Mr. C. Lance Terry
Senior Vice President
& Principal Nuclear Officer
TXU Electric
Attn: Regulatory Affairs Department
P. O. Box 1002
Glen Rose, TX 76043

February 20, 2001

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES), UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: INSTALLATION OF LASER WELDED SLEEVES AS AN ALTERNATIVE TO PLUGGING DEFECTIVE STEAM GENERATOR TUBES (TAC NOS. MA9950 AND MA9951)

Dear Mr. Terry:

The Commission has issued the enclosed Amendment No. 83 to Facility Operating License No. NPF-87 and Amendment No. 83 to Facility Operating License No. NPF-89 for CPSES, Units 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated September 6, 2000, as supplemented by letters dated December 14, 2000, and January 25, 2001.

The amendments change CPSES, Units 1 and 2, TS 5.5.9, "Steam Generator Tube Surveillance Program," to permit installation of laser welded tube sleeves in the CPSES Unit 1 steam generators as an alternative to plugging defective tubes, and TS 5.6.10, "Steam Generator Tube Inspection Report," is revised to address reporting requirements for repaired tubes. Also, an editorial correction is made to TS Table 5.5-2.

A copy of our related Safety Evaluation is enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely, /R/
David H. Jaffe, Senior Project Manager, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-445 and 50-446

Enclosures: 1. Amendment No. 83 to NPF-87
2. Amendment No. 83 to NPF-89
3. Safety Evaluation

cc w/encls: See next page

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NRR-058

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 20, 2001

Mr. C. Lance Terry
Senior Vice President
& Principal Nuclear Officer
TXU Electric
Attn: Regulatory Affairs Department
P. O. Box 1002
Glen Rose, TX 76043

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES), UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: INSTALLATION OF LASER WELDED SLEEVES AS AN ALTERNATIVE TO PLUGGING DEFECTIVE STEAM GENERATOR TUBES (TAC NOS. MA9950 AND MA9951)

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The amendments change CPSES, Units 1 and 2, TS 5.5.9, "Steam Generator Tube Surveillance Program," to permit installation of laser welded tube sleeves in the CPSES Unit 1 steam generators as an alternative to plugging defective tubes, and TS 5.6.10, "Steam Generator Tube Inspection Report," is revised to address reporting requirements for repaired tubes. Also, an editorial correction is made to TS Table 5.5-2.

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Sincerely,

David H. Jaffe, Senior Project Manager, Section 1
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Enclosures: 1. Amendment No. 83 to NPF-87
2. Amendment No. 83 to NPF-89
3. Safety Evaluation

cc w/encls: See next page

Comanche Peak Steam Electric Station

cc:

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

TXU ELECTRIC

COMANCHE PEAK STEAM ELECTRIC STATION, UNIT NO. 1

DOCKET NO. 50-445

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 83
License No. NPF-87

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by TXU Electric dated September 6, 2000, as supplemented by letters dated December 14, 2000, and January 25, 2001, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. NPF-87 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 83 , and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. TXU Electric shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Gramm

Robert A. Gramm, Chief, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: February 20, 2001

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

TXU ELECTRIC

COMANCHE PEAK STEAM ELECTRIC STATION, UNIT NO. 2

DOCKET NO. 50-446

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 83
License No. NPF-89

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by TXU Electric dated September 6, 2000, as supplemented by letters dated December 14, 2000, and January 25, 2001, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. NPF-89 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 83 , and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. TXU Electric shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A Gramm

Robert A. Gramm, Chief, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: February 20, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 83

TO FACILITY OPERATING LICENSE NO. NPF-87

AND AMENDMENT NO. 83

FACILITY OPERATING LICENSE NO. NPF-89

DOCKET NOS. 50-445 AND 50-446

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
5.0-13	5.0-13
5.0-15a	5.0-15a
5.0-16	5.0-16
5.0-17	5.0-17
-----	5.0-17a
5.0-19	5.0-19
-----	5.0-19a
5.0-36	5.0-36

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Tube Surveillance Program

Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program.

The provisions of SR 3.0.2 are applicable to the SG Surveillance Program test frequencies.

a. Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 5.5-1.

b. Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 5.5-2 or 5.5-3. Table 5.5-2 applies to all tubes except repaired tubes (Unit 1 only) which are covered by Table 5.5-3. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 5.5.9d., and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 5.5.9e. The tubes selected for each inservice inspection per Table 5.5-2 shall include at least 3% of all the expanded tubes and at least 3% of the remaining number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

   1. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas;

   2. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

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

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

(continued)

5.5 Programs and Manuals

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

3. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 5.5-2 during the shutdown subsequent to any of the following conditions:

   a) Primary-to secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.5.2, or

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

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

   d) A main steam line or feedwater line break.

e. Acceptance Criteria

1. As used in this specification:

   a) 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;

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

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

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

   e) Defect means an imperfection of such severity that it exceeds the plugging or (for Unit 1 only) repair limit. A tube containing a defect is defective;

(continued)

5.5 Programs and Manuals

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

f) Plugging or Repair Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging or (for Unit 1 only) repaired by sleeving and is equal to 40% of the wall thickness. The plugging limit for laser welded sleeves is equal to 43% of the nominal wall thickness. This definition does not apply to that portion of the Unit 1 tubing that meets the definition of an F* tube. This definition does not apply to tube support plate intersections for which the voltage-based plugging criteria are being applied. Refer to 5.5.9e.1m) for the repair limit applicable to these intersections;

g) 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 5.5.9d.3, above;

h) Tube Inspection means an inspection of the steam generator tube from the tube end (hot leg side) completely around the U-bend to the top support of the cold leg. For a tube repaired by sleeving (for Unit 1 only), the tube inspection shall include the sleeved portion of the tube;

i) 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;

j) F* Distance (Unit 1 only) is the distance of the hardroll expanded portion of a tube which provides a sufficient length of non-degraded tube expansion to resist pullout of the tube from the tubesheet. The F* distance is equal to 1.13 inches, plus an allowance for eddy current measurement uncertainty, and is measured down from the top of the tubesheet, or the bottom of the roll transition, whichever is lower in elevation;

k) F* Tube (Unit 1 only) is that portion of the tubing in the area of the tubesheet region below the F* distance with a) degradation below the F* distance equal to or greater than 40%, b) which has no indication of degradation within the F* distance, and c) that remains inservice;

(continued)

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:

$$V_{MURL} = \frac{V_{SL}}{1.0 + NDE + Gr\frac{[CL - \Delta t]}{CL}}$$

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

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 and Westinghouse letter WPT-16094 dated March 20, 2000 and WCAP-15090, Rev. 1.

(continued)

5.5 Programs and Manuals

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

2. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 5.5-2.

(continued)

5.5 Programs and Manuals

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

TABLE 5.5-2
STEAM GENERATOR TUBE INSPECTION

	1ST SAMPLE INSPECTION		2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.	C-1	None	N.A.	N.A.	N.A.	N.A.
	C-2	Plug or repair* defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N.A.	N.A.
			C-2	Plug or repair* defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug or repair* defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N.A.	N.A.
	C-3	Inspect all tubes in this S.G., plug or repair* defective tubes and inspect 2S tubes in each other S.G. Notification to NRC pursuant to 10CFR50.72(b)(2)	All other S.G.s are C-1	None	N.A.	N.A.
			Some S.G.s C-2 but no additional S.G. C-3	Perform action for C-2 result of second sample	N.A.	N.A.
			Additional S.G. is C-3	Inspect all tubes in each S.G. and plug or repair* defective tubes. Notification to NRC pursuant to 10CFR50.72(b)(2)	N.A.	N.A.

(continued)

S = 12/n% Where n is the number of steam generators inspected during an inspection
* for Unit 1 only

5.5 Programs and Manuals

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

TABLE 5.5-3
STEAM GENERATOR REPAIRED TUBE INSPECTION FOR UNIT 1 ONLY

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

(continued)

(1) Each repair method is considered a separate population for determination of initial inservice inspection and scope expansion.

5.6 Reporting Requirements (continued)

5.6.7 Not used

5.6.8 PAM Report

When a report is required by the required actions of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.9 Not used

5.6.10 Steam Generator Tube Inspection Report

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged, repaired or designated as an F* tube in each steam generator shall be reported to the Commission;

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

   1) Number and extent of tubes and (for Unit 1 only) sleeves 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 to the Commission pursuant to 10 CFR 50.72(b)(2) within four hours of initial discovery, and in a report 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.

(continued)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 83 TO

FACILITY OPERATING LICENSE NO. NPF-87

AND AMENDMENT NO. 83 TO

FACILITY OPERATING LICENSE NO. NPF-89

TXU ELECTRIC

COMANCHE PEAK STEAM ELECTRIC STATION, UNITS 1 AND 2

DOCKET NOS. 50-445 AND 50-446

1.0 INTRODUCTION

By application dated September 6, 2000, as supplemented by letters dated December 14, 2000, and January 25, 2001, TXU Electric (the licensee) requested changes to the Technical Specifications (TSs) for the Comanche Peak Steam Electric Station (CPSES), Units 1 and 2. The changes revise CPSES, Units 1 and 2, TS 5.5.9, "Steam Generator (SG) Tube Surveillance Program," to permit installation of laser welded tube sleeves in the CPSES Unit 1 SGs as an alternative to plugging defective SG tubes. Also, TS 5.6.10, "Steam Generator Tube Inspection Report," is revised to address reporting requirements for repaired tubes. Additionally, an editorial correction is made to TS Table 5.5-2. The supplemental letters dated December 14, 2000, and January 25, 2001, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on November 1, 2000 (65 FR 65350).

2.0 BACKGROUND

CPSES Unit 1 has four Westinghouse Model D4 SGs. The nominal outside diameter of the SG tubes is 3/4 inch. The tubes are fabricated with mill annealed Inconel Alloy 600. SG tubes form a part of the reactor coolant pressure boundary. After a period of use, the tubes may degrade due to a corrosive environment or loading conditions and, thus, require repair or removal from service by plugging. The proposed laser welded sleeve design is a method of restoring a defective SG tube to a condition consistent with the design requirements of the tube, which is part of the reactor coolant pressure boundary. As such, General Design Criterion (GDC) 14 of Appendix A to Title 10, Code of Federal Regulations (10CFR) Part 50 applies to the laser welded sleeve repair method. GDC 14 requires that the reactor coolant pressure boundary be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

To satisfy GDC 14 and 10 CFR 50.55a, the sleeve needs to satisfy several sections of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code). The sleeve should be qualified for service in accordance with IWA-4000 in Section XI of the ASME Code. In addition, Section XI references Section III of the ASME Code for component design requirements that govern the original SG tubes and are applicable to the sleeve design. The sleeve must be analyzed using appropriate Section III criteria of the ASME Code for structural integrity under design, operating, and accident loading conditions. The resulting stresses in the sleeve and sleeve wall thickness must satisfy corresponding Section III allowable stresses. The laser welding should satisfy the qualification standard for welding procedures, welders, and welding operators in Section IX of the ASME Code.

Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR [Pressurized Water Reactor] Steam Generator Tubes," August 1976, provides guidance for determining the structural integrity of defective tubes. The sleeve is a part of a repaired tube; therefore, RG 1.121 applies to the structural integrity of sleeves.

The Nuclear Regulatory Commission (NRC) staff has approved the use of Westinghouse laser welded sleeve repairs at several nuclear power plants similar to CPSES.

3.0 EVALUATION

The NRC staff has reviewed the following areas of the sleeve repair method: design, materials of construction, structural integrity, leakage integrity, corrosion testing, plugging limits, and sleeve inspections. In addition, the NRC staff has evaluated the effects of SG sleeving on normal reactor operation, transients, and accidents. These topics are discussed below.

3.1 Design

The design of the Westinghouse laser welded sleeve is based on the design rules of Section III, Subsection NB, of the ASME Code. The Westinghouse laser welded sleeves are of three basic designs: full length tubesheet sleeve (FLTS), elevated tubesheet sleeve (ETS) and tube support plate sleeve (TSS). The FLTS spans a tube from the primary surface of the tubesheet to a point above the secondary-side surface of the tubesheet. The ETS spans a tube from a location within the tubesheet, about 14 inches up from the primary surface of the tubesheet, to a point above the secondary-side surface of the tubesheet. The TSS spans a tube about centered on a tube support plate intersection.

The FLTS and ETS are secured in the parent tube by first hydraulically expanding the upper and lower portions of the sleeve. The hydraulic expansion brings the sleeve into contact with the parent tube. A continuous circumferential weld is made by a laser head in the area of the hydraulically expanded region of the upper joint which is stress relieved with post weld heat treatment. The weld structurally supports the sleeve and at the same time forms a seal. On the lower end of the FLTS and ETS, the sleeve is joined to the tube by hydraulic expansion and hard rolling to provide structural integrity under all plant conditions. The TSS has a hydraulic expansion region at each end of the sleeve within which the weld is placed. The weld configuration is the same as the weld in the FLTS and ETS.

3.2 Materials of Construction

The sleeve material is a nickel-iron-chromium alloy. The sleeves are fabricated from thermally treated Alloy 690, an ASME Code-approved material (ASME SB-163) covered by ASME Code Case N-20. The NRC staff endorsed the use of Code Case N-20, Revision 3, in RG 1.85, "Materials Code Case Acceptability, ASME Code Section III, Division I." The Alloy 690 thermally treated material exhibits resistance to stress corrosion cracking in the SG environment and has been a preferred alloy for tubing in new and replacement SGs since the late 1980's.

3.3 Structural Integrity

To qualify the laser welded sleeves, Westinghouse performed the following structural evaluations: finite element model development, a heat transfer and thermal stress evaluation, a primary stress intensity evaluation, a primary plus secondary stress range evaluation, and a fatigue evaluation for mechanical and thermal conditions.

Westinghouse also analyzed the impact of a tubesheet bow on radial contact pressure for ETSs. The ETSs are installed in the upper half of the tubesheet, where the tubesheet bow, due to primary-to-secondary pressure differential during normal operation, tends to increase the diameter of the holes drilled in the tubesheet. This diameter increase due to tubesheet bowing tends to decrease the initial contact pressure between the sleeve/tube and tube/tubesheet. However, these contact pressures are also influenced by system pressures and differential thermal expansions among the sleeve, tube, and tubesheet. Westinghouse performed a finite element analysis of the tubesheet, channel head, and lower shell to determine the impact of the tubesheet bow on radial contact pressure. An analysis has been completed for each tube size for a given set of primary-to-secondary side pressures. The analysis showed that tubesheet bowing has no adverse effect on radial contact pressure.

Westinghouse performed the following tests on the lower joint for FLTS and ETS: the primary-to-secondary leak resistance test, the secondary-to-primary "onset of significant leakage" test, and the sleeve pullout test. The purpose of the primary-to-secondary leak resistance testing is to determine, for a potential perforation in a section of a tube spanned by the sleeve, the leakage for normal operation, for feedline break/steamline break, and for pressures bounding the SG initial primary side hydrostatic pressure test. The secondary-to-primary leak test is performed to determine the sleeve-to-tube interference fit radial contact pressure. Sleeve pullout testing is a direct determination of the resistance to a pullout of the sleeve joint. The pullout tests showed that the lower joint for FLTS and ETS will maintain structural integrity under normal and accident conditions. The leak test results are discussed in the Leakage Integrity section herein.

Another structural integrity issue related to the laser weld sleeves is the weld width. In a letter to Wisconsin Public Service Corporation dated June 27, 2000, the NRC staff approved an amendment regarding Westinghouse laser welded sleeves and laser weld repair at the Kewaunee Nuclear Power Plant (Kewaunee). One of the issues in the Kewaunee amendment is related to the laser weld width. The laser weld is autogenous, that is, no filler material is used in the weld. The weld joint is made by applying a power source to a laser weld head, thereby liquidizing sleeve and tube material. The joint is formed by fusion of the molten sleeve metal with molten tube metal. Chemical analysis of the solidified weld metal has confirmed that it conforms to the requirements for nickel-chromium-iron Alloy 690. Westinghouse has shown that the laser welding is qualified to the requirements of Section IX of the ASME Code.

In 1998, Westinghouse determined that the original finite element analysis of the weld under-predicted the shear stress in the laser weld, which was qualified with a minimum weld width of 0.015 inch, such that the calculated stresses exceeded the ASME Code design allowable stresses. Westinghouse initiated a "design-by-test" verification program to demonstrate the acceptability of the laser weld having a minimum specified weld width of 0.015 inch. The NRC staff raised a concern regarding the approach being used.

Subsequently, Westinghouse performed structural analysis to characterize the average weld width that would be necessary to show compliance with the ASME Code design-by-analysis requirements. The structural analysis showed that an average weld width of 0.021 inches meets all of the design-by-analysis requirements of the ASME Code (no required structural tests). Furthermore, based on the results of the burst testing program, Westinghouse demonstrated that a weld with an average width of 0.019 inches meets the ASME Code design-by-test requirements. Based on confirmatory calculation, the NRC staff agrees with the above weld widths.

In a letter to the NRC staff dated February 28, 2000, and in a letter to the licensee dated March 20, 2000, Westinghouse stated that it is committed to revise field inspection procedure to include a criterion for an average width of each weld in order to meet the requirements of Section III of the ASME Code for design-by-analysis. Any welds determined to have an average weld width of less than 0.021 inches will be subject to an engineering disposition process. Special considerations may be made that result in infrequently accepting welds with average widths as small as, but no less than 0.019 inches.

The Westinghouse analyses and testing show that the primary stress intensities for the laser welded tube assembly satisfy all of the allowable ASME Code primary stress limits. The analyses also show that ASME Code, Section III, limits for the maximum primary to secondary stress intensity range and cumulative fatigue are satisfied during all plant conditions.

As discussed in WCAP-15090, Revision 1, the pressure and temperature loading conditions for CPSES Unit 1 SG tubes are bounded by the loading conditions in the generic structural analyses as shown in WCAP-13698, Revision 3. Therefore, the structural integrity of the laser weld sleeves for CPSES Unit 1 will be maintained under various plant conditions.

3.4 Leakage Integrity

The upper joint of the FLTS and ETS is inherently leak tight because of the weld. The TSS has a weld in the upper and lower joint and is considered leak tight. The lower joint of the FLTS and ETS is a mechanical seal and is not considered leak tight, but leak limiting. The requirements in 10 CFR Part 100 specify certain primary-to-secondary leakage limits under accident conditions. For CPSES Unit 1, the 10 CFR Part 100 leak limit is 27.79 gpm. The plant TS also specifies certain primary-to-secondary leakage limits under normal operation conditions. The primary-to-secondary leakage limit in the CPSES TS is 150 gallons per day per SG.

The Westinghouse leak rate tests, as discussed above, have consistently demonstrated that any rolled joint leakage, should it occur, is small comparing to 10 CFR Part 100 limits and TS leakage limits. Additionally, operating experience with sleeves has demonstrated actual performance to be essentially leak tight during all plant conditions.

3.5 Corrosion Testing

Westinghouse has performed a number of bench and autoclave tests to evaluate the corrosion resistance of the welded sleeve-tube joint. These tests included accelerated corrosion tests on actual sleeve samples whose inside surface and outside surface were subjected to corrosive solutions. The corrosion tests were conducted to determine the effect of the mechanical expansion and weld residual stresses and the condition of the weld and weld heat affected zone because a concern of the laser welded sleeve is the magnitude of the residual stresses from the sleeve installation process. The corrosion tests showed that Alloy 690 material performs better than Alloy 600 material in terms of corrosion resistance.

3.6 Plugging Limits

The sleeve plugging limit is defined as the imperfection depth in the sleeve at or beyond which the sleeved tube shall be removed from service. Westinghouse calculated the sleeve plugging limit in accordance with RG 1.121 and ASME Code Section III. On the basis of current operating conditions, Westinghouse calculated a sleeve plugging limit of 44 percent of the sleeve wall thickness for CPSES Unit 1. The licensee proposed to implement a plugging limit of 43 percent sleeve wall thickness in the TS. The NRC staff finds the proposed TS limit acceptable because it is more conservative than the Westinghouse calculated value.

3.7 Sleeve Inspections

In WCAP-13698, Revision 3, Westinghouse specified a pre-service sample inspection for welds using the ultrasonic technique (UT) as a part of the sleeve installation procedure. Westinghouse also specified that a Cecco-5/bobbin probe be used in the baseline sleeve inspection. The NRC staff raised a concern about the adequacy of a sample UT inspection to assure that all welds will meet the acceptance criteria. The NRC staff also raised a question about the effectiveness of the Cecco-5 probe in the sleeve inspection. In the December 14, 2000, supplemental letter, the licensee stated that it will not implement the UT sample inspection. Instead, it will ultrasonically inspect all welds after sleeve installation. Further, the licensee stated that the plus point probe will be used in the baseline and inservice inspection and that the Cecco-5 probe will not be used. The licensee stated in the supplement to the applicaton dated December 14, 2000, that it will control this commitment in its commitment management program. The NRC staff finds the licensee's responses to these two inspection issues acceptable.

For the inservice inspection, the licensee proposes to implement, in Table 5.5-3, "Steam Generator Repaired Tube Inspection For Unit 1 Only," a sleeve sample inspection which is based on Electric Power Research Institute SG guidelines. The EPRI guidelines provide for a conservative sample inspecton of SG tube sleeves. Therefore, the NRC staff finds the sleeve sample inspection acceptable.

3.8 Effects of SG Sleeving on Normal Reactor Operation, Transients, and Accidents

Without provisions for tube repair by sleeving, SG tubes with indications of degradation in excess of the plugging limit defined in TS 5.5.9e.1.f) would have to be removed completely from service by using a SG tube plug. Removal of the tube from service results in a reduction of reactor coolant flow through the SG, which affects the heat transfer efficiency of the SG, and a corresponding decrease in flow through the Reactor Coolant System (RCS). Repair of a tube by sleeving maintains the tube in service and results in less impact on (1) reactor coolant flow rate, and (2) heat transfer between the RCS and the SG; therefore, the use of sleeving in lieu of plugging allows more tubes to be repaired before operational limits are reached. As stated in the TS Bases 3.4.1, the limits placed on RCS pressure, temperature, and flow rate,[1] will ensure that the minimum departure from nucleate boiling ratio will be met for analyzed transients such as loss of coolant flow (complete and partial loss of forced reactor coolant flow, and reactor coolant pump shaft seizure and break) and dropped rod events. The minimum RCS total flow rate limits correspond to that assumed for departure from nucleate boiling analyses.

For CPSES Unit 1, TS 3.4.13 limits primary to secondary leakage through one SG to 150 gallons per day. This limit minimizes the potential for large primary-to-secondary leakage during a main steam line break, when the differential pressure across the SG is the greatest. Also, TS Bases B.3.4.13e states that, based upon non-destructive testing examination uncertainties, bobbin coil voltage distribution, and crack growth rate from the previous inspection of non-sleeved SG tubes, the expected leak rate following a steam line break is limited to below 27.79 gallons per minute (gpm) (calculated at room temperature conditions). Maintaining leakage within the 27.79 gpm limit will ensure that offsite doses will remain within 10 CFR Part 100 guidelines and within control room dose (GDC 19) guidelines. Leak testing was not performed on laser welded joints, since as a welded structure, the joint is considered leak tight. Also, confirmatory mechanical and leak testing will be conducted supporting the installation of elevated tubesheet sleeves at CPSES Unit 1.

The licensee's September 6, 2000, application states that the hypothetical consequences of failure of a sleeve would be bounded by the current SG tube rupture (SGTR) analysis included

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[1] TS 3.4.1 requires that the RCS total flow rate be greater than or equal to 389,700 gallons per minute (gpm) and equal to or greater than the limit specified in the Core Operating Limits Report (COLR). The COLR is prepared by the licensee as required by TS 5.6.5 for each operating cycle. On October 12, 1999, CPSES submitted the COLR for Unit 1, Cycle 8. The COLR specifies that the limits for RCS total flow be equal to or greater than 397,200 gpm based on precision heat balance and greater than 317,000 gpm based on an elbow tap differential pressure measurement prior to MODE 1 after the refueling outage. The limits were developed using NRC-approved methodologies specified in TS 5.6.5b, Items 5 and 9 through 19. These limits were determined by the licensee such that all applicable limits of the safety analysis are met.

in the CPSES Final Safety Analysis Report (FSAR). Due to the slight reduction in diameter caused by the sleeve wall thickness, it is expected that primary coolant release rates would be slightly less than assumed for the SGTR analysis (depending on the break location), and therefore, would result in lower total primary fluid mass release to the secondary system. Also, tubesheet sleeves would reduce the primary fluid flow through the sleeved tube assembly due to the diameter reduction the fluid would have to pass on its way to the break area. The overall effect would be reduced SGTR release rates.

Reports WCAP-13698, Revision 3, and WCAP-15090, Revision 1, address the hydraulic equivalency of SG sleeves relative to SG plugs. Since the NRC staff has not had sufficient time to review and approve the two reports, with regard to the hydraulic equivalency issue, the licensee provided additional conservatism in their supplemental letter dated January 25, 2001, with regard to the number of sleeves that may be installed. In this letter the licensee committed to using a hydraulic equivalency ratio of four (4) sleeved tubes to one (1) plugged tube when calculating the hydraulic equivalency of SG sleeves relative to SG plugs. In addition, the number of sleeved tubes in any one SG will be limited to 50 percent of the allowed SG tube plugging allowance.[2] This commitment is more conservative than the hydraulic equivalency presented by the licensee in the two referenced technical reports.

The NRC staff finds this commitment acceptable since it provides a high degree of confidence that the contribution of sleeving to the 10 percent SG plugging allowance will not be exceeded.

3.9 Review for Risk Implications

Although this license amendment request was not risk-informed, it was qualitatively screened by the NRC staff for potential risk implications in accordance with the guidance in Appendix D to Standard Review Plan Chapter 19. The staff review did not identify any:

- significant changes to allowed outage times (e.g., outside the range previously approved at similar plants), the probability of any initiating event, the probability of success for any mitigative action, change to any functional recovery time, or any operator action requirement;

- significant change to functional requirements or redundancy;

- significant change to operations that affect the likelihood of undiscovered failures;

- significant effects on a basis for successful safety function; or

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[2] By letter dated January 8, 2001, the licensee provided the Annual Report of Changes in Peak Cladding Temperature. The analyses for Unit 1 were performed with a SG tube plugging allowance of 10 percent of the total number of SG tubes. As set forth in its analysis, the licensee determined that the peak clad temperatures for large and small break LOCAs were below the 10 CFR 50.46 Acceptance Criteria of 2200 degrees Fahrenheit.

- "special circumstances" under which compliance with existing regulations may not produce the intended or expected level of safety and plant operation may pose an undue risk to public health and safety.

Therefore, this amendment was not referred for a more detailed risk evaluation as part of the license amendment review.

3.10 Commitments

In reviewing the application dated September 6, 2000, as supplemented, the NRC staff noted that the licensee made commitments regarding activities associated with the proposed use of laser welded sleeves. The commitments that the NRC staff considers to be safety significant are as follows:

1. TXU Electric will use a hydraulic equivalency ratio of four (4) sleeved tubes to one (1) plugged tube when calculating Steam Generator Tube Plugging and the number of sleeved tubes in any one steam generator will be limited to 50% of the allowed Steam Generator Tube Plugging.

2. The plus point probe will be used for UT and the cecco-5 probe will not be used.

3. The weld width limit of 0.021 inches will be implemented at CPSES Unit 1via the site-specific procedures. TXU Electric will perform a 100% pre-service inspection of the sleeves. The inspection of the laser weld will be done ultrasonically.

The NRC staff finds that reasonable controls for the implementation and for subsequent evaluation of proposed changes pertaining to the above regulatory commitments are best provided by the licensee's administrative processes, including its commitment management program. The above regulatory commitments do not warrant the creation of regulatory requirements. The staff notes that pending industry and regulatory guidance pertaining to 10 CFR 50.71(e) may call for some information relative to the above commitments to be included in a future update of the CPSES Updated FSAR.

4.0 PROPOSED TECHNICAL SPECIFICATION CHANGES

Changes to TS pages are discussed below:

TS 5.5.9b—This section is revised by adding a reference to new Table 5.5-3, "Steam Generator Repaired Tube Inspection For Unit 1 Only." This table provides inservice inspection scope for sleeved tubes. The licensee proposed to delete a requirement to reinspect previous defects or imperfections in the tube area that are repaired by sleeving. This deletion is acceptable because the sleeve becomes the new pressure boundary for the previous defects. The licensee also proposed to delete a requirement that specifies that when referring to a SG tube the sleeve shall be considered as part of the tube if the tube has been repaired per TS 5.5.9e.1.n). This requirement is redundant considering the tube repair is defined in various TS sections; therefore, its deletion is acceptable.

TS 5.5.9e.1.e)—This section is revised by adding "...or (for Unit 1 only) repair..." This change indicates that the proposed sleeve repair is applicable to Unit 1 SGs only.

TS 5.5.9e.1.f)—This section is revised by adding a requirement that the plugging limit for laser welded sleeves is equal to 43 percent of the nominal sleeve wall thickness.

TS 5.5.9e.1.h)—This section is revised by adding a requirement that for a tube repaired by sleeving (for Unit 1 only), the tube inspection shall include the sleeved portion of the tube.

TS 5.5.9e.1.n)—This section is added to the TS to define the tube repair. Additionally, it is required that acceptable tube repairs will be performed in accordance with the process described in Westinghouse WCAP-13698, Revision 3; WCAP-15090, Revision 1; and Westinghouse letter WPT-16094, dated March 20, 2000.
Table 5.5-2—This table is revised by adding "...or repair..." to various action statements. In addition, the table is revised by changing "C-3" to "C-2" under the second sample inspection category, which is an editorial correction.

TS 5.6.10—This section is revised by adding a requirement that the results of sleeve inspection, which include number of sleeves inspected and identification of sleeved tubes, shall be submitted to the NRC in a report within 12 months following the completion of the inspection.

The NRC staff finds that the proposed changes to the TSs are consistent with the submitted analyses, and are therefore acceptable. The various editorial changes to the TS also are acceptable.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Texas State official was notified of the proposed issuance of the amendments. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

Except for the change to TS 5.6.10 the amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding. (65 Fed. Reg. 65,350 (Nov. 1, 2000). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). With respect to the change to TS 5.6.10, the amendments relate to changes in recordkeeping, reporting, or administrative procedures or requirements. Accordingly, except for the change to TS 5.6.10, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(10). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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Date: February 20, 2001