ULNRC-3357

ATTACHMENT 1

TECHNICAL SPECIFICATION CHANGES

(MARKED-UP)

ULNRC-3357

REVISION 1

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. When applying the exceptions of 4.4.5.2.a through 4.4.5.2.c, previous defects or imperfections in the area repaired by sleeving are not considered an area requiring inspection. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

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

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

ULNRC-3357

REVISION 1

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 from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube 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;

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

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

or repair — or sleeve

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

or Repair — by plugging or repaired by sleeving

6) Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to 48% of the ~~nominal~~ tube wall thickness~~%~~.

40%

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 Specification 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~~ For a tube repaired by sleeving, the tube inspection shall include the sleeved portion of the tube; and

The plugging limit for laser welded sleeves is equal to 39% of the nominal sleeve wall thickness.

ULNRC-3357.

REVISION 1

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 1

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug or repair by sleeving 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 and sleeves 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.

INSERT 1

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 in Westinghouse Technical Report WCAP-14596-P, "Laser Welded Elevated Tube Sheet Sleeves For Westinghouse Model F Steam Generators." March 1996 (W Proprietary)

ULNRC-3357

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 defective tubes and inspect additional 2S tubes in this S. G. or repair	C–1	None	N. A.	N. A.
			C–2	Plug defective tubes and inspect additional 4S tubes in this S. G.	C–1	None
					C–2	Plug 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 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 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

ULNRC-3357

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 gpm UNIDENTIFIED LEAKAGE.

c. ~~1 gpm~~ 600 gpd total reactor-to-secondary leakage through all steam generators not isolated from the Reactor Coolant System and ~~500~~ 150 gallons per day through any one steam generator,

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

e. 8 gpm per RC pump CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 ± 20 psig, and

f. The leakage from each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1 shall be limited to 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm, at a Reactor Coolant System pressure of 2235 ± 20 psig.*

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

ACTION:

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

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

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the above limit, reduce the leakage rate to within limits within 4 hours, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 12 hours with an RCS pressure of less than 600 psig.

---

*Test pressures less than 2235 psig but greater than 150 psig are allowed. Observed leakage shall be adjusted for the actual test pressure up to 2235 psig assuming the leakage to be directly proportional to pressure differential to the one-half power.

ULNRC-3357

REVISION 1

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.

Unscheduled inservice inspections are performed on each steam generator following: (1) reactor to secondary tube leaks; (2) a seismic occurrence greater than the Operating Basis Earthquake; and (3) a loss-of-coolant accident requiring actuation of the Engineered Safety Features, which for this Specification is defined to be a break greater than that equivalent to the severance of a 1" inside diameter pipe, or, for a main steamline or feedline, a break greater than that equivalent to a steam generator safety valve failing open; to ensure that steam generator tubes retain sufficient integrity for continued operation. Transients less severe than these do not require inspections because the resulting stresses are well within the stress criteria established by Regulatory Guide 1.121, which unplugged steam generator tubes must be capable of withstanding.

150

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.

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 repair will be required for all tubes with imperfections exceeding the plugging limit. ~~of 48% of the tube nominal wall thickness.~~ Steam generator

STEAM GENERATORS (Continued)

tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness. Results from WCAP-10043 have been used to establish plugging limit.

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

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

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

3/4.4.6.2 OPERATIONAL LEAKAGE

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

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

The total steam generator tube leakage limit of ~~1 gpm~~ 600 gpd for all steam generators not isolated from the RCS ensures that the dosage contribution from the tube leakage will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of either a steam generator tube rupture or steam line break. The ~~1 gpm~~ 600 gpd limit is ~~consistent with~~ conservative compared to the assumptions used in the analysis of these accidents. The ~~500~~ 150 gpd leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

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

The CONTROLLED LEAKAGE limitation restricts operation when the total flow from the reactor coolant pump seals exceeds 8 gpm per RC pump at a nominal RCS pressure of 2235 psig. This limitation ensures adequate performance of the RC pump seals.

The plugging or repair limit for the pressure boundary portion of laser welded sleeves is determined to be 39% through-wall (by NDE). The laser welded sleeve repair limit applicable to the pressure boundary portion of the sleeve is established in WCAP-14596. Appropriate NDE techniques are also discussed in WCAP-14596.

ULNRC-3357

ATTACHMENT 2

TECHNICAL SPECIFICATION CHANGES

(RE-TYPED)

ULNRC-3357

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. When applying the exceptions of 4.4.5.2.a through 4.4.5.2.c, previous defects or imperfections in the area repaired by sleeving are not considered an area requiring inspection. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

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

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

ULNRC-3357

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 from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube 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;

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

4) % Degradation means the percentage of the tube 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 defect is defective;

6) Plugging or Repair Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging or repaired by sleeving and is equal to 40% of the nominal tube wall thickness. The plugging limit for laser welded sleeves is equal to 39% of the nominal sleeve 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 Specification 4.4.5.3c., above;

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

ULNRC-3357

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

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

a. Laser welded sleeving as described in Westinghouse Technical Report WCAP-14596-P, "Laser Welded Elevated Tube Sheet Sleeves For Westinghouse Model F Steam Generators." March 1996 (W Proprietary)

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug or repair by sleeving 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 and sleeves 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 4.4-2

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

ULNRC-3357

ULNRC-3357

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 gpm UNIDENTIFIED LEAKAGE,

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

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

e. 8 gpm per RC pump CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 ± 20 psig, and

f. The leakage from each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1 shall be limited to 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm, at a Reactor Coolant System pressure of 2235 ± 20 psig.*

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

ACTION:

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

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

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the above limit, reduce the leakage rate to within limits within 4 hours, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 12 hours with an RCS pressure of less than 600 psig.

---

* Test pressures less than 2235 psig but greater than 150 psig are allowed. Observed leakage shall be adjusted for the actual test pressure up to 2235 psig assuming the leakage to be directly proportional to pressure differential to the one-half power.

ULNRC-3357

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.

Unscheduled inservice inspections are performed on each steam generator following: (1) reactor to secondary tube leaks; (2) a seismic occurrence greater than the Operating Basis Earthquake; and (3) a loss-of-coolant accident requiring actuation of the Engineered Safety Features, which for this Specification is defined to be a break greater than that equivalent to the severance of a 1" inside diameter pipe, or, for a main steamline or feedline, a break greater than that equivalent to a steam generator safety valve failing open; to ensure that steam generator tubes retain sufficient integrity for continued operation. Transients less severe than these do not require inspections because the resulting stresses are well within the stress criteria established by Regulatory Guide 1.121, which unplugged steam generator tubes must be capable of withstanding.

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

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 repair will be required for all tubes with imperfections exceeding the plugging or repair limit. Steam generator tube inspections of operating plants have demonstrated

ULNRC-3357

STEAM GENERATORS (Continued)

the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness. Results from WCAP-10043 have been used to establish plugging limit.

The plugging or repair limit for the pressure boundary portion of laser welded sleeves is determined to be 39% through-wall (by NDE). The laser welded sleeve repair limit applicable to the pressure boundary portion of the sleeve is established in WCAP-14596. Appropriate NDE techniques are also discussed in WCAP-14596.

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

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

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

3/4.4.6.2 OPERATIONAL LEAKAGE

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

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

The total steam generator tube leakage limit of 600 gpd for all steam generators not isolated from the RCS ensures that the dosage contribution from the tube leakage will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of either a steam generator tube rupture or steam line break. The 600 gpd limit is conservative compared to the assumptions used in the analysis of these accidents. The 150 gpd leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

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

The CONTROLLED LEAKAGE limitation restricts operation when the total flow from the reactor coolant pump seals exceeds 8 gpm per RC pump at a nominal RCS pressure of 2235 psig. This limitation ensures adequate performance of the RC pump seals.

ULNRC-3357

ATTACHMENT 3

SAFETY EVALUATION

SAFETY EVALUATION

INTRODUCTION

This proposed amendment revises the Surveillance Requirements of Technical Specification (TS) 3/4.4.5 "Steam Generators" and 3.4.6.2 "Operational Leakage" and associated Bases as appropriate, to address the installation of laser welded tube sleeves (LWSs) in the Callaway Plant steam generators.

This license amendment request revises TS 3/4.4.5, 3.4.6.2, and associated Bases to include:

a. Laser welded sleeving per Westinghouse WCAP-14596 as an approved tube repair method,

b. The associated sleeve wall depth based plugging limit value and inspection requirements,

c. Reduction of the tube plugging limit from 48% to 40% through wall (of the nominal tube wall thickness) to be consistent with NUREG-1431, "Standard Technical Specifications for Westinghouse Plants", and,

d. The reduction of the primary to secondary normal operational leakage limit from 500 to 150 gpd per steam generator.

Currently, tubes with indications of degradation in excess of the plugging criteria are removed from service by plugging. Removal of a tube from service results in a reduction of reactor coolant flow through the steam generator. This small reduction in flow can impact 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 via sleeving maintains the tube heat transfer area and results in a much smaller RCS flow reduction. Therefore, the use of sleeving in lieu of plugging helps to assure that minimum flow rates are maintained in excess of that required for operation at full power. Any combination of sleeving and plugging, up to a level such that the effect will not reduce the minimum reactor coolant flow rate to below the current TS limit or below the plugging limits analyzed in the Callaway Safety Analysis Report is acceptable. The sleeve/plug equivalency results are contained in WCAP-14596.

BACKGROUND

Callaway has Westinghouse Model F steam generators which utilize 11/16" OD x 0.040" nominal wall thickness tubes. The first ten rows of tubes at Callaway are thermally treated Alloy 600 (1326 tubes), while the remainder of the tubes (4300 tubes) are mill annealed Alloy 600. The Callaway tubes are hydraulically expanded within the tubesheet region. The pressure utilized for the expansion process is designed to provide a radial preload between the tube and tubesheet such that the tube to tubesheet gap is completely reduced during all conditions. The current Callaway TS require steam generator tubes with eddy current indications of 48% through wall or greater to be removed from service.

This amendment proposes to permit the installation of Alloy 690 elevated LWSs within the tubesheet area of the steam generators at the Callaway Plant. This steam generator tube repair method secures to the original tube a short length of tubing with an outer diameter slightly smaller than the inside diameter of the parent tube, thereby spanning the degraded area. LWS has been determined to be an effective repair method for degraded steam generator tubes and has been licensed and installed at plants with both 3/4" and 7/8" outside diameter steam generator tubes. Laser welded sleeves are installed by first hydraulically expanding the sleeve against the tube to bring the tube and sleeve into intimate contact for weld quality purposes. Minimal levels of tube deformation are introduced in the hydraulic expansion process. The upper joint (located above the tubesheet in the tube free span region) is then welded, followed by a roll expansion at the lower joint, which is located within the tubesheet region. The bottom edge of the elevated tubesheet sleeve will be located approximately 6 inches below the top of the tubesheet (TTS) and extend to approximately 6 inches above the TTS. The only expected location of indications at Callaway is the top of tubesheet expansion transition region.

EVALUATION

Generic Structural Assessment

The LWS described in WCAP-14596 has been designed to Section III, Subsection NB-3300, of the 1989 Edition of the ASME Code. Fatigue and stress analyses of the sleeved tube assemblies have been completed in accordance with the requirements of Section III, Subsection NB-3200 of the 1989 Edition of the ASME Code. Both the sleeve and weld are evaluated. The structural evaluation considers the effects of operation upon the assembly by considering cases of free and fixed tube support conditions and intact and separated tube conditions upon the applied stresses. The results of the primary stress intensity evaluation, primary plus secondary stress intensity range evaluation and fatigue evaluation indicate that the ASME Code allowable limits are not exceeded. That is, stress intensities are bounded by the Code minimum limits for SB-163 (Alloy 690) material and cumulative fatigue usage is less than 1.0. Therefore, the design of the sleeve pressure boundary meets the design objectives of the original tubing.

Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes" and the ASME Code are used to develop the plugging limit of the sleeve 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. No credit for the presence of the parent tube behind the sleeve is assumed when performing the minimum wall/burst evaluation.

The sleeve and weld structural analysis utilizes a generic set of design and transient loading inputs which are intended to bound all plants with Westinghouse Model F steam generators. The temperature and pressure variances used in the assumed operating conditions and generic transients are bounding.

An ultrasonic inspection of the free span welds is performed prior to placing the sleeve in service 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. In addition, a fatigue analysis was performed for the assembly with 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 bounding for an operating plant. The results of the fatigue analysis indicate acceptable usage factors for the entire range of permitted weld thickness.

Leakage Assessment

The LWS joint is an autogenous tube-to-sleeve weld. Leakage testing of 3/4" and 7/8" LWS assemblies under conditions considered to be more severe than expected during all operating plant conditions has shown that the laser welded joint does not introduce additional primary to secondary leakage during a postulated steam line break event. LWS tube assemblies were subjected to thermal and fatigue cycling and then leak tested at pressure differentials of up to 3110 psi, which far exceeds the expected maximum feed line break or steam line break pressure differential. Leakage testing has also shown that the non-welded elevated tubesheet sleeve lower joint is essentially leaktight during all plant conditions. 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. Essentially no leakage was detected in any non-welded tubesheet sleeve lower joint at 600°F in hydraulically expanded tubes after both thermal and fatigue loading. Primary to secondary leakage through non-welded tubesheet sleeve lower joints would be expected to be negligible at SLB, normal operating, and at 0% power conditions, and therefore will not significantly contribute to offsite doses in the event of postulated SLB. Due to time limitations, leakage testing of LWS specimens for 11/16" tubes has not been performed. These tests are scheduled to be completed before the Callaway Plant Refuel 8 outage, which is currently scheduled for Fall 1996. The leakage test results will be provided to NRC upon completion. The testing program for 11/16" tubes will be consistent with the previously performed LWS leakage tests.

Corrosion Assessment

Thermally treated Alloy 600 and Alloy 690 sleeved tube assemblies have performed well historically with regard to corrosion. Accelerated corrosion test results show the free span laser welded joint with post weld heat treatment (PWHT) 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. These factors suggest postulated sleeve degradation, even in a non-heat treated condition, would occur at a relatively slow rate, and be detected by routine eddy current inspection. The heat treatment process is designed to achieve sufficient stress reduction such that rapid crack initiation and propagation in the joint is not expected. Test data indicates acceptable corrosion resistance at temperatures as low as 1250°F. On the tube ID surface, which is where the weld cooling stresses are concentrated, temperatures during heat treatment will exceed the OD temperature, thereby providing a more effective heat treatment in this area. PWHT parent tube far field stresses can affect the susceptibility of the weld to cracking. The steam generator geometry, fixity conditions at the tube supports, and PWHT temperature all influence the resultant far field stresses. In order to develop significant post-process far field residual stresses, it must be assumed that the tube is locked at the tube support plates (TSPs). The post-process tube far field residual stresses (assuming the tubes are locked at the first TSP) for a heat treatment temperature of 1350°F at Callaway Plant are similar to other plants.

Mechanical Integrity Assessment

Mechanical testing of 3/4" and 7/8" laser weld and hybrid expansion joint (HEJ) sleeves indicates that the axial load bearing capability of these joints individually exceeds the most limiting theoretical pressure end cap loading established by RG 1.121. Both the lower joint (hydraulic expansion plus roll expansion, commonly known as an HEJ joint) and the free span laser weld joint (LWJ) separately have load bearing characteristics which exceed the most limiting RG 1.21 loading scenario. 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 sleeves restores the integrity of the primary pressure boundary and the tube is leaktight. All welds must be produced a minimum distance from any detected tube degradation as described in WCAP-14596. The structural analysis and mechanical performance of the sleeves are based on installation in the hot leg of the steam generators.

Mechanical integrity tests for the 11/16" LWS will be provided upon completion. It is expected that the 11/16" LWS design will perform similar to the 3/4" and 7/8" designs.

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 WCAP-14596. Historically at Callaway, only the top of tubesheet region has experienced stress corrosion cracking. The sleeve free span (structural) weld joints are located several inches above this area, and should not be affected by any previously identified degradation mechanism which resulted in the tube's removal from service. The analysis also supports sleeve installation in a separated tube, therefore, the extent of the originally identified degradation indication should not affect sleeve installation. Conformance to surface finish requirements for the lower joint helps 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.

EVALUATION

The proposed changes to the TS do not involve an Unreviewed Safety Question because operation of Callaway Plant with this change would not:

1. Increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report.

The elevated tubesheet LWS configuration has been designed and analyzed in accordance with the requirements of the ASME Code. The applied stresses and fatigue usage for the sleeve and weld are bounded by the limits established in the ASME Code. ASME Code minimum material property values are used for the structural and plugging limit analysis. Ultrasonic inspection is used to verify that minimum weld fusion zone thickness are produced. Mechanical testing has shown that the individual joint structural strength of Alloy 690 LWS under normal, upset and faulted conditions provides margin to the acceptance limits. These acceptance limits bound the most limiting (3 times normal operating pressure differential) burst margin recommended by RG 1.121. Therefore, each individual joint provides for structural integrity exceeding RG recommendations. Leakage testing for 7/8" and 3/4" tube sleeves has demonstrated that no unacceptable levels of primary to secondary leakage are expected during any plant condition, including the case where the seal weld is not produced in the lower joint of the tubesheet sleeve. Similar tests of 11/16" tube sleeves will be completed prior to Refuel 8.

The sleeve minimum acceptable wall thickness (used for developing the depth-based plugging limit for the sleeve) is determined using the guidance of Regulatory Guide 1.121 and the pressure stress equation of Section III of the ASME Code. 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 (design) conditions. A bounding set of design and transient loading input conditions was used for the minimum wall thickness evaluation in the generic evaluation. Evaluation of the minimum acceptable wall thickness for normal, upset and postulated accident condition loading per the ASME Code indicates these conditions are bounded by the design condition requirement minimum wall thickness.

A bounding tube wall degradation growth rate per cycle and an eddy current uncertainty has been assumed for determining the sleeve TS plugging limit. The sleeve wall degradation extent determined by eddy current examination, which would require plugging sleeved tubes, is developed using the guidance of RG 1.121 and is defined in WCAP-14596 to be 39% throughwall of the sleeve nominal wall thickness.

The consequences of failure of the sleeve joint are bounded by the current steam generator tube rupture analysis included in the Callaway FSAR. Due to the slight reduction in diameter caused by the sleeve wall thickness, primary coolant release rates would be slightly less than assumed for the steam generator tube rupture analysis (depending on the break location), and therefore, would result in lower total primary fluid mass release to the secondary system.

The proposed change does not adversely impact any other previously evaluated design basis accident or the results of LOCA and non-LOCA accident analyses for the current TS minimum reactor coolant system flow rate. The results of the analyses and testing demonstrate that the sleeve assembly is an acceptable means of maintaining tube integrity. Furthermore, per Regulatory Guide 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes" 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.

Corrosion testing of laser welded sleeve joints indicates that the corrosion resistance (relative to roll transition control samples) can be increased by greater than a factor of ten with the application of a post weld heat treatment. All free span laser welds will receive a post weld heat treatment. Therefore, rapid corrosion degradation of the free span laser weld joint region is not expected. Recently performed corrosion testing of LWS joints in locked (at the first TSP structure) tube conditions indicates that with PWHT, the stress corrosion cracking initiation potential in the weld region of the parent tube is reduced and the cracking resistance is enhanced. Similar test results and conclusions would be expected for Callaway based on the similarity of designs and expected tube far field residual stresses.

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 LWS will not increase the probability or consequences of an accident previously evaluated.

2. Create the possibility for an accident or malfunction of equipment of a different type than any previously evaluated in the Safety Analysis Report.

Sleeving will not adversely affect any plant component. Stress and fatigue analysis of the repair has shown that the ASME Code and Regulatory Guide 1.121 criteria are not exceeded. Implementation of LWS 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 each sleeve joint 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 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.

Implementation of LWS will reduce the potential for primary to secondary leakage during a postulated steam line break while not significantly impacting available primary coolant flow area in the event of a LOCA. By effectively isolating degraded areas of the tube 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 reduces primary coolant flow, the reduction is far below that caused by plugging. Therefore, far greater primary coolant flow area is maintained through sleeving versus plugging.

3. Reduce the margin of safety as defined in the basis for any Technical Specification.

The LWS repair of degraded steam generator tubes has been shown by analysis to restore the integrity of the tube bundle consistent with its original design basis condition, i.e., tube/sleeve operational and faulted condition stresses are bounded by the ASME Code requirements and the repaired tubes are leaktight. The safety factors used in the design of sleeves for the repair of degraded tubes are consistent with the safety factors in the ASME 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. Similar tests of 11/16" tube sleeves will be completed prior to Refuel 8. 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-14596.

In addition, since the installed sleeve represents 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 has been reviewed based on the installation of sleeves up to the level of steam generator tube plugging coincident with the minimum reactor flow rate and the Callaway Safety Analysis.

Provisional requirements cited in other NRC Safety Evaluation Reports addressing the implementation of sleeving have required the reduction of the individual steam generator normal operation primary to secondary leakage limit from 500 to 150 gpd. Consistent with these evaluations, Union Electric will reduce the per steam generator leak rate limit of 500 gpd in TS 3.4.6.2.c to 150 gpd. The establishment of this leakage limit at 150 gpd provides additional safety margin.

Finally, Union Electric will reduce the tube plugging limit from 48% through wall to 40% through wall to be consistent with NUREG-1431. The establishment of the plugging limit at 40% through wall provides additional safety margin.

CONCLUSION

Based on the preceding analysis it is concluded that operation of the Callaway Plant following the installation of Alloy 690 laser welded sleeves within the tubesheet region of the steam generators 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, or reduce any margins to plant safety. Therefore, given the above discussions as well as those presented in the Significant Hazards Consideration, the proposed change does not adversely affect or endanger the health or safety of the general public or involve an Unreviewed Safety Question.

ULNRC-3357

ATTACHMENT 4

SIGNIFICANT HAZARDS EVALUATION

SIGNIFICANT HAZARDS EVALUATION

INTRODUCTION

This proposed amendment revises the Surveillance Requirements of Technical Specification (TS) 3/4.4.5 "Steam Generators" and 3.4.6.2 "Operational Leakage" and associated Bases as appropriate, to address the installation of laser welded tube sleeves (LWSs) in the Callaway Plant steam generators.

This license amendment request revises TS 3/4.4.5, 3.4.6.2, and associated Bases to include:

a. Laser welded sleeving per Westinghouse WCAP-14596 as an approved tube repair method,

b. The associated sleeve wall depth-based plugging limit value and inspection requirements,

c. Reduction of the tube plugging limit from 48% to 40% through wall (of the nominal tube wall thickness) to be consistent with NUREG-1431, "Standard Technical Specifications for Westinghouse Plants", and,

d. The reduction of the primary to secondary normal operational leakage limit from 500 to 150 gpd per steam generator.

Currently, tubes with indications of degradation in excess of the plugging criteria are removed from service by plugging. Removal of a tube from service results in a reduction of reactor coolant flow through the steam generator. This small reduction in flow can impact 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 via sleeving maintains the tube heat transfer area and results in a much smaller RCS flow reduction. Therefore, the use of sleeving in lieu of plugging helps to assure that minimum flow rates are maintained in excess of that required for operation at full power. Any combination of sleeving and plugging, up to a level such that the effect will not reduce the minimum reactor coolant flow rate to below the current TS limit or below the plugging limits analyzed in the Callaway Safety Analysis Report is acceptable. The sleeve/plug equivalency results are contained in WCAP-14596.

BACKGROUND

Callaway has Westinghouse Model F steam generators which utilize 11/16" OD x 0.040" nominal wall thickness tubes. The first ten rows of tubes at Callaway are thermally treated Alloy 600 (1326 tubes), while the remainder of the tubes (4300 tubes) are mill annealed Alloy 600. The Callaway tubes are hydraulically expanded within the tubesheet region. The pressure utilized for the expansion process is designed to provide a radial preload between the tube and tubesheet such that the tube to tubesheet gap is completely reduced during all conditions. The current Callaway TS require steam generator tubes with eddy current indications of 48% through wall or greater to be removed from service.

This amendment proposes to permit the installation of Alloy 690 elevated LWSs within the tubesheet area of the steam generators at the Callaway Plant. This steam generator tube repair method secures to the original tube a short length of tubing with an outer diameter slightly smaller than the inside diameter of the parent tube, thereby spanning the degraded area. LWS has been determined to be an effective repair method for degraded steam generator tubes and has been licensed and installed at plants with both 3/4" and 7/8" outside diameter steam generator tubes. Laser welded sleeves are installed by first hydraulically expanding the sleeve against the tube to bring the tube and sleeve into intimate contact for weld quality purposes. Minimal levels of tube deformation are introduced in the hydraulic expansion process. The upper joint (located above the tubesheet in the tube free span region) is then welded, followed by a roll expansion at the lower joint, which is located within the tubesheet region. The bottom edge of the elevated tubesheet sleeve will be located approximately 6 inches below the top of the tubesheet (TTS) and extend to approximately 6 inches above the TTS. The only expected location of indications at Callaway is the top of tubesheet expansion transition region.

EVALUATION

Generic Structural Assessment

The LWS described in WCAP-14596 has been designed to Section III, Subsection NB-3300, of the 1989 Edition of the ASME Code. Fatigue and stress analyses of the sleeved tube assemblies have been completed in accordance with the requirements of Section III, Subsection NB-3200 of the 1989 Edition of the ASME Code. Both the sleeve and weld are evaluated. The results of the primary stress intensity evaluation, primary plus secondary stress intensity range evaluation and fatigue evaluation indicate that the ASME Code allowable limits are not exceeded. That is, stress intensities are bounded by the Code minimum limits for SB-163 (Alloy 690) material and cumulative fatigue usage is less than 1.0. Therefore, the design of the sleeve pressure boundary meets the design objectives of the original tubing.

Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes" and the ASME Code are used to develop the plugging limit of the sleeve 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. No credit for the presence of the parent tube behind the sleeve is assumed when performing the minimum wall/burst evaluation.

The sleeve and weld structural analysis utilizes a generic set of design and transient loading inputs which are intended to bound all plants with Westinghouse Model F steam generators. The temperature and pressure variances used in the assumed operating conditions and generic transients are bounding.

An ultrasonic inspection of the free span welds is performed prior to placing the sleeve in service 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. In addition, a fatigue analysis was performed for the assembly with 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 bounding for an operating plant. The results of the fatigue analysis indicate acceptable usage factors for the entire range of permitted weld thickness.

Leakage Assessment

The LWS joint is an autogenous tube-to-sleeve weld. Leakage testing of 3/4" and 7/8" LWS assemblies under conditions considered to be more severe than expected during all operating plant conditions has shown that the laser welded joint does not introduce additional primary to secondary leakage during a postulated steam line break event. LWS tube assemblies were subjected to thermal and fatigue cycling and then leak tested at pressure differentials of up to 3110 psi, which far exceeds the expected maximum feed line break or steam line break pressure differential. Leakage testing has also shown that the non-welded elevated tubesheet sleeve lower joint is essentially leaktight during all plant conditions. 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. Essentially no leakage was detected in any non-welded tubesheet sleeve lower joint at 600°F in hydraulically expanded tubes after both thermal and fatigue loading. Primary to secondary leakage through non-welded tubesheet sleeve lower joints would be expected to be negligible at SLB, normal operating, and at 0% power conditions, and therefore will not significantly contribute to offsite doses in the event of postulated SLB. Due to time limitations, leakage testing of LWS specimens for 11/16" tubes has not been performed. These tests are scheduled to be completed before the Callaway Plant Refuel 8 outage, which is currently scheduled for Fall 1996. The leakage test results will be provided to NRC upon completion. The testing program for 11/16" tubes will be consistent with the previously performed LWS leakage tests.

Corrosion Assessment

Thermally treated Alloy 600 and Alloy 690 sleeved tube assemblies have performed well historically with regard to corrosion. Accelerated corrosion test results show the free span laser welded joint with post weld heat treatment (PWHT) 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. These factors suggest postulated sleeve degradation, even in a non-heat treated condition, would occur at a relatively slow rate, and be detected by routine eddy current inspection. The heat treatment process is designed to achieve sufficient stress reduction such that rapid crack initiation and propagation in the joint is not expected. Test data indicates acceptable corrosion resistance at temperatures as low as 1250°F. On the tube ID surface, which is where the weld cooling stresses are concentrated, temperatures during heat treatment will exceed the OD temperature, thereby providing a more effective heat treatment in this area. PWHT parent tube far field stresses can affect the susceptibility of the weld to cracking. The steam generator geometry, fixity conditions at the tube supports, and PWHT temperature all influence the resultant far field stresses. In order to develop significant post-process far field residual stresses, it must be assumed that the tube is locked at the tube support plates (TSPs). The post-process tube far field residual stresses (assuming the tubes are locked at the first TSP) for a heat treatment temperature of 1350°F at Callaway Plant are similar to other plants.

Mechanical Integrity Assessment

Mechanical testing of 3/4" and 7/8" laser weld and hybrid expansion joint (HEJ) sleeves indicates that the axial load bearing capability of these joints individually exceeds the most limiting pressure end cap loading established by RG 1.121. Both the lower joint (hydraulic expansion plus roll expansion, commonly known as an HEJ joint) and the free span laser weld joint (LWJ) separately have load bearing characteristics which exceed the most limiting RG 1.21 loading scenario. 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 TSP sleeves and/or tubesheet sleeves restores the integrity of the primary pressure boundary and the tube is leaktight. All welds must be produced a minimum distance from any detected tube degradation as described in WCAP-14596. The structural analysis and mechanical performance of the sleeves are based on installation in the hot leg of the steam generators.

Mechanical integrity tests for the 11/16" LWS will be provided upon completion. It is expected that the 11/16" LWS design will perform similar to the 3/4" and 7/8" designs.

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 WCAP-14596. Historically at Callaway, only the top of tubesheet region has experienced stress corrosion cracking. The sleeve free span (structural) weld joints are located several inches above this area, and should not be affected by any previously identified degradation mechanism which resulted in the tube's removal from service. The analysis also supports sleeve installation in a separated tube, therefore, the extent of the originally identified degradation indication should not affect sleeve installation. Conformance to surface finish requirements for the lower joint helps 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.

EVALUATION

The proposed changes to the TS do not involve a significant hazards consideration because operation of Callaway Plant with these changes would not:

1. Involve a significant increase in the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report.

The elevated tubesheet LWS configuration has been designed and analyzed in accordance with the requirements of the ASME Code. The applied stresses and fatigue usage for the sleeve and weld are bounded by the limits established in the ASME Code. ASME Code minimum material property values are used for the structural and plugging limit analysis. Ultrasonic inspection is used to verify that minimum weld fusion zone thickness are produced. Mechanical testing has shown that the individual joint structural strength of Alloy 690 LWS under normal, upset and faulted conditions provides margin to the acceptance limits. These acceptance limits bound the most limiting (3 times normal operating pressure differential) burst margin recommended by RG 1.121. Therefore, each individual joint provides for structural integrity exceeding RG recommendations. Leakage testing for 7/8" and 3/4" tube sleeves has demonstrated that no unacceptable levels of primary to secondary leakage are expected during any plant condition, including the case where the seal weld is not produced in the lower joint of the tubesheet sleeve. Similar tests of 11/16" sleeves will be completed prior to Refuel 8.

The sleeve minimum acceptable wall thickness (used for developing the depth-based plugging limit for the sleeve) is determined using the guidance of Regulatory Guide 1.121 and the pressure stress equation of Section III of the ASME Code. 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 (design) conditions. A bounding set of design and transient loading input conditions was used for the minimum wall thickness evaluation in the generic evaluation. Evaluation of the minimum acceptable wall thickness for normal, upset and postulated accident condition loading per the ASME Code indicates these conditions are bounded by the design condition requirement minimum wall thickness.

A bounding tube wall degradation growth rate per cycle and an eddy current uncertainty has been assumed for determining the sleeve TS plugging limit. The sleeve wall degradation extent determined by eddy current examination, which would require plugging sleeved tubes, is developed using the guidance of RG 1.121 and is defined in WCAP-14596 to be 39% throughwall of the sleeve nominal wall thickness.

The consequences of failure of the sleeve joint are bounded by the current steam generator tube rupture analysis included in the Callaway FSAR. Due to the slight reduction in diameter caused by the sleeve wall thickness, primary coolant release rates would be slightly less than assumed for the steam generator tube rupture analysis (depending on the break location), and therefore, would result in lower total primary fluid mass release to the secondary system.

The proposed change does not adversely impact any other previously evaluated design basis accident or the results of LOCA and non-LOCA accident analyses for the current TS minimum reactor coolant system flow rate. The results of the analyses and testing demonstrate that the sleeve assembly is an acceptable means of maintaining tube integrity. Furthermore, per Regulatory Guide 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes" 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.

Corrosion testing of laser welded sleeve joints indicates that the corrosion resistance (relative to roll transition control samples) can be increased by greater than a factor of ten with the application of a post weld heat treatment. All free span laser welds will receive a post weld heat treatment. Therefore, rapid corrosion degradation of the free span laser weld joint region is not expected. Recently performed corrosion testing of LWS joints in locked (at the first TSP structure) tube conditions indicates that with PWHT, the stress corrosion cracking initiation potential in the weld region of the parent tube is reduced and the cracking resistance is enhanced. Similar test results and conclusions would be expected for Callaway based on the similarity of designs and expected tube far field residual stresses.

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 LWS will not increase the probability or consequences of an accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any previously evaluated in the Safety Analysis Report.

Sleeving will not adversely affect any plant component. Stress and fatigue analysis of the repair has shown that the ASME Code and Regulatory Guide 1.121 criteria are not exceeded. Implementation of LWS 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 each sleeve joint 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 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.

Implementation of LWS will reduce the potential for primary to secondary leakage during a postulated steam line break while not significantly impacting available primary coolant flow area in the event of a LOCA. By effectively isolating degraded areas of the tube 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 reduces primary coolant flow, the reduction is far below that caused by plugging. Therefore, far greater primary coolant flow area is maintained through sleeving versus plugging.

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

The LWS repair of degraded steam generator tubes has been shown by analysis to restore the integrity of the tube bundle consistent with its original design basis condition, i.e., tube/sleeve operational and faulted condition stresses are bounded by the ASME Code requirements and the repaired tubes are leaktight. The safety factors used in the design of sleeves for the repair of degraded tubes are consistent with the safety factors in the ASME 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. Similar tests of 11/16" sleeves will be completed prior to Refuel 8. 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-14596.

In addition, since the installed sleeve represents 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 has been reviewed based on the installation of sleeves up to the level of steam generator tube plugging coincident with the minimum reactor flow rate and the Callaway Safety Analysis.

Provisional requirements cited in other NRC Safety Evaluation Reports addressing the implementation of sleeving have required the reduction of the individual steam generator normal operation primary to secondary leakage limit from 500 to 150 gpd. Consistent with these evaluations, Union Electric will reduce the per steam generator leak rate limit of 500 gpd in TS 3.4.6.2.c to 150 gpd. The establishment of this leakage limit at 150 gpd provides additional safety margin.

Finally, Union Electric will reduce the tube plugging limit from 48% through wall to 40% through wall to be consistent with NUREG-1431. The establishment of the plugging limit at 40% through wall provides additional safety margin.

CONCLUSION

Given the above discussions, the proposed change does not adversely affect or endanger the health or safety of the general public or involve a significant hazards consideration.

ULNRC-3357

ATTACHMENT 5

ENVIRONMENTAL CONSIDERATION

ENVIRONMENTAL CONSIDERATION

This proposed amendment revises the Surveillance Requirements of Technical Specification (TS) 3/4.4.5 "Steam Generators" and 3.4.6.2 "Operational Leakage" and associated Bases as appropriate, to address the installation of laser welded tube sleeves (LWSs) in the Callaway Plant steam generators.

The proposed amendment involves changes with respect to the use of facility components located within the restricted area, as defined in 10 CFR 20, and changes surveillance requirements. Union Electric has determined that the proposed amendment does not involve:

(1) A significant hazard consideration, as discussed in Attachment 4 of this amendment application;

(2) A significant change in the types or significant increase in the amounts of any effluents that may be released offsite;

(3) A significant increase in individual or cumulative occupational radiation exposure, as discussed in Attachment 3 of this amendment application.

Accordingly, the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or enviromental assessment need be prepared in connection with the issuance of this amendment.

ULNRC-3357

ATTACHMENT 8

WESTINGHOUSE AUTHORIZATION LETTER, CAW-96-940, AND
ACCOMPANYING AFFIDAVIT, DATED APRIL 3, 1996