Nuclear Operating Company

South Texas Project Electric Generating Station P.O. Box 289 Wadsworth, Texas 77483

January 24, 2001
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File No.: G21.02.01
10 CFR 50.90

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

South Texas Project
Unit 2
Docket No. STN 50-499
Supplement to Proposed Amendment to South Texas Project Technical Specification 3/4.4.5 - Modify Acceptance Criteria for Repair of Steam Generator Tubes at Certain Intersections of Tubes and Tube Support Plates (TAC No. MA8271)

In February 2000, STP Nuclear Operating Company (STPNOC) proposed to revise the Technical Specifications to implement 3-volt alternate repair criteria for certain Unit 2 steam generator tubes for one fuel cycle (Reference 1). WCAP-15163, Rev. 1, which was attached to Reference 1, provided the technical basis for the proposed 3-volt criteria.

The NRC submitted an informal request for additional information (RAI) from the Materials Branch regarding the 3-volt alternate repair criteria, WCAP-15163, and the proposed Technical Specification changes. The NRC also submitted a formal RAI (Reference 2) that identified several questions regarding vibration and related topics, RELAP-5 and its application, steam generator thermal response, other analysis-related items, and probabilistic analysis considerations.

On November 17, 2000, STPNOC met with NRC management and staff, and presented additional justification for the application of RELAP-5 and a review of the conservatism included in our approach. The NRC noted that the application of RELAP-5 to determine tube support plate loading during a main steam line break was not inappropriate, but that validation of the code version against available test data was necessary prior to its application to provide input hydraulic loading for the tube support plate displacement analysis for Unit 2.

In response to the NRC position, STPNOC initiated a bounding analysis based on conservative first-principles assumptions to determine bounding hydraulic loads and tube support plate deflections that do not depend on the use of RELAP-5. The bounding analysis was performed because the time required to develop the complete validation of the specific application of RELAP-5 as requested in Reference 2 would prevent timely implementation of the alternate repair criteria for which approval had been requested. The objective of the bounding analysis was to demonstrate the significant margins for the probability of burst that exist for the 3-volt criteria.

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To reduce uncertainties due to cross-flow, the application of the 3-volt criteria was limited to the hot leg of the three tube support plates above the flow distribution baffle in the steam generators (plates C, F, and J). To add further conservatism, sixteen tubes at each of the three tube support plates will be expanded to lock them in place. The tube support plate displacement analysis is a static, elastic analysis that assumes unit loading; therefore, the results of this analysis can be extrapolated within limits. STPNOC presented the bounding analysis approach and preliminary results to NRC management and staff on December 8, 2000. On January 10, 2001, STPNOC informally submitted an addendum to WCAP-15163, Rev. 1, which documents the bounding analysis and, in conjunction to the two meetings described above, responds to Reference 2.

STPNOC submits herein a supplemental Technical Specification change package that reflects application of the 3-volt criteria for the three lowest tube support plates rather than the five plates as proposed in Reference 1. The marked-up Technical Specification pages also incorporate comments received from the NRC. The "Determination of No Significant Hazards Consideration" has been revised to reflect the changes proposed by this supplement, but the conclusion that the alternate repair criteria present no significant hazards remains valid. Likewise, the determination that the supplemental change package satisfies criteria of 10CFR51.22(c)(9) for categorical exclusion from environmental assessment remains valid.

Other administrative changes have been made to the steam generator Technical Specification reporting requirements to reflect the 10CFR50.72 rule change and additional requests by the NRC. Effective January 23, 2001, in accordance with 10CFR50.72(b)(3)(ii), the NRC must be notified within eight hours of any event or condition that results in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded. This requirement used to be stated in 10CFR50.72(b)(2)(i) as requiring notification within four hours. This reporting requirement applies to the discovery that more than 10% of the total number of steam generator tubes inspected are degraded tubes or that more than 1% of the inspected tubes are defective. Technical Specification Tables 4.4-2 and 4.4-3 have been revised to reflect the new reporting requirement. The NRC also requested that two additional reports regarding inspection results for the Model E steam generators be made if necessary.

The STP Plant Operations Review Committee has reviewed and approved the revised change package.

In accordance with 10CFR50.91(b), STPNOC is providing a copy of this letter and its attachments to the State of Texas.

If there are any questions regarding this submittal, please contact Mr. Mark Kanavos, Manager, Replacement Steam Generator Project Engineering, at (361) 972-7181.

J. J. Sheppard
Vice President,
Engineering & Technical Services

References:

1. Letter, J.J. Sheppard to NRC Document Control Desk, "Proposed Amendment to South Texas Project Technical Specification 3/4.4.5 - Modify Acceptance Criteria for Repair of Steam Generator Tubes at Certain Intersections of Tubes and Tube Support Plates," NOC-AE-000702, dated February 21, 2000

2. Letter, T. Kim to W. Cottle, "South Texas Project, Unit 2 - Request for Additional Information re: License Amendment Request Associated with Modifying Alternate Repair Criteria of Steam Generator Tubes at Certain Intersections of Tubes and Tube Support Plates (TAC No. MA8271)," AE-NOC-00000699, dated October 31, 2000

Attachments:

1. Affidavit
2. Description of Technical Specification Changes with Safety Evaluation
3. Determination of No Significant Hazard Consideration
4. Annotated Technical Specifications
5. Annotated Technical Specification Bases
6. Reconstituted Technical Specification and Bases Pages

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U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555-0001

ATTACHMENT 1

AFFIDAVIT

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter	)	
	)	
STP Nuclear Operating Company, et al.	)	Docket No. STN 50-499
	)	
South Texas Project Unit 2	)	

AFFIDAVIT

I, J. J. Sheppard, being duly sworn, hereby depose and state that I am Vice President, Engineering & Technical Services, of STP Nuclear Operating Company; that I am duly authorized to sign and file with the Nuclear Regulatory Commission the attached supplemental Technical Specification change to modify acceptance criteria for repair of certain steam generator tube segments; that I am familiar with the content thereof; and that the matters set forth therein are true and correct to the best of my knowledge and belief.

J. J. Sheppard
Vice President,
Engineering & Technical Services

STATE OF TEXAS	)
	)
COUNTY OF Matagorda	)

Subscribed and sworn to before me, a Notary Public in and for the State of Texas, this 24th day of January, 2001.

LINDA RITTENBERRY
Notary Public, State of Texas
My Commission Expires
OCT. 9, 2001

Linda Rittenberry
Notary Public in and for the
State of Texas

ATTACHMENT 2

DESCRIPTION OF
TECHNICAL SPECIFICATION CHANGES
WITH
SAFETY EVALUATION

BACKGROUND

STP Nuclear Operating Company (STPNOC) requests permission to amend the South Texas Project (STP) Unit 2 license to implement 3-volt alternate repair criteria (ARC). This 3-volt ARC will apply only to Model E steam generator tubes experiencing outer diameter stress corrosion cracking (ODSCC) at the intersections of tube hot-legs and tube support plates (TSP) C, F, and J. It will be in effect only until STPNOC replaces the currently installed Model E steam generators with new Model Δ94 steam generators in the fall of 2002.

Displacement of TSP C, F, and J in Westinghouse Model E steam generators during a steam line break (SLB) event has been demonstrated to be very small. Portions of tube hot-legs passing through these TSP are circumferentially constrained by the TSP, reducing to negligible levels the probability that they will burst during a design basis SLB event. STP Unit 2 tube integrity assessments using 3-volt ARC will take credit for this limited TSP displacement. Primary-to-secondary leakage during an SLB will be calculated as free-span leakage in the remaining tube spans.

DESCRIPTION OF PROPOSED CHANGE

STPNOC proposes to amend STP technical specifications (TS) so that steam generator tubes with eddy-current inspection indications of ≤ 3.0 volts can be left in service if the indications are at intersections of tube hot-legs with TSP C, F, and J. The underlying premise for this proposal is:

- A tube with degradation indicating ≤ 3.0 volts that is captured within the thickness of a closely surrounding tube support plate cannot expand sufficiently to pose a credible threat of either tube rupture or leakage exceeding allowable limits.
- TSP C, F, and J do not deflect significantly relative to any tube during normal operation or design-basis accident conditions.
- Tube segments normally located within the thickness of TSP C, F, and J will remain there and will not expose a significant length of a postulated crack at the edge of the TSP.

Consequently, those portions of tube hot-legs circumferentially constrained by TSP C, F, and J and have degradation indicating ≤ 3.0 volts cannot rupture, or leak at rates exceeding allowable limits. This premise has been validated through calculations, laboratory testing, and analysis of actual data from operating steam generator tubes.

The specific effect of this proposal on STP TS is summarized as follows:

1) TS 3/4.4.5.2.d.1) through 3/4.4.5.2.d.4) are added to reflect specific requirements in WCAP-15163, Rev. 1, Table 2-2.

2) In TS 3/4.4.5 STEAM GENERATOR, section 4.4.5.4.a.11, delete the last sentence, which reads, "At tube support plate intersections, the plugging (repair) limit is based on maintaining steam generator tube serviceability as described below," and add INSERT A as a second paragraph in this section. The new paragraph stipulates the locations to which the existing repair criteria continue to apply. Additionally, specific requirements from WCAP-15163, Rev. 1, Table 2-2 are added to TS 3/4.4.5.4.a.11.d).

3) In TS 3/4.4.5 STEAM GENERATOR, section 4.4.5.4.a.11, immediately following Note 2, add INSERT B. This appends a paragraph and two lettered sub-paragraphs, e) and f), that provide for use of the 3.0 volt ARC at tube hot-leg intersections with support plates C, F, and J.

4) TS 3/4.4.5.5.d.1.a) and -b); 3/4.4.5.5.d.5.a) and -b); and 3/4.4.5.5.e. are added to reflect specific requirements from WCAP-15163, Rev. 1, Table 2-2.

5) TS 3/4.4.5.5.d.6) and -7) are added to reflect an NRC request for additional inspection results.

6) Tables 4.4-2 and 4.4-3 are revised to reflect a rule change in 10CFR50.72 regarding 8-hour reports.

7) In the REACTOR COOLANT SYSTEM BASES for TS 3/4.4.5 STEAM GENERATORS, page B 3/4 4-3, 4th paragraph from the top, replace the 2nd sentence with INSERT C. This provides the structural margins and Westinghouse topical references used as the bases for application of 3.0 volt ARC.

8) Note 1 and Note 2, page 3/4 4-16a, are indented to align with paragraph d) immediately preceding.

SAFETY EVALUATION

STPNOC herewith provides the analysis required by 10CFR 50.91(a)(1), which demonstrates that the proposed license amendment for use of ARC does not represent a significant hazard. Application of the proposed 3-volt ARC relies on limited TSP displacement during an SLB event. Tube burst probability and leakage rate associated with this limited displacement are well within the allowable burst probability and leakage rate licensed for STPNOC.

Note: References in this submittal to a rotating pancake coil (RPC) probe also refer to any probe that is the functional equivalent thereof.

Reference 1 describes the basis for the proposed use of 3-volt ARC, incorporating the following considerations:

- A 3-volt repair limit for tube hot-leg intersections with TSP C, F, and J
- Repair of flaws detected at intersections of tube hot-legs and TSP C, F, and J if bobbin-coil-probe indications are > 3.0 volts, regardless of the results of rotating-pancake-coil (RPC) probe confirmation.
- RPC inspection of intersections with mechanically induced dent signals > 5.0 volts and with bobbin mixed residual signals that could potentially mask fault indications near and above voltage repair limits.
  - Use of stainless steel TSP in South Texas Project Unit 2 steam generators eliminates corrosion denting as a consideration; therefore, no special inspection requirements related to corrosion induced denting are required.

- Repair of:
  - indications found during RPC inspection at the intersection of tubes and TSP where mechanically induced dents are >5.0 volts, and
  - indications found during RPC inspection at the intersection of tubes and TSP where large mixed residual signals mask detection by a bobbin-coil probe.
- No exclusion of 3-volt ARC application near TSP wedge locations.
  - No TSP wedge locations in STP Unit 2 steam generators are subject to plastic deformation capable of causing tube damage during LOCA + SSE loading conditions.
- Continued exclusion of fifteen tubes in steam generator D of STP Unit 2 from application of ODSCC ARC because they are made of thermally treated Alloy 600 instead of the mill annealed Alloy 600 of which the remaining tubes are made.
  - No pulled-tube data is available to confirm ODSCC morphology for thermally treated Alloy 600.
  - Thermally treated Alloy 600 is less susceptible to stress corrosion attack than mill annealed Alloy 600. This metallurgical consideration makes it unlikely that ODSCC will occur at TSP intersections with these tubes and none has been detected to date.
  - TS section 4.4.5.4.a.11 captures this exclusion in the Tube Support Plate Plugging Limit where it states that this limit is for the disposition of "a *mill annealed* alloy 600 steam generator tube…" (emphasis added).
- Bobbin-coil probe inspections of intersections of all tube hot-legs with TSP, all tube hot-legs with FDB, and all tube cold-legs with TSP down to the lowest cold-leg TSP having ODSCC.
  - Determination of the lowest cold-leg TSP intersections having ODSCC indications is based on the performance of at least a 20% random sampling of tubes inspected over their full length.
- RPC probe inspection of all flaws that are detected at intersections of tube hot-legs and TSP C, F, and J and have bobbin-coil probe indications > 3.0 volts.
- RPC inspection of a minimum, total for all four steam generators, of 100 tube hot-legs intersecting with TSP C, F, and J that have bobbin coil probe indications ≤ 3.0 volts.
- Evaluation of RPC data to confirm that responses within the confines of the TSP are typical of ODSCC.

The proposed amendment increases the voltage limit for steam generator tube ARC in Technical Specification 3/4.4.5, "Steam Generators," and the associated Bases. This amendment specifies tube inspection requirements and acceptance criteria to describe the level of degradation at which a tube experiencing ODSCC at TSP C, F, and J must be removed from service in the South Texas Project Unit 2 Model E steam generators.

Model E2 Steam Generator Design

STP Unit 2 steam generators are Westinghouse Model E2 preheat design. Each steam generator contains 4,851 Alloy 600 U-tubes of 0.75 inch OD by 0.043-inch wall, providing 68,000 sq. ft of heat transfer area per steam generator. All tubes are mill-annealed except for fifteen tubes in steam generator 2D that are thermally treated.

Linear portions of the inverted-U-shaped steam generator tubes pass through TSP at various levels to prevent lateral tube motion. During normal operation, there is a small pressure drop across each TSP or baffle plate. This pressure drop causes a slight elastic displacement of the TSP relative to the tubes. The magnitude of this elastic deflection at a specific evaluated location on a TSP depends on loading and the effects of normal TSP support geometry. During a postulated design basis accident, such as SLB, pressure differential can cause increased deflection in unsupported regions of certain TSP. This increased deflection could expose degraded tube spans that are normally circumferentially constrained by their respective TSP. For these TSP, degraded tube spans are treated as if they were in the free-span of the tube.

Tube Degradation Characterization

The STP Unit 2 program for tube removal and examination will comply with the guidance of GL 95-05, Section 4.0, "Tube Removal and Examination/Testing."

Steam Generator Tube Integrity

Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," and RG 1.83, Rev. 1, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes," are used in development of voltage-based ARC for South Texas Project Unit 2. They serve as the bases for determining whether steam generator tube integrity remains within acceptable limits. RG 1.121 describes a method accepted by NRC staff for meeting General Design Criteria (GDC) 14, 15, 31, and 32 through reducing the probability and consequence of steam generator tube rupture. This is done by determining the safe in-service inspection limit for tube wall degradation, beyond which tubes should be removed from service by plugging. This regulatory guide applies tube burst load safety factors that are consistent with ASME Section III requirements .

For tube degradation occurring at TSP elevations in the South Texas Project Unit 2 steam generators, tube burst criteria are inherently satisfied during normal operating conditions. The TSP closely surrounds the tube in that region and precludes tube deformation beyond the diameter of the drilled hole, thus precluding tube burst. Analysis shows that displacement of TSP C, F, and J during a design basis SLB is sufficiently limited to essentially preclude tube burst at the locations of their intersections with tubes (Reference 1).

STP has performed a TSP displacement analysis during a design basis SLB to assess tube burst probabilities. These bounding tube burst probabilities for Unit 2 steam generators conservatively assume that all intersections of tube hot-legs for which the 3-volt criteria apply have through-wall cracks and that these cracks will be exposed to the extent of the calculated limited displacements allowed by tube expansions. Analysis shows that TSP loads are highest if an SLB occurs at hot standby, rather than full power, conditions.

South Texas Project Unit 2 steam generator TSP are vertically supported using various means:

- Fourteen stayrods/spacers vertically support each TSP.
- Eighteen additional stayrods vertically support plates in the pre-heater region.
- Sixteen additional hot-leg tubes will be expanded at the TSP to lock the TSP in place. The expanded tubes will then be plugged.
- Vertical bars above and below each plate are welded to the inside of the wrapper, to the impingement plate, and to the partition plate.
- Wedges located at the periphery of each plate are welded to the inside of the wrapper and to the impingement and partition plates. These wedges primarily provide lateral support for the TSP, but also provide resistance to upward motion of the plates because the small ends of the wedges face downward.

TSP are constrained by stayrods and spacer bars. Stayrods are straight bars threaded into the tubesheet and extend to the full height of the linear portion of the tube bundle. They are secured by a nut on the upper side of the topmost TSP, plate R. The lower end of the center stayrod is threaded into a special coupling welded to the top of the partition plate. Around the outside of the stayrods and between each of the TSP are spacers. Non-pre-heater stayrods and spacers have no rigid link between the spacer and the support plates.

In the pre-heater region, stayrods are segmented rods tack welded to the flow distribution baffle that run between the various baffle plates. Each segment threads into the lower end of the stayrod immediately above it. Unlike the full-bundle stayrods that pass through the TSP without interaction, pre-heater stayrods interface with each plate due to plate geometric differences. Unlike full-bundle stayrods, there are no spacers surrounding pre-heater stayrods.

A finite element model that simulates the structural response of the tube bundle was used to determine relative tube/plate motion during design basis SLB. Due to hot-leg-to-cold-leg asymmetry caused by the pre-heater, this model includes 180° of the tube bundle.

Analyses reveal that TSP loads are higher for an SLB occurring at hot standby than for one occurring at full power. At hot standby, the juxtaposition of TSP with tube cracks located within the thickness of the TSP is essentially the same as at cold shutdown. Every known inspection of SG at cold shutdown shows that ODSCC in non-dented tube portions located within the thickness of TSP tend to be centered in the TSP. Therefore, the location of TSP relative to the tubes at cold shutdown is essentially the same relative location as at full power operation during which the cracks are formed, which is also the relative location during hot standby. These inspections indicate that there is little relative movement between tubes and TSP throughout the operating cycle. Thus, the structural analysis calculates tube/TSP relative motion based on tube/TSP positions at initiation of the SLB transient.

For ODSCC occurring within the thickness of TSP, STPNOC implemented voltage-based ARC for Unit 2 in License Amendment 83 according to the guidelines of NRC GL 95-05. For steam generators with 3/4" tubing, a conservative ARC limit of 1.0 volt was established to meet a 10^{-2} probability of burst. This avoids exceeding allowable leakage rate or offsite dose limits during the limiting accident. For ODSCC occurring at intersections of tube hot-legs with TSP C, F,

and J, results of analyses (Reference 1) reveal that TSP stability limits tube burst probability to $<< 10^{-2}$. Calculation of this negligible burst probability includes the extremely conservative assumption that all tubes have through-wall cracks at their TSP and that these cracks are exposed by the limited displacement during an SLB. Thus, repair limits to preclude burst are not necessary and tube repair limits may be directed at limiting accident condition leakage to acceptable levels.

At some level of degradation, with commensurately higher bobbin voltage level, it becomes possible for axial loads resulting from the pressure differential across the tubes to result in axial tensile separation of the tube. This tensile load requirement establishes the applicable structural limit voltage for the limited displacement-based plugging criterion. Tensile tests to measure the force required to separate a tube with cellular corrosion patches are used to establish a lower bound structural limit. Additionally, for some pulled tubes with cellular and/or inter-granular attack (IGA) tube-wall degradation, the tensile capability of the tube can be conservatively calculated from the non-corroded cross-section of the tube. This method assumes that the degraded portions do not contribute to the axial load carrying capability of the tube. From available data it is clear that the tube pressure differential necessary to cause circumferential ruptures, i.e., axial separation at the plane of the circumferential rupture, is well above 3 times normal operating pressure differential limit. This occurs at bobbin voltages exceeding 35 volts (Reference 1). Because of the size of the database, a lower bound structural limit of greater than 17 volts (additional safety factor of 2) is conservatively established. This restricts the upper range of the tube repair limit until additional data is obtained to refine the structural limit.

Destructive examination of tubes pulled from intersections that were not plugged during two years, or more, before the tube pull has not found IGA of significant depth. In those tubes plugged for at least two years before being pulled, only one plant was identified as having IGA of significant depth. Even in this instance, IGA was approximately one-half as deep as the IGSCC that dominated the corrosion morphology. Conceptually, a circumferential crack could also limit the structural capability of a tube; however, circumferential cracking has not been found at non-dented intersections. Tube denting at TSP intersections in the STP Unit 2 steam generators is not anticipated because the TSP are made from stainless steel.

For tube hot-leg intersections at other than TSP C, F, and J, it is adequate and conservative to continue to use the currently approved 1-volt ARC, which assumes free-span leakage for ODSCC during the design basis SLB.

Implementation of 3-volt ARC at TSP C, F, and J for STP Unit 2 must result in a leak rate that remains within acceptable limits during all plant operating conditions. As with the 1.0-volt free-span ARC limit for STP Unit 2, implementation of the proposed 3-volt ARC for intersections of tube hot-legs and TSP C, F, and J is expected to result in negligible leakage during normal operating conditions. It will also remain within established limits during design basis conditions. This remains true even if there are indications of potential through-wall cracks.

<u>SLB Leak Rate and Tube Burst Probability Considerations</u>

GL 95-05 requires the licensee to perform an SLB leak rate analysis before returning to power after an outage during which a steam generator inspection was conducted. Results of the analysis are to be submitted in a report to the NRC within 90 days following restart (breaker closure). An

operational assessment must also be performed and the results included in this 90-day report. An operational assessment is a determination of the total leak rate and the total burst probability associated with each steam generator, extrapolated to the last day of the next operating cycle.

Any indication for which the probability of burst is extrapolated to become 1.0 or greater at any time during the coming operating cycle is an "overpressurized" indication. SLB leak rate calculations for tube hot-leg indications found at TSP C, F, and J will use the free-span leakage methodology provided by GL 95-05, except that a value determined through testing will be added to the overall SLB leakage determination to compensate for "overpressurized" indications. For each of these "overpressurized" indications, the value to be added to the total predicted leak rate is the bounding leak rate identified for indications restricted from burst (IRB). The total SLB leak rate limit for the faulted STP Unit 2 SG is 15.4 gpm (Reference 2).

An operational assessment is a complex calculation that extrapolates growth of the current level of tube degradation over the next operating cycle, based on a calculated growth rate distribution for the degradation. It also calculates probability of burst and total leakage over the period between this and the next inspection. If an operational assessment can be completed before returning the steam generators to service, then no other assessment is required, and results are provided to the NRC with the 90-day report. Alternatively, if time doesn't allow completion of an operational assessment prior to returning steam generators to service, an interim 'condition monitoring assessment' may be performed, since a condition monitoring assessment is a less complex calculation and requires less time to complete. A condition monitoring assessment calculates total leak rate and tube burst probability for each steam generator using as-found data collected during the inspection.

The steam generators may be returned to service if results of either type of assessment are satisfactory. However, if steam generators are returned to service based on results of a condition monitoring assessment, an operational assessment still must be completed and its results provided to the NRC in the 90-day report. If leak rates calculated during the outage exceed established limits, STPNOC must report these results to the NRC before restart, and perform an assessment of their safety significance.

A finite probability exists that a crack in a tube in an operating steam generator could open significantly more than a similar crack in the sample population of tubes that were laboratory-tested to establish a correlation between bobbin-coil-probe voltage indications and SLB leak rates. The probability that a crack at the intersection of a tube and TSP will open to the limits of the tube-to-TSP gap is equivalent to the probability that a similar crack in a free-span tube section will burst.

An indication restricted from burst (IRB) is a through-wall tube crack within the intersection of tube and TSP of a size that could burst under SLB conditions if it were located in a free-span portion of the tube. However, IRB tube walls are prevented from bursting by closely surrounding TSP that restrict their expansion within the limit of the tube-wall to TSP gap.

In conjunction with 3-volt ARC development, the Electric Power Research Institute (EPRI) conducted a test program (Reference 3) to determine the IRB bounding leak rate and its sensitivity to TSP displacement. Testing demonstrated that leakage from an IRB is limited to a rate less than that of a similar free-span indication. During a limiting SLB event, steam generator

depressurization can cause TSP to deflect from their nominal position. In some cases, this deflection can be sufficient to expose tube cracks normally surrounded by TSP. However, TSP C, F, and J have been shown to have small deflections during the limiting SLB event. The effect of these small deflections on probability of burst is negligible and, thus, cracks at intersections with TSP C, F, and J remain constrained from bursting during accident conditions. These test results substantiate the suitability of increasing ARC to 3.0 volts for IRB at TSP C, F, and J.

During laboratory tests, fifteen steam generator tube specimens were tested. Eight of the specimens were 7/8" diameter tubes and seven were 3/4" diameter tubes. Tube specimens were made of mill annealed Alloy 600, the prototypical steam generator tubing material. The three processes used in preparing specimens were: 1) accelerated corrosion; 2) accelerated corrosion followed by fatigue to increase the length of the crack; and 3) laser cutting. These tests simulated a cracked tube at a TSP, conservatively assuming the maximum diametrical clearance between tube and TSP. The tests were configured to provide a 0.025" gap at the side of the tube with the crack to minimize the restriction provided by the TSP. The longest through-wall crack tested, 0.809 inches, was greater than any crack that could form at any TSP intersection. Testing was performed with the entire crack contained within the thickness of the TSP, with one end of the crack aligned with the edge of the TSP, and with the crack tip positioned outside the TSP.

These tests determined that the limiting indication to which a high voltage ARC would apply has a leak rate of 5.5 gpm at 2560 psid. Based on limitations imposed by the pressurizer power-operated relief valve (PORV) setpoints, maximum pressure differential for South Texas Project Unit 2 is 2405 psid. At 2405 psid the bounding leak rate is 5.0 gpm. Tests of IRB demonstrate that this bounding leak rate is constant for TSP displacements up to 0.21 inch, which conservatively bounds the calculated displacements of TSP C, F, and J. For through-wall cracks that produce the bounding leak rate, bobbin-probes are expected to indicate a minimum of 8 volts. This is significantly greater than the 3-volt limit being requested here, and for cracks much shorter that those evaluated in the IRB tests.

The leak rate from a crack is an exponential function of the through-wall length of the crack, neglecting TSP interaction. Testing has demonstrated that if a tube has a longer crack together with a shorter crack, the longer crack dominates the leak rate and the shorter crack contributes only slightly to the leak rate. The combined leak rate from a principal crack and a subordinate crack is much less than the leakage from a single crack the length of which is equal to the sum of the lengths of the two cracks. Similarly, in tubes with multiple cracks the principal crack dominates structural behavior in that region of the tube, controlling overall leakage. Together, these considerations demonstrate that there is no need to adjust the bounding leak rate to compensate for multiple through-wall indications.

Leakage from tube hot-legs at intersections with TSP C, F, and J during the limiting SLB is determined using a variant of the free-span leakage calculation methodology provided in the Westinghouse methods report. This methodology conforms to NRC Generic Letter 95-05 and uses Monte Carlo simulation to accurately model significant parameters, such as distribution of tube-flaw indications, expected future growth of tube-flaws, and measurement uncertainty. The variant conservatively assigns the bounding leakage-flow value of 5.0 gpm for all IRB, regardless of their bobbin-coil-probe voltage reading. For other indications, the model determines whether there is a probability that it will leak. For indications determined to have a probability of leaking,

the model assigns a leak rate based on correlation of the common logarithm of the leak rate to the common logarithm of the bobbin-coil-probe measurement amplitude, in conformance with GL 95-05.

Burst Probability

Tube burst probability analyses are required for tube cold-leg indications at TSP, all indications at the flow distribution baffle (FDB), and tube hot-leg indications at TSP L through R. Monte Carlo burst calculations for tube hot-leg intersections with TSP C, F, and J are not required because the limited displacement of these plates under postulated SLB loading allows only a negligible probability of tube burst. The resulting burst probabilities will be compared to the GL 95-05 reporting guideline of 10^{-2}.

Stayrod/Spacer Stresses

Since elastic response is the basis of the dynamic analysis, calculations establish that the stayrods and spacers, which provide significant support for the plates, remain elastic throughout the transient, and stayrod and spacer stresses are well within ASME allowable limits.

Tube Support Plate and Vertical Bar/Wedge Weld Stresses

Stresses to which the TSP are exposed are relevant in determining whether the elastic solution is appropriate. Thus, TSP stress values were calculated for the time at which maximum displacement occurs. As with stayrods and spacers, TSP stresses are calculated to be well within limits. Stresses were also calculated for the vertical-bar and wedge-weld to wrapper contact points and found to be within limits.

In addressing combined LOCA + SSE effects on steam generator components as required by GDC 2, analysis has shown that tube collapse may occur in certain regions of the steam generators of some plants. This collapse is caused by TSP plastic deformation in the region of the TSP wedge supports. Plastic deformation occurs when TSP experience large lateral loads concentrated at wedge support points on the periphery of a TSP undergoing combined loading effects of a LOCA rarefaction wave and SSE. Deformation impinges on TSP apertures through which tubes pass, deflecting tube walls inward. The resulting pressure differential across deformed tube walls may cause some tubes to collapse.

There are two issues associated with steam generator tube collapse. First, collapse of steam generator tubing reduces RCS flow. RCS flow reduction increases resistance to heat flow from the core during a LOCA, increasing Peak Clad Temperature (PCT). Second, partial through-wall tube cracks could become full through-wall tube cracks during tube deformation or collapse. Tubes in regions affected by this phenomenon are usually excluded from evaluation under 3-volt ARC. STP Model E steam generator design does not produce this plastic deformation, thus is not subject to tube collapse. No STP Unit 2 tubes are excluded, for this reason, from application of the proposed 3-volt ARC.

Steam Generator Internals Inspection

Past visual inspections of steam generator internals confirm that components and structures forming the load path credited in use of 3-volt ARC for Unit 2 have not been degraded. STPNOC performed a variety of secondary side inspections during past refueling outages and

found no degradation of the load path. Table 10.2 of Reference 1 summarizes the results of these inspections. The details are provided in Reference 4 (noted below) and will be included in Section 6 of WCAP-15163, Revision 1, Addendum 1. It is estimated that performing internals inspections on the Unit 2 steam generators for one fuel cycle before their replacement would cost $440,000 and increase radiation exposure to workers by 12 rem. These expenditures are not prudent in light of the existing inspection data compiled both at STP and by the industry. Additionally, expansion of the sixteen hot-leg tubes at TSP C, F, and J obviates the need to perform internal stayrod inspections.

Stainless steel TSP in the South Texas Project Unit 2 steam generators eliminates corrosion denting as a consideration. Therefore, no special inspection requirements related to corrosion-induced dents are required, and no exclusion zones due to denting are required.

Additional Considerations

The proposed amendment will provide additional benefits. It will:

- reduce non-essential tube plugging and associated occupational radiation exposure,
- minimize the time that steam generators are open to containment atmosphere,
- conserve reactor coolant flow margin needed for design basis accidents,
- maximize flow rates for full power operation, and
- reduce the length of plant outages.

REFERENCES:

1. WCAP-15163, Rev. 1 "Technical Support for Implementing High Voltage Alternate Repair Criteria at Hot Leg Limited Displacement Tube Support Plate Intersections for South Texas Unit 2, Model E Steam Generators," March 1999, including Addendum 1
2. Letter, D.A. Leazar to NRC Document Control Desk, "Response to NRC's Request for Additional Information (TAC Nos. MA0967 and MA0968)," NOC-AE-000228, dated July 15, 1998
3. TR-107625, SG Indications Restricted from Burst (IRB) Leak Test Report, Final Report, EPRI, September 1998
4. Letter, T.H. Cloninger to NRC Document Control Desk, "90-Day Response to Generic Letter 97-06, 'Degradation of Steam Generator Internals'," NOC-AE-000117, dated March 30, 1998

ATTACHMENT 3

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

In accordance with the criteria set forth in 10CFR50.92, the STP Nuclear Operating Company (STPNOC) has reviewed these proposed Technical Specification changes and determined they do not represent a significant hazards consideration. Conformance of the proposed amendment to the standards for a determination of no significant hazard as defined by the criteria set forth in 10 CFR 50.92 is shown in the following discussions addressed to each criterion:

1) **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   During the limiting design-basis steam-line-break (SLB) event, South Texas Project (STP) Unit 2 steam generator tube burst criteria are inherently satisfied for marginally degraded (primarily axially-oriented ODSCC) tube spans at certain tube support plate (TSP) intersections.

   Steam generator tubes pass through holes drilled in the TSP. The inside diameter (ID) of the drilled holes closely approximates the outside diameter (OD) of the tubes. Generally, the TSP precludes those tube spans within the drilled holes from deforming beyond the diameters of the drilled holes, thus, precluding tube burst in the restrained regions. However, design basis SLB events may vertically displace a TSP, removing its support from the tube spans passing through it. The deflections of the affected hot leg support plates are small and remain essentially stationary during all conditions. Tube spans included within the drilled holes are restrained during the limiting SLB event. Thus, the tube burst margin for intersections of tube hot-legs and TSP C, F, and J is independent of voltage-related growth rates and the proposed 3-volt ARC is compliant with RG 1.121 criteria.

   For the calculated displacement of the affected support plates, tube hot-leg spans enclosed within TSPs C, F, and J have a negligible tube burst probability of much less than 10^{-5} collectively. This is orders of magnitude less than the 10^{-2} probability-of-burst criterion specified by GL 95-05 and represents negligible axial tube burst probabilities for affected tube hot-leg spans intersecting TSPs. Thus, repair limits to preclude burst are not needed and tube repair limits may be based primarily on limiting leakage to acceptable levels during accident conditions.

   Cracks that include cellular corrosion may yield to axial loads, resulting in tensile tearing of the tube at that location. A tensile load requirement to prevent this establishes a structural limit for the tube expansion-based plugging criterion. In order to establish a lower bound for the structural limit, tensile tests were used to measure the force required to separate a tube that exhibits cellular corrosion. Additionally, pulled tubes with cellular and/or inter-granular attack (IGA) tube wall degradation were evaluated and the tensile strength of the tube conservatively calculated from the remaining non-corroded cross-section of the tube. This calculation assumes that the degraded portions contribute nothing to the axial load carrying ability of the tube. Data from these tests shows that circumferential cracks exhibiting bobbin-coil-probe-indication-voltages greater than 35 volts require tube-pressure-differentials well

above the operating limit of 3-times-normal differential pressure in order to produce circumferential ruptures (i.e., axial separation at the plane of the crack). This proposal specifies a structural limit of 17 volts (safety factor of 2) to ensure conservative results for repairs at intersections of tubes with TSP C, F, and J.

GL 95-05 states that licensees must perform SLB leak rate and tube burst probability analyses before returning to power from outages during which they perform steam generator inspections. Licensees must include the results in a report to the NRC within 90 days after restart. If an analysis reveals that leak-rate or burst-probability exceeds limits, the licensee must report it to the NRC and assess the safety significance of this finding. Model E steam generator SLB leak rates are calculated for indications found at intersections of tube hot-legs and TSP. Both SLB leak rate and tube burst probability are calculated for tube hot-leg intersections. For the burst contribution of TSP C, F, and J, a bounding collective burst probability is used in lieu of Monte Carlo techniques.

It has been established that the design basis main SLB outside of containment and upstream of the MSIV produces the limiting radiological consequence from any tube leakage that may be postulated to exist at the initiation of an accident. With use of 3-volt ARC, STPNOC will calculate the maximum primary-to-secondary leakage for the last day of the coming steam-generator service-cycle and use this value to calculate the radiological consequence of the limiting SLB event. This methodology will ensure that site boundary doses for this accident remain within an acceptable fraction of the 10 CFR 100 guidelines and that doses to the control room operators remain within GDC 19 limits.

The changes in Technical Specifications reporting requirements to reflect the 10CFR50.72 rule change and additional NRC reporting requests are administrative in nature, and do not affect the safe operation of the plant.

Based on the above, STPNOC concludes that operation of South Texas Project Unit 2 in accordance with the proposed license amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2) Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

Use of the proposed steam generator tube 3-volt ARC does not significantly change circumstances or conclusions assumed by the plant design basis. Application of the 3-volt ARC does not significantly increase the probability of either single or multiple tube ruptures. Steam generator tube integrity remains adequate for all plant operating conditions.

STPNOC has confirmed that the allowed post-accident primary-to-secondary leakage rate for SLB events results in the limiting offsite and control room doses for South Texas Project Unit 2. A projected SLB leak rate of 15.4 gpm is calculated to produce doses ≤ 90% of the currently licensed South Texas Project Unit 2 dose limits (Reference 2). STPNOC TS impose a normal leak rate limit of 150 gpd (0.1 gpm) per steam generator to minimize the potential for excessive leakage during all plant conditions. The 150-gpd limit provides added margin to accommodate contingent leakage should a stress corrosion crack grow at a greater than expected rate or extend outside the TSP. Leakage trending consistent with EPRI Report

TR-04788, "PWR Primary-to-Secondary Leak Guidelines," has been established for South Texas Project Unit 2.

Since steam generator tube integrity will meet GL 95-05 requirements and be confirmed through in-service inspection and primary-to-secondary leakage monitoring, the proposed license amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The changes in Technical Specifications reporting requirements to reflect the 10CFR50.72 rule change and additional NRC reporting requests are administrative in nature, and do not affect the safe operation of the plant.

3) **Does this change involve a significant reduction in a margin of safety?**

RG 1.121 describes a method for meeting GDC 14, 15, 31, and 32 by reducing the probability or consequences of steam-generator tube-rupture through application of criteria for removing degraded tubes from service. These criteria set limits of degradation for steam generator tubing through in-service inspection. Analyses show that tube integrity will remain consistent with the criteria of Regulatory Guide 1.121 after implementation of the proposed 3-volt ARC. Even under the worst case ODSCC occurrence at TSP elevations, 3-volt ARC will not cause or significantly increase probability of a steam-generator tube-rupture event.

In addressing combined LOCA + SSE effects on steam generator components as required by GDC 2, analysis has shown that tube collapse may occur in certain regions of the steam generators of some plants. This collapse is caused by TSP plastic deformation in the region of the TSP wedge supports. Plastic deformation occurs when TSP experience large lateral loads concentrated at wedge support points on the periphery of a TSP undergoing combined loading effects of a LOCA rarefaction wave and SSE. Deformation impinges on TSP apertures through which tubes pass, deflecting tube walls inward. The resulting pressure differential across deformed tube walls may cause some tubes to collapse.

There are two issues associated with steam generator tube collapse. First, collapse of steam generator tubing reduces RCS flow. RCS flow reduction increases resistance to heat flow from the core during a LOCA, increasing Peak Clad Temperature (PCT). Second, partial through-wall tube-cracks could become full through-wall tube-cracks during tube deformation or collapse. Tubes in regions affected by this phenomenon are usually excluded from evaluation under 3-volt ARC. STP Model E steam generator design does not produce this plastic deformation, thus is not subject to tube collapse. No STP Unit 2 tubes are excluded, for this reason, from application of the proposed 3-volt ARC.

End of Cycle (EOC) distribution of crack indications at affected TSP elevations will be confirmed to allow no more than the acceptable primary-to-secondary leakage rate during all plant conditions and not adversely affect radiological dose consequences. For the limiting SLB event, STPNOC will calculate leak rates as free-span leakage for ODSCC indications at tube and TSP intersections. The calculations will use GL 95-05 leak rate methods with an additional component for potentially overpressurized indications at TSP C, F, and J.

Inspections conducted in accordance with RG 1.83, Rev. 1, for 3-volt ARC at hot-leg intersections and 1-volt ARC at cold-leg intersections will be supplemented by:

1) enhanced eddy current inspection procedures to achieve consistency in voltage normalization,
2) eddy current inspection of 100% of tubes found, using inspection of a 20% tube sample, to have ODSCC at intersections with TSP, and
3) a required RPC inspection of the larger indications to confirm that the principal degradation mechanism continues to be ODSCC.

Plugging steam generator tubes reduces RCS flow margin. As previously noted, increasing repair limits for indications found at TSP intersections will reduce the number of tubes that must be plugged. Thus, 3-volt ARC will conserve RCS flow margin, preserving operational and safety benefits that would otherwise be reduced by unnecessary plugging.

The changes in Technical Specifications reporting requirements to reflect the 10CFR50.72 rule change and additional NRC reporting requests are administrative in nature, and do not affect the safe operation of the plant.

Therefore, the proposed license amendment does not result in a significant increase in dose consequences represented in the current licensing basis, and does not involve a significant reduction in margin of safety.

CONCLUSION

STPNOC analyses, testing, and assessments demonstrate that using 3-volt ARC to evaluate indications of steam generator tube ODSCC degradation found at intersections of tube hot-legs and TSPs C, F, and J is acceptable and presents no significant hazard, as defined in 10 CFR 50.92.

ATTACHMENT 4

ANNOTATED
TECHNICAL SPECIFICATIONS

ANNOTATED TECHNICAL SPECIFICATIONS

The following Technical Specification pages are attached to show the specific changes made as a result of incorporating NRC comments on Attachment 4 of Reference 1 or as a result of the bounding calculation. The pages used for this mark-up are the reconstituted Technical Specification and Bases pages provided in Attachment 6 of the original submittal.

Each markup has the source of the change annotated as follows:

1) NRC editorial comment
2) NRC request that the information in Table 2-2 of WCAP-15163, Rev. 1 be transferred to the Technical Specifications
3) The bounding calculation that addressed using the alternate repair criteria only for hot-leg tube intersections with plates C, F, and J rather than with plates C, F, J, L, and M (i.e., "plates C through M")

Tables 4.4-2 and 4.4-3 are also included as mark-ups to reflect the new reporting requirements of 10CFR50.72(b)(3)(ii).

Pages:

3/4 – 12	(No changes)
3/4 – 13	
3/4 – 13a	
3/4 – 14	(No changes)
3/4 – 15	(No changes)
3/4 – 16	
3/4 – 16a	
3/4 – 16b	
3/4 – 16c	
Table 4.4-2	
Table 4.4-3	

NO CHANGES ON THIS PAGE

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 generator(s) to OPERABLE status prior to increasing Tavg above 200°F.

SURVELLANCE 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 and Table 4.4-3. 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 reinspection. The tubes selected for each inservice inspection shall include at least 3% of the total number of nonrepaired tubes in all steam generators and (for Model E steam generators only) 20% of the total number of repaired 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:

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

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

3) A tube Inspection (pursuant to Specification 4.4.5.4a.9) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

4) For Model E steam generators only, 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.

c. The tubes selected as the second and third samples (if required by Table 4.4-2 or Table 4.4-3) during each inservice inspection may be subjected to a partial tube inspection provided:

1) The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found, and

2) The inspections include those portions of the tubes where imperfections were previously found.

NRC Editorial

WCAP Table 2-2

d. For Model E steam generators only, implementation of the steam generator tube/tube support plate repair criteria requires a 100-percent bobbin coil inspection for the flow distribution baffle plate intersections, for the hot-leg tube support plate intersections, and for the 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-percent random sampling of tubes inspected over their full length.

Info added from WCAP Table 2-2 at NRC request

1) All intersections with mechanically induced dent signals greater than 5 volts identified by bobbin coil inspection shall be inspected by rotating pancake coil (or equivalent).

2) All intersections with large mixed residuals that could potentially mask flaw responses at or above the voltage repair limits shall be inspected by rotating pancake coil (or equivalent).

Bounding calculation

3) At the flow distribution baffle intersections, at the cold leg support plate intersections, and at the hot leg support plate intersections with support plates ~~N~~ L through R (as identified in Figure 5.1 of WCAP-15163, Revision 1), tubes with degradation attributed to axially-oriented ODSCC within the bounds of the tube support plate with a bobbin voltage greater than the lower voltage repair limit (defined in 4.4.5.4.a.11) shall be inspected by rotating pancake coil (or equivalent).

Info added from WCAP Table 2-2

4) At the hot leg support plate intersections with support plates C, F, and J (as identified in Figure 5.1 of WCAP-15163, Revision 1), tubes with degradation attributed to axially-oriented ODSCC within the bounds of the tube support plate with a bobbin voltage greater than 3 volts shall be inspected by rotating pancake coil (or equivalent) eddy current probe. An additional 100 tube intersections with support plates C, F, and J with degradation attributed to axially-oriented ODSCC within the bounds of the tube support plate with a bobbin voltage less than 3 volts (100 total over all steam generators, not necessarily selected at random) shall be inspected by rotating pancake coil (or equivalent).

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

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

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

NO CHANGES ON THIS PAGE

SURVEILLANCE REQUIREMENTS (Continued)

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

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

   Note: Inservice inspection is not required during the steam generator replacement outage.

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

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

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

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

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

   4) A main steam line or feedwater line break.

NO CHANGES ON THIS PAGE

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.4 Acceptance Criteria

a. As used in this specification:

1) Tubing or Tube means that portion of the tube or sleeve which forms the primary system to secondary system pressure boundary;

2) 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;

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

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

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

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

7) Plugging Limit or Repair Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging or (for Model E steam generators only) repaired by sleeving in the affected area because it may become unserviceable prior to the next inspection. The plugging or repair limit imperfection depths are specified in percentage of the nominal wall thickness as follows:

a. original tube wall	40%
b. Westinghouse laser welded sleeve wall	40%

For Model E steam generators, this definition does not apply to tube support plate intersections for which the voltage-based repair criteria are being applied. Refer to 4.4.5.4.a.11 for the repair limit applicable to these intersections.

8) 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;

9) 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;

10) 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.

11) For Model E steam generators only, Tube Support Plate Plugging Limit is used for the disposition of a mill annealed alloy 600 steam generator tube for continued service that is experiencing predominately axially oriented outside diameter stress corrosion cracking confined within the thickness of the tube support plates.

Bounding calculation

NRC Editorial

At the flow distribution baffle intersections, at the cold leg support plate intersections, and at the hot leg support plate intersections with support plates ~~N~~ L through R (as identified in Figure 5.1 of WCAP-15163, Revision 1), the plugging (repair) limit is based on maintaining steam generator tube serviceability as described in a), b), c) and d) below:

a) Steam generator tubes, whose degradation is attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with bobbin voltage less than or equal to the lower voltage repair limit (Note 1), will be allowed to remain in service.

b) Steam generator tubes, whose degradation is attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than the lower voltage repair limit (Note 1), will be repaired or plugged, except as noted in 4.4.5.4.a.11.c below.

c) Steam generator tubes, with indications of potential degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than the lower voltage repair limit (Note 1) but less than or equal to the upper repair voltage limit (Note 2), may remain in service if a rotating pancake coil inspection does not detect degradation. Steam generator tubes, with indications of outside diameter stress corrosion cracking degradation with bobbin voltage greater than the upper voltage repair limit (Note 2) will be plugged or repaired.

d) If an unscheduled mid-cycle inspection is performed, the mid-cycle repair limits apply instead of the limits identified in 4.4.5.4.a.11.a, 4.4.5.4.a.11.b, and 4.4.5.4.a.11.c. The mid-cycle repair limits will be determined from the equations for mid-cycle repair limits of NRC Generic Letter 95-05, Attachment 2, page 3 of 7. Implementation of these mid-cycle repair limits should follow the same approach as in TS 4.4.5.4.a.11.a, 4.4.5.4.a.11.b, and 4.4.5.4.a.11.c.

Note 1: The lower voltage repair limit is 1.0 volt for 3/4-inch diameter tubing.

Info added from WCAP Table 2-2

Note 2: The upper voltage repair limit (V_{URL}) is calculated for each inspection according to the methodology in Generic Letter 95-05 as supplemented. V_{URL} may differ at the TSPs and flow distribution baffle. Voltage growth rate shall be the larger of the average growth rates experienced in the two prior cycles, but not less than 30% per effective full power year.

NRC comment

Bounding calculation

For Unit 2 Cycle 9 only, at the hot leg support plate intersections with support plates C, F, and J~~, L, and M~~ (as identified in Figure 5.1 of WCAP-15163, Revision 1), the plugging (repair) limit is based on maintaining steam generator tube serviceability as described in e) and f) below:

NRC Editorial

e) Steam generator tubes, whose degradation is attributed to axially oriented outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage less than or equal to 3.0 volts may remain in service.

NRC Editorial

f) Steam generator tubes, whose degradation is attributed to axially oriented outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than 3.0 volts shall be plugged or repaired regardless of whether or not a rotating pancake coil inspection detects degradation.

12) Tube Repair refers to a process that reestablishes tube serviceability for Model E steam generators only. Acceptable tube repair will be performed in accordance with the methods described in Westinghouse Reports WCAP-13698, Revision 2, "Laser Welded Sleeves for 3/4 Inch Diameter Tube Feedring-Type and Westinghouse Preheater Steam Generators," April 1995 and WCAP-14653, "Specific Application of Laser Welded Sleeves for South Texas Project Power Plant Steam Generators," June 1996, including post-weld stress relief;

Tube repair includes the removal of plugs that were previously installed as a corrective or preventive measure. A tube inspection per 4.4.5.4.a.9 is required prior to returning previously plugged tubes to service.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions [plug or (for Model E steam generators only) repair all tubes exceeding the plugging or repair limit and all tubes containing through-wall cracks] required by Table 4.4-2 and Table 4.4-3.

4.4.5.5 Reports

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged or repaired in each steam generator shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2;

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following the completion of the inspection. This Special Report shall include:

1) Number and extent of tubes inspected,

2) Location and percent of wall-thickness penetration for each indication of an imperfection, and

3) Identification of tubes plugged or repaired.

c. Results of steam generator tube inspections which-fall into Category C-3 shall be reported in a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days and prior to resumption of plant operation. This report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

d. For Model E steam generators, implementation of the voltage-based repair criteria to tube support plate intersections, notify the 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 steam line break) for the next operating cycle. The calculation(s) shall be done using:

Bounding calculation

Info added from WCAP Table 2-2

a) The methodology of Generic Letter 95-05 for intersections at the flow distribution baffles, at the applicable cold leg support plates, and at the hot leg support plates ~~N~~ L through R; and

SURVEILLANCE REQUIREMENTS (Continued)

Info added from WCAP Table 2-2

b) The methodology of Generic Letter 95-05 modified for potential overpressurized tubes as described in WCAP-15163, Revision 1, for hot leg intersections at support plates C, F, and J.

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

3) If indications are identified that extend beyond the confines of 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 1×10^{-2}, notify the NRC and provide an assessment of the safety significance of the occurrence. The calculation(s) shall be done using:

Info added from WCAP Table 2-2

Bounding calculation

a) The methodology of Generic Letter 95-05 for intersections at the flow distribution baffles, at the applicable cold leg support plates, and at the hot leg support plates ~~N~~ L through R; and

b) A total main steam line break tube burst probability of 1×10^{-5} ~~1.7×10^{-14}~~ for hot leg intersections at support plates C, F, and J.

NRC comment

6) If cracking is observed in the tube support plates.

7) If steam generator internals inspections are conducted and if indications detrimental to the integrity of the load path necessary to support the 3-volt alternate repair criteria are found, notify the NRC and provide an assessment of the safety significance of the occurrence.

Info added from WCAP Table 2-2

e. For Model E steam generators, submit a report to the Staff that addresses "Information to be Provided Following Each Restart" per Generic Letter 95-05, 6.b, within 90 days following outage breaker closure.

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~~ Notify NRC pursuant to 10CFR50.72(b)(3)(ii)	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~~ Notify NRC pursuant to 10CFR50.72(b)(3)(ii)	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.

Table 4.4-3

MODEL E STEAM GENERATOR REPAIRED TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required
A minimum of 20% of repaired tubes [1]	C-1	None	N.A.	N.A.
	C-2	Plug defective repaired tubes and inspect 100% of the repaired tubes in this S.G.	C-1	None
			C-2	Plug defective repaired tubes
			C-3	Perform action for C-3 result of first sample
	C-3	Inspect all repaired tubes in this S.G., plug defective repaired tubes and inspect 20% of the repaired tubes in each other S.G. ~~Notification to NRC pursuant to 50.72 (b)(2) of 10 CFR Part 50.~~ Notify NRC pursuant to 10CFR50.72(b)(3)(ii)	All other S.G.s are C-1	None
			Some S.G.s C-2 but no additional S.G. are C-3	Perform action for C-2 result of first sample
			Additional S.G. is C-3	Inspect all repaired tubes in each S.G. and plug defective repaired tubes. ~~Notification to NRC pursuant to 50.72 (b)(2) of 10 CFR Part 50.~~ Notify NRC pursuant to 10CFR50.72(b)(3)(ii)

[1] Each repair method is considered a separate population for determination of scope expansion.

ATTACHMENT 5

ANNOTATED
TECHNICAL SPECIFICATION
BASES

ANNOTATED TECHNICAL SPECIFICATIONS BASES

The following Technical Specification Bases pages are attached to show the specific change made as a result of the bounding calculation. The pages used for this mark-up are the reconstituted Technical Specification Bases pages provided in Attachment 6 of the original submittal.

Pages

B 3/4 4-2a (No changes)

B 3/4 4-3

B 3/4 4-3a (No changes)

B 3/4 4-4 (No changes)

RELIEF VALVES (Continued)

C. Manual control of the block valve to: (1) unblock an isolated PORV to allow it to be used for manual control of reactor coolant system pressure (Item A), and (2) isolate the PORV with excessive seat leakage (Item B).

D. Manual control allows a block valve to isolate a stuck-open PORV.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to minimize 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 3.4.6.2.c limitation of steam generator tube leakage between the Reactor Coolant System and the Secondary Coolant System. Cracks having a primary-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 primary-to-secondary leakage as low as 150 gallons per day per steam generator can readily be detected. 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 (for Model E steam generators only) repaired. Defective tubes in Model E steam generators may be repaired by a Westinghouse laser welded sleeve. The technical bases for sleeving repair are described in Westinghouse Reports WCAP-1 3698, Revision 2, "Laser Welded Sleeves for 3/4 Inch Diameter Tube Feedring-Type and Westinghouse Preheater Steam Generators, April 1995 and WCAP-14653, "Specific Application of Laser Welded Sleeves for South Texas Project Power Plant Steam Generators," June 1996.

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. Except as discussed below, plugging or (for Model E steam generators only) repair will be required for all tubes with imperfections exceeding the plugging or repair limit of 40% of the original tube nominal wall thickness. If a tube contain's a Westinghouse laser welded sleeve with imperfection exceeding 40% of nominal wall thickness, it must be plugged. The basis for the sleeve plugging limit for Model E steam generators is based on Regulatory Guide 1.121 analysis, and is described in the Westinghouse sleeving technical reports listed above. Steam generator tube inspections of operating

STEAM GENERATORS (Continued)

plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness. Repaired tubes are also included in the inservice tube inspection program.

For Model E steam generators only, the voltage-based repair limits of SR 4.4.5 implement the guidance in GIL 95-05 and are applicable only to Westinghouse-designed steam generators (SGs) with outside diameter stress corrosion cracking (ODSCC) located at the tube-to-tube support plate intersections. The criteria of GIL 95-05 are also applicable to the Unit 2 flow distribution plate intersections. The voltage-based repair limits are not applicable to other forms of SG tube degradation nor are they applicable to ODSCC that occurs at other locations within the SG. Additionally, the repair criteria apply only to indications where the degradation mechanism is dominantly axial ODSCC with no significant cracks extending outside the thickness of the support plate. Refer to GIL 95-05 for additional description of the degradation morphology.

Implementation of SR 4.4.5 for Model E steam generators requires a derivation of the voltage structural limit from the burst versus voltage empirical correlation and then 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 of the tube at flow distribution baffle intersections, (which have large tube to plate clearances) is based on a $3\Delta P_{NO}$ structural margin. For tubes at the cold leg tube support plate intersections and the hot leg intersections at plates ~~N~~ L through R for which the small clearances provide constraint against tube burst during normal operation, the structural limit is based on a $1.43\Delta P_{SLB}$ structural margin. For the hot leg intersections at plates C, F and J~~, L, and M~~ with the limited displacement of the lower tube support plates demonstrated by analyses in WCAP-15163, Rev. 1, Addendum 1, the constraint of the tube support plate reduces the burst probability of those tubes having axially oriented ODSCC indications that are confined within the tube support plate to negligible levels and the tube repair limit is not required to prevent tube burst. The need for tube repair is dictated by the need to satisfy allowable steam line break leakage limits.

Bounding calculation

For those intersections where the possibility of tube burst must be considered (i.e., at the flow distribution baffle, at cold leg intersections, and at the hot leg intersections at plates ~~N~~ L through R), the voltage structural limit must be adjusted downward to obtain the upper voltage repair limit 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:

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

NO CHANGES ON THIS PAGE

STEAM GENERATORS (Continued)

where V_{GR} represent 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 in SR 4.4.5.4.a. 11.e should only be used during unplanned inspections of Model E steam generators in which eddy current data is acquired for indications at the tube support plates.

SR 4.4.5.5 implements several reporting requirements for Model E steam generators recommended by GL 95-05 for situations which the NRC: wants to be notified prior to returning the SGs to service. For the purpose 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 SGs to service. Note that if leakage and conditional burst probability were calculated using the EOC voltage distribution for the purposes of addressing the GL section 6.a.1 and 6.a.3 reporting criteria, then the results of the projected EOC voltage distribution should be provided per the GL section 6.b.(c) criteria.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission in a Special Report pursuant to Specification 6.9.2 within 30 days -and 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.

OPERATIONAL LEAKAGE (Continued)

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 leakage limits incorporated into SR 4.4.6 are more restrictive than the standard operating leakage limits and were implemented in conjuction with the application of voltage-based repair criteria and laser-welded sleeving to Model E steam generators. They were intended to provide an additional margin to accommodate a crack which might grow at a greater than expected rate or unexpectedly extend outside the thickness of the tube support plate. Hence, the reduced leakage limit, when combined with an effective leak rate monitoring program, provides additional assurance that should a significant leak be experienced in service, it will be detected, and the plant shut down in a timely manner. The additional margin provided by the reduced leakage limit will be retained with the D94 steam generators.

The steam generator tube leakage limit of 150 gpd for each stearn generator 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 150 gpd limit per steam generator 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 stearn 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 specified allowed leakage from any RCS pressure isolation valve is sufficiently low to ensure early detection of possible inseries check valve failure. It is apparent that when pressure isolation is provided by two inseries check valves and when failure of one valve in the pair can go undetected for a substantial length of time, verification of valve integrity is required. Since these valves are important in preventing overpressurization and rupture of the ECCS low pressure piping which could result in a LOCA that bypasses containment, these valves should be tested periodically to ensure low probability of gross failure.

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valve is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining

ATTACHMENT 6

RECONSTITUTED TECHNICAL SPECIFICATION AND BASES PAGES

[Note: These pages represent the Technical Specification with amendments incorporated, and are provided for the reviewer's convenience.]

3/4.4.5 STEAM GENERATOR

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 generator(s) to OPERABLE status prior to increasing Tavg above 200°F.

SURVELLANCE 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 and Table 4.4-3. 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 reinspection. The tubes selected for each inservice inspection shall include at least 3% of the total number of nonrepaired tubes in all steam generators and (for Model E steam generators only) 20% of the total number of repaired 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:

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

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

3) A tube Inspection (pursuant to Specification 4.4.5.4a.9) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

4) For Model E steam generators only, 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.

c. The tubes selected as the second and third samples (if required by Table 4.4-2 or Table 4.4-3) during each inservice inspection may be subjected to a partial tube inspection provided:

1) The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found, and

2) The inspections include those portions of the tubes where imperfections were previously found.

d. For Model E steam generators only, implementation of the steam generator tube/tube support plate repair criteria requires a 100-percent bobbin coil inspection for the flow distribution baffle plate intersections, for the hot-leg tube support plate intersections, and for the 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-percent random sampling of tubes inspected over their full length.

1) All intersections with mechanically induced dent signals greater than 5 volts identified by bobbin coil inspection shall be inspected by rotating pancake coil (or equivalent).

2) All intersections with large mixed residuals that could potentially mask flaw responses at or above the voltage repair limits shall be inspected by rotating pancake coil (or equivalent).

3) At the flow distribution baffle intersections, at the cold leg support plate intersections, and at the hot leg support plate intersections with support plates L through R (as identified in Figure 5.1 of WCAP-15163, Revision 1), all distorted support plate indications (DSIs) with a bobbin voltage greater than the lower voltage repair limit (defined in 4.4.5.4.a.11) shall be inspected by rotating pancake coil (or equivalent).

4) At the hot leg support plate intersections with support plates C, F, and J (as identified in Figure 5.1 of WCAP-15163, Revision 1), all DSIs greater than 3 volts shall be examined by rotating pancake coil (or equivalent) eddy current probe. An additional 100 intersections with support plates C, F, and J with DSIs less than 3 volts (100 total over all steam generators, not necessarily selected at random) shall be inspected by rotating pancake coil (or equivalent).

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

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

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

SURVEILLANCE REQUIREMENTS (Continued)

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

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

   Note: Inservice inspection is not required during the steam generator replacement outage.

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

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

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

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

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

   4) A main steam line or feedwater line break.

4.4.5.4 Acceptance Criteria

a. As used in this specification:

1) Tubing or Tube means that portion of the tube or sleeve which forms the primary system to secondary system pressure boundary;

2) 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;

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

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

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

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

7) Plugging Limit or Repair Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging or (for Model E steam generators only) repaired by sleeving in the affected area because it may become unserviceable prior to the next inspection. The plugging or repair limit imperfection depths are specified in percentage of the nominal wall thickness as follows:

a. original tube wall	40%
b. Westinghouse laser welded sleeve wall	40%

For Model E steam generators, this definition does not apply to tube support plate intersections for which the voltage-based repair criteria are being applied. Refer to 4.4.5.4.a.11 for the repair limit applicable to these intersections.

8) 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;

9) 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;

10) 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.

11) For Model E steam generators only, Tube Support Plate Plugging Limit is used for the disposition of a mill annealed alloy 600 steam generator tube for continued service that is experiencing predominately axially oriented outside diameter stress corrosion cracking confined within the thickness of the tube support plates.

At the flow distribution baffle intersections, at the cold leg support plate intersections, and at the hot leg support plate intersections with support plates L through R (as identified in Figure 5.1 of WCAP-15163, Revision 1), the plugging (repair) limit is based on maintaining steam generator tube serviceability as described in a), b), c) and d) below:

a) Steam generator tubes, whose degradation is attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with bobbin voltage less than or equal to the lower voltage repair limit (Note 1), will be allowed to remain in service.

b) Steam generator tubes, whose degradation is attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than the lower voltage repair limit (Note 1), will be repaired or plugged, except as noted in 4.4.5.4.a.11.c below.

c) Steam generator tubes, with indications of potential degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than the lower voltage repair limit (Note 1) but less than or equal to the upper repair voltage limit (Note 2), may remain in service if a rotating pancake coil inspection does not detect degradation. Steam generator tubes, with indications of outside diameter stress corrosion cracking degradation with bobbin voltage greater than the upper voltage repair limit (Note 2) will be plugged or repaired.

d) If an unscheduled mid-cycle inspection is performed, the mid-cycle repair limits apply instead of the limits identified in 4.4.5.4.a.11.a, 4.4.5.4.a.11.b, and 4.4.5.4.a.11.c. The mid-cycle repair limits will be determined from the equations for mid-cycle repair limits of NRC Generic Letter 95-05, Attachment 2, page 3 of 7. Implementation of these mid-cycle repair limits should follow the same approach as in TS 4.4.5.4.a.11.a, 4.4.5.4.a.11.b, and 4.4.5.4.a.11.c.

Note 1: The lower voltage repair limit is 1.0 volt for 3/4-inch diameter tubing.

Note 2: The upper voltage repair limit (V_{URL}) is calculated for each inspection according to the methodology in Generic Letter 95-05 as supplemented. V_{URL} may differ at the TSPs and flow distribution baffle. Voltage growth rate shall be the larger of the average growth rates experienced in the two prior cycles, but not less than 30% per effective full power year.

For Unit 2 Cycle 9 only, at the hot leg support plate intersections with support plates C, F, and J (as identified in Figure 5.1 of WCAP-15163, Revision 1), the plugging (repair) limit is based on maintaining steam generator tube serviceability as described in e) and f) below:

e) Steam generator tubes, whose degradation is attributed to axially oriented outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage less than or equal to 3.0 volts may remain in service.

f) Steam generator tubes, whose degradation is attributed to axially oriented outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than 3.0 volts shall be plugged or repaired regardless of whether or not a rotating pancake coil inspection detects degradation.

12) Tube Repair refers to a process that reestablishes tube serviceability for Model E steam generators only. Acceptable tube repair will be performed in accordance with the methods described in Westinghouse Reports WCAP-13698, Revision 2, "Laser Welded Sleeves for 3/4 Inch Diameter Tube Feedring-Type and Westinghouse Preheater Steam Generators," April 1995 and WCAP-14653, "Specific Application of Laser Welded Sleeves for South Texas Project Power Plant Steam Generators," June 1996, including post-weld stress relief;

Tube repair includes the removal of plugs that were previously installed as a corrective or preventive measure. A tube inspection per 4.4.5.4.a.9 is required prior to returning previously plugged tubes to service.

SURVEILLANCE REQUIREMENTS (Continued)

b. The steam generator shall be determined OPERABLE after completing the corresponding actions [plug or (for Model E steam generators only) repair all tubes exceeding the plugging or repair limit and all tubes containing through-wall cracks] required by Table 4.4-2 and Table 4.4-3.

4.4.5.5 Reports

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged or repaired in each steam generator shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2;

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following the completion of the inspection. This Special Report shall include:

   1) Number and extent of tubes inspected,

   2) Location and percent of wall-thickness penetration for each indication of an imperfection, and

   3) Identification of tubes plugged or repaired.

c. Results of steam generator tube inspections which-fall into Category C-3 shall be reported in a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days and prior to resumption of plant operation. This report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

d. For Model E steam generators, implementation of the voltage-based repair criteria to tube support plate intersections, notify the 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 steam line break) for the next operating cycle. The calculation(s) shall be done using:

      a) The methodology of Generic Letter 95-05 for intersections at the flow distribution baffles, at the applicable cold leg support plates, and at the hot leg support plates L through R; and

b) The methodology of Generic Letter 95-05 modified for potential overpressurized tubes as described in WCAP-15163, Revision 1, for hot leg intersections at support plates C, F, and J.

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

3) If indications are identified that extend beyond the confines of 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 1×10^{-2}, notify the NRC and provide an assessment of the safety significance of the occurrence. The calculation(s) shall be done using:

a) The methodology of Generic Letter 95-05 for intersections at the flow distribution baffles, at the applicable cold leg support plates, and at the hot leg support plates L through R; and

b) A total main steam line break tube burst probability of 1×10^{-5} for hot leg intersections at support plates C, F, and J.

6) If cracking is observed in the tube support plates.

7) If steam generator internals inspections are conducted and if indications detrimental to the integrity of the load path necessary to support the 3-volt alternate repair criteria are found, notify the NRC and provide an assessment of the safety significance of the occurrence.

e. For Model E steam generators, submit a report to the Staff that addresses "Information to be Provided Following Each Restart" per Generic Letter 95-05, 6.b, within 90 days following outage breaker closure.

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. Notify NRC pursuant to 10CFR50.72(b)(3)(ii)	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. Notify NRC pursuant to 10CFR50.72(b)(3)(ii)	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.

Table 4.4-3

MODEL E STEAM GENERATOR REPAIRED TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required
A minimum of 20% of repaired tubes [1]	C-1	None	N.A.	N.A.
	C-2	Plug defective repaired tubes and inspect 100% of the repaired tubes in this S.G.	C-1	None
			C-2	Plug defective repaired tubes
			C-3	Perform action for C-3 result of first sample
	C-3	Inspect all repaired tubes in this S.G., plug defective repaired tubes and inspect 20% of the repaired tubes in each other S.G. Notify NRC pursuant to 10CFR50.72(b)(3)(ii)	All other S.G.s are C-1	None
			Some S.G.s C-2 but no additional S.G. are C-3	Perform action for C-2 result of first sample
			Additional S.G. is C-3	Inspect all repaired tubes in each S.G. and plug defective repaired tubes. Notify NRC pursuant to 10CFR50.72(b)(3)(ii)

[1] Each repair method is considered a separate population for determination of scope expansion.

STEAM GENERATORS (Continued)

plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness. Repaired tubes are also included in the inservice tube inspection program.

For Model E steam generators only, the voltage-based repair limits of SR 4.4.5 implement the guidance in GIL 95-05 and are applicable only to Westinghouse-designed steam generators (SGs) with outside diameter stress corrosion cracking (ODSCC) located at the tube-to-tube support plate intersections. The criteria of GIL 95-05 are also applicable to the Unit 2 flow distribution plate intersections. The voltage-based repair limits are not applicable to other forms of SG tube degradation nor are they applicable to ODSCC that occurs at other locations within the SG. Additionally, the repair criteria apply only to indications where the degradation mechanism is dominantly axial ODSCC with no significant cracks extending outside the thickness of the support plate. Refer to GIL 95-05 for additional description of the degradation morphology.

Implementation of SR 4.4.5 for Model E steam generators requires a derivation of the voltage structural limit from the burst versus voltage empirical correlation and then 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 of the tube at flow distribution baffle intersections, (which have large tube to plate clearances) is based on a $3\Delta P_{NO}$ structural margin. For tubes at the cold leg tube support plate intersections and the hot leg intersections at plates L through R for which the small clearances provide constraint against tube burst during normal operation, the structural limit is based on a $1.43\Delta P_{SLB}$ structural margin. For the hot leg intersections at plates C, F, and J with the limited displacement of the lower tube support plates demonstrated by analyses in WCAP-15163, Rev. 1, Addendum 1, the constraint of the tube support plate reduces the burst probability of those tubes having axially oriented ODSCC indications that are confined within the tube support plate to negligible levels and the tube repair limit is not required to prevent tube burst. The need for tube repair is dictated by the need to satisfy allowable steam line break leakage limits.

For those intersections where the possibility of tube burst must be considered (i.e., at the flow distribution baffle, at cold leg intersections, and at the hot leg intersections at plates L through R), the voltage structural limit must be adjusted downward to obtain the upper voltage repair limit 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} represent 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.