May 22, 1996

Mr. William T. Cottle
Group Vice-President, 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. 83 TO FACILITY OPERATING LICENSE NO. NPF-76 (TAC NO. M94535)

Dear Mr. Cottle:

The Commission has issued the enclosed Amendment No. 83 to Facility Operating License No. NPF-76 for the South Texas Project, Unit 1 (STP). The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated January 22, 1996, as supplemented April 4 and May 2, 1996.

The amendment modifies the steam generator tube plugging criteria in TS 3/4.4.5, Steam Generators, the allowable primary-to-secondary leakage in TS 3/4.4.6.2, Operational Leakage, and the associated Bases. These changes allow the implementation of alternate steam generator tube plugging criteria for the tube support plate/tube intersections 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 Janet Kennedy
for 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. 83 to NPF-76
2. Safety Evaluation

cc w/encls: See next page

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Document Name: STP94535.AMD

OFC	LA/PD4-1	PM/PD4-1	PM/PD4-1	OGC
NAME	PNoonan	JKennedy JK	TAlexion TA	
DATE	5/20/96	5/20/96	5/20/96	5/21/96
COPY	YES/NO	YES/NO	YES/NO	YES/NO

Signed 5/22/96 JLK

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DF01 1/1 CP-1

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

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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. 83
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 April 4 and May 2, 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.

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*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. 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 with full implementation within 10 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Janet Kennedy for

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 22, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 83

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 4-13a
3/4 4-15	3/4 4-15
3/4 4-16	3/4 4-16
---	3/4 4-16a
---	3/4 4-16b
3/4 4-20	3/4 4-20
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 4-4	B 3/4 4-4

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) 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.

e. For Unit 1, 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 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. For Unit 1, 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.

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.

(11) For Unit 1, Tube Support Plate Plugging Limit is used for the disposition of an 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 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) Certain 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.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 or 2.0 volts for 7/8-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.

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.

SURVEILLANCE REQUIREMENTS (Continued)

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 Unit 1, 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.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.6.1 The following Reactor Coolant System Leakage Detection Systems shall be OPERABLE:

a. The Containment Atmosphere Gaseous Radioactivity Monitoring System,

b. The Containment Normal Sump Level and Flow Monitoring System, and

c. The Containment Atmosphere Particulate Radioactivity Monitoring System.

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

ACTION:

a. With a. or c. of the above required Leakage Detection Systems inoperable, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed for gaseous and particulate radioactivity at least once per 24 hours when the required Gaseous or Particulate Radioactive Monitoring System is inoperable; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With b. of the above required Leakage Detection Systems inoperable, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With a. and c. of the above required Leakage Detection Systems inoperable:

1) Restore either Monitoring System (a. or c.) to OPERABLE status within 72 hours and

2) Obtain and analyze a grab sample of the containment atmosphere for gaseous and particulate radioactivity at least once per 24 hours, and

3) Perform a Reactor Coolant System water inventory balance at least once per 8 hours.

Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The Leakage Detection Systems shall be demonstrated OPERABLE by:

a. Containment Atmosphere Gaseous and Particulate Monitoring Systems performance of CHANNEL CHECK, CHANNEL CALIBRATION, and DIGITAL CHANNEL OPERATIONAL TEST at the frequencies specified in Table 4.3-3, and

b. Containment Normal Sump Level and Flow Monitoring System performance of CHANNEL CALIBRATION at least once per 18 months.

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 gpm UNIDENTIFIED LEAKAGE,

c. For Unit 1, 150 gallons per day of primary-to-secondary leakage through any one steam generator, and for Unit 2, 1 gpm total reactor-to-secondary leakage through all steam generators and 500 gallons per day through any one steam generator,

d. 10 gpm IDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. 0.5 gpm leakage per nominal inch of valve size up to a maximum of 5 gpm at a Reactor Coolant System pressure of 2235 ± 20 psig from any Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1.*

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

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE and leakage from Reactor Coolant System Pressure Isolation Valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed manual or deactivated automatic valves, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

*Test pressures less than 2235 psig but greater than 150 psig are allowed. Observed leakage shall be adjusted for the actual test pressure up to 2235 psig assuming the leakage to be directly proportional to pressure differential to the one-half power.

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.

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 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

BASES

STEAM GENERATORS (Continued)

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.

For Unit 1, 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 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.

STEAM GENERATORS (Continued)

The mid-cycle equation in SR 4.4.5.4.a.11.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 vale is sufficiently low to ensure early detection of additional leakage.

OPERATIONAL LEAKAGE (Continued)

For Unit 1, 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.

For Unit 2, 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 valves 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. 83

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 April 4 and May 2, 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 (STP). The letter dated May 2, 1996, provided clarifying information that did not change the initial proposed no significant hazards consideration determination nor did it expand the amendment request beyond its original scope.

The proposed amendment would revise Technical Specifications (TSs) 3/4.4.5 and 3/4.4.6.2 including associated Bases 3/4.4.5 and 3/4.4.6.2 to allow the implementation of steam generator (SG) voltage-based repair criteria for the tube support plate/tube intersections for STP Unit 1. The voltage-based steam generator tube repair criteria allows axially oriented outside diameter stress corrosion cracking (ODSCC) confined within the thickness of the tube support plates (TSPs) to remain in service based on the magnitude of the bobbin coil voltage response. The allowed primary-to-secondary operational leakage from any one steam generator will be reduced from 500 gallons per day (gpd) to 150 gpd. The licensee has stated that the proposed amendment request is consistent with the guidance provided in Generic Letter (GL) 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking [ODSCC]."

2.0 VOLTAGE-BASED STEAM GENERATOR TUBE REPAIR CRITERIA

2.1 Discussion

The NRC staff documented its generic position on voltage-based limits for ODSCC affecting the SG tubes at the TSP elevations in GL 95-05 and its supporting documentation. This approach takes no credit for the TSPs in preventing and/or reducing the likelihood of a tube from bursting and/or leaking during postulated accident conditions. In essence it assumes that the degradation affecting the SG tubes at the TSP elevation is in the tube free span.

The licensee's proposed amendment requests a permanent change to the TSs to incorporate the voltage-based repair criteria described in GL 95-05. The guidance contained in GL 95-05 specifies, in part, that: (1) the repair criteria is only applicable to predominantly axially oriented ODSCC located within the bounds of the TSPs; (2) licensees should perform an evaluation to confirm that the SG tubes will retain adequate structural and leakage integrity until the next scheduled inspection; (3) licensees should adhere to specific inspection criteria to ensure consistency in methods between inspections; (4) tubes must be periodically removed from the SGs to verify the morphology of the degradation and provide additional data for structural and leakage integrity evaluations; (5) the operational leakage limit should be reduced; and (6) specific reporting requirements shall be followed, some of which will be incorporated into the plant TSs.

The licensee's current proposal requests a permanent amendment to the TSs and, in general, incorporates the guidance of GL 95-05. Exceptions and alternatives to the methodology specified in GL 95-05 are described below.

2.2 Evaluation

The licensee has proposed to follow the requested actions of GL 95-05 for implementing the voltage-based plugging criteria. GL 95-05, however, permits licensees to implement various alternatives to the methodology specifically stated in the GL. For example, licensees can, subject to NRC approval, (1) choose to implement the voltage-based tube repair criteria at tube-to-flow distribution baffle plate intersections; (2) choose to implement an alternative to the probability of detection value of 0.6; (3) choose to include only a fraction (rather than all) of the bobbin indications which were not confirmed with a rotating pancake coil (RPC) probe in the determination of the beginning-of-cycle voltage distribution; (4) choose to implement an alternative to the main steam line break leak rate of 2496 liters per hour assigned to the V.C. Summer tube R28C41; (5) choose to implement an alternative to the probe wear criteria which requires all tubes since the last successful probe wear check to be reinspected with a new calibrated probe when a probe is found to be out of specification; (6) choose to use probe sizes different than the nominal probe size; and (7) choose to implement an industry alternative to the tube pull program specified in GL 95-05.

With respect to the items listed above, the licensee has elected not to implement the voltage-based tube repair criteria at the flow distribution baffle plate intersections; however, the licensee stated that if, in the future, they elect to apply the repair criteria to flow distribution baffle plate intersections, the technical bases will be submitted for NRC review and approval. Furthermore, the NRC has not approved the use of (1) alternatives to the probability of detection value of 0.6; (2) a generic alternative to include only a fraction of the bobbin indications which were not confirmed with an RPC probe in the determination of the beginning-of-cycle voltage distribution (note that an alternative was approved for Beaver Valley Unit 1; however, generic implementation was not approved); (3) an alternative to the 2496 liters per hour leak rate for V.C. Summer tube R28C41; (4) probe sizes different than the nominal probe size; and (5) an industry alternative to the tube pull program specified in GL 95-05. As a result, the licensee has proposed to use the methodology specified in the GL. The staff finds this acceptable and notes that if alternatives are approved by the staff on a generic basis they may be used by the licensee. The NRC has generically approved an alternative to the probe wear criteria specified in GL 95-05 by letter dated March 18, 1996, from Brian W. Sheron (NRC) to Mr. Alex Marion (Nuclear Energy Institute). As a result, the licensee can implement either this approved alternative or the methodology specified in GL 95-05. In addition, the staff's letter dated March 18, 1996, addressed a methodology for controlling new probe variability which the staff found acceptable for implementation.

In GL 95-05, the NRC indicated that licensees should (1) submit the methodology for calculating the conditional burst probability for NRC review and approval; (2) relate burst pressure to bobbin voltage using an empirical model (the currently approved model consists of determining a linear first-order equation between the burst pressure and the logarithm (base 10) of the bobbin voltage; this model may need to be changed as additional information is acquired; the alternative model is subject to NRC approval); (3) relate probability of leakage to the bobbin voltage using an empirical model (the currently approved model consists of using a log-logistic function to fit the data; this model may need to be changed as additional information is acquired; the alternative model is subject to NRC approval); (4) relate conditional leak rate to bobbin voltage using an empirical model (the currently approved model consists of determining a linear first-order equation between the logarithm (base 10) of the conditional leak rate and the logarithm (base 10) of the bobbin voltage; this model may need to be changed as additional information is acquired; the alternative model is subject to NRC approval); (5) not exclude data based on data exclusion criteria 3a, 3b, and 3c unless approved by the NRC; and (6) justify lowering the I-131 limits in the TSs to a value below 0.35 microcuries per gram (μCi/g), if applicable.

With respect to the items listed above, the licensee has submitted a methodology for calculating the conditional burst probability and the staff's review of this methodology is documented below. In addition, the licensee has submitted their methodology for calculating the end-of-cycle voltage distribution and the total leak rate during postulated accident conditions (e.g., main steam line break). A review of these methodologies is also documented below. With respect to the burst pressure, probability of leakage, and conditional leak rate correlations, the licensee has proposed to use, based on currently available data, a linear first-order equation to relate the burst pressure and the logarithm (base 10) of the bobbin voltage, a log-logistic function to fit the probability of leakage data; and a linear first-order equation to relate the logarithm (base 10) of the conditional leak rate and the logarithm (base 10) of the bobbin voltage, respectively. The staff finds this acceptable for the current databases; however, the adequacy of these correlations should be assessed as additional data is acquired and, if the model(s) require changing as a result of this additional information, the revised model(s) should be submitted for NRC review and approval per GL 95-05. With respect to the data exclusion criteria, the licensee did not exclude any data from the databases based on data exclusion criteria 3a, 3b, and 3c; and, as a result, the staff finds this acceptable. With respect to the I-131 limits in the TSs, the licensee did not request to lower the I-131 limits; therefore, no additional justification was necessary.

With respect to the methodologies for calculating the end-of-cycle voltage distribution, the probability of burst, and the total leak rate during postulated accident conditions, the staff has the following comments. The licensee has indicated that it will use the probabilistic methodology specified in WCAP-14277 for calculating the conditional probability of burst and the total leak rate during postulated accident conditions. The staff has reviewed these probabilistic methodologies which involves Monte Carlo simulations and has concluded that they are consistent with the methodology outlined in GL 95-05 and, therefore, are acceptable. Similarly, the methodology for calculating the end-of-cycle voltage distribution is consistent with the methodology outlined in GL 95-05 and, therefore, is acceptable. To provide additional assurance of the adequacy of these calculational methodologies, the staff notes that it may periodically verify the results of these calculations and assess the effectiveness of the methodologies as indicated in GL 95-05.

GL 95-05 also recommends that licensees use updated databases (e.g., burst pressure, probability of leakage, and conditional leak rate databases) in their tube integrity evaluations (e.g., calculation of tube repair limits, conditional burst probability, and total leakage under postulated accident conditions). The industry is currently working on a generic process for updating the applicable databases. Once developed, the staff will review the adequacy of this process. Comments have been supplied to the industry on this issue by a letter dated August 4, 1995, from Brian W. Sheron to Mr. Alex Marion. Pending completion of the development of the industry process for updating the applicable databases, the staff has reviewed the data supplied by the licensee and has found it to be acceptable. The staff notes that if the generic industry process for updating the databases is approved by the staff, this process would provide the mechanism for assuring NRC approval with the databases used by the licensee for application of this repair criteria in future outages.

2.3 Conclusion

The staff has previously evaluated the acceptability of the voltage-based tube repair criteria that the licensee is proposing as documented in GL 95-05. As a result, based on this and the above evaluation, the staff finds the licensee's proposal acceptable. Further technical details on the voltage-based tube repair criteria methodology are contained within GL 95-05 and its supporting technical documentation.

3.0 TUBE LOCATIONS BEING EXCLUDED FROM THE ALTERNATE REPAIR CRITERIA

3.1 Discussion

In accordance with GL 95-05, the alternate repair criteria (ARC) cannot be applied at TSP locations where tubes may collapse or deform following a postulated loss-of-coolant accident (LOCA) plus a safe shutdown earthquake (SSE) event. Specifically, the tube locations adjacent to wedge supports at the upper TSPs are of primary concern due to the potential yielding of the plate and subsequent deformation of the tubes during a main steam line break (MSLB). Consequently, an evaluation of the support plates in these regions has been performed. The staff has reviewed the licensee's submittal including Framatome Technologies, Inc. Topical Report BAW-10204P, "South Texas Project Tube Repair Criteria for ODSCC at Tube Support Plates." The licensee's submittal contained the analysis methodology and identification of tube locations excluded from the ARC at STP Unit 1.

3.2 Evaluation

The loads on the SG tubes at TSP intersections during a LOCA are primarily due to the cumulative effect of pressure waves initiated at the pipe break locations and the compressive differential pressure across the tubes. The resultant effect of the pressure waves is to cause in-plane bending loads on the SG tubes at the upper TSPs. In addition, the pipe break hydraulic forces cause a shaking of the reactor coolant system (RCS) as a whole, which further transmits inertial loads to the TSPs. A dynamic load factor is applied to the LOCA loads which are then probabilistically combined with the seismic loads via the square root-of-the-sum-of-the-squares (SRSS) method.

As specified in General Design Criteria (GDC) 4, dynamic effects of pipe ruptures in nuclear power plant units may be excluded from the design basis provided it is demonstrated that the probability of pipe rupture is extremely low under conditions consistent with the design of piping. Dynamic effects covered by GDC 4 include missile generation, pipe whipping, pipe break reaction forces, jet impingement forces, decompression waves within the ruptured pipe and dynamic pressurization in cavities and subcompartments. The NRC has concluded in References 2 and 3 that STP is in compliance with GDC 4. As such, the staff believes that the probability of a rupture of the primary reactor coolant piping and the surge line is extremely low. Hence, the dynamic effects of postulated pipe ruptures of the large primary piping and the surge line are eliminated from the design basis at STP. The design loadings for the analysis of the upper tube support plate at STP, was based on a 12-inch diameter, schedule 140, attachment line break.

LOCA loads for STP were based on loads from a similar replacement recirculating steam generator (RSG) analytical model for another plant. Some of the conservatisms in the analysis, according to the licensee, include the following:

- The loads were based on a larger attachment line break (14-inch diameter schedule 140 versus the actual STP 12-inch diameter schedule 140).

- A stress-strain curve based on the American Society of Mechanical Engineers (ASME) Code minimum yield and tensile strength properties for the support plate was used. It is considered likely that the actual material property values are greater than Code minimum allowables.

- The increase in yield strength due to the rapidly applied load was neglected.

- It was assumed that the tube deformation is equal to the TSP hole deformation in the finite element analysis even though a gap may exist at the intersection. Also, the TSP stiffness neglected any contribution provided by the tubing.

- It was assumed that the interface between the support plate and wedges is frictionless even though the wedges were snugly installed and are securely welded to wrapper support blocks.

- It was assumed that the entire LOCA pressure wave loading is acting at the top support plate only.

With regard to the item relating to the ASME Code minimum yield and tensile strength properties stated above, the staff does not consider its usage as a quantifiable conservatism and expects the licensee to normally use the remaining properties in the course of performing calculations. The licensee's contention of increase in yield strength due to rapidly-applied loads is also not considered to be a quantifiable conservatism, unless supported by analytic or test data for the specific application. The staff finds the remaining conservatisms in the analysis stated above reasonable and acceptable and the analytic model applicable to the STP replacement steam generators.

The ANSYS finite element computer code was utilized to generate an inelastic model of the steam generator and evaluate the loadings on the tubes in the vicinity of the wedge supports. The seismic loading evaluation was previously performed and the results are contained in Reference 7. The in-plane seismic loadings on the TSPs were determined on the basis of a time history analysis. In determining the combined effects of seismic and LOCA loads on the TSPs, each of the lower support plates was conservatively assumed to exhibit this worst case loading as well. However, since the wedge groups are vertically aligned, the number of tubes affected is small. The tubes located in the vicinity of the wedge supports undergo the maximum deformation. Tubes that are projected to deform greater than a certain critical magnitude under combined LOCA plus SSE loads were considered to be unacceptable. This acceptance criterion based on a critical magnitude of deformation relies on analysis and previous test data. It is similar to the acceptance criterion developed for similar SGs for other plants and it has been reviewed and found acceptable by the staff. The calculations performed in Reference 6 have identified all tubes exceeding this acceptance criterion. A summary of the excluded tubes on this basis is also provided in Reference 1. TSP locations and designations for identifying the elevation of the wedge group locations, as well as the circumferential locations of the wedge groups for TSPs have been identified. A submittal from another plant shows the limiting wedge group exclusion region to be approximately 30 tubes for a similar steam generator with slightly higher loading conditions from a large primary pipe break (Reference 8). The exclusion region for STP is nearly 1.5 times as large for the limiting wedge group. The staff finds the number of excluded tubes at STP to be reasonable and acceptable.

3.3 Conclusion

The staff has reviewed the licensee's analysis relative to the effects of accident loads at specific tube support plate locations where ARC cannot be applied. Based on this review, the staff finds that the licensee's analytical methodology and identification of tube locations excluded from the ARC are acceptable.

3.4 References

1. Report BAW-10204P Rev. 02, "South Texas Report Tube Repair Criteria at Tube Support Plates for ODSCC Framatome Technology, Inc," dated January 1996.

2. NUREG-0781, "Safety Evaluation Report Related to the Operation of South Texas Project, Units 1 & 2," Supplement No. 2, January 1987.

3. NUREG-0781, "Safety Evaluation Report Related to the Operation of South Texas Project, Units 1 & 2," Supplement No. 4, July 1987.

4. HL&P to USNRC Letter ST-HL-AE-3016, "Pressurizer Surge Line Thermal Stratification," March 14, 1989.

5. USNRC to HL&P Letter, NRC Bulletin 88-11, "Pressurizer Surge Line Thermal Stratification - South Texas Project, Units 1 and 2 (TAC No. 72168)," September 17, 1990.

6. BWNT Document 32-1236240, "Calculation for Wedge Deformation in W-E RSG's."

7. HL&P Document No. 120 (1) 00019-CWN, "Model E2 Steam Generator Stress Report," and addendum.

8. NRC Letter from George F. Dick, Jr. to D. L. Farrar "Issuance of Amendments (TAC Nos. M90052 and M90053)," dated October 24, 1994; Amendment No. 66, Docket No. STN 50-454 p. 15.

4.0 ASSESSMENT OF RADIOLOGICAL CONSEQUENCES

4.1 Discussion

The licensee performed an assessment of the radiological dose consequences of a main steam line break accident in support of its amendment request to apply a voltage-based repair limit for the STP Unit 1 steam generator TSP intersections experiencing outside diameter stress corrosion cracking. That assessment was based upon a primary to secondary leakage of 5.0 gpm initiated by a main steam line break accident and 0.42 gpm (600 gpd) allowed by TS 3.4.6.2. The licensee conservatively assumed that the 0.42 gpm leakage was divided into 0.147 gpm to the faulted steam generator and 0.273 to the intact steam generators. The licensee found the radiological dose consequences acceptable, assuming allowable activity levels in the primary coolant of 60 μCi/g dose equivalent I-131 for the pre-existing spike condition and 1.0 μCi/g dose equivalent I-131 for the accident-initiated spike condition.

4.2 Evaluation

The staff has independently calculated the doses resulting from a MSLB accident using the methodology in Standard Review Plan (SRP) 15.1.5, Appendix A. The staff performed two separate assessments. The first assessment was based upon a pre-existing iodine spike activity level of 60 μCi/g of dose equivalent I-131 in the primary coolant. The second assessment was based upon an accident-initiated iodine spike. Both assessments utilized dose conversion factors listed in Regulatory Guide 1.109 (1977) for the calculation of dose equivalent I-131 in the primary and secondary coolants, as required by the Unit 1 TSs.

For the accident-initiated spike assessment, the staff assumed that the accident initiated an increase in the release rate of iodine from the fuel by a factor of 500 over the release rate to maintain an activity level of 1 μCi/g of dose equivalent I-131 in the primary coolant. For each assessment, the staff calculated doses for individuals located at the Exclusion Area Boundary (EAB) and at the Low-Population Zone (LPZ). The control room operator's thyroid dose was also calculated. The parameters which were utilized in the staff's assessment are presented in Table 1. The radiological doses for each of the assessments are presented in Table 2.

Previous MSLB accident analyses for STP Unit 1 conservatively assumed additional coolant iodine activity as a result of potential fuel failures. However, the applicant's previous analysis showed no departure from nucleate boiling (DNB) occurring as a result of an MSLB. The staff has reviewed this analysis and agrees with the licensee's assessment. Although use of alternate plugging criteria may change the amount of allowable steam generator tube leakage during plant operation, this change will not affect the transient DNB value following a design basis MSLB. Because there is no fuel failure for this event, the staff did not consider this scenario when reviewing this license amendment request.

4.3 Conclusion

The staff's calculations, as shown in Table 2, show that the thyroid doses for the EAB and LPZ are within the acceptance criteria presented in SRP 15.1.5, Appendix A of NUREG-0800 for both the pre-existing spike and the accident-initiated spike cases. The control room operator thyroid doses are also within the acceptance criteria presented in SRP 6.4 of NUREG-0800. Since the calculated doses meet these acceptance criteria, the staff concludes that a leak rate of 5.15 gpm is an acceptable limit for the maximum primary to secondary leakage initiated by the MSLB accident.

5.0 PROPOSED CHANGES TO TS 3/4.4.5 AND 3/4.4.6.2 AND ASSOCIATED BASES

GL 95-05 provided model TS changes based on the NUREG-0452, Revision 4a, "Standard Technical Specifications (STS) for Westinghouse Pressurized Water Reactors." STP Unit 1 proposed the following TS changes:

- New surveillance requirement (SR) 4.4.5.2.b.4 requires future bobbin coil inspection of all tubes left in service as a result of the application of voltage-based repair criteria;

- New SR 4.4.5.2.e requires a 100 percent bobbin coil inspection for all hot leg tube support plate intersections and all cold leg intersections down to the lowest cold leg tube support plate with known ODSCC indications. The determination of the 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;

- Modify SR 4.4.5.4.a.6 to include an exception to the current plugging limits so that the definition does not apply to the region of the tube subject to the TSP intersections since the voltage-based repair criteria applies to this region;

- New SR 4.4.5.4.a.11 provides limitations applicable for the TSP voltage-based repair criteria limit;

- New SR 4.4.5.5.d addresses additional reporting criteria for those tubes where the TSP voltage-based repair criteria has been applied;

- Modify TS 3.4.6.2.c by changing the 1 gpm limit and by changing the 500 gpd limit for leakage through any one steam generator to 150 gpd for Unit 1 only;

- Modify TS Bases 3/4.4.5, Steam Generators, to reflect the reduction in Unit 1 daily steam generator leakage limits from 500 gpd to 150 gpd, delete "by radiation monitors of steam generator blowdown," and to add a reference to SRs for voltage-based repair criteria;

- Modify TS Bases 3/4.4.6.2, Operational Leakage, to address the new Unit 1 steam generator leakage limits.

The above TS and Bases changes meet the guidance provided by the staff in GL 95-05. The proposed TS and Bases changes modify the STP Unit 1 TS to reflect the use of voltage-based repair criteria for steam generator tubes affected by ODSCC. The staff has reviewed the above TS and Bases changes and finds them acceptable.

6.0 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.

7.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (61 FR 16651 and 61 FR 17735). 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.

8.0 CONCLUSION

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

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Date: May 22, 1996

TABLE 1

INPUT PARAMETERS FOR SOUTH TEXAS PROJECT UNIT 1 EVALUATION OF A MAIN STEAMLINE BREAK ACCIDENT

1. Primary coolant concentration of 60 μCi/g of dose equivalent ^{131}I.

   Pre-existing Spike Value (μCi/g)

^{131}I	=	45.14
^{132}I	=	52.67
^{133}I	=	71.48
^{134}I	=	10.72
^{135}I	=	39.50

2. Volume of primary coolant and secondary coolant.

Primary Coolant Volume	13,103 ft^3
Primary Coolant Temperature	592.0 °F
Secondary Coolant Steam Volume	5,245 ft^3
Secondary Coolant Liquid Volume	2,742 ft^3
Secondary Coolant Steam Temperature	556.0 °F
Secondary Coolant Feedwater Temperature	440.0 °F

3. TS limits for DE ^{131}I in the primary and secondary coolant:

Primary Coolant DE ^{131}I concentration (μCi/g)	1.0
Secondary Coolant DE ^{131}I concentration (μCi/g)	0.1

4. TS value for the primary to secondary leak rate:

Primary to secondary leak rate, any one SG	150 gpd
Primary to secondary leak rate, total all SGs	600 gpd

5. Maximum primary to secondary leak rate to the faulted and intact Sgs assumed for the MSLB analysis:

Faulted SG (gpm)	5.15
Intact Sgs (gpm)	0.27

6. Iodine Partition Factor

Faulted SG	1.0
Intact SG	0.01
Primary to Secondary Leakage	1.0

TABLE 1

**INPUT PARAMETERS FOR SOUTH TEXAS PROJECT UNIT 1
EVALUATION OF A MAIN STEAMLINE BREAK ACCIDENT**
(continued)

7. Steam Released to the Environment

Faulted SG (0 - 2 hours)	5.15 gpm
Faulted SG (2 - 8 hours)	5.15 gpm
Intact Sgs (0 - 2 hours)	484,000 lbs plus primary to secondary leakage
Intact Sgs (2 - 8 hours)	1,106,000 lbs plus primary to secondary leakage

8. Release Rate for 1.0 μCi/g of Dose Equivalent ^{131}I

		Ci/hr
^{131}I	=	12.4
^{132}I	=	80.9
^{133}I	=	28.7
^{134}I	=	38.7
^{135}I	=	28.0

9. Letdown Flow Rate (gpm) = 100

10. Atmospheric Dispersion Factors (sec/m^3)

EAB (0-2 hrs)	1.40×10^{-4}
LPZ (0-8 hrs)	1.90×10^{-5}
Control Room (0-8 hrs)	1.70×10^{-2}*

11. Control Room Parameters

Filter Efficiency (%)	
• makeup filter	95 (elemental I)
• recirculation filter	95 (elemental I)
Volume (ft^3)	280,000
Makeup flow (cfm)	1800**
Recirculation Flow (cfm)	9000
Unfiltered Inleakage (cfm)	10
Occupancy Factor (0-8 hrs)	1

* Calculation based on Murphy-Campe methodology assuming a point source and point receptor. This estimate is conservative because it does not take into account enhanced atmospheric dispersion caused by the high release pressure and temperature of the effluent.

** 235 cfm of this flow does not pass through the control room recirculation filter units.

TABLE 2

CALCULATED THYROID DOSES FOR SOUTH TEXAS PROJECT UNIT 1 MAIN STEAMLINE BREAK ACCIDENT

	DOSE (rem)	
LOCATION	Pre-Existing Spike	Accident-Initiated Spike
EAB	4.9*	3.1**
LPZ	2.6*	6.4**
Control Room**	10.5	24.4

* NUREG-0800 Acceptance Criterion = 300 rem thyroid

** NUREG-0800 Acceptance Criterion = 30 rem thyroid