ATTACHMENT B

MARKED UP PAGES FOR

PROPOSED CHANGES TO APPENDIX A

TECHNICAL SPECIFICATIONS FOR

FACILITY OPERATING LICENSES

NPF-37 AND NPF-66

NPF-72 AND NPF-77

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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:

INSERT A

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~~.

INSERT B

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.

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.~~ INSERT C

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.

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:

Insert A

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.~~

Insert B

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

11) Tube Support Plate Interim Criteria Limit is used in Unit 1 for the disposition of a steam generator tube for continued service that is experiencing ODSCC confined within the thickness of the tube support plates. For application of the tube support plate interim plugging criteria limit, the tube's disposition for continued service will be based upon standard bobbin coil probe signal amplitude of flaw-like indications. Pending incorporation of the voltage verification requirements in ASME standard verifications, an ASME standard calibrated against the laboratory standard will be utilized in Unit 1 steam generator inspections for consistent voltage normalization.*

1. A tube can remain in service with a flaw-like bobbin coil signal amplitude of less than or equal to 1.0 volt, regardless of the depth of the tube wall penetration, provided Item 3 below is satisfied.

2. A tube can remain in service with a flaw-like bobbin coil signal amplitude greater than 1.0 volt but less than or equal to 2.7 volts provided an RPC inspection does not detect degradation and provided Item 3 below is satisfied.

*The plant specific guidelines used for all inspections shall be consistent with the eddy current guidelines in Appendix A of WCAP-13854.

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. 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.~~ Insert C

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.

INSERT A

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

INSERT B

in a Babcock & Wilcox Nuclear Technologies Technical Report currently approved by the NRC, subject to the limitations and restrictions as noted by the NRC staff.

INSERT C

the current Westinghouse or Babcock & Wilcox Nuclear Technologies Technical Reports.

ATTACHMENT C

EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATIONS
FOR PROPOSED CHANGES TO
APPENDIX A TECHNICAL SPECIFICATIONS OF
FACILITY OPERATING LICENSE
NPF-37, NPF-66, NPF-72, AND NPF-77

Commonwealth Edison (ComEd) has evaluated this proposed amendment and determined that it involves no significant hazards considerations. According to Title 10 Code of Federal Regulations Part 50 Section 92 Sub-Section c (10 CFR 50.92 (c)), a proposed amendment to an operating license involves no significant hazards considerations if operation of the facility in accordance with the proposed amendment would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated; or

2. Create the possibility of a new or different kind of accident from any accident previously evaluated; or

3. Involve a significant reduction in a margin of safety.

A. INTRODUCTION

Byron and Braidwood requested approval of two Steam Generator (SG) sleeving processes in a previous amendment submitted to the Nuclear Regulatory Commission (NRC) in August of 1993. The NRC issued a Safety Evaluation Report authorizing the use of both sleeving processes. Commonwealth Edison requests an editorial change in the approved Technical Specifications to remove references to a specific vendor Technical Report revision. Removing references to a specific vendor Technical Report revision will allow the use of any new installation process made to the approved sleeving techniques without requiring a TS amendment. The new process must first be accepted by the NRC to be authorized for use at Byron and Braidwood stations

B. 10 CFR 50.92 ANALYSIS

1. **The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.**

The proposed changes to the Steam Generator section of Technical Specifications do not affect any accident initiators or precursors and do not alter the design assumptions for the systems or components used to mitigate the consequences of an accident. These changes are editorial changes to the requirements currently identified in the Technical Specifications. The requirements approved by the NRC will not be reduced by this request. The proposed change maintains the administrative controls necessary to ensure safe plant operation.

The original amendment requested tubesheet sleeves and tube support plate sleeves as an alternate tube repair method for Byron and Braidwood Units 1 and 2. The steam generator sleeves approved for installation use the Westinghouse process (laser welded joints) or the Babcock & Wilcox Nuclear Technologies (BWNT) process of kinetically welded joints. The sleeve configuration was designed and analyzed in accordance with the criteria of Regulatory Guide (RG) 1.121 and the design requirements of Section III of the American Society of Mechanical Engineers (ASME) Code. Fatigue and stress analyses of the sleeved tube assemblies for both processes produced acceptable results documented in the current Westinghouse and the BWNT Technical Reports. The proposed Technical Specifications change to allow the use of the current NRC approved laser welded or kinetically welded sleeving process does not adversely impact any other previously evaluated design basis accident or the results of these analyses. Therefore, the editorial changes to the referenced sleeving Technical Reports will not increase the probability of occurrence of an accident previously evaluated.

2. **The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.**

The proposed changes are considered to be administrative changes. All the requirements described in Technical Specifications "Acceptance Criteria" for the Steam Generators will continue to be implemented as described in the current Technical Reports.

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Referencing the current Westinghouse or BWNT Sleeving Technical Reports currently approved by the NRC and subject to the limitations and restrictions as noted by the NRC, has no effect upon any design transient or accident analyses. The proposed changes do not affect the design or operation of any system, structure, or component in the plant. There are no changes to parameters governing plant operation and no new or different type of equipment will be installed.

The use of the proposed sleeving processes will not introduce significant or adverse changes to the plant design basis. Stress and fatigue analyses of the repair have shown the ASME Code and RG 1.121 allowable values are met. Implementation of the currently approved laser welded or kinetically welded sleeving will continue to maintain the overall tube bundle structural integrity at a level consistent with that of the originally supplied tubing. Repair of a tube with a sleeve does not provide a mechanism which would result in an accident outside of the area affected by the sleeve. Any hypothetical accident as a result of potential tube or sleeve degradation in the repaired portion of the tube is bounded by the existing steam generator tube rupture accident analysis. The tube rupture accident analysis accounts for the installation of sleeves and the impact on current plugging level analyses. This sleeve design does not affect any other component or location on the tube outside of the immediate area repaired.

Thus, the possibility of a new or different type of accident from any accident previously evaluated is not created.

3. **The proposed change does not involve a significant reduction in a margin of safety.**

The proposed change is administrative in nature and has no impact on the margin of safety of any Technical Specification. Specific technical reports are no longer referenced in Technical Specifications. An editorial change is made to TS referencing the current NRC approved vendor Technical Report, subject to the limitations and restrictions noted by the NRC. The initial conditions and methodologies used in the accident analyses remain unchanged.

The laser welded and kinetically welded sleeving repair of degraded steam generator tubes has been shown by analysis to restore the integrity of the tube bundle to its original design basis condition. The safety factors used in the design of sleeves for the repair of degraded tubes are consistent with the safety factors in the ASME Boiler and Pressure Vessel Code used in steam generator design. The design of the tube sleeves has been verified by testing to preclude leakage during normal and postulated accident conditions. Installation of either type of vendor sleeve using the current approved process will continue to maintain the structural integrity of the steam generators tubes.

Thus, these changes do not involve a significant reduction in the margin of safety.

Therefore, based on the above evaluation, Commonwealth Edison has concluded that these changes do not involve any significant hazards considerations.

ATTACHMENT D

ENVIRONMENTAL ASSESSMENT FOR

PROPOSED CHANGES TO

APPENDIX A TECHNICAL SPECIFICATIONS OF

FACILITY OPERATING LICENSES

NPF-37, NPF-66, NPF-72, AND NPF-77

Commonwealth Edison has evaluated the proposed amendment against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10CFR51.21. It has been determined that the proposed change meets the criteria for a categorical exclusion as provided for under 10CFR51.22(c)(9).

This determination is based on the fact that the proposed change does not involve a significant hazards consideration as discussed in Attachment C to this letter, will not involve significant changes in the types or amounts of any radioactive effluents, does not affect any of the permitted release paths, and does not involve a significant increase in individual or cumulative occupational exposure. Therefore, this change meets the categorical exclusion requirements permitted by 10CFR51.22(c)(9).