



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. 96
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 February 16, 1998, as supplemented by letters dated April 2, July 15, and August 13, 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.

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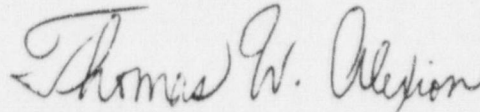
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C.(2) of Facility Operating License No. NPF-76 is hereby amended to read as follows:

2. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 96 , 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 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION



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: September 24, 1998



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. 83
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 February 16, 1998, as supplemented by letters dated April 2, July 15, and August 13, 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.

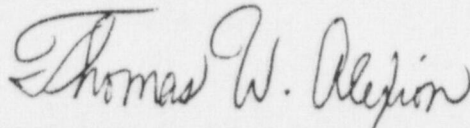
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C.(2) of Facility Operating License No. NPF-80 is hereby amended to read as follows:

2. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 83 , 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 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION



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: September 24, 1998

ATTACHMENT TO LICENSE AMENDMENT NOS. 96 AND 83

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. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE

3/4 4-13
3/4 4-15
3/4 4-16
3/4 4-16a
3/4 4-16b
B 3/4 4-3
B 3/4 4-4

INSERT

3/4 4-13
3/4 4-15
3/4 4-16
3/4 4-16a
3/4 4-16b
B 3/4 4-3
B 3/4 4-4

REACTOR COOLANT SYSTEM

STEAM GENERATORS

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.
- 4) 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 Unit 1, any tube allowed to remain in service per Acceptance Criterion 11 (of Technical Specification 4.4.5.4a) 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.
- e. 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.

<u>Category</u>	<u>Inspection Results</u>
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.

REACTOR COOLANT SYSTEM

STEAM GENERATORS

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.

REACTOR COOLANT SYSTEM

STEAM GENERATORS

SURVEILLANCE REQUIREMENTS (Continued)

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 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%

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.12 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; and

REACTOR COOLANT SYSTEM

STEAM GENERATORS

SURVEILLANCE REQUIREMENTS (Continued)

- 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) 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 O.
- 12) 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.12.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.

REACTOR COOLANT SYSTEM

STEAM GENERATORS

SURVEILLANCE REQUIREMENTS (Continued)

- d) Certain Unit 1 intersections as identified in Framatome Technologies, Inc. Topical Report BAW-10204P, "South Texas Project Tube Repair Criteria for ODSCC At Tube Support Plates" 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.
- e) 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.12.a, 4.4.5.4.a.12.b, and 4.4.5.4.a.12.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.12.a, 4.4.5.4.a.12.b, and 4.4.5.4.a.12.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.

- 13) Tube Repair refers to a process that reestablishes tube serviceability. 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 Feeding-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 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 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.

REACTOR COOLANT SYSTEM

BASES

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.

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.

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 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.

REACTOR COOLANT SYSTEM

BASES

STEAM GENERATORS (Continued)

The mid-cycle equation in SR 4.4.5.4.a.12.e should only be used during unplanned inspections in which eddy current data is acquired for indications at the tube support plates.

SR 4.4.5.5 implements several reporting requirements 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.

REACTOR COOLANT SYSTEM

BASES

OPERATIONAL LEAKAGE (Continued)

The leakage limits incorporated into SR 4.4.6 are more restrictive than the standard operating leakage limits and are 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 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