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. 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:

Insert B Here

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:

Insert A Here

9308180226 930813
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P PDR

SURVEILLANCE REQUIREMENTS (Continued)

Insert C Here

~~1) All nonplugged tubes that previously had detectable wall penetrations (greater than 20% of wall thickness),~~

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

Insert D Here

~~3)~~ 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.

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 ~~a~~ an unrepaired defect is defective;

Insert E Here

~~6) Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to 40% of the nominal tube 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~~; and~~

Insert F Here

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.

Insert G Here

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) ~~and all tubes containing through-wall cracks)~~ 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 minimum 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. Insert H

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. Insert I Here ~~Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness~~.

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.

REACTOR COOLANT SYSTEM

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. 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:

Insert B Here

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:

Insert A Here

SURVEILLANCE REQUIREMENTS (Continued)

Insert C Here

~~1) All nonplugged tubes that previously had detectable wall penetrations (greater than 20% of wall thickness),~~

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

Insert D Here

~~3)~~ 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.

SURVEILLANCE REQUIREMENTS (Continued)

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;

Insert E Here

~~6) Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to 40% of the nominal tube 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; ~~and~~

Insert F Here

Insert G Here

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.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug, all tubes exceeding the plugging or repair limit) ~~and all tubes containing through-wall cracks~~) or repair in the affected area 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 minimum 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. Insert H

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. ~~Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.~~ Insert I Here

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.

INSERT A

* 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.

INSERT B

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.

INSERT C

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,

INSERT D

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

INSERT E

6) <u>Plugging or Repair Limit</u> 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.

INSERT F

For a tube that has been repaired by sleeving, the tube inspection shall include the sleeved portion of the tube, and

INSERT G

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.

INSERT H

The technical bases for sleeving are described in Westinghouse report WCAP-13698 Rev. 1 and Babcock & Wilcox Topical Report BAW-2045PA Rev. 1.

INSERT I

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.

ATTACHMENT C

10CFR50.92 SIGNIFICANT HAZARDS CONSIDERATION

The amendment does not involve a significant hazards consideration 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.

Commonwealth Edison has reviewed the attached significant hazards considerations written by Westinghouse and B&W and finds them acceptable. Commonwealth Edison agrees with their conclusion that this amendment does not contain a significant hazards consideration.

Please note that a separate Byron and Braidwood significant hazards consideration analysis was supplied by B&W, however, both documents are virtually identical.

CECo would like to clarify one item in the Westinghouse significant hazards consideration document. On page 4 of 10, Westinghouse states: "Primary to secondary leakage through non-welded tubesheet sleeve lower joints would not be expected at 0% power (T_{hot} = 557°F)." It should be clearly noted that primary to secondary leakage is not expected at any power level based on the test results documented in WCAP-13698, Revision 1 and in Reference 5.3 of the subject significant hazards consideration (NSD-JLH-3267, "Confirmatory Leak Testing - 3/4" LWS HEJ for Byron/Braidwood," July 23, 1993).

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JUL 2 9 1993

Westinghouse Electric Corporation

Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

CAE-93-193
CCE-93-212
ET-NSL-OPL-I-93-390
July 26, 1993

Mr. D. L. Shamblin
Commonwealth Edison Company
1400 Opus Place, Suite 500
Downers Grove, IL 60515

Commonwealth Edison Company
Byron/Braidwood Stations
Laser Welded Sleeving - Revised Significant Hazards Consideration Analysis

Reference 1: Westinghouse letter CAE-93-171/CCE-93-192, "Laser Welded Sleeving - Licensing Documents, dated 6/9/93.

Dear Mr. Shamblin:

Enclosed please find Significant Hazards Consideration Analysis, 93-112 SHC Rev. 2, performed in accordance with 10CFR50.91 (a)(1) and 10CFR50.92 (c), to demonstrate that a proposed license amendment to implement repair of tubes using laser welded tube sleeves represents no significant hazards consideration. Changes from the Rev. 1 version (transmitted via Reference 1) are identified in the margins.

If you have any questions, do not hesitate to call.

Very truly yours,

B S Humphries

B. S. Humphries, Manager
Commonwealth Edison Projects
Customer Projects Department

B. E. Rarig/cld
Enclosures

cc: J. C. Blomgren
J. F. Reiss
P. M. McHale
S. E. Bishop
D. E. St. Clair
G. R. Masters
F. G. Lentine
L. A. Petrie

CLD850:BER:072693

93-112 SHC Rev. 2
Customer Reference No(s).

Westinghouse Reference No(s).

WESTINGHOUSE NUCLEAR SAFETY
SIGNIFICANT HAZARDS CONSIDERATION (SHC)

1) NUCLEAR PLANT(S) Byron and Braidwood Units 1 and 2

2) SUBJECT Laser Welded Sleeving

3) TECHNICAL SPECIFICATIONS CHANGED 3/4.4.5 "Steam Generators"

4) A written analysis of the significant hazards consideration, in accordance with the three factor test of 10CFR50.92, of a proposed license amendment to implement the subject change has been prepared and is attached. On the basis of the analysis the checklist below has been completed.

Will operation of the plant in accordance with the proposed amendment:

4.1) Yes__ No X Involve a significant increase in the probability or consequences of an accident previously evaluated;

4.2) Yes__ No X Create the possibility of a new or different kind of accident from any accident previously evaluated;

4.3) Yes__ No X Involve a significant reduction in a margin of safety.

5) REFERENCE DOCUMENTS:

5.1) WCAP-13698 Rev. 1, "Laser Welded Sleeves for 3/4" Diameter Tube Feedring-Type and Westinghouse Preheater Steam Generators," May 1993

5.2) Westinghouse Safety Evaluation Checklist, SECL-93-112, "Laser Welded Sleeving," May 1993

5.3) NSD-JLH-3267, "Confirmatory Leak Testing - 3/4" LWS HEJ for Byron/Braidwood," July 23, 1993

6) Comments: None

7) SAFETY EVALUATION APPROVAL LADDER:

7.1) Prepared by (Nuclear Safety): RSLapides Date: 7-23-93
R. S. Lapides

7.2) Reviewed by (Nuclear Safety): G.W. White Date: 7/23/93
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LASER WELDED SLEEVING
BYRON/BRAIDWOOD UNITS 1 AND 2
SIGNIFICANT HAZARDS CONSIDERATION ANALYSIS

1.0 INTRODUCTION

A license amendment is proposed to permit the installation of Alloy 690 laser welded tube sleeves at degraded tube support plate intersections and within the tubesheet area of the steam generators at Byron and Braidwood, Units 1 and 2. Per the current Technical Specifications, steam generator tubes with eddy current indications of 40% through wall or greater must be removed from service. Laser welded tube sleeves can be installed to repair degraded steam generator tubes either at the tube support plate intersections, within the tubesheet area, or a combination of both within the same tube. Laser welded sleeving has been determined to be an effective method of degraded steam generator tube repair and has been successfully implemented at plants with 7/8" outside diameter tubes.

2.0 DESCRIPTION OF THE AMENDMENT REQUEST

As required by 10 CFR 50.91 (a)(1), this analysis is provided to demonstrate that a proposed license amendment to implement repair of tubes using laser welded tube sleeves for the steam generators at Byron and Braidwood, Units 1 and 2, represents no significant hazards consideration. In accordance with 10 CFR 50 92(c), implementation of the proposed license amendment was analyzed using the following standards and found not to: 1) involve a significant increase in the probability or consequences for an accident previously evaluated; 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 repair method has been developed which secures to the original tube a short length of tubing with an outer diameter slightly smaller than the inside diameter of the tube, spanning the degraded area of the parent tube. The tube support plate sleeve is attached to the degraded tube by producing an autogenous weld between the original tube and sleeve. Tube support plate sleeve welds are produced in the free span sections of the tube. The free span welds provide the structural joint between the tube and sleeve and also provide positive (leaktight) leakage integrity. The tubesheet sleeve is secured and supported structurally at the upper section by a free span autogenous weld performed identically to the tube support plate sleeve welds while the lower joint is secured by a hybrid expansion joint (HEJ). A seal weld can also be included within the tubesheet sleeve lower joint at an elevation coincident with the approximate midpoint of the tubesheet cladding. However, the HEJ supplies the necessary structural requirements for the lower joint. Both the lower HEJ and free span laser weld joints (LWJ) separately have strength exceeding the structural requirements for the sleeve. Therefore, it can be postulated that a loss of structural integrity in one of the sleeve joints will not result in a loss of structural integrity for the sleeve. The sleeve structural integrity requirements include safety factors inherent to the requirements of the ASME Code. Installation of tube support plate sleeves and/or tubesheet sleeves restores the integrity of the primary pressure boundary to a condition consistent with that of the originally supplied tubing. All welds must be produced a minimum distance from any detected tube degradation as described in Reference 1. The structural analysis and mechanical performance of the sleeves are based on installation in the hot leg of the steam generators. The tube support plate sleeves have been qualified to be installed in the second from highest support plate elevations and also include the flow distribution baffle.

Tubes with indications of degradation in excess of the plugging criteria would have to be removed from service, according to Technical Specification tube plugging criteria without provision for tube repair by sleeving. Removal of a tube from service results in a reduction of reactor coolant flow through the steam generator. This small reduction in flow has an impact on the margin in the reactor coolant flow through the steam generator in LOCA analyses and on the heat transfer efficiency of the steam generator. Repair of a tube with sleeving maintains the tube in service and results in a much smaller flow reduction. Therefore, the use of sleeving in lieu of plugging minimizes loss of margin in reactor coolant system flow and assist in assuring that minimum flow rates are maintained in excess of that required for operation at full power. Any combination of sleeving and plugging utilized at Byron and Braidwood, Units 1 and 2, up to a level such that the effect of sleeving will not reduce the minimum reactor coolant flow rate to below the current Technical Specification limit is acceptable. Also, minimizing the reduction in flow has operational benefits by limiting the increase in heat flux across the tubes remaining in service. Increased heat fluxes have been associated with an increased potential for tube corrosion.

The proposed amendment would modify Technical Specifications 3/4.4.5 "Steam Generators", and Bases B 3/4.4.5, "Steam Generators", to provide the sleeve/tube inspection requirements and acceptance criteria to determine the level of degradation which would require the sleeve to be removed from service.

3.0 EVALUATION

3.1 Generic Structural Assessment

During the development of laser welded sleeving, Section III of the ASME Code was used for the bounding stress and fatigue levels for the sleeve. By showing that the sleeve design meets all facets of the applicable subsections of Section III of the Code, the sleeve design meets the design requirements of the original tubing. Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes", is used to develop the plugging limit of the sleeve determined by NDE, should sleeve wall degradation occur. Potentially degraded sleeves are shown (by analysis) to retain burst strength in excess of three times the normal operating pressure differential at end of cycle conditions.

The structural analysis utilized a generic set of loading inputs which conservatively bound the operating regimes of all plants with Westinghouse Model D and E steam generators. The requirements of Regulatory Guide 1.83, "Inservice Inspection of PWR Steam Generator Tubes", are implemented, and a baseline eddy current inspection of the installed sleeves is performed prior to operation. An ultrasonic inspection of the free span weld joints is also performed prior to operation. The ultrasonic inspection is used to verify that the minimum acceptable fusion zone thickness of the weld is achieved. This minimum weld fusion zone thickness has been shown by analysis to satisfy the requirements of the ASME Code with regard to acceptable stress levels during operating and accident conditions. As stated previously, a generic set of loading conditions was used for structural analysis of the sleeved tube assembly. The values for primary to secondary pressure differential and T_{hot} and T_{cold} represented bounding or design values and are considered conservative for all plants with Model D and E steam generators. In addition, a fatigue analysis was performed for the assembly, the critical location being the free span laser weld. The loading cycles that were applied to the sleeve assembly analysis were those for a 40 year plant life cycle. Therefore, the fatigue analysis is conservative for an operating plant. The results of the fatigue analysis indicate acceptable usage factors for the entire range of permitted weld thicknesses.

Leakage testing under conditions considered to be more severe than expected during all operating plant conditions has shown that the laser welded sleeve does not introduce additional primary to secondary leakage during a postulated steam line break event. Laser welded sleeved tube assemblies were subjected to thermal and fatigue cycling and then leak tested at pressure differences of up to 3110 psi, which far exceeds the expected FLB/SLB pressure differential of a maximum of 2600 psi. No leakage was detected in any welded joint (both free span and tubesheet joints). Leakage testing has also shown that the seal weld of the tubesheet sleeve lower joint is not required in order to preclude leakage during normal operation or accident conditions at 600°F. Non-welded lower joint tubesheet sleeve/tube leakage test specimens were subjected to both fatigue and thermal cycling tests prior to final leak rate evaluation testing. The load level applied during the fatigue testing exceeded the maximum axial load applied to the sleeve during the most severe pressure loading condition. Thermal cycling tests simulated a standard plant heatup/cooldown cycle. No leakage was detected in any non-welded tubesheet sleeve lower joint at 600°F after both thermal and fatigue loading. Primary to secondary leakage through non-welded tubesheet sleeve lower joints would not be expected at 0% power (T_{hot} = 557°F).

3.2 Specific Structural Assessment

In comparing the plant specific loads to the loads used in the generic analysis (Reference 1) for evaluating the sleeve primary stresses, the generic loads are found to umbrella the plant specific values for Byron and Braidwood, Units 1 and 2.

In evaluating the maximum range of stress and fatigue, the number of transients, as well as the temperature and pressure fluctuations are significant. A comparison of the transient cycles considered in the generic analysis to the applicable transients for Byron Units 1 & 2 and Braidwood Units 1 & 2 shows that the generic analysis considers a larger number of transients, and in general, more transient cycles, than are applicable to Byron and Braidwood, Units 1 and 2.

Relative to the temperature and pressure fluctuations, the transient definitions are defined in terms of changes in applicable parameters from an initial starting point, typically normal operation. Comparison of the pressure fluctuations shows the generic analysis and Byron and Braidwood, Units 1 & 2, values to be comparable. Because the generic analysis considers more transients and generally more transient cycles than Byron and Braidwood, Units 1 & 2, the generic analysis is concluded to be applicable to Byron and Braidwood, Units 1 & 2.

3.3 Sleeving of Previously Plugged Indications

The sleeve installation requirements applicable to active tubes which have been identified as containing degradation indications which exceed the repair limit are no different for the sleeving of previously plugged tubes. A new "baseline" inspection of the entire tube length must be performed prior to sleeve installation in a previously plugged tube. The location of the identified tube degradation indication must be verified to be a minimum distance from the weld joints (same for active tubes), as defined in Reference 1. Historically, the areas of the tube which have suffered corrosion degradation indications are the tube support plate intersections, the expansion transition and the sections of tube within the thickness of the tubesheet where secondary side contaminants have collected in operating crevices. The sleeve free span (structural) weld joints are not located in these areas, and should not be affected by any previously identified degradation mechanism which resulted in the tube's removal from service. The analysis has also supported sleeve installation in a separated tube, therefore, the extent of the originally identified degradation indication should not affect sleeve installation. Additionally, the area of the tube where the tube plug was installed must be visually inspected prior to sleeve installation. Surface finish requirements for this area have been developed which help to maintain the ability of the joint to form a leaktight seal. Conformance to the surface finish requirements for the lower joint will help to ensure a leaktight sleeve joint, regardless of whether or not the seal weld has been produced. The ability of the weld to sufficiently penetrate the tube wall has been shown by test in cases where a localized gap of several mils existed between the tube and sleeve. The penetrating capabilities of the weld will also help to ensure a leaktight joint in cases where slight surface imperfections due to tube plug removal may be present.

Thermally treated Alloy 600 and Alloy 690 sleeved tube assemblies have performed well historically with regard to corrosion. There are no reported instances of sleeve degradation for the greater than 25,000 sleeves that Westinghouse has installed in the U.S. Accelerated corrosion test results show the free span laser welded joint (with post weld heat treatment) is capable of exhibiting a resistance to corrosion of greater than 10 times that of rolled tube transitions. Accelerated corrosion tests also show that non-heat treated laser welded free span joints exhibit resistance to stress corrosion cracking equal to or greater than rolled tube transitions. Corrosion testing of stress relieved laser welded sleeve free span joints exhibit a resistance to corrosion cracking of approximately 10 times that of rolled tube transitions. These factors suggest postulated sleeve degradation would occur at a rate less than rolled transitions, and the potential for a sleeve with accelerated degradation rate characteristics more severe than roll transitions is negligible.

4.0 ANALYSIS

Conformance of the proposed amendments to the standards for a determination of no significant hazard as defined in 10 CFR 50.92 (three factor test) is shown in the following:

4.1 Operation of Byron and Braidwood Units 1 and 2 in accordance with the proposed license amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The tubesheet and/or tube support plate intersection laser welded sleeve configuration has been designed and analyzed in accordance with the requirements of the ASME Code. Fatigue and stress analyses of the sleeved tube assemblies produced acceptable results. Mechanical testing has shown that the structural strength of Alloy 690 sleeves under normal, faulted and upset conditions is within acceptable limits. Leakage testing for 7/8" and 3/4" tube sleeves has demonstrated that primary to secondary leakage is not expected during all plant conditions, including the case where the seal weld is not produced in the lower joint of the tubesheet sleeve.

The sleeve minimum acceptable wall thickness is determined using the criteria of Regulatory Guide 1.121 and the pressure stress equation of Section III of the ASME Code. With respect to the design of the sleeve, the limiting requirement of Regulatory Guide 1.121 which applies to part throughwall degradation is that the minimum acceptable wall must maintain a factor of safety of three against tube failure under normal operating conditions. A bounding set of input conditions which envelop the operating parameters of all Model D and E steam generators was used for the minimum wall thickness evaluation in the generic evaluation. The minimum acceptable tube wall thickness determined by the Regulatory Guide 1.121 evaluation is 35% of the original sleeve wall thickness. Evaluation of the minimum acceptable wall thickness for postulated combined accident condition loadings, per Reg Guide 1.121 recommendations, shows that the minimum wall requirement for SLB/FLB + SSE loadings, as well as the loading conditions during a Loss of Load transient, is bounded by the normal operating condition requirement of 35% minimum wall thickness (Reference 1).

According to Regulatory Guide recommendations, an allowance for non destructive evaluation (NDE) uncertainty and operational growth of existing tube wall degradation indications within the sleeve must be accounted for in determining a sleeve plugging limit based on NDE. While there have been no reported Westinghouse HEJ sleeves which have been plugged due to degradation of the sleeve, a conservative tube wall degradation growth rate of 10% through-wall per cycle and an eddy current uncertainty of 10% has been assumed for determining the sleeve Technical Specification plugging limit.

The sleeve wall degradation extent based on bounding generic conditions, determined by eddy current examination, which would require plugging sleeved tubes is defined to be 45% (100% - (structural limit of 35% + NDE uncertainty of 10% + growth of 10%)). Removal of tubes/sleeves from service when degradation indications reach the plugging limit assures that the minimum acceptable wall thickness will not be exceeded during the next subsequent cycle of operation.

A 44% (depth of penetration) through wall sleeve plugging limit is established for Byron and Braidwood Units 1 and 2. The difference in allowable throughwall extent is due to a Byron/Braidwood plugging analysis that resulted in pressures that fall outside of the envelope of the generic analysis. Therefore, the plugging limit for sleeves is bounded by the current Technical Specification plugging limit for tubes of 40%.

A conservative leak-before-break evaluation has been performed for the sleeved tube assembly, using plant specific values for operating regimes of the Model D4 steam generators at Byron/Braidwood Unit 1 and the D5 steam generators at Byron/Braidwood Unit 2. The evaluation is considered conservative in that no credit for the parent tube is assumed in determining the burst pressure of the sleeved tube assembly. The leak-before-break criteria compares the postulated throughwall crack length which will leak at a specified value at normal operating conditions, thereby permitting adequate leakage detection and safe shutdown of the plant prior to a crack achieving a length equal to the critical crack length which could be postulated to burst at steam line break conditions.

Leak rates for the sleeves are a function of sleeve geometry, material strength properties, and several operating parameters. The operating parameters of significance are the primary and secondary side pressures and the primary side temperature. For the present operating conditions at Byron/Braidwood, the limiting primary-to-secondary leakage rate bounded by the current technical specification limit. Byron Units 1 & 2 and Braidwood Units 1 & 2 presently have a technical specification limit for steam generator tube leakage of 500 gpd. Therefore, for the present operating conditions, the technical specification limit is governing.

Despite the fact that leak-before-break is considered to be applicable to the sleeved tube assembly, historically no primary to secondary leakage or degradation has been reported in Westinghouse sleeves. Furthermore, the hypothetical consequences of failure of the sleeve would be bounded by the current steam generator tube rupture analysis included in the Byron and Braidwood FSAR. Due to the slight reduction in diameter caused by the sleeve wall thickness, it is expected that primary coolant release rates would be slightly less than assumed for the steam generator tube rupture analysis, and therefore, would result in lower total primary fluid mass release to the secondary system. Additionally, further conservatisms would be included if the break were postulated to occur at an elevation higher than where sleeves are installed. Combinations of tubesheet sleeves and tube support plate sleeves would reduce the primary fluid flow through the sleeved tube assembly due to the series of diameter reductions the fluid would have to pass on its way to the break area. The overall effect would be reduced steam generator tube rupture release rates.

The proposed Technical Specification change to support the installation of Alloy 690 laser welded sleeves does not adversely impact any other previously evaluated design basis accident or the results of LOCA and non-LOCA accident analyses for the current Technical Specification minimum reactor coolant system flow rate. The results of the analyses and testing, as well as plant operating experience demonstrate that the sleeve assembly is an acceptable means of maintaining tube integrity. Plugging limit criteria are established using the guidance of Regulatory Guide 1.121. Furthermore, per Regulatory Guide 1.83 recommendations, the sleeved tube can be monitored through periodic inspections with present eddy current techniques. These measures demonstrate that installation of sleeves spanning degraded areas of the tube will restore the tube to a condition consistent with its original design basis.

Conformance of the sleeve design with the applicable sections of the ASME Code and results of the leakage and mechanical tests, support the conclusion that installation of laser welded tube sleeves will not increase the probability or consequences of an accident previously evaluated. Depending upon the break location for a postulated steam generator tube rupture event, implementation of tube sleeving could act to reduce the radiological consequences to the public due to reduced flow rate through a sleeved tube compared tube a non-sleeved tube based on the restriction afforded by the sleeve wall thickness.

4.2 The proposed license amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Implementation of laser welded sleeving will not introduce significant or adverse changes to the plant design basis. Stress and fatigue analysis of the repair has shown the ASME Code and Regulatory Guide 1.121 allowable values are met. Implementation of laser welded sleeving maintains overall tube bundle structural and leakage integrity at a level consistent to that of the originally supplied tubing during all plant conditions. Leak and mechanical testing of sleeves support the conclusions of the calculations that the sleeve retains both structural and leakage integrity during all conditions. Sleeving of tubes does not provide a mechanism resulting in an accident outside of the area affected by the sleeves. Any hypothetical accident as a result of potential tube or sleeve degradation in the repaired portion of the tube is bounded by the existing tube rupture accident analysis. Since the sleeve design does not affect any other component or location of the tube outside of the immediate area repaired, in addition to the fact that the installation of sleeves and the impact on current plugging level analyses is accounted for, the possibility that laser welded sleeving creates a new or different type of accident is not supported.

Implementation of laser welded sleeving will reduce the potential for primary to secondary leakage during a postulated steam line break while maintaining available primary coolant flow area in the event of a LOCA. By removing from service degraded intersections through repair, the potential for steam line break leakage is reduced.

These degraded intersections now are returned to a condition consistent with the Design Basis. While the installation of a sleeve causes a reduction in primary coolant flow, the reduction is far below the reduction incurred by plugging. Therefore, far greater primary coolant flow area is maintained through sleeving.

4.3 The proposed license amendment does not involve a significant reduction in a margin of safety.

The laser welded sleeving repair of degraded steam generator tubes as identified in WCAP-13698, Rev. 1, 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 tubesheet sleeve lower joints for the 3/4" and 7/8" sleeves have been verified by testing to preclude leakage during normal and postulated accident conditions.

The portions of the installed sleeve assembly which represent the reactor coolant pressure boundary can be monitored for the initiation and progression of sleeve/tube wall degradation, thus satisfying the requirements of Regulatory Guide 1.83. The portion of the tube bridged by the sleeve joints is effectively removed from the pressure boundary, and the sleeve then forms the new pressure boundary. The areas of the sleeved tube assembly which require inspection are defined in WCAP-13698, Rev. 1.

In addition, since the installed sleeve represent a portion of the pressure boundary, a baseline inspection of these areas is required prior to operation with sleeves installed.

The effect of sleeving on the design transients and accident analyses have been reviewed based on the installation of sleeves up to the level of steam generator tube plugging coincident with the minimum reactor flow rate. The installation of sleeves is to be evaluated as the equivalent of some level of steam generator tube plugging. Evaluation of the installation of sleeves is based on the determination that LOCA evaluations for the licensed minimum reactor coolant flow bound the effect of a combination of tube plugging and sleeving up to an equivalent of the actual steam generator tube plugging limit. Information provided in WCAP-13698, Rev. 1, describes the method to determine the flow equivalency for all combinations of tubesheet and tube support plate sleeves in order that the minimum flow requirements are met.

5.0 CONCLUSION

Based on the preceding analysis it is concluded that operation of Byron and Braidwood Units 1 and 2 following the installation of Alloy 690 laser welded sleeves at the tube support elevations and within the tubesheet region of the steam generators, in accordance with the proposed amendment does not increase the probability of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, nor reduce any margins to plant safety. Therefore, the license amendment does not involve a Significant Hazards Consideration as defined in 10CFR50.92.

B&W NUCLEAR SERVICE COMPANY

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Reviewer is Independent.

Remarks:

BWNS NON-PROPRIETARY

This document provides an evaluation for the BWNS tube sheet and tube support plate sleeves which concludes, in accordance with 10CFR50.92(c), there are no significant hazard considerations for this application at the Commonwealth Edison Byron Station. Byron is a four loop Westinghouse NSSS with Model D4 and D5 steam generators for Unit 1 and 2, respectively.

Tubesheet (TS) and tube support plate (TSP) sleeves are a viable method to maintain steam generator (S/G) tubing integrity. The sleeve serves as the pressure boundary and allows the degraded tube to remain in-service for heat transfer and core cooling. S/G tubing and sleeve NDE is performed on a scheduled basis; wall loss or deformation is trended accordingly.

The BWNS TSP and TS sleeves are kinetically welded at both ends. The sleeve design allows for tube plugging or stabilization at a later date should that need arise.

RECORD OF REVISION

REV.NO/NAME	SECTION	DESCRIPTION OF CHANGE
00/RF PENN	ALL	ORIGINAL ISSUE.
01/RF PENN	All Pages	Classification change from Proprietary-2 to Non-Proprietary
	1.0	Removed proprietary information and resolved flow/heat transfer loss as "insignificant". Provided "recirculating steam generator" for RSG acronym. Substituted Alloy 690 for Inconel 690.
	6.0	Referenced non-proprietary document.

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1.0 INTRODUCTION

Sleeves are an integral part of the steam generator tubing repair process allowing the tube to remain in service. This continues the heat removal capabilities and core cooling conditions desired by the steam generator design. The BWNS sleeves for tubesheet (TS) and tube support plates (TSP) are fabricated from Alloy 690 thermally treated (TT) and are licensed [6.1] in accordance with the requirements of the USNRC This original sleeve design was directed toward application in Westinghouse Model D steam generators with 3/4 inch OD tubes.

Since that time, over 5000 BWNS kinetic sleeves have been successfully installed. In-service inspection at outages subsequent to installation have shown that the sleeve has an excellent record of performance. This field experience has also allowed BWNS to identify areas for improvement, primarily in installation rates and quality of the sleeve-to-tube joints. These modifications to the design have resulted in a sleeve that can be located at tube support plates in addition to the tubesheet. 'his modified design is proposed for application at Byron.

BWNS has evaluated this proposed amendment for the Byron Station and determined that it involves no significant hazards considerations. According to 10CFR50.92(c), a proposed amendment to an operating license involves no significant hazards 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.

The proposed amendment does make changes to the Station Operating Technical Specifications for in-service inspection for the reactor coolant system steam generators. Operationally, the sleeve has an insignificant influence on the primary coolant flow rate parameters.

For the sleeving scope the technical specification review involves (A) the inspection criteria for the sleeve in lieu of the steam generator tubing, (B) the heat transfer capability of the sleeve, (C) the reactor coolant pump flow parameters, and (D) the unlikely event that a sleeve defect can cause a primary coolant release to the secondary side of the steam generator. As tube plugging was previously approved and performed at Byron, the sleeving effect is considered insignificant for the steam generator heat transfer loss and RCP flow performance.

1.1 Description of Change

The BWNS design has two different length sleeves, with the shorter for the outer perimeter tubes restricted by bowl-to-tubesheet headheight. TS and TSP sleeves are manufactured from thermally treated (for corrosion resistance) Alloy 690 tubing and have been qualified for spanning parent tube defects.

Although sleeving slightly diminishes fluid flow capability, the impact is insignificant as compared to the fluid flow and heat transfer losses due to plugging [6.1].

The installation steps for the sleeve are:

- ECT Inspect The Tube
- Clean The Tube
- Sleeve Insertion and Kinetic Welds
- Stress Relief
- ECT Inspect Sleeve/Tube

The sleeve installation is totally remote using the BWNS ROGER™ or COBRA™ manipulators. Typically these steps are performed for a large batch of sleeves, thus minimizing the number of tool changes on the manipulator.

Eddy current inspection is initially performed on the tube to be sleeved in order to confirm that there are no obstructions and that there are no defects of a specified percent through wall or greater in the weld regions.

Following this, the tubes are cleaned in the regions to be welded using a flexible hone. The hone is rotated and oscillated at the weld regions using a flexible shaft through a snorkel.

The sleeve is inserted into the tube using an actuator mounted onto the manipulator. During insertion, a hardstop on the sleeve assembly contacts the bottom of the actuator, assuring the sleeve is properly positioned. The welding occurs by remotely detonating the two kinetic welding devices. After welding, the actuator removes the disposable parts of the sleeve welding assembly from the tube and delivers it to the manway for disposal.

Stress relief of the freespan weld occurs next. This is done using a resistance heater positioned in the sleeve at the weld location. Proper positioning is assured by utilizing a hardstop mounted on the heater which contacts the end of the sleeve. The tube temperature is controlled by correlation to thermocouples mounted on the heater which provide continuous feedback for the computer control.

The final installation step is eddy current inspection of the sleeve/tube combination. This inspection serves several functions. The new sleeve pressure boundary is inspected for defects and to establish a baseline for future inspections. The upper and lower expansions are dimensionally verified. Finally, the axial position of the sleeve is verified. All of these items are performed with one pass of the ECT probe.

1.2 Affected Systems

The sleeves are qualified for use in the Westinghouse D-Series recirculating steam generators. The installation of the sleeve enhances the safety of the plant by keeping the degraded steam generator tube in-service. Safety is improved because the risk of a tube rupture is reduced due to the double barrier established between the primary and secondary fluids.

The steam generator is the key affected system. There is an insignificant reduction of both fluid flow and heat transfer surface, but these are much less then the loss due to one plugged tube.

RCP pump/motor performances and turbine valve settings are not affected by a typical sleeving campaign.

Reactor core nucleonic computations, i.e. for boric acid calculations or control rod positioning, are also not affected by sleeving.

Secondary side operation for feedwater, auxiliary feedwater, blowdown, and outage maintenance are not affected by sleeve installation.

1.3 Failure Modes

The sleeves are installed in tubes with defects or degradation. The sleeve then serves as fluid boundary between primary and secondary coolant in addition to providing a structural strength. The sleeve is designed to prevent primary water from entering the secondary side of the steam generator should the parent tube develop a through wall defect.

The first scenario is the sleeve is placed in service and the mechanical bonding is not adequate. This would result in primary to secondary leakage between a tight tolerance fit.

The second scenario is the sleeve is placed in service and the tube degradation allows the tube to leak or sever between the sleeve attachments. The sleeve is qualified to maintain both structural loads and fluid boundary flows under all aspects of design and transient conditions.

The third scenario is the outer tube becomes severed inside the tubesheet at the lower sleeve attachment and the sleeve remains intact. This condition was qualified by the sleeve design and testing; the sleeve is fully capable of maintaining structural and fluid boundary integrity for all design and transient conditions.

The fourth scenario is the tube severs in the freespan adjacent to the sleeve attachment, allowing a flowpath for reactor coolant to the secondary side of the steam generator. The sleeve prolongation will stay engaged in the severed tube end to limit the leak flow area and prevent tube whip. This condition is identified as a tube leak/tube rupture in the station licensing submittal for power operation. The plant operating technical specifications identifies leakage rates for a primary to secondary source and shutdown criteria.

1.4 Qualification

The sleeves are manufactured from Alloy 690 TT and are designed and tested for all operational and design considerations. Analyses were performed on the recirculating steam generator (RSG) sleeve. The analyses consist of a design stress analysis; analysis to support fatigue testing per Appendix II of the ASME Code; analysis of flow-induced vibration of sleeved tubes; analysis of a degraded sleeve for the plugging criteria of RG 1.121; analysis of the effect of sleeves on RSG heat transfer flow; and a certified stress report.

An additional specific percent of wall thickness is deducted as combined allowance for postulated degradation due to corrosion and for eddy current testing inaccuracy. Therefore, a defect plugging limit for the original sleeve wall is established.

The adequacy of the sleeve attachments to withstand cyclic loadings were demonstrated by means of a fatigue test per ASME Code, Section III, Appendix II-1500.

The sleeves are designed to accommodate all loads that any normal undegraded steam generator tube may experience due to normal plant conditions and all anticipated transients specified for the steam generator. The fatigue test load calculations determine a conservative maximum loading for a sleeve in any steam generator tube. These calculations include both pressure and thermal loading.

Loads due to thermal and pressure transients were calculated for three sleeved tube cases. Where possible, transients were grouped together and the number of cycles adjusted accordingly. The sleeved tube cases analyzed are listed below.

- Non-Leaking Outer Tube
- Leaking Outer Tube
- Severed Outer Tube

The specific combination of geometry and operating conditions which resulted in the highest load for a given transient grouping was then used in the mechanical test program.

Test load ranges and required cycles are obtained per ASME Appendix II. Increased operational load range or number of cycles is required based on the number of test assemblies and various factors relating the test conditions to the actual operating conditions. The load sets used a fatigue test factor applied to the operational load range to obtain the test load range.

The pressure cycling portion of the fatigue test is based on hundreds of cycles of heatup. The 2500 psi range of test pressure conservatively represents the pressure effects of normal and accident service life.

The flow-induced vibration (FIV) analysis for the sleeve evaluated fluid-elastic stability margin, vortex shedding and random vibration. The analysis compared a virgin undegraded tube to a tube sleeved at the secondary face of the tubesheet on the hot leg and cold leg of the steam generator. For the sleeved tube, the tube was considered severed between the two sleeve joints.

2.0 IMPACT ON ACCIDENTS EVALUATED AS THE DESIGN BASIS

The sleeves have been designed, analyzed, and tested for design and operating conditions. This document provides evaluation for effects on the reactor fuel, reactor coolant system, steam generators, and technical specifications. The result is the plant is in a safer environment by sleeving vs. plugging as the sleeved tube continues to provide core cooling and heat transfer capability. At worst case a combination of tube and sleeve rupture could occur with the result being a primary to secondary leak. The probability of this combination is considered unlikely.

Should a tube/sleeve leak occur the impact is bounded by the ruptured tube evaluation submitted by the utility for the operating license. No new or unreviewed accident conditions are created by the sleeve addition. The potential for a tube leak or rupture is not increased from the original submittal, thus there is no impact on accidents evaluated as the design basis.

Thus, 10CFR50.92(c)(1) is satisfied.

3.0 POTENTIAL FOR CREATING AN UN-ANALYZED EVENT

There is not a potential for creating an un-analyzed event. The failure modes discussed indicate the events that can occur are already analyzed. The worst case scenario is a failed sleeve leak in a severed tube resulting in a primary coolant release to the secondary side of the steam generator with the reactor at full power. This is not considered to be a new scenario for a reactor accident. There is not an increased probability of an accident by installing either tubesheet or tube support plate sleeves in the Byron Station steam generators.

Thus, 10CFR50.92(c)(2) is satisfied.

4.0 IMPACT ON MARGIN OF SAFETY

Based on previous responses, installing either tubesheet or tube support plate sleeves will increase the protective boundaries of the steam generators. A tube with degradation can be sleeved, and the sleeve then enhances the safety of the plant while keeping degraded tubes in service. Safety is improved because a double barrier is established between the primary and secondary fluids, which reduces the risk of tube rupture. The installation of the TS and TSP sleeves into the parent tube will not affect the acceptance limits for the tube protective boundary nor reduce the margin of safety.

Thus, 10CFR50.92(c)(3) is satisfied.

5.0 CONCLUSIONS

The BWNS kinetic sleeve licensed for application in D-series steam generators design is such that the installation at Byron Station Model D4 and D5 is conservatively bounded by the original qualification. This design involves no new significant hazards as evaluated in accordance with 10CFR50.92(c).

6.0 REFERENCES

6.1 BWNS Document 43-2045A-01, "Recirculating Steam Generators Kinetic Sleeve Qualification For 3/4 Inch OD Tubes", Non-Proprietary.

BWNT-20440-6 (11/89)

B&W NUCLEAR SERVICE COMPANY

ENGINEERING INFORMATION RECORD

Document Identifier 51- 1223765-01

Title EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATIONS - BRAIDWOOD STATION

PREPARED BY:		REVIEWED BY:	
Name	R. F. PENN, JR.	Name	J. A. LAUFF
Signature R. F. Penn Jr.	Date 7/21/93	Signature J. A. Lauff	Date 7/21/93

Technical Manager Statement: Initials JAF

Reviewer is Independent.

Remarks:

BWNS NON-PROPRIETARY

This document provides an evaluation for the BWNS tube sheet and tube support plate sleeves which concludes, in accordance with 10CFR50.92(c), there are no significant hazard considerations for this application at the Commonwealth Edison Braidwood Station. Braidwood is a four loop Westinghouse NSSS with Model D4 and D5 steam generators for Unit 1 and 2, respectively.

Tubesheet (TS) and tube support plate (TSP) sleeves are a viable method to maintain steam generator (S/G) tubing integrity. The sleeve serves as the pressure boundary and allows the degraded tube to remain in-service for heat transfer and core cooling. S/G tubing and sleeve NDE is performed on a scheduled basis; wall loss or deformation is trended accordingly.

The BWNS TSP and TS sleeves are kinetically welded at both ends. The sleeve design allows for tube plugging or stabilization at a later date should that need arise.

RECORD OF REVISION

REV.NO/NAME	SECTION	DESCRIPTION OF CHANGE
00/RF PENN	ALL	ORIGINAL ISSUE.
01/RF PENN	All Pages	Classification change from Proprietary-2 to Non-Proprietary
	1.0	Removed proprietary information and resolved flow/heat transfer loss as "insignificant". Provided "recirculating steam generator" for RSG acronym. Substituted Alloy 690 for Inconel 690.
	6.0	Referenced non-proprietary document.

BWNS NON-PROPRIETARY

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BWNS NON-PROPRIETARY

1.0 INTRODUCTION

Sleeves are an integral part of the steam generator tubing repair process allowing the tube to remain in service. This continues the heat removal capabilities and core cooling conditions desired by the steam generator design. The BWNS sleeves for tubesheet (TS) and tube support plates (TSP) are fabricated from Alloy 690 thermally treated (TT) and are licensed [6.1] in accordance with the requirements of the USNRC. This original sleeve design was directed toward application in Westinghouse Model D steam generators with 3/4 inch OD tubes.

Since that time, over 5000 BWNS kinetic sleeves have been successfully installed. In-service inspection at outages subsequent to installation have shown that the sleeve has an excellent record of performance. This field experience has also allowed BWNS to identify areas for improvement, primarily in installation rates and quality of the sleeve-to-tube joints. These modifications to the design have resulted in a sleeve that can be located at tube support plates in addition to the tubesheet. This modified design is proposed for application at Braidwood.

BWNS has evaluated this proposed amendment for the Braidwood Station and determined that it involves no significant hazards considerations. According to 10CFR50.92(c), a proposed amendment to an operating license involves no significant hazards 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 fr a any accident previously evaluated; or

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

The proposed amendment does make changes to the Station Operating Technical Specifications for in-service inspection for the reactor coolant system steam generators. Operationally, the sleeve has an insignificant influence on the primary coolant flow rate parameters.

For the sleeving scope the technical specification review involves (A) the inspection criteria for the sleeve in lieu of the steam generator tubing, (B) the heat transfer capability of the sleeve, (C) the reactor coolant pump flow parameters, and (D) the unlikely event that a sleeve defect can cause a primary coolant release to the secondary side of the steam generator. As tube plugging was previously approved and performed at Braidwood, the sleeving effect is considered insignificant for the steam generator heat transfer loss and RCP flow performance.

1.1 Description of Change

The BWNS design has two different length sleeves, with the shorter for the outer perimeter tubes restricted by bowl-to-tubesheet headheight. TS and TSP sleeves are manufactured from thermally treated (for corrosion resistance) Alloy 690 tubing and have been qualified for spanning parent tube defects.

Although sleeving slightly diminishes fluid flow capability, the impact is insignificant as compared to the fluid flow and heat transfer losses due to plugging [6.1].

The installation steps for the sleeve are:

- ECT Inspect The Tube
- Clean The Tube
- Sleeve Insertion and Kinetic Welds
- Stress Relief
- ECT Inspect Sleeve/Tube

The sleeve installation is totally remote using the BWNS ROGER™ or COBRA™ manipulators. Typically these steps are performed for a large batch of sleeves, thus minimizing the number of tool changes on the manipulator.

Eddy current inspection is initially performed on the tube to be sleeved in order to confirm that there are no obstructions and that there are no defects of a specified percent through wall or greater in the weld regions.

Following this, the tubes are cleaned in the regions to be welded using a flexible hone. The hone is rotated and oscillated at the weld regions using a flexible shaft through a snorkel.

The sleeve is inserted into the tube using an actuator mounted onto the manipulator. During insertion, a hardstop on the sleeve assembly contacts the bottom of the actuator, assuring the sleeve is properly positioned. The welding occurs by remotely detonating the two kinetic welding devices. After welding, the actuator removes the disposable parts of the sleeve welding assembly from the tube and delivers it to the manway for disposal.

Stress relief of the freespan weld occurs next. This is done using a resistance heater positioned in the sleeve at the weld location. Proper positioning is assured by utilizing a hardstop mounted on the heater which contacts the end of the sleeve. The tube temperature is controlled by correlation to thermocouples mounted on the heater which provide continuous feedback for the computer control.

The final installation step is eddy current inspection of the sleeve/tube combination. This inspection serves several functions. The new sleeve pressure boundary is inspected for defects and to establish a baseline for future inspections. The upper and lower expansions are dimensionally verified. Finally, the axial position of the sleeve is verified. All of these items are performed with one pass of the ECT probe.

1.2 Affected Systems

The sleeves are qualified for use in the Westinghouse D-Series recirculating steam generators. The installation of the sleeve enhances the safety of the plant by keeping the degraded steam generator tube in-service. Safety is improved because the risk of a tube rupture is reduced due to the double barrier established between the primary and secondary fluids.

The steam generator is the key affected system. There is an insignificant reduction of both fluid flow and heat transfer surface, but these are much less then the loss due to one plugged tube.

RCP pump/motor performances and turbine valve settings are not affected by a typical sleeving campaign.

Reactor core nucleonic computations, i.e. for boric acid calculations or control rod positioning, are also not affected by sleeving.

Secondary side operation for feedwater, auxiliary feedwater, blowdown, and outage maintenance are not affected by sleeve installation.

1.3 Failure Modes

The sleeves are installed in tubes with defects or degradation. The sleeve then serves as fluid boundary between primary and secondary coolant in addition to providing a structural strength. The sleeve is designed to prevent primary water from entering the secondary side of the steam generator should the parent tube develop a through wall defect.

The first scenario is the sleeve is placed in service and the mechanical bonding is not adequate. This would result in primary to secondary leakage between a tight tolerance fit.

The second scenario is the sleeve is placed in service and the tube degradation allows the tube to leak or sever between the sleeve attachments. The sleeve is qualified to maintain both structural loads and fluid boundary flows under all aspects of design and transient conditions.

The third scenario is the outer tube becomes severed inside the tubesheet at the lower sleeve attachment and the sleeve remains intact. This condition was qualified by the sleeve design and testing; the sleeve is fully capable of maintaining structural and fluid boundary integrity for all design and transient conditions.

The fourth scenario is the tube severs in the freespan adjacent to the sleeve attachment, allowing a flowpath for reactor coolant to the secondary side of the steam generator. The sleeve prolongation will stay engaged in the severed tube end to limit the leak flow area and prevent tube whip. This condition is identified as a tube leak/tube rupture in the station licensing submittal for power operation. The plant operating technical specifications identifies leakage rates for a primary to secondary source and shutdown criteria.

1.4 Qualification

The sleeves are manufactured from Alloy 690 TT and are designed and tested for all operational and design considerations. Analyses were performed on the recirculating steam generator (RSG) sleeve. The analyses consist of a design stress analysis; analysis to support fatigue testing per Appendix II of the ASME Code; analysis of flow-induced vibration of sleeved tubes; analysis of a degraded sleeve for the plugging criteria of RG 1.121; analysis of the effect of sleeves on RSG heat transfer flow; and a certified stress report.

An additional specific percent of wall thickness is deducted as combined allowance for postulated degradation due to corrosion and for eddy current testing inaccuracy. Therefore, a defect plugging limit for the original sleeve wall is established.

The adequacy of the sleeve attachments to withstand cyclic loadings were demonstrated by means of a fatigue test per ASME Code, Section III, Appendix II-1500.

The sleeves are designed to accommodate all loads that any normal undegraded steam generator tube may experience due to normal plant conditions and all anticipated transients specified for the steam generator. The fatigue test load calculations determine a conservative maximum loading for a sleeve in any steam generator tube. These calculations include both pressure and thermal loading.

Loads due to thermal and pressure transients were calculated for three sleeved tube cases. Where possible, transients were grouped together and the number of cycles adjusted accordingly. The sleeved tube cases analyzed are listed below.

- Non-Leaking Outer Tube
- Leaking Outer Tube
- Severed Outer Tube

The specific combination of geometry and operating conditions which resulted in the highest load for a given transient grouping was then used in the mechanical test program.

Test load ranges and required cycles are obtained per ASME Appendix II. Increased operational load range or number of cycles is required based on the number of test assemblies and various factors relating the test conditions to the actual operating conditions. The load sets used a fatigue test factor applied to the operational load range to obtain the test load range.

The pressure cycling portion of the fatigue test is based on hundreds of cycles of heatup. The 2500 psi range of test pressure conservatively represents the pressure effects of normal and accident service life.

The flow-induced vibration (FIV) analysis for the sleeve evaluated fluid-elastic stability margin, vortex shedding and random vibration. The analysis compared a virgin undegraded tube to a tube sleeved at the secondary face of the tubesheet on the hot leg and cold leg of the steam generator. For the sleeved tube, the tube was considered severed between the two sleeve joints.

2.0 IMPACT ON ACCIDENTS EVALUATED AS THE DESIGN BASIS

The sleeves have been designed, analyzed, and tested for design and operating conditions. This document provides evaluation for effects on the reactor fuel, reactor coolant system, steam generators, and technical specifications. The result is the plant is in a safer environment by sleeving vs. plugging as the sleeved tube continues to provide core cooling and heat transfer capability. At worst case a combination of tube and sleeve rupture could occur with the result being a primary to secondary leak. The probability of this combination is considered unlikely.

Should a tube/sleeve leak occur the impact is bounded by the ruptured tube evaluation submitted by the utility for the operating license. No new or unreviewed accident conditions are created by the sleeve addition. The potential for a tube leak or rupture is not increased from the original submittal, thus there is no impact on accidents evaluated as the design basis.

Thus, 10CFR50.92(c)(1) is satisfied.

3.0 POTENTIAL FOR CREATING AN UN-ANALYZED EVENT

There is not a potential for creating an un-analyzed event. The failure modes discussed indicate the events that can occur are already analyzed. The worst case scenario is a failed sleeve leak in a severed tube resulting in a primary coolant release to the secondary side of the steam generator with the reactor at full power. This is not considered to be a new scenario for a reactor accident. There is not an increased probability of an accident by installing either tubesheet or tube support plate sleeves in the Braidwood Station steam generators.

Thus, 10CFR50.92(c)(2) is satisfied.

4.0 IMPACT ON MARGIN OF SAFETY

Based on previous responses, installing either tubesheet or tube support plate sleeves will increase the protective boundaries of the steam generators. A tube with degradation can be sleeved, and the sleeve then enhances the safety of the plant while keeping degraded tubes in service. Safety is improved because a double barrier is established between the primary and secondary fluids, which reduces the risk of tube rupture. The installation of the TS and TSP sleeves into the parent tube will not affect the acceptance limits for the tube protective boundary nor reduce the margin of safety.

Thus, 10CFR50.92(c)(3) is satisfied.

5.0 CONCLUSIONS

The BWNS kinetic sleeve licensed for application in D-series steam generators design is such that the installation at Braidwood Station Model D4 and D5 is conservatively bounded by the original qualification. This design involves no new significant hazards as evaluated in accordance with 10CFR50.92(c).

6.0 REFERENCES

6.1 BWNS Document 43-2045A-01, "Recirculating Steam Generators Kinetic Sleeve Qualification For 3/4 Inch OD Tubes", Non-Proprietary.

ATTACHMENT D

ENVIRONMENTAL ASSESSMENT

Commonwealth Edison has evaluated the proposed amendment against the criteria for and 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).

The sleeving process does result in radioactive waste which is considered disposable and cannot be reused. This waste is described in WCAP 13698 Rev. 1 and Topical Report BAW 2045PA Rev. 1. The amount of waste created using the sleeving process is comparable to that created by tube plugging.

The proposed change does not involve a significant hazards consideration as discussed in Attachment C to this letter. Also, this proposed amendment will not involve significant changes in the types or amounts of any radioactive effluents nor does it affect any of the permitted release paths. In addition, this change does not involve a significant increase in individual or cumulative occupational exposure, therefore, this change meets the categorical exclusion permitted by 10CFR51.22(c)(9).

ATTACHMENT E

WESTINGHOUSE DOCUMENTATION

1. Westinghouse Electric Corporation Authorization Letter, CAW-93-480, dated May 27, 1993, "APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE", regarding WCAP-13698, Revision 1, "Laser Welded Sleeves for 3/4 Inch Diameter Tube Feedring-Type and Westinghouse Preheater Steam Generators", dated May 1993 (Proprietary).

2. Westinghouse Affidavit, CAW-93-480, dated May 27, 1993.

3. Westinghouse Proprietary Information Notice.

4. Westinghouse Copyright Notice.

5. WCAP-13698, Revision 1, "Laser Welded Sleeves for 3/4 Inch Diameter Tube Feedring-Type and Westinghouse Preheater Steam Generators", dated May 1993 (Proprietary).

6. WCAP-13699, Revision 1, "Laser Welded Sleeves for 3/4 Inch Diameter Tube Feedring-Type and Westinghouse Preheater Steam Generators", dated May 1993 (Non-Proprietary).