Florida Power
A Progress Energy Company

Crystal River Nuclear Plant
Docket No. 50-302
Operating License No. DPR-72

Ref: 10 CFR 50.90
10 CFR 50.36

March 21, 2001
3F0301-02

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Subject: License Amendment Request #252, Revision 0
Once Through Steam Generator Tube Surveillance Program,
Tube Repair Roll (Re-Roll) Process

Dear Sir:

Pursuant to 10 CFR 50.90, Florida Power Corporation (FPC) hereby submits a request for an amendment to the Crystal River Unit 3 (CR-3) Operating License No. DPR-72. The attached License Amendment Request (LAR) #252, Revision 0, requests a change to the CR-3 Improved Technical Specifications which are required by 10 CFR 50.36. Specifically, CR-3 proposes a repair roll (re-roll) process for the CR-3 Once Through Steam Generator (OTSG) tubes applicable to the upper and lower tubesheets of the OTSGs.

The request to approve the re-roll repair process for CR-3 is technically supported by proprietary Topical Report BAW-2303P, Revision 4, "OTSG Repair Roll Qualification Report." This B&W Owners Group (BWOG) topical report was previously submitted to the NRC by Oconee Nuclear Stations 1, 2 and 3, and Arkansas Nuclear One, Unit One.

Additionally, this request is technically supported by Topical Report BAW-2374, Revision 1,"Risk Informed Assessment of Once-Through Steam Generator Tube Thermal Loads Due to Breaks in Reactor Coolant System Upper Hot Leg Large-Bore Piping," which was submitted by Framatome ANP on March 12, 2001, for NRC review and approval on a generic basis. This topical report provides a risk informed basis for the acceptability of postulated thermal loads on OTSG tubes, tube repair products such as the re-roll, and tube-to-tubesheet joints induced by a loss-of-coolant accident (LOCA) in the large bore piping of the reactor coolant system (RCS) upper hot leg.

The CR-3 Plant Nuclear Safety Committee (PNSC) has reviewed and approved this LAR.

A001

FPC is requesting NRC approval of this LAR by August 30, 2001, to allow sufficient time for outage planning prior to Refueling Outage 12 planned for fall 2001. FPC intends to implement this LAR during Refueling Outage 12.

If you have any questions regarding this submittal, please contact Mr. Sid Powell, Supervisor, Licensing and Regulatory Programs at (352) 563-4883.

Sincerely,

Dale E. Young

Dale E. Young
Vice President, Crystal River Nuclear Plant

DEY/lvc

xc: Regional Administrator, Region II
NRR Project Manager
Senior Resident Inspector

Attachments:

A. Description of Proposed Change, Background, Reason for Request, and Evaluation of Request
B. No Significant Hazards Consideration Determination
C. Environmental Impact Evaluation
D. Proposed Revised Improved Technical Specifications Change Pages, Strikeout / Shadowed Format
E. Proposed Revised Improved Technical Specifications Change Pages, Revision Bar Format
F. NRC Questions Regarding Topical Report BAW-2303P, Revision 4

STATE OF FLORIDA

COUNTY OF CITRUS

Dale E. Young states that he is the Vice President, Crystal River Nuclear Plant for Progress Energy; that he is authorized on the part of said company to sign and file with the Nuclear Regulatory Commission the information attached hereto; and that all such statements made and matters set forth therein are true and correct to the best of his knowledge, information, and belief.

Dale E. Young

Dale E. Young
Vice President
Crystal River Nuclear Plant

Sworn to and subscribed before me this 21st day of March, 2001, by Dale E. Young.

LISA A. MORRIS
Notary Public, State of Florida
My Comm. Exp. Oct. 25, 2003
Comm. No. CC 879691

Lisa A Morris

Signature of Notary Public
State of Florida

LISA A MORRIS

(Print, type, or stamp Commissioned Name of Notary Public)

Personally Known X -OR- Produced Identification ______

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302/LICENSE NUMBER DPR-72

ATTACHMENT A

LICENSE AMENDMENT REQUEST #252, REVISION 0

ONCE THROUGH STEAM GENERATOR TUBE SURVEILLANCE PROGRAM, TUBE REPAIR ROLL (RE-ROLL) PROCESS

Description of Proposed Change, Background, Reason for Request, and Evaluation of Request

LICENSE AMENDMENT REQUEST (LAR) #252, REVISION 0
ONCE THROUGH STEAM GENERATOR TUBE SURVEILLANCE PROGRAM,
TUBE REPAIR ROLL (RE-ROLL) PROCESS

BACKGROUND

Approval of this LAR will accomplish the following related to the re-roll process: 1.) Supplement the previously approved re-roll repair process by providing revised exclusion zones for the installation of re-rolls, 2.) Qualify the installation of multiple re-rolls in a single tube, and 3.) Qualify the installation of re-rolls in the lower tubesheets. A short history of this issue is provided below.

Crystal River Unit 3 (CR-3) is a Babcock and Wilcox (B&W) designed pressurized water reactor with Model 177 FA Once Through Steam Generators (OTSGs). For this design, the primary coolant enters the steam generators at the top of the tubes and exits at the bottom of the tubes, where the primary coolant is directed back to the reactor coolant pumps and the reactor vessel. The functions of the OTSGs are to provide a pressure boundary between the reactor coolant and the secondary side fluid, confine the fission and activation products within the reactor coolant system, and to provide heat transfer capability and a heat sink to remove the heat produced during power operations. They also provide main and emergency feedwater flow paths and heat transfer capability for normal or emergency cooldown.

The OTSG tubes are mill annealed Alloy 600 (Inconel) which have been sensitized as a result of the full vessel post-fabrication heat treatment. The original tube-to-tubesheet joint consists of a roll expansion of one to two inches in length with a seal weld (fillet) between the tube and the primary side tubesheet cladding. The upper tubesheets are 24 inch thick carbon steel with a minimum primary side Inconel clad of 5/16 inch in the "A" OTSG and 15/32 inch in the "B" OTSG. Each OTSG has 15,531 tubes with a nominal outer diameter of 0.625 inches and a nominal wall thickness of 0.034 inches.

On June 28, 1999, the Nuclear Regulatory Commission (NRC) issued License Amendment No. 180 approving for CR-3 the use of repair rolls as described in B&W Owners Group (BWOG) Topical Report BAW-2303P, Revision 3, "OTSG Repair Roll Qualification Report," and Topical Report BAW-2342P, "OTSG Repair Roll Qualification Report Addendum A." The Addendum addressed tube loads generated by a Small Break Loss of Coolant Accident (SBLOCA) which had not been previously considered during the qualification of the repair roll (re-roll), and could be more limiting regarding axial tube loads than those from a Main Steam Line Break (MSLB). CR-3 installed 1547 re-rolls in the upper tubesheet of both OTSGs during the 11th refueling outage conducted during the fall of 1999.

Recently, Framatome Technologies Incorporated (FTI) re-qualified the re-roll process using the limiting load at each B&W plant (SBLOCA or MSLB). FTI completed testing and re-qualification of the re-roll process, as documented in Revision 4 to Topical Report BAW-2303P. This revision of BAW-2303P has been reviewed and approved by the NRC as part of a Duke Energy Corporation LAR submitted on September 12, 2000, for Oconee Nuclear Stations 1, 2, and 3, References 1 to 5.

As part of the re-qualification process, FTI evaluated the re-roll for installation in the upper and lower tubesheets and for installation of multiple re-rolls in a single tube. The re-roll is qualified to potential axial loading conditions expected during the most limiting of either a SBLOCA or MSLB. Testing completed by FTI for the re-qualification of the re-roll was performed to measure the loads at which tube slippage would occur, to measure leakage for the re-roll joints that did not slip, and to measure the leakage of the re-roll joints if tube slippage did occur. The re-roll slippage under faulted conditions constitutes a change in the design criteria compared to the original qualification of the re-roll. However, Appendix B of BAW-2303P, Revision 4 shows that for CR-3, no re-rolls installed in qualified locations are predicted to slip during faulted accident transients. The re-rolls installed under the previous revision of BAW-2303P are bounded by revision 4.

The effects of a break in large-bore Reactor Coolant System (RCS) piping for OTSGs were evaluated in Topical Report BAW-2374, Revision 0, July 2000, "Justification for Not Including Postulated Breaks in Large-Bore Reactor Coolant System Piping in the Licensing Basis for Existing and Replacement Once Through Steam Generators." The report was prepared by FTI for the BWOG and submitted on July 7, 2000, for NRC review and approval. The topical report requested a change to the licensing basis of B&W designed nuclear power plants such that the effects of a break in large-bore RCS piping need not be included for repair products of existing OTSGs. Approval of BAW-2374 would not result in changes to the CR-3 Improved Technical Specifications (ITS).

Since the NRC staff had not finalized their review of BAW-2374, Revision 0, the NRC staff imposed several license conditions to the Oconee Unit 1 re-roll license amendment issued on December 2000. The BWOG revised Topical Report BAW-2374, and submitted Topical Report BAW-2374, Revision 1, "Risk Informed Assessment of Once-Through Steam Generator Tube Thermal Loads Due to Breaks in Reactor Coolant System Upper Hot Leg Large-Bore Piping," on March 12, 2001. This revision of the topical report will resolve previous open items such that no license conditions will be necessary in order to obtain approval of the re-roll process for CR-3.

Oconee also committed to verify that their plant-specific Emergency Operating Procedures (EOPs) are consistent with the descriptions in BAW-2374 in regard to the key operator actions for mitigation of the accident sequence of concern. CR-3 has reviewed the plant-specific EOPs to ensure that they are consistent with the descriptions in BAW-2374, Revision 0, in regards to the key operator actions required to transfer the Emergency Core Cooling System (ECCS) suction from the Borated Water Storage Tank to the Containment Sump, and verify secondary system isolation to contain primary-to-secondary leakage. Therefore, a similar commitment is no longer necessary.

Attachment F provides CR-3 responses to questions developed by the NRC during the review of Topical Report BAW-2303P, Revision 4. These questions were posed to two BWOG member utilities who requested the use of the topical report as the basis for the re-roll repair process.

SUMMARY OF CHANGES TO THE ITS

ITS 5.6.2.10, "STEAM GENERATOR (OTSG) TUBE SURVEILLANCE PROGRAM"

Description of Specification Change

Revise ITS 5.6.2.10.4.a.11.b to state the applicability of the re-roll to the upper and lower tubesheets, to incorporate BAW-2303P, Revision 4, and to clarify that multiple re-rolls may be performed in a single OTSG tube. The changes to Pages 5.0-17 and 5.0-17A will read as follows:

> **"Installation of repair rolls in the upper and lower tubesheets in accordance with BAW-2303P, Revision 4. The repair process (single, overlapping, or multiple roll) may be performed in each tube. The repair roll area will be examined using eddy-current methods following installation. The repair roll must be free of imperfections and degradation for the repair to be considered acceptable.**
>
> **The repair roll in each tube will be inspected during each subsequent inservice inspection while the tube with a repair roll is in service. The repair roll will be considered a specific limited area and will be excluded from the random sampling. No credit will be taken for meeting the minimum sample size.**
>
> **If primary-to-secondary leakage results in a shutdown and the cause is determined to be degradation in a repair roll, 100% of the repair rolls in that OTSG shall be examined. If that inspection results in entering Category C-2 or C-3 for specific limited area inspection, as detailed in Table 5.6.2-3, 100% of the repair rolls shall be examined in the other OTSG."**

Reason for Request

The proposed changes to ITS 5.6.2.10.4.a.11.b are to update the requirements for the re-roll process used in the CR-3 OTSG Tube Surveillance Program to the requirements established in Topical Report BAW-2303P, Revision 4.

Evaluation of Request

Installation of re-rolls in accordance with BAW-2303P, Revision 4, are being proposed by this LAR as a method to repair defective tubes with unacceptable indications within the tubesheets. The augmented inspection described in proposed ITS 5.6.2.10.4.a.11.b will ensure an appropriate level of monitoring is performed on the re-roll areas and tube structural integrity is maintained. The information provided in the following evaluation addresses the transients considered in Topical Report BAW-2303P, Revision 4, and discusses design characteristics and inspection aspects pertinent to the re-roll. Further detailed information is available in the topical report.

Qualification

Each transient case (normal operating and accident) corresponds to a set of temperatures, pressures and resulting tubesheet deflections that may cause a change to the contact pressure of the rolled joint relative to the condition at installation. Differential dilation is the term used to describe the alteration in interference fit between the tube and the tubesheet bore.

Eight test cases were developed that represented and bounded 2800 pairs of differential dilations from normal operating and accident transient analyses. Finite Element analyses results were reviewed to determine a bounding set of dilation test cases. A bounding set of corresponding axial loads were developed for all normal operating and accident transients for the OTSG.

The cooldown transient axial tensile loads and differential dilations bound the normal operating transients for slip tests. The time of maximum tube load was selected as the bounding time point for the slip tests.

SBLOCA and MSLB transients bound the accident transients. The time of maximum tube load was selected as the bounding time point for the SBLOCA slip tests. For the CR-3 MSLB transient, the time of maximum tube load for slip tests and the time of maximum pressure differential for leak tests were the same. All leak tests were conservatively performed at the maximum possible pressure differential of 2575 psig, derived from the safety relief valve set point plus a 3% allowance for set point tolerance.

Joint Slippage

The purpose of the slip tests was to verify that the re-roll could withstand axial loads during normal operating pressure and accident conditions without benefit from the original roll or the tube-to-tubesheet weld. The tested condition did not take credit for the heel transition by representing a full circumferential severance at the end of the effective 1 inch roll (primary side). Applicable tubesheet bore dilations were achieved and an axial load was applied using either a swage-lock fitting or an inside diameter (ID) gripper attached to the free end of the tube. Tube movement was monitored during the tests and verified by measuring the depth of the tube end after each test.

BAW-2303P, Revision 4, shows that for CR-3 slippage is not a concern. Therefore, leakage due to slippage will not be calculated.

Application of Leakage

The purpose of leak testing was to quantify leak rates for re-rolls for accident conditions. All leak tests were conservatively performed at the maximum possible pressure differential of 2575 psig, derived from the safety relief valve set point plus a 3% allowance for set point tolerance. A seal plate was installed, the block pressurized to 2625 psig (2575 psig plus uncertainty measurement) and the tube end was sealed so that the leak path was through the re-roll for a 20 minute period. If required, additional axial load was applied. The bounding non-slip leak rates were developed by calculating a one-sided 95% upper confidence limit in the statistical analysis.

Inspection

The re-roll area will be examined using eddy-current methods as described in BAW-2303P, Revision 4. The re-roll must be free of imperfections and degradation for the repair to be considered acceptable and the tube returned to service. The re-roll is considered a specific limited area as defined in the ITS. As a specific limited area, the re-roll in each tube will be inspected during each subsequent inservice inspection while the tube with a re-roll is in service. The augmented inspection is added to ensure an appropriate level of monitoring is performed on the re-roll areas.

The re-roll will be excluded from the random sampling and will not be credited for meeting the minimum sample size. Excluding the re-roll inspections from the minimum sample size requirements and random tube inspection results category is consistent with the current CR-3 ITS 5.6.2.10.2.d requirements for a specific limited area. The re-rolls are distinguished from the remainder of the tube bundle by unique physical construction. Not taking credit for these inspections for the minimum sample size ensures that a larger population of tubes will be examined during the inservice inspection. Not including the re-roll inspection results in the overall inspection results classification will prevent bias of the random inspection results due to a specific limited area in the OTSGs.

Design Basis Accidents

The MSLB was analyzed as the most limiting design basis accident for BAW-2303P, Revision 3. Recently, FTI evaluated RCS accident analyses to identify the maximum OTSG tube tensile loads. The tube loads were re-analyzed for Revision 4, resulting in a new set of design loads for each BWOG plant. Based on these analyses and evaluations, the pressurizer surge line break, which is categorized as a SBLOCA, will produce the most limiting tensile tube load. As previously discussed, LBLOCA issues applicable to OTSGs are addressed in References 6 and 7.

Defense in Depth & Safety Margin

The re-roll methodology described in BAW-2303P, Revision 4, was qualified in accordance with the guidance of draft NRC Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," dated August 1976. A test pressure of 3 times normal operating differential pressure and 1.4 times MSLB differential pressure was applied to the sample tubes. The margin of safety against tube rupture under normal operating conditions, recommended in RG 1.121, should be equal to or greater than 3 at any tube location where defects have been detected, and be consistent with the margin of safety determined by the stress limits specified in NB-3225 of Section III of the Boiler and Pressure Vessel Code of the American Society of Mechanical Engineers. Normal structural loads imposed on the tube-to-tubesheet roll are derived from the differential pressure between the primary and secondary sides of the tubes. Cyclic loading from transients (e.g., startup/shutdown) is also considered in the qualification of the roll joints.

Description of Re-roll Process

The re-roll is a process whereby a new primary-to-secondary pressure boundary joint is established inboard of the tube degradation by creating a new roll joint within the tube at a point closer to the secondary face of the tubesheet. The new leak limiting roll joint is established to remove the area of degradation from pressure boundary service. The re-roll process was previously qualified by FTI for B&W OTSGs with Inconel Alloy 600 material tubes in the upper tubesheet and approved by the NRC as an approved repair technique. FTI Topical Report BAW-2303P, Revision 4 "OTSG Repair Roll Qualification Report," provides the technical basis for the re-qualification of the re-roll process, including single, overlapping and multiple rolls.

The repair is based on establishing a mechanical joint capable of carrying normal operating and accident condition structural loads without taking credit for support from the original tube expansion and tube seal weld. The single re-roll joints (1 inch) and overlapping re-roll joints (1.625 inch) were qualified to carry structural loads and minimize potential leakage for the OTSG tubes.

The full length of the re-roll in a tube will establish a new pressure boundary for the defective or degraded tubes. Upon completion of the re-roll, the original tube-to-tubesheet joint will no longer be considered part of the pressure boundary and will be outside of the inspection area of interest. The physical length of the expander will control the re-roll length. The re-roll will be situated entirely inboard of the original roll (there will be no overlap with the existing roll).

Multiple Re-rolls and Lower Tubesheet Installation

Both the single and overlapping re-rolls may be installed in the upper and lower tubesheets. Multiple re-rolls may be installed in a tube, except for the exclusion zone as identified in Figure B-1 and B-2 of BAW-2303P, Revision 4. Multiple re-rolls may be installed in the same tubesheet with a new re-roll inboard of previous re-rolls. Re-rolls may also be installed in the upper and lower tubesheet of the same tube.

The installation of re-rolls in the lower tubesheet region is the same process as the upper tubesheet. Consideration was given to what effect the deposits in the crevice would have on structural integrity and leakage of the re-roll. Leak rate data from testing conducted in 1999 was evaluated, Reference 2. The test sample included tubes with and without crevice deposits. The test results showed that for the OTSG installation process, a clean crevice leaks more than a packed crevice. The decreased leakage for the packed crevice is attributed to sludge providing a partial seal between the tube and the tubesheet that would be an otherwise open flow path in a clean crevice. An evaluation of joint strength data from 1999, from testing performed on clean crevice samples for pre-fatigue and post-fatigue conditions and packed crevice samples for post-fatigue conditions, showed a maximum of 10% difference in joint strength for the tested conditions. For the configurations tested, the results showed that the pre-fatigue, clean crevice sample resulted in the minimum joint strength. Therefore, testing performed without deposits is conservative for both the upper and lower tubesheets regarding joint integrity and leakage.

Risk Assessment

CR-3 has evaluated the risk implications of using the proposed re-roll process on severe accident vulnerabilities, including OTSG tube structural integrity and pressure-retaining characteristics. The risk of severe accidents causing degradation or post-accident leakage of the OTSG tubes repaired by this process is not increased. This is because the proposed repair process effectively removes the area of degradation from service by creating a new pressure boundary with at least the same design capabilities as the original pressure boundary.

Description of Specification Change

Delete the following words from ITS 5.6.2.10.3:

> **"...except, a one-time change for Cycle 11 is granted to modify the scheduled inspection frequency from a calendar-based interval to an interval of 21.6 months of operating time at a temperature of 500°F or above (measured at the hot leg side). This will allow the OTSG tube inspection to coincide with Refuel Outage 11R."**

Reason for Request

The above paragraph was added to ITS 5.6.2.10.3 by License Amendment No. 176, Reference 8, for Cycle 11 only. The approved interval which consisted of 21.6 months of operating time at a temperature of 500°F or above was only applicable until Refueling Outage 11 (October 1, 1999).

Evaluation of Request

The deletion of the above paragraph restores the text of ITS 5.6.2.10.3 to the wording that existed prior to License Amendment No.176. Therefore, it is considered an editorial change.

Description of Specification Change

Delete the second paragraph of ITS 5.6.2.10.4.b. which reads as follows:

> **"There are a number of OTSG tubes that have the potential to exceed the tube plugging/repair limit as a result of tube end anomalies. Defective tubes will be repaired or plugged during the next outage of sufficient duration. An evaluation has been performed which confirms that operability of the CR-3 OTSGs will not be impacted with those tubes in service."**

Reason for Request

ITS 5.6.2.10.4.b is no longer applicable. This paragraph was inserted by License Amendment No. 169, as a one-time change to the ITS.

Evaluation of Request

In the summer of 1997, Florida Power Corporation (FPC) detected tube end anomalies in the CR-3 OTSGs. In LAR #228, FPC requested to leave the OTSG tubes affected by tube end anomalies (TEC) in service until either Refueling Outage 11 or an outage of sufficient duration, if such an outage would occur before Refueling Outage 11 (fall 1999). The request was approved by License Amendment No. 169.

Subsequently, FPC requested approval for alternate repair criteria to leave TEC in service. The request was approved by License Amendment No. 188. Under the alternate repair criteria, axially-oriented indications are allowed to remain in service without repair while tubes with circumferential, mixed mode or volumetric, indications are repaired. Thus the alternate repair criteria, approved by License Amendment No. 188, supercedes the change approved by License Amendment No. 169. Therefore, the proposed deletion of the second paragraph of ITS 5.6.2.10.4.b is appropriate.

References

1. Duke Energy Corporation to NRC letter dated September 12, 2000, Oconee Nuclear Station, Units 1, 2, and 3, Docket Numbers 50-269, 50-270, and 50-287, "License Amendment Request for Technical Specification 5.5.10.e.6, Steam Generator Tube Surveillance Program (TSCR 2000-07)."

2. Duke Energy Corporation to NRC letter dated October 26, 2000, Oconee Nuclear Station, Units 1, 2, and 3, Docket Nos. 50-269, 50-270, and 50-287, "Response to NRC Questions on License Amendment Request for Technical Specification 5.5.10.e.6 and Topical Report BAW-2303P, Revision 4."

3. Duke Energy Corporation to NRC letter dated November 10, 2000, Oconee Nuclear Station, Units 1, 2, and 3, Docket Nos. 50-269, 50-270, and 50-287, "Response to NRC Questions on License Amendment Request for Technical Specification 5.5.10.e.6 and Topical Report BAW-2303P, Revision 4."

4. Duke Energy Corporation to NRC letter dated December 8, 2000, Oconee Nuclear Stations 1, 2, and 3, Docket Nos. 50-269, 50-270, and 50-287, "Supplemental Information Regarding License Amendment Request for Technical Specification 5.5.10.e.6 and Topical Report BAW-2303P, Revision 4."

5. NRC to Duke Energy Corporation letter dated December 15, 2000, "Oconee Nuclear Station, Units 1, 2, 3 RE: Issuance of Amendments (TAC NOS. MA9969, MA9970, and MA9971)."

6. B&W Owners Group to NRC letter dated July 7, 2000, "Justification for Not Including Postulated Breaks in Large-Bore Reactor Coolant System Piping in the Licensing Basis for Existing and Replacement Once Through Steam Generators."

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1. Duke Energy Corporation to NRC letter dated September 12, 2000, Oconee Nuclear Station, Units 1, 2, and 3, Docket Numbers 50-269, 50-270, and 50-287, "License Amendment Request for Technical Specification 5.5.10.e.6, Steam Generator Tube Surveillance Program (TSCR 2000-07)."

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3. Duke Energy Corporation to NRC letter dated November 10, 2000, Oconee Nuclear Station, Units 1, 2, and 3, Docket Nos. 50-269, 50-270, and 50-287, "Response to NRC Questions on License Amendment Request for Technical Specification 5.5.10.e.6 and Topical Report BAW-2303P, Revision 4."

4. Duke Energy Corporation to NRC letter dated December 8, 2000, Oconee Nuclear Stations 1, 2, and 3, Docket Nos. 50-269, 50-270, and 50-287, "Supplemental Information Regarding License Amendment Request for Technical Specification 5.5.10.e.6 and Topical Report BAW-2303P, Revision 4."

5. NRC to Duke Energy Corporation letter dated December 15, 2000, "Oconee Nuclear Station, Units 1, 2, 3 RE: Issuance of Amendments (TAC NOS. MA9969, MA9970, and MA9971)."

6. B&W Owners Group to NRC letter dated July 7, 2000, "Justification for Not Including Postulated Breaks in Large-Bore Reactor Coolant System Piping in the Licensing Basis for Existing and Replacement Once Through Steam Generators."

7. Framatome ANP to NRC letter dated March 12, 2001, submitting Topical Report BAW-2374, "Risk Informed Assessment of Once-Through Steam Generator Tube Thermal Loads Due to Breaks in Reactor Coolant System Upper Hot Leg Large-Bore Piping."

8. NRC to FPC letter dated May 5, 1999, "Crystal River Unit 3 - Issuance of Amendment Regarding Steam Generator Tube Surveillance Program Inspection Interval (TAC NO. MA4702)."

9. NRC to FPC letter dated July 30, 1998, "Crystal River Unit 3 - Issuance of Amendment Regarding Steam Generator Tube End Anomalies (TAC NO. MA 2123)."

10. NRC to FPC letter dated October 1, 1999, "Issuance of Amendment Regarding Alternate Repair Criteria for Steam Generator Tubing (TAC NO. MA5395)."

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302/LICENSE NUMBER DPR-72

ATTACHMENT B

LICENSE AMENDMENT REQUEST #252, REVISION 0

ONCE THROUGH STEAM GENERATOR TUBE SURVEILLANCE PROGRAM, TUBE REPAIR ROLL (RE-ROLL) PROCESS

No Significant Hazards Consideration Determination

NO SIGNIFICANT HAZARDS CONSIDERATION

An evaluation of this proposed License Amendment Request (LAR) has been performed in accordance with 10 CFR 50.91(a)(1) regarding significant hazard considerations using the standards in 10 CFR 50.92(c). A discussion of these standards as they relate to this LAR follows:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated.*

   The re-roll process is a method to create a new primary-to-secondary pressure boundary joint in the upper tubesheet of Babcock & Wilcox (B&W) Once Through Steam Generators (OTSGs) manufactured with Inconel Alloy 600 tubes. The new pressure boundary is established by the re-roll to remove degradation of the existing roll joint from pressure boundary service. The re-roll process has been previously qualified as an acceptable repair methodology for use in the upper tubesheet of the Crystal River Unit 3 (CR-3) OTSGs by License Amendment No. 180. This proposed LAR incorporates Revision 4 of Topical Report BAW-2303P, "OTSG Repair Roll Qualification Report." This proposed LAR also addresses several editorial changes which do not impact the current CR-3 accident analyses.

   The qualification of the OTSG tube re-roll methodology is based on establishing a mechanical joint length that will carry all structural loads imposed on the OTSG tubes while maintaining the required margins during normal and accident conditions. A series of tests and analyses were performed to establish the minimum acceptable length of the OTSG tube re-roll. Tests performed included leak, tensile, fatigue, ultimate load and eddy-current measurement uncertainty. The analyses evaluated plant operating and faulted load conditions. OTSG tube leakage remains bounded by the evaluation presented in the CR-3 Final Safety Analysis Report (FSAR) for a main steam line break (MSLB). The current CR-3 Improved Technical Specifications (ITS) include a description of the required inspection program for the OTSG tube re-rolls. The required ITS inspections following OTSG tube re-roll installation, and during future inservice inspections, ensure continuous monitoring of these tubes such that in service degradation of tubes repaired by the re-roll process will be detected. Based on the qualification testing and analyses performed, as well as the industry experience with the use of the OTSG tube re-roll processes, there are no new safety issues associated with the use of the re-roll methodology. The probability of a steam generator tube rupture is not increased by the re-roll since it is a repair process not applied to defective OTSG tube areas. This repair process establishes a new pressure boundary roll joint which is free of degradation. Therefore, this change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. *Create the possibility of a new or different kind of accident from previously evaluated accidents.*

   The re-roll process creates no new failure modes or accident scenarios. The new pressure boundary joint created by the re-roll process has been demonstrated, by testing and analysis, to provide structural and leakage integrity equivalent to the original design and construction for all normal operating and accident conditions. Furthermore, testing and analysis demonstrate that the re-roll process creates no new adverse effects for the repaired tube and does not change the design or operating characteristics of the OTSGs. BAW-2303P, Revision 4, addresses limiting events for steam generator re-roll repairs. These events include Main Steam Line Break, Small Break Loss of Coolant Accident and other transients on the B&W Once Through Steam Generators. Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. *Involve a significant reduction in a margin of safety.*

   The re-roll process effectively removes the defective/degraded area of the tube from service by establishing a new pressure boundary. The re-roll interface created with the tubesheet satisfies the necessary structural, leakage and heat transfer requirements. Implementation of BAW-2303P, Revision 4, will result in assurance that parameters affecting the integrity of steam generator tubes continue to meet safety analyses and industry codes and standards. Therefore, the FSAR analyzed accident scenarios remain bounding, and the use of the re-roll process does not significantly reduce the margin of safety.

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302/LICENSE NUMBER DPR-72

ATTACHMENT C

LICENSE AMENDMENT REQUEST #252, REVISION 0

ONCE THROUGH STEAM GENERATOR TUBE SURVEILLANCE PROGRAM,
TUBE REPAIR ROLL (RE-ROLL) PROCESS

Environmental Impact Evaluation

ENVIRONMENTAL IMPACT EVALUATION

10 CFR 51.22(c)(9) provides criteria for identification of licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed amendment to an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed amendment would not:

(i) involve a significant hazards consideration,

(ii) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, and

(iii) result in a significant increase in individual or cumulative occupational radiation exposure.

Florida Power Corporation has reviewed this proposed License Amendment Request (LAR) #252, Revision 0, and concludes it meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(c), no environmental impact statement or environmental assessment needs to be prepared in connection with this request.

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302/LICENSE NUMBER DPR-72

ATTACHMENT D

LICENSE AMENDMENT REQUEST #252, REVISION 0

ONCE THROUGH STEAM GENERATOR TUBE SURVEILLANCE PROGRAM, TUBE REPAIR ROLL (RE-ROLL) PROCESS

Proposed Revised Improved Technical Specifications Change Pages

Strikeout / Shadowed Format

~~**Strikeout Text**~~	**Indicates deleted text**
Shadowed Text	**Indicates added text**

5.6 Procedures, Programs and Manuals

5.6.2.10 OTSG Tube Surveillance Program (continued)

C-3 More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

3. The above-required inservice inspections of OTSG tubes shall be performed at the following frequencies~~ except, a one-time change for Cycle 11 is granted to modify the scheduled inspection frequency from a calendar-based interval to an interval of 21.6 months of operating time at a temperature of 500 °F or above (measured at the hot leg side). This will allow the OTSG tube inspection to coincide with Refuel Outage 11R~~:

   a. Inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under all volatile treatment (AVT) conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category, or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

   b. If the inservice inspection of an OTSG, conducted in accordance with Table 5.6.2-2 or Table 5.6.2-3 requires a third sample inspection whose results fall in Category C-3, the inspection frequency shall be reduced to at least once per 20 months. The reduction in inspection frequency shall apply until a subsequent inspection demonstrates that a third sample inspection is not required. If the C-3 inspection results classification is due to including new tubes with TEC indications that meet the criteria to remain in-service, no reduction in inspection frequency is required.

   c. Additional unscheduled inservice inspections shall be performed on each OTSG in accordance with the first sample inspection specified in Table 5.6.2-2 or Table 5.6.2-3 during the shutdown subsequent to any of the following conditions:

      1. Primary-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.12,

      2. A seismic occurrence greater than the Operating Basis Earthquake,

      3. A loss-of-coolant accident requiring actuation of the engineered safeguards, or

      4. A main steam line or feedwater line break.

(continued)

5.6 Procedures, Programs and Manuals

5.6.2.10 OTSG Tube Surveillance Program (continued)

8. Plugging/Repair Limit means the extent of pressure boundary degradation beyond which the tube shall either be removed from service by installation of plugs or the area of degradation shall be removed from service (a new pressure boundary established) using an Approved Repair Technique. The plugging/repair limit is 40% through-wall for all pressure boundary degradation.

9. 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 main steam line or feedwater line break, as specified in 5.6.2.10.3.c, above.

10. Tube Inspection means an inspection of the OTSG tube pressure boundary.

11. Approved Repair Technique means a technique, other than plugging, that has been accepted by the NRC as a methodology to remove or repair degraded or defective portions of the pressure boundary and to establish a new pressure boundary. Following are Approved Repair Techniques:

    a) Sleeve installation in accordance with the B&W process (or method) described in report BAW-2120P. No more than five thousand sleeves may be installed in each OTSG.

    b) Installation of repair rolls in the upper and lower tubesheets in accordance with ~~the Framatome Technologies Incorporated processes (or methods) described in reports~~ BAW-2303P, Revision 4~~. and BAW-2342P~~. The repair process (~~either~~ single, ~~roll~~ overlapping, or ~~double~~ multiple roll) may be performed ~~once per~~ in each tube. The repair roll area will be examined using eddy-current methods following installation. The repair roll must be free of imperfections and degradation for the repair to be considered acceptable.

(continued)

5.6 Procedures, Programs and Manuals

5.6.2.10 OTSG Tube Surveillance Program (continued)

The repair roll in each tube will be inspected during each subsequent inservice inspection while the tube with a repair roll is in service. The repair roll will be considered a specific limited area and will be excluded from the random sampling. No credit will be taken for meeting the minimum sample size.

If primary-to-secondary leakage results in a shutdown of the plant and the cause is determined to be degradation in a repair roll, 100% of the repair rolls in that OTSG shall be examined. If that inspection results in entering Category C-2 or C-3 for specific limited area inspection, as detailed in Table 5.6.2-3, 100% of the repair rolls shall be examined in the other OTSG.

12. Tube End Cracks (TEC) are those crack-like eddy current indications, circumferentially and/or axially oriented, that are within the Inconel clad region of the primary face of the upper and lower tubesheets, but do not extend into the carbon steel-to Inconel clad interface.

b. The OTSG shall be determined OPERABLE after completing the corresponding actions (plug or repair all tubes exceeding the plugging/repair limit) required by Table 5.6.2-2 (and Table 5.6.2-3 if the provisions of Specification 5.6.2.10.2.d are utilized).

~~There are a number of OTSG tubes that have the potential to exceed the tube plugging/repair limit as a result of tube end anomalies. Defective tubes will be repaired or plugged during the next outage of sufficient duration. An evaluation has been performed which confirms that operability of the CR-3 OTSGs will not be impacted with those tubes in service.~~

c. Inservice tubes with pit-like IGA indications in the "B" OTSG first span shall be monitored for growth of these indications by using a test probe equivalent to the high frequency bobbin probe used in the 1997 inspection. The indicated percentage throughwall value from the current inspection shall be compared to the indicated percentage throughwall value from the 1997 inspection.

(continued)

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302/LICENSE NUMBER DPR-72

ATTACHMENT E

LICENSE AMENDMENT REQUEST #252, REVISION 0

ONCE THROUGH STEAM GENERATOR TUBE SURVEILLANCE PROGRAM, TUBE REPAIR ROLL (RE-ROLL) PROCESS

Proposed Revised Improved Technical Specifications Change Pages

Revision Bar Format

5.6 Procedures, Programs and Manuals

5.6.2.10 OTSG Tube Surveillance Program (continued)

C-3 More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

3. The above-required inservice inspections of OTSG tubes shall be performed at the following frequencies:

   a. Inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under all volatile treatment (AVT) conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category, or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

   b. If the inservice inspection of an OTSG, conducted in accordance with Table 5.6.2-2 or Table 5.6.2-3 requires a third sample inspection whose results fall in Category C-3, the inspection frequency shall be reduced to at least once per 20 months. The reduction in inspection frequency shall apply until a subsequent inspection demonstrates that a third sample inspection is not required. If the C-3 inspection results classification is due to including new tubes with TEC indications that meet the criteria to remain in-service, no reduction in inspection frequency is required.

   c. Additional unscheduled inservice inspections shall be performed on each OTSG in accordance with the first sample inspection specified in Table 5.6.2-2 or Table 5.6.2-3 during the shutdown subsequent to any of the following conditions:

      1. Primary-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.12,

      2. A seismic occurrence greater than the Operating Basis Earthquake,

      3. A loss-of-coolant accident requiring actuation of the engineered safeguards, or

      4. A main steam line or feedwater line break.

(continued)

5.6 Procedures, Programs and Manuals

5.6.2.10 OTSG Tube Surveillance Program (continued)

8. Plugging/Repair Limit means the extent of pressure boundary degradation beyond which the tube shall either be removed from service by installation of plugs or the area of degradation shall be removed from service (a new pressure boundary established) using an Approved Repair Technique. The plugging/repair limit is 40% through-wall for all pressure boundary degradation.

9. 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 main steam line or feedwater line break, as specified in 5.6.2.10.3.c, above.

10. Tube Inspection means an inspection of the OTSG tube pressure boundary.

11. Approved Repair Technique means a technique, other than plugging, that has been accepted by the NRC as a methodology to remove or repair degraded or defective portions of the pressure boundary and to establish a new pressure boundary. Following are Approved Repair Techniques:

    a) Sleeve installation in accordance with the B&W process (or method) described in report BAW-2120P. No more than five thousand sleeves may be installed in each OTSG.

    b) Installation of repair rolls in the upper and lower tubesheets in accordance with BAW-2303P, Revision 4. The repair process (single, overlapping, or multiple roll) may be performed in each tube. The repair roll area will be examined using eddy-current methods following installation. The repair roll must be free of imperfections and degradation for the repair to be considered acceptable.

(continued)

5.6 Procedures, Programs and Manuals

5.6.1.1 OTSG Tube Surveillance Program (continued)

The repair roll in each tube will be inspected during each subsequent inservice inspection while the tube with a repair roll is in service. The repair roll will be considered a specific limited area and will be excluded from the random sampling. No credit will be taken for meeting the minimum sample size.

If primary-to-secondary leakage results in a shutdown of the plant and the cause is determined to be degradation in a repair roll, 100% of the repair rolls in that OTSG shall be examined. If that inspection results in entering Category C-2 or C-3 for specific limited area inspection, as detailed in Table 5.6.2-3, 100% of the repair rolls shall be examined in the other OTSG.

12. Tube End Cracks (TEC) are those crack-like eddy current indications, circumferentially and/or axially oriented, that are within the Inconel clad region of the primary face of the upper and lower tubesheets, but do not extend into the carbon steel-to Inconel clad interface.

b. The OTSG shall be determined OPERABLE after completing the corresponding actions (plug or repair all tubes exceeding the plugging/repair limit) required by Table 5.6.2-2 (and Table 5.6.2-3 if the provisions of Specification 5.6.2.10.2.d are utilized).

c. Inservice tubes with pit-like IGA indications in the "B" OTSG first span shall be monitored for growth of these indications by using a test probe equivalent to the high frequency bobbin probe used in the 1997 inspection. The indicated percentage throughwall value from the current inspection shall be compared to the indicated percentage throughwall value from the 1997 inspection.

(continued)

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302/LICENSE NUMBER DPR-72

ATTACHMENT F

LICENSE AMENDMENT REQUEST #252, REVISION 0

ONCE THROUGH STEAM GENERATOR TUBE SURVEILLANCE PROGRAM, TUBE REPAIR ROLL (RE-ROLL) PROCESS

NRC Questions Regarding Topical Report BAW-2303P, Revision 4

NRC Questions Regarding Topical Report BAW-2303P, Revision 4

During the review of Topical Report BAW-2303P, Revision 4, the NRC developed a set of questions. These questions were posed to two B&W Owners Group (BWOG) member utilities (Duke Energy Corporation and Entergy) who requested to use the topical report as technical justification for implementing the re-roll repair process. Provided below are the Crystal River Unit 3 (CR-3) responses to these NRC questions.

Questions to Duke Energy Corporation:

Questions specific to BAW-2303P Revision 4

1. *On page 1-1, Framatome states that eddy current inspections of OTSG tubes have resulted in the detection of indications within the upper and lower tubesheet region. Since this license amendment request seeks to remove the restriction on lower tube sheet area re-rolling that was part of the NRC approval of prior re-roll amendments, please discuss your detection of indications (if any) in lower tubesheet and characterization of the degradation.*

CR-3 Response:

Indications of volumetric intergranular attack (IGA) have been identified in the lower tubesheet crevice near the secondary face; however, all of these indications are in exclusion zones for the repair roll and therefore would not be candidates for repair roll application.

2. *On page 1-2, Framatome states that volumetric indications attributed to Intergranular Attack (IGA) have been identified in the unexpanded portion of the tube within the tubesheet crevice. Please discuss the impact of IGA on the re-rolling operation and exclusion zones.*

CR-3 Response:

The presence of IGA in a tube does not exclude the tube from being repaired by re-roll. Re-roll repair can be implemented provided the re-roll is not installed in an exclusion zone and the re-roll itself is free of indications. Based on pre-installation Eddy Current Testing (ECT) inspection, the selected re-roll area is free from indications and the re-roll area is inspected again after installation to verify that the area is free from indications. CR-3 Improved Technical Specifications (ITS) 5.6.2.10, require subsequent inservice inspections of the re-roll area to insure that the re-roll remains free of defects.

3. *On page 2-1, Framatome states that:*

"There are three overlapping roll configurations that may be installed. The compressive load is minimized by installing an inboard roll followed by an overlapping outboard roll for a total addition compressive load of 21 lbs. Installing an outboard roll, followed by an overlapping inboard roll results in a total additional compressive load of 50 lbs."

Please provide an explanation as to why the sequence of rolling, i.e., inboard roll followed by overlapping outboard roll vis-a-vis outboard roll followed by overlapping inboard roll, would result in doubling the compressive load from 21 to 50 lbs.

CR-3 Response:

If the inboard repair roll is installed first, the tube is "locked" into place resulting in 21 lbs. compressive load over the length of the tube; thus the second outboard repair roll does not impart any additional compressive load over the length of the tube. If the outboard repair roll is installed first, 21 lbs. of compressive load results, then the second, inboard roll imparts an additional 29 lbs. of compressive load over the length of the tube due to material pushed out beyond the toe transition of the second roll. The compressive loads that result from installation of the repair rolls were based on test measurements of tube elongation resulting from repair roll installation for the configurations noted above.

4. *Comparing Table 3-1 "B&W OTSG Performance Characteristics" in BAW-2303P, Rev. 4 with Table 4.1 "B&W OTSG Performance Characteristics" in BAW-2303P, Rev. 3, please discuss the deletion of 25 psia secondary side pressure for MSLB and 15 psia for primary side pressure for SBLOCA.*

CR-3 Response:

In Revision 4 of the repair roll topical report, the maximum pressure of either the primary or secondary side is conservatively taken as the primary-to-secondary pressure difference; thereby, maximizing the pressure difference. The MSLB secondary pressure of 25 psia and SBLOCA primary side pressure of 15 psia were conservatively ignored when establishing the design pressure differences.

5. *On page 4-13, Framatome states:*

"Differential dilation is a term that is used to refer to the interface between the tube OD and the tubesheet bore diameter, which allows a comparison of the relative interface of the joint for any transient condition. The differential dilation is equal to the tubesheet bore dilation (due to tubesheet bowing and free thermal growth) minus the tube dilation (due to internal pressure and free thermal growth). A positive value indicates that the increase in bore diameter is greater than the increase in the tube OD with a reduced interference within the rolled joint. A negative value indicates that the tube free expansion would be greater than the bore expansion resulting in an increase in the interference pressure of the rolled joint. The differential dilations are expressed as diametrical changes along two perpendicular axes. "Radial" refers to the dilation along the radius from OTSG center to the tube centerline and "circumferential" refers to the dilation perpendicular to the radial dilation."

In Tables 5-3 to 5-8, please clarify the physical representation of both the major and minor differential dilations having positive values. Does this mean that there is reduced (or no) contact between the tube and tubesheet at this point. Please clarify the relationship between the "major and minor" differential dilations

(Tables 5-1, 5-2, 5-3, 5-4, 5-6) and "radial and circumferential" differential dilations (Tables 4-1 and 4-2) .

CR-3 Response:

Positive differential dilations in both directions indicate the contact is reduced in both directions. "Major" and "minor" are related to the magnitude of the dilations, with the larger dilation (larger reduction in contact) referred to as the major dilation. "Radial" and "circumferential" relate to the direction of the dilation in the generator. "Radial" dilation refers to the dilation along the radius from the SG center to the tube centerline. The "circumferential" dilation refers to the direction perpendicular to the radial dilation. The major dilation may be in either the radial or circumferential direction depending on the location of the tube in the SG. Regarding qualification of the repair roll, the magnitude of the dilations is evaluated (referenced as major and minor) with the direction of the dilation having no impact on the results.

6. *On page 5-1, Framatome states that:*

"However, the lower tubesheet crevice is known to contain solid particles in the sludge that collects in this region. Previous testing had demonstrated that leak rates are much higher for repair rolls without crevice deposits. Therefore, leak tests performed without crevice deposits provided conservative leak rates for upper tubesheet and lower tubesheet repair rolls. In addition, previous testing has shown that the joint strength is higher for rolled joints with deposits. Therefore, testing without crevice deposits is conservative for both leakage and structural integrity."

This license amendment request seeks to remove the restriction on lower tube sheet area re-rolling. The NRC staff has noted in a RAI for a previous re-roll submittal that operating experience with re-rolled tubes in other PWRs indicates that crevice deposits may be a significant contributor to a reduction in the leakage integrity of re-rolled tubes. Please discuss the basis for the assumption of superior leakage integrity and joint strength for repair rolls with crevice deposits. Provide the results of the previous testing cited in the above discussion. Include a discussion of any actual inspection results of pulled tubes and/or upper tubesheet crevice areas.

CR-3 Response:

Please see the response for Question 7 below, and the revised response provided in Page 13 of this Attachment.

7. *On page 5-1, Framatome states that:*

"Previous testing has shown that cyclic loading associated with normal operating and steam generator transient conditions does not degrade the integrity of the repair roll. Cyclic loading has been shown to result in higher joint strength for both high yield and low yield tubing. Previous repair roll leak test resulted in higher leakage for test samples without deposits that were not subjected to cyclic loading prior to testing than for samples with deposits that were subjected to cyclic loading prior to testing. Therefore, all leak and load testing to support this qualification of the repair was conservatively performed on samples that were not subjected to cyclic loading."

Please discuss the basis for the assumption of superior leakage integrity and joint strength for repair rolls subjected to cyclic loading. Provide the results of previous testing cited in the above discussion.

CR-3 Response:

The test configuration was selected based on leak and load tests performed using the same roll installation process as that used for the currently operating OTSGs.

Leak test data from testing conducted in 1999 was evaluated that included samples with and without crevice deposits, pre-fatigue and post-fatigue. The test results clearly show that for the OTSG installation process, a clean crevice leaks more than a packed crevice, both in the pre-fatigue and post-fatigue cases. The resulting leak rate from the clean crevice, pre-fatigue samples was 11 times greater than the leak rate from the packed crevice samples (with or without fatigue). The decreased leakage for the packed crevice is attributed to sludge providing a partial seal between the tube and tubesheet that would be an otherwise open flow path in a clean crevice.

An evaluation of joint strength test data from 1999 from testing performed on clean crevice samples for pre-fatigue and post fatigue conditions, and packed crevice samples for post-fatigue conditions showed a maximum of 10% difference in joint strength for the tested conditions. For the configurations tested, the results showed that the pre-fatigue, clean crevice sample resulted in the minimum joint strength.

Since the current qualification allows the joint to slip, qualification of the repair roll is based primarily on leakage with joint strength as a secondary factor. Therefore, the test configuration (clean crevice) was selected which resulted in the highest leakage. The test configuration results in conservative leak rates for the lower tubesheet (LTS) and bounding leak rates for the upper tubesheet (UTS). In addition, the leak rates are applied very conservatively by assuming a 360°, 100% through-wall (TW) circumferential crack at the heel transition (primary side) of every repair roll and taking no credit for the seal weld.

8. *On page 9-1, Framatome states that:*

"Two single 1-inch repair rolls or any overlapping repair roll that results in a maximum of 50 lbs additional compressive load may be installed at a qualified location in any one tube. Additional repair rolls may be installed on a case-by-case basis by evaluating for acceptable compressive tube loads."

The license amendment request seeks to remove the limitation of only one reroll per steam generator (SG) tube. Please discuss how to evaluate acceptable maximum compressive tube loads, to determine how many additional repair rolls may be installed in a single tube.

CR-3 Response:

CR-3 evaluated the worst case compressive tube load, which occurs in the periphery during heat-up, and allowed the compressive load on that tube to increase an additional 50 lbs. This is not a significant increase compared to the compressive load due to the transient and therefore was concluded to be acceptable.

Compressive loads in the center of the SG are less than the compressive loads in the periphery and there is no reason to limit the tube load at these locations to less than that allowed in the periphery. Therefore, additional repair rolls could be installed in the center of the SG as long as the total resulting load is less than the load in the worst case periphery tube. Additional repair rolls in the center would be evaluated on a case-by-case basis, depending on the transient load for that tube. Florida Power Corporation has decided to limit repair rolls at CR-3 to a configuration resulting in a maximum of 50 lbs. additional compressive load for any location.

9. *What is the Finite Element (FE) computer code used for analyzing the general structural behavior of the OTSG during the various operating and accident transients? Describe in detail how the computer code was validated, benchmarked, or compared.*

CR-3 Response:

Version 5.4 of the "ANSYS" finite element software code was used for determining the OTSG tubesheet-to-tube differential dilations and tube loads.

Per FTI procedures, computer programs such as ANSYS must be independently verified as performing properly. To independently demonstrate the correct execution of the ANSYS program, several small problems are executed using the same software, software features, and hardware that were used in the final solution runs. These types of problems are chosen to confirm behavior of the same types of elements and loads as are employed in the transient/structural analysis of the OTSG.

In addition, the resulting average shell temperature and tube-to-shell ΔT from the ANSYS thermal analysis were compared to those resulting from the detailed thermal hydraulic analysis of the accident transients (RELAP results).

There is no actual OTSG tube load or dilation experimental data to benchmark against.

10. *In Section 4.1 related to the development of the finite element (FE) model of the BAW-2303P, Revision 4, it is stated that the vertical support provided to the tube support plate (TSP) by the tie-rods and spacers was not included in the model. Describe how the 42 tie-rods and spacers are distributed and connected to the upper and lower tubesheets. For the case of MSLB transient at Oconee Units 1, 2, and 3, in the absence of actual representation of the tie-rods and spacers in the FE model, provide the basis for your description of the behavior of the tie-rods and spacers under the MSLB dynamic and thermal conditions, and their effects on the tubesheets in general. Specifically, discuss the predicted effects of these tierods and spacers on the tubesheets in the local area surrounding the tie-rods and spacers for the determination of maximum tube load, differential dilations (radial and circumferential) and, therefore, on the determination of the repair roll exclusion zones.*

CR-3 Response:

Typically, the tie-rods are distributed with 6 rods (3 pairs equally spaced in the circumferential direction) near the center of the tubesheet (and tube support plates, TSP's) @ R =4.5"; 12 rods (6 pairs equally spaced in the circumferential direction) at an intermediate location with an approximate radius of 23 inches, and 24 rods (12 pairs equally spaced in the circumferential direction) toward the outer region at an approximate radius of 44 inches.

The 42 tie-rods (5/8 inch diameter solid rods) are anchored at the lower tubesheet primary face by a ¼ inch minimum thickness plug weld. Each tie-rod extends from the lower tubesheet upward through 15 sets of tube support plates and associated vertical support spacer sleeves. The upper end of the tie-rod is threaded and is anchored to the 15th tube support plate by a nut/washer combination. The overall length of the tie-rod is approximately 50 ft. 6 inches and thus does not reach (or interface with) the upper tubesheet.

For an event such as a MSLB, the dynamic effects of the break occur relatively quickly. For example, the mechanical loads occurring due to the initial pressure wave occur in less than the first second. Subsequently, the thermal response of the OTSG shell and tube material occurs over an even greater time span, which typically takes several minutes to reach its most severe loading. Because of this time-phasing of transient effects, the dynamic loads and thermal response loads do not occur simultaneously. Therefore, the thermal/pressure transient analysis performed to determine the maximum axial tube loads and associated hole dilations is not impacted by the presence of the tie-rods.

Moreover, the collective axial stiffness of the 42 tie-rods (5/8 inches diameter x 50.5 ft. long) is relatively very low. Thus, even if the tie-rods were to become loaded, they are not structurally capable of imparting a load significant enough to affect the flexure of the tubesheet (i.e., dilations are not affected).

For these reasons, the tie-rods were not included in the transient analysis for determining tube load and hole dilations.

QUESTIONS DISCUSSED DURING THE NOVEMBER 1, 2000 CONFERENCE CALL

We understand thermal conduction in the tubesheet region for a steam line break was calculated with the following modeling:

- *Two-dimensional nodalization was used as shown in Figure 4-1 with no variation in the horizontal plane (which would provide a 3 dimensional nodalization).*

- *RELAP 5 provided primary and secondary side fluid conditions and the resulting temperatures and film coefficients were input as boundary conditions for the conduction calculation.*

- *Tube to tube sheet heat transfer was calculated with a constant conductance between the tube outer diameter and the tube sheet.*

We further understand that maximum calculated tube sheet temperature non-uniformity between the tubes and the center of the ligaments was about 6°F.

We have the following questions:

1. *Are the above understandings correct?*

CR-3 Response:

Yes, the above understandings are correct with the following clarifications:

Statement #2:
The tubesheet temperatures were imposed as the primary fluid temperature. See further discussion below under Question #2.

Statement #3:
Rather than a "constant" conductance, a more applicable description is "an isothermal condition between the tube and tubesheet."

Sentence following statement #3 above:
A conservative assessment of the bounding MSLB transient was performed to determine the temperature difference between the tube and adjacent tubesheet ligament. At the time of maximum tube load and dilations (~605 seconds), the temperature difference was 0°F in the rolled portion of the tube and 15°F in the unrolled portion. Additional discussion on this issue is provided under Question #2 below.

2. *The tube to tubesheet configuration is illustrated in Figure 2-1 of Topical Report BAW-2303P, Revision 4, where the tube is in intimate contact with the tubesheet over an inch or so as shown at the top of the figure, but the remaining almost two feet is separated by a gap that will provide poor heat transfer in comparison to the intimate contact region. What is the effect of this configuration in comparison to the configuration you assumed.*

CR-3 Response:

The actual configuration, including the gap between the tube and tubesheet bore, was considered in the detailed transient analysis. As stated above, an assessment of the tube-to-tubesheet bore interface was performed. Based on that assessment, it was concluded that it was conservative to assume that the perforated portion of the tubesheet follows the temperature of the primary fluid that passes through it. The effect of the gap between the tube and tubesheet was considered in the analysis and was determined to have a negligible effect on the temperature of the tubesheet at the time of interest for the MSLB transient. Therefore, the model was simplified to assign the temperature of the primary fluid to the entire tubesheet. Additional substantiation of this method is provided below.

The temperature profile of the tubesheet is a function of four interdependent factors:

1) The relative heat transfer of the tubesheet surface area and the tube bore surface area,

2) The radial gradient between the tube wall to tubesheet ligaments (horizontal direction),

3) The change in primary fluid temperature through the thickness of the tubesheet (vertical direction), and

4) The thermal gradient through the tubesheet thickness (vertical direction).

Relative Heat Transfer Areas

Due to the massive surface area of the tubesheet bores, the temperature of the fluid within the tubes controls the temperature of the tubesheet. The temperature of the fluid/steam on the primary and secondary faces of the tubesheet does not contribute significantly to the temperature of the tubesheet. A simple calculation (approximate values) of the relative heat transfer areas shows that the surface area of the bores is 60 times greater than the surface area of the tubesheet face. (Dimensions are provided in inches.)

Tubesheet surface area = $\pi(58^2) - (\pi/4)(0.635^2)15500 = 5660\ in^2$, percent of area on tubesheet face = 1.5% or 0.75% per face, percent of area within tubesheet bore = 98.5%, ratio of tube bore area to tubesheet face area = 66 to 1

Based on the relative heat transfer areas shown above, the radial heat transfer of the fluid within the tube is the controlling factor for the temperature profile of the tubesheet. Specifically, the temperature of the tubesheet ligaments is a function of the heat transfer of the primary fluid to the tube ID, conductance across the tube wall, conductance across the nominal 0.005 annulus between the tube OD and the tubesheet bore ID (for unrolled sections only), and conductance of the tubesheet ligaments. Further discussion of the temperature profile between the tube and tubesheet ligaments follows.

Radial Thermal Gradient Between the Tube and Tubesheet Ligament

A detailed one-dimensional heat transfer analysis of both the rolled and unrolled portions of the tube was performed to determine the radial gradient between the tube and the tubesheet ligament. The evaluation of the unrolled portion of the tube conservatively assumed a steam-filled 0.005-inch gap between the tube OD

and the tubesheet bore. The analysis was conservative in that it did not credit any radiant heat transfer across the gap, did not credit any condensation that may take place, and also did not credit any convection of the steam within the annulus. All of these processes would increase the heat transfer and reduce the temperature lag between the tubesheet and the tube (or bulk fluid temperature).

The results of the bounding Oconee MSLB time history evaluation show that the maximum radial gradient between the tube wall and the center of the tubesheet ligament occur early in the MSLB transient (first 100 seconds) as shown in the following table.

RADIAL TEMPERATURE GRADIENTS BETWEEN THE TUBE AND TUBESHEET LIGAMENT

Time	Tube to Tubesheet Ligament Gradient of 0°		Comment
0 - 100 sec	Rolled Portion	5°	Early in Transient, No Significant Tube Load
	Unrolled Portion	80°	
605 sec	Rolled Portion	0°	Time of Maximum Tube Thermal Load
	Unrolled Portion	15°	

The difference in the radial gradient between the rolled portion of the tube and the unrolled portion of the tube produces a through-thickness gradient in the tubesheet which is discussed further below.

Primary Fluid Temperature Change through the Tubesheet Thickness

Since the tube length within the tubesheets is not directly exposed to the secondary side fluid temperature, the change in primary fluid temperature through the relatively short distance of the tubesheet thickness (24inches) is negligible. Based on NRC questions regarding the fluid temperature gradient through the tubesheet thickness, FTI developed a detailed model to quantify the maximum change in temperature as the primary fluid passes through the tubesheet. These simulations showed a change in fluid temperature of less than 2°F from the primary face to the secondary face at any time during the transient. Therefore, the change in the temperature of the primary fluid through the thickness of the tubesheet has a negligible effect on the temperature profile of the tubesheet.

Thermal Gradient through the Thickness of the Tubesheet

As stated above, the primary fluid contained within the tubes of the tubesheet controls the temperature profile of the tubesheet. Since there is very little temperature variation of the primary fluid within the tubesheet, any temperature gradient across the thickness of the tubesheet is equal to the difference in radial heat transfer between the rolled and unrolled portions of the tubes. As provided in the preceding discussion, the radial gradient in the rolled portion of the tube is less than 5°F throughout the MSLB transient. In the unrolled portion of the tube, the radial gradient is approximately 80°F at the beginning of the MSLB transient and only 15°F at the time of maximum tube load and dilation. Thus, the majority of the tubesheet thickness (23 of 24 inches) will be at the uniform temperature associated with the unrolled portion of the tube and only a very small length (1 inch) will have a temperature associated with the rolled portion of the tube. This produces a through-thickness gradient of 80°F early in the transient and 15°F at the time of maximum tube loads with the primary face being cooler than the secondary face in both cases (for the upper and the lower tubesheets).

To provide quantitative support for the uniform temperature applied in the transient analysis, an assessment of the effects of through-thickness gradients on the previously calculated tube loads and dilations has been performed. The FE analysis model described in Section 4.0 of the topical report was used and solved for the gradient conditions at the time points mentioned above for the Oconee MSLB transient. The model was solved twice:

1) utilizing the engineering judgement that the tubesheet metal is at primary fluid temperature (i.e., no through-thickness gradients) and

2) adding the through-thickness tubesheet gradients to the model (i.e., as an alternative to the engineering judgement of assigning tubesheet temperatures).

The key results of the two solutions, axial tube load and tube hole dilations, were compared to isolate the effect of the tubesheet through-thickness gradients. A summary of the comparison is shown in the following table.

**Comparison Of Bounding Oconee MSLB Tube Loads And Dilations
As Analyzed (No Tubesheet Gradient) Case Versus Tubesheet Gradient Case**

Transient Time Point	Description	Through-Thickness Gradient Modeled	% Difference in Max. Axial Tube Load	% Difference in Max. Bore Dilation
1 minute	Time of max. through-thickness gradient	80°F 0°F[1]	-0.8% (including gradient reduces max. load)	-10% (including gradient reduces max. dilation)
10 minutes	Time of max. tube load/dilation	15°F 0°F[1]	-0.1% (including gradient reduces max. load)	-3% (including gradient reduces dilation)
Note 1: The analysis assigns the tubesheets a uniform temperature equal to the temperature of the contained primary fluid				

The results of the comparative analysis indicate that both the tube axial loads and the dilations are smaller when the through-thickness gradients are applied. The results also show that the gradients have only a minor effect on the maximum tube loads and dilations. Therefore, the assumption that the tubesheet temperature equals the primary fluid temperature, which was used in determining tube loads and dilations for the repair roll topical report, is not only valid, but also conservative.

3. *There are a number of "inert" tubes because they have been plugged. What is the effect of these tubes on your tubesheet response calculations?*

CR-3 Response:

Plugged tubes result in a global effect on the general behavior of the tubesheet and a local effect on the difference in temperature between the tube and adjacent tubesheet ligament.

The global effect of plugged tubes on the general behavior of the tubesheet was directly evaluated as part of the tube load analysis. As stated in Section 4.7 of the topical report, two different cases of 25% plugged tubes and a 0% plugging case were considered for the general behavior of the tubesheet and the resulting tube loads and dilations. The loads and dilations used for qualification of the repair roll reflect the bounding plugging conditions.

The local effects of plugging were not directly evaluated in the finite element analysis. During the MSLB transient, plugged tubes remain hot while unplugged tubes are rapidly cooled. Thus, plugged tubes could result in a slight increase in the temperature difference between the tube and the tubesheet ligament, thus increasing the tube-to-tubesheet differential dilations and decreasing the interference in the repair roll joint. For tubes with repair rolls that are adjacent to only a couple of plugged tubes, the effect is concluded to be minimal; while the maximum effect occurs for a repair roll that is surrounded by plugged tubes. The

periphery in the Oconee OTSGs would be most influenced by the local effects of adjacent plugs due to the density of plugging in this area.

Effect on Leak Rates

Test results show that leakage is driven primarily by the pressure differential and varies minimally as a function of differential dilations and tube loads. Therefore, a slight increase in differential dilations at any location would have a negligible effect on the leak rates applied to the repair rolls.

As noted previously, the applied leak rates are very conservative. The leak rates assume a 360°, 100% through-wall circumferential flaw at the heel transition of each repair roll and no credit is taken for the original roll joint or the tube-to-tubesheet weld. Since most of the degradation in the roll transitions is small, axial crack-like indications attributed to Primary Water Stress Corrosion Cracking (PWSCC), this is a very conservative approach. In the same manner, all tubes with an axial load in excess of the tested joint strength load are assumed to slip and a post-slip leak rate is applied without taking credit for the original roll or the tube-to-tubesheet weld. The tube loads and dilations are based on a conservative thermal hydraulic analysis that maximizes feedwater flow, minimizes feedwater temperatures, and does not isolate feedwater to the broken steam generator. Finally, the leak rates were evaluated at the maximum pressure differential, though the maximum pressure differential does not occur until late in the transient after the time of maximum tube loads and dilations and assumes no operator intervention. The pressure differential through most of the transient is less than 1200 psi. The bounding leak rates are conservatively applied to repair rolls in both OTSGs. The conservatism described above bounds any increase in leak rates due to slightly increased differential dilations due to the local effects of plugged tubes.

Therefore, the local effects of plugged tubes do not impact the results and conclusions of the repair roll qualification.

4. *We have no information on RELAP 5 nodalization. Typically, tubes are modeled in RELAP 5 with nodes that are several feet long. Is that the case here? If so, is that a fine enough nodalization to predict tube sheet behavior?*

   Your response should address the thermal expansion effects in conjunction with tube sheet response to pressure difference.

CR-3 Response:

FTI does not use RELAP 5 to predict the structural behavior of the tubesheet. As described in response to question #2, the primary fluid temperature was assigned to the tubesheet, T_{HOT} for the upper tubesheet and T_{COLD} for the lower tubesheet. The primary fluid temperatures were taken directly from the RELAP 5 analysis.

Based on NRC questions regarding the primary fluid temperature gradient through the tubesheet thickness, FTI developed a detailed model to quantify the maximum fluid temperature change as the fluid passes through the tubesheet. These simulations showed a temperature change of less than 2°F at any time during the transient as discussed in response to question #2.

Revision To The Response To Question #7 Contained In The Duke Letter To The NRC Dated October 26, 2000

Additional information on question #7 (includes response to question #6).

On page 5-1, Framatome states that:

"Previous testing has shown that cyclic loading associated with normal operating and steam generator transient conditions does not degrade the integrity of the repair roll. Cyclic loading has been shown to result in higher joint strength for both high yield and low yield tubing. Previous repair roll leak test resulted in higher leakage for test samples without deposits that were not subjected to cyclic loading prior to testing than for sample with deposits that were subjected to cyclic loading prior to testing. Therefore, all leak and load testing to support this qualification of the repair was conservatively performed on samples that were not subjected to cyclic loading."

Please discuss the basis for the assumption of superior leakage integrity and joint strength for repair rolls subjected to cyclic loading. Provide the results of previous testing cited in the above discussion.

CR-3 Response:

The test configuration was selected based on leak and load tests that were performed using the same repair roll installation process as that currently used for the OTSGs. A review of the leak test data and joint strength data is provided below.

Leak Test Data

An evaluation was performed of leak test data from testing conducted in 1999 that included samples with and without crevice deposits, pre-fatigue and post-fatigue. The test results showed that for the OTSG repair roll installation process, a clean crevice leaks more than a packed crevice, both in the prefatigue and post-fatigue cases. The resulting leak rate from the clean crevice, pre-fatigue samples was an order of magnitude greater than the leak rate from the packed crevice samples (with or without fatigue). (See data table that follows.) The decreased leakage for the packed crevice is attributed to sludge providing a partial seal between the tube and tubesheet that would be an open flow path in a clean crevice.

FTI Tube Hole Dilation Leak Test Summary (test pressure 2580 psi)

Average Leak Rate Without Crevice Deposits Dilated in^3/hr		Average Leak Rate With Crevice Deposits Dilated in^3/hr	
Pre-Fatigue	Post-Fatigue	Pre-Fatigue	Post-Fatigue
0.3152	**0.0851**	**0.0094**	**0.0288**

Joint Strength Data

An evaluation of joint strength test data from 1999 from testing performed on clean crevice samples for pre-fatigue and post-fatigue conditions and packed crevice samples for post-fatigue conditions showed a maximum of 10% difference in joint strength for the tested conditions. For the configurations tested, the results showed that the pre-fatigue, clean crevice sample resulted in the minimum joint strength. This is also the configuration that resulted in the maximum leakage.

FTI Summary of Average Repair Roll Joint Strength

Condition	Average Joint Strength (lbs.)
Post-Fatigue, Dilated, Without Deposits	4785
Post-Fatigue, Dilated, With Deposits	4408
Pre-Fatigue, No Dilations, Without Deposits	4296

Qualification of the repair roll is based primarily on leakage, with joint strength as a secondary factor. Therefore, the test configuration (clean crevice, pre-fatigue) was selected that resulted in the highest leakage. The test configuration results in conservative leak rates for the lower tubesheet (LTS) and bounding leak rates for the upper tubesheet (UTS). The leak rates are applied very conservatively by assuming a 360°, 100% TW circumferential crack at the heel transition of every repair roll and taking no credit for any tube-to-tubesheet weld. Additional conservatism results from leak testing at the maximum pressure differential (2575 psi), which occurs late in the MSLB transient after the time of maximum tube loads and dilations. The pressure differential during most of the transient is less than 1200 psi. The conservative leak rate is applied to repair rolls in both generators.