH DIAMETER TUBE FEEDING-TYPE AND WESTINGHOUSE PREHEATER STEAM GENERATORS" FOR BRAIDWOOD STATION (UNITS 1 AND 2) AND BYRON STATION (UNITS 1 AND 2) REVIEW

2.3.1 SCOPE OF REVIEW

WCAP-13698, Revision 1, is a generic sleeving report. The staff reviewed the data pertaining to Westinghouse Models D4 and D5 SGs and the specific design and installation parameters at Byron and Braidwood. Byron Unit 1 and Braidwood Unit 1 use the Westinghouse Model D4 SG with full-depth hardroll-expanded tubes installed in the tubesheet, which is necessary for the installation of ETS. Byron Unit 2 and Braidwood Unit 2 use the Westinghouse Model D5 SG with tubes hydraulically expanded to the tubesheet. These joints were heat treated during initial assembly. Since ETS will not be used in the Model D5 SG, the staff did not review the installation of ETS into hydraulically expanded tubes.

2.3.2 SLEEVE DESIGN AND ANALYSIS

The laser welded sleeves (both tubesheet and tube support plate sleeves) have been analyzed to maintain steam generator tube structural and leakage integrity during all plant conditions. The function of the sleeve is to restore the integrity of the pressure boundary in the region between the sleeve joints, to a level which is consistent with the original tube. The sleeve joint design is qualified through laboratory testing and analysis. Analytical verification has been performed using design and operating transient parameters selected to envelop loads imposed during normal operation, upset conditions, and accident conditions for Byron and Braidwood. Fatigue and stress analyses of the sleeved tube assemblies have been completed in accordance with the requirements of ASME Code, Section III.

The analyses include a primary stress intensity evaluation, primary plus secondary stress intensity range evaluation, and a fatigue evaluation for mechanical and thermal conditions which encompass the loading conditions for Byron and Braidwood SGs. In addition, the load cycles used in the fatigue analysis represent 40-year life-cycle conditions and, therefore, are considered conservative for operating plants. ASME Code, Sections III, including Code Case N-395, and Section XI, were utilized to address the laser-weld qualification process. The results of the qualification testing, analyses, and plant operating experience demonstrate that the sleeving process with PWHT and an augmented inspection program is an acceptable means of restoring integrity to a steam generator tube.

2.3.3 CORROSION

Corrosion testing was performed in high-temperature, high-pressure autoclaves. A primary-to-secondary pressure differential which bounds the conditions in the Byron and Braidwood SGs was applied, and a doped steam environment was utilized to accelerate crack propagation. The material used was Alloy 600 MA tube-to-Alloy 690 TT sleeve as fabricated with and without PWHT. No corrosion data was presented for Alloy 600 TT tube-to-Alloy 690 TT sleeve joints; however, it is well documented that Alloy 600 MA is more susceptible to PWSCC than the same material in the TT condition. The accelerated corrosion tests indicate that a freespan laser-welded joint is slightly less susceptible to primary water stress-corrosion cracking (PWSCC) than mechanically expanded tube joints at the rolled-to-unrolled transition in the tubesheet. Limited testing shows that the freespan laser-welded joint can be made more resistant to corrosion by subjecting the joint to a PWHT. A laser-produced seal weld of the tubesheet sleeve lower joint (if used) could be performed according to identical parameters used for the upper laser-welded joint, except that heat sink characteristics of the tubesheet would render a PWHT ineffective. Test results of the lower laser-welded joint show slightly less susceptibility to corrosion than the mechanically expanded tube joint.

During a telephone call with the licensee on December 23, 1993, the staff requested that additional information be provided, showing the effectiveness of PWHT on the tubes used in Byron and Braidwood SG. In a January 19, 1994, submittal, the licensee presented data showing a decrease in the degree of PWSCC cracking in U-bends with increasing PWHT soak times and temperatures. The data showed that U-bends manufactured from heats with a history of PWSCC would be less susceptible to cracking after a PWHT. The data also showed that 2 simulated sleeve-to-tube conduction-limited laser-welded and PWHT (5-minute soak time at 1400°F) joints were subjected to steam environment for 1000 hours resulting in a shallow stress corrosion crack in one of the joints. The data supports a position that a PWHT of Alloy 600 MA tubes manufactured from a given heat, improves the crack resistance of that heat. The data did not describe the effects microstructure, chemistry, and joint crevices have on PWSCC. The data focused only on limited accelerated corrosion results, which may not bound all the Alloy 600 MA or TT material conditions that may exist in the Byron and Braidwood steam generators. In the absence of an explanation or supporting test data that relates manufacturing processing, chemical analysis, yield strength, and material microstructure to PWSCC susceptibility in SG tubes, the NRC staff will require that a 1400°F minimum soak temperature with a 5-minute minimum soak time PWHT be used on freespan laser-welded joints.

The PWHT of MA tubes provides stress relief and promotes carbide formation, thus enhancing the tube's ability to withstand PWSCC; however, the joint between the sleeve and tubing provides a crevice. Tests show that degradation occurs in the crevice and is located several mils from the laser-welded joint and extends into the tube. The staff believes that additional corrosion testing is needed to establish the design life for the sleeved tubes in the presence of a crevice. The testing should determine the effects that the conditions identified in the above paragraph have on PWSCC initiation and growth, and should include the associated stress intensity values.

2.4 B&W, BAW-2045PA REVISION 1, "RECIRCULATING STEAM GENERATORS KINETIC SLEEVE QUALIFICATION FOR 3/4 INCH TUBES" FOR BRAIDWOOD STATION (UNITS 1 AND 2) AND BYRON STATION (UNITS 1 AND 2) REVIEW

2.4.1 BACKGROUND

The BWNT process uses a kinetic welding technique to secure the sleeves inside SG tubes. Technical justification supporting the process is given in B&W Topical Report BAW-2045PA, Revision 1, "Recirculating Steam Generator Kinetic Sleeve Qualification for 3/4-Inch OD Tubes." The staff has approved the Topical Report for referencing.

Previously, the staff had approved the use of the BWNT kinetic sleeving process as an alternative to plugging tubes in a number of nuclear plants based on its review and approval for referencing of the topical report. The details of the sleeving process are described in B&W topical report BAW-2045PA, Rev. 1, which the staff approved on June 18, 1992, as being suitable for referencing in licensing documents. The staff's topical report approval letter stated that the staff did not intend to repeat the review of the matters described in the report and found acceptable when the report appears as a reference in license applications, except to assure that the material presented is applicable to the specific plant involved and further we stated that our acceptance applied only to the matters described in the report.

2.4.2 BASIS FOR ADDITIONAL REVIEW

Recent plant operating experience has caused the staff to re-open its review of the kinetic weld sleeving process, and to impose certain new conditions on its use. In the kinetic welding process used by BWNS, an explosive charge expands the sleeve and fuses the sleeve to the tube. The process leaves residual stresses in the parent tube in the vicinity of the explosive weld and must be followed by a post weldheat treatment (PWHT) in order to relieve the residual stresses in the parent tube.

As described in NRC Information Notice 94-05, "Potential Failure of Steam Generator Tubes With Kinetically Welded Sleeves," a leak from an Alloy 600 MA parent steam generator tube with a BWNS kinetically welded Alloy 690 TT sleeve was the cause of the August 22, 1993, McGuire, Unit 1 shutdown. After the tube and sleeve were removed it was confirmed that the leak was from a circumferential crack just above the kinetic weld in the parent Alloy 600 MA tube. Destructive examination of the sleeved tube at McGuire indicated that the most probable root cause of the parent tube leak was the inherent susceptibility of the parent tube to stress corrosion cracking in combination with a stress relief temperature at the low end of the qualified PWHT range.

A kinetically sleeved Alloy 600 MA steam generator tube cracked and leaked at the Trojan Nuclear Plant in 1992 as a consequence of a process error that caused the complete omission a postweld stress relief.

On January 23, 1994, McGuire Unit 1 shut down again because of a 104-gallon per day (gpd) SG tube leak. Eddy current testing with a cross wound coil indicated that this leak was from a circumferential crack in the parent tube located just above the weld joint between the tube and a BWNS kinetically welded sleeve that had been installed in 1991. The circumferential character of the crack appears to be similar to the previous 170 gpd tube leak that occurred in the Unit 1 "A" SG on August 21, 1993.

The BWNS kinetic sleeve process was originally qualified for a range of Alloy 600 materials whose corrosion susceptibilities were believed at the time to include the most limiting case material in operating plants. With this latest information BWNS concluded that the tube from McGuire Unit 1 was more susceptible to corrosion than the tubes used in the original qualification testing. BWNS therefore requalified the PWHT at a higher temperature which is intended to remove virtually all of the residual stress induced by the kinetic welding.

The licensee presented microhardness data on kinetic welded tubes with and without PWHT in BWNT Document #51-1228682-00, "Evaluation of BWNT's Kinetic Sleeving Process" and BWNT Document #51-1228708-00, "Evaluation of Stress Relief Temperature for Kinetic Sleeves." The licensee states that these data indicate that the new PWHT process would remove virtually all of the residual stress from the explosive deformation and weld resulting from the kinetic welding process.

Because the kinetics of the recovery and re-crystallization of explosively deformed metals is more rapid than conventionally deformed metals, it was stated that microhardness measurements are a dependable indicator of the reduction in residual stresses. The staff has not been able to pursue the theoretical basis for this argument nor has it been able to obtain the original references containing the experimental verification of this thesis.

In the qualification of the new stress relief process, a correlation is made between service experience of in-situ stress relieved steam generator tube U-bends and tests conducted in what is believed to be an accelerated caustic environment, corrosion test environment. The staff believes that additional justification is needed to demonstrate the validity of these tests to the qualification of the sleeves for 40 years of steam generator service.

The licensee and BWNT have not provided sufficient long term corrosion testing to bound all heats of the Alloy 600 MA and TT tubing, to establish the suitability of creviced Alloy 600 MA and TT in service, to establish the process parameters of the PWHT, or to establish the minimum stress intensity for crack initiation under extended service conditions.

Therefore, as with the laser welded sleeves, additional corrosion testing is needed to establish the design life of kinetically welded sleeve assemblies, using samples containing crevice conditions and microstructure such as those that exist in installed sleeves. In addition, the NRC staff will require a

1400°F minimum soak temperature with a 5-minute minimum soak time PWHT be used on freespan kinetically welded joints.

2.5 NDE INSPECTABILITY

NDE of the laser welded sleeves is utilized to confirm the adequacy of the sleeve installation and determine if the sleeves have degraded in service. Ultrasonic acceptance and eddy current inspections are used to determine acceptability, to establish baseline inspection, and for subsequent inservice inspections. The sleeved tube can be monitored in accordance with the criteria in RG 1.83 (Revision 1) to determine its suitability for continued service. The inspection of the sleeve necessitates the use of an eddy current bobbin probe, having a diameter that can pass through the inside diameter of the sleeve. The licensee stated that the sensitivity of the eddy current inspections will be 20% of wall thickness based on calibration standards having 0.187 inch diameter flat bottom holes at all locations of the pressure boundary, including those behind the sleeve extension. Changes in the eddy current signature of the sleeve-to-tube joint region would require further inspection by alternate techniques in order to accept the tube for service.

The licensee has committed to validate the adequacy of any system that is used for periodic inservice inspection of the sleeves and, as deemed appropriate, to upgrade testing methods as better methods are developed and validated for commercial use. The new methods can be applied; as long as they can be demonstrated to provide the same or greater degree of inspection accuracy as the methods currently reviewed and approved by the NRC.

In view of the concerns regarding corrosion which are discussed in paragraph 2.3.5 above, the staff requires that the license be amended to reflect an inservice inspection of a minimum of 20 percent of a random sample of the laser-welded sleeves 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 (minimum) of the unsampled sleeves should be inspected, and if an imperfection of 40 percent or greater depth is detected in the second sample, all remaining sleeves should be inspected. The inservice inspection is required until the licensee demonstrates the corrosion resistance for laser-welded joints in tubes that bound the material parameters of the tubes installed at Byron and Braidwood SG. If conformance with the requirements of the plant TS for tube structural integrity is not confirmed, the tubes containing the sleeves in question should be removed from service.

2.6 LEAKAGE

Historically, SG sleeve-to-tube joint were basically mechanical seals and were not considered leak-tight. The NRC has not required that sleeve repairs be leak-tight but only leak-limiting. The staff evaluated leakage based on plant-specific TS requirements for primary-to-secondary leakage limits under normal and accident conditions. The licensee has analyzed the leakage effects of the HEJ, the HEJ with laser-welded joint, the freespan laser-welded joint and kinetically welded joint. The analyses show that even under extreme postulated conditions, the three joints will maintain satisfactory integrity against leakage. The continuous circumferential laser welded joints are inherently leak-tight.

Degraded tubes that were restored to operation as a result of sleeving are susceptible to additional degradation in the same SG environment. The sleeve is designed to extend past the welded joint and into the tubing. In the event that a sleeved tube fails near or at the weld, the sleeve extension will restrict tube movement and leakage. Leakage monitoring devices are intended to alert plant personnel to implement the appropriate procedures. However, based on experience with various causes of leakage through tubes including experience related to tube repaired by sleeving, the staff has concluded that the current primary-to-secondary leakage limits in the Byron and Braidwood TS are not sufficient to detect early stages of sleeve degradation. To reach a satisfactory conclusion regarding the acceptability of the sleeving application, particularly in view of the discussion on corrosion in paragraph 2.3.5 above, the staff requested that the license be amended to reflect a primary-to-secondary leakage limit of 150 gallons per day per steam generator.

2.7 SLEEVE MINIMUM WALL THICKNESS DETERMINATION

The minimum acceptable wall thickness for the sleeve was determined using the criteria of RG 1.121 for partial through-wall degradation, Appendix I of Section III of the ASME Code for allowable material strength values, and Sub-article NB-3300 of Section III of the ASME Code for the pressure stress equation. A bounding set of input conditions, which encompass the operating parameters of Byron and Braidwood SGs, was used for the minimum wall-thickness evaluation. The wall-thickness calculations were performed for faulted, upset, and normal operating conditions. The minimum acceptable wall-thickness was determined to be 40 percent of the nominal wall-thickness of the sleeve.

2.8 PLUGGING LIMIT DETERMINATION FOR SLEEVED TUBES

According to RG 1.121 criteria, an allowance for non-destructive examination (NDE) uncertainty and operational growth of existing tube wall degradation within the sleeve must be accounted for in determining a sleeve plugging limit based on NDE. A conservative tube-wall-degradation growth rate of 10 percent through-wall per cycle and an eddy current uncertainty of 10 percent have been assumed for determining the TS plugging limit for sleeves.

The sleeve-wall degradation based on the bounding conditions, determined by eddy current examination, that would require plugging sleeved tubes is defined to be 40 percent. The plugging limit is determined by subtracting the eddy current uncertainty and assumed crack growth rate for an additional cycle from the through-wall penetration of degradation, which corresponds to the minimum acceptable wall thickness. Removing the sleeved tubes from service when degradation reaches the plugging limit provides assurance that the minimum acceptable wall thickness will not be exceeded during the next cycle of operation. In the event that degradation is determined by another technique, other than eddy current examination, the uncertainty of that technique would have to be established.

3.0 SUMMARY

On the basis of its analysis, the staff concludes that for Braidwood (Units 1 and 2) and Byron (Units 1 and 2) the repair of SG tubes using Westinghouse conduction-limited laser-welded sleeves and B&W kinetically-welded sleeves in accordance with the proposed amendment, with minimum soak time and temperature PWHT and inservice inspection described in this SE is acceptable. However, the staff finds that additional confirmatory testing is necessary to establish the design life of the sleeved tubes and to confirm that the inspection and leak detection requirements are sufficient. The inservice inspections and primary-to-secondary leakage limit requirements discussed previously must be in place to adequately monitor SG sleeve performance. Also, corrosion testing to confirm long term suitability for service of these sleeve assemblies must be completed.

By letter dated February 24, 1994, CECo committed to submit a license amendment request to include the following conditions in Byron and Braidwood station licenses:

1. Amend the license to reflect a primary-to-secondary leakage limit of 150 gallons per day.

2. Amend the license to reflect an inservice inspection as described in section 2.8 of this SE.

3. Add a condition to the license to conduct additional corrosion testing as described in sections 2.3.3 and 2.4.2 of this SE.

The licensee committed to submit the above amendment requests no later than 90 days from the date of issuance of this amendment. In addition, the licensee committed to:

> 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 NRC staff finds licensee's commitment to implement the conditions stated in this SE acceptable.

4.0 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.

5.0 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 effluent 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 (58 FR 57846). 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.

6.0 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.

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