## 4.2 ASME CODE CLASS IN-SERVICE INSPECTION AND TESTING

**APPLICABILITY**

Applies to in-service structural surveillance of the ASME Code Class components and supports and functional testing of pumps and valves.

**OBJECTIVE**

To assure the continued integrity and operational readiness of ASME Code Class 1, 2 and 3 components.

**SPECIFICATION**

a. ASME Code Class 1, 2 and 3 Components and Supports

1. In-service inspection of ASME Code Class 1, Class 2 and Class 3 components and supports shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a(g), except where relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). The testing and surveillance of shock suppressors (snubbers) is detailed in TS 3.14 and TS 4.14.

2. In-service testing of ASME Code Class 1, Class 2 and Class 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a(f), except where relief has been granted by the Commission pursuant to 10 CFR 50.55a(f)(6)(i).

3. Surveillance testing of pressure isolation valves:

   a. Periodic leakage testing[1] on each valve listed in Table TS 3.1-2 shall be accomplished prior to entering the OPERATING mode after every time the plant is placed in the COLD SHUTDOWN condition for refueling, after each time the plant is placed in a COLD SHUTDOWN condition for 72 hours if testing has not been accomplished in the preceding 9 months, and prior to returning the valve to service after maintenance, repair, or replacement work is performed.

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[1]To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

b. Whenever integrity of a pressure isolation valve listed in Table TS 3.1-2 cannot be demonstrated, the integrity of the remaining pressure isolation valve in each high pressure line having a leaking valve shall be determined and recorded daily. In addition, the position of the other closed valve located in the high pressure piping shall be recorded daily.

b. Steam Generator Tubes

Examinations of the steam generator tubes shall be in accordance with the in-service inspection program described herein. The following terms are defined to clarify the requirements of the inspection program.

Imperfection is an exception to the dimension, finish, or contour required by drawing or specification.

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

% Degradation is an estimated % of the tube wall thickness affected or removed by degradation.

Degraded Tube means a tube contains an imperfection $\geq$ 20% of the nominal wall thickness caused by degradation.

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

Tube Inspection means an inspection of the steam generator tube from the point of entry (e.g., hot leg side) completely around the U-bend to the top support of the opposite leg (cold leg).

Tube is the Reactor Coolant System pressure boundary past the hot leg side of the tubesheet and before the cold leg side of the tubesheet.

Plugged Tube is a tube intentionally removed from service by plugging in the hot and cold legs because it is defective, or because its continued integrity could not be assured.

Regaired Tube is a tube that has been modified to allow continued service consistent with plant Technical Specifications regarding allowable tube wall degradation, or to prevent further tube wall degradation. A tube without repairs is a nonrepaired tube. This definition does not apply to the portion of the tube below the F* or EF* distance provided the tube is not degraded (i.e., no detectable degradation permitted) within the F* distance for F* tubes and within the EF* distance for EF* tubes.

Laser Weld Repaired Sleeved Tube is a tube with a Westinghouse mechanical hybrid expansion joint sleeve that has been returned to operable status by use of a laser welded repair process.

F* Distance is the distance of the expanded portion of a tube which provides a sufficient length of undegraded tube expansion to resist pullout of the tube from the tubesheet. The F* distance is equal to 1.12 inches (plus an allowance for NDE uncertainty) and is measured downward from the bottom of the uppermost roll transition. The F* distance applies to roll expanded regions below the midpoint of the tubesheet.

F* Tube is a tube with degradation below the F* distance, equal to or greater than 50%, and has no indications of degradation within the F* distance.

EF* Distance is the distance of the expanded portion of a tube which provides a sufficient length of undegraded tube expansion to resist pullout of the tube from the tubesheet. The EF* distance is equal to 1.44 inches (plus an allowance for NDE uncertainty) and is measured downward from the bottom of the uppermost roll transition. The EF* distance applies to roll expanded regions above the midpoint of the tubesheet.

EF* Tube is a tube with degradation below the EF* distance, equal to or greater than 50%, and has no degradation within the EF* distance.

1. Steam Generator Sample Selection and Inspection

   The in-service inspection may be limited to one steam generator on a rotating schedule encompassing the number of tubes determined in TS 4.2.b.2.a provided the previous inspections indicated that the two steam generators are performing in a like manner.

2. Steam Generator Tube Sample Selection and Inspection

   The tubes selected for each in-service inspection shall:

   a. Include at least 3% of the total number of nonrepaired tubes, in both steam generators, and 20% of the total number of repaired tubes in both steam generators. The tubes selected for these inspections shall be selected on a random basis except as noted below and in TS 4.2.b.2.b.

      Indications left in service as a result of application of the tube support plate voltage-based repair criteria shall be inspected by bobbin coil probe during all future REFUELING outages.

   b. Concentrate the inspection by selection of at least 50% of the tubes to be inspected from critical areas where experience in similar plants with similar water chemistry indicates higher potential for degradation.

c. Include the inspection of all non-plugged tubes which previous inspections revealed in excess of 20% degradation. The previously degraded tubes need only be inspected about the area of previous degradation indication if their inspection is not employed to satisfy 4.2.b.2.a and 4.2.b.2.b above.

Implementation of the steam generator tube/tube support plate repair criteria requires a 100% bobbin coil inspection for hot leg and cold leg tube support plate intersections down to the lowest cold leg tube support plate with known outside diameter stress corrosion cracking (ODSCC) indications. The determination of the lowest cold-leg tube support plate intersections having ODSCC indications shall be based on the performance of at least a 20% random sampling of tubes inspected over their full length.

d. In addition to the sample required in 4.2.b.2.a through 4.2.b.2.c, all tubes which have had the F*, or EF*, criteria applied will be inspected each outage in the uppermost tubesheet roll expanded region. These tubes may be excluded from 4.2.b.2.c provided the only previous wall penetration of >20% was located below the F* or EF* distance. F* and EF* tubes will be inspected for a minimum of 2 inches below the bottom of the uppermost roll transition. The results of F* or EF* tube inspections are not to be used as a basis for additional inspection per Table TS 4.2-2 or Table TS 4.2-3.

e. In addition to the sample required in 4.2.b.2.a through 4.2.b.2.c, all laser weld repaired sleeved tubes will be inspected at the first in-service inspection following the repair. Subsequent inspections will include a minimum sample size consistent with 4.2.b.2.a.

f. The second and third sample inspections during each in-service inspection may be less than the full length of each tube by concentrating the inspection on those areas of the tubesheet array and on those portions of the tubes where tubes with imperfections were previously found.

g. If a tube does not permit the passage of the eddy current inspection probe the entire length and through the U-bend, this shall be recorded and an adjacent tube shall be inspected. The tube which did not allow passage of the eddy current probe shall be considered degraded.

The results of each sample inspection shall be classified into one of the following three categories. For non-repaired tubes, actions shall be taken as described in Table 4.2-2. For repaired tubes, actions shall be taken as described in Table 4.2-3.

| 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 must exhibit significant (>10%) further wall penetrations to be included in the above percentage calculations.

3. Inspection Frequencies

The above required in-service inspections of steam generator tubes shall be performed at the following frequencies:

a. In-service inspections shall be performed at refueling intervals not more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the pre-service 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 in-service inspection of a steam generator conducted in accordance with Table 4.2-2 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 a subsequent inspection meets the conditions specified in 4.2.b.3.a and the interval can be extended to a 40-month period.

c. Additional, unscheduled in-service inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.2-2 during the shutdown subsequent to any of the following conditions:

   1. Primary-to-secondary tube leaks (not including leaks originating from tube-to-tubesheet welds) in excess of the limits of TS 3.1.d and TS 3.4.a.1.C or

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

3. A loss-of-coolant accident requiring actuation of the engineering safeguards, where the cooldown rate of the Reactor Coolant System exceeded 100°F/hr, or

4. A main steam line or feedwater line break, where the cooldown rate of the Reactor Coolant System exceeded 100°F/hr.

d. If the type of steam generator chemistry treatment is changed significantly, the steam generators shall be inspected at the next outage of sufficient duration following 3 months of power operation since the change.

4. Plugging Limit Criteria

The following criteria apply independently to tube and sleeve wall degradation except as specified in TS 4.2.b.5 for the tube support plate intersections for which voltage-based plugging criteria are applied or for degradation except as specified in TS 4.2.b.6 for tubesheet crevice region in which the F* and EF* criteria is applied.

a. Any tube which, upon inspection, exhibits tube wall degradation of 50% or more shall be plugged or repaired prior to returning the steam generator to service. If significant general tube thinning occurs, this criterion will be reduced to 40% wall degradation. Tube repair shall be in accordance with the methods described in WCAP-14685 Revision 1, "Laser Welded Repair of HEJ Sleeves for Kewaunee Nuclear Power Plant," WCAP-11643, "Kewaunee Steam Generator Sleeving Report (Mechanical Sleeves)," CEN-413-P, "Kewaunee Steam Generator Tube Repair Using Leak Tight Sleeves," or WCAP-13088, Revision 3, "Westinghouse Series 44 and 51 Steam Generator Generic Sleeving Report."

b. Any Westinghouse mechanical hybrid expansion joint (HEJ) sleeve which, upon inspection, exhibits wall degradation of 24% or more shall be plugged or repaired prior to returning the steam generator to service. For disposition of parent tube indications (PTI), the following requirements will apply:

1. HEJ sleeved tubes with circumferential indications located within the upper hardroll lower transition shall be inspected with a non-destructive examination (NDE) technique capable of measuring the sleeve ID difference between the sleeve hardroll peak diameter, and the sleeve ID at the elevation of the PTI. If this diameter change is $\geq$ 0.003" (plus an allowance for NDE uncertainty), the indication may remain in service provided the faulted loop steam line break (SLB) leakage limit from all sources is not exceeded. A SLB leakage allowance of 0.025 gpm shall be assumed for each indication left in service regardless of length or depth. For tubes where the diameter difference is > 0.013", SLB leakage can be neglected.

2. HEJ sleeved tubes with a sleeve ID difference of < 0.003" (plus an allowance for NDE uncertainty) between the sleeve ID hardroll peak diameter and sleeve ID at the elevation of the PTI shall be plugged or repaired prior to returning the steam generator to service.

3. HEJ sleeved tubes with axial indications located within the parent tube pressure boundary as defined on Figure TS 4.2-1 shall be plugged or repaired prior to returning the steam generator to service.

4. HEJ sleeved tubes with parent tube indications located outside of the parent tube pressure boundary as defined on Figure TS 4.2-1 may remain in service.

c. Any Combustion Engineering leak tight sleeve which, upon inspection, exhibits wall degradation of 40% or more shall be plugged prior to returning the steam generator to service. This plugging limit applies to the sleeve up to and including the weld region.

d. Any Westinghouse laser welded sleeve which, upon inspection, exhibits wall degradation of 25% or more, shall be plugged prior to returning the steam generator to service. This plugging limit applies to the sleeve up to and including the weld.

5. <u>Tube Support Plate Plugging Limit</u>

The following criteria are used for the disposition of a steam generator tube for continued service that is experiencing predominantly axially oriented outside diameter stress corrosion cracking confined within the thickness of the tube support plates. At tube support plate intersection, the repair limit is based on maintaining steam generator tube serviceability as described below:

a. Degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with bobbin voltage $\leq$ 2.0 volts will be allowed to remain in service.

b. Degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage > 2.0 volts will be repaired or plugged except as noted in TS 4.2.b.5.c below.

c. Indications of potential degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage > 2.0 volts but ≤ the upper voltage repair limit, may remain in service if a rotating pancake coil inspection does not detect degradation. Indications of outside diameter stress corrosion cracking degradation with a bobbin voltage > the upper voltage repair limit will be plugged or repaired.

d. If an unscheduled mid-cycle inspection is performed, the following repair limits apply instead of TS 4.2.b.5.a, b and c. The mid-cycle repair limits are determined from the following equation:

$$V_{MURL} = \frac{V_{SL}}{1.0 + NDE + Gr\left(\frac{CL - \Delta t}{CL}\right)}$$

$$V_{MLRL} = V_{MURL} - (V_{URL} - 2.0)\left(\frac{CL - \Delta t}{CL}\right)$$

Where:

$V_{MURL}$ = mid-cycle upper voltage repair limit based on time into cycle
$V_{SL}$ = structural limit voltage
NDE = 95% cumulative probability allowance for NDE uncertainty
Gr = average growth rate per cycle length
CL = cycle length (time between scheduled inspections)
$\Delta t$ = length of time since last scheduled inspection during which $V_{URL}$ and $V_{LRL}$ were implemented
$V_{MLRL}$ = mid-cycle lower voltage repair limit based on $V_{MURL}$ and time into cycle
$V_{URL}$ = upper voltage repair limit

Implementation of these mid-cycle repair limits should follow the same approach as in TS 4.2.b.5.a, b and c.

NOTE: The upper voltage repair limit is calculated according to the methodology in Generic Letter 95-05 as supplemented.

6. F* and EF* Tubesheet Crevice Region Plugging Criteria

The following criteria are to be used for disposition or repair of steam generator tubes experiencing degradation in the tubesheet crevice region.

a. Tubes with indications of degradation within the roll expanded region below the midpoint of the tubesheet may remain in service provided the distance from the bottom of the uppermost roll transition to the tip of the crack is greater than 1.12" (plus an allowance for NDE uncertainty). This criteria is called the F* criteria and applies to the factory roll expansion, or to additional roll expansions formed as an extension of the original roll. Any degradation existing below the F* (plus an allowance for NDE uncertainty) is acceptable for continued service.

b. Indications of degradation not repairable by 4.2.b.6.a may be repaired using the EF* criteria. The EF* region is located a minimum of 4" below the top of the tubesheet, and is formed by an additional roll expansion of the tube in the originally unexpanded length. Tubes with indications of degradation within the EF* region may remain in service provided the distance from the bottom of the uppermost roll transition to the tip of the crack is greater than 1.44" (plus an allowance for NDE uncertainty). Any degradation existing below EF* (including uncertainty) is acceptable for continued service.

7. Reports

a. Following each in-service inspection of steam generator tubes, if there are any tubes requiring plugging or repairing, the number of tubes plugged or repaired shall be reported to the Commission within 30 days. This report shall include the tubes for which the F* or EF* criteria were applied.

b. The results of the steam generator tube in-service inspection shall be included in the Annual Operating Report for the period in which this inspection was completed. This report shall include:

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of a degradation.

3. Identification of tubes plugged.

4. Identification of tubes repaired.

c. Results of a steam generator tube inspection which fall into Category C-3 require prompt (within 4 hours) notification of the Commission consistent with 10 CFR 50.72(b)(2)(i). A written follow up report shall be submitted to the Commission consistent with Specification 4.2.b.7.a, using the Licensee Event Report System to satisfy the intent of 10 CFR 50.73(a)(2)(ii).

d. For implementation of the voltage-based repair criteria to tube support plate intersections, notify the NRC staff prior to returning the steam generators to service should any of the following conditions arise:

1. If estimated leakage based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds the leak limit (determined from the licensing basis dose calculation for the postulated main steamline break) for the next operating cycle.

2. If circumferential crack-like indications are detected at the tube support plate intersections.

3. If indications are identified that extend beyond the confines the tube support plate.

4. If indications are identified at the tube support plate elevations that are attributable to primary water stress corrosion cracking.

5. If the calculated conditional burst probability based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds $1 \times 10^{-2}$, notify the NRC and provide an assessment of the safety significance of the occurrence.

**BASIS**

The plant was not specifically designed to meet the requirements of Section XI of the ASME Code; therefore, 100% compliance may not be feasible or practical. However, access for in-service inspection was considered during the design and modifications have been made where practical to make provisions for maximum access within the limits of the current plant design. Where practical, the inspection of ASME Code Class 1, Class 2 and Class 3 components is performed in accordance with Section XI of the ASME Code. If a code required inspection is impractical, a request for a deviation from the requirement is submitted to the Commission for approval.

The basis for surveillance testing of the Reactor Coolant System pressure isolation valves identified in Table TS 3.1-2 is contained within "Order for Modification of License" dated April 20, 1981.

Technical Specification 4.2.b

These Technical Specifications provide the inspection and repair/plugging requirements for the steam generator tubes at the Kewaunee Nuclear Power Plant. Fulfilling these specifications will assure the KNPP steam generator tubes are inspected and maintained in a manner consistent with current NRC regulations and guidelines including the General Design Criteria in 10 CFR Part 50, Appendix A.

General Design Criterion (GDC) 14 "Reactor Coolant Pressure Boundary," and GDC 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," require that the reactor coolant pressure boundary have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. Also, GDC 15, "Reactor Coolant System Design," requires that the Reactor Coolant System and associated auxiliary, control, and protection systems be designed with sufficient margin to ensure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences. Furthermore, GDC 32 "Inspection of Reactor Coolant System Pressure Boundary," requires that components that are part of the reactor coolant pressure boundary be designed to permit periodic inspection and testing of critical areas to assess their structural and leak tight integrity.

The NRC has developed guidance for steam generator tube inspections and maintenance including Regulatory Guides 1.83 and 1.121. Regulatory Guide 1.83, "In-service Inspection of Pressurized Water Reactor Steam Generator Tubes," forms the basis for many of the requirements in this section and should be consulted prior to any revisions. Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," defines the minimum wall thickness in a steam generator tube, and may be applied to tube sleeves in determining their minimum wall thickness.

Technical Specification 4.2.b.1

If the steam generators are shown to be performing in a like manner, it is appropriate to limit the inspection to one steam generator on a rotating schedule. Economic savings as well as reductions in personnel exposure and outage duration can be realized.

Technical Specification 4.2.b.2

Periodic inspection of the steam generator tubes allows evaluation of their service condition. As operational experience has become available it is evident that certain types of steam generators are susceptible to generic degradation mechanisms. Site specific steam generator tube degradation has also occurred throughout the industry. The inspection program at Kewaunee is designed to identify both generic and site specific tube degradation mechanisms.

Steam generator tube surveillance at Kewaunee is generally performed using eddy current techniques. Various methods of eddy current (EC) testing are used to inspect steam generator tubes for wall degradation. EC methods have improved considerably since Kewaunee began commercial operation in 1974. Single frequency EC testing with a single probe and X-Y plotter have evolved into multifrequency techniques with assorted probe types and sophisticated software to allow more accurate volumetric tube examinations. Profilometery techniques are also being developed which detect imperfections in a tube's original geometry. WPSC is committed to utilize advancing EC testing technology, as appropriate, to assure accurate determination of the steam generator tubes' service condition.

Technical Specification 4.2.b.3

Steam generator tube inspections are generally scheduled during refueling outages at the Kewaunee Nuclear Power Plant. The tubes scheduled for a given inspection are based upon their service condition determined during previous inspections, and operational experience from other plants with similar steam generators and water chemistry. Identification of degraded steam generator tube conditions results in augmentation of the inspection effort as well as increasing the frequency of subsequent inspections. In this manner, steam generator tube surveillance is consistent with service conditions.

There are several operational occurrences or transients that will require subsequent steam generator tube inspections. These inspections are required as a result of excessive primary-to-secondary leakage or transients imposing large mechanical and thermal stresses on the tubes.

Technical Specification 4.2.b.4

Steam generator tubes found with less than the minimum wall thickness criteria determined by analysis, as described in WCAP-7832[1][2], must either be repaired to be kept in service or removed from service by plugging.

Steam generator tube plugging is a common method of preventing primary-to-secondary steam generator tube leakage and has been utilized since the inception of PWR nuclear reactor plants. This method is relatively uncomplicated from a structural/mechanical standpoint as flow is cut off from the affected tube by plugging it in the hot and cold leg faces of the tubesheet.

To determine the basis for the sleeve plugging limit, the minimum sleeve wall thickness was calculated in accordance with the ASME Code and is consistent with Draft Regulatory Guide 1.121 (August 1976).

For the Westinghouse mechanical sleeves, the sleeve plugging limit of 31% is applied to the sleeve as shown on Figure TS 4.2-1. For the Combustion Engineering leak tight sleeves, a plugging limit of 40% is applied to the sleeve and weld region. The sleeve plugging limits allow for eddy current testing inaccuracies and continued operational degradation per Draft Regulatory Guide 1.121 (August 1976).

Repair by sleeving, or other methods, has been recognized as a viable alternative for isolating unacceptable tube degradation and preventing tube leakage. Sleeving isolates unacceptable degradation and extends the service life of the tube, and the steam generator. Tube repair, by sleeving in accordance with WCAP-11643[3], CEN-413-P[4], and WCAP-13088[5], has been evaluated and analyzed as acceptable. The Westinghouse mechanical hybrid expansion joint (HEJ) sleeve spans the degraded area of the parent tube in the tubesheet region. The sleeves are either 36", 30" or 27" to allow access permitted by channel head bowl geometry. The sleeve is hydraulically expanded and hard rolled into the parent tubing.

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[1]WCAP 7832, "Evaluation of Steam Generator Tube, Tube Sheet, and Divider Plate Under Combined LOCA Plus SSE Conditions."

[2]E. W. James, WPSC, to A. Schwencer, NRC, dated September 6, 1977.

[3]WCAP 11643, Kewaunee Steam Generator Sleeving Report, Revision 1, November 1988 (Proprietary).

[4]CEN-413-P, "Kewaunee Steam Generator Tube Repair Using Leak Tight Sleeves," January 1992 (Proprietary).

[5]WCAP 13088, Revision 3, "Westinghouse Series 44 and 51 Steam Generator Generic Sleeving Report," January 1994.

The pressure boundary for HEJ sleeves is shown on Figure TS 4.2-1. The pressure boundary used to disposition parent tube indications (PTIs) detected in the upper joint of HEJ sleeved tubes is discussed in WCAP-14641[6]. The pressure boundary will allow PTIs located such that there is a minimum diameter change of 0.003 inch (plus an allowance for NDE uncertainty) between the peak diameter of the sleeve hardroll, and the diameter at the elevation of the PTI, to remain in service. The 0.003 inch interference lip is derived from structural and leakage testing. When inspecting and dispositioning the PTIs, the acceptance criteria will be adjusted to account for measurement uncertainties associated with the technique used to measure the relative change in ID sleeve diameters. During field application, the PTI elevation will be measured by comparing the diameter reported at the peak amplitude of the flaw, and the diameter at the center of the plus point coil's field, and using the more conservative of the two diameters to perform the $\Delta D$ determination. Application of the pressure boundary for HEJ sleeved tubes provides allowance for leakage in a faulted loop during a postulated steam line break (SLB) event. A SLB leakage of 0.025 gpm is assumed for each applicable indication. Steam line break leakage from all sources must be calculated to be < 34 gpm in the faulted loop. Maintenance of the 34 gpm limit ensures off-site doses will remain within a small fraction of the 10 CFR Part 100 guidelines for a SLB.

Recent inspection information has indicated a potential for the parent tube behind the upper HEJ region to develop service induced degradation. For parent tube degradation within or below the upper HEJ hardroll lower transition, tube operability can be restored by fusing the sleeve and tube using a laser welding process, effectively isolating the degradation below the weld. The laser weld repair is performed identical to the initial installation of laser welded sleeves. The laser repair weld for degraded parent tubes with installed HEJ sleeves has been shown to meet the weld qualification, stress and fatigue requirements of the ASME code. All laser weld repaired HEJ sleeved tubes will receive a post weld stress relief at the weld location, consistent with the process described in WCAP-14685 Revision 1[7].

There are three types of Combustion Engineering leak tight sleeves. The first type, the straight tubesheet sleeve, spans the degraded area of the parent tube in the tubesheet crevice region. The sleeve is welded to the parent tube near each end. The second type of sleeve is the peripheral tubesheet sleeve. The sleeve is initially curved as part of the manufacturing process and straightened as part of the installation process. The third type of sleeve, the tube support plate sleeve, spans the degraded area of the tube support plate and is installed up to the sixth support plate. This sleeve is welded to the parent tube near each end of the sleeve.

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[6]WCAP-14641, "HEJ Sleeved Tube Structural Integrity Criteria: Diameter Interference at PTIs," April 1996.

[7]WCAP 14685 Revision 1, "Laser Welded Repair of HEJ Sleeves for Kewaunee Nuclear Power Plant," January 1997 (Proprietary).

Two types of Westinghouse laser welded sleeves can be installed, tube support plate sleeves and tubesheet sleeves.

The tube support plate sleeve is 12" long and spans the degraded area of the tube adjacent to the support plate intersection. The tube support plate sleeve is hydraulically expanded and laser welded at each end. The pressure boundary portion of the tube support plate sleeve is the weld and the sleeve section between the welds. Tubesheet sleeves extend from the tube end to above the top of the tubesheet. Standard and bowed or peripheral tubesheet sleeves can be installed. The upper or free span joint is hydraulically expanded and laser welded. The lower joint is hydraulically expanded and roll expanded. Standard tubesheet sleeves extend from 27" to 36" in length while bowed tubesheet sleeves extend from 30" to 36" in length. The pressure boundary portion of the tubesheet sleeve is the weld and below, down to the tubesheet primary face.

The hydraulic equivalency ratios for the application of normal operating, upset, and accident condition bounding analyses have been evaluated. Design, installation, testing, and inspection of steam generator tube sleeves requires substantially more engineering than plugging, as the tube remains in service. Because of this, the NRC has defined steam generator tube repair to be an Unreviewed Safety Question as described in 10 CFR 50.59(a)(2). As such, other tube repair methods will be submitted under 10 CFR 50.90; and in accordance with 10 CFR 50.91 and 92, the Commission will review the method, issue a significant hazards determination, and amend the facility license accordingly. A 90-day time frame for NRC review and approval is expected.

Technical Specification 4.2.b.5

The repair limit of tubes with degradation attributable to outside diameter stress corrosion cracking contained within the thickness of the tube support plates is conservatively based on the analysis documented in WCAP-12985, "Kewaunee Steam Generator Tube Plugging Criteria for ODSCC at Tube Support Plates" and EPRI Draft Report TR-100407, Rev.1, "PWR Steam Generator Tube Repair Limits - Technical Support Document for Outside Diameter Stress Corrosion Cracking at Tube Support Plates." Application of these criteria is based on limiting primary-to-secondary leakage during a steam line break to ensure the applicable 10 CFR Part 100 limits are not exceeded.

The voltage-based repair limits of TS 4.2.b.5 implement the guidance in Generic Letter 95-05 and are applicable only to Westinghouse-designed steam generators with outside diameter stress corrosion cracking (ODSCC) located at the tube-to-tube support plate intersections. The voltage-based repair limits are not applicable to other forms of tube degradation nor are they applicable to ODSCC that occurs at other locations within the steam generators. Additionally, the repair criteria apply only to indications where the degradation mechanism is predominantly axial ODSCC with no indications extending outside the thickness of the support plate. Refer to GL 95-05 for additional description of the degradation morphology.

Implementation of TS 4.2.b.5 requires a derivation of the voltage structural limit from the burst versus voltage empirical correlation and the subsequent derivation of the voltage repair limit from the structural limit (which is then implemented by this surveillance).

The voltage structural limit is the voltage from the burst pressure/bobbin voltage correlation, at the 95 percent prediction interval curve reduced to account for the lower 95/95 percent tolerance bound for tubing material properties at 650°F (i.e., the 95 percent LTL curve). The voltage structural limit must be adjusted downward to account for potential flaw growth during an operating interval and to account for NDE uncertainty. The upper voltage repair limit, $V_{URL}$, is determined from the structural voltage limit by applying the following equation:

$$V_{URL} = V_{SL} - V_{GR} - V_{NDE}$$

Where $V_{GR}$ represents the allowance for flaw growth between inspections and $V_{NDE}$ represents the allowance for potential sources of error in the measurement of the bobbin coil voltage. Further discussion of the assumptions necessary to determine the voltage repair limit are discussed in GL 95-05.

The mid-cycle equation should only be used during unplanned inspection in which eddy current data is acquired for indications at the tube support plates.

Technical Specification 4.2.b.6

Tubes with indications of degradation in either the original factory roll expansion in the tubesheet or the unexpanded portion of tube within the tubesheet may be dispositioned for continued service or repaired through application of the F* or EF* criteria. The F* and EF* criteria are described in WCAP-14677[8]. The F* and EF* criteria are established using guidance consistent with RG 1.121. Neither the F* or EF* criteria will significantly contribute to offsite dose following a postulated main steam line break such that contributions from these sources need to be included in offsite dose analyses. Inherent to these criteria is the ability to perform an additional roll expansion of the tube, either as an extension of the original factory roll expansion, in which case F* criteria applies, or in the area starting approximately 4" below the top of the tubesheet, in which case EF* criterion apply. The additional roll expansion procedure can be applied over existing degradation, provided the F* or EF* requirements for non-degraded roll expansion lengths of 1.12" (plus an allowance for NDE uncertainty) and 1.44" (plus an allowance for NDE uncertainty), respectively, are satisfied. The NDE uncertainty applied to the F* and EF* distance is a function of the eddy current probe and technique used. Current state-of-the art inspection technology will be used with implementation of the F* and EF* criteria. The uncertainty in such inspections has been shown to be as small as 0.06", however, for field application, an eddy current uncertainty of 0.20" will be applied. Any and all indications of degradation existing below the F* or EF* distance is acceptable for continued service.

Technical Specification 4.2.b.7

Category C-3 inspection results are considered abnormal degradation to a principal safety barrier and are therefore reportable under 10 CFR 50.72(b)(2)(i) and 10 CFR 50.73(a)(2)(ii).

TS 4.2.b.7.d implements several reporting requirements recommended by GL 95-05 for situations which NRC wants to be notified prior to returning the steam generators to service. For TS 4.2.b.7.d.3 and 4, indications are applicable only where alternate plugging criteria is being applied. For the purposes of this reporting requirement, leakage and conditional burst probability can be calculated based on the as-found voltage distribution rather than the projected end-of-cycle voltage distribution (refer to GL 95-05 for more information) when it is not practical to complete these calculations using the projected EOC voltage distributions prior to returning the steam generators to service. Note that if leakage and conditional burst probability were calculated using the measured EOC voltage distribution for the purposes of addressing GL Sections 6.a.1 and 6.a.3 reporting criteria, then the results of the projected EOC voltage distribution should be provided per GL Section 6.b(c) criteria.

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[8] WCAP 14677, F* and Elevated F* Tube Alternate Repair Criteria for Tubes With Degradation Within the Tubesheet Region of the Kewaunee Steam Generators, June 1996 (Proprietary).