STP Nuclear Operating Company
South Texas Project Electric Generating Station P.O. Box 289 Wadsworth, Texas 77483

February 21, 2000
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10 CFR 2.790
10 CFR 50.90
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U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

South Texas Project
Unit 2
Docket No. STN 50-499
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

STP Nuclear Operating Company (STPNOC) proposes to revise South Texas Project (STP) technical specifications to implement a 3.0-volt alternate-repair-criteria (ARC) methodology for Unit 2 Model E steam generators. The new ARC methodology will apply only until we replace these steam generators during the outage currently scheduled to commence in fall of 2002.

STPNOC proposes to amend its license to apply 3.0-volt ARC for evaluation of intersections of tube hot-legs with selected tube-support-plates (TSP). It will apply only to intersections of TSP C through TSP M with tube hot-legs that exhibit outer-diameter-stress-corrosion-cracking (ODSCC) characteristics and only until the steam generators are replaced.

In support of the new ARC for STP Unit 2, STPNOC performed a tube integrity assessment. This assessment demonstrates that TSP C through M displacement is limited to < 0.15 inches during a design basis steam line break (SLB) event. The stability of these TSP reduces to negligible levels the likelihood of tube-burst at their intersection with tube hot-legs. Except for those indications restricted from burst at intersection of tube hot-legs with TSP C through M, STPNOC will calculate SLB primary-to-secondary leakage as free-span leakage.

The new ARC will provide an improved method for evaluating irregularities in steam generator tube outer surfaces that exhibit ODSCC characteristics in stable and heavily supported sections of the tubes. Use of this improved methodology also reduces thermal cycling of the reactor-coolant-pressure-boundary by avoiding unnecessary compliance-driven mid-cycle inspections.

Note that all references in this submittal to rotating pancake coil (RPC) probes also refer to other probes that are the functional equivalent thereof.

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Attached to this letter are documents requiring special handling. These are:

- CAW-99-1328, Westinghouse authorization letter, affidavit, Proprietary Information Notice and Copyright Notice (Attachment 7),
- EPRI letter, "Request for Withholding of Proprietary Document; 10CFR2.790(a)(4)," from Govinda Srikantiah to U.S. Nuclear Regulatory Commission, dated February 1, 2000, and associated EPRI affidavit, signed and dated February 1, 2000 (Attachment 8)
- WCAP-15163, Westinghouse topical report, proprietary (Attachment 9),
- WCAP-15164, Westinghouse topical report, non-proprietary (Attachment 10),
- TR-107625, "SG Indications Restricted from Burst (IRB) Leak Test Report," Final Report, September 1998, Electric Power Research Institute (EPRI), proprietary (Attachment 11),
- TR-107625, "SG Indications Restricted from Burst (IRB) Leak Test Report," Final Report, September 1998, Electric Power Research Institute (EPRI), non-proprietary (Attachment 12),

Five copies each of each of these documents are included for distribution to and use of NRC staff in review of this request.

WCAP-15163 contains information proprietary to Westinghouse Electric Company, and is accompanied by an affidavit signed by Westinghouse, the owner of the information. The affidavit sets forth the basis upon which information contained in WCAP-15163 may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.790 of Title 10 of the Code of Federal Regulations (10 CFR).

TR-107625 contains information proprietary to the Electric Power Research Institute (EPRI), and is accompanied by an affidavit signed by EPRI, the owner of the information. The affidavit sets forth the basis upon which information contained in TR-107625 may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.790 of Title 10 of the Code of Federal Regulations (10 CFR).

Accordingly, it is respectfully requested that the information, which is proprietary to Westinghouse or EPRI, be withheld from public disclosure in accordance with 10 CFR 2.790.

Correspondence with respect to the copyright on proprietary aspects of the Westinghouse WCAP topical reports listed above or the supporting Westinghouse affidavit should reference CAW-99-1328 and should be addressed to N. J. Liparulo, Manager of Equipment Design and Regulatory Engineering, Westinghouse Electric Company, P. O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Correspondence with respect to the copyright on proprietary aspects of the EPRI TR-107625 report listed above or the supporting EPRI affidavit should reference and should be addressed to Electric Power Research Institute, Hillview Drive, Palo Alto, CA.

STPNOC has reviewed the proposed amendment pursuant to 10CFR50.92 and determined that it does not involve a significant hazard. In addition, STPNOC has determined that the proposed amendment satisfies criteria of 10CFR51.22(c)(9) for categorical exclusion from environmental assessment. The STP Plant Operations Review Committee and the Nuclear Safety Review Board have reviewed and approved the proposed amendment.

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

STPNOC requests that the NRC review and approve this proposed amendment by November 30, 2000, and also requests the customary 30 day implementation period.

If questions arise regarding this proposed amendment, please contact either Mr. S. E. Thomas at (361) 972-7162 or me at (361) 972-8757.

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

JJS/SMH/MTVN

Attachments:

1. Affidavit
2. Description of Technical Specification Changes with Safety Evaluation
3. Determination of No Significant Hazard
4. Annotated Technical Specifications
5. Annotated Technical Specification Bases
6. Reconstituted Technical Specification and Bases Pages
7. Westinghouse letter CAW-99-1328, "APPLICATION FOR WITHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE," dated March 23, 1999
8. EPRI letter, "Request for Withholding of Proprietary Document; 10CFR2.790(a)(4)," from Govinda Srikantiah to U.S. Nuclear Regulatory Commission, dated February 1, 2000, and associated EPRI affidavit, signed and dated February 1, 2000
9. WCAP-15163, Revision 1, "Technical support for Implementing High Voltage Alternate Repair Criteria at Hot Leg Limited Displacement TSP Intersections for South Texas Plant Unit 2, Model E Steam Generator," March 1999, (Proprietary)
10. WCAP-15164, Revision 1, "Technical support for Implementing High Voltage Alternate Repair Criteria at Hot Leg Limited Displacement TSP Intersections for South Texas Plant Unit 2, Model E Steam Generator," March 1999, (Non-Proprietary)
11. TR-107625, "SG Indications Restricted from Burst (IRB) Leak Test Report," Final Report, Electric Power Research Institute (EPRI), September 1998, (Proprietary)
12. TR-107625, "SG Indications Restricted from Burst (IRB) Leak Test Report, Final Report, Electric Power Research Institute (EPRI)," September 1998, (Non-Proprietary)

cc:

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* U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
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* Attachments 9 through 12 of Westinghouse and EPRI reports included.

ATTACHMENT 1

AFFIDAVIT

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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

AFFIDAVIT

I, J. J. Sheppard, being duly sworn, hereby depose and say that I am Vice President, Nuclear Engineering and Technical Services, of STP Nuclear Operating Company; that I am duly authorized to sign and file with the Nuclear Regulatory Commission the attached proposed 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

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

STATE OF TEXAS	)
	)
COUNTY OF	)

Subscribed and sworn to before me, a Notary Public in and for the State of Texas, this 21st day of February, 2000.

NOTARY PUBLIC STATE OF TEXAS
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 through M. 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 through M in Westinghouse Model E steam generators during a steam line break (SLB) event has been demonstrated to be < 0.15 inch. 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 tube eddy-current inspection indications of ≤ 3.0 volts can be left in service if found at intersections of tube hot-legs with TSP C through M. 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 through M 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 through M 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 through M 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.

Note 1 and Note 2 on TS page 3/4 4-16a are incorrectly aligned with the left margin and should be realigned. This change has no effect other than to indent the notes for proper alignment to facilitate ease of reading.

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

1) 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.

2) 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 through M.

3) 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.

4) 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 10 CFR 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 through M
  - Although TSP R meets displacement criteria for the 3-volt repair limit, we exclude it for convenience because it is located above TSP Q.
- Repair of flaws detected at intersections of tube hot-legs and TSP C through M 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 residuals 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 through M 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 through M 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 through M 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 Section 4.0 of GL 95-05, entitled "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, "In Service 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 through M during a design basis SLB is sufficiently limited to essentially preclude tube burst at the locations of their intersections with tubes (Reference 1).

Although a TSP displacement of approximately 0.3 inch during a design basis SLB limits total axial tube burst probability to a negligible 10^{-5}, displacement of ≤ 0.15 inch yields a tube burst probability of less than 10^{-10} for a single tube. These bounding tube burst probabilities for South Texas Project Unit 2 steam generators conservatively assume that all intersections of tube hot-legs with TSP have through-wall cracks and that these cracks will be exposed by TSP displacement. However, when TSP C through M are subjected to transient dynamic loads during a design basis SLB, displacement is shown to be <0.15 inch even though the hydraulic loading effect is calculated using a conservative multiplier of 1.5.

Dynamic loading of a TSP is proportional to the hydraulic pressure drop across the TSP. Hydraulic TSP loads used in TSP displacement analyses were obtained using RELAP 5/MOD 3 Code. RELAP/MOD 3 Code simulates a two fluid, non-equilibrium system condition. Use of the calculated pressure differences, with a 1.5 multiplier to calculate plate loading, envelops the results of calculation sensitivity studies and conservatively bounds plate loading that might occur during a steam line break event. 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.
- 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.

As noted above, 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 are 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 in 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. Analysis shows that on the tube hot-leg side only Plates N, P and Q exceed the conservatively selected limit of 0.150 inch during an SLB at hot standby. Only plate Q exceeds this displacement limit for an SLB at full power.

For ODSCC occurring within the thickness of TSP, STPNOC implemented voltage-based ARC for Unit 2 according to the guidelines of NRC GL 95-05. For steam generators with 3/4" tubing, a conservative ARC limit of 1.0 volt is 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 through M, results of analyses (Reference 1) reveal that TSP stability limits tube burst probability to $< 10^{-10}$ for a single tube. 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 during TSP displacement. Axial tube burst probability is negligibly small when TSP displacement is limited to ≤ 0.15 inch; 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 the 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 factor of safety 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 through M, 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 through M 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 through M 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.

SLB Leak Rate and Tube Burst Probability Considerations

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 included 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 through M 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 inspections. 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, we 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, and 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 through M have been shown to deflect <0.15 inches during the limiting SLB event. The affect of <0.15 inches deflection on probability-of-burst is negligible and, thus, cracks at intersections with TSP C through M 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 through M.

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.

In these tests it was 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 valves (PORV) setpoints, maximum pressure differential for South Texas Project Unit 2 is 2405 psid. At 2405 psid the bounding leak rate is then 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 displacement limit of 0.15 inch proposed for high voltage ARC. 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 throughwall 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 through M 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 indication restrained from burst (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 FDB, and tube hot-leg indications at TSP N through R. Burst calculations for tube hot-leg intersections with TSP C through M 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 NRC GL 95-05 reporting guidelines 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. ASME Code minimum yield stresses for stayrods and spacers are 34.0 ksi and 26.9 ksi, respectively. Model E stayrod and spacer stresses are well within 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

Inspection of steam generator internals confirms that components and structures forming the load path credited in use of 3-volt ARC for Unit 2 have not been degraded. These inspections also verify that actual TSP displacement is not greater than that calculated by the TSP displacement analysis. STPNOC performed a variety of secondary side inspections during past refueling outages and found no degradation of the load path. Tables 10.1 and 10.2 of Reference 1 summarize results of these inspections. STPNOC will continue to perform inspections, including one during the outage implementing 3-volt ARC, or during the preceding outage. Visual portions of these inspections will emphasize the lower TSP regions where most ODSCC occurs, and will be complete to the extent that access to components, radiation exposure, and risk of leaving foreign objects in the steam generator, will allow. Tables 10.4 and 10.5 list the additional inspections.

Stainless steel TSP in the South Texas Project Unit 2 steam generators eliminate 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
2. STPNOC letter dated 7/15/98, NOC-AE-000228, Response to NRC Request for Additional Information
3. TR-107625, SG Indications Restricted from Burst (IRB) Leak Test Report, Final Report, EPRI, September 1998

ATTACHMENT 3

DETERMINATION OF NO SIGNIFICANT HAZARD

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

In accordance with the criteria set forth in 10CFR50.92, the STP Nuclear Operating Company (STPNOC) has evaluated 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. For TSP C through M, maximum displacement during a postulated SLB event is less than 0.15 inch. Because TSP C through M 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 through M is independent of voltage related growth rates and the proposed 3-volt ARC is compliant with RG 1.121 criteria.

   Given a TSP displacement of < 0.15 inch, tube hot-leg spans enclosed within TSP C through M have a negligible tube burst probability of less than 10^{-10} for a single tube. This is eight 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 tube hot-leg spans intersecting TSP C through M. 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 through M.

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 with FDB, hot-leg intersections with TSP N through R, and indications found at intersections of tube cold-legs with any TSP.

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.

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.

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, as discussed above in the Safety Evaluation section, in the part on "SLB Leak Rate and Tube Burst Probability Considerations."

Inspections conducted in accordance with RG 1.83, Rev. 1, using 3-volt ARC for intersections of tube hot-legs with TSP C through M, and using 1-volt ARC at remaining hot-leg and 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.

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 TSP C through M is acceptable and presents no significant hazard, as defined in 10 CFR 50.92.

ATTACHMENT 4

ANNOTATED

TECHNICAL SPECIFICATIONS

ANNOTATED TECHNICAL SPECIFICATIONS

The below listed Technical Specification pages are annotated to show the specific changes proposed by this request and are attached herewith. Page 3/4 – 16c was added to accommodate text overflow from previous page. Other included pages, not listed below, provide context to aid review of the request, but are not changed.

Pages:

3/4 – 16

3/4 – 16a

3/4 – 16b (received unchanged text overflow from previous page)

3/4 – 16c (page added to accommodate overflow from previous page)

INSERT A:

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 through R, the plugging (repair) limit is based on maintaining steam generator tube serviceability as described in a), b), c) and d) below:

INSERT B:

At the hot leg support plate intersections with support plates C through M, 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 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 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.

NO CHANGE

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 tube support plate intersections, the plugging (repair) limit is based on maintaining steam generator tube serviceability as described below:~~

INSERT A

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.

INDENT NOTES

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 according to the methodology in Generic Letter 95-05 as supplemented. V_{URL} may differ at the TSPs and flow distribution baffle.

INSERT B

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;

TEXT ON THIS PAGE SHIFTED DOWN BY OVERFLOW FROM PREVIOUS PAGE. NO CHANGES TO TEXT CONTENT.

SURVEILLANCE REQUIREMENTS (Continued)

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.

   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 x 10^{-2}, notify the NRC and provide an assessment of the safety significance of the occurrence.

ATTACHMENT 5

ANNOTATED

TECHNICAL SPECIFICATION

BASES

ANNOTATED TECHNICAL SPECIFICATIONS BASES

The below listed Technical Specification Bases pages are annotated to show the specific changes proposed by this request and are attached herewith. Other included pages, not listed below, provide context to aid review of the request, but are not changed.

Pages:

B3/4 4–3

INSERT C:

The structural limit for the flow distribution baffle intersections (which have large tube to plate clearances) is based on a $3\Delta P_{NO}$ structural margin. For the cold leg tube support plate intersections and the hot leg intersections at plates N through R for which the small clearances provide constraint against tube burst during normal operation, the structural limit is based on a 1.43 ΔP_{SLB} structural margin. For the hot leg intersections at plates C through M with the limited displacement of the lower tube support plates demonstrated by analyses in WCAP-15163, Rev. 1, the constraint of the tube support plate reduces the tube burst probability 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 N 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.

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-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.

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 contains 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 GL 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 GL 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 GL 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).

INSERT C

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

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

where V_{GR} 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.

No CHANGE

STEAM GENERATORS (Continued)

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.

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.

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.]

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 N through R, 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.12.a, 4.4.5.4.a.12.b, and 4.4.5.4.a.12.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.12.a, 4.4.5.4.a.12.b, and 4.4.5.4.a.12.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 according to the methodology in Generic Letter 95-05 as supplemented. V_{URL} may differ at the TSPs and flow distribution baffle.

At the hot leg support plate intersections with support plates C through M, 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 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 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.

   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.

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 GL 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 GL 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 GL 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 structural limit for the flow distribution baffle intersections (which have large tube to plate clearances) is based on a $3\Delta P_{NO}$ structural margin. For the cold leg tube support plate intersections and the hot leg intersections at plates N 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 through M with the limited displacement of the lower tube support plates demonstrated by analyses in WCAP-15163, Rev. 1, the constraint of the tube support plate reduces the tube burst probability 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 N 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.

ATTACHMENT 7

WESTINGHOUSE LETTER CAW-99-1328, "APPLICATION FOR WITHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE," DATED MARCH 23, 1999

Westinghouse Electric Company,
a division of CBS Corporation

Box 355
Pittsburgh Pennsylvania 15230-0355

March 23, 1999

CAW-99-1328

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

Attention: Mr. Samuel J. Collins

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject WCAP-15163, "Technical Support for Implementing High Voltage Alternate Repair Criteria at Hot Leg Limited Displacement TSP Intersections for South Texas Plant Unit 2 Model E Steam Generator", March 1999

Dear Mr. Collins:

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-99-1328 signed by the owner of the proprietary information, Westinghouse Electric Company, a division of CBS Corporation ("Westinghouse"). The affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.790 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying Affidavit by South Texas Project Nuclear Operating Company.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-99-1328, and should be addressed to the undersigned.

Very truly yours,

H. A. Sepp, Manager
Regulatory and Licensing Engineering

Enclosures

cc: T. Carter/NRC (5E7)

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared Henry A. Sepp, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company, a division of CBS Corporation ("Westinghouse"), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

Henry A. Sepp

Henry A. Sepp, Manager
Regulatory and Licensing Engineering

Sworn to and subscribed
before me this 23rd day
of March, 1999

Janet A. Schwab
Notary Public

Notarial Seal
Janet A. Schwab, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires May 22, 2000
Member, Pennsylvania Association of Notaries

(1) I am Manager, Regulatory and Licensing Engineering, in the Nuclear Services Division, of the Westinghouse Electric Company, a division of CBS Corporation ("Westinghouse"), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rulemaking proceedings, and am authorized to apply for its withholding on behalf of the Westinghouse Energy Systems Business Unit.

(2) I am making this Affidavit in conformance with the provisions of 10CFR Section 2.790 of the Commission's regulations and in conjunction with the Westinghouse application for withholding accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by the Westinghouse Energy Systems Business Unit in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.790 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information which is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10CFR Section 2.790, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in WCAP-15163, "Technical Support for Implementing High Voltage Alternate Repair Criteria at Hot Leg Limited Displacement TSP Intersections for South Texas Plant Unit 2 Model E Steam Generator", March 1999 transmitted by the South Texas Project Nuclear Operating Company letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk, Attention Mr. Samuel J. Collins. The proprietary information as submitted for use the South Texas Project Nuclear Operating Company for the South Texas Project Nuclear Power Plant Unit 2 is expected to be applicable in other licensee submittals in response to certain NRC requirements for implementation of the 3 volt alternate repair criteria at hot leg tube support plate intersections.

This information is part of that which will enable Westinghouse to:

(a) Provide justification for application of higher voltage alternate repair criteria for ODSCC at tube support plates.

(b) Provide complete licensing packages to support license amendments.

Further this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of similar information to its customers for purposes of implementing higher voltage alternate repair criteria at TSPs.

(b) Westinghouse can sell support and defense of the technology to its customers in the licensing process.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar licensing services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended for developing testing and analytical methods and performing tests.

Further the deponent sayeth not.

Proprietary Information Notice

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.790 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification of claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) contained within parentheses located as a subscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower cases letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.790(b)(1).

Copyright Notice

ATTACHMENT 8

EPRI LETTER,
"REQUEST FOR WITHHOLDING OF PROPRIETARY DOCUMENT; 10CFR2.790(A)(4),"
FROM GOVINDA SRIKANTIAH TO U.S. NUCLEAR REGULATORY COMMISSION,
DATED FEBRUARY 1, 2000,
AND ASSOCIATED EPRI AFFIDAVIT,
SIGNED AND DATED FEBRUARY 1, 2000

EPRI

February 1, 2000

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Request for Withholding of Proprietary Document; 10CFR2.790(a)(4)

Dear Sir or Madam:

This is a request under 10CFR2.790(a)(4) that the NRC withhold from public disclosure the information identified in the enclosed affidavit consisting of EPRI owned Proprietary Information contained within EPRI Report TR-107625 and entitled "Steam Generator Indications Restricted from Burst (IRN) Leak Test Report." The report is referred to below as the "Information." The Information is marked PROPRIETARY and CONFIDENTIAL.

The Report is for NRC's internal use and may be used only for the purpose for which it is disclosed by EPRI. The Report should not be otherwise used or disclosed to any person outside the NRC without prior written permission from EPRI.

Enclosed is an affidavit to support EPRI's request for withholding. If you have any questions relating to the propriety of this request or the enclosed affidavit, please do not hesitate to contact Mark D. Fox at (650) 855-8957. Questions on the technical aspects of the Information may be addressed to me at (650) 855-2109.

Yours truly,

G Srikantiah

Govinda Srikantiah
Manager, Steam Generator Dynamics
Science & Technology division

Enclosures

EPRI

AFFIDAVIT

RE: Request for Withholding of Proprietary Document; 10CFR2.790(a)(4) TR-107625 "Steam Generator Indications Restricted from Burst (IRB) Leak Test Report."

I, JOHN BATEMAN, being duly sworn, depose and state as follows:

I am the Director of the Science & Technology Division at Electric Power Research Institute ("EPRI") and I have been specifically delegated responsibility for reviewing the records, documents and information sought under this affidavit to be withheld (the "Information") and authorized to apply for their withholding on behalf of EPRI. This affidavit is submitted to the Nuclear Regulatory Commission ("NRC") pursuant to 10 CFR 2.790 (a)(4) based on the fact that the Information consists of trade secrets of EPRI owned by EPRI and that the NRC will receive the Information from EPRI under privilege and in confidence.

The Information, which EPRI requests the NRC to withhold, consists of EPRI owned Proprietary Information contained within EPRI Report TR-107625 and entitled "Steam Generator Indications Restricted from Burst (IRB) Leak Test Report." The Information has been marked as PROPRIETARY AND CONFIDENTIAL.

EPRI desires to disclose the Information to the NRC for informational purposes to assist the NRC. EPRI would welcome any discussions between EPRI and the NRC related to the Information that the NRC desires to conduct.

The basis for which the Report should be withheld from the public is set forth below:

(i) The Information has been held in confidence by EPRI. EPRI intends to provide copies of the Information to EPRI members and to one or more EPRI contractors. EPRI members and contractors are bound by confidentiality agreements to preserve the confidentiality of proprietary and confidential documents received from EPRI. Receipt of the Information by such members and contractors will not impair the proprietary and confidential nature of the Information nor will such receipt impair the value of the Information as trade secrets. In addition, EPRI may license the Information to organizations that are not EPRI members.

(ii) The Information is of a type customarily held in confidence by EPRI and there is a rational basis therefor. The Information is of a type that EPRI considers to be trade secrets. Such Information is customarily held in confidence by EPRI because to disclose it would prevent EPRI from licensing the Information at fees which would allow EPRI to recover its investment. If consultants and other businesses providing services in the electric power industry were able to obtain the Information, they would be able to use it commercially for profit and avoid spending the large amount of money that EPRI was required to spend to obtain the Information. The rational basis that EPRI has for classifying information as a trade secrets is the Uniform Trade Secrets Act which California adopted in 1984 and which has been adopted by over twenty states. The Uniform Trade Secrets Act defines a "trade secret" as follows:

> "Trade secret" means information, including a formula, pattern, compilation, program, device, method, technique, or process, that:
>
> > (1) Derives independent economic value, actual or potential, from not being generally known to the public or to other persons who can obtain economic value from its disclosure or use; and
> >
> > (2) Is the subject of efforts that are reasonable under the circumstances to maintain its secrecy.

(iii) The Information will be transmitted and received by the NRC in confidence. The purpose is to maintain the confidentiality of the Information.

(iv) The Information is not available in public sources. EPRI developed the Information only after making a determination that the Information was not available from public sources. EPRI was required to spend a large amount of money through payments to contractors. In addition, EPRI was required to use a large amount of time of EPRI employees. Finally, the Information was developed only after a long period of effort.

(v) A public disclosure of the Information would be highly likely to cause substantial harm to EPRI's competitive position. The Information can be properly acquired or duplicated by others only with an equivalent investment of time and effort.

EPRI

I have read the foregoing and the matters stated therein are true and correct to the best of my knowledge, information and belief. I make this affidavit under penalty of perjury under the laws of the United States of America and under the laws of the State of California.

Executed at 3412 Hillview Avenue, Palo Alto, being premises and place of business of the Electric Power Research Institute:

February 1, 2000

John Bateman

Subscribed and sworn before me this day: February 1, 2000

Sumi Yamashita, Notary Public

MGC-1
THE GREAT SEAL OF THE STATE OF CALIFORNIA
EUREKA
Sumi Yamashita
Comm. #1093096
NOTARY PUBLIC - CALIFORNIA
SANTA CLARA COUNTY
Comm. Exp. March 31, 2000