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

July 3, 2002
NOC-AE-02001355
10CFR50.90

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852

South Texas Project
Unit 1
Docket No. STN 50-498
Response to Request for Additional Information

Reference: Letter, T. J. Jordan to NRC Document Control Desk, "License Amendment Request - Revised Proposed Amendment to Technical Specification 4.4.5.3a," dated June 20, 2002 (NOC-AE-02001351)

The referenced letter submitted a license amendment request for a change in the steam generator inservice inspection frequency requirements in Technical Specification (TS) 4.4.5.3a for South Texas Project Unit 1. The change would allow a one-time inspection interval of once per 40 months for the steam generator tube inspection performed immediately following 1RE10.

The NRC staff has informally requested additional information regarding the license amendment request. The attachment to this letter provides our response. If there are any questions regarding this response, please contact Mr. Mark Kanavos, Manager, Modifications and Design Basis Engineering at (361) 972-7181 or me at (361) 972-7902.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on: July 3, 2002

T. J. Jordan
Vice President,
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jtc

Attachment: Response to Request for Additional Information

STI: 31463776 A047

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ATTACHMENT

Response to Request for Additional Information

1. ***During a conference call with the licensee on June 17, 2002, the staff expressed a concern with reviewing the amendment for STP Unit 2. The staff recognizes the STP Unit 2 steam generators will be replaced with the same model as the Unit 1 steam generators; however, the staff explained to the licensee that the preservice and first inservice inspection results are critical information required for our evaluation of this type of technical specification change request. The licensee has not, yet, performed the first inservice inspection. To facilitate the staff's review of the Unit 1 request, the licensee indicated they would withdraw the amendment request for Unit 2. If the licensee were to revise this decision, additional staff review and questions would be necessary.***

Response

STP Nuclear Operating Company submitted a revised license amendment request (Reference 1) to apply to Unit 1 only.

2. ***In Attachment 1 of the submittal, the licensee briefly describes the steam generator (SG) inspection scope for STP Unit 1 during the 1RE10 outage as a full-length inspection of 100% of the steam generator tubes. The staff assumes this inspection scope refers to a bobbin coil probe inspection. The staff requests the licensee to provide a more detailed discussion of the inspection scope, including the following information:***

   a. ***Describe the scope of any rotating pancake coil (RPC) probe inspections including RPC of the hot leg top-of-tubesheet region, the low-row U-bend tubes, dings and dents. If any of these regions were not inspected, please provide the technical justification for excluding these areas from the inspection scope during the 1RE10 outage.***

Response

Rotating pancake coil (+Point) testing of all bobbin possible indications (I-codes) was performed in all four steam generators (SGs) during 1RE10. Manufacturing buff marks (MBMs) and dents/dings (> 0.75 volt) were compared to the baseline signals, and if any changes were detected, they were assigned I-codes for further +Point examination.

Neither full nor sample +Point inspections were performed at the hot leg top-of-tubesheet (TTS) expansion area, at low-row U-bends, or at dings and dents, unless I-codes were located at these areas. The technical bases for not including these areas are described in the South Texas 1RE10 Degradation Assessment, Appendix F (Reference 2), which was faxed to the NRC on June 17, 2002. To summarize Appendix F, stress corrosion cracking is not regarded as a potential damage mechanism at the TTS transition, low-row U-bends, or in dents and dings in the Delta 94 SGs. The bobbin probe inspection is regarded as qualified for detection of the volumetric indications. To date there has been no confirmed identification of stress corrosion cracking in the Delta Series, Model F, or F-Type replacement SGs currently in service.

b. Please include a discussion of the 75 +Point coil examinations that were mentioned in the submittal.

Response

Included in the 1RE10 +Point examination were five I-codes at dents, nine at dings, and 32 at MBMs. Twenty-eight non-quantifiable possible indications (NQIs) were found at locations other than at dents, dings, and MBMs. One possible loose part (PLP) was examined by +Point in SG "D" at support plate 6 and the six surrounding tubes at support plate 6 were examined by +Point to confirm no additional loose part signals. These I-codes were inspected by +Point at the following tube locations:

- Hot leg (HL) free span 45 inspections
- Cold leg (CL) free span 23 inspections
- HL support plate 8 inspections
- CL support plate 4 inspections
- Row 2 U-bend 1 inspection
- CL TTS 2 inspections

No defects or degradation were confirmed by any of these +Point inspections. The details of the inspection results were provided to the NRC in Reference 3.

3. In the submittal, the licensee states that "[n]o defective or degraded tubes were indicated" during the 1RE10 outage of STP Unit 1. Utilizing the Technical Specification definition of degraded and defective would imply that no tubes contained defects greater than 20% throughwall.

a. Did the inspection results reveal defects (i.e., service-induced degradation) less than 20% throughwall? If so, please list all indications that were identified and sizing estimates, if appropriate.

Response

No wear or service-induced wall reduction was detected in any of the four SGs during 1RE10.

b. Describe inspection results identified that were not attributed to service-induced degradation and were left in service (i.e., dents, manufacturing burnish marks, etc.). Briefly describe the actions taken that led to the decision to leave them in service (e.g., history review, supplemental eddy current inspection, etc.).

Response

Manufacturing buff marks greater than or equal to one volt on the 150 kHz absolute that were found to be in the baseline bobbin data results and had not changed were re-identified as MBMs and left in service. If the signal was new or a change had occurred, the signal was called a manufacturing buff mark possible indication (MBI) and subjected to +Point examination. MBMs less than one volt were also compared to the preservice inspection (PSI) database for change and were called MBIs if change had occurred. These were further examined by +Point. No cracks or degradation were detected in any MBM and all MBMs were left in service.

Dings and dents greater than 0.75 volts on the 630/150 kHz mix were monitored for distortion or change from the baseline. Those that exhibited change were reported as dent/ding possible indications (DNIs) and subjected to +Point testing. All NQI signals were also compared to the baseline data. Those that either changed or did not exist in the baseline data were subjected to +Point examination. There were no crack-like indications detected by +Point examination in any of the four SGs.

4. In the submittal, the operating conditions of V. C. Summer recirculating steam generators (RSGs) and Westinghouse Delta model RSGs are compared to the current STP Unit 2 model "E" steam generators. The staff assumes the licensee believes the operating conditions are also similar to STP Unit 1. Please provide information that shows that the operating conditions of STP Unit 1 are similar to those of V. C. Summer and other Westinghouse Delta model RSGs.

Response

The operating conditions that most affect stress corrosion cracking are temperature, primary and secondary chemistry, and to a much lesser degree, tube differential pressure. The average hot leg temperatures do not vary more than twelve degrees among these plants:

South Texas	620°F
Shearon Harris	620°F
V. C. Summer	619°F
Arkansas - 2	608°F

Although each plant has its own method of chemistry control, each operates within the strict bounds of the EPRI Primary and Secondary Chemistry Guidelines. If SG tube degradation or cracking were identified in one of these plants, a detailed comparison between the subject plant and STP would be performed in accordance with the requirement of the EPRI Guidelines to determine susceptibility for similar degradation. If extended operation based upon this evaluation could not be justified, then a SG inspection would be performed at the next scheduled refueling outage.

5. ***In the submittal, the licensee states that Westinghouse Delta model RSGs have been stress corrosion free after six calendar years of operation.***

   a. ***The staff requests the licensee to list the other Delta models RSGs, other than STP Unit 1 and V. C. Summer, to which the submittal referred and to discuss their operating experience.***

Response

The six years of operation referred to V. C. Summer. However, subsequent to the initial STP amendment request, inspection after the first operating cycle at Arkansas-2 was completed in April 2002. The only reported tube degradation was wear on two tubes (4% and 18%) from a small loose part that was removed. Arkansas-2 does not have the same type of feedwater spray nozzles as described in the response to Question 7. Thus, the STP feedwater spray nozzle would have captured a loose part similar to the one removed at Arkansas-2.

   b. ***What process is in place to gain relevant industry operating experience that may effect the operational assessment of STP Unit 1 during an extended inspection interval?***

Response

The STP SG Engineering organization maintains close communications with other utility representatives through direct participation on EPRI SGMP member committees. If degradation is discovered at another station with thermally treated Alloy 690 SG tubing, it would be immediately communicated to our Engineers, and an Operational Assessment would be performed to determine susceptibility at STP and whether an inspection would be required.

6. ***In the submittal, the licensee highlights the results of the October 2000 V. C. Summer inspection results. The V. C. Summer replacement steam generators have been in service since 1994. What are the results from 1994 through 2000 with respect to service-induced degradation? Provide a list including outage identified, flaw type, location and sizing estimates.***

Response

Refer to Table 1 at the end of this Attachment.

7. ***In the submittal, the licensee states that the Delta 94 steam generator design incorporates features to minimize the development of loose parts during operation and maintenance. Briefly respond to the following related questions.***

   a. How do these design features minimize the development of loose parts?

Response

Feedwater enters the Delta 94 SG through 34 spray nozzles on a ring header. The spray nozzles are vertical cylinders on the top of the feedring, each with an outside diameter of 2.875 inches and height of approximately 6 inches. There are 130 holes with a diameter of 0.29 inch in the outer surface of each cylindrical nozzle. The nozzles will trap loose parts that might be introduced into the SG from the feedwater system. As described in the submittal, small objects that would pass through the 0.29-inch holes would flow through the tube bundle (gap of 0.293 inch between tubes) and would be unlikely to produce tube wear.

Likewise, auxiliary feedwater is introduced into the SG through a single cylindrical nozzle with a diameter of 6.625 inches and height of approximately 13.5 inches. There are 560 holes with a diameter of 0.29 inch in the outer surface of the cylindrical nozzle. Like the feedwater spray nozzles, the auxiliary feedwater spray nozzle will trap foreign objects that are large enough to produce tube wear.

The use of threaded members inside the SG has been minimized. In all cases threaded members are secured by welding or mechanical means to minimize the potential for loose parts in service. Threaded members used on STP have been used on many Westinghouse steam generators with no history of degradation in service.

b. From what materials are these features constructed?

Response

Both the main feedwater and the auxiliary feedwater spray nozzles are fabricated from thermally treated Alloy 690 material. The tube support plates and baffle plates are manufactured from 405 stainless steel to minimize corrosion potential.

c. What inspection plans, if any, are in place to verify the structural integrity of these new features during the extended SG ISI interval? If there are no plans to inspect these features, please discuss the technical basis for this position.

Response

A secondary-side foreign object search and retrieval operation was performed in all four SGs after RSG installation during 1RE09. Several loose parts that were left during the fabrication of the SGs were found and removed. No signs of damage during shipment were seen.

An extensive internal visual inspection was performed during 1RE10 in one SG. The objectives were to verify that the upper steam generator internal welds and parts were not cracked or eroded during the first cycle of operation and to obtain data on deposits. No problems were identified during this inspection. In addition, inspections were performed at the TTS prior to sludge lancing in all four SGs as described in the submittal.

V. C. Summer Delta 75 SGs have not experienced internal structural degradation nor have other Westinghouse replacement SGs.

A loose part was detected and removed in the April 2002 Arkansas-2 inspection. The piece was identified as a curled machining chip believed to have been introduced into the SG from the feedwater system.

Based upon this good industry experience with replacement SGs, there are no STP plans to perform secondary internal structural integrity inspections during the interval between tube inservice inspections.

8. ***In the submittal, the licensee describes a loose part indication below the sixth hot leg support plate that could not be visually investigated. To justify leaving this loose part in the SG, the licensee performed a bounding analysis to provide assurance that the loose part would not cause significant wear over the proposed operating period. In the analysis, the licensee assumes that the loose part is a metal gasket banding piece located at the "worst SG tube location." The staff requests the licensee to discuss where the "worst SG tube location" is and the basis for assuming that the loose part is a metal gasket banding piece.***

Response

The worst assumed location was at a tube that exhibits the limiting amplitudes of vibration and cross flow velocity. It was also assumed that the tube had an existing 20% throughwall degradation, which is a conservative limit of wear detection with bobbin exam. Additional conservative assumptions included that the object would remain in the same location (once tube wear begins) and that only the tube would experience wear.

The loose part was assumed to be a gasket banding piece because similar banding pieces were found on the TTS in SG "A" and it would be small enough to reach this location. The gap between the tubes is only 0.293 inch and a larger object could not have reached this area deep in

the tube bundle. The possible loose part signal was not detected on the six adjacent tubes that were examined, which provided further confirmation that the piece was very small.

References

1. Letter, T. J. Jordan to NRC Document Control Desk, "License Amendment Request - Revised Proposed Amendment to Technical Specification 4.4.5.3a," dated June 20, 2002 (NOC-AE-02001351)

2. "South Texas 1RE10 Degradation Assessment," dated September 20, 2001

3. Letter, T. J. Jordan to NRC Document Control Desk, "Special Report - 1RE10 Refueling Outage Inservice Inspection Results for Steam Generator Tubing," dated January 22, 2002 (NOC-AE-02001254) (Accession Number ML020390361)

4. Letter, G. J. Taylor to NRC Document Control Desk, "V. C. Summer Special Report (SPR 960004)," dated May 13, 1996 (PDR 9605170213 960513)

5. Letter, G. J. Taylor to NRC Document Control Desk, "V. C. Summer Special Report (SPR 970001)," dated November 3, 1997 (PDR 9711070068 971103)

6. Letter, G. J. Taylor to NRC Document Control Desk, "V. C. Summer Special Report (SPR 1999-003)," dated April 29, 1999 (PDR 9905050083 990429)

7. Letter, S. A. Byrne to NRC Document Control Desk, "V. C. Summer Special Report (SPR 2000-005)," dated November 8, 2000 (Accession Number ML003769321)

Table 1 – Results of V. C. Summer Tube Inspections Since 1994 Steam Generator Replacement

Outage	Flaw Type	Location	Sizing	# Plugged
April 1996 (Reference 4) ~ 22% of SG "A" ~ 16% of SG "B"	 Imperfection indication * Imperfection indication * * Present in baseline	 A-R113C72 @ TSP6+10.09 B-R52C11 @ TSP9+1.09	 Not quantifiable < 20% of NTW	 0 0
October 1997 (Reference 5) ~ 30% of SG "C"	 None			 0
April 1999 (Reference 6) ~ 40% of SG "A" ~ 40% of SG "B"	 None None			 0 0
October 2000 (Reference 7) 100% of SG "A" 100% of SG "B" 322 tubes TTS ~ 5% HL 100% of SG "C" 14 tubes Row 1 +P U-bend +P w/ HFC and MRC ~ 20% Row 1	 Small piece of wire removed Wear-like indication Wear-like indication Wear-like indication No tube expansion (NTE) NTE NTE NTE NTE * Found to be apparent in reviewing 1994 PSI	 A-R89C62/R88C63 @ TTS A-R19C140 @ AV7 * A-R26C139 @ AV2 * A-R26C139 @ AV7 * A-R25C26 A-R25C31 A-R94C51 C-R99C100 C-R57C96	 est. 9% of NTW est. 5% of NTW est. 9% of NTW	 1 1 1 1 1