Docket File

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

September 22, 1999

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), UNIT 1 - ISSUANCE OF AMENDMENTS RE: IMPLEMENTATION OF THE 1.0 VOLT STEAM GENERATOR TUBE REPAIR CRITERIA (TAC NOS. MA4843 AND MA4844)

Dear Mr. Terry:

The Commission has issued the enclosed Amendment No. 70 to Facility Operating License No. NPF-87 and Amendment No. 70 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 February 12, 1999, as supplemented by letter dated June 14, 1999.

The amendments change TS 3.4.13, "RCS Operational Leakage," TS 5.5.9, "Steam Generator (SG) Tube Surveillance Program," and TS 5.6.10, "Steam Generator Tube Inspection Report," to implement the 1.0 Volt Steam Generator Tube Repair Criteria for CPSES, Unit 1.

As a part of the implementation of the 1.0 Volt Steam Generator Tube Repair Criteria for CPSES, Unit 1, TXU Electric (TXU) requested NRC staff approval to use a voltage-dependent probability of detection instead of the constant probability of detection of 0.6 specified in Generic Letter 95-05. The voltage-dependent probability of detection approach calculates the distribution of bobbin indications as a function of voltage at the beginning of cycle. The NRC staff is currently addressing such an approach generically through the Nuclear Energy Institute.

DF01

CP1

Until the NRC staff makes a generic determination regarding probability of detection, the licensee should use a constant value of 0.6. The NRC staff and the licensee have discussed this matter and have agreed that the licensee will use a constant value of .6 until this issue is resolved.

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,

/s/

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. 70 to NPF-87
2. Amendment No. 70 to NPF-89
3. Safety Evaluation

cc w/encls: See next page

DISTRIBUTION
Docket File
PUBLIC
PDIV-1 Reading
W.Beckner,TSB
L.Hurley,RIV
R.Scholl (e-mail SE only)
SRichards (cover ltr only)
OGC
ACRS
G.Hill (4)
K.Brockman,RIV
J.Kilcrease,RIV

*See previous concurrence
***No major changes to SE

To receive a copy of this document, indicate "C" in the box								
OFFICE	PDIV-1/PM		PDIV-1/LA	C	EMCB/SPSB	OGC*		PDIV-1/SC	C
NAME	D.Jaffe:db		L.Berry		***	RWeisman		R.Gramm RG	
DATE	9/22/99		9/22/99		8/14, 8/4/99	9/16/99		9/22/99	

DOCUMENT NAME: G:\PDIV-1\ComanchePeak\amdma4843.wpd
OFFICIAL RECORD COPY

cc:
Senior Resident Inspector
U.S. Nuclear Regulatory Commission
P. O. Box 2159
Glen Rose, TX 76403-2159

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011

Mrs. Juanita Ellis, President
Citizens Association for Sound Energy
1426 South Polk
Dallas, TX 75224

Mr. Roger D. Walker
Regulatory Affairs Manager
TXU Electric
P. O. Box 1002
Glen Rose, TX 76043

George L. Edgar, Esq.
Morgan, Lewis & Bockius
1800 M Street, N.W.
Washington, DC 20036-5869

Honorable Dale McPherson
County Judge
P. O. Box 851
Glen Rose, TX 76043

Office of the Governor
ATTN: John Howard, Director
Environmental and Natural
Resources Policy
P. O. Box 12428
Austin, TX 78711

Arthur C. Tate, Director
Division of Compliance & Inspection
Bureau of Radiation Control
Texas Department of Health
1100 West 49th Street
Austin, TX 78756-3189

Jim Calloway
Public Utility Commission of Texas
Electric Industry Analysis
P. O. Box 13326
Austin, TX 78711-3326

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

TXU ELECTRIC COMPANY

COMANCHE PEAK STEAM ELECTRIC STATION, UNIT 1

DOCKET NO. 50-445

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 70
License No. NPF-87

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

   A. The application for amendment by TXU Electric Company (TXU Electric) dated February 12, 1999, as supplemented by letter dated June 14, 1999, 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. 70, 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: September 22, 1999

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

TXU ELECTRIC COMPANY

COMANCHE PEAK STEAM ELECTRIC STATION, UNIT 2

DOCKET NO. 50-446

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 70
License No. NPF-89

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

   A. The application for amendment by TXU Electric Company (TXU Electric) dated February 12, 1999, as supplemented by letter dated June 14, 1999, 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. 70 , 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: September 22, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 70

TO FACILITY OPERATING LICENSE NO. NPF-87

AND AMENDMENT NO. 70

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
3.4-33	3.4-33
5.0-14	5.0-14
5.0-15	5.0-15*
---	5.0-15a*
5.0-16	5.0-16
---	5.0-16a
5.0-17	5.0-17
5.0-36	5.0-36
---	5.0-36a

* no change - overflow page

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. 1 gpm unidentified LEAKAGE;

c. 10 gpm identified LEAKAGE;

d. 1 gpm total primary to secondary LEAKAGE through all steam generators for Unit 2 (SGs); and

e. 150 gallons per day for Unit 1 and 500 gallons per day for Unit 2 primary to secondary LEAKAGE through any one SG.

APPLICABILITY: MODES 1, 2, 3, and 4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Required Action and associated Completion Time of Condition A not met. OR Pressure boundary LEAKAGE exists.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

5.5 Programs and Manuals

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

c) A tube inspection (pursuant to Specification 5.5.9.4a.8 shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

d) Indications left in service as a result of the application of the tube support plate voltage repair criteria shall be inspected by bobbin probe during all future refueling outages.

3. The tubes selected as the second and third samples (if required by Table 5.5.9-2 during each inservice inspection may be subjected to a partial tube inspection provided:

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

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

4. Implementation of the steam generator tube/tube support plate repair criteria requires a 100% bobbin coil inspection for hot-leg and cold-leg tube support plate intersections down to the lowest cold-leg support with known outside diameter stress corrosion cracking (ODSCC) indications. The Determination of the lowest cold leg tube support plate intersections having ODSCC indications shall be based on the performance of at least a 20% random sampling of the tubes inspected over their full length.

(continued)

5.5 Programs and Manuals

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

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

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

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

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

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

2. If the results of the inservice inspection of a steam generator conducted in accordance with Table 5.5-2 at 40-month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 5.5.9.3a.; the interval may then be extended to a maximum of once per 40 months; 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.

d. 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 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 Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to 40% of the nominal tube wall thickness. This definition does not apply to tube support plate intersections for which the voltage-based plugging criteria are being applied. Refer to 5.5.9.d.1.j 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 5.5.9.3c, 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;

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; and

j) For Unit 1 only, the Tube Support Plate Plugging Limit is used for the disposition of alloy 600 steam generator tubes for continued service that are experiencing predominantly axially oriented outside diameter stress corrosion cracking confined within the thickness of the tube support plates and flow distribution baffle (FDB). At tube support plate intersections (and FDB), the plugging limit is based on maintaining steam generator tube serviceability as described below:

1. Steam generator tubes, whose degradation is attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with bobbin voltages less than or equal to the lower voltage repair limit [1.0 volt], will be allowed to remain in service.

(continued)

5.5 Programs and Manuals

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

2. Steam generator tubes, whose degradation is attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with the bobbin voltage greater than the lower voltage repair limit [1.0 volt], will be repaired, except as noted in 5.5.9.d.1.j.3 below.

3. Steam generator tubes with indications of potential degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than the lower voltage repair limit [1.0 volt] but less than or equal to the upper voltage repair limit*, may remain inservice if a rotating pancake coil inspection does not detect degradation. Steam generator tubes, with indications of outside diameter stress corrosion cracking degradation with a bobbin voltage greater than the upper repair limit** will be plugged or repaired.

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.

(continued)

* The upper voltage repair limit is calculated according to the methodology in GL 95-05 as supplemented.

** V_{URL} will differ at the TSPs and flow distribution baffle.

5.5 Programs and Manuals

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

5. If an unscheduled mid-cycle inspection is performed, the following mid-cycle repair limits apply instead of the limits identified in 5.5.9.d.1.j.1, 5.5.9.d.1.j.2, and 5.5.9.d.1.j.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 5.5.9.d.1.j.1, 5.5.9.d.1.j.2, and 5.5.9.d.1.j.3.

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)

5.6 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 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 inspected,

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

   3) Identification of tubes plugged.

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)

5.6 Reporting Requirements (continued)

5.6.10 Steam Generator Tube Inspection Report (continued)

d. For implementation of the voltage based repair criteria to tube support plate intersections, notify the staff prior to returning the steam generators to service should any of the following conditions arise:

1. If estimated leakage based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds the leakage limit (determined from the licensing basis dose calculation for the postulated main steam line break) for the next operating cycle.

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

3. If indications are identified that extend beyond the confines of the tube support plate.

4. If indications are identified at the tube support plate elevations that are attributable to primary water stress corrosion cracking.

5. If the calculated conditional burst probability based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds 1×10^{-2}, notify the NRC and provide an assessment of the safety significance of the occurrence.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 70 TO

FACILITY OPERATING LICENSE NO. NPF-87

AND AMENDMENT NO. 70 TO

FACILITY OPERATING LICENSE NO. NPF-89

TXU ELECTRIC COMPANY

COMANCHE PEAK STEAM ELECTRIC STATION, UNITS 1 AND 2

DOCKET NOS. 50-445 AND 50-446

1.0 INTRODUCTION

By application dated February 12, 1999, as supplemented by letter dated June 14, 1999, TXU Electric Company (TXU Electric, the licensee) requested changes to the Technical Specifications (TSs) for the Comanche Peak Steam Electric Station (CPSES), Units 1 and 2. The proposed changes would change TS 3.4.13, "RCS [Reactor Coolant System] Operational Leakage," TS 5.5.9, "Steam Generator (SG) Tube Surveillance Program," and TS 5.6.10, "Steam Generator Tube Inspection Report," to implement a 1.0 Volt Steam Generator Tube Repair Criteria for CPSES, Unit 1. The requested amendment would permit the use of voltage-based alternate repair criteria for defective steam generator tubes at CPSES, Unit 1. The voltage-based alternate repair criteria allows steam generator tubes with axially oriented outside diameter stress corrosion cracking (ODSCC) confined within the thickness of the tube support plates to remain in service based on the magnitude of the bobbin coil voltage response. These changes apply to the CPSES, Unit 1, TSs and administratively to Unit 2, since CPSES, Units 1 and 2, share the same TSs.

The June 14, 1999, supplement provided clarifying information that did not change the scope of the February 12, 1999, application and the initial proposed no significant hazards consideration determination.

2.0 BACKGROUND

The traditional strategy for ensuring adequate structural and leakage integrity of the steam generator tubes has been to establish a minimum wall thickness requirement in accordance with NRC Regulatory Guide 1.121, "Bases for Plugging Degraded PWR [pressurized water reactor] Steam Generator Tubes." Development of minimum wall thickness requirements is governed by analyses assuming a uniform thinning of the tube wall. This assumed degradation mechanism is inherently conservative for certain forms of steam generator tube degradation. Steam generator tube flaw acceptance criteria (i.e., plugging limits) are specified in the TSs.

Conservative plugging limits may lead to removing degraded tubes from service that may otherwise have adequate structural and leakage integrity for further service.

In early 1990's, the NRC staff developed generic criteria for voltage-based limits for ODSCC confined within the thickness of the tube support plates. The NRC staff published several conclusions regarding voltage-based repair criteria in draft NUREG-1477, "Voltage-Based Interim Plugging Criteria for Steam Generator Tubes" and in a draft generic letter titled "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes." The latter document was published for public comment in the *Federal Register* on August 12, 1994 (59 FR 41520). On August 3, 1995, the NRC staff issued Generic Letter (GL) 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking," which took into consideration public comments on the draft generic letter cited above, domestic operating experience under the voltage-based repair criteria, and additional data from European nuclear power plants.

The guidance of GL 95-05 does not set depth-based limits on predominantly axially oriented ODSCC at tube support plate locations; rather it relies on empirically derived correlations between bobbin coil voltage and tube burst pressure and correlations between bobbin coil voltage and leak rate. The guidance in GL 95-05 ensures structural and leakage integrity continue to be maintained at acceptable levels consistent with the requirements of Part 50 to Title 10 of the *Code of Federal Regulations* (10 CFR) and the guideline values in 10 CFR Part 100. Since the voltage-based repair criteria do not incorporate a limit on minimum tube wall thickness, there is the possibility for tubes with through-wall cracks to remain in service. Because of the increased likelihood of such flaws, the staff included provisions for augmented steam generator tube inspections and more restrictive operational leakage limits.

GL 95-05 provides an approach on voltage-based limits for ODSCC affecting the steam generator tubes at the tube support plate elevations. This approach takes no credit for the tube support plates in preventing or reducing the likelihood of a tube from bursting or leaking during postulated accident conditions. In essence it assumes that the degradation affecting the steam generator tubes at the tube support plate elevation is in the tube free span.

GL 95-05 specifies, in part, that (1) the repair criteria are only applicable to predominantly axially oriented ODSCC located within the bounds of the tube support plates; (2) licensees should perform an evaluation to confirm that the steam generator tubes will retain adequate structural and leakage integrity until the next scheduled inspection; (3) licensees should adhere to specific inspection criteria to ensure consistency in methods between tube inspections; (4) tubes must be periodically removed from the steam generators to verify the morphology of the degradation and provide additional data for structural and leakage integrity evaluations; (5) the operational leakage limit in the TSs should be reduced; (6) licensees should implement an operational leakage monitoring program; and (7) specific reporting requirements should be incorporated into the TSs.

CPSES, Units 1 and 2, are Westinghouse 4-loop pressurized water reactor plants that use model D4 steam generators with 3/4-inch diameter mill annealed Alloy 600 tubing and drilled hole carbon steel tube support plates. These steam generators have a flow distribution baffle plate located about 8 inches above the top of the tubesheet.

3.0 EVALUATION

The NRC staff has evaluated the acceptability of the licensee's 1.0 Volt Steam Generator Tube Repair Criteria for CPSES, Unit 1. Since the tube repair criteria rest on certain assumptions regarding steam generator leakage, and since such leakage affects post accident doses, the NRC has evaluated the acceptability of the licensee's revised dose assessment.

3.1 Applicability of GL 95-05 To Flow Distribution Baffle Plates

GL 95-05 does permit application of the voltage-based repair criteria at the flow distribution baffle plate provided the licensee addresses the potential for higher flaw growth rates at the flow distribution baffle plate. Two plants have experienced high flaw growth rates at their flow distribution baffle plate intersections.

The first event occurred at a European plant. The event was attributed to the incomplete rinsing of the steam generator after removing copper during a secondary side cleaning process. The inadequate rinsing process resulted in large amounts (up to 1 inch deep) of highly concentrated copper products to become deposited atop the flow distribution baffle plate. After the plant returned to power, a primary-to-secondary leak was detected and the plant was shut down to address the leakage. Axial outside diameter stress corrosion cracking was observed immediately above the flow distribution baffle by eddy current testing.

The second event involved a United States (U.S.) plant that observed large flaw growth in tubes in one steam generator at the flow distribution baffle plate. The cause of the cracking was attributed to high copper concentrations, caustic crevice conditions, and misalignment of a flow distribution baffle hole. The misaligned baffle hole led to partial packing of the holes and contact between the tube and the hole. The source of the high copper ingress was attributed to ammonia breakthrough in a steam generator demineralizer bed, which supplied large quantities of ammonia and sodium species into the feedwater. The ammonia accelerated the transport of copper species from the copper moisture separator reheater tubes to the steam generator. Copper acts as an oxidizer, thereby accelerating the corrosion process. The sodium-chloride molar ratio was found to be elevated during a major portion of the cycle with the high flaw growth rates. The licensee of the U.S. plant concluded that the high growth rate of indications was attributed to accumulation of caustic material in a crevice caused by the sodium transport from the demineralizer bed, the presence of copper, and a misaligned baffle plate.

The licensee has performed chemical cleaning at CPSES, Unit 1, and found no copper or copper alloy in the feedwater train. The licensee has minimized copper in the secondary system and has met the Electric Power Research Institute's guidelines for secondary chemistry control. The licensee stated that no copper species should be present on the secondary side of the Unit 1 steam generators and accelerated corrosion caused by copper species is not expected.

The licensee has assessed the potential for high growth rates of cracks at the flow distribution baffle plate intersections. In its response to the December 31, 1997, issuance of GL 97-06, "Degradation of Steam Generator Internals," the licensee stated that it has performed visual inspection and found no contact between the flow distribution baffle holes and the tube. Also, the licensee has not detected any bobbin indications in the tubes at baffle plate intersections in the most recent inspection in the CPSES, Unit 1, steam generators. CPSES, Unit 1, has been operated for seven cycles. The licensee has found insignificant flaw growth from inspection results reported from other nuclear plants whose steam generators use a flow distribution baffle plate. Based on the submitted information, the NRC staff agrees with the licensee's assessment. Accordingly, the NRC staff judges that the potential for high flaw growth at the flow distribution baffle plate intersections is low.

3.2 Tube Repair Limits (Background)

The proposed criteria will (1) permit tubes having indications confined to within the thickness of the tube support plates with bobbin voltages less than or equal to 1.0 volts to remain in service; (2) permit steam generator tubes having indications confined to within the thickness of the tube support plates with bobbin voltages greater than 1.0 volts but less than or equal to the upper voltage limit to remain in service if a motorized rotating pancake coil probe or acceptable alternative inspection does not detect degradation; and (3) require steam generator tubes having indications confined to within the thickness of the tube support plates with bobbin voltages greater than the upper voltage limit be plugged or repaired.

The lower voltage limit of 1.0 volts is consistent with the recommended value specified in GL 95-05 for 3/4-inch steam generator tubing. The upper voltage limit is derived based on the lower 95 percent prediction interval of the burst pressure versus bobbin voltage correlation, adjusted for lower bound material properties evaluated at the 95 percent confidence level. The upper voltage limit is further reduced to account for uncertainty in the nondestructive examination technique and flaw growth over the next operating cycle. Using this reduced lower prediction bound curve, the structural limit is determined for a free span burst pressure of 1.43 times main steamline break differential pressure. The proximity of the tube support plate prevents burst during normal operating conditions. The industry periodically updates the database for burst pressure and bobbin voltage when the destructive test data from pulled tubes are available; therefore, the upper voltage limit may vary as additional data are incorporated into the database.

The increased gap between the tube and flow distribution baffle plate does not provide sufficient constraint such that burst will not occur within the flow distribution baffle plate. Therefore, to determine the upper voltage repair limit for the flow distribution baffle plate follows the same methodology as for tube support plate except that the tube structural limit should be determined for a free span burst pressure of 3 times normal operational differential pressure or 1.43 times main steamline break differential pressure, whichever is more limiting in accordance with the guidance in Regulatory Guide 1.121 and GL 95-05.

3.3 Alternatives to GL 95-05 - Inspection Issues

The licensee has proposed to follow guidance in GL 95-05 except for considerations regarding probe variability, probe wear, and probability of detection.

3.3.1 Probe Variability

With respect to probe variability considerations as given in Section 3.c.2 of Attachment 1 to GL 95-05, the licensee proposed to follow an approach developed through the Nuclear Energy Institute (NEI). The proposed procedures and methodology are described in the January 23 and October 15, 1996, letters from Alex Marion of NEI to Brian Sheron of the NRC. The industry indicated that testing of new probes using only the primary frequency was not sufficient. This was based on a review of data used originally to support the notion that only the primary frequency was required to test new probes to satisfy the voltage variability specification of plus or minus 10 percent of the nominal response. The proposed approach specifies that the voltage responses from the primary frequency and mix frequency channel of new probes be within plus or minus 10 percent of the nominal voltage response. The nominal voltage responses were established as the average voltages obtained from the American Society of Mechanical Engineers standard drilled hole flaws for at least 10 production probes. In a letter from Brian Sheron of the NRC to David Modeen of NEI dated July 29, 1997, the NRC indicated that the NEI-proposed approach is acceptable because it satisfies the intent of probe variability in Section 3.c.2 of Attachment 1 to GL 95-05. Therefore, the licensee's proposal to follow the industry approach is acceptable to the NRC staff.

3.3.2 Probe Wear

Section 3.c.3 of Attachment 1 to GL 95-05 provides guidance for consideration of probe wear. The licensee proposed to use an alternative to Section 3.c.3. The alternative approach, developed through NEI, specifies that if the probe does not satisfy the voltage variability criterion for wear of plus or minus 15 percent limit before its replacement, all steam generator tubes that exhibited flaw signals with voltage responses measured at 75 percent or greater of the lower repair limit (i.e., 1 volt) must be reinspected with a bobbin probe satisfying the plus or minus 15 percent wear standard criterion. The voltages from the reinspection should be used as the basis for tube repair. The NRC staff completed a review of the NEI-proposed alternative method and concluded that the approach is acceptable as discussed in a letter from Brian Sheron of the NRC to Alex Marion of NEI dated February 9, 1996, because NEI's approach satisfied Section 3.c.3 of Attachment 1 to GL 95-05. The NRC staff finds that licensee's proposal to follow the NEI approach is acceptable.

3.3.3 Alternate Probability of Detection

The licensee requested NRC staff approval to use a voltage-dependent probability of detection instead of the constant probability of detection of 0.6 specified in GL 95-05. The voltage-dependent probability of detection approach calculates the distribution of bobbin indications as a function of voltage at the beginning of a cycle. The NRC staff is currently addressing such an approach generically through NEI. The NRC staff, however, has not made any determination regarding this matter. Accordingly, the NRC staff is deferring action on the licensee's request regarding probability of detection, but expects that any generic determination on this issue will apply to the licensee's request. Until the NRC staff makes a generic determination regarding probability of detection, the licensee should use a constant value of 0.6. The NRC staff and the licensee have discussed this matter and have agreed that the licensee will use a constant value of 0.6 until this issue is resolved.

3.4 <u>Structural and Leakage Integrity Assessments</u>

The staff focuses the implementation of the voltage-based repair criteria on maintaining tube structural integrity during the full range of normal, transient, and postulated accident conditions with adequate allowance for eddy current test uncertainty and flaw growth projected to occur during the next operating cycle. Tube structural limits based on Regulatory Guide 1.121 criteria include maintaining a margin of safety of 1.43 against tube failure under postulated accident conditions and maintaining a margin of safety of 3 against burst during normal operation. Because GL 95-05 addresses steam generator tubes affected with ODSCC confined to within the thickness of the steam generator tube support plate during normal operation, the NRC staff concluded that the structural constraint provided by the tube support plate ensures all tubes to which the voltage-based criteria apply will retain a margin of 3 with respect to burst under normal operating conditions. For a postulated main steamline break accident, however, the steam generator tube support plate may displace axially during steam generator blowdown such that the ODSCC affected portion of the tubing may no longer be fully constrained by the tube support plate. Accordingly, it is appropriate to consider the ODSCC affected regions of the tubes as free standing tubes for the purpose of assessing burst integrity under postulated main steamline break conditions.

In order to ensure that the structural and leakage integrity of the steam generator tube will be maintained until the next scheduled inspection, GL 95-05 specifies a methodology to determine the conditional burst probability and the total primary-to-secondary leak rate from an affected steam generator during a postulated main steamline break event occurring just prior to the next scheduled inspection outage. To complete GL 95-05 prescribed assessments, the licensee proposes to follow the methodology described in WCAP-14277, Revision 1, "SLB [steam line break] Leak Rate and Tube Burst Probability Analysis Methods for ODSCC at Tube Support Plate Intersections," dated December 1996. The NRC staff finds the methodology in WCAP-14277, Revision 1, acceptable as addressed in the March 13, 1997, NRC staff Safety Evaluation regarding the 1.0 Volt Steam Generator Tube Repair Criteria for the D. C. Cook nuclear power facility.

GL 95-05 specifies that the structural and leakage integrity assessments should use the latest available data from destructive examination of tubes removed from Westinghouse-designed steam generators. The NRC staff has agreed with NEI on a protocol by which the industry will periodically update the ODSCC database used to perform GL 95-05 specified calculations. The protocol ensures that the latest available data from destructive examination of tubes is considered. The licensee stated that it will follow the protocol. The staff finds this acceptable.

3.4.1. Conditional Probability of Burst

The licensee will use the methodology described in Revision 1 of WCAP-14277 for performing a probabilistic analysis to quantify the potential for steam generator tube ruptures given a main steamline break event. The results of the probabilistic analysis will be compared to a threshold value of $1x10^{-2}$ per cycle in accordance with GL 95-05. The NRC staff concludes that the licensee's proposed methodology for calculating the conditional burst probability is consistent with the guidance in GL 95-05 and is, therefore, acceptable.

3.4.2 Accident Leakage

The licensee will use the methodology described in Revision 1 of WCAP-14277 for calculating the steam generator tube leakage from the faulted steam generator during a postulated main steamline break event. The model consists of two major components: (1) a model predicting the probability that a given indication will leak as a function of voltage; and (2) a model predicting leak rate as a function of voltage, given that leakage occurs. The NRC staff concludes that the licensee's proposed methodology for calculating the steam generator tube leakage is consistent with the guidance in GL 95-05 and is, therefore, acceptable.

3.4.3 Primary-to-Secondary Leakage During Normal Operation

Because the voltage-based repair criteria would allow degraded tubes to remain in service, the degraded tubes may develop through-wall cracks during an operational cycle, thus creating the potential for primary-to-secondary leakage during normal operation, transients, or postulated accidents. Therefore, as a defense-in-depth measure, GL 95-05 specifies that the operational leakage limits of the TSs be limited to 150 gallons per day from any one steam generator. The licensee has proposed to limit primary-to-secondary leakage to 150 gallons per day from any one steam generator. The NRC staff concludes that proposed TS 3.4.13, which limits steam generator primary-to-secondary leakage to 150 gallons per day per steam generator, is consistent with GL 95-05. Because this change reduces allowable leakage and will reduce the consequences of any through-wall cracks in steam generator tubes, the NRC staff concludes that the change to TS 3.4.13 is acceptable.

3.5 Tube Pulls

To confirm the nature of the degradation occurring at the steam generator tube support plate elevations, tubes are periodically removed from the steam generators for destructive analysis. Tube pulls can confirm that the nature of the degradation being observed at these locations is predominantly axially oriented ODSCC, provide data for assessing the reliability of the inspection methods, and supplement the existing databases (e.g. burst pressure, probability of leakage, and leak rate).

GL 95-05 specifies that licensees should remove at least two pulled steam generator tube specimens with the objective of retrieving as many intersections as practical (a minimum of four intersections) during the steam generator inspection outage that implements the voltage-based repair criteria or during an inspection outage preceding initial application of the voltage-based criteria. On an ongoing basis, additional steam generator tube specimen removals (minimum of two intersections) should be obtained at the first refueling outage following 34 effective full power months of operation or at the maximum interval of three refueling outages after the previous tube pull. Alternatively, the licensee may participate in an industry-sponsored tube pull program endorsed by the NRC staff as described in GL 95-05. Currently, no tubes have been removed from the CPSES units. In its response to GL 95-05, the licensee stated that it will comply with Section 4 of GL 95-05 with regard to tube pulls. The NRC staff finds this acceptable.

3.6 Changes to the TSs Associated with the 1.0 Volt Steam Generator Tube Repair Criteria and Reporting

To implement the 1.0 Volt Steam Generator Tube Repair Criteria, the licensee has proposed the following changes to the TSs:

TS 5.5.9.b.2.d - This proposed change requires that steam generator tubes with indications left in service as a result of the application of the alternate repair criteria will be inspected by bobbin probes.

TS 5.5.9.b.4 - This proposed change specifies regions of the steam generator tubes that need to be inspected for outside diameter stress corrosion cracking indications.

TS 5.5.9.d.1.f - This proposed change specifies that the current plugging limit does not apply to steam generator tube support plate intersections for which the voltage-based alternate repair criteria are being applied.

TS 5.5.9.d.j - This proposed change includes specific bobbin voltage criteria regarding which steam generator tubes should remain in service and which steam generator tubes should be repaired. It also includes specific voltage limits if a mid-cycle inspection is required.

ITS 5.6.10.d - This proposed change requires the licensee to notify the NRC staff before returning the steam generators to service if certain indications are detected, estimated leakage exceeds the leakage limit, or the calculated conditional burst probability exceeds 1 X 10E-2.

B 3.3.13 - This is a change to the Bases section. The change discusses the technical basis of the alternate repair criteria, the voltage structural limit, and the reporting requirement.

The above proposed TS changes and Bases change are consistent with the model TSs in GL 95-05 and, for reasons set forth in Sections 3.1 through 3.5 of this Safety Evaluation, are acceptable.

3.7 Dose Assessment

The licensee revised the main steamline break (MSLB) accident radiological dose consequences analysis to assume an accident leak rate that results in calculated doses approaching the limits in 10 CFR Part 100 for the pre-accident iodine spike case, or a small fraction (10 percent) of 10 CFR Part 100 limits for the accident-initiated iodine spike case. This analysis provided for a maximum primary-to-secondary leak rate limit against which the predicted end-of-cycle leakage would be compared. The amendment request proposed that CPSES, Unit 1, be approved to operate based upon an accident-initiated primary-to-secondary leak rate of 27.79 gallons per minute (gpm) (at room temperature and pressure) in the faulted steam generator and an allowable value for the primary to secondary leakage in each of the three intact SGs equal to the proposed TS limit (150 gallons per day (gpd) per SG). The licensee found the radiological dose consequences of incorporating these changes to be acceptable based on the NRC acceptance criteria for doses at the Exclusion Area Boundary (EAB), Low Population Zone (LPZ), and the control room.

The staff reviewed the licensee's MSLB evaluation and performed confirmatory calculations to determine the acceptability of the licensee's methodology and resulting doses. Consistent with the guidance in NUREG-0800, Standard Review Plan (SRP) Section 15.1.5, Appendix A, the licensee analyzed two cases:

(1) an MSLB with a preaccident iodine spike of 60 μCi/gm dose equivalent (DE) I-131 in the primary coolant, and

(2) an MSLB with an accident initiated iodine spike with an appearance rate 500 times the TS equilibrium primary coolant activity of 1.0 μCi/gm DE I-131.

The current CPSES Final Safety Analysis Report analyzed the consequences of a third scenario based on assumed failure of the clad in 5 percent of the fuel rods. Because no fuel failures are attributed to the MSLB, the licensee did not consider this scenario in this reanalysis. The NRC staff agrees and finds this acceptable.

The licensee's analysis assumed a primary-to-secondary leak rate initiated by an MSLB accident for the faulted steam generator of 27.79 gpm (measured at room temperature conditions), and also assumed the TS primary-to-secondary leakage value of 150 gpd per steam generator in the intact steam generators. The licensee performed the revised radiological analysis using thyroid dose conversion factors from the International Commission on Radiation Protection Publication 30 (ICRP-30). The staff has generally accepted the use of ICRP-30 dose conversion factors, and such use is consistent with current industry standards as addressed in the NRC staff Safety Evaluation associated with Amendment No. 163 for the St. Lucie, Unit 1, nuclear power facility issued on August 18, 1999. The licensee's calculated doses are within the acceptance criteria, as shown in Table 1 attached. The licensee's calculated offsite dose consequences meet the acceptance criteria given in SRP 15.1.5, Appendix A, namely, (1) preaccident iodine spike calculated dose consequences are within 10 CFR Part 100 standards and (2) for an accident-initiated iodine spike, such consequences are a small fraction (10 percent) of the 10 CFR Part 100 standards. The control room doses meet the acceptance criteria given in 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 19.

The staff determined that the assumptions used in the licensee's MSLB accident analysis are acceptable. The staff also performed confirmatory calculations using the licensee's assumptions and the methodology associated with SRP 15.1.5, Appendix A, and confirmed the licensee's results.

3.8 Proposed TS for Steam Generator Primary-to-Secondary Leak

As set forth above, the NRC staff concludes that for CPSES, Unit 1, the radiological dose consequences of an MSLB with a TS value for total primary-to-secondary leakage of 150 gpd per steam generator and an accident-initiated leak rate of 27.79 gpm are within the criteria given in SRP 15.1.5, Appendix A, and GDC-19. Accordingly, the adoption of the 150 gpd, per steam generator, leakage rate, in proposed TS 3.4.13, is acceptable.

3.9 Summary

In summary, the licensee submitted proposed TSs to permit the use of the voltage-based repair criteria for steam generator tubes at CPSES, Unit 1. The NRC staff has reviewed the proposed TSs and concludes that the proposed alternate repair criteria are consistent with GL 95-05 and are acceptable. The NRC staff concludes that adequate structural and leakage integrity can be assured, consistent with applicable regulatory requirements, for indications to which the voltage-based repair criteria will be applied. The NRC staff approves the proposed voltage-based repair criteria based, in part, on the licensee being able to successfully demonstrate after each inspection outage that the conditional probability of burst and the primary-to-secondary leakage during a postulated MSLB will be acceptable in accordance with the guidance in GL 95-05.

Concerning the use of a voltage-dependent probability of detection, the staff is currently addressing such an approach generically through NEI. The licensee may use a voltage-dependent probability of detection if and when it is approved by the NRC. Until that occurs, the licensee should use a constant value of 0.6.

4.0 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.

5.0 ENVIRONMENTAL CONSIDERATION

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 (64 FR 24202 dated May 5, 1999). The amendments also change recordkeeping, reporting, or administrative procedures or requirements. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (c)(10). 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.

6.0 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.

Attachment: Table 1

Principal Contributors: J. Tsao
M. Hart
D. Jaffe

Date: September 22, 1999

Table 1
Licensee Calculated MSLB Doses

	Dose(rem)	Acceptance Criterion (rem)
EAB (0-2 hr)		
Thyroid: PIS[(1)]	43.50	300
Thyroid: GIS[(2)]	29.05	30
Whole Body	0.25	2.5
LPZ (0-8 hr)		
Thyroid: PIS[(1)]	22.26	300
Thyroid: GIS[(2)]	22.48	30
Whole Body	0.11	2.5
Control Room (0-8 hr)		
Thyroid: PIS[(1)]	6.72	30
Thyroid: GIS[(2)]	6.68	30
Whole Body	0.01	5

(1) Pre-accident iodine spike of 60 μCi/gm DEI-131 in the primary coolant

(2) Accident initiated iodine spike with an appearance rate 500 times the equilibrium RCS specific activity of 1.0 μCi/gm DEI-131.