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April 4, 1996
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File No.: G20.02.01
10 CFR 50.90,
10 CFR 50.92,
10 CFR 51

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

South Texas Project
Unit 1
Docket No. STN 50-498
Revised Proposed Amendment To Incorporate Voltage-Based
Repair Criteria In Unit 1 Technical Specifications 3.4.5 and 3.4.6.2

Reference: Letter from Mr. T. H. Cloninger, South Texas Project, to the U.S. Nuclear Regulatory Commission (ST-HL-AE-5269), "South Texas Project , Unit 1, Docket No. STN 50-398, Unit 1 Technical Specifications 3.4.5 and 3.4.6.2" dated January 22, 1996.

The South Texas Project (STP) proposes to amend its Operating License NPF-76, Unit 1, by incorporating the revised attached proposed changes to Technical Specifications 3.4.5 and 3.4.6.2. This revised proposed amendment is submitted in response to questions addressed by the NRC Staff on the referenced submittal. The changes that have been made in this revised proposed amendment submittal as opposed to the referenced previous submittal are basically wording of the Justification, Safety Evaluation, and No Significant Hazards Consideration Determination in Attachment 2. Although the wording in this submittal has changed to be consistent with Generic Letter 95-05, the conclusions of the Safety Evaluation and No Significant Hazards Consideration Determination of the referenced previous submittal remain valid. In both the Safety Evaluation and the No Significant Hazards Consideration Determination, a better description of the STP actions and plans in meeting the requirements of Generic Letter 95-05 have been included. The marked-up Technical Specification in Attachment 3 has been reworded to be consistent with the wording in Generic Letter 95-05.

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." The purpose of this amendment is to modify the Unit 1 Steam Generator tube plugging criteria in Technical Specification 3.4.5, Steam Generators and

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the allowable leakage Technical Specification 3.4.6.2, Operational Leakage, and the associated Bases. These changes will allow the implementation of steam generator tube voltage-based repair criteria for the tube support plate/tube intersections for Unit 1 based on the guidance provided in Generic Letter 95-05.

STP has reviewed the attached proposed amendment pursuant to 10 CFR 50.92 and determined that it does not involve a significant hazards consideration. In addition, STP has determined that the proposed amendment satisfies the criteria of 10 CFR 51.22(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.

A revised Topical Report, BAW-10204P, proprietary to Framatome Technologies, Inc. (FTI), will be submitted under a different cover letter.

In accordance with 10 CFR 50.91(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.H. R. Pate at (512) 972-7787 or myself at (512) 972-8434.

W. T. Cottle
Executive Vice President,
and General Manager
Nuclear

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.

c:

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ATTACHMENT 1

AFFIDAVIT

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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

AFFIDAVIT

I, W. T. Cottle, being duly sworn, hereby depose and say that I am Executive Vice President and General Manager, Nuclear, 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.

W T Cottle

W. T. Cottle
Executive Vice President,
and General Manager
Nuclear

STATE OF TEXAS	)
	)
COUNTY OF MATAGORDA	)

Subscribed and sworn to before me, a Notary Public in and for the State of Texas, this 4th day of April, 1996.

EUNICE I. SARGENT
Notary Public, State of Texas
My Commission Expires 10/13/98

Eunice I. Sargent
Notary Public in and for the
State of Texas

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ATTACHMENT 2

SAFETY EVALUATION
AND
NO SIGNIFICANT HAZARDS
CONSIDERATION DETERMINATION
FOR
STP UNIT 1 VOLTAGE-BASED
REPAIR 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 steam generator voltage-based repair 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. 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

Technical 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. Unnecessarily plugged tubes reduce steam generator heat removal capability in both accident conditions and normal operations. To preclude this, voltage-based repair criteria for Westinghouse Steam Generator tubes affected by ODSCC is proposed.

Voltage-based repair criteria for Westinghouse Steam Generator tubes affected by ODSCC involves a correlation between eddy current bobbin coil signal amplitude (voltage) versus tube burst pressure and leak rate. The principal parameter is bobbin voltage amplitude which is correlated with tube burst capability and leakage potential. The voltage-based repair 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 voltage-based repair criteria is based on compliance with the NRC Generic Letter 95-05 and also is described in Topical Report BAW-10204-P, "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, will be provided at a future. At some 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, Sequoyah, 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 voltage-based repair criteria, recognizing that higher limits have been demonstrated to provide adequate margins in accordance with applicable regulatory requirements. The proposed voltage-based repair criteria is provided in accordance with the following for Unit 1:

1. Implementation of the steam generator tube/tube support voltage-based repair criteria requires a 100 % bobbin coil inspection for all hot leg tube support plate intersections and all cold leg tube support plate intersections down to the lowest cold leg tube support plate with known outer 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% random sampling of tubes inspected over their full length.

2. The tube support plate voltage-based repair criteria limit is used for the disposition of an alloy 600 steam generator tube for continued service that is experiencing predominately axially oriented ODSCC 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 ODSCC within the bounds of the tube support plate with bobbin voltage less than or equal to the lower voltage repair limit (1 volt for the STP Unit 1 steam generator 3/4-inch tubes), will be allowed to remain in service.

   b. Steam generator tubes, whose degradation is attributed to ODSCC within the bounds of the tube support plate with a bobbin voltage greater than the lower voltage repair limit (1 volt for STP Unit 1 steam generator 3/4-inch tubes), will be repaired or plugged except as noted in item 2.c below.

   c. Steam generator tubes, with indications of potential degradation attributed to ODSCC within the bounds of the tube support plate with a bobbin voltage greater than the lower voltage repair limit (1 volt for STP Unit 1 steam generator 3/4-inch tubes), but less than or equal to the upper repair voltage limit, may remain in service if a rotating pancake coil inspection does not detect degradation. Steam generator tubes, with indications of ODSCC degradation with bobbin voltage greater than the upper voltage repair limit (as calculated in accordance with the methodology in Generic Letter 95-05 as supplemented) will be plugged or repaired.

3. 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 limits of 10 CFR 100 and control room doses within GDC 19 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. Projected steam generator tube leakage during a steam line break will be calculated as prescribed in Generic Letter 95-05.

4. 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.

5. 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.

6. 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.

7. If an unscheduled mid-cycle inspection is performed, the mid-cycle repair limits apply instead of the limits identified in Technical Specifications 4.4.5.4.10.a, 4.4.5.4.10.b, and 4.4.5.10.c. The mid-cycle repair limits will be determined from the equations for mid-cycle repair limits in Generic Letter 95-05, Attachment 2, page 3 of 7. Implementation of these mid-cycle repair limits should follow the same approach as in Technical Specifications 4.4.5.4.10a, 4.4.5.10.b, and 4.4.5.10.c.

STP inspections are consistent with Generic Letter 95-05, Section 3, Inspection Criteria, and reference 5, Appendix A. STP has identified differences with justifications for the differences and clarifications of any aspects of STP inspections which vary from reference 5 due to changes in industry practices and further development of technology. STP will provide this information under a different cover letter.

SAFETY ANALYSIS

The proposed change modifies the steam generator surveillance requirements to allow implementation of the tube support plate voltage-based repair criteria 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 voltage-based repair criteria. 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 known 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 voltage-based repair 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 voltage-based repair 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 voltage-based repair criteria has been applied. Specification 3.4.6.2.c has been modified by changing the 1 gpm limit and by changing the 500 gallons per day limit for leakage through any one steam generator to 150 gallons per day for Unit 1 only. The revised leakage limits are consistent with the methodology addressed in draft NUREG-1477 and Generic Letter 95-05. 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 requirements for voltage-based repair 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 is based on minimizing the potential for a large leakage event during a Main Steam Line Break.

In the development of the voltage-based repair 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 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 degradation, 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 voltage-based repair 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 degradation. Generic Letter 95-05 requirements for use of the latest NRC approved database will be met and will ensure plant safety for future cycle structural repair limits. 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. STP is not applying for voltage-based repair criteria for flow distribution baffle intersections.

For the voltage-based repair criteria developed for the steam generator tubes, no leakage is expected during normal operating conditions even with the presence of through wall degradation. 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.

- <u>Steam Generator Leak Detection Program Adequacy:</u>
  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 the Data Analysis Guidelines

The data acquisition and the analysis guidelines will be in accordance with those given in Framatome Technologies, Inc. Topical Report, BAW 10204P 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". A "Qualified Data Analyst" qualification is not required for supervision. All analysts will take and pass the site specific data analysis course prior to beginning work. The site specific data analysis course will sensitize the analysts to identify indications attributable to primary water stress corrosion cracking (PWSCC) and to recognize the potential for PWSCC to occur at dented tube support plate intersections. This information will be contained in the current "Steam Generator Eddy Current Data Analysis Training Manual" or its equivalent.

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 ~ 54% axial defect were burst tested. No other defects such as PWSCC, circumferential degradation or deep IGA were found in the support plate intersections. Helium leak testing indicated no through-wall penetrations and thus hot leak rate testing was not performed. This information has been made available to EPRI in support of development of an industry wide tube pull database.

STP is committed to future cycle tube pulls as required by Generic Letter 95-05 or to participation in an industry tube pull program should such a program be approved by the NRC as allowed by Generic Letter 95-05.

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 and 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 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 10 CFR 50.91, 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 10 CFR 50.92 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 current structural limit of 4.7 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 next projected end-of-cycle (EOC) voltage compares favorably with the current structural limit considering the EPRI voltage growth rate for indications at STP. Using the methodology of 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 Generic Letter 95-05 is approximately 20 percent. The growth allowance will be 30 percent/EFPY or a STP Unit 1 plant specific growth value, to be calculated in accordance with Generic Letter 95-05, which ever is greater. The use of 30%/EFPY growth is conservative when compared to the actual STP growth experience. Each succeeding cycle upper voltage repair limit will also be conservatively established based on Generic Letter 95-05 methodology. By adding NDE uncertainty allowances and a growth allowance to the repair limit, the structural limit can be validated.

The upper voltage repair limit could be applied to bobbin coil voltages between the lower and upper repair limits to leave such indications in service independent of RPC confirmation. However, RPC confirmed indications will be conservatively removed from service consistent with Generic Letter 95-05.

Leakage Considerations

As part of the implementation of voltage-based repair criteria, the distribution of EOC degradation 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 the applicable dose limits of 10 CFR 100 and GDC 19. This limit was calculated using the Technical Specification RCS Iodine-131 transient spiking values consistent with NUREG-0800. Application of the voltage-based repair criteria requires the projection of postulated MSLB leakage based on the projected EOC voltage distribution from the beginning of cycle voltage distribution. Projected EOC voltage distribution is developed using the most recent EOC eddy current results and a voltage measurement uncertainty. Draft NUREG-1477 and Generic Letter 95-05 require that all indications, to which voltage-based repair criteria is applied, must be included in the leakage projection.

The projected MSLB leakage rate calculation methodology prescribed in Westinghouse WCAP-14277 or Generic Letter 95-05 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 bobbin coil eddy current test uncertainties, voltage growth, and an assumed probability of detection (POD) of 0.6. The fitted log-logistic probability of leakage correlation will be used to establish the STP MSLB leak rate for each cycle. This leak rate will be used for comparison with a bounding allowable leak rate in the faulted loop which would result in radiological consequences which are within the dose limits of 10 CFR 100 for offsite doses and GDC 19 for control room doses. Due to the relatively low voltage levels of indications at STP to date and low voltage growth rates, it is expected that the actual calculated leakage values will be far less than this limit for each successive cycle.

Therefore, implementation of voltage-based repair criteria does not adversely affect steam generator tube integrity and the radiological consequences will remain below the limits of 10 CFR 100 and GDC 19. 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 voltage-based repair 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 nor multiple tube rupture event would likely occur in a steam generator where voltage-based repair criteria 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 permissible degradation is exceeded.

The predicted EOC leakage for STP is based on 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 degradation 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 voltage-based repair criteria 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 voltage-based repair criteria 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 degradation 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 indications at the TSP elevations for each successive cycle will be confirmed to result in acceptable primary-to-secondary leakage during all plant conditions.

In addressing the combined effects of loss of coolant accident (LOCA) and safe shutdown earthquake (SSE) on the steam generators, 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 tubes 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 degradation 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.

The analysis results in Framatome Technologies, Inc. Topical Report, BAW 10204P, 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 voltage-based repair criteria. Thus, existing tube integrity requirements apply to these tubes and the margin of safety is not reduced. 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. 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 known outer diameter stress corrosion cracking (ODSCC) indications at each cycle. 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 bobbin voltages greater than 1.0 volt. This further establishes the principal degradation morphology as ODSCC.

Implementation of voltage-based repair criteria at TSP intersections will decrease the number of tubes which must be repaired at each subsequent inspection. Since the installation of tube plugs, to remove ODSCC degraded tubes from service, reduces the RCS flow margin, voltage-based repair criteria implementation will help preserve the margin of flow.

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 the dose limits of 10 CFR 100 for offsite doses and GDC 19 for control room doses. 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.