The Light company
Houston Lighting & Power South Texas Project Electric Generating Station P. O. Box 289 Wadsworth, Texas 77483

January 22, 1996
ST-HL-AE-5269
File No.: G20.02.01
10CFR50.90,
10CFR50.92,
10CFR51

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

South Texas Project
Units 1
Docket No. STN 50-498
Unit 1 Technical Specifications 3.4.5 and 3.4.6.2

The South Texas Project (STP) proposes to amend its Operating License NPF-76, Unit 1, by incorporating the attached proposed change to Technical Specifications 3.4.5 and 3.4.6.2. This amendment is consistent with the guidance provided in NRC Generic Letter 95-05, "Voltage-Based Repair Criteria For Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking" for Alternate Plugging Criteria (APC). The purpose of this amendment is to modify the Steam Generator tube plugging criteria in Technical Specification 3.4.5, Steam Generators and the allowable leakage for Unit 1 in Technical Specification 3.4.6.2, Operational Leakage and the associated Bases. These changes will allow the implementation of alternate steam generator tube plugging criteria for the tube support plate/tube intersections for Unit 1.

STP has reviewed the attached proposed amendment pursuant to 10CFR50.92 and determined that it does not involve a significant hazards consideration. In addition, STP has determined that the proposed amendment satisfies the criteria of 10CFR51.22(c)(9) for categorical exclusion from the requirement for an environmental assessment. The STP Nuclear Safety Review Board has reviewed and approved the proposed changes.

The required affidavit, along with a Safety Evaluation and No Significant Hazards Consideration Determination associated with the proposed changes, and the marked up affected pages of the Technical Specifications are included as attachments to this letter.

As Attachment 4 contains information, BAW-1024P, Revision 2, proprietary to Framatome Technologies, Inc. (FTI), it is supported by an affidavit signed by FTI, the owner of the information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.790 of the Commission's regulations.

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Change: NRC PDR 1 INP

Accordingly, it is respectfully requested that the information which is proprietary to FTI be withheld from public disclosure in accordance with 10 CFR Section 2.790 of the Commission's regulations.

Correspondence with respect to the copyright or proprietary aspects of the items listed above or the supporting FTI Affidavit should reference BAW-10204P, Revision 2 and should be addressed to J. Taylor, Manager, Licensing Services, Framatome Technologies, Inc. (formerly B&W Nuclear Technologies), P. O. Box 10935, Lynchburg, Virginia 24506-0935.

In accordance with 10CFR50.91(b), STP is providing the State of Texas with a copy of this proposed amendment.

If you should have any questions concerning this matter, please call Mr. M. A. McBurnett at (512) 972-7206 or myself at (512) 972-8787.

T. H. Cloninger
Vice President,
Nuclear Engineering

HRP/lf

Attachment:

1. Affidavit
2. Safety Evaluation and No Significant Hazards Consideration Determination
3. Mark-ups of Proposed Change to Technical Specifications 3.4.5 and 3.4.6.2.
4. BAW-10204P, Revision 2, "South Texas Project Tube Repair Criteria for ODSCC at Tube Support Plates", Proprietary and Affidavit
5. BAW-10204, Revision 2, "South Texas Project Tube Repair Criteria for ODSCC at Tube Support Plates", Non-Proprietary

c: *

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* Without Attachments 4 & 5
** 2 Copies Attachments 4 & 5
*** Attachment 5 only

ATTACHMENT 1

AFFIDAVIT

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
Houston Lighting & Power	)	Docket Nos. 50-498
Company, et al.,	)	
	)	
South Texas Project	)	
Units 1 and 2	)	

AFFIDAVIT

I, T. H. Cloninger, being duly sworn, hereby depose and say that I am Vice President, Nuclear Engineering, of Houston Lighting & Power Company; that I am duly authorized to sign and file with the Nuclear Regulatory Commission the attached revision to proposed changes to Technical Specification 3.4.5 and 3.4.6.2; that I am familiar with the content thereof; and that the matters set forth therein are true and correct to the best of my knowledge and belief.

T. H. Cloninger
Vice President,
Nuclear Engineering

STATE OF TEXAS)
)
)

Subscribed and sworn to before me, a Notary Public in and for the State of Texas, this 22nd day of January, 1996.

LINDA RITTENBERRY
Notary Public, State of Texas
My Commission Expires 10/9/97

Linda Rittenberry
Notary Public in and for the
State of Texas

TSC-96\5269

AFFIDAVIT OF JAMES H. TAYLOR

A. My name is James H. Taylor. I am Manager of Licensing Services for Framatome Technologies, Inc. (FTI), and as such, I am authorized to execute this Affidavit.

B. I am familiar with the criteria applied by FTI to determine whether certain information of FTI is proprietary and I am familiar with the procedures established within FTI to ensure the proper application of these criteria.

C. In determining whether an FTI document is to be classified as proprietary information, an initial determination is made by the Unit Manager, who is responsible for originating the document, as to whether it falls within the criteria set forth in Paragraph D hereof. If the information falls within any one of these criteria, it is classified as proprietary by the originating Unit Manager. This initial determination is reviewed by the cognizant Section Manager. If the document is designated as proprietary, it is reviewed again by Licensing personnel and other management within FTI as designated by the Manager of Licensing Services to assure that the regulatory requirements of 10 CFR Section 2.790 are met.

D. The following information is provided to demonstrate that the provisions of 10 CFR Section 2.790 of the Commission's regulations have been considered:

(i) The information has been held in confidence by FTI. Copies of the document are clearly identified as proprietary. In addition, whenever FTI transmits the information to a customer, customer's agent, potential customer or regulatory agency, the transmittal requests the recipient to hold the information as proprietary. Also, in order to strictly limit any potential or actual customer's use of proprietary information, the following provision is included in all proposals submitted by FTI, and an applicable version of the proprietary provision is included in all of FTI's contracts:

"Purchaser may retain Company's proposal for use in connection with any contract resulting therefrom, and, for that purpose, make such copies thereof as may be necessary. Any proprietary information concerning Company's or its Supplier's products or manufacturing processes which is so designated by Company or its Suppliers and disclosed to Purchaser incident to the performance of such contract shall remain the property of Company or its Suppliers and is disclosed in confidence, and Purchaser shall not publish or otherwise disclose it to others without the written approval of Company, and no rights, implied or otherwise, are granted to produce or have produced any products or to practice or cause to be practiced any manufacturing processes covered thereby.

Notwithstanding the above, Purchaser may provide the NRC or any other regulatory agency with any such proprietary information as the NRC or such other agency may require; provided, however, that Purchaser shall first give Company written notice of such proposed disclosure and Company shall have the right to amend such proprietary information so as to make it non-proprietary. In the event that Company cannot amend such proprietary information, Purchaser shall prior to disclosing such information, use its best efforts to obtain a commitment from NRC or such other agency to have such information withheld from public inspection.

Company shall be given the right to participate in pursuit of such confidential treatment."

(ii) The following criteria are customarily applied by FTI in a rational decision process to determine whether the information should be classified as proprietary. Information may be classified as proprietary if one or more of the following criteria are met:

a. Information reveals cost or price information, commercial strategies, production capabilities, or budget levels of FTI, its customers or suppliers.

b. The information reveals data or material concerning FTI research or development plans or programs of present or potential competitive advantage to FTI.

c. The use of the information by a competitor would decrease his expenditures, in time or resources, in designing, producing or marketing a similar product.

d. The information consists of test data or other similar data concerning a process, method or component, the application of which results in a competitive advantage to FTI.

e. The information reveals special aspects of a process, method, component or the like, the exclusive use of which results in a competitive advantage to FTI.

f. The information contains ideas for which patent protection may be sought.

The document(s) listed on Exhibit "A", which is attached hereto and made a part hereof, has been evaluated in accordance with normal FTI procedures with respect to classification and has been found to contain information which falls within one or more of the criteria enumerated above. Exhibit "B", which is attached hereto and made a part hereof, specifically identifies the criteria applicable to the document(s) listed in Exhibit "A".

(iii) The document(s) listed in Exhibit "A", which has been made available to the United States Nuclear Regulatory Commission was made available in confidence with a request that the document(s) and the information contained therein be withheld from public disclosure.

(iv) The information is not available in the open literature and to the best of our knowledge is not known by Combustion Engineering, EXXON, General Electric, Westinghouse or other current or potential domestic or foreign competitors of FTI.

(v) Specific information with regard to whether public disclosure of the information is likely to cause harm to the competitive position of FTI, taking into account the value of the information to FTI; the amount of effort or money expended by FTI developing the information; and the ease or difficulty with which the information could be properly duplicated by others is given in Exhibit "B".

E. I have personally reviewed the document(s) listed on Exhibit "A" and have found that it is considered proprietary by FTI because it contains information which falls within one or more of the criteria enumerated in Paragraph D, and it is information which is customarily held in confidence and protected as proprietary information by FTI. This report comprises

information utilized by FTI in its business which afford FTI an opportunity to obtain a competitive advantage over those who may wish to know or use the information contained in the document(s).

James H Taylor
JAMES H. TAYLOR

State of Virginia)
) SS. Lynchburg
City of Lynchburg)

James H. Taylor, being duly sworn, on his oath deposes and says that he is the person who subscribed his name to the foregoing statement, and that the matters and facts set forth in the statement are true.

James H Taylor
JAMES H. TAYLOR

Subscribed and sworn before me
this 9th day of January 1996.

Brenda C. Cardoza
Notary Public in and for the City
of Lynchburg, State of Virginia.

My Commission Expires July 31, 1999

EXHIBITS A & B

EXHIBIT A

1. FTI Topical Report BAW-10204P, Revision 2, "South Texas Project Tube Repair Criteria for ODSCC at Tube Support Plates," January 1996.

EXHIBIT B

The above listed document contains information which is considered Proprietary in accordance with Criteria c, d, and e of the attached affidavit.

ATTACHMENT 2

SAFETY EVALUATION
AND
NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION
FOR
STP UNIT 1 ALTERNATE TUBE PLUGGING CRITERIA

DESCRIPTION OF AMENDMENT REQUEST

The proposed amendment would revise Specifications 3/4.4.5 and 3.4.6.2 including associated Bases 3/4.4.5 and 3/4.4.6.2 to allow the implementation of alternate steam generator tube plugging criteria for the tube support plate/tube intersections for Unit 1. The allowed primary-to-secondary operational leakage from any one steam generator will be reduced from 500 gallons per day (gpd) to 150 gpd and the total allowed primary-to-secondary operational leakage through all steam generators would be reduced from one gallon per minute (1440 gpd) to 600 gpd. This amendment is consistent with the guidance provided in NRC Generic Letter 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking."

This proposed amendment applies to Unit 1 only.

BACKGROUND

Previous inservice inspections and examinations of the steam generator tubes have identified intergranular stress corrosion cracking (IGSCC) on the outer diameter of the tubes at the tube support plate (TSP) intersections. This particular form of IGSCC is known as outer diameter stress corrosion cracking (ODSCC) and is a degradation phenomenon found in a number of nuclear power plant steam generators. Various tubes, including tube-to-TSP intersections, have been removed from affected steam generators from numerous nuclear plants for examination and testing. Each of the pulled tubes was sectioned and metallographically examined. The examinations have revealed multiple, segmented, and axial cracks with short lengths for the deepest penetrations. The ODSCC is generally confined to within the thickness of the TSPs, consistent with the corrosion mechanism which involves the concentration of impurities, including caustics, in the tube-to-TSP crevices. There is some potential for shallow ODSCC for a short distance above or below the TSP. This has been observed in the TSP intersections of some pulled tubes from another plant.

The steam generator tube specimens pulled from STP Unit 1 in 1993 and 1995 have shown only limited intergranular attack (IGA) associated with the ODSCC. However, more significant IGA has been observed to occur with ODSCC on some pulled tube specimens from other plants. These results suggest that the degradation developed as IGA plus stress corrosion cracking (SCC). This combination of IGA plus SCC was seen when maximum IGA depths were greater than 25 percent. A large number (greater than 100) of axial cracks around the circumference are commonly found on these tubes. The maximum depth of IGA is typically one-half to one-third of the SCC depth. Patches of cellular IGA/ODSCC formed by combined axial and circumferential orientation of microcracks are frequently found in pulled tube examinations. Axial crack segments have been the dominant flaw feature affecting the structural integrity of the pulled tube specimens as evidenced by results of burst tests of the pulled TSP intersections prior to sectioning. Testing of tubes with ODSCC has demonstrated a high margin to failure and evaluations have shown that existing tube plugging criteria would cause unnecessary and inappropriate tube plugging.

JUSTIFICATION

Specification 4.4.5.4.a.6, Plugging Limit, requires that tubes with imperfections exceeding 40 percent of the nominal tube wall thickness be removed from service. This criterion would result in unnecessarily plugging significant numbers of steam generator tubes affected with ODSCC at TSPs. Unnecessary plugged tubes reduce steam generator heat removal capability in both accident conditions and normal operations. To preclude this, implementation of an alternate plugging criteria (APC) is proposed.

The alternate plugging criteria involves a correlation between eddy current bobbin coil signal amplitude (voltage) versus tube burst pressure and leak rate. The principal parameter is voltage amplitude which is correlated with tube burst capability and leakage potential. The plugging criteria are developed by EPRI from testing of laboratory induced ODSCC specimens, extensive examination of pulled tubes from operating steam generators, and field experience from leakage due to indications at the tube support plates.

The APC is based on compliance with the NRC Generic Letter 95-05 which also is described in Topical Report BAW-10204-P, Rev. 2, "South Texas Project Tube Plugging Criteria for ODSCC at Tube Support Plates," which addresses the criterion provided in draft NUREG-1477, "Voltage-Based Interim Plugging Criteria for Steam Generator Tubes," to maintain steam generator tube serviceability. Topical Report BAW-10204-P, Rev. 2 is being provided with this submittal for information. At a future time, this Topical Report may be replaced with an equivalent report developed by Westinghouse in accordance with NRC Generic Letter 95-05. In this case, the Commission will be notified in writing and the Westinghouse Topical Report will be made available upon request. The methodology employed follows the industry degradation specific management methodology developed by EPRI and is similar to that implemented for Byron Unit 1, Beaver Valley Unit 1, and Farley Unit 1. The proposed bobbin coil voltage criteria detailed in this proposed amendment to the Technical Specifications reflects a conservative approach for the South Texas Project APC, recognizing that higher limits have been demonstrated to provide adequate margins in accordance with applicable regulatory requirements. The proposed alternate tube plugging criteria is provided in accordance with the following:

1. For Unit 1, implementation of the tube support plate alternate plugging criteria limit 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 outer diameter stress corrosion cracking (ODSCC) indications. The determination of the tube support plate intersections having ODSCC indications shall be based on the performance of at least a 20% random sampling of tubes inspected over their full length.

2. Distorted support plate indications (DSIs) confined within the thickness of the tube support plate with bobbin voltages less than or equal to 1.0 volt will be allowed to remain in service.

3. DSIs confined within the thickness of the tube support plate with a bobbin voltage of greater than 1.0 volt will be plugged except as noted in Item 5.

4. As part of a sample inspection program to help ensure that additional degradation modes are not occurring, all flaw indications with bobbin voltages greater than 1.0 volt but less than or equal to 2.85 volts will be inspected by rotating pancake coil-probe (RPC) inspection.

5. DSIs confined within the thickness of the tube support plate with a bobbin voltage greater than 1.0 volt but less than or equal to 2.85 volts may remain in service if an RPC inspection does not detect a flaw. DSIs with a bobbin voltage greater than 2.85 volts will be plugged.

6. For each upcoming cycle, the end-of-cycle voltage distribution will be established based upon the previous end-of-cycle eddy current data. Based upon this distribution, postulated steam generator tube leakage during a steam line break will be estimated based on the guidance of NRC Generic Letter 95-05. Projected leakage must remain below a level which results in offsite dose estimates remaining within the applicable dose limits. Should this estimation exceed the applicable dose limits, the highest voltage indications will be successively plugged until the leakage estimation drops below the applicable dose limits. While projected steam generator tube leakage during a steam line break will be calculated as prescribed in NRC Generic Letter 95-05, STP is also requesting that the current EPRI recommended leak rate calculational methodology be approved for Unit 1 use (NRC Generic Letter 95-05, Reference 3; PWR Steam Generator Tube Repair Limits - Technical Support Document for Outside Diameter Stress Corrosion Cracking at Tube Support Plates, Rev. 2).

7. An overall tube burst probability during a postulated steam line break event will be calculated and compared to the threshold of 1×10^{-2} defined in the NRC Generic Letter 95-05.

8. Tubes left in service as a result of application of the alternate tube support plate plugging criteria shall be inspected by bobbin coil probe during all future refueling outages.

9. The alternate plugging criteria will not be applied for tubes within the calculated tube support plate plastic deformation exclusion zone.

SAFETY ANALYSIS

The proposed change modifies the steam generator surveillance requirements to allow implementation of the alternate plugging criteria (APC) for Unit 1. Surveillance Requirement 4.4.5.2.b.4 was added to require future bobbin coil inspection of all tubes left in service as a result of the application of APC. Surveillance Requirement 4.4.5.2.d has been added to require 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 outer diameter stress corrosion cracking (ODSCC) indications. The determination of the tube support plate intersections having ODSCC indications shall be based on the performance of at least a 20% random sampling of tubes inspected over their full length. Surveillance Requirement 4.4.5.4.a.6 has been modified by including an exception to the current plugging limits so that the definition does not apply to the region of the tube subject to the tube support plate intersections since the alternate tube plugging criteria applies to this region. Surveillance Requirement 4.4.5.4.a.10 has been added to provide the limitations applicable for the "Tube Support Plate Alternate Plugging Criteria Limit." Surveillance Requirement 4.4.5.5.d has been added to address additional reporting criteria for those tubes where the tube support plate APC has been applied. Specification 3.4.6.2.c has been modified by changing (for Unit 1 only) the 1 gpm limit for combined leakage through all steam generators to 600 gallons per day and by changing (for Unit 1 only) the 500 gallons per day limit for leakage through any one steam generator to 150 gallons per day. The revised leakage limits are consistent with the methodology addressed in draft NUREG-1477. Bases 3/4.4.5, Steam Generators, has been modified to reflect the reduction in Unit 1 daily steam generator leakage limits from 500 gallons per day to 150 gallons per day, to delete "by radiation monitors of steam generator blowdown," and add a reference to Surveillance Requirement 4.4.5.4.a.10 for APC criteria. Bases 3/4.4.6.2, Operational Leakage, has been modified to address the new Unit 1 steam generator leakage limits. The added paragraph states that the new 150 gallons per day steam generator leakage limit (600 gallons per day total) is based on minimizing the potential for a large leakage event during a Main Steam Line Break.

In the development of the alternate plugging criteria, draft Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes" and RG 1.83 "Inservice Inspection of PWR Steam Generator Tubes" are used as the bases for determining that steam generator integrity considerations are maintained within acceptable limits. RG 1.121 describes a method, acceptable to the NRC staff, for meeting General Design Criteria (GDC) 14, 15, 31, and 32. The probability and consequences of steam generator tube rupture are reduced by determining the limiting safe conditions of degradation of steam generator tubing, beyond which tubes with unacceptable cracking, as established by inservice inspection, would be removed from service by plugging. This regulatory guide uses safety factors on loads for tube burst that are consistent with the requirements of Section III of the ASME Code. For the degradation occurring in the steam generator tube support plate elevations, tube burst criteria are inherently satisfied during normal

operating conditions by the presence of the tube support plate. The presence of the tube support plate enhances the integrity of the degraded tubes in that region by precluding tube deformation beyond the diameter of the drilled hole. It is not certain whether the tube support plate would function to provide a similar constraining effect during accident condition loadings. Therefore, no credit is taken in the development of the plugging criteria for the presence of the tube support plate during accident condition loadings. Conservatively, based on the existing database, burst testing shows that the safety requirements for tube burst margins during both normal and accident condition loadings can be satisfied with bobbin coil signal amplitudes of about 4.70 volts or less, regardless of the depth of tube wall penetration cracking. RG 1.83 describes a method acceptable for implementing GDC 14, 15, 31, and 32 through periodic inservice inspection for detection of significant tube wall degradation.

For the alternate tube plugging criteria developed for the steam generator tubes, no leakage is expected during normal operating conditions even with the presence of through wall cracks. This is the case because the stress corrosion cracking occurring in the tubes at the support plate elevations in the steam generators are short, tight, axially oriented macrocracks separated by ligaments of material. Relative to the expected leakage during accident condition loadings, the limiting event with respect to primary-to-secondary leakage is a postulated steam line break event.

The following items support this proposed license amendment.

1. <u>Chemistry</u>

   STP has undertaken steps to help mitigate steam generator tubing corrosion. Plant design was upgraded during construction to:

   - add a full flow feedwater dearator for dissolved oxygen control,
   - add cation condensate polishers in addition to the full flow mixed bed condensate polishers,
   - double the capacity of the Steam Generator Blowdown System to 1% of Main Steam flow,
   - remove copper components from the secondary system, and
   - use all volatile treatment (AVT)

   During the past two (2) years, alternate amine pH control was implemented to reduce iron transport. Current information included in the EPRI Secondary Chemistry Guidelines (e.g. molar ratio) is used to monitor the effectiveness of the chemistry program.

2. Steam Generator Leakage Monitoring

Steam generator leakage monitoring at STP employs a sampling program in conjunction with radiation monitors permanently installed on the Condenser Air Removal System, (RT8027), the Unit Vent Monitor (RT8010), the Steam Generator Blowdown (SGBD) Flash Tank (RT8043), and employing N-16 Primary to Secondary Leak Monitors permanently installed on each of the four main steam lines (RT8130B, RT8131B, RT8132B, and RT8133B). The STP program for detection and mitigation of steam generator tube leak events was upgraded earlier in response to industry lessons learned, such as IEN 91-043. The STP program for early leak detection provides for prompt detection and response, minimizing the likelihood of a steam generator tube rupture event. (Note: In addition to the monitors described below, additional monitors which are less sensitive to small leaks have been provided on each of the four main steam lines and on the four Steam Generator Blowdown lines. These are provided primarily for detection of a Steam Generator Tube Rupture event and are not discussed in detail.)

- Sampling:
  Each steam generator is routinely sampled for various purposes, including the detection of tube leaks and determination of secondary specific radioactivity once every 72 hours during operation in modes 1, 2, 3, and 4.

- Steam Generator Blowdown (SGBD) Radiation Monitor:
  The SGBD Radiation Monitor continuously checks the steam generator blowdown flash tank effluent. This monitor provides indication and alarms locally and in the Control Room. The SGBD Radiation Monitor detects water activation products as well as corrosion activation products and fission products. It is sensitive to leakage as low as five gallons per day. An alert or high alarm would be an indication of a primary-to-secondary leak.

- Condenser Air Removal System Radiation Monitor:
  The Condenser Air Removal System is provided with a radiation monitor which continuously monitors the effluent line from the Condenser Vacuum Pump. This monitor is designed to detect low levels of noble gas radioactivity and is sensitive to leaks as low as five gallons per day. An alarm from this detector indicates a primary-to-secondary system leak.

- Unit Vent Monitor:
  The Unit Vent Monitor is provided with a radiation monitor which samples the plant vent stack prior to discharge to the environment and monitors for particulates, iodine, and noble gases. The Unit Vent Monitor provides sampling capability of plant effluents in compliance with NUREG-0737, Item II.F.1.

- <u>N-16 Radiation Monitor:</u>
The N-16 gamma detectors provide continuous indication of individual steam generator primary-to-secondary leakage. The N-16 gamma detectors provide real time indication in the Control Room of steam generator leak rate in gallons per day and are used when reactor power is greater than or equal to 25 percent. The STP N-16 monitors are reactor power compensated for accurate leakage trending during power level reductions and increases. A recorder monitors the N-16 detector readings and provides a trend recording of steam generator leak rate. The N-16 monitors alarm in the Cold Chemistry Lab, from which they are controlled, while monitor readings are continuously available in the Control Room via the plant computer.

- <u>Station Response to a Steam Generator Tube Leak:</u>
Abnormal radiation in a steam generator indicates primary-to-secondary leakage. This can be shown by trends or alarms on main steam line N-16 monitors, the Condenser Vacuum Pump Effluent Monitor, the Steam Generator Blowdown Radiation Monitor, or from chemistry samples. A large leak would be indicated by feedwater flow being less than steam flow, decreasing feed flow, a mismatch in charging and letdown flow, or decreasing feed regulating valve position in conjunction with a stable steam generator level. These symptoms, however, would more likely be noticed with a tube rupture event. Procedures provide actions to mitigate the entire spectrum of steam generator tube leaks from the threshold of detectability up to a steam generator tube rupture event.

Upon any confirmed indication of leakage, the frequency of monitoring and sampling is increased in a manner proportionate to the severity of the leak. Additional confirmatory/diagnostic samples would be taken from the steam generator blowdown, and from the Condenser Air Removal System effluent. Operations begins to closely monitor the N-16 monitors in the Control Room.

- <u>Training:</u>
The operator training program has been upgraded previously to reflect training scenarios based on actual STP plant response to previous steam generator leak events. Plant operators and Chemical Analysis technicians have been trained in the use of the recently added N-16 monitors and in the upgraded station procedures for response to steam generator leaks.

- Steam Generator Leak Detection Program Adequacy:
  The plant leak rate monitors and procedures provide the required indications and alarms to ensure Reactor Coolant System leakage is detected early, while the leakage rate is low. In addition, leakage verification is provided by STP chemistry procedures which provide alternate means of calculating and confirming Reactor Coolant System leakage. These measures maximize assurance that leak evaluation and mitigation can occur before small leaks propagate to steam generator tube rupture events.

3. Eddy Current Test and Data Analysis Guidelines

   The data acquisition and analysis guidelines will be in accordance with those developed by the industry APC program and/or described in applicable STP Engineering bases documents for type of calibration, recording, and analysis requirements.

4. Eddy Current Data Analyst Training and Qualifications

   All analysts will be qualified per SNT-TC-1A and a minimum of 90% will be qualified as "Qualified Data Analyst". All analysts will take and pass the site specific Data Analysis Course prior to beginning work. A "Qualified Data Analyst" qualification is not required for supervision.

5. Tube Pulls

   The results of destructive examination of 18 tube support plate intersections from four tube pulls performed in 1993 and three tubes pulls performed in 1995 are included in BAW-10204-P. From these STP pull samples, one ~70% through wall axial defect and one with ~ 54% axial defect were burst tested. This information has been made available to EPRI in support of development of an industry wide tube pull data base.

   Draft NUREG-1477 states that removing tubes during each outage for examination and testing is important to enhance and validate the empirical burst and leakage correlations, to confirm that axial ODSCC continues to be the dominant degradation method at the tube support plate intersections, and to provide additional data for assessing the reliability of the inspection methods. Accordingly, STP pulled a total of three tubes from the Unit 1 steam generators prior to the start of Cycle 6. These specimens were used for additional burst testing and morphology confirmation.

The proposed amendment may preclude occupational radiation exposure that would otherwise be incurred by plant workers involved in tube plugging operations. It would minimize the loss of margin in the reactor coolant flow through the steam generator and is therefore safety enhancing as compared to less conservative plugging of otherwise structurally adequate tubes. The proposed amendment would avoid loss of margin in reactor coolant system flow and, therefore, assist in demonstrating that minimum flow rates are maintained in excess of that required for operation at

full power. Reduction in the amount of tube plugging required can reduce the length of plant outages and reduce the time that the steam generator is open to the containment environment during an outage. STP has determined that this methodology is applicable to our steam generators and provides a safe and effective alternative to plugging.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Pursuant to 10CFR50.91, this analysis provides a determination that the proposed change to the Technical Specifications described previously, does not involve any significant hazards consideration as defined in 10CFR50.92 as described below:

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Structural Considerations

   Industry testing of model boiler and operating plant tube specimens for free span tubing at room temperature conditions show typical burst pressures in excess of 5000 psi for indications of outer diameter stress corrosion cracking with voltage measurements at or below the structural limit of 4.0 volts. One model boiler specimen with a voltage amplitude of 19 volts also exhibited a burst pressure greater than 5000 psi. Burst testing performed on one intersection pulled from STP Unit 1 in 1993 with a 0.51 volt indication yielded a measured burst pressure of 8900 psi at room temperature. Burst testing performed on another intersection pulled from STP Unit 1 in 1995 with a 0.48 volt indication yielded a measured burst pressure of 9950 psi at room temperature.

   The projected end-of-cycle (EOC) voltage compares favorably with the 4.7 volt structural limit considering the EPRI voltage growth rate for indications at STP. Using the methodology of the NRC Generic Letter 95-05, the structural limit is reduced by allowances for uncertainty and growth to develop a beginning-of-cycle (BOC) repair limit which should preclude EOC indications from growing in excess of the structural limit. The non-destructive examination (NDE) uncertainty to be applied per EPRI is approximately 20 percent. The EPRI recommended growth allowance of 30 percent/EFPY is also to be applied. This growth value is conservative for STP Unit 1 based on previous inspection history. By adding NDE uncertainty allowances and a crack growth allowance to the repair limit, the structural limit can be validated. Therefore, the maximum allowable BOC repair limit (RL) based on the structural limit of 4.7 volts can be represented as:

   RL + (0.20 x RL) + (0.45* x RL) = 4.7 volts, which yields RL of 2.85 volts.

   * The 30% growth rate for 1 EFPY was scaled up to the cycle length used at South Texas.

This repair limit (2.85 volts) reasonably could be applied for APC implementation to repair bobbin indications greater than the 1.0 volt criterion specified by NRC Generic Letter 95-05 and is independent of RPC confirmation of the indications. STP has chosen to use a steam generator tube upper repair limit of 2.85 volts to assess tube integrity for those bobbin indications which are above 1.0 volt but do not have confirming RPC calls. This 2.85 volt upper limit for non-confirmed RPC calls is consistent with the NRC Generic Letter 95-05. Since the upper bound for repair of non-confirmed RPC is limited to a value far less than the structural limit associated with a full alternate criteria, the establishment of the repair limits are determined to be reasonable and conservative with respect to the industry pulled tube data base used.

Leakage Considerations

As part of the implementation of APC, the distribution of EOC cracking indications at the TSP intersections has been used to calculate the primary-to-secondary leakage which is bounded by the maximum leakage required to remain within applicable dose limits. This limit was calculated using the Technical Specification RCS Iodine-131 transient spiking values consistent with NUREG-0800. Application of the APC criteria requires the projection of postulated MSLB leakage based on the projected EOC voltage distribution for the beginning of cycle. Projected EOC voltage distribution is developed using the most recent EOC eddy current results and a voltage measurement uncertainty. Draft NUREG-1477 requires that all indications to which APC is applied must be included in the leakage projection.

The projected MSLB leakage rate calculation methodology prescribed in EPRI TR-100407 will be used to calculate the EOC leakage. A Monte Carlo approach will be used to determine the EOC leakage, accounting for all of the ECT uncertainties, voltage growth, and an assumed probability of detection (POD) of 0.6 for a 1.0 volt repair limit. The fitted logarithmic function probability of leakage correlation will be used to establish the STP MSLB leak rate used for comparison with a bounding allowable leak rate in the faulted loop which would result in radiological consequences which are within applicable dose limits. Due to the relatively low voltage levels of indications at STP and low voltage growth rates, it is expected that the actual calculated leakage values will be far less than this limit.

Therefore, implementation of APC does not adversely affect steam generator tube integrity and implementation will be shown to result in acceptable dose consequences. The proposed amendment does not result in any increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

Implementation of the proposed steam generator tube alternate plugging criteria for ODSCC at the TSP intersections does not introduce any significant changes to the plant design basis. Use of the criteria does not provide a mechanism which could result in an accident outside of the region of the TSP elevations since no ODSCC has been identified outside the thickness of the TSPs. It is therefore expected that for all plant conditions, neither a single or multiple tube rupture event would occur in a steam generator where APC has been applied.

Specifically, STP will implement, for Unit 1, a maximum leakage rate of 150 gpd per steam generator (SG) to help preclude the potential for excessive leakage during all plant conditions. The current technical specification limits on primary-to-secondary leakage at operating conditions are 1 gpm for all steam generators or 500 gpd for any one SG. The RG 1.121 criterion for establishing operational leakage rate limits governing plant shutdown is based upon leak-before-break (LBB) considerations to detect a free span crack before potential tube rupture as a result of faulted plant conditions. The 150 gpd limit is intended to provide for leakage detection and plant shutdown in the event of an unexpected crack propagation resulting in excessive leakage. RG 1.121 acceptance criteria for establishing operating leakage limits are based on LBB considerations such that plant shutdown is initiated if the permissible crack is exceeded.

The predicted EOC leakage for STP is based on a the calculated growth rate and does not take credit for the TSP proximity during normal operation. Thus, the 150 gpd limit provides for plant shutdown prior to reaching critical crack lengths. Additionally, this leak-before-break evaluation assumes that the entire crevice area is uncovered during the secondary side blowdown of a MSLB. Typically, it is expected for the vast majority of intersections that only partial uncovery will occur. Thus, the proximity of the TSP will enhance the burst capacity of the tube.

Steam generator tube integrity is continually maintained through inservice inspection and primary-to-secondary leakage monitoring. Any tubes falling outside the APC repair limits are removed from service. Therefore, the possibility of a new or different kind of accident from any accident previously developed is not created.

3. Does the change involve a significant reduction in a margin of safety?

The use of the voltage based bobbin probe for dispositioning ODSCC degraded tubes within TSP intersections by APC is demonstrated to maintain steam generator tube integrity in accordance with the requirements of RG 1.121. RG 1.121 describes a method acceptable to the NRC staff for meeting GDCs 14, 15, 31, and 32 by reducing the probability or the consequences of steam generator tube rupture. This is accomplished by

determining the limiting conditions of degradation of steam generator tubing, as established by inservice inspection, for which tubes with unacceptable cracking are removed from service. Upon implementation of the criteria, even under the worst case conditions, the occurrence of ODSCC at the TSP elevation is not expected to lead to a steam generator tube rupture event during normal or faulted plant conditions. The EOC distribution of crack indications at the TSP elevations will be confirmed to result in acceptable primary-to-secondary leakage during all plant conditions and that radiological consequences are not adversely impacted.

In addressing the combined effects of loss of coolant accident (LOCA) and safe shutdown earthquake (SSE) on the steam generator component (as required by GDC 2), it has been determined that tube collapse may occur in the steam generators at some plants. This is the case at STP as the TSP may become deformed as a result of lateral loads at the wedge supports at the periphery of the plate due to the combined effects of the LOCA rarefaction wave and SSE loadings. The resulting secondary-to-primary pressure differential on the deformed tubes may cause some of the tube to collapse.

There are two concerns associated with steam generator tube collapse. First, the collapse of steam generator tubing reduces the RCS flow area through the tubes. The reduction in flow area increases the resistance to flow of steam from the core during a LOCA which, in turn, may potentially increase peak clad temperature (PCT). Second, there is a potential that through wall cracks in tubes could sufficiently enlarge during tube deformation or collapse, causing sufficient in-leakage of secondary water back to the core which dilutes the poisoning effect of boron injection from the emergency cooling system. Again, an increase in core PCT may result.

Consequently, since the LBB methodology is applicable to the STP reactor coolant loop piping, the probability of breaks in the primary loop piping is sufficiently low that they need not be considered in the structural design of the plant. The analysis identified tubes located adjacent to wedge regions that are subject to potential collapse during combined LOCA and SSE. These tubes will be excluded from application of APC. Thus, existing tube integrity requirements apply to these tubes and the margin of safety is not reduced.

Implementation practices using the bobbin probe voltage based tube plugging criteria bounds RG 1.83 considerations by:

1) Using enhanced eddy current inspection guidelines consistent with those used by EPRI in developing the correlations. This provides consistency in voltage normalization,

2) Performing 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 outer diameter stress corrosion cracking (ODSCC) indications. The determination of the tube support plate intersections having ODSCC indications shall be based on the performance of at least a 20% random sampling of tubes inspected over their full length, and

3) Incorporating RPC inspection for all tubes with larger indications left in service. This further establishes the principal degradation morphology as ODSCC.

Implementation of APC at TSP intersections will decrease the number of tubes which must be repaired. Since the installation of tube plugs (to remove ODSCC degraded tubes from service) reduces the RCS flow margin, APC implementation will help preserve the margin of flow that would otherwise be reduced.

For each cycle the projected EOC primary-to-secondary leak rate allowed is bounded by a leak rate which limits the radiological consequences of a EOC MSLB to within applicable dose limits. Therefore, this change does not involve a significant reduction in the margin to safety.

It is therefore concluded that the proposed license amendment request does not result in a significant reduction in the margin of safety as defined in the plant Final Safety Analysis Report or Technical Specifications.

IMPLEMENTATION PLAN

STP requests this amendment be given an expeditious review and approval prior to May 1996 for the upcoming Unit 1 refueling outage. STP requests 10 days for implementation.