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Ref. # 10CFR50.90

CPSES-200102509
Log # TXX-01086
File # 236

October 23, 2001

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

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES)
DOCKET NOS. 50-445 AND 50-446
LICENSE AMENDMENT REQUEST (LAR) 01-012
STEAM GENERATOR TUBE REPAIR USING LEAK TIGHT SLEEVES

Gentlemen:

Pursuant to 10CFR50.90, TXU Electric hereby requests an amendment to the CPSES Unit 1 Operating License (NPF-87) and CPSES Unit 2 Operating License (NPF-89) by incorporating the attached change into the CPSES Unit 1 and 2 Technical Specifications (TS).

The proposed change will revise TS 5.5.9, "Steam Generator Tube Surveillance Program" to permit tube sleeving repair techniques developed by Combustion Engineering Nuclear Power, LLC., (now owned by Westinghouse and will be referred to as Westinghouse hereafter) to be used at CPSES. The proposed change, when approved, will allow installation of a leak tight tube sleeve as described in Westinghouse report CEN-630-P as an alternative to plugging defective steam generator tubes.

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Attachment 1 is the required Affidavit. Attachment 2 provides a detailed description of the proposed changes, a safety analysis of the changes, and TXU Electric's determination that the proposed changes do not involve a significant hazard consideration.

Attachment 3 provides the affected Technical Specification pages, marked-up to reflect the proposed changes. Attachment 4 provides a retyped copy of affected Technical Specification pages with the proposed changes.

TXU Electric requests approval of the proposed License Amendment by July 30, 2002 to be implemented within 60 days of the issuance of the license amendment. This approval date supports the CPSES Unit 1 outage which is scheduled for the fall of 2002. The amendment is not required to complete the outage and restart the unit, but if the requested license amendment is not received, certain steam generator tubes may have to be plugged rather than sleeved. Please note that the subject tube repair methodology has been reviewed by the NRC staff and a generic safety evaluation has been issued, refer to NRC letter dated February 10, 2000 [ADAMS document number ML0036837650].

In accordance with 10CFR50.91(b), TXU Electric is providing the State of Texas with a copy of this proposed amendment.

This communication contains no new or revised commitments. Should you have any questions, please contact Obaid Bhatty at (254) 897-5839 or Robert Kidwell at (254) 897-5310.

Sincerely,

C. L. Terry

By: Roger D. Walker
Roger D. Walker
Regulatory Affairs Manager

OAB/ob

Attachments:
1. Affidavit
2. Description and Assessment
3. A markup of Technical Specifications Pages
4. Retyped Technical Specification Pages

cc: E. W. Merschoff, Region IV
C. E. Johnson, Region IV
D. H. Jaffe, NRR
Resident Inspectors, CPSES

Mr. Arthur C. Tate
Bureau of Radiation Control
Texas Department of Public Health
1100 West 49th Street
Austin, Texas 78704

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)		
	)		
TXU Electric	)	Docket Nos.	50-445
	)		50-446
(Comanche Peak Steam Electric Station,	)	License Nos.	NPF-87
Units 1 & 2)	)		NPF-89

AFFIDAVIT

Roger D. Walker being duly sworn, hereby deposes and says that he is Regulatory Affairs Manager of TXU Electric, the licensee herein; that he is duly authorized to sign and file with the Nuclear Regulatory Commission this License Amendment Request 01-012; that he is familiar with the content thereof; and that the matters set forth therein are true and correct to the best of his knowledge, information and belief.

Roger D. Walker

Roger D. Walker
Regulatory Affairs Manager

STATE OF TEXAS)
)
COUNTY OF)
Somervell

Subscribed and sworn to before me, on this 23rd day of October, 2001.

Gayle R. Peck Jespersen

Notary Public

Gayle R. Peck Jespersen
Notary Public, State of Texas
My Comm. Expires 03/16/02

ATTACHMENT 2 to TXX-01086

DESCRIPTION AND ASSESSMENT

DESCRIPTION AND ASSESSMENT

1.0 INTRODUCTION

Proposed change LAR-01-012 is a request to revise Technical Specifications (TS) 5.5.9, "Steam Generator Tube Surveillance Program."

The evaluations performed in support of this License Amendment Request do not result in changes to the FSAR per 10CFR50.71(e), the guidance provided by Regulatory Guide 1.181 "Content of the Updated Final Safety Analysis Report in Accordance with 10 CFR 50.71(e)," and NEI 98-03, "Guidelines for Updating Final Safety Analysis Reports."

2.0 DESCRIPTION OF PROPOSED AMENDMENT

The proposed change will revise TS 5.5.9, "Steam Generator Tube Surveillance Program" to permit tube sleeving repair techniques developed by Westinghouse to be used at Comanche Peak Steam Electric Station (CPSES). Sleeving is a steam generator tube repair method where a length of tubing (sleeve), having an outer diameter slightly smaller than the inside of the steam generator tube, is installed spanning the degraded region of the parent tube. Paragraph, "n) Tube Repair," invokes an additional applicable topical report for welded sleeves, other than the laser welded sleeving process previously approved by the NRC. The proposed change, when approved, will allow installation of a Westinghouse leak tight tube sleeve as an alternative to plugging or laser welded repaired sleeve for the defective steam generator tubes.

3.0 BACKGROUND

Pressurized water reactor (PWR) steam generators have experienced tube degradation related to corrosion phenomena, such as wastage, pitting, intergranular attack, stress corrosion cracking, and crevice corrosion, along with other phenomena such as denting and vibration wear. Tubes that experience excessive degradation reduce the integrity of the primary-to-secondary pressure boundary. Eddy current examination is used to measure the extent of tube degradation. When the reduction in tube wall thickness reaches a calculated value commonly known as the plugging criteria, the tube is considered defective and corrective action is taken.

Currently, the corrective action taken at many PWRs, including CPSES, is to remove the degraded tube (not addressed by a licensed Alternate Repair Criteria (ARC)) from service by installing plugs at both ends of the tube. The installation of steam generator tube plugs removes the heat transfer surface of the plugged tube from service and leads to a reduction in the primary coolant flow available for core cooling.

An alternative to plugging tubes is to repair defective steam generator tubes using Westinghouse Leak Tight Sleeves. Sleeving is a steam generator tube repair method where a length of tubing (sleeve) having an outer diameter slightly smaller than the inside of the steam generator tube is installed inside the parent tube spanning the degraded region. Installation of steam generator sleeves does not greatly affect the heat transfer capability or the primary coolant flow rate through the tube being sleeved; therefore, a large number of sleeves can be installed without significantly affecting the operation of the reactor coolant system. The sleeve spans the degraded section of the tube and maintains the structural integrity of the steam generator tube under normal and accident conditions and limits or prevents primary-to-secondary leakage through the sleeved section of the tube should the degradation progress through-wall. This repair method has been approved for use at several other U.S. Nuclear Power Plants, including Palo Verde and Prairie Island. (See Section 9.0 for additional examples.)

The proposed amendment would modify Technical Specification 5.5.9 "Steam Generator (SG) Tube Surveillance Program" to permit an additional method for installation of welded tube sleeves other than the current laser welded sleeving process.

This revision will allow installation of tube sleeves as an alternative to plugging to repair defective steam generator tubes. This revision will also allow removal of plugs that were previously installed as a corrective or preventive measure for the purpose of sleeving the tube. A tube inspection will be required prior to returning previously plugged tubes to service. 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. The proposed amendment will permit CPSES to use Leak Tight Sleeves developed by Westinghouse.

4.0 REGULATORY REQUIREMENTS AND GUIDANCE

10 CFR 50.55a requires components which are a part of the primary pressure boundary to be built to the requirements of Section III of the ASME Boiler and Pressure Vessel Code.

The associated materials and processes meet the rules of Section II of the ASME Code and Code Case N-20-3. The NRC has previously endorsed Code Case N-20 in Regulatory Guide 1.85. The design of the sleeve is predicated by the requirements of Section III, NB-3200, "Analysis" and NB-3300, "Wall Thickness". The ASME Boiler and Pressure Vessel Code provides criteria for evaluation of the stress levels in the tubes for design, normal operating, and postulated accident conditions. Any modification, repair or replacement of these components must also meet the requirements of the ASME Code to assure that the basis on which the unit was originally evaluated is unchanged. Essential welding variables, defined in Section IX of the ASME Code, Code Case N-395 which was endorsed by the NRC via Regulatory Guide 1.84, and Section XI, IWB-4300 were used to develop the weld process. The margin of safety is provided, in part, by the inherent safety factors in the criteria and requirements of the ASME Code.

Draft Regulatory Guide 1.121, issued for comment, entitled "Bases for Plugging Degraded PWR Steam Generator Tubes", addresses tubes with both part through-wall and through-wall cracking.

Regulatory Guide 1.83, Rev. 1, "Inservice Inspection of Pressurizer Water Reactor Steam Generator Tubes" (and the Comanche Peak Units 1 and 2 Technical Specifications) is used as the basis to determine the inservice inspection requirements for the sleeve. Additionally, CPSES FSAR section 5.4.2 describes the design and the Inservice Inspection of the Steam Generators.

5.0 TECHNICAL ANALYSIS

The CPSES Unit 1 steam generators are Westinghouse Model D4 steam generators. The Model D4 steam generators installed in CPSES Unit 1 have tubes that are either full depth hard rolled (~90% of the tubes) or full depth WEXTEX expanded (~10% of the tubes) in the tubesheet and are made of mill annealed Alloy 600 material. The tube support plates for the Unit 1 steam generators are of the drilled hole, carbon steel type.

There are two major types of Westinghouse Leak Tight Sleeves for steam generator tube repair. CEN-630-P, "Repair of ¾" O.D. Steam Generator Tubes Using Leak Tight Sleeves," Rev. 2, dated June 1997, describes in detail the design and testing of these sleeves applicable to CPSES. The analysis was performed for Westinghouse designed steam generators with ¾ inch outer diameter, 0.048-inch wall, Alloy 600 tubes. Westinghouse report CEN-630-P, Reference 10.1, provides a detailed description of the design, installation, and testing associated with the Westinghouse Leak Tight Sleeves. The sleeve material is thermally treated Alloy 690. The first type of sleeve spans the parent steam generator tube at the top of the tube sheet. This sleeve is welded to the tube near the upper end of the sleeve and is hard rolled into the tube within the steam generator tube sheet. A shorter sleeve of the same design is used to span defective areas of a steam generator tube, which exists just above the tube sheet. The second type of sleeve spans degraded areas of the steam generator tube at a tube support plate or in a free span section of tube. This leak tight sleeve is welded to the steam generator tube near each end of the sleeve. The steam generator tube with the installed welded and/or hard rolled sleeve meets the structural requirements of tubes that are not degraded.

The principal accident associated with this proposed change is the steam generator tube rupture (SGTR) accident. The consequences associated with the SGTR event are discussed in CPSES Final Safety Analysis Report Section 15.6.3, "Steam Generator Tube Failure." The SGTR event is a breach of the barrier between the reactor coolant system and the main steam system. The integrity of this barrier is significant from the standpoint of radiological safety in that a leaking steam generator tube allows the transfer of reactor coolant into the main steam system. In the event of a SGTR, radioactivity contained in the reactor coolant mixes with water in the shell side of the affected steam generator. This radioactivity is transported by steam to the turbine and then to the condenser, or directly to the condenser via the turbine bypass valves, or directly to the

atmosphere via the atmospheric dump valves, main steam safety valves, or the auxiliary feedwater pump turbine exhaust. Non-condensable radioactive gases in the condenser are removed by the condenser evacuation system and discharged to the plant vent. The use of Westinghouse Leak Tight Sleeves will allow the repair of degraded steam generator tubes such that the function and integrity of the tube is maintained; therefore, the SGTR accident is not affected.

The consequences of a hypothetical failure of a Westinghouse Leak Tight Sleeve and/or associated steam generator tube would be bounded by the current SGTR analysis described above. Due to the slight reduction in diameter caused by the sleeve wall thickness, primary coolant release rates would be slightly less than assumed for the SGTR analysis (depending on break location), and therefore, would result in lower total primary fluid mass release to the secondary system. A main steam line break (MSLB) or feed line break (FLB) will not cause a SGTR since the sleeves are analyzed for a maximum accident differential pressure greater than that predicted in the CPSES safety analysis. The impact of sleeving on steam generator performance, heat transfer, and flow restriction is minimal and insignificant compared to plugging.

The proposed technical specification change to allow the use of Westinghouse Leak Tight Sleeves does not adversely impact any other previously evaluated design basis accident. The structural analyses of the sleeves demonstrate that their design meets all applicable American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) criteria for the steam generator pressure, temperature, and flow design conditions, and establishes the minimum reactor coolant pressure boundary wall thickness requirements. As described in detail in Westinghouse report CEN-630-P, Reference 10.1, the results of the analyses and testing, as well as plant operating experience, demonstrate that CE Leak Tight Sleeves are an acceptable means of maintaining tube integrity. Sleeved tube plugging limit criteria are established using the guidance of Draft Regulatory Guide (RG) 1.121, "Basis for Plugging Degraded PWR Steam Generator Tubes." Furthermore, per Regulatory Guide 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes" recommendations, the sleeved tube can be monitored through periodic inspections with present eddy current techniques. These measures ensure that installation of sleeves spanning degraded areas of the tube will restore the tube to a condition consistent with its original design basis.

The material selected for both types of sleeves is thermally treated Alloy 690 due to its corrosion resistance properties. Historically, thermally treated Alloy 690 has been used successfully for steam generator tubes, tube plugs, and tube sleeves.

Westinghouse conducted a number of bench and autoclave tests to evaluate the corrosion resistance of the welded sleeve joint. Of particular interest is the effect of the mechanical expansion/weld residual stresses and the condition of the weld and weld heat affected zone. Tests have been performed on welded joints with and without a post-weld heat treatment. There was no detectable indication of sleeve or joint corrosion or aggravated tube corrosion. The specific

details of the corrosion performance of the thermally treated Alloy 690 material are contained in Section 6 of Westinghouse report CEN-630-P, Reference 10.1.

The sleeve dimensions, materials, and joints were designed to the applicable ASME Code. An extensive analysis and test program was undertaken to prove the adequacy of both the welded and welded-hard rolled sleeve. This program determined the effect of normal operating and postulated accident conditions on the sleeve-tube assembly, as well as the adequacy of the assembly to perform its intended function. The proposed sleeving provides for a substitution in kind for a portion of a steam generator tube.

Installation of Westinghouse Leak Tight Sleeves has no significant effect on the configuration of the plant and does not affect the way in which the plant is operated. Design criteria were established prior to performing the analysis and test program which, if met, would prove that both sleeve types are acceptable repair techniques. Based upon the results of the analytical and test programs described in Westinghouse report CEN-630-P, Reference 10.1, the two sleeve types fulfill their intended function as leak tight structural members and meet or exceed the established design criteria.

Evaluation of the sleeved tubes indicates no detrimental effects on the sleeve-tube assembly resulting from reactor coolant system flow, coolant chemistries, or thermal and pressure conditions. The sleeves are designed to be leak tight and therefore have no impact on steam generator leakage limits. Structural analyses of the sleeve-tube assembly, using the demonstrated margins of safety, have established its integrity under normal and accident conditions. The structural analyses performed are applicable to shorter sleeves installed at the top of the tubesheet and the tube support sleeves, which may be installed at CPSES. The detailed analyses for the different sleeve types and lengths are included in Section 8 of Reference 10.1.

Welding development has been performed on clean tubing, dirty tubing which has been taken from potboiler tests, and contaminated tubing taken from a steam generator. Westinghouse installed their first welded sleeves in a demonstration program at Ringhals Unit 2 in May 1984. Westinghouse's sleeving history is shown in Table 2-1 of Westinghouse report CEN-630-P, Reference 10.1. Since 1985, no sleeve that has been accepted based on nondestructive examination (NDE) has been removed from service due to degradation.

Mechanical tests using ASME Code stress allowables were performed on mockup steam generator tubes containing sleeves to provide qualified test data describing the basic properties of the completed assemblies. These tests determined axial load, collapse, burst, and thermal cycling capability. A minimum of three tests of each type was performed. The demonstrated load capacity of the assemblies provided an adequate safety factor for normal operating and postulated accident conditions. The load capability of the upper and lower sleeve joints is sufficient to withstand thermally induced stresses in the weld resulting from the temperature differential between the sleeve and the tube, and pressure-induced stresses resulting from normal operating

and postulated accident conditions. The burst and collapse pressures of the sleeve provide a large safety factor over limiting pressure differential. Mechanical testing revealed that the installed sleeve would withstand the cyclical loading resulting from power changes in the plant and other transients.

The effects of sleeve installation on steam generator heat removal capability and reactor coolant system flow rate are discussed in Section 10 of Westinghouse report CEN-630-P, Reference 10.1, which in summary states that the installation of the sleeves does not substantially affect the primary system flow rate or the heat transfer capability of the steam generators. Heat removal capability and reactor coolant system flow rate were considered for installation of one to three sleeves in a steam generator tube. These aforementioned information was used to provide hydraulic equivalency of plugs and installed sleeves, or the sleeve/plug ratio. Table 10-1 of Westinghouse report CEN-630-P, Reference 10.1, provides the sleeve/plug ratio. After sleeves are installed, an ultrasonic and eddy current examination is performed. The ultrasonic examination is used to confirm fusion of sleeve to the tube after welding. The eddy current examination serves as baseline to determine if there is sleeve degradation in later operating years. The steam generator tube will be plugged if the sleeve installation is not successful or if there is unacceptable degradation of a sleeve or sleeved steam generator tube. Standard steam generator tube plugs may be used to remove a sleeved tube from service.

Based on past usage and extensive testing, the Westinghouse Leak Tight Sleeves provide satisfactory repair of defective steam generator tubes. Design criteria were established based on the requirements of ASME Code and Draft Regulatory Guide 1.121. Qualified nondestructive examination will be used to perform necessary repair sleeve and tube inspections for defect detection and to verify proper installation of the repair sleeve. In-service inspection requirements for the sleeves are being added to the Technical Specifications consistent with current industry practices.

6.0 REGULATORY ANALYSIS

Total plant allowable primary to secondary leakage rates, derived from the requirements of 10 CFR 100, are determined on a plant specific basis. Offsite doses during either a main steam line break, or tube rupture event are not to exceed a small fraction of 10 CFR 100 limits. Since the free span welded joints form a hermetic seal between the sleeve and tube, and the tubesheet sleeve lower joints were shown to indicate leaktight performance during operating and faulted condition temperatures and pressure, the installation of welded sleeves will not contribute to offsite doses during either a postulated steam line break or any other faulted or upset condition.

The technical analysis performed by TXU Electric herein for the installation of the welded sleeves into the CPSES Unit 1 steam generators will provide a level of leak tightness and individual tube integrity equal to that of a non-degraded tube, and such will not adversely affect the safe

operation of the steam generators or the entire plant, and thus continues to be compliant with the regulatory requirements.

Conclusion

The requirements of the draft RG 1.121 are extended to the Westinghouse Leak Tight Sleeve in order to determine the level of degradation which will require removal of the sleeve from service by plugging. By utilizing the requirements for sleeve design according to the ASME Code and the draft Regulatory Guide 1.121 to define acceptance criteria, the design of the sleeve meets the requirements of General Design Criteria (GDC) 14, "Reactor Coolant Pressure Boundary", GDC 15, "Reactor Coolant System Design", and GDC 31, "Fracture Prevention of Reactor Coolant Pressure Boundary."

7.0 NO SIGNIFICANT HAZARDS DETERMINATION

TXU Electric has evaluated whether or not a significant hazards consideration is involved with the proposed changes by focusing on the three standards set forth in 10CFR50.92 as discussed below:

1. Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   The Westinghouse Leak Tight Sleeves are designed using the applicable American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code and, therefore, meet the design objectives of the original steam generator tubing. The applicable design criteria for the sleeves conforms to the stress limits and margins of safety of Section III of the ASME code. Mechanical testing has shown that the structural strength of repair sleeves under normal, upset, and faulted conditions provides margin to the acceptance limits. These acceptance limits bound the most limiting (three times normal operating pressure differential) burst margin recommended by Draft Regulatory Guide 1.121. Burst testing of sleeved tubes has demonstrated that no unacceptable levels of primary-to-secondary leakage are expected during any plant condition.

   Evaluation of the repaired steam generator tubes indicates no detrimental effects on the sleeve or sleeve-tube assembly from reactor coolant system flow, primary or secondary coolant chemistries, thermal conditions or transients, or pressure conditions as may be experienced at CPSES. Corrosion testing of sleeve-tube assemblies indicates no evidence of sleeve or tube corrosion considered detrimental under anticipated service conditions.

   The installation of the proposed sleeves is controlled via the sleeving vendor's proprietary processes and equipment. The Westinghouse process has been in use since 1984 and has

been implemented more than 24 times for the installation of over 4,200 sleeves. The CPSES steam generator design was reviewed and found to be compatible with the installation processes and equipment.

The implementation of the proposed amendment has no significant effect on either the configuration of the plant or the manner in which it is operated. The consequences of a hypothetical failure of the sleeved tube is bounded by the current steam generator tube rupture (SGTR) analysis described in the CPSES FSAR. Due to the slight reduction in diameter caused by the sleeve wall thickness, primary coolant release rates would be slightly less than assumed for the steam generator tube rupture analysis, depending on the break location, and therefore, would result in lower total primary fluid mass release to the secondary system. A main steam line break or feed line break will not cause a SGTR since the sleeves are analyzed for a maximum accident differential pressure greater than that predicted in the CPSES safety analysis. The proposed reduction of the steam generator primary to secondary operational leakage limit provides added assurance that leaking flaws will not propagate to burst prior to commencement of plant shutdown.

In conclusion, based on the discussion above, these changes will not significantly increase the probability or consequences of an accident previously evaluated.

2. Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No

The Westinghouse Leak Tight Sleeves are designed using the applicable ASME Code as guidance; therefore, they meet the objectives of the original steam generator tubing. As a result, the functions of the steam generators will not be significantly affected by the installation of the proposed sleeves. The proposed repair sleeves do not interact with any other plant systems. Any accident as a result of potential tube or sleeve degradation in the repaired portion of the tube is bounded by the existing tube rupture accident analysis. The continued integrity of the installed sleeve is periodically verified by the Technical Specification requirements.

The implementation of the proposed amendment has no significant effect on either the configuration of the plant or the manner in which it is operated. As discussed above, the reduced primary to secondary leakage limit is considered a conservative change in the plant limiting conditions for operation. Therefore, TXU Electric concludes that this proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Do the proposed changes involve a significant reduction in a margin of safety?

Response: No

The repair of degraded steam generator tubes with Westinghouse Leak Tight Sleeves restores the structural integrity of the degraded tube under normal operating and postulated accident conditions. The design safety factors utilized for the repair sleeves are consistent with the safety factors in the ASME Code used in the original steam generator design. The portions of the installed sleeve assembly that represents the reactor coolant pressure boundary can be monitored for the initiation and progression of sleeve/tube wall degradation. Use of the previously identified design criteria and design verification testing assures that the margin of safety is not significantly different from the original steam generator tubes. The proposed sleeve inspection requirements are more stringent than existing requirements for inspection of the steam generator tubes, and the reduction in the operational limit for primary to secondary leakage through the steam generator tubes is more conservative than current requirements. Therefore, TXU Electric concludes that the proposed change does not involve a significant reduction in a margin of safety.

EPRI qualified eddy current techniques will be used for the detection of tube degradation in 3/4 inch welded sleeved tubes. Alternate inspection techniques, may be used as they become available, as long as it can be demonstrated that the technique used provides the same degree or greater degree of inspection rigor.

The effect of sleeving on the design transients and accident analyses were reviewed and found to remain valid up to the level of steam generator tube plugging consistent with the minimum reactor flow rate as specified in Technical Specification 3.4.1. Continued compliance with the RCS flow limits of Technical Specification 3.4.1 is assured through precision flow measurements.

Because all relevant safety analyses were reviewed and found to remain valid, and because the appropriate design margins are maintained through compliance with the relevant ASME Code requirements, it is concluded that the proposed change does not involve a significant reduction in a margin of safety.

Based on the above evaluations, TXU Electric concludes that the activities associated with the above described changes present no significant hazards consideration under the standards set forth in 10CFR50.92 and accordingly, a finding by the NRC of no significant hazards consideration is justified.

8.0 ENVIRONMENTAL CONSIDERATION

TXU Electric has determined that the proposed amendment would change requirements with respect to the installation or use of a facility component located within the restricted area, as defined in 10CFR20, or would change an inspection or surveillance requirement. TXU Electric has evaluated the proposed changes and has determined that the changes do not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amount of effluent that may be released offsite, or (iii) a significant increase in the individual or cumulative occupational radiation exposure. Accordingly, the proposed change meets the eligibility criterion for categorical exclusion set forth in 10CFR51.22 (c)(9). Therefore, pursuant to 10CFR51.22 (b), an environmental assessment of the proposed change is not required.

9.0 PRECEDENTS

The NRC staff has previously reviewed identical and closely similar documents supporting requests for changes to the TS at other plants. The bulk of the technical and regulatory issues for the present request are identical to those reviewed in previous Safety Evaluations (SEs) concerning the use of Westinghouse welded sleeves. Details of prior staff evaluations of leak tight sleeves may be found in the staff safety evaluations for Waterford Steam Electric Station, Unit 3, docket number 50-382, dated December 14, 1995; Byron Nuclear Power Station, Units 1 and 2 and Braidwood Nuclear Power Station, Units 1 and 2, docket numbers 50-454, 50-455, 50-456 and 50-457, dated April 12, 1996; Kewaunee Nuclear Power Plant, docket No. 50-305, dated June 7, 1997; Prairie Island Units 1 and 2, docket numbers 50-282 and 50-306, dated November 4, 1997; Beaver Valley Unit 1, docket number 50-334, dated November 25, 1997; San Onofre Units 2 and 3, docket numbers 50-361 and 50-362, dated August 26, 1998 and Palo Verde Units 1, 2 and 3, docket numbers 50-528, 50-529, and 50-530, dated August 5, 1999.

10.0 REFERENCES

10.1 CEN- 630-P, Rev. 2, “Repair of 3/4" O.D. Steam Generator Tubes Using Leaktight Sleeves,” June 1997.

10.2. Westinghouse Letter Report No. CSE-01-023, “Effects of Comanche Peak 1 and 2 (D4 & D5) Steam Generator Parameters on the Generic TIG Welded Sleeves”, March 12, 2001.

ATTACHMENT 3 TO TXX-01086

MARKUP OF TECHNICAL SPECIFICATION PAGE

1 Page

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Tube Surveillance Program (continued)

4. Certain intersections as identified in WPT-15949 will be excluded from application of the voltage-based repair criteria as it is determined that these intersections may collapse or deform following a postulated LOCA + SSE event.

5. If an unscheduled mid-cycle inspection is performed, the following mid-cycle repair limits apply instead of the limits identified in 5.5.9e.1.m)1., 5.5.9e.1.m)2., and 5.5.9e.1.m)3. The midcycle repair limits are determined from the following equations:

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

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

where:

V_{URL}	=	upper voltage repair limit
V_{LRL}	=	lower voltage repair limit
V_{MURL}	=	mid-cycle upper voltage limit based on time into cycle
V_{MLRL}	=	mid-cycle lower voltage repair limit based on V_{MLRL} and time into cycle
Δt	=	length of time since last scheduled inspection during which V_{URL} and V_{LRL} were implemented
CL	=	cycle length (the time between two scheduled steam generator inspections)
V_{SL}	=	structural limit voltage
Gr	=	average growth per cycle
NDE	=	95-percent cumulative probability allowance for nondestructive examination uncertainty (i.e., a value of 20-percent has been approved by the NRC)

Implementation of these mid-cycle repair limits should follow the same approach as in TS 5.5.9e.1.m)1., 5.5.9e.1m)2., and 5.5.9e.1.m)3.

n. Tube Repair (for Unit 1 only) refers to a process that establishes tube serviceability. Acceptable tube repairs will be performed in accordance with the process described in Westinghouse WCAP-13698, Rev. 3,Westinghouse Letter WPT-16094 dated March 20, 2000 ~~and~~ , WCAP-15090, Rev. 1, *and CEN-630-P, Rev 2 dated June 1997.*

(continued)

ATTACHMENT 4 to TXX-01086

RETYPED TECHNICAL SPECIFICATION PAGE

1 Page

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Tube Surveillance Program (continued)

4. Certain intersections as identified in WPT-15949 will be excluded from application of the voltage-based repair criteria as it is determined that these intersections may collapse or deform following a postulated LOCA + SSE event.

5. If an unscheduled mid-cycle inspection is performed, the following mid-cycle repair limits apply instead of the limits identified in 5.5.9e.1.m)1., 5.5.9e.1.m)2., and 5.5.9e.1.m)3. The midcycle repair limits are determined from the following equations:

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

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

where:

V_{URL}	=	upper voltage repair limit
V_{LRL}	=	lower voltage repair limit
V_{MURL}	=	mid-cycle upper voltage limit based on time into cycle
V_{MLRL}	=	mid-cycle lower voltage repair limit based on V_{MLRL} and time into cycle
Δt	=	length of time since last scheduled inspection during which V_{URL} and V_{LRL} were implemented
CL	=	cycle length (the time between two scheduled steam generator inspections)
V_{SL}	=	structural limit voltage
Gr	=	average growth per cycle
NDE	=	95-percent cumulative probability allowance for nondestructive examination uncertainty (i.e., a value of 20-percent has been approved by the NRC)

Implementation of these mid-cycle repair limits should follow the same approach as in TS 5.5.9e.1.m)1., 5.5.9e.1m)2., and 5.5.9e.1.m)3.

n. Tube Repair (for Unit 1 only) refers to a process that establishes tube serviceability. Acceptable tube repairs will be performed in accordance with the process described in Westinghouse WCAP-13698, Rev. 3,Westinghouse Letter WPT-16094 dated March 20, 2000, WCAP-15090, Rev. 1, and CEN-630-P, Rev 2 dated June 1997.

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