September 10, 2001

Mr. Dale E. Young, Vice President
Crystal River Nuclear Plant (NA1B)
ATTN: Supervisor, Licensing & Regulatory Programs
15760 W. Power Line Street
Crystal River, Florida 34428-6708

SUBJECT: CRYSTAL RIVER UNIT 3 - ISSUANCE OF AMENDMENT REGARDING REROLL REPAIR FOR ONCE-THROUGH STEAM GENERATOR TUBING (TAC NO. MB1519)

Dear Mr. Young:

The Commission has issued the enclosed Amendment No. 198 to Facility Operating License No. DPR-72 for Crystal River Unit 3 (CR-3). The amendment consists of changes to the existing Technical Specifications (TS) in response to your letter dated March 21, 2001, as supplemented June 28, 2001. Florida Power Corporation (FPC) submitted a request for changes to the CR-3 Improved Technical Specifications (ITS) to implement a reroll process to repair degraded steam generator tubes and allow the reroll repairs to be used in both the upper and lower tubesheets.

On the basis of submitted information, the U.S. Nuclear Regulatory Commission (NRC) concludes that the proposed TS changes regarding reroll repair for degraded roll joints in the CR-3 steam generators are acceptable because FPC has (1) demonstrated through an acceptable qualification program that the reroll satisfies Title 10 *Code of Federal Regulations* Part 50, Appendix A, General Design Criterion-14, and conforms to the guidance in Regulatory Guide 1.121; and (2) agreed to regulatory commitments providing to report to the NRC circumferential flaws detected in the rerolled steam generator tubes. FPC provided these regulatory commitments in your letter to the NRC dated June 28, 2001.

A copy of the Safety Evaluation is enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/
John M. Goshen, Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-302

Enclosures:
1. Amendment No.   to DPR-72
2. Safety Evaluation

cc w/enclosures: See next page

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ACCESSION NO. ML012270329 * see previous concurrence

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DATE	09/06/01	09/06/01	09/06/01	08 /29 /01		

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FLORIDA POWER CORPORATION
CITY OF ALACHUA
CITY OF BUSHNELL
CITY OF GAINESVILLE
CITY OF KISSIMMEE
CITY OF LEESBURG
CITY OF NEW SMYRNA BEACH AND UTILITIES COMMISSION,
CITY OF NEW SMYRNA BEACH
CITY OF OCALA
ORLANDO UTILITIES COMMISSION AND CITY OF ORLANDO
SEMINOLE ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-302

CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 198
License No. DPR-72

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Florida Power Corporation, et al. (the licensees) dated March 21, 2001, as supplemented June 28, 2001, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-72 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 198, are hereby incorporated in the license. Florida Power Corporation shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Richard P. Correia, Chief, Section 2
Project Directorate II
Division of Project Licensing Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the
Technical Specifications

Date of Issuance: September 10, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 198

FACILITY OPERATING LICENSE NO. DPR-72

DOCKET NO. 50-302

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

Remove	Insert
5.0-15	5.0-15
5.0-17	5.0-17
5.0-17A	5.0-17A

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 198 TO FACILITY OPERATING LICENSE NO. DPR-72

FLORIDA POWER CORPORATION, ET AL.

CRYSTAL RIVER UNIT NO. 3 NUCLEAR GENERATING PLANT

DOCKET NO. 50-302

1.0 INTRODUCTION

By letter dated March 21, 2001, as supplemented by letter dated June 28, 2001, Florida Power Corporation (FPC), submitted a request for changes to the Crystal River Unit 3 (CR-3) Improved Technical Specifications (ITS). The requested changes would revise the CR-3 ITS by allowing the reroll repair process for steam generator (SG) tubes to be used multiple times for a single tube and would allow such tube repairs in the upper and lower tubesheet regions. The proposed reroll methodology for CR-3 is similar to the reroll repair methodology that the Nuclear Regulatory Commission (NRC) has approved for the Oconee and Arkansas Nuclear One, Unit 1 nuclear power plants.

By License Amendment No. 180 dated June 28, 1999, the NRC approved the use of repair rolls in the upper tubesheet of the CR-3 SGs, as analyzed in Babcock & Wilcox Owners Group (B&WOG) Topical Report BAW-2303P, Revision 3, “Once Through Steam Generator Repair Roll Qualification Report,” and Topical Report, BAW-2342P, “Once Through Steam Generator Repair Roll Qualification Report Addendum A.” BAW-2342P 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. The SBLOCA loads could be more limiting than those from a main steam line break (MSLB). In addition, the MSLB transient has been re-analyzed, resulting in a new set of design loads. The B&WOG requested Framatome Technologies, Inc. (FTI) to re-qualify the reroll process using the limiting load for each B&W plant and submitted to the NRC Topical Report BAW-2303P, Revision 4. The report contains technical bases for the CR-3 license amendment request for repair rolls to be installed in both the upper and lower tubesheets and for multiple repair rolls in a single tube. The analysis in BAW-2303P, Revision 4, demonstrates that it is acceptable for a tube that has been repaired with a roll to slip under faulted conditions (but such a tube is not projected to slip under normal operating conditions), which constitutes a change in design criteria compared to the original evaluation. Repair rolls that have been installed under BAW-2303P, Revision 3, remain acceptable based on the criteria contained in BAW-2303P, Revision 4. (See Section 3.1 of this safety evaluation.)

By evaluating reroll repairs only for MSLB and SBLOCA faulted conditions, BAW-2303P, Revision 4, implicitly credits the results of Topical Report BAW-2374, Revision 0, “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," which was submitted to the NRC by B&WOG by letter dated July 7, 2000. BAW-2374, Revision 0, provides the risk-informed bases for excluding the Large Break Loss of Coolant Accident (LBLOCA) from some design considerations. BAW-2374 explains that rerolled tubes may slip in the tubesheet during some LBLOCA scenarios if there is degradation (such as circumferential cracking) in the tube that prevents the tube seal weld from carrying the axial load that results from the event. By letter dated November 27, 2000, the B&WOG provided additional information related to
BAW-2374, Revision 0.

The NRC has not approved BAW-2374, Revision 0. However, based on the risk-informed arguments presented in BAW-2374, Revision 0, the NRC accepts that the reroll repairs at CR-3 may slip during an LBLOCA, resulting in an increase in leakage past the reroll. CR-3 has included in their application a regulatory commitment to demonstrate that the expected leakage following an LBLOCA is acceptable, based on the as-found condition of their SGs. Section 3.4 of this safety evaluation (SE) contains the NRC's evaluation of the risk-informed arguments presented in BAW-2374, Revision 0. On March 12, 2001, the B&WOG submitted BAW-2374, Revision 1, which is currently under staff review.

2.0 BACKGROUND

CR-3 has two model 177FA SGs manufactured by B&W. The SG tubes were fabricated from mill-annealed Alloy 600 material and were restrained by the roll expansion joints in the upper and lower tubesheets. The original tube-to-tubesheet rolls were expanded by a hardroll process and are about 1 inch in axial length extended into the upper or lower tubesheet from the tube end. The upper and lower tubesheets are about 24 inches thick, and a seal weld at the primary face of each tubesheet prevents primary-to-secondary leakage around the hardroll expansions.

Title 10, *Code of Federal Regulations* (10 CFR), Part 50, Appendix A, General Design Criteria (GDC)-14 requires that the reactor coolant pressure boundary be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. A significant portion of the reactor coolant pressure boundary is maintained by SG tubes that have experienced various levels of degradation. Draft NRC Regulatory Guide (RG) 1.121 provides guidance on an acceptable method for establishing the limiting conditions of tube degradation. In addition, the TS require periodic inspections of SG tubes. The TS also require that those tubes with defects in excess of the repair limits (e.g., flaws equal to or greater than 40 percent through-wall of tube wall thickness) be repaired or removed from service.

The original joint between the tube and tubesheet is an interference fit constructed by roll expanding the tube into the bore of the tubesheet, followed by a seal weld at the primary face of the tubesheet. The tube-to-tubesheet roll joint provides sufficient strength to maintain adequate structural and pressure boundary integrity.

Industry experience has shown that defects have developed in the tube-to-tubesheet roll joints as a result of various degradation processes. In general, tubes with degraded roll joints are either removed from service or repaired. The NRC has accepted alternate repair criteria allowing repaired tubes with degraded roll joints to remain in service provided that the repaired tubes can maintain adequate structural and leakage integrity under loadings from normal operation, anticipated operational occurrence, and postulated accident conditions. Such roll joints are said to be "qualified."

RG 1.121 recommends that the margin of safety for tube rupture under normal operating conditions be equal to or greater than 3 at any tube location where defects have been detected. For postulated accidents, RG 1.121 recommends that the margin of safety against tube rupture be consistent with the margin of safety determined by the stress limits specified in American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section III NB-3225. Structural loads imposed on the tube-to-tubesheet roll under normal operating conditions primarily result from the differential pressure between the primary and secondary sides of the tubes. Cyclic loadings from transients (e.g., startup/shutdown) were also considered in the qualification of the roll joints.

3.0 EVALUATION

3.1 Qualification Program

In support of License Amendment No. 180, FPC performed a qualification program presented in BAW-2303P, Revision 3, that demonstrated that the strength of the roll joints was satisfactory in accordance with RG 1.121. FPC applied loads to sample tubes to simulate or exceed normal, thermal and pressure cycling transients, and postulated accident conditions. In accordance with RG 1.121, room temperature hydrostatic pressure tests were performed at a pressure exceeding 3 times normal operating pressure and 1.43 times MSLB pressure. The purpose of this test was to look for gross leakage or structural failure of the joints. No mechanical change or gross leakage in the samples was noted. The original analysis in BAW-2303P, Revision 3, had assumed no joint slippage as the design basis for rerolls.

For the current license amendment request, the licensee developed a qualification program presented in Revision 4 of the BAW-2303P to demonstrate repair roll joint integrity through slip and leak tests. The program consisted of (1) establishing tube loads for the qualification tests, (2) preparing a mockup to simulate tubesheet conditions for qualification tests, and (3) performing verification tests and analyses.

In the qualification program, the licensee considered the impact of tubesheet bowing on the roll joints because the tubesheet bore diameter can change during certain operating conditions. The combined effects of primary-to-secondary pressure differential and thermal loads may cause the tubesheet to bow in one direction or the other, which can lead the tubesheet bore to dilate or shrink. When the tubesheet bore is dilated, the contact stress between the roll joint and the tubesheet would decrease and, thereby, reduce the pullout resistance of the roll joint. The resulting bowing effect can produce a dilation of the tubesheet bore in the region of the tube-to-tubesheet joint, which may reduce the load carrying capability of the rolled joint. Various dilations were included in the test block to evaluate this behavior.
FPC constructed a mockup that consisted of perforated cruciform metal blocks, which allowed simulation of tubesheet bore dilations by applying a biaxial load to the block. Alloy 600 tube samples were inserted into the block that simulated the tube-to-tubesheet configuration in the field. The tubes were expanded into the tubesheet using an expanding tool that had the same critical dimensions as the tool used in the field. The repair roll design and installation for

Revision 4 of BAW-2303P is the same process as that described in Revision 3. To obtain conservative leakage results, the sample tubes were roll expanded using a spacer such that there was no heel transition in the tested repair roll. By removing the heel transition, the tested condition represented a complete circumferential sever at the end of the effective roll (primary side). After tube installation, the blocks were subjected to various heat-up and cooldown thermal cycles.

FPC performed testing with a clean crevice between the outside diameter of the tube samples and the tubesheet bore. A clean crevice in the mockup was determined to be conservative on the basis of a proprietary analysis conducted in 1999 using the same repair roll installation process as that currently used for the SGs. To justify the reroll repair in the lower tubesheet region, FPC evaluated the effects of crevice deposits on leakage and joint strength. Leak tests were performed for samples with and without crevice deposits, pre-fatigue and post-fatigue, using a representative material in the crevice deposits. The leak tests showed that for the once through steam generator (OTSG) repair roll installation process, a clean crevice leaks more than a packed crevice, both in the pre-fatigue and post-fatigue cases. 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. Similarly, the joint strength test results showed that the pre-fatigue, clean crevice sample resulted in the minimum joint strength. Based on 1999 test results provided by the licensee, the NRC found that the test plans with the clean crevice in the post-fatigue case, as described in Revision 4 of BAW-2303P, would bound the leakage analysis and that lower tubesheet area rerolling is acceptable.

To quantify leak rates for repair rolls subjected to accident conditions, applicable tubesheet bore dilations were achieved with representative pressures adjusted for uncertainties. The maximum pressure differential provides a bounding leak rate for all transients. The tube end was sealed so that the leak path was through the repair roll. To obtain conservative leakage results, the sample tubes were roll expanded using a spacer such that there was no heel transition in the tested repair roll. Leak tests were performed at room temperature. Room temperature leak tests are expected to be conservative based on higher temperatures increasing the joint tightness due to thermal expansion differences between the Inconel 600 tubes and the carbon steel tubesheet.

The licensee applied an axial load to the tube specimen in the test block with various simulated tubesheet bore dilation to verify that the repair roll could withstand anticipated axial loads during normal operation and accident conditions. A full circumferential severed tube was modeled for the testing, which is conservative for structural and leakage integrity because the majority of the degradation within the tubesheet is from short, axial cracks. The testing assesses the joint strength of a repair roll without taking any credit for the original roll expansion or the tube-to-tubesheet weld. Tube movement was monitored during the test and verified by measuring the depth of the tube end after each test.

On the basis of its qualification program, the licensee established that either a single or double roll repair will carry all structural loads and minimize potential leakage. For a double roll, a second repair roll is installed that overlaps a single repair roll. Both the single and double repair rolls may be installed in the upper tubesheet or lower tubesheet. The need to use a double roll depends on the location of the tube within the tube bundle. Using a double roll increases the joint strength because of the longer area of tube-to-tubesheet contact and the increased joint strength will accommodate larger applied loads. Having the option to use a double reroll in addition to the traditional use of a single reroll decreases the number of tubes that would be considered as part of an exclusion zone for applicability of reroll as an alternate repair criteria. The qualification program establishes bounding leak rates for rerolls longer than 1 inch, which the licensee will use in ensuring that it maintains leakage below TS limits, as further discussed below. On the basis of the qualification program, the NRC considers the elimination of FPC's current requirement that the reroll be 1 inch in length to be acceptable.

FPC 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. FPC concluded that this is not a significant increase compared to the compressive load due to the transient and is acceptable. Compressive loads in the center of the SG are less than the compressive loads in the periphery. FPC stated that there is no reason to limit the tube load at the center to less than that allowed in the periphery. 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. FPC has decided to limit repair rolls at CR-3 to a configuration resulting in a maximum of 50 lbs additional compressive for any location. On the basis of this evaluation, the NRC finds it acceptable to remove the limitation of only one reroll per SG tube from the existing CR-3 ITS.

3.2 Structural and Leakage Integrity

On the basis of the results of the qualification testing, FPC has determined roll lengths sufficient to ensure adequate margins of structural and leakage integrity. FPC determined the amount of slip for a tube with a new hardroll expansion based on the possible combination of loadings. Field experience to date has shown that the majority of the flaws in the OTSGs within the tubesheet have been found to be short and axial in orientation.

With regard to structural integrity, the licensee demonstrated through slip tests that the limiting load for differential dilations consists of a major dilation and a minor dilation in the plane perpendicular to the tube. Differential dilations that are greater than the tested dilations resulted in an exclusion zone simply because test data is not available for such differential dilations.

With regard to leakage integrity, FPC assumes that all tubes with an axial load in excess of the tested joint strength load are assumed to slip. In addition, a post-slip leak rate is applied without taking credit for the original roll or the tube-to-tubesheet weld. The qualification tests predicted a steady-state leak rate for each repair roll. The NRC finds this approach acceptable because the predicted leak rate assumes a 360-degree, 100-percent through-wall circumferential flaw at the upper edge of the reroll and takes no credit for the original rolls or tube-to-tubesheet seal welds. Most of the flaw indications in the original roll transitions have been found to be small and axially oriented, which are attributed to primary water stress corrosion cracking; therefore, the NRC found that this is conservative and is an adequate approach.

A post-slip leak rate was applied to all repair rolls that have the potential to slip, regardless of whether a circumferential crack is actually present. The repair roll will not actually slip unless a large circumferential flaw is present. For the bounding load analyzed, the total leakage (and, therefore, the total number of repair rolls allowed) would be limited by the ITS limits. The leak rate from each single repair roll or overlapping repair roll that is serving as a pressure boundary is summed to obtain a total leak rate for the SG.

3.3 Field Installation and Inspection

FPC proposed to repair tubes in the same manner as those repairs performed under Revision 4 of BAW-2303P. FPC will install single, overlapping, or multiple rerolls in the tubes that have degradation in or near the original roll or reroll region. The repaired roll is typically installed using a manipulator and a tool head, monitored by a control system that tracks the position and monitors the torque of the roll expander. The torque is automatically controlled during the rerolling and is recalibrated after installation of a certain number of rerolls to ensure the minimum torque is maintained to produce proper fit.

Following the installation, FPC will inspect all rerolls using eddy current methods to ensure proper diametral expansion and positioning of the reroll repair joint. The proposed CR-3 ITS requires that the repair roll must be free of imperfections and degradation for the repair to be considered acceptable. The CR-3 ITS also requires that 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.

3.4 LBLOCA Considerations

In the above discussions of tubesheet hole dilations and leakage evaluations, the faulted conditions under consideration were limited to MSLB and SBLOCA. BAW-2303, Revision 4, does not evaluate the performance of rerolls following an LBLOCA. Instead, BAW-2303, Revision 4, implicitly credits Topical Report BAW-2374, Revision 0, which provides risk-informed arguments to justify excluding the LBLOCA from consideration as a faulted condition. The NRC has not approved BAW-2374 for referencing in a plant's licensing basis. However, the NRC has reviewed the risk-informed arguments in BAW-2374 as they relate to the reroll repairs at CR-3. The NRC performed its review in accordance with RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," as described below.

RG 1.174 contains general guidance for using probabilistic risk assessments (PRAs) in risk-informed decision-making on plant-specific changes to the licensing basis, and defines a licensing basis change as "modifications to a plant's design, operation, or other activities that require NRC approval." RG 1.174 also provides an acceptable approach to analyzing and evaluating proposed licensing basis changes.

According to the guidelines in RG 1.174, the applicant proposing the licensing basis change should perform an analysis of the proposed change using a combination of traditional engineering analyses with supporting insights from PRA methods. RG 1.174 states that when using risk-informed decision making, the proposed changes are expected to meet a set of key principles. These principles are:

a. The change meets current regulations unless an exemption is requested.

b. The change is consistent with the defense-in-depth philosophy.

c. The change maintains sufficient safety margins.

d. The increase in core damage frequency (CDF) or risk is small.

e. The impact should be monitored using performance measurement strategies.

3.4.1 Evaluation

BAW-2374, Revision 0, which in the context of these amendments is used to support the technical basis of the proposed reroll design specified in BAW-2303P, Revision 4, eliminates consideration of the thermal and pressure loads that result from a postulated LBLOCA. While loading conditions resulting from pipe break events are not included in design considerations specified by the ASME Code, they are required by NRC regulation and evaluated in accordance with ASME Code principles. Pressure loads are classified as primary stresses per the ASME Code and the Code requires evaluation of primary stresses for all conditions. However, the pressure loads on an OTSG are small during an LBLOCA when compared to other licensing basis events. Therefore, excluding the pressure loads resulting from an LBLOCA would not result in a decrease in the existing structural margins. However, due to differential thermal expansion during an LBLOCA event, significant thermal stresses may develop in some SG components. While ASME Code guidelines would classify these thermal stresses as secondary stresses and permit them to be excluded from the structural analysis when considering faulted conditions, the NRC has taken the position that for SG tubes and tube repair methods, including rerolls, these thermally-induced stresses are significant and should be considered in facility licensing bases. Eliminating consideration of the thermal stresses resulting from a postulated LBLOCA from the design of SG tubes and tube repair methods could result in a decrease in design structural margins.

The NRC has reviewed the engineering evaluations provided by the B&WOG for the reroll repairs as described in Appendix D of BAW-2374. Although the information in BAW-2374 (particularly that regarding reroll operating experience) appeared to be applicable only to upper tubesheet reroll repairs, the B&WOG confirmed by letter dated November 27, 2000, that all of the conclusions reached in the report were equally applicable to proposed lower tubesheet reroll repairs as well.

The engineering analysis in BAW-2374 regarding the performance of rerolls during the LBLOCA assumed that the tube was completely severed just to the primary system side of the reroll repair. The B&WOG considered this to be a conservative assumption since to date no reroll repair has been installed in a tube with a complete severance. In addition, critical flaw size calculations have suggested that a very large circumferential flaw would have to exist (approximately 60 percent through-wall and 150 degrees in extent) for the LBLOCA loads to cause tube severance. If complete severance did not occur just to the primary system side of the reroll repair, additional margin beyond that discussed below would exist, since load could be transmitted to the original roll joint and fillet weld.

Considering only the structural integrity provided by the reroll repair joint, in the event of the limiting LBLOCA the axial (differential thermal expansion) loads and dilations placed on the reroll joint would result in the load carrying capacity of the joint being exceeded. As a result, the reroll joint would be expected to slip within the tubesheet until the displacement-controlled thermal expansion loads were relieved. This was conservatively estimated by the B&WOG to entail a slippage of approximately 1.5 inches. Hence, the B&WOG concluded that, provided current exclusion zone criteria in BAW-2303P are followed (which do not permit reroll repairs within 2 inches of the secondary-side face of the tubesheet), the reroll repair joint would remain within the tubesheet and the slipped tubes would not experience gross structural damage.

The B&WOG also assessed the leakage integrity of the reroll repairs during the LBLOCA. Because of the differences in tubesheet bore on tube dilation resulting from the thermal loads associated with the LBLOCA, some loss of connection between the two was expected. In the event that tube severance had occurred just to the primary system side of the reroll repair joint and joint slippage had occurred, this would permit leakage between the primary and secondary sides of the SG. However, during an LBLOCA, only a small pressure differential would be expected to exist wherein the primary side was at a higher pressure than the secondary side. Assuming a conservative gap between the rolled tube and tubesheet bore of 0.001 inch and a representative pressure and temperature, the B&WOG topical report noted that the limiting leakage rate was determined to be 0.06 gallons per minute per slipped tube. Since the likelihood of a slipped tube was considered to be small and the leakage associated with a slipped tube was also considered to be small, the B&WOG concluded that the leakage integrity of the reroll repairs was acceptable for LBLOCA events.

The NRC examined the engineering evaluation provided by the B&WOG. The NRC determined that sufficient information had been provided to conclude that adequate structural integrity of the tube-to-tubesheet reroll joints could be maintained and that only a limited amount of leakage may be expected to result from joint slippage during an LBLOCA scenario. The NRC noted that the B&WOG estimate for leakage per slipped tube (0.06 gpm) appeared to be conservative. However, the NRC concluded that the number of tubes expected to slip in the event of an LBLOCA was indeterminate.

To date, reroll repairs have only been installed in upper tubesheets. Operational experience has shown that the majority of cracking at elevations above reroll joints has been axially-oriented cracking in the primary-side (upper) reroll transition region of the reroll and axially or circumferentially-oriented cracking in the heat affected zone (HAZ) of the tube-to-tubesheet fillet weld. The instances of circumferentially-oriented cracking in the fillet weld HAZ have been limited in extent and insufficient to lead to complete tube severance. However, this does not preclude the possibility that future in-service inspection results may show more significant circumferential cracking in the weld HAZ region, in the secondary-side (lower) transition region of original upper tubesheet rolls, or in either the primary-side (upper) or secondary-side (lower) reroll transition region of upper tubesheet reroll repairs. Similar conclusions can be made regarding the potential for degradation in lower tubesheet original roll and reroll regions as well.

Present SG inspection activities will be continued to ensure that, should significant circumferential cracking occur in the HAZ in the future, it will be identified, evaluated, and reported to the NRC. Pending the completion of the review of BAW-2374, it is the NRC's position that when individual licensees intend to install reroll repairs, a best-estimate evaluation must be performed to demonstrate that the as-found condition of the licensee's SGs (based on the most recent inspection results) is such that the technical bases for concluding that the amount of leakage (as described in Sections 3.4.1.1 and 3.4.1.2) that would occur in the event of an LBLOCA would be acceptable.

3.4.1.1 Defense-in-Depth Considerations

BAW-2374 demonstrates that rerolls could slip and leak following an LBLOCA, but this would not result in a significant degradation of the SG tube pressure boundary. BAW-2374 also explains that traditional defense-in-depth considerations would be maintained, specifically that a sequence of independent failures must occur in order for core damage or large radiological release to result from tube damage during an LBLOCA. For core damage to result, these events include the extremely low frequency pipe rupture event itself, a secondary side isolation failure, and a failure of recovery actions that would prevent sump depletion (which would take considerable time for the leak rates discussed above). For large early release, the failures include the pipe rupture, a failure of secondary system isolation, a failure of the emergency core cooling system low pressure recirculation system, and an unscrubbed release pathway via the secondary side/balance of plant (note that the leakage past the reroll repairs is a tortuous path).

BAW-2374 also demonstrates that, when considering the spectrum of LBLOCAs, the limiting rupture size/location from the standpoint of causing rerolls to slip does not correspond to the limiting rupture size/location from the standpoint of potential core damage (e.g., limiting in 10 CFR 50.46 analysis). Hence, while containment integrity may be slightly diminished as a result of reroll joint slippage, the likelihood of fuel cladding failure from the less challenging LBLOCA scenario is also diminished.

In addition, the proposed amendment includes a regulatory commitment (see Section 3.5 of this SE) for the licensee to demonstrate that, based on the condition of the SGs, an acceptable amount of leakage would be expected in the event of an LBLOCA at CR-3. In this context, “acceptable leakage” means the best estimate leakage (in the event of an LBLOCA) would not result in a significant increase in radionuclide release (e.g., in excess of 10 CFR Part 100 limits). For these reasons, the NRC finds that defense-in-depth is maintained.

3.4.1.2 Safety Margins

BAW-2374 noted that the design and repair of the OTSGs will continue to be governed by the requirements of Section III and Section XI of the ASME Boiler and Pressure Vessel Code along with staff guidance provided in draft NRC RG 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes." In general, draft RG 1.121 is based on the requirements of the ASME Code, and, in addition, specifies that SG tubes shall have a margin to burst of 3.0 for normal operating conditions and a margin of 1.4 for faulted conditions.

The NRC finds the evaluation of reroll performance during an LBLOCA acceptable. Approximately 1 to 1.5 inches of reroll joint slippage would be expected if the original roll and fillet weld do not carry the axial loads. While permitting such slippage does not maintain the same margins of structural integrity as the original roll and fillet weld (which would not slip), the staff concludes that the margins maintained are sufficient to ensure that gross failure of the SG tube containment boundary does not occur. The NRC also concludes that the evaluation of the expected leakage behavior of the reroll joints was reasonable, particularly considering the small differential pressures during the event. Finally, the commitment included in the proposed amendments (see Section 3.5 of this SE) includes the licensee demonstrating, based on the condition of its SGs, that an acceptable amount of leakage would be expected in the event of an LBLOCA. In this context, “acceptable leakage” means the best estimate leakage (in the event of an LBLOCA) would not result in a significant increase in radionuclide release (e.g., in excess of 10 CFR Part 100 limits). For these reasons, the NRC finds that sufficient safety margins will be maintained at CR-3 for reroll repairs in the event of an LBLOCA.

3.4.1.3 Change in Risk

BAW-2374, Revision 0, contains a bounding risk analysis to estimate the potential risk contribution (i.e., change in risk) by assuming a loss of SG tube integrity due to tube loads induced by LBLOCA (LBLOCA-induced SG tube rupture (SGTR)). The risk analysis uses CDF and large early release frequency (LERF) as the metrics for comparison to the acceptance guidelines of RG 1.174. In the risk analysis, event sequences associated with the postulated LOCA-induced SGTR scenarios were quantified to estimate the potential increase in CDF and LERF. This risk analysis conservatively assumes that the LOCA-induced SGTR is a catastrophic failure of the SG tube pressure boundary. This is significantly more challenging from the standpoint of losing coolant inventory than the limited SG pressure boundary leakage from rerolls described above.

In Section 3.4 of BAW-2374, two LOCA-induced SGTR scenarios were identified for quantitative assessment to determine the change in risk. Both scenarios begin with a LOCA in the upper region of the reactor coolant system (RCS) hot leg (i.e., "candy cane"). The RCS is refilled by the low pressure injection (LPI) subsystem of the emergency core cooling system (ECCS), which induces a SGTR in the broken RCS loop. In the first scenario, secondary side isolation failure and failure of operators to provide makeup water to the reactor building sump leads to eventual depletion of sump inventory through the secondary side, which causes ECCS failure and late core damage but no large early release. In the second scenario, secondary side isolation failure occurs, coupled with an independent failure of ECCS recirculation after depletion of the borated water storage tank, which leads to early core damage and possible large early release. The NRC determined that these two LOCA-induced SGTR scenarios adequately represented the sequences of events for a bounding risk analysis of possible loss of OTSG tube integrity due to a large-bore RCS pipe break.

One other possible scenario, such as core damage caused by boron dilution from the secondary side (Generic Issue 141 of NUREG-0933, "A Prioritization of Generic Safety Issues"), was reviewed and determined to be not applicable to this issue for several reasons. When the SG tubes experience the high tube-to-shell differential temperature following the upper hot leg break, the secondary side pressure would be lower than the primary pressure (based on the B&WOG’s thermal-hydraulic evaluation contained in Appendix A to BAW-2374). This condition would not result in significant tube failures or allow sufficient leakage to dilute the RCS from the injection of non-borated water. Furthermore, there would be guidance in emergency operating procedures to terminate feedwater flow to the OTSG.

The changes in CDF and LERF for the two LOCA-induced SGTR sequences were estimated by quantifying the cutset combinations containing the LOCA frequency, OTSG tube failure, secondary side isolation failure, failure of operator recovery actions to isolate the faulted SG and replenish primary inventory (in late CDF sequence), independent failure of LPI recirculation (in LERF sequence), and the conditional probability of large early release. The NRC reviewed the probability assumptions for each basic event in the cutset equations for the two scenarios and determined that conservative probability estimates for all of the basic events were used in the quantitative risk analysis. BAW-2374, Revision 0, uses an initiating event frequency of 8 x 10^{-7} per reactor-year, which is based on a 36-inch large pipe using the Beliczey-Schulz correlation to account for the frequency of through-wall cracks in piping based on historical experience data (NUREG/CR-5750) and the conditional probability of any rupture given a through-wall crack. This analysis assumed one through-wall crack to have occurred in a 36-inch diameter pipe, which was taken as conservative since, according to the topical report, “no TW (through-wall) cracks have been experienced in pipes larger than 8 inches.” The NRC does not accept this basis for establishing the estimated frequency for 36-inch pipe breaks of 8 x 10^{-7} per calendar year because the staff has not concluded that the Beliczey and Schultz correlation alone provides a sufficient basis for calculating this frequency.

Instead, the NRC concluded that the expected frequency for 36-inch pipe rupture is less than 1 x 10^{-6} per reactor-year. This conclusion is based on consideration of leak-before-break (LBB) approvals granted for all B&WOG facility main coolant loops in the mid-1980s. LBB evaluations have been accepted by the NRC for the purposes of satisfying 10 CFR Part 50, Appendix A, GDC-4 to, "demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping." In the rulemaking that implemented this provision into GDC 4, an extremely low probability of piping system rupture was equated to a frequency of 1 x 10^{-6} per reactor year or less. This probability is related to the frequency of failure at any location within the piping system analyzed for LBB. Consequently, the probability of a piping system rupture at the limiting location in the "candy cane" portion of the main coolant loop hot leg would be expected to be significantly less than 1 x 10^{-6} per reactor year. Hence, for the purpose of this SE, the staff accepts the B&WOG estimated frequency for 36-inch pipe breaks of 8 x 10^{-7} per calendar year.

It should be noted, however, that through-wall cracking was recently discovered in a 34-inch main coolant loop hot leg to reactor pressure vessel nozzle weld at the V.C. Summer Nuclear Station (Westinghouse design), which may call into question certain conclusions that have been made regarding the frequency of large-bore piping rupture. The NRC will evaluate the results of the V.C. Summer root cause analysis to determine if any generic conclusions can be drawn regarding the probability of large-bore piping rupture. If generic implications are found, the NRC staff may conclude that it is necessary to reevaluate the technical basis for establishing large-bore pipe rupture frequencies.

Based on this LOCA frequency estimate and conservative probability estimates for other events in the cutset equations, the change in CDF was estimated to be 8 x 10^{-10} per reactor-year and the change in LERF was estimated to be 4 x 10^{-11} per reactor-year. These quantitative risk changes are considered as very small risk increases according to the risk acceptance guidelines in RG 1.174 and are acceptable.

3.4.1.4 Compliance with Regulations

To determine if an exemption were necessary, the NRC considered whether permitting rerolled tube joints to slip complies with the regulations. With respect to GDC-14, based on the technical information set forth above, the staff concluded that the RCS pressure boundary at CR-3 continues to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. Therefore, the results of the topical report confirms that GDC-14 continues to be satisfied. With respect to GDC-30, the NRC has concluded that the reroll repairs at CR-3 are still designed and fabricated to the highest practicable standards as previously approved in Licensee Amendment No. 180, on June 28, 1999. Therefore, GDC-30 is satisfied.

The NRC also considered whether compliance with the requirements of 10 CFR 50.46 and Appendix K to 10 CFR Part 50 would be an issue at CR-3, because the licensee did not request an exemption from these requirements. In the letter dated November 27, 2000, the B&WOG discusses continued compliance with 10 CFR 50.46, given that the hot leg "candy cane" break scenarios described in Appendix A to BAW-2374 can result in leakage past rerolls. The B&WOG presented analyses, based on quantitative sensitivity studies performed with FTI approved large break and small break evaluation models, and supplemented by qualitative analyses, to show that the consequences (with respect to 10 CFR 50.46(b)) of hot leg "candy cane" breaks are bounded by those of the existing licensing basis cold leg guillotine beaks.

The NRC concludes that these quantitative and qualitative analyses adequately demonstrate that the subject LOCA scenarios (large and small break) are not bounding with regard to 10 CFR 50.46(b) criteria, and do not represent new limiting scenarios. The NRC also concludes that the qualitative and quantitative analytical methods employed by FTI satisfy the model requirements of 10 CFR 50.46(a)(1)(i) and (ii) for the specific analyses that the NRC reviewed. Based on this, the NRC concludes that LOCA analyses provided in support of BAW-2374 meet the requirements of 10 CFR 50.46 for CR-3, and are, therefore, acceptable.

Based on this analysis, the NRC has concluded that use of Topical Report BAW-2303P, Revision 4, which does not consider LBLOCA loads in the design of the reroll repairs, does not require an exemption to the regulations.

The NRC also notes that the analyses described in the B&WOG's letter dated November 27, 2000, rely on operator action, as instructed by plant Emergency Response Procedures (ERPs), to achieve and maintain long term core cooling per 10 CFR 50.46(b). In order to demonstrate the applicability of the BAW-2374 LOCA analyses, FPC stated that it has verified that the CR-3 plant-specific ERPs are consistent with the descriptions in BAW-2374 in regard to the key operator actions for mitigation of the accident sequence of concern. FPC has also confirmed that the CR-3 ERPs are consistent with the B&WOG's November 27, 2000, letter with respect to compliance with 10 CFR 50.46. The NRC concludes that the licensee has sufficiently resolved the NRC's concerns related to compliance with 10 CFR 50.46 for an LBLOCA and SBLOCA at CR-3.

3.4.1.5 Integrated Decision Making

The NRC had considered removal of LBLOCA loads from the reroll design, consistent with the proposed use of Topical Report BAW-2303, Revision 4, with respect to the integrated decision making criteria in RG 1.174. The NRC has concluded that this change is a practical solution for addressing the thermal loads caused by an LBLOCA and their impact on the design of the SG repair method. Upon implementation of this amendment, FPC will be able to use practical and acceptable repair methods (e.g., rerolls) at CR-3 and avoid premature plugging of SG tubes. This benefit outweighs the change in CDF and LERF, which is considered very small by RG 1.174 criteria. Further, adequate margin and defense-in-depth are maintained and there is no impact on the LOCA analysis performed to satisfy 10 CFR 50.46 at CR-3.

3.4.1.6 Implementation and Monitoring

CR-3 has not proposed any changes to the existing monitoring programs. Implementation of the proposed amendments will not result in any changes in plant operation, inspections, or design. Inspection and monitoring programs which impact this SE can be broken down into two areas: (1) those that apply to RCS piping and (2) those that apply to the SG primary-to-secondary pressure boundary.

RCS piping will continue to be inspected in accordance with the inservice inspection program, as required by the ITS. Additionally, the CR-3 ITS contain RCS leakage limits and require plant shutdown if those limits are exceeded. As required by CR-3's licensing basis as it applies to LBB approvals, CR-3 has RCS leakage detection systems that can detect RCS leakage before any postulated flaws reach a size that could challenge the structural integrity of the RCS under faulted conditions. Further, primary system leakage is included within the performance indicators of the NRC's Revised Oversight Program, which ensures that appropriate emphasis will be given to any unacceptable change in RCS leakage. For these reasons, it is concluded that the existing inspection and monitoring programs at CR-3 will ensure a low probability of degradation of the RCS piping that could lead to an LBLOCA.

The SG tubes will continue to be inspected and plugged or repaired as required by the CR-3 ITS. This will include inspections of the pressure boundary components, including the original tube-to-tubesheet roll transition region and fillet weld or the tube-to-tubesheet reroll transition region, as appropriate. In addition, CR-3 must have an adequate inspection program for repaired tubes (including plugs) to verify that the primary-to-secondary leakage following an LBLOCA is within acceptable limits in order to provide an adequate basis for evaluating compliance with the technical arguments in BAW-2374 that were relied upon by this SE. Finally, the CR-3 ITS will continue to impose SG leakage limits and require plant shutdown if the limits are exceeded. For these reasons, the NRC has concluded that CR-3's inspection and monitoring programs ensure that the SG primary-to-secondary pressure boundary will be adequately maintained to support the conclusions of this SE.

3.4.1.7 Conformance to RG 1.174

RG 1.174 describes an acceptable method for assessing the nature and impact of licensing basis changes by a licensee when the licensee chooses to support these changes with risk information. RG 1.174 identifies a four-element approach for evaluating such changes, and these four elements are aimed at addressing the five key principles of risk-informed regulation that plant changes are expected to meet. NRC review has determined that the risk-informed arguments in BAW-2374, Revision 0, that the staff relied on for this SE are consistent with RG 1.174 as discussed below:

Element 1: Element 1 of the RG 1.174 approach recommends that the licensee define the proposed change.

BAW-2303P, Revision 4, describes the faulted conditions that were evaluated in the design of the reroll repairs for CR-3. BAW-2303P, Revision 4, implicitly relies upon BAW-2374 to exclude LBLOCA and only consider MSLB and SBLOCA as the faulted conditions. The NRC finds Element 1 is satisfied.

Element 2: Element 2 provides for the performance of an engineering analysis.

Under this element, the licensee performs qualitative and quantitative analyses using traditional engineering approaches, and techniques associated with the use of PRA findings. Further, this element recommends that the licensee satisfy the principles set forth in Section 2 of RG 1.174. This includes, for example, establishment of a reasonable balance between prevention, mitigation, and avoidance of over-reliance on programmatic activities.

Appendix A of BAW-2374 describes the thermal-hydraulic analysis of an LBLOCA with respect to its effect on primary-to-secondary leakage if SG tube leakage occurs. Appendix D of BAW-2374 describes the impact of the thermal-hydraulic loads on the SG tubes and repair methods. Based on these evaluations, the LBLOCA will have only a minor impact on the integrity of the SG reroll repairs (e.g., result in minor leakage). Further, the reroll repairs are still designed to handle MSLB and LOCAs of attached piping (SBLOCAs). Therefore, the SG tubes continue to mitigate the effect of an accident without over reliance on programmatic activities. The NRC finds that the analysis criteria of this element are satisfied.

RG 1.174 states that in implementing risk-informed decision making, plant changes are expected to meet a set of key principles. The following paragraphs summarize these principles and the NRC findings related to these principles.

- Principle 1 states that the proposed change must meet current regulations unless it is explicitly related to a requested exemption or rule change. As described above, the NRC has concluded that permitting rerolls to slip during an LBLOCA meets the current regulations without requiring an exemption pursuant to 10 CFR 50.12. Therefore, principle 1 is satisfied.

- Principle 2 states that the proposed change must be consistent with the defense-in-depth philosophy. The NRC has concluded that these amendments are, in general, consistent with the defense-in-depth philosophy in that (a) any LBLOCA-induced SG tube leakage would not result in a significant reduction in the effectiveness of the SG tube containment barrier, and (b) a sequence of independent failures would need to occur in order for core damage or large early release to occur. Therefore, principle 2 is satisfied.

- Principle 3 states that the proposed change shall maintain sufficient safety margins. As set forth above, the NRC has concluded that the change maintains sufficient safety margins to ensure that gross failure of the SG tube containment boundary function does not occur. Therefore, principle 3 is satisfied.

- Principle 4 states that when proposed changes result in an increase in CDF or LERF, the increases should be small and consistent with the intent of the Commission's Safety Goal Policy Statement. As explained above, the NRC concludes that the increases in CDF and LERF are considered very small. Therefore, principle 4 is satisfied.

- Principle 5 states that the impact of the proposed change should be monitored using performance measurement strategies. CR-3 has not proposed any changes to the existing monitoring programs, which are already sufficient to monitor the integrity of the RCS pressure boundary and the SG tubes. However, CR-3 has committed to demonstrate that, based on the condition of the SGs, an acceptable amount of leakage is expected in the event of an LBLOCA. Therefore, principle 5 is satisfied.

Element 3: Element 3 is the definition of the implementation and monitoring program.

The primary goal of this element is to ensure that no adverse safety degradation occurs because of the proposed change. The NRC has determined that the existing monitoring programs are sufficient to monitor the integrity of the RCS and SG tubes. However, in accordance with the regulatory commitments included in the letter dated June 28, 2001, CR-3 will verify that the expected primary-to-secondary leakage is acceptable based on the current condition of their SGs. Element 3 is satisfied.

Element 4: Element 4 is the submittal of the proposed change.

CR-3 submitted a request for the change by letter dated March 21, 2001, through reference to Topical Report BAW-2303P, Revision 4, which relies on the technical arguments in BAW-2374. The application was supplemented by the letter dated June 28, 2001. Therefore, Element 4 is satisfied.

3.4.2 Summary of LBLOCA Considerations

In summary, the NRC concludes that a break in the large-bore RCS hot leg could lead to large axial loads on the SG tubes due to the temperature difference between the SG tubes and the SG shell. The result, based on technical bases in BAW-2374 as applied to the reroll repairs at CR-3 would be a minor degradation in the SG tube pressure boundary in a condition where the pressure difference across the tubes is small.

The NRC does not require the licensee to consider the loads and dilations caused by an LBLOCA in analyzing the CR-3 reroll repairs, and the NRC finds it acceptable that slippage will occur during the limiting LBLOCA if other factors (such as circumferential cracking) prevent the tube seal weld from carrying the axial load. CR-3 has made a regulatory commitment that they will demonstrate, based on the condition of its SGs (including the number of SG tubes that are expected to slip), that the total primary-to-secondary leakage following an LBLOCA is acceptable (i.e., adequate margin and defense-in-depth is maintained). For the purpose of this evaluation, “acceptable leakage” means the best estimate leakage (in the event of an LBLOCA) would not result in a significant increase in radionuclide release (e.g., in excess of 10 CFR Part 100 limits). The NRC finds that adequate margin of safety and defense-in-depth is maintained, and that the increase in risk as measured by CDF and LERF is small.

Additionally, excluding LBLOCA from consideration in designing rerolls is not intended to affect CR-3’s approved LOCA evaluation models or the analysis performed to demonstrate compliance with the requirements of 10 CFR 50.46. It is not intended to exclude any pipe break sizes or locations from the LOCA analyses that were performed to demonstrate compliance with 10 CFR 50.46 or exclude evaluation of consequent SG tube degradation from consideration in those analyses.

Therefore, the NRC finds that the LBLOCA, as described in Appendix A to BAW-2374, does not need to be considered in the design of CR-3's reroll repairs provided the regulatory commitments contained in letter dated June 28, 2001, are implemented.

3.5 Regulatory Commitments

In the June 28, 2001, letter, FPC made the following commitments:

1. Following each inservice inspection of SG tubes but prior to returning the CR-3 steam generators to service, FPC will verbally notify the NRC of the following:

    a. Number of tubes with circumferential cracking indications inboard of the roll repair.

    b. Number of tubes with circumferential cracking indications in the original roll region, including the zone adjacent to the tube-to-tubesheet seal weld if no re-roll is present.

    c. Determination of the best-estimate total leakage that would result from an analysis of the limiting Large Break Loss of Coolant Accident (LBLOCA) based on as-found circumferential cracking in the original tube-to-tubesheet rolls, tube-to-tubesheet re-roll repairs, and the zones adjacent to the seal welds.

2. Demonstrate that the primary-to-secondary leakage following an LBLOCA, as described in Appendix A to Topical Report BAW-2374, Revision 1, is acceptable based on the as-found condition of the steam generators. This is required to demonstrate that adequate margin and defense-in-depth are maintained. For the purpose of this evaluation, "acceptable" means a best estimate of the leakage expected due to an LBLOCA where that leakage would not result in a significant increase of radionuclide release (e.g., in excess of 10 CFR Part 100 limits). A summary of this evaluation shall be provided to the NRC following completion of steam generator tube inservice inspection with the report required by Improved Technical Specification 5.7.2.e.

These regulatory commitments will ensure that FPC will perform an adequate evaluation to demonstrate that gross structural failure and leakage of the reroll repair joints will not occur in the event of an LBLOCA pending the resolution of this issue during the review of BAW-2374. This evaluation will demonstrate that adequate safety margins and defense-in-depth are maintained in the design and installation of the reroll repairs at the CR-3. FPC recognizes that further NRC review of BAW-2374, Revision 0, and BAW-2374, Revision 1, may require it to modify the regulatory commitments or otherwise involve additional actions to conform with final NRC conclusion on the topical report. The NRC has concluded that adequate controls for these actions are provided by the licensee's commitment management program.

3.6 Proposed Improved Technical Specification Change

To implemented the revised reroll process, the licensee proposed the following changes to the ITS:

ITS 5.6.2.10.11.b This section is revised to permit the 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 staff finds the changes to this ITS section acceptable on the basis of the staff evaluation of Topical Report BAW-2303P, Revision 4, and limited review of Topical Report BAW-2374, Revision 0, for its application to the proposed reroll activity at the CR-3, as discussed in this SE.

As a part of the amendment request, the licensee submitted the following changes to the ITS that are not related to the reroll process:

ITS 5.6.2.10.3 This section is revised by deleting the following: ". . . 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." The staff finds that this one-time exception is no longer applicable and is acceptable for removal.

ITS 5.6.2.10.4.b The following paragraph of this section is deleted: ". . . [T]here 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 OTSG will not be impacted with those tubes in service." By License Amendment No. 169, the paragraph was inserted as a one-time change to the ITS for refueling outage 11 or an outage of sufficient duration, if such an outage would occur before refueling outage 11. Subsequently, in License Amendment No. 188, the NRC approved a permanent alternate repair criteria for the tube end cracking. The NRC finds the removal of this paragraph is acceptable because the alternate repair criteria, approved by License Amendment No. 188, supersedes the changes approved by License Amendment No. 169.

3.7 Mechanical Integrity Evaluation

For the previous license amendments granted for OTSG tube rerolling at Oconee Nuclear Station, Units 1, 2, and 3 and Arkansas Nuclear One, Unit 1, the licensees of those facilities performed qualification programs presented in BAW-2303P that demonstrated the strength of the roll joints met the specifications stated in RG 1.121. In this program, test specimens were loaded under normal, upset and postulated accident conditions. Hydrostatic pressure tests were performed at pressures exceeding 3 times normal operating and 1.43 times main steam line break pressures as stipulated in RG 1.121. The purpose of these tests was to look for gross leakage or structural failure of the joints. No mechanical change or gross leakage in the samples was noted.

The original analysis in BAW-2303P, Revision 3, had assumed no joint slippage as the design basis for rerolls. In the current request for amendment, FPC developed a qualification program presented in Revision 4 of the BAW-2303P to demonstrate repair roll joint integrity through slip and leak tests. The program consisted of (1) establishing tube loads for the qualification tests, (2) preparing a mockup to simulate tubesheet conditions for qualification tests, and (3) performing verification tests and analyses of the test results.

FTI developed a finite element (FE), linear-elastic, axisymmetric model of an overall OTSG, including the tube bundle, the tubesheets, shell, heads, and support skirt, to quantify the general structural behavior of the OTSG during various operating and accident transients. The NRC did not review the details of the FE model, thermal-hydraulic analyses, and structural analyses. However, in the topical report, FTI provided a general summary of assumptions in the development of the FE model, assessments of the parameters addressing the significant effects of different features among the OTSGs, and a general summary of the results of the thermal-hydraulic and structural analyses.

FE thermal analysis was performed to model the general structural behavior of the OTSG, including deflections and axial tube loads, as well as the local structural behavior (hole dilations). The key results of the finite element analyses included (1) axial tube loads as a function of tubesheet radial position, (2) tube-to-tubesheet hole differential dilations as a function of tubesheet radial position, and (3) tube-to-tubesheet hole differential dilations as a function of depth into the tubesheet. Differential dilation is a term that is used to refer to the interface between the tube outside diameter and the tubesheet bore diameter, which allows a comparison of the relative interface of the joint for any transient condition. The limiting accident transient for load-carrying capacity of the repair roll is a function of differential dilations and axial tube load, which are used to determine plant-specific exclusion zones for repair roll.

The results of the finite element analysis indicate that for CR-3 OTSGs, the SBLOCA is the most limiting accident from the tube loads and, hence, reroll repair perspective. The reason for this lies in its design features. Some key design differences between CR-3 and the Oconee units account for the fact that the MSLB was determined to be the limiting tube load event for the latter. These differences are as follows.

1. CR-3 has a safety-grade, single failure proof Emergency Feedwater Initiation and Control (EFIC) system, while Oconee does not have such a system.

   EFIC actuates components in response to sudden changes in secondary conditions. A decrease in main steam pressure to approximately 600 psig will initiate a series of actions to isolate the faulted OTSG. EFIC will initiate a closure of the Main Steam Isolation Valves (MSIVs), which would limit the induced cooldown. EFIC would also terminate main feedwater flow to the OTSGs, and only permits emergency feedwater flow to the intact OTSG, via the Feed-Only-Good-Generator (FOGG) logic. In contrast, the lack of an EFIC or similar system does not readily assure the positive and automatic isolation of the secondary to limit cooldown effects - including those on tube load.

2. CR-3 has MSIVs, while Oconee does not have them. Each of CR-3's OTSGs has two main steam lines, for a total of four potential break locations. Each line has an MSIV. At the onset of an MSLB, the MSIVs on the faulted OTSG close first, but the intact OTSG is isolated by its MSIVs a few seconds later. The former ensures that the blowdown is limited to only one of the two main steam lines on the faulted OTSG. The latter isolation ensures that uncontrolled cooldown paths do not exist

through the intact OTSG. This series of actions ensures that the impact of the cooldown is lessened. In contrast, the lack of MSIVs does not readily assure the limitation of the blowdown; thus exacerbating cooldown effects.

These are the principal reasons why the SBLOCA is the limiting event at CR-3 from the tube load perspective.

In its response to the NRC's request for additional information dated November 10, 2000, concerning the potential flow-induced loadings on SG tubes, the licensee provided a summary of the results of their proprietary analysis. The flow-induced loadings affect the initial conditions for the FE thermal analysis, which, in turn, determine the axial tube loads and differential dilations. FPC also stated that the significant cross-flow loads on the periphery tube section between the secondary face of the upper tubesheet and the cylindrical baffle during the first few seconds of the MSLB transient will result in some plastic deformation of the tubes. The resulting bowing of the tubes imparts an axial tension load of approximately one-tenth of the axial load. The licensee contends that the evaluation of the tube loads without accounting for the plastic deformation was a conservative approach.

This approach is justified because a typical stress-strain diagram for structural material (e.g., steel) shows that as strain increases, stress drops off slightly after reaching a maximum value. Thus, instead of having the stress value mitigated by this behavior, no credit is taken for deformation. However, from leakage considerations, plastic deformation tends to exacerbate leakage. At the onset of the event, flows induced by the blowdown through the broken steam line would exert lateral forces that would tend to bow the faulted OTSG's tubes, particularly those on the periphery. As the blowdown subsides, some tubes in the faulted OTSG would remain deformed. A bowed tube would tend to alter the tight interference fit between the tube outer diameter and the tubesheet bore, introducing dilation and increasing the leakage. Thus for an evaluation of leakage, accounting for plastic deformation would be conservative.

As stated earlier, FPC used the results of the structural analyses to construct a mockup to simulate tubesheet conditions for qualification tests. The finite element analyses results were reviewed to determine a bounding set of dilation test cases. Then a set of corresponding bounding axial loads were developed, which together with the tubesheet bore dilations effectively bound all normal operating and accident transients for the OTSG. The test matrix was developed from a set of applied loads for each slip test case and a combination of internal pressure and applied load for leak tests. The test sequence progressed from less severe conditions (tubesheets bore dilations and/or axial loads) to more severe conditions. When tube movement was noted, the initial sequence of tests was terminated for that sample. FTI performed testing to (1) measure the loads at which tube slippage would occur, (2) measure leakage for reroll joints that did not slip, and (3) measure leakage if tube slippage did occur. The test data were compiled and summarized to develop slip and leakage criteria to qualify installation of a repair roll on a plant-specific basis. The repair roll is allowed to slip under specific faulted conditions.

3.7.1 Mechanical Integrity Evaluation Conclusion

On the basis of its review as discussed above, the NRC finds that the assumptions made in the development of the finite element analytical model, the analytical methodology, and the results of the structural analysis are reasonable and acceptable. In addition, the bounding set of loads to evaluate the dilation tests envelope normal operating and accident transients for the OTSG.

4.0 STATE CONSULTATION

Based upon a letter dated March 8, 1991, from Mary E. Clark of the State of Florida, Department of Health and Rehabilitative Services, to Deborah A. Miller, Licensing Assistant, U.S. Nuclear Regulatory Commission, the State of Florida does not desire notification of issuance of license amendments.

5.0 ENVIRONMENTAL CONSIDERATIONS

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (66 FR 20006). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

FPC proposed to implement an alternate repair method using a hardroll expansion process to repair tubes having indications of tube degradation in the original roll or repair roll (or both) in regions of the upper or lower tubesheets. The technical basis for the proposed reroll method is documented in Topical Report BAW-2303P, Revision 4.

The NRC has determined that (1) FPC's alternate repair criteria were established on the basis of the qualification tests that used specimens simulating the actual tube-to-tubesheet joint configuration of the SGs, (2) the loads for structural and leakage tests were specified and applied in accordance with RG 1.121, and (3) the proposed changes to the ITS satisfy all regulatory requirements applicable to SG tube integrity.

On the basis of submitted information, the NRC concludes that the proposed ITS changes regarding reroll repair for degraded roll joints in the SGs at CR-3 are acceptable because FPC has demonstrated through an acceptable qualification program that the reroll satisfies GDC-14 of Appendix A to 10 CFR Part 50 and conforms to the guidance of RG 1.121.

The NRC has reviewed the finite element analysis model of a OTSG including the tube bundle, tubesheets, shell, heads, and support skirt to quantify the general structural behavior of the OTSG during various operating and accident transients. In addition, the finite element analysis results were reviewed to determine bounding loads for tube slippage/leakage tests. On the basis of its review, the NRC finds this to be acceptable.

It should be noted, however, as indicated in Section 3.4.1.3 of this SE, that through-wall cracking in a main coolant loop nozzle weld at the V.C. Summer facility may call into question certain conclusions that have been made regarding the frequency of large-bore piping rupture. The NRC will evaluate the results of the V.C. Summer root cause analysis to determine if any generic conclusions can be drawn regarding the probability of large-bore piping rupture. If generic implications are found, the NRC may conclude that it is necessary to reevaluate the technical basis for establishing large-bore pipe rupture frequencies.

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

7.0 REFERENCES

1. NRC Draft RG 1.121, “Bases for Plugging Degraded PWR Steam Generator Tubes,” August 1976.

2. FTI Topical Report, BAW-2374, “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,” July 2000.

3. FTI Topical Report, BAW-2303P, Revision 4, “OTSG Repair Roll Qualification Report,” August 2000. (Proprietary)

4. Letter from D. Firth (FTI) to NRC, “Additional Information to Topical Report BAW-2374, “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,” November 27, 2000.

5. FTI 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,” March 2001.

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