September 4, 1997

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

SUBJECT: SOUTH TEXAS PROJECT, UNITS 1 AND 2 - AMENDMENT NOS. 90 AND 77 TO FACILITY OPERATING LICENSE NOS. NPF-76 AND NPF-80 (TAC NOS. M95401 AND M95402)

Dear Mr. Cottle:

The Commission has issued the enclosed Amendment Nos. 90 and 77 to Facility Operating License Nos. NPF-76 and NPF-80 for the South Texas Project, Units 1 and 2. The amendments consist of changes to the Technical Specifications in response to your application dated May 17, 1996, as supplemented June 14, 1996, March 17, July 29, and July 30, 1997.

The amendments modify Technical Specification Section 3/4.4.5 Steam Generators, 3/4.4.6 Reactor Coolant System Leakage, and associated Bases to allow installation of tube sleeves as an alternative to plugging to repair defective steam generator tubes.

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,
Orig. signed by
Janet L. Kennedy, Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

DF01

Docket Nos. 50-498 and 50-499

Enclosures:
1. Amendment No. 90 to NPF-76
2. Amendment No. 77 to NPF-80
3. Safety Evaluation

cc w/encls: See next page

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

September 4, 1997

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Mr. William T. Cottle
Houston Lighting & Power Company

South Texas, Units 1 & 2

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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. 90
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 May 17, 1996, as supplemented June 14, 1996, March 17, July 29, and July 30, 1997, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

---

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

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

   2. <u>Technical Specifications</u>

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 90, 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.

FOR THE NUCLEAR REGULATORY COMMISSION

Janet L. Kennedy

Janet L. Kennedy, 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 4, 1997

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. 50-499

SOUTH TEXAS PROJECT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 77
License No. NPF-80

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 May 17, 1996, as supplemented June 14, 1996, March 17, July 29, and July 30, 1997, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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

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

   2. Technical Specifications

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 77, 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.

FOR THE NUCLEAR REGULATORY COMMISSION

Janet L. Kennedy

Janet L. Kennedy, 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 4, 1997

ATTACHMENT TO LICENSE AMENDMENT NOS. 90 AND 77

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	INSERT
3/4 4-12	3/4 4-12
3/4 4-13	3/4 4-13
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 4-18	3/4 4-18
-	3/4 4-18a
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 4-3a
B 3/4 4-4	B 3/4 4-4
B 3/4 4-5	B 3/4 4-5

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 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

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. 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.

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%

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.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

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 0.

12) 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.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 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.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 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.

13) <u>Tube Repair</u> 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 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 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 <u>Reports</u>

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;

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 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.

TABLE 4.4-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	No			Yes		
No. of Steam Generators per Unit	Two	Three	Four	Two	Three	Four
First Inservice Inspection	All			One	Two	Two
Second & Subsequent Inservice Inspections	One[1]			One[1]	One[2]	One[3]

TABLE NOTATIONS

1. The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

2. The other steam generator not inspected during the first inservice inspection shall be inspected. The third and subsequent inspections should follow the instructions described in 1 above.

3. Each of the other two steam generators not inspected during the first inservice inspections shall be inspected during the second and third inspections. The fourth and subsequent inspections shall follow the instructions described in 1 above.

Table 4.4-2

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.	C-1	None	N.A.	N.A.	N.A.	N.A.
	C-2	Plug or repair defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N.A.	N.A.
			C-2	Plug or repair defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug or repair defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N.A.	N.A.
	C-3	Inspect all tubes in this S.G., plug or repair defective tubes and inspect 2S 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	N.A.	N.A.
			Some S.G.s C-2 but no additional S.G. are C-3	Perform action for C-2 result of second sample	N.A.	N.A.
			Additional S.G. is C-3	Inspect all tubes in each S.G. and plug or repair defective tubes. Notification to NRC pursuant to 50.72 (b)(2) of 10 CFR Part 50	N.A.	N.A.

$S=3\frac{N}{n}\%$ where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection.

Table 4.4-3

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.

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 Instrumentation shall be OPERABLE:

a. One Containment Atmosphere Radioactivity Monitor (gaseous or particulate), and

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

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

ACTION:

a. With the required containment atmosphere radioactivity monitor inoperable perform the following actions or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours:

   1) Restore one containment atmosphere monitoring system to OPERABLE status within 30 days and,

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

   3) Perform a Reactor Coolant System water inventory balance at least once per 24 hours.

b. With the required containment normal sump level and flow monitoring system inoperable perform the following actions or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours:

   1) Restore the containment normal sump and flow monitoring system to OPERABLE status within 30 days and,

   2) Perform a Reactor Coolant System water inventory balance at least once per 24 hours.

c. With both a. and b. inoperable, enter 3.0.3.

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. 150 gallons per day of primary-to-secondary leakage 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 or repaired. Defective tubes 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 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 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.

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.

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.

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 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 Units 1 and 2, 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 valves 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

CHEMISTRY (Continued)

the chemistry within the Steady-State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride, and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with containment concentration levels in excess of the Steady-State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady-State Limits.

The Surveillance Requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the reactor coolant ensure that the resulting 2-hour doses at the SITE BOUNDARY will not exceed an appropriately small fraction of 10 CFR Part 100 dose guideline values following a steam generator tube rupture accident in conjunction with an assumed steady-state reactor-to-secondary steam generator leakage rate of 150 gpd per steam generator. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the STPEGS site, such as SITE BOUNDARY location and meteorological conditions, were not considered inthis evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the reactor coolant's specific activity greater than 1 microCurie/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER.

The sample analysis for determining the gross specific activity and $\bar{E}$ can exclude the radioiodines because of the low reactor coolant limit of 1 microCurie/gram DOSE EQUIVALENT I-131, and because, if the limit is exceeded, the radioiodine level is to be determined every 4 hours. If the gross specific activity level and radioiodine level in the reactor coolant were at their limits, the radioiodine contribution would be approximately 1%. In a release of reactor coolant with a typical mixture of radioactivity, the actual radioiodine contribution would probably be about 20%. The exclusion of radionuclides with half-lives less than 15 minutes from these determinations has

SPECIFIC ACTIVITY (Continued)

been made for several reasons. The first consideration is the difficulty to identify short-lived radionuclides in a sample that requires a significant time to collect, transport, and analyze. The second consideration is the predictable delay time between the postulated release of radioactivity from the reactor coolant to its release to the environment and transport to the SITE BOUNDARY, which is relatable to at least 30 minutes decay time. The choice of 15 minutes for the half-life cutoff was made because of the nuclear characteristics of the typical reactor coolant radioactivity. The radionuclides in the typical reactor coolant have half-lives of less than 4 minutes or half-lives of greater than 14 minutes, which allows a distinction between the radionuclides above and below a half-life of 15 minutes. For these reasons the radionuclides that are excluded from consideration are expected to decay to very low levels before they could be transported from the reactor coolant to the SITE BOUNDARY under any accident condition.

Based upon the above considerations for excluding certain radionuclides from the sample analysis, the allowable time of 2 hours between sample taking and completing the initial analysis is based upon a typical time necessary to perform the sampling, transport the sample, and perform the analysis of about 90 minutes. After 90 minutes, the gross count should be made in a reproducible geometry of sample and counter having reproducible beta or gamma self-shielding properties. The counter should be reset to a reproducible efficiency versus energy. It is not necessary to identify specific nuclides. The radiochemical determination of nuclides should be based on multiple counting of the sample within typical counting basis following sampling of less than 1 hour, about 2 hours, about 1 day, about 1 week, and about 1 month.

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the reactor coolant is below the lift pressure of the atmospheric steam relief valves. The Surveillance Requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G:

1. The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.4-2 and 3.4-3 for the service period specified thereon:

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 90 AND 77 TO

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

HOUSTON LIGHTING & POWER COMPANY

CITY PUBLIC SERVICE BOARD OF SAN ANTONIO

CENTRAL POWER AND LIGHT COMPANY

CITY OF AUSTIN, TEXAS

DOCKET NOS. 50-498 AND 50-499

SOUTH TEXAS PROJECT, UNITS 1 AND 2

1.0 INTRODUCTION

By application dated May 17, 1996, as supplemented by letters dated June 14, 1996, March 17, July 29 and July 30, 1997, the Houston Lighting & Power Company, et al. (the licensee) requested changes to the Technical Specifications (Appendix A to Facility Operating License Nos. NPF-76 and NPF-80) for the South Texas Project, Units 1 and 2 (STP-1 and -2). The proposed changes would revise Technical Specification (TS) Section 3/4.4.5 Steam Generators, 3/4.4.6 Reactor Coolant System Leakage, and associated Bases by allowing the use of Westinghouse (W) designed laser welded sleeves to repair defective steam generator (SG) tubes. The June 14, 1996 supplemental submittal provided the plant-specific W reports for the specific application of laser welded SG tube sleeving at STP discussed in the original submittal. The March 17, 1997 supplemental submittal provided corrected TS pages that include references to the topical sleeve reports and sleeve inservice inspection requirements. The July 29 and July 30, 1997 submittals provided responses to a Request for Additional Information from the staff dated June 30, 1997. The supplemental submittals did not affect the staff's initial no significant hazards consideration determination.

The licensee submitted WCAP-13698, Revision 2, "Laser Welded Sleeves for 3/4 Inch Diameter Tube Feedring-Type and Westinghouse Preheater Steam Generators," dated April 1995 (proprietary), which provides the technical basis for licensing the use of W designed laser welded sleeves to repair degraded SG tubes. This report summarizes the generic design, structural, and thermal-hydraulic analyses for three distinct types of sleeves. It includes a discussion of the supporting mechanical, leakage and corrosion test results and describes the sleeve installation processes and sleeve inspection methodology. Revision 2 of WCAP-13698 addresses laser welded sleeves for use in Combustion Engineering (CE) feedring-type SGs and for W Models D3, D4, D5, E1 and E2 preheater-type SGs, all of which use 3/4 inch outside diameter SG tubing. The licensee also submitted WCAP-14653, "Specific Application of Laser Welded Sleeves for the South Texas Power Plant Steam Generators," dated June 1996 (proprietary), which provides the technical bases supporting the licensing of W designed laser welded sleeves as described in Revision 2 of WCAP-13698 for use at STP-1 and -2.

The staff previously reviewed identical and closely similar W documents supporting requests for changes to the TS at other plants. The bulk of the technical and regulatory issues for the present request are identical to those reviewed in previous Safety Evaluations (SEs) concerning the use of W laser welded sleeves. Details of prior staff evaluations of W sleeves may be found in the SEs for Byron and Braidwood Nuclear Power Stations, Units 1 and 2, Docket Nos. 50-454, -455, -456 and -457, dated March 8, 1994; Maine Yankee Nuclear Power Plant, Docket No. 50-309, dated May 22, 1995; and Calvert Cliffs Nuclear Power Plant, Units 1 and 2, Docket Nos. 50-317 and 50-318, dated March 22, 1996. These evaluations apply to the proposed STP license amendment.

This SE discusses only those issues warranting revision, amplification, or inclusion based upon current experience. A summary of the principal technical issues regarding the design and use of W laser welded sleeves follows.

2.0 BACKGROUND

A sleeve is a tube slightly smaller in diameter than a SG tube that is inserted into a SG tube to bridge a degraded or susceptible section. The length of a sleeve is selected according to the individual installation circumstance. Generally, they vary in length between one and three feet. The sleeve becomes the pressure boundary and thereby restores the structural integrity of a degraded or potentially degraded portion of the original SG tube.

Prior to the development of sleeve technology, licensees removed defective SG tubes from service by plugging. However, this reduced the heat transfer area. The reduction in heat transfer (or other thermal-hydraulic operating parameters) can be tolerated up to a point before other system consequences of the reduced SG performance become limiting. Beyond this limit, a utility had to make operational changes resulting in reduced electrical generating capacity of the affected unit.

Because sleeves have minimal effect upon the thermal-hydraulics of a SG, their use is essentially unrestricted. This means a licensee may restore degraded sections of SG tubes to like new condition without experiencing a serious penalty with regard to unit generating capacity. This has led to increased use of sleeves versus plugs where practical. Recently, some foreign and domestic plants have installed sleeves in previously unprecedented numbers, up to nearly 100 percent of the SG tubes on a single unit.

The licensee's proposal addressed the use of three basic sleeve designs: a full length tubesheet sleeve (FLTS), an elevated tubesheet sleeve (ETS) and a tube support sleeve (TSS). The FLTS spans from the end of the tube, at the bottom surface of the tubesheet, to a point above the secondary side surface of the tubesheet. The ETS spans from a location within the tubesheet, approximately 14 inches above the tube end, to a point above the secondary side surface of the tubesheet. The TSS is installed centered approximately on a tube support intersection or in a freespan section of SG tube. All sleeve types are first secured by hydraulically expanding the upper and lower portions of the sleeve. The hydraulic expansion brings the sleeve ends into contact with the parent tube in preparation for subsequent welding or rolling. The FLTS and the ETS are installed by means of two different joint types: an autogenous laser weld at the freespan end of the sleeve (the upper joint) and a rolled joint (mechanically expanded) at the tubesheet end of the sleeve (the lower joint). The TSS is laser welded to the SG tube at each freespan end of the sleeve. The material of construction for the sleeve is a nickel-iron-chromium alloy, alloy 690, a Code approved material (ASME SB-163), incorporated in ASME Code Case N-20.

3.0 SUMMARY OF PREVIOUS REVIEWS

Previous staff evaluations of W laser welded sleeves addressed the technical adequacy of the sleeves in the principal areas of pressure retaining component design: structural requirements, material of construction, welding and post weld heat treatment, and sleeve plugging limits. The staff found the analyses and tests submitted to address these areas of component design to be acceptable as summarized below.

3.1 Structural Requirements

The sleeves function to restore the structural integrity of the tube pressure boundary. Consequently, W performed structural analyses for a variety of loadings including design pressure, operating transients, and other parameters selected to envelope loads imposed during normal operating, upset, and accident conditions at STP-1 and -2. The stress analyses of sleeved tube assemblies documented in Revision 2 of WCAP-13698 were performed in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III. W cites these analyses, along with the results of qualification testing and previous plant operating experience, to demonstrate the capability of the sleeved tube assembly of restoring SG tube structural integrity.

3.2 Material of Construction

The sleeves are fabricated from thermally treated alloy 690, a Code approved material (ASME SB-163) covered by ASME Code Case N-20. The staff found the use of alloy 690 is an improvement over the alloy 600 material used in the original SG tubing. Corrosion tests conducted under Electric Power Research Institute (EPRI) sponsorship confirmed test results regarding the improved corrosion resistance of alloy 690 over that of alloy 600. Accelerated stress corrosion tests in caustic and aqueous chloride solutions also indicated alloy

690 resists general corrosion in aggressive environments. Isothermal tests in high purity water have shown that, at normal stress levels, alloy 690 has high resistance to intergranular stress corrosion cracking (IGSCC) in extended high temperature exposure. The NRC concluded, as a result of these laboratory corrosion tests, that alloy 690 is acceptable for use in nuclear power plants. The NRC endorsed the use of Code Case N-20 in Regulatory Guide (RG) 1.85, "Materials Code Case Acceptability, ASME Section III, Division 1." The NRC staff has approved use of alloy 690 tubing in replacement SGs as well as sleeving applications.

3.3 Welding and Post Weld Heat Treatment

W employs automatic autogenous laser welding to join the sleeve to the parent tube in the freespan regions. W specifically qualified and demonstrated the application of this process to the sleeve design during laboratory tests employing full scale sleeve/tube mockups. Qualification of the welding procedures and welding equipment operators was performed in accordance with the requirements of the ASME Code, Section IX.

Accelerated corrosion tests confirm a post weld heat treatment (PWHT) significantly improves the IGSCC resistance of the alloy 600 parent tube material in the weld zone. A PWHT reduces the residual stresses resulting from welding. Residual stresses from forming operations (such as bending, welding, etc.) are known to be a principal contributor to IGSCC in alloy 600. Performance of a PWHT greatly reduces the residual stresses from welding thereby enhancing the IGSCC resistance of the alloy 600 portion of the weld zone. (The alloy 690 sleeve material is highly resistant to IGSCC either with or without PWHT.) In its July 29, 1997 submittal, the licensee committed to performing PWHT of the laser welded joint in accordance with the W topical sleeve reports. This commitment is reflected in the proposed TSs.

The rolled joint used to join the sleeve to the tube within the tubesheet effectively isolates the alloy 600 of the parent tube from the environment and thus is not susceptible to IGSCC. Stress relief of these joints is unwarranted.

3.4 Sleeve Plugging Limits

The licensee determined the sleeve minimum acceptable wall thickness using the criteria of RG 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," and ASME Code Section III allowable stress values and pressure stress equations. According to RG 1.121 criteria, an allowance for nondestructive evaluation (NDE) uncertainty and postulated operational growth of tube wall degradation within the sleeve must be accounted for when using NDE to determine sleeve plugging limits. Therefore, the licensee assumed a conservative tube wall combined allowance for postulated degradation growth and eddy current uncertainty of 20% through wall per cycle for the purpose of determining the sleeve plugging limit. The sleeve plugging limit calculated for STP-1 and -2 used the most limiting of normal, upset, or faulted conditions to determine a plugging criterion of 62% of the sleeve nominal wall thickness. The licensee proposes to use a value of 40% of the sleeve nominal wall thickness as the TS plugging limit. Removal of tubes and/or sleeves from service when degradation reaches the plugging limit provides assurance the minimum acceptable wall thickness will not be violated during the next subsequent cycle of operation.

4.0 DISCUSSION

Experience with all types of SG tube sleeves has led to several areas of concern outside the scope of basic sleeve design and qualification discussed above. These include instances of cracking in sleeved SG tubes, service life predictions for sleeved SG tubes, application of PWHT and the effect of tube lockup, nondestructive examination of sleeves, and primary-to-secondary leakage limits.

4.1 Cracking in Sleeved SG Tubes

Recent experiences at two U.S. plants indicate the alloy 600 SG tube may be susceptible to IGSCC at the sleeve freespan joint of a tubesheet sleeve. The affected joints are of the mechanically expanded type. These employ a hydraulic expansion followed by a hard roll in the center of the hydraulically expanded region. The hard roll forms the structural joint and leak limiting seal. Inner-diameter initiated cracks have been detected in the alloy 600 parent tube material at the lower hard roll transition and lower hydraulic transition of the freespan joints. The cracks were detected after 4 to 7 years of service. Since a number of sleeved tubes with this joint type have operated up to 14 years in one of the affected units with no such degradation, it is clear that not all such sleeved tubes are likely to develop cracks after a given service interval. Accelerated corrosion tests of laser welded sleeve joints have shown the hydraulic transition to have little or no susceptibility to IGSCC. Service times exceeding 8 years have been achieved for sleeved tubes with laser welded joints at U.S. plants. The staff is monitoring these developments for potential impact on welded sleeve installations.

4.2 Service Life Predictions for Sleeved SG Tubes

The staff considers the sleeving methods unable to assure an unlimited service life for a repaired tube. The conservative view is sleeving creates new locations in the parent tube which may be susceptible to IGSCC after new incubation times are expended. Incubation times are not quantified. They are observed to vary between individual steam generators and the various tubes within, based upon prior experiences with U-bend and roll transition cracking.

This staff conclusion that sleeving has limited service life is due to the circumstances of the sleeving processes. Sleeve installation methods can enhance one or two of the conditions necessary for IGSCC. The primary contributor is the residual stress resulting from the various joining methods. Secondarily, the local environment of the tube may be altered as a result of the formation of a wetted crevice between the tube and sleeve. Remediation of these contributors would benefit sleeved tube life. Of the two, stress relieving may be the most beneficial given the underlying causes of IGSCC and present sleeve designs. As discussed earlier, the sleeve installation procedure includes a PWHT of the weld joints to increase the resistance to IGSCC.

4.3 PWHT and Tube Lockup

Recent field experience with the installation of welded sleeves with PWHT indicated SG tubes may be constrained ("tube lockup") in their tube support plates. The result of such tube locking is distortion of the tube (bowing or bulging) during the PWHT. After the heat treatment is completed, the bow or bulge remains. Measurements of the bowing and bulging have shown them to be of negligible values. These distortions have been analyzed and found to be immaterial to the examination, operation, and safety of the sleeved tubes.

Along with the observed distortion (bowing or bulging) is a residual stress remaining after the heat treatment is completed. Strain gage measurements of this residual stress have shown it to be moderate compared to that resulting from welding without subsequent PWHT. This issue was the subject of additional testing and analysis related to the use of laser welded sleeves at the Maine Yankee facility during a sleeve installation project. Based upon the finding that many tubes are fixed in the tube support plates, W modified their sleeve installation procedure on the assumption that all tubes are locked. The modified installation procedure, described in WCAP-14653, minimizes the residual stress of PWHT regardless of tube condition.

4.4 Nondestructive Examination of Sleeves

The licensee proposes using ultrasonic testing (UT) and eddy current testing (ECT) as part of the nondestructive examination (NDE) of sleeved tubes prior to service. UT is performed after welding to confirm the laser welds are consistent with critical process dimensions and are of acceptable weld quality. W presented data on a UT system demonstrating post weld examinations of the sleeve/tube assembly will be adequate. Standards which included undersized welds were used in the qualification of the UT technique. The results of the qualification tests demonstrate the system can confirm there is a continuous metallurgical bond between the sleeve and tube and that the weld size (width) meets minimum acceptable dimensions.

ECT is then used to demonstrate presence of upper and lower hydraulic expansions, demonstrate presence of lower rolled joint, verify proper location of weld, verify post weld heat treatment, verify lack of process anomalies such as blow holes or weld cracking, and establish the baseline inspection data for future inspections. In performing ECT inspections, the licensee follows the Electric Power Research Institute's "PWR Steam Generator Tube Examination Guidelines," in that Appendix G qualified personnel and Appendix H qualified ECT techniques will be used. For the pre-service examination of the sleeve welds, South Texas will use the Plus Point coil in the eddy current probe to verify lack of weld process anomalies. The staff finds these examination methods adequate to monitor SG tube integrity.

For future sleeve inservice inspections, the licensee will be following the most current revision of the EPRI guidelines in terms of inspection scope and expansion criteria as well as personnel and technique qualifications. The licensee proposes modifying the TS to incorporate sleeve/tube inspection scope and expansion criteria that meet staff expectations. The staff finds this acceptable.

4.5 Primary-to-Secondary Leakage Limits

While a laser weld should be inherently leak-tight, the lower (rolled) joint of a tubesheet sleeve may not be leak tight. The topical sleeve reports describe leakage test results and plant-specific confirmatory test results that demonstrate adequate leakage integrity of the lower rolled joints. These results apply directly to STP-1 which has SG tubes hard rolled within the tubesheet. A confirmatory test program will be performed at STP-2 prior to sleeve installation to verify the applicability of these test results to a plant with SG tubes hydraulically expanded within the tubesheet, as described in the licensee's July 30, 1997 submittal.

Degraded tubes restored to operation as a result of sleeving are susceptible to additional degradation in the same SG environment. The sleeve is designed to extend past the upper weld joint and into the tubing. In the event a sleeved tube failed at the weld, the sleeve extension restricts tube movement and leakage. Leakage monitoring devices alert plant personnel to implement the appropriate procedures. However, based on experience with various causes of leakage through SG tubes including experience related to tubes repaired by sleeving, the staff has required licensees amend the TS to reflect a primary-to-secondary leakage limit of 150 gallons per day (gpd). With respect to the staff position regarding primary-to-secondary leakage limits for sleeving amendments, the licensee already implemented a change to the TS adopting a 150 gpd per SG leakage limit for STP-1. As part of this TS request, the licensee plans to adopt a 150 gpd per SG leakage limit for STP-2. The staff finds this acceptable.

5.0 CHANGES TO THE TECHNICAL SPECIFICATIONS

The licensee's proposal would revise TS Sections 3/4.4.5 and 3/4.6.2 as follows:

TS 4.4.5.2 would be revised to change the inservice inspection sample size for repaired SG tubes in all steam generators to 20% and to reflect the addition of TS Table 4.4-3 which will be used to classify the inspection results for repaired SG tubes.

TS 4.4.5.4 would be revised to add a definition of tubing or tube and modify the definition of plugging limit or repair limit by specifying plugging or repair limit imperfection depths specified in percentage of the nominal wall thickness. This change would be added to specify that any Westinghouse laser welded sleeve which upon inspection exhibits imperfection exceeding 40% of nominal wall thickness, it must be plugged prior to returning the SG to service.

TS 4.4.5.4, Note 2 would be revised to add tube repair (sleeving) in accordance with 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 the South Texas Project Power Plant Steam Generators," June 1996, including the post weld heat treatment, as acceptable methods for SG tube repair.

TS Table 4.4-2 would be revised to specify its use for nonrepaired SG tubes only.

TS Table 4.4-3 would be added to specify the inspection scope expansion criteria for repaired SG tubes.

TS 3.4.6.2 would be revised to reflect a primary-to-secondary leakage limit of 150 gpd through any one SG.

The TS Bases and the Table of Contents would also be revised consistent with the changes described above.

The staff reviewed the TS changes discussed above and found they consistently incorporate the W laser welded sleeving methodology discussed in this SE and will provide adequate assurance of SG tube integrity. Therefore, the proposed changes are acceptable.

6.0 SUMMARY

The NRC staff concludes the proposed sleeving repair, as described in the W topical sleeve reports WCAP-13698 and WCAP-14653, will produce sleeved tubes with acceptable metallurgical properties, structural and leakage integrity and corrosion resistance. The NRC staff also finds acceptable the proposed preservice and future inspections of the sleeved SG tubes.

7.0 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.

8.0 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 a surveillance requirement. 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 proposed findings that the amendments involve no significant hazards consideration, and there has been no public comment on such findings (61 FR 26938 and 62 FR 17235). 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.

9.0 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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