May 14, 1996

Mr. William T. Cottle
Executive Vice-President &
General Manager, Nuclear
Houston Lighting & Power Company
South Texas Project Electric
Generating Station
P. O. Box 289
Wadsworth, TX 77483

SUBJECT: SOUTH TEXAS PROJECT, UNIT 1 - AMENDMENT NO. 82 TO FACILITY OPERATING LICENSE NO. NPF-76 (TAC NO. M94537)

Dear Mr. Cottle:

The Commission has issued the enclosed Amendment No. 82 to Facility Operating License No. NPF-76 for the South Texas Project, Unit 1. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated January 22, 1996, as supplemented by letter dated April 18, 1996.

The amendment modifies the steam generator tube plugging criteria in TS 3/4.4.5, Steam Generators, and the associated Bases, to allow the implementation of alternate steam generator tube plugging criteria for the tube-to-tubesheet joints (known in the industry as F*) for Unit 1.

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,
Original signed by
Thomas W. Alexion, Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-498

Enclosures: 1. Amendment No. 82 to NPF-76
2. Safety Evaluation

cc w/encls: See next page

DISTRIBUTION:

Docket File	GHill (4)	PUBLIC	JTsao
PNoonan	TAlexion (2)	CGrimes	PDIV-1 r/f
ACRS	WBeckner	LHurley, RIV	JKilcrease, RIV f/r
OGC	JRoe	EAdensam (EGA1)	JDyer, RIV

Document Name: STP94537.AMD

OFC	LA/PD4-1	PM/PD4-1	OGC NLO w/noted
NAME	PNoonan	TAlexion/sp	MYoung
DATE	5/14/96	5/8/96	5/13/96
COPY	YES/NO	YES/NO	YES/NO

TWA signed on 5/14/96

OFFICIAL RECORD COPY

9605170307 960514
PDR ADOCK 05000498
P PDR

)125

COPY

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

May 14, 1996

Mr. William T. Cottle
Executive Vice-President &
General Manager, Nuclear
Houston Lighting & Power Company
South Texas Project Electric
Generating Station
P. O. Box 289
Wadsworth, TX 77483

SUBJECT: SOUTH TEXAS PROJECT, UNIT 1 - AMENDMENT NO. 82 TO FACILITY OPERATING LICENSE NO. NPF-76 (TAC NO. M94537)

Dear Mr. Cottle:

The Commission has issued the enclosed Amendment No. 82 to Facility Operating License No. NPF-76 for the South Texas Project, Unit 1. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated January 22, 1996, as supplemented by letter dated April 18, 1996.

The amendment modifies the steam generator tube plugging criteria in TS 3/4.4.5, Steam Generators, and the associated Bases, to allow the implementation of alternate steam generator tube plugging criteria for the tube-to-tubesheet joints (known in the industry as F*) for Unit 1.

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,

Thomas W. Alexion

Thomas W. Alexion, Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-498

Enclosures: 1. Amendment No. 82 to NPF-76
2. Safety Evaluation

cc w/encls: See next page

Mr. William T. Cottle
Houston Lighting & Power Company

South Texas, Units 1 & 2

cc:

Mr. David P. Loveless
Senior Resident Inspector
U.S. Nuclear Regulatory Commission
P. O. Box 910
Bay City, TX 77414

Mr. J. C. Lanier/M. B. Lee
City of Austin
Electric Utility Department
721 Barton Springs Road
Austin, TX 78704

Mr. M. T. Hardt
Mr. W. C. Gunst
Central Public Service Board
P. O. Box 1771
San Antonio, TX 78296

Mr. G. E. Vaughn/C. A. Johnson
Central Power and Light Company
P. O. Box 289
Mail Code: N5012
Wadsworth, TX 74483

INPO
Records Center
700 Galleria Parkway
Atlanta, GA 30339-3064

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 1000
Arlington, TX 76011

Mr. Joseph M. Hendrie
50 Bellport Lane
Bellport, NY 11713

Judge, Matagorda County
Matagorda County Courthouse
1700 Seventh Street
Bay City, TX 77414

Jack R. Newman, Esq.
Morgan, Lewis & Bockius
1800 M Street, N.W.
Washington, DC 20036-5869

Mr. Lawrence E. Martin
General Manager, Nuclear Assurance Licensing
Houston Lighting and Power Company
P. O. Box 289
Wadsworth, TX 77483

Rufus S. Scott
Associate General Counsel
Houston Lighting and Power Company
P. O. Box 61867
Houston, TX 77208

Joseph R. Egan, Esq.
Egan & Associates, P.C.
2300 N Street, N.W.
Washington, DC 20037

Office of the Governor
ATTN: Andy Barrett, Director
Environmental 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

J. W. Beck
Little Harbor Consultants, Inc.
44 Nichols Road
Cohasset, MA 02025-1166

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

HOUSTON LIGHTING & POWER COMPANY

CITY PUBLIC SERVICE BOARD OF SAN ANTONIO

CENTRAL POWER AND LIGHT COMPANY

CITY OF AUSTIN, TEXAS

DOCKET NO. 50-498

SOUTH TEXAS PROJECT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 82
License No. NPF-76

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

   A. The application for amendment by Houston Lighting & Power Company* (HL&P) acting on behalf of itself and for the City Public Service Board of San Antonio (CPS), Central Power and Light Company (CPL), and City of Austin, Texas (COA) (the licensees), dated January 22, 1996, as supplemented by letter dated April 18, 1996, 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.

---

*Houston Lighting & Power Company is authorized to act for the City Public Service Board of San Antonio, Central Power and Light Company and City of Austin, Texas and has exclusive responsibility and control over the physical construction, operation and maintenance of the facility.

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-76 is hereby amended to read as follows:

   2. Technical Specifications

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 82, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee 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 to be implemented within 10 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas W. Alexion

Thomas W. Alexion, Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: May 14, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 82

FACILITY OPERATING LICENSE NO. NPF-76

DOCKET NO. 50-498

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain marginal lines indicating the areas of change. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE	INSERT
3/4 4-13	3/4 4-13
3/4 4-16	3/4 4-16
B 3/4 4-2a	B 3/4 4-2a
B 3/4 4-3	B 3/4 4-3

SURVEILLANCE REQUIREMENTS (Continued)

3) A tube inspection (pursuant to Specification 4.4.5.4a.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.

c. The tubes selected as the second and third samples (if required by Table 4.4-2) during each inservice inspection may be subjected to a partial tube inspection provided:

1) 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

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

d. For Unit 1, any tube allowed to remain in service per Acceptance Criterion 10 (of Technical Specification 4.4.5.4) shall be inspected via the rotating pancake coil (RPC) eddy current method over the F* distance. Such tubes are exempt from eddy current inspection over the portion of the tube below the F* distance which is not structurally relevant.

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 one 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.

SURVEILLANCE REQUIREMENTS (Continued)

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

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. 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;

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 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 4.4.5.3a.; the interval may then be extended to a maximum of once per 40 months; and

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:

   1) Primary-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2, or

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

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

   4) A main steam line or feedwater line break.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.4 Acceptance Criteria

a. As used in this specification:

1) 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;

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

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

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

5) Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective;

6) 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;

7) Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in Specification 4.4.5.3c., above;

8) Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg; and

9) 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.

SURVEILLANCE REQUIREMENTS (Continued)

(10) F* criteria [For Unit 1 only] Tube degradation below a specified distance from the hard roll contact point at or near the top-of-tubesheet (the F* distance) can be excluded from consideration to the acceptance criteria stated in this section (i.e., plugging of such tubes is not required). The methodology for determination for the F* distance as well as the list of tubes to which the F* criteria is not applicable is described in detail in Topical Report - BAW 10203P, Revision 0.

b. 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 4.4-2.

4.4.5.5 Reports

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2;

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following the completion of the inspection. This Special Report shall include:

1) Number and extent of tubes inspected,

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

3) Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3 shall be reported in a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days and prior to resumption of plant operation. This report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

RELIEF VALVES (Continued)

C. Manual control of the block valve to: (1) unblock an isolated PORV to allow it to be used for manual control of reactor coolant system pressure (Item A), and (2) isolate the PORV with excessive seat leakage (Item B).

D. Manual control allows a block valve to isolate a stuck-open PORV.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the Reactor Coolant System and the Secondary Coolant System (primary-to-secondary leakage = 500 gallons per day per steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 500 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

STEAM GENERATORS (Continued)

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit of 40% of the tube nominal wall thickness. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Exclusion of certain areas of Unit 1 tubes from consideration has been analyzed using an F* criteria. The criteria allows service induced degradation deep within the tubesheet to remain in service. The analysis methodology determines the length of sound fully rolled expanded tubing required in the uppermost area within the tubesheet to preserve needed structural margins for all service conditions. The remainder of the tube, below the F* distance, is considered not structurally relevant and is excluded from consideration to the customary plugging criteria of 40% throughwall.

The amount of primary to secondary leakage from tubes left in service by application of the F* criterion has been determined by verification testing. This leakage has been considered in the calculation of postulated primary to secondary leakage under accident conditions. Primary to secondary leakage during accident conditions is limited such that the associated radiological consequences as a result of this leakage is less than the 10 CFR 100 limits.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission in a Special Report pursuant to Specification 6.9.2 within 30 days and prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS Leakage Detection Systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. These Detection Systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.6.2 OPERATIONAL LEAKAGE

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, OPERATIONAL LEAKAGE (Continued)

the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 gpm. This threshold value is sufficiently low to ensure early detection of additional leakage.

The total steam generator tube leakage limit of 1 gpm for all steam generators not isolated from the RCS ensures that the dosage contribution from the tube leakage will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of either a steam generator tube rupture or steam line break. The 1 gpm limit is consistent with the assumptions used in the analysis of these accidents. The 500 gpd leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

The 10 gpm IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the Leakage Detection Systems.

The specified allowed leakage from any RCS pressure isolation valve is sufficiently low to ensure early detection of possible in-series check valve failure. It is apparent that when pressure isolation is provided by two in-series check valves and when failure of one valve in the pair can go undetected for a substantial length of time, verification of valve integrity is required. Since these valves are important in preventing overpressurization and rupture of the ECCS low pressure piping which could result in a LOCA that bypasses containment, these valves should be tested periodically to ensure low probability of gross failure.

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valve is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 82 TO

FACILITY OPERATING LICENSE NO. NPF-76

HOUSTON LIGHTING & POWER COMPANY

CITY PUBLIC SERVICE BOARD OF SAN ANTONIO

CENTRAL POWER AND LIGHT COMPANY

CITY OF AUSTIN, TEXAS

DOCKET NO. 50-498

SOUTH TEXAS PROJECT, UNIT 1

1.0 INTRODUCTION

By application dated January 22, 1996, as supplemented by letter dated April 18, 1996, Houston Lighting & Power Company, et.al., (the licensee) requested changes to the Technical Specifications (Appendix A to Facility Operating License No. NPF-76) for the South Texas Project, Unit 1. The proposed changes would modify the steam generator tube plugging criteria in Technical Specification (TS) 3/4.4.5, Steam Generators, and the associated Bases, to allow the implementation of alternate steam generator tube plugging criteria for the tube-to-tubesheet joints (known in the industry as F*) for Unit 1.

The April 18, 1996, supplement provided clarifying information and did not change the initial no significant hazards consideration determination.

2.0 BACKGROUND

The licensee proposed an alternative repair criteria for defects found in the steam generator tube expansion region within the tubesheet. Under the amendment request, steam generator tubes with defects in excess of the current plugging limits could remain in service without repair provided the defects exist below a specified distance (F* distance). The F* distance is measured from the secondary face of the tubesheet or the top of the last hardroll, whichever is lower. The licensee presented a qualification test program, which was performed by Babcock and Wilcox Nuclear Technologies, to show that the proposed F* distance satisfies the necessary structural and leakage integrity requirements of Appendix A to 10 CFR Part 50 and the existing plant TSs. The test program was documented in Topical Report, BAW-10203P, Revision 0, which was included in the submittal.

The proposed F* criteria apply to Unit 1 only. Unit 1 uses Westinghouse E-Series steam generators. The tubes are fabricated from low temperature mill annealed Alloy 600 material. The tube-to-tubesheet joints in the Unit 1 steam generators are fully expanded by a hardroll process. The tube-to-tubesheet joints in the Unit 2 steam generators are expanded by a hydraulic process, which is not covered by this proposed amendment.

3.0 DISCUSSION

General Design Criterion (GDC) 14 of Appendix A to 10 CFR Part 50 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. Regulatory Guide (RG) 1.121 provides guidance on an acceptable method for establishing the limiting safe conditions of tube degradation of steam generator tubing. Surveillance requirements in the STP TSs require a periodic inspection of steam generator tubes. If a tube is found to have defects in excess of the TSs limits (i.e., 40 percent through-wall), it is required to be repaired or removed from service.

Steam generator tubes comprise a significant portion of the reactor coolant pressure boundary. Maintenance of this boundary is provided by the integrity of the steam generator tube wall and tube-to-tubesheet joint. The joint between the tube and tubesheet is an interference fit constructed by roll expanding the tube into the bore of the tubesheet. The tubes were inserted into the bore of the tubesheet followed by seal welding at the primary face of the tubesheet. Step rolls were then performed on the tube to fully expand the tube in the bore of the tubesheet. The tubes are restrained in the bore of the tubesheet by the elastic springback of the tubesheet. The tube-to-tubesheet joint provides sufficient strength to maintain adequate structural and pressure boundary integrity.

The industry experience has shown that defects have been developed in the tube-to-tubesheet joints by various degradation processes. The staff believes that tubes having degradation in the joint may remain in service provided that the degradation is below a specified distance and that the undegraded portion of the tube in the joint can maintain adequate structural and leakage integrity under loadings from normal operation, anticipated operational occurrences, and postulated accident conditions. RG 1.121 recommends that the margin of safety against tube rupture under normal operating conditions should not be less than three at any tube location where defects have been detected. For postulated accidents, RG 1.121 recommends that the margin of safety against tube rupture be consistent with the margin of safety determined by the stress limits specified in NB-3225 of Section III of the Boiler and Pressure Vessel Code of the American Society of Mechanical Engineers (ASME). Structural loads imposed on the tube-to-tubesheet joint primarily result from the differential pressure between the primary and secondary sides of the tubes. The peak postulated loading occurs during a main steam line break due to a lowering of the secondary side pressure. However, normal operating loads, cyclic loading from transients (e.g., startup/shutdown), and potential thermal expansion loads can also be significant. The analysis supporting the F* criteria should address all loading conditions necessary to maintain adequate integrity of the tube-to-tubesheet joint.

The elastic preload between the tube and tubesheet not only prevents pullout of the tube from the tubesheet, but also provides a leak tight barrier minimizing the potential for primary to secondary coolant leakage. With sufficient length of hardroll, the tube-to-tubesheet joint will not allow any leakage under normal and faulted conditions. If a through-wall crack is present in the joint, primary coolant may leak to the secondary side. Under the proposed F* criteria, the licensee has to demonstrate that leakage integrity of the joint is maintained under all analyzed conditions.

4.0 EVALUATION

4.1 Qualification Test Program

The licensee focused the qualification analyses and testing program for the F* criteria on (1) establishing axial tube loads based on operating and faulted conditions, (2) performing mechanical tests necessary to verify the F* criteria, (3) adjusting results from mechanical tests for actual steam generator conditions, and (4) performing eddy current inspection to determine the uncertainties associated with length measurements for final F* distance determination.

4.1.1 Fabrication of Test Specimens

The licensee fabricated several mockup blocks from the material with the same properties as the tubesheet material used in the Westinghouse E-Series steam generators. The blocks were 4 inches thick having a 4 by 4 array of holes to simulate the tubesheet. The F* tubes were located in the middle of the test block and surrounded by the undegraded test tubes. The F* tubes were installed as two separate sections in the bores of the blocks. The tube sections were inserted from the top and bottom of the blocks and they were roll expanded in the bore. The separation between tube sections represented a full 360 degree through-wall crack at the F* distance. Various installation parameters such as tubesheet bore diameter, tubesheet bore surface finish, block temperature, and tubing yield strength were considered to address ranges of possible steam generator conditions. After the tubes were expanded into the blocks, the blocks were heated to simulate the effects of actual steam generator service temperature.

4.1.2 Testing for F* Determination

The licensee used a combination of internal pressure and axial loading to simulate the applied loads on the steam generator tubes. The radial preload stress, thermal effect, internal (primary side) pressure effect, and tubesheet bow effect were considered. The radial preload stress was calculated from the tube springback distance. The springback distance was determined by cutting the block away from the test tube and comparing the difference between the outer diameter of the expanded tube and the bore of the block. The thermal effect was determined from the differential thermal expansion of the tube and tubesheet in the radial direction. The primary-to-secondary differential pressure acts to both increase and decrease the strength of the tube-to-tubesheet joint. The primary pressure in the tube increases the radial stress on the outside surface of the tube and thus increases the joint strength. However, the primary-to-secondary differential pressure causes tubesheet bow and, therefore, dilates the tubesheet bore. When the bore dilates, the joint strength is reduced. Overall, the primary pressure would slightly reduce the strength of the tube-to-tubesheet joint.

A series of tensile tests were conducted to determine strength of the tube-to-tubesheet joint under the maximum load, locked tube load, pressure cycling load, and ultimate load. For the maximum load test, axial and pressure loads were applied to the joint. The axial loads included RG 1.121 recommended safety factors of 1.43 and 3 for the faulted and normal conditions, respectively. For the locked tube test, the tube was assumed to be locked at a tube support plate and the loads caused by the difference in thermal expansion of the tube, tube support plate, and associated tube assembly were used. For the pressure cycling test, the test tubes were subjected to cycling pressure simulating transient conditions and joint slippage was monitored to determine the structural integrity of the joints. For the ultimate load test, the test tubes were internally pressurized and subjected to an increasing axial tensile load until failure. Failure was defined as a relative movement of a specific distance between the tube and tubesheet. The tested engagement lengths showed acceptable structural adequacy for all loading conditions.

As part of the test program, the test tubes were subjected to leak rate testing. The tubes were internally pressurized to simulate differential pressures during normal operating and faulted conditions. The acceptance criteria for these tests were based on the 1 gallon per minute requirement of the plant TSs. Low leak rates were observed for the normal operating condition and faulted condition.

Applying the F* distance to the tubes in the field would require measuring the tube length by eddy current probes. The licensee stated that any uncertainties associated with eddy current measurement beyond an eddy current indication will be considered in the F* distance. To determine the measurement uncertainty, the licensee used various probes with various frequencies and rate of probe pull. A maximum measurement uncertainty was obtained and was included in the final F* distance calculation.

The licensee considered the effects of boric acid corrosion on the steam generator tubes because if primary coolant leaked through the tube wall and came in contact with the carbon steel tubesheet, stress corrosion cracking may occur in the tubesheet. On the basis of previous industry studies, the licensee concluded that any corrosion that might occur is limited only to localized degradation in the bore of the tubesheet with an insignificant depth. The licensee concluded that this would not affect the F* tube nor the structural integrity of the tubesheet.

4.1.3 F* Test Results

Based on the results of the leakage rate and mechanical testing, the licensee determined a nominal engagement length necessary to ensure adequate margins of structural integrity as reported in BAW-10203P, Revision 0. The length also included uncertainties that accounts for limited sample size, statistical scatter in the data, and eddy current inspection error.

In addition, the licensee identified a number of tubes having "wavy" roll expansion in the Unit 1 steam generators as listed in Table 3.3.3 of BAW-10203P, Revision 0. The wavy tubes do not have full 360 degree roll contact in the tubesheet because of certain rolling anomalies. The licensee has not qualified these tubes for structural and leakage integrity and has requested that the wavy tubes be excluded from the application of the F* criteria. The staff finds this exclusion acceptable.

4.2 Evaluation of Proposed Technical Specification Changes

The proposed F* criteria in the applicable TS sections are as follows:

TS 4.4.5.2.d:

> For Unit 1, any tube allowed to remain in service per Acceptance Criterion 10 (of Technical Specification 4.4.5.4) shall be inspected via the rotating pancake coil (RPC) eddy current method over the F* distance. Such tubes are exempt from eddy current inspection over the portion of the tube below the F* distance which is not structurally relevant.

TS 4.4.5.4.a.10:

> F* Criteria [For Unit 1 only] Tube degradation below a specified distance from the hard roll contact point at or near the top-of-tubesheet (the F* distance) can be excluded from consideration to the acceptance criteria stated in this section (i.e., plugging of such tubes is not required). The methodology for determination for the F* distance as well as the list of tubes to which the F* criteria is not applicable is described in detail in Topical Report BAW-10203P, Revision 0.

The licensee also proposed changes to the associated Bases. The proposed Bases describes areas of Unit 1 tubes that are not structurally relevant and therefore are excluded from consideration to the customary plugging criteria, and describes the consideration of primary to secondary leakage from tubes left in service by the application of the F* criterion and the associated radiological consequences during postulated accident conditions.

The staff has determined that 1) the licensee's F* criteria were established on the basis of the qualification tests that used specimens simulating the actual tube-to-tubesheet joint configuration of the South Texas steam generators, 2) the applied loads for structural and leakage assessment were specified in accordance with RG 1.121, and 3) the licensee has committed to reinspect the F* tubes during all future outages to ensure integrity of the F* tubes. The continued inspection of the F* tubes will minimize the potential for the degradation to occur inside the F* distance of all the F* tubes.

4.3 Summary

The proposed F* criteria would permit steam generator tubes to remain in service with degradation in excess of the current plugging limit provided the degradation exists below the F* distance. The proposed F* distance as reported in Topical Report, BAW-10203, Revision 0, is measured down into the tubesheet from the secondary face of the tubesheet or the top of the last hardroll, whichever is lower. The staff concludes that the proposed F* criteria is acceptable for Unit 1 because the licensee has demonstrated through an acceptable qualification test program that the F* criteria satisfy GDC 14 and the guidance in RG 1.121. The licensee may incorporate the proposed changes into the TSs and Bases for the South Texas Project Electric Generating Station, Unit 1.

5.0 STATE CONSULTATION

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

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves 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 amendment involves no significant hazards consideration, and there has been no public comment on such finding (61 FR 7553). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: J. Tsao

Date: May 14, 1996