Mr. William T. Cottle
President and Chief Executive Officer
STP Nuclear Operating Company
South Texas Project Electric
Generating Station
P. O. Box 289
Wadsworth, TX 77483

April 19, 1999

SUBJECT: SOUTH TEXAS PROJECT, UNITS 1 AND 2 - AMENDMENT NOS. 107 AND 94 TO FACILITY OPERATING LICENSE NOS. NPF-76 AND NPF-80 (TAC NOS. MA3761 AND MA3762)

Dear Mr. Cottle:

The Commission has issued the enclosed Amendment Nos. 107 and 94 to Facility Operating License Nos. NPF-76 and NPF-80 for the South Texas Project, Units 1 and 2 (STP). The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated September 30, 1998.

The amendments change steam generator tube inspection surveillance-related TSs (and associated Bases pages). The future installation of the new Δ94 steam generators necessitates changes to the steam generator tube sample selection and inspection requirements; inservice inspection frequencies; acceptance criteria; and inspection reporting requirements.

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, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-498 and 50-499

Enclosures:
1. Amendment No. 107 to NPF-76
2. Amendment No. 94 to NPF-80
3. Safety Evaluation

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

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cc:

Mr. Cornelius F. O'Keefe
Senior Resident Inspector
U.S. Nuclear Regulatory Commission
P. O. Box 910
Bay City, TX 77414

A. Ramirez/C. M. Canady
City of Austin
Electric Utility Department
721 Barton Springs Road
Austin, TX 78704

Mr. M. T. Hardt
Mr. W. C. Gunst
City 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 400
Arlington, TX 76011

D. G. Tees/R. L. Balcom
Houston Lighting & Power Co.
P. O. Box 1700
Houston, TX 77251

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. T. H. Cloninger
Vice President
Engineering & Technical Services
STP Nuclear Operating Company
P. O. Box 289
Wadsworth, TX 77483

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

Jon C. Wood
Matthews & Branscomb
One Alamo Center
106 S. St. Mary's Street, Suite 700
San Antonio, TX 78205-3692

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

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

STP NUCLEAR OPERATING COMPANY

DOCKET NO. 50-498

SOUTH TEXAS PROJECT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 107
License No. NPF-76

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

   A. The application for amendment by STP Nuclear Operating Company* acting on behalf of itself and for Houston Lighting & Power Company (HL&P), the City Public Service Board of San Antonio (CPS), Central Power and Light Company (CPL), and City of Austin, Texas (COA) (the licensees), dated September 30, 1998, 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.

---

*STP Nuclear Operating Company is authorized to act for Houston Lighting & Power Company (HL&P), 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. 107 , 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, and shall be implemented following the replacement of Unit 1 Model E steam generators with Model Δ94 steam generators and prior to Unit 1 operation with the Δ94 steam generators installed.

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: April 19, 1999

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

STP NUCLEAR OPERATING COMPANY

DOCKET NO. 50-499

SOUTH TEXAS PROJECT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 94
License No. NPF-80

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

   A. The application for amendment by STP Nuclear Operating Company* acting on behalf of itself and for Houston Lighting & Power Company (HL&P), the City Public Service Board of San Antonio (CPS), Central Power and Light Company (CPL), and City of Austin, Texas (COA) (the licensees), dated September 30, 1998, 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.

---

*STP Nuclear Operating Company is authorized to act for Houston Lighting & Power Company (HL&P), 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-80 is hereby amended to read as follows:

   2. Technical Specifications

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 94 , 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 and shall be implemented following the replacement of Unit 1 Model E steam generators with Model Δ94 steam generators and prior to Unit 1 operation with the Δ94 steam generators installed.

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: April 19, 1999

ATTACHMENT TO LICENSE AMENDMENT NOS. 107 AND 94

FACILITY OPERATING LICENSE NOS. NPF-76 AND NPF-80

DOCKET NOS. 50-498 AND 50-499

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.

REMOVE	INSERT
3/4 4-11	3/4 4-11*
3/4 4-12	3/4 4-12
3/4 4-13	3/4 4-13
3/4 4-13a	3/4 4-13a*
3/4 4-14	3/4 4-14
3/4 4-15	3/4 4-15
3/4 4-16	3/4 4-16
3/4 4-16a	3/4 4-16a
3/4 4-16b	3/4 4-16b
3/4 4-18a	3/4 4-18a
B 3/4 4-2a	B 3/4 4-2a
B 3/4 4-3	B 3/4 4-3
B 3/4 4-3a	B 3/4 4-3a
B 3/4 4-4	B 3/4 4-4

* Overleaf pages provided to maintain document completeness. No changes contained on these pages.

SURVEILLANCE REQUIREMENTS

4.4.4.1 In addition to the requirements of Specification 4.0.5, each PORV shall be demonstrated OPERABLE at least once per 18 months by:

a. Performing a CHANNEL CALIBRATION on the actuation channel, and

b. Operating the valve through one complete cycle of full travel.

4.4.4.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed in order to meet the requirements of ACTION b. or c. in Specification 3.4.4.

3/4.4.5 STEAM GENERATOR

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.5.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2 and Table 4.4-3. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. When applying the exceptions of 4.4.5.2.a through 4.4.5.2.c, previous defects or imperfections in the area repaired by sleeving are not considered an area requiring reinspection. The tubes selected for each inservice inspection shall include at least 3% of the total number of nonrepaired tubes in all steam generators and (for Model E steam generators only) 20% of the total number of repaired tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

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

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

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

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

SURVEILLANCE REQUIREMENTS (Continued)

3) A tube inspection (pursuant to Specification 4.4.5.4a.9) 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.

4) For Model E steam generators only, indications left in service as a result of application of the tube support plate voltage-based repair criteria shall be inspected by bobbin coil probe during all future refueling outages.

c. The tubes selected as the second and third samples (if required by Table 4.4-2 or Table 4.4-3) 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 Model E steam generators only, implementation of the steam generator tube/tube support plate repair criteria requires a 100-percent bobbin coil inspection for hot-leg and cold leg tube support plate intersections down to the lowest cold-leg tube support plate 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-percent random sampling of tubes inspected over their full length.

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.

SURVEILLANCE REQUIREMENTS (Continued)

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 following steam generator replacement shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality after the steam generator replacement. 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 degradatiaon has occurred, the inspection interval may be extended to a maximum of once per 40 months;

   Note: Inservice inspection is not required during the steam generator replacement outage.

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-seconday 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.

4.4.5.4 Acceptance Criteria

a. As used in this specification:

1) Tubing or Tube means that portion of the tube or sleeve which forms the primary system to secondary system pressure boundary;

2) 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;

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

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

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

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

7) Plugging Limit or Repair Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging or (for Model E steam generators only) repaired by sleeving in the affected area because it may become unserviceable prior to the next inspection. The plugging or repair limit imperfection depths are specified in percentage of the nominal wall thickness as follows:

a. original tube wall	40%
b. Westinghouse laser welded sleeve wall	40%

For Model E steam generators, this definition does not apply to tube support plate intersections for which the voltage-based repair criteria are being applied. Refer to 4.4.5.4.a.11 for the repair limit applicable to these intersections.

8) 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;

9) 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;

10) 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.

11) For Model E steam generators only, Tube Support Plate Plugging Limit is used for the disposition of a mill annealed alloy 600 steam generator tube for continued service that is experiencing predominately axially oriented outside diameter stress corrosion cracking confined within the thickness of the tube support plates. At tube support plate intersections, the plugging (repair) limit is based on maintaining steam generator tube serviceability as described below:

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

   b) Steam generator tubes, whose degradation is 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 (Note 1), will be repaired or plugged, except as noted in 4.4.5.4.a.11.c below.

   c) 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 (Note 1) but less than or equal to the upper repair voltage limit (Note 2), may remain in service if a rotating pancake coil inspection does not detect degradation. Steam generator tubes, with indications of outside diameter stress corrosion cracking degradation with bobbin voltage greater than the upper voltage repair limit (Note 2) will be plugged or repaired.

SURVEILLANCE REQUIREMENTS (Continued)

d) If an unscheduled mid-cycle inspection is performed, the mid-cycle repair limits apply instead of the limits identified in 4.4.5.4.a.11.a, 4.4.5.4.a.11.b, and 4.4.5.4.a.11.c. The mid-cycle repair limits will be determined from the equations for mid-cycle repair limits of NRC Generic Letter 95-05, Attachment 2, page 3 of 7. Implementation of these mid-cycle repair limits should follow the same approach as in TS 4.4.5.4.a.11.a, 4.4.5.4.a.11.b, and 4.4.5.4.a.11.c.

Note 1: The lower voltage repair limit is 1.0 volt for 3/4-inch diameter tubing.

Note 2: The upper voltage repair limit (V_{URL}) is calculated according to the methodology in Generic Letter 95-05 as supplemented. V_{URL} may differ at the TSPs and flow distribution baffle.

12) Tube Repair refers to a process that reestablishes tube serviceability for Model E steam generators only. Acceptable tube repair will be performed in accordance with the methods described in Westinghouse Reports WCAP-13698, Revision 2, "Laser Welded Sleeves for 3/4 Inch Diameter Tube Feedring-Type and Westinghouse Preheater Steam Generators," April 1995 and WCAP-14653, "Specific Application of Laser Welded Sleeves for South Texas Project Power Plant Steam Generators," June 1996, including post-weld stress relief;

Tube repair includes the removal of plugs that were previously installed as a corrective or preventive measure. A tube inspection per 4.4.5.4.a.9 is required prior to returning previously plugged tubes to service.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions [plug or (for Model E steam generators only) repair all tubes exceeding the plugging or repair limit and all tubes containing through-wall cracks] required by Table 4.4-2 and Table 4.4-3.

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 or repaired in each steam generator shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2;

REACTOR COOLANT SYSTEM

STEAM GENERATORS

SURVEILLANCE REQUIREMENTS (Continued)

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 or repaired.

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.

d. For Model E steam generators, 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 leak 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.

Table 4.4-3

MODEL E STEAM GENERATOR REPAIRED TUBE INSPECTION

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

[1] Each repair method is considered a separate population for determination of scope expansion.

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 minimize 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 3.4.6.2.c limitation of steam generator tube leakage between the Reactor Coolant System and the Secondary Coolant System. 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 as low as 150 gallons per day per steam generator can readily be detected. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged or (for Model E steam generators only) repaired. Defective tubes in Model E steam generators may be repaired by a Westinghouse laser welded sleeve. The technical bases for sleeving repair are described in Westinghouse Reports WCAP-13698, Revision 2, "Laser Welded Sleeves for 3/4 Inch Diameter Tube Feedring-Type and Westinghouse Preheater Steam Generators," April 1995 and WCAP-14653, "Specific Application of Laser Welded Sleeves for South Texas Project Power Plant Steam Generators," June 1996.

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. Except as discussed below, plugging or (for Model E steam generators only) repair will be required for all tubes with imperfections exceeding the plugging or repair limit of 40% of the original tube nominal wall thickness. If a tube contains a Westinghouse laser welded sleeve with imperfection exceeding 40% of nominal wall thickness, it must be plugged. The basis for the sleeve plugging limit for Model E steam generators is based on Regulatory Guide 1.121 analysis, and is described in the Westinghouse sleeving technical reports listed above. Steam generator tube inspections of operating

STEAM GENERATORS (Continued)

plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness. Repaired tubes are also included in the inservice tube inspection program.

For Model E steam generators only, the voltage-based repair limits of SR 4.4.5 implement the guidance in GL 95-05 and are applicable only to Westinghouse-designed steam generators (SGs) with outside diameter stress corrosion cracking (ODSCC) located at the tube-to-tube support plate intersections. The criteria of GL 95-05 are also applicable to the Unit 2 flow distribution plate intersections. The voltage-based repair limits are not applicable to other forms of SG tube degradation nor are they applicable to ODSCC that occurs at other locations within the SG. Additionally, the repair criteria apply only to indications where the degradation mechanism is dominantly axial ODSCC with no significant cracks extending outside the thickness of the support plate. Refer to GL 95-05 for additional description of the degradation morphology.

Implementation of SR 4.4.5 for Model E steam generators requires a derivation of the voltage structural limit from the burst versus voltage empirical correlation and then the subsequent derivation of the voltage repair limit from the structural limit (which is then implemented by this surveillance).

The voltage structural limit is the voltage from the burst pressure/bobbin voltage correlation, at the 95-percent prediction interval curve reduced to account for the lower 95/95-percent tolerance bound for tubing material properties at 650°F (i.e., the 95-percent LTL curve). The voltage structural limit must be adjusted downward to account for potential flaw growth during an operating interval and to account for NDE uncertainty. The upper voltage repair limit; V_{URL}, is determined from the structural voltage limit by applying the following equation:

$$V_{URL} = V_{SL} - V_{GR} - V_{NDE}$$

where V_{GR} represent the allowance for flaw growth between inspections and V_{NDE} represents the allowance for potential sources of error in the measurement of the bobbin coil voltage. Further discussion of the assumptions necessary to determine the voltage repair limit are discussed in GL 95-05.

STEAM GENERATORS (Continued)

The mid-cycle equation in SR 4.4.5.4.a.11.e should only be used during unplanned inspections of Model E steam generators in which eddy current data is acquired for indications at the tube support plates.

SR 4.4.5.5 implements several reporting requirements for Model E steam generators recommended by GL 95-05 for situations which the NRC wants to be notified prior to returning the SGs to service. For the purpose of this reporting requirement, leakage and conditional burst probability can be calculated based on the as-found voltage distribution rather than the projected end-of-cycle voltage distribution (refer to GL 95-05 for more information) when it is not practical to complete these calculations using the projected EOC voltage distributions prior to returning the SGs to service. Note that if leakage and conditional burst probability were calculated using the EOC voltage distribution for the purposes of addressing the GL section 6.a.1 and 6.a.3 reporting criteria, then the results of the projected EOC voltage distribution should be provided per the GL section 6.b.(c) criteria.

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, 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.

OPERATIONAL LEAKAGE (Continued)

The leakage limits incorporated into SR 4.4.6 are more restrictive than the standard operating leakage limits and were implemented in conjuction with the application of voltage-based repair criteria and laser-welded sleeving to Model E steam generators. They were intended to provide an additional margin to accommodate a crack which might grow at a greater than expected rate or unexpectedly extend outside the thickness of the tube support plate. Hence, the reduced leakage limit, when combined with an effective leak rate monitoring program, provides additional assurance that should a significant leak be experienced in service, it will be detected, and the plant shut down in a timely manner. The additional margin provided by the reduced leakage limit will be retained with the Δ94 steam generators.

The steam generator tube leakage limit of 150 gpd for each steam generator 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 150 gpd limit per steam generator is conservative compared to the assumptions used in the analysis of these accidents. The 150 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 NOS. 107 AND 94 TO

FACILITY OPERATING LICENSE NOS. NPF-76 AND NPF-80

STP NUCLEAR OPERATING COMPANY

DOCKET NOS. 50-498 AND 50-499

SOUTH TEXAS PROJECT, UNITS 1 AND 2

1.0 INTRODUCTION

By application dated September 30, 1998, STP Nuclear Operating Company, et.al., (the licensee) requested changes to the Technical Specifications (TSs) (Appendix A to Facility Operating License Nos. NPF-76 and NPF-80) for the South Texas Project, Units 1 and 2 (STP). The proposed changes would revise TS Section 3/4.4.5, "Steam Generator" Surveillance Requirements. The future installation of the new Δ94 steam generators at STP necessitates changes to the steam generator tube sample selection and inspection requirements; inservice inspection frequencies; acceptance criteria; and inspection reporting requirements.

2.0 BACKGROUND

The Westinghouse Model E steam generators will be replaced with Westinghouse Model Δ94 steam generators. Unit 1 steam generator replacement is scheduled to occur at the end of Cycle 9, currently planned for spring of the year 2000. Unit 2 replacement is scheduled to occur at the end of Cycle 9, currently planned for the year 2002. The Δ94 steam generators have a number of different design features including a different tubing material, a different tube diameter and thickness, and a different tube support plate design.

The licensee proposes to amend TS Surveillance Requirements (SRs) for Specification 3/4.4.5, "Steam Generator" to support steam generator replacement. This proposed change limits the applicability of provisions for voltage-based repair criteria and laser-welded sleeving to Model E steam generators only (not applicable to the Δ94 steam generators), and deletes the F* alternate repair criteria. It also clarifies the inservice inspection schedule as it pertains to replacement steam generators (Model Δ94). The licensee proposes that this amendment be implemented following replacement of the Unit 1 Model E steam generators with Model Δ94 steam generators and prior to Unit 1 operation with the Δ94 steam generators installed.

As discussed above, the Δ94 steam generators have a number of different design features. As a result of these design differences, the analyses performed for use of voltage-based repair criteria for tube-to-tube support plate intersections, F* alternate repair criteria for tube-to-tubesheet joints, and tube repair by laser-welded sleeving are not applicable to the Δ94 steam generators.

The technical/topical reports and correspondence submitted by the licensee in association with the license amendments for F* alternate repair criteria, the voltage-based repair criteria, and laser-welded sleeving contain various commitments associated with the implementation of these license amendments. Following steam generator replacement in each respective unit, these commitments will no longer apply to that unit. The tubing of the fabricated Δ94 steam generators will be examined by eddy current methods in accordance with ASME Section XI requirements (i.e., preservice inspection will be performed).

Description of Proposed Changes

The proposed steam generator tube inspection surveillance-related TS changes (and associated BASES changes) resulting from replacement of the existing Model E steam generators with Δ94 steam generators are as follows:

SR 4.4.5.2, Steam Generator Tube Sample Selection and Inspection
SR 4.4.5.3, Inspection Frequencies
SR 4.4.5.4, Acceptance Criteria
SR 4.4.5.5, Reports
BASES for Specification 3/4.4.5, Steam Generators
BASES for Specification 3/4.4.6.2, Operational Leakage

The proposed changes delete provisions for the F* alternate repair criteria, and make the provisions in the TSs for voltage-based repair criteria and laser-welded sleeving applicable only to Model E steam generators, not to the Δ94 steam generators. Also, clarification of the TS describing the first inservice inspections is necessary to reflect steam generator replacement. In addition, the Steam Generator Tube Inspection Report requirements must be similarly revised to make the reporting requirements added for the voltage-based repair criteria applicable to Model E steam generators only. Finally, unrelated to the Replacement Steam Generator Project, an incorrect referenced Specification number is corrected in SR 4.4.5.2.b.3, and a superfluous "and" is deleted from SR 4.4.5.4.a.9. These proposed changes are administrative in nature.

The Bases section for Steam Generators has also been revised, as appropriate, to reflect the above TS changes. In addition, the Bases section for Operational Leakage, that discusses more restrictive operating leakage limits (for existing Model E steam generators), has been revised to specify that even with installation of new Δ94 steam generators, this more restrictive operating leakage limit will still apply.

In addition, two typographical errors in Bases section 3/4.4.6.2 are being corrected. The words "value" and "values" were inadvertently misspelled by the NRC during the issuance of Amendment 83 (May 22, 1996).

3.0 EVALUATION

The Δ94 steam generators have several design differences that were made to significantly improve the resistance to, or to eliminate, tube degradation mechanisms experienced by the Model E steam generators. These differences render the analyses which were the basis for the tube support plate voltage-based repair criteria, the F* alternate repair criteria, and laser-welded sleeving, not applicable to the Δ94 steam generators.

The Model E steam generators have tube support plates with drilled tube holes. The Δ94 steam generators have tube support plates with broached tube holes. The broached tube holes provide for open flow around the tube at the support plate intersections, with three short flat-contact points remaining to provide adequate support for the tube for all design conditions. The Model E steam generator tubes are 0.750" diameter, 0.043" thick mill annealed Alloy 600. (Note: fifteen tubes installed in Unit 2 steam generator "D" are thermally treated Alloy 600). The Δ94 steam generators tubes are 11/16" diameter, 0.040" thick thermally treated Alloy 690 tubes. The basis for the Unit 1 STP voltage-based repair criteria presumes the tubing diameter, tubing thickness, tubing material and tube-to-tube support plate intersection design used in the Model E steam generators. Generic Letter 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking," is specific to Westinghouse steam generators with Alloy 600 tubing material and drilled-hole tube support plates, and therefore does not apply to the Δ94 steam generators. Accordingly, the TS changes implemented by Amendment No. 83 to the Unit 1 Operating License, which changed the steam generator tube surveillance TSs to implement the voltage-based repair criteria for Unit 1, will not apply after steam generator replacement. Similarly, Amendment No. 83 to the Unit 2 Operating License which implemented a 1-volt voltage-based repair criteria for Unit 2 will not apply after steam generator replacement on Unit 2. Restoring the original TS repair limit for applicability to indications at tube-to-tube support plate intersections after installation of the Δ94 steam generators is therefore appropriate. Thus, the proposed revisions to TSs 4.4.5.2, 4.4.5.4, and 4.4.5.5, to make voltage-based repair applicable to Model E steam generators only are acceptable.

The basis for laser-welded sleeving at STP presumes the tubing diameter, thickness and material used in the Model E steam generators. Accordingly, the TS changes implemented by Amendment No. 90 to the Unit 1 Operating License and Amendment 77 to the Unit 2 Operating License, which changed the steam generator tube surveillance TSs to implement the Westinghouse laser-welded sleeving process for both Unit 1 and Unit 2, do not apply after installation of the Δ94 steam generators. The proposed changes that restore the original TS remedy for defective tubes after installation of the Δ94 steam generators and make laser-welded sleeving applicable only to Model E steam generators are acceptable.

The Unit 1 Model E steam generator tube-to-tubesheet joint was accomplished using a full-depth hard roll. The Δ94 steam generator tube-to-tubesheet joint is accomplished using full-depth hydraulic expansion. This design difference results in lower stresses in the replacement steam generator's expanded tubing and expansion transition which reduces the likelihood of stress corrosion cracking in this area; however, the basis for the F* alternate repair criteria presumes that the tube-to-tubesheet joint is a full-depth hard roll. Accordingly, the TS changes implemented by Amendment No. 82 to the Unit 1 Operating License will not apply after steam generator replacement. This proposed TS change will be implemented after replacement of the Unit 1 Model E steam generators with Δ94 steam generators. Therefore, the proposed changes to TSs 4.4.5.2 and 4.4.5.4 that restore the original TS repair limits for applicability to indications in the tubesheet area for Unit 1 after installation of the Δ94 steam generators are acceptable.

The plugging/repair limits for the Δ94 steam generators will consist solely of the through-wall criteria and will require that tubes be plugged (not sleeved) when imperfections exceed the plugging limit of 40% of the tube nominal wall thickness. This is the same as the Model E steam generator 40% plugging limit which is currently in the TS. Therefore, no TS change regarding this criteria is being requested. The licensee performed analyses in accordance with the ASME Code and Regulatory Guide 1.121 to substantiate the validity of the 40% plugging limit for Δ94 steam generators. The original 40% plugging limit should be applied to the Δ94 steam generators.

As part of the license amendment for the voltage-based repair criteria for Unit 1, and later as part of the license amendment for laser-welded sleeving for Unit 2, the primary-to-secondary leakage limit of TS 3.4.6.2.d was lowered from 500 gpd in any single steam generator (combined leakage limited to 1 gpm or 1440 gpd) to 150 gpd in any single steam generator. This change was made to provide increased margin (i.e., earlier shutdown) following identification of increasing primary-to-secondary leakage, reducing the likelihood that an occurrence of low level primary-to-secondary leakage might propagate to a tube rupture event. Although this change was made in association with the implementation of the license amendments for voltage-based repair criteria and laser-welded sleeving on existing Model E steam generators, no change (relaxation) to TS 3.4.6.2.d is requested in this submittal to account for the new Δ94 steam generators. The licensee proposes to retain the increased margin against tube rupture events.

Clarifications to the inspection frequency requirements of SR 4.4.5.3 are proposed. The existing wording of SR 4.4.5.3.a includes a specific one-time inservice inspection interval requirement which is only applicable to the first cycle at the beginning of unit life. Since steam generator replacement is analogous to initial unit start-up as related to the steam generator surveillance, it is appropriate to update the wording here to clearly indicate that the same initial interval requirement applies following steam generator replacement. Further, a note is added to clarify that no steam generator inservice inspection will be performed during the steam generator replacement outage. Inspecting the Model E steam generators as they are being removed from the plant would serve little purpose, since the ultimate corrective action for any degradation found is being implemented (i.e., replacement). Inservice inspection of the new Δ94 steam generators during the replacement outage is not possible, since they will not have been in service. The note will ensure that no confusion exists in the event that more than 24 calendar months pass between the last inservice inspection of the Model E steam generators and the first inservice inspection of the Δ94 steam generators.

Based on its review, the staff concludes that the proposed changes continue to meet the requirements for safe operation and are acceptable.

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 and change surveillance requirements. The Nuclear Regulatory Commission 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 (63 FR 59595). 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 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.

Principal Contributors: Marsha Gamberoni
Cheryl Beardslee

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