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

March 28, 2001

Mr. Craig G. Anderson
Vice President, Operations ANO
Entergy Operations, Inc.
1448 S. R. 333
Russellville, AR 72801

SUBJECT: ARKANSAS NUCLEAR ONE, UNIT NO. 1 - ISSUANCE OF AMENDMENT RE: THE USE OF THE REROLL REPAIR PROCESS FOR STEAM GENERATOR TUBES (TAC NO. MB0097)

Dear Mr. Anderson:

The Commission has issued the enclosed Amendment No. 212 to Facility Operating License No. DPR-51 for Arkansas Nuclear One, Unit No. 1 (ANO-1). The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated September 28, 2000, as supplemented by letter dated February 19, 2001. In addition, you provided in a letter dated October 26, 2000, a non-proprietary version of the topical report BAW-2303, "OTSG [Once-Through Steam Generator] Repair Roll Qualification Report," Revision 4 (ADAMS Accession No. ML003765879). The topical report provided part of your justification for this amendment.

The amendment revises the TS for ANO-1 to allow a revised reroll repair process for the steam generators. This amendment allows the reroll repair process to be used multiple times for a single tube and would allow the repairs in both the upper and lower tubesheets.

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

Sincerely,

William Reckley

William Reckley, Project Manager, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-313

Enclosures: 1. Amendment No. 212 to DPR-51
2. Safety Evaluation

cc w/encls: See next page

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

ENTERGY OPERATIONS INC.

DOCKET NO. 50-313

ARKANSAS NUCLEAR ONE, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 212
License No. DPR-51

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

   A. The application for amendment by Entergy Operations, Inc. (the licensee) dated September 28, 2000, as supplemented October 26, 2000, and February 19, 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 license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

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

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-51 is hereby amended to read as follows:

   2. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 212, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

John G. Nakoski for

Robert A. Gramm, Chief, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: March 28, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 212

FACILITY OPERATING LICENSE NO. DPR-51

DOCKET NO. 50-313

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

Remove	Insert
110m	110m
110n	110n

4.18.5 Acceptance Criteria

a. As used in this specification:

1. Tubing or Tube means that portion of the tube or sleeve which forms the primary system to secondary system pressure boundary.

2. Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

3. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either the inside or outside of a tube.

4. Degraded Tube means a tube containing imperfections ≥ 20% of the nominal wall thickness caused by degradation, except where all degradation has been spanned by the installation of a sleeve or repaired by a rerolled joint.

   The reroll repair process can be used to repair tubes with defects in the upper and lower tubesheet areas as described in topical report, BAW-2303P, Revision 4.

5. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

6. Defect means an imperfection of such severity that it exceeds the plugging limit except where the imperfection has been spanned by the installation of a sleeve. A tube containing a defect in its pressure boundary is defective.

7. Plugging Limit means the imperfection depth at or beyond 40% of the nominal tube wall thickness for which the tube shall be sleeved, rerolled, or removed from service because it may become unserviceable prior to the next inspection. This does not apply during Cycle 16 to ODIGA indications within the defined region of the upper tubesheet. These indications shall be assessed for continued plant operation in accordance with topical report BAW-10235P, Revision 1.

   Axially-oriented TEC indications in the tube that do not extend beyond the adjacent cladding portion of the tube sheet into the carbon steel portion are not included in this definition. These indications shall be assessed for continued plant operation in accordance with topical report BAW-2346P, Rev. 0.

8. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in Specification 4.18.4.c.

9. Tube Inspection means an inspection of the steam generator tube from the point of entry completely to the point of exit. For tubes that have been repaired by the reroll process within the tubesheets, that portion of the tube outboard of the new roll can be excluded from future periodic inspection requirements because it is no longer part of the pressure boundary once the repair roll is installed.

Bases

The surveillance requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

In general, steam generator tubes that are degraded beyond the repair limit can either be plugged, sleeved, or rerolled. The steam generator (SG) tubes that are plugged are removed from service by the installation of plugs at both ends of the associated tube and thus completely removing the tube from service. When the tube end cracking (TEC) alternate repair criteria is applied, axially-oriented indications found not to extend from the tube sheet cladding region into the carbon steel region may be left in service under the guidelines of topical report BAW-2346P, Rev. 0. When the upper tubesheet outer diameter intergranular attack (ODIGA) alternate repair criteria is applied, indications found within the defined region of the upper tubesheet may be left in service under the guidelines of topical report BAW-10235P, Revision 1. The defined region begins one inch above the upper tubesheet secondary face and ends at the nearest tube roll transition. Following a SG inspection, an operational assessment is performed to ensure primary-to-secondary leak rates will be maintained within the assumptions of the accident analysis.

Degraded steam generator tubes can also be repaired by the installation of sleeves which span the area of degradation and serve as a replacement pressure boundary for the degraded portion of the tube, thus permitting the tube to remain in service.

Degraded steam generator tubes can also be repaired by the rerolling of the tube in the upper or lower tubesheet to create a new roll area and pressure boundary for the tube. The portion of the tube that is outboard of the repair roll is the portion of the tube closest to the primary side of the tubesheet and includes tubing from the tube end up to and including the heel expansion transition. The 1-inch repair roll is considered to be within the pressure boundary. If more than one repair roll is present, the outboard portion includes tubing from the tube end to the heel transition and the beginning of the 1-inch repair roll that is farthest from the primary side of the tubesheet. The rerolling repair process will be used to repair defects in the upper and lower tubesheet in accordance with BAW-2303P, Revision 4.

All tubes which have been repaired using the reroll process will have the new roll area inspected during future inservice inspections. Defective or degraded tube indications found in the new roll and any indications found in the original roll need not be included in determining the Inspection Results Category for the generator inspection.

The reroll repair process can be used to repair tubes with defects in the upper and lower tubesheet areas. Installation of multiple repair rolls in a single tube is acceptable. The new roll area must be free of detectable degradation in order for the repair to be considered acceptable. After the new roll area is initially deemed acceptable, future degradation in the new roll area will be analyzed to determine if the tube is defective and needs to be removed from service or repaired. The reroll repair process is described in the topical report, BAW-2303P, Revision 4.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 212 TO

FACILITY OPERATING LICENSE NO. DPR-51

ENTERGY OPERATIONS, INC.

ARKANSAS NUCLEAR ONE, UNIT NO. 1

DOCKET NO. 50-313

1.0 INTRODUCTION

By letter dated September 28, 2000, as supplemented by letters dated October 26, 2000, and February 19, 2001, Entergy Operations, Inc. (Entergy or the licensee), submitted a request for changes to the Arkansas Nuclear One, Unit No. 1 (ANO-1), Technical Specifications (TSs). The requested changes would revise the ANO-1 TSs by allowing the reroll repair process for steam generator (SG) tubes to be used multiple times for a single tube and would allow the repairs in both the upper and lower tubesheets.

The letter dated October 26, 2000, provided a non-proprietary version of the topical report BAW-2303, "OTSG [Once-Through Steam Generator] Repair Roll Qualification Report," Revision 4 (ADAMS Accession No. ML003765879). A proprietary version of the report was included in the licensee's application dated September 28, 2000. The letter dated February 19, 2001, provided clarifying information to support the staff's review of the application and did not change the staff's initial proposed no significant hazards consideration determination.

By letter dated December 15, 2000, the NRC approved a reroll repair methodology for Oconee Nuclear Station, Unit Nos. 1, 2, and 3 (Oconee). Entergy proposed a similar reroll repair methodology as that approved for the Oconee units.

1.1 Relationship of this Amendment Request 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"

By License Amendment No.190 dated April 10, 1998, the Nuclear Regulatory Commission (NRC) staff approved the use of repair rolls in the upper tubesheet, as analyzed in BAW-10232P, "OTSG Repair Roll Qualification Report (Including Hydraulic Expansion Report)," Revision 00, for ANO-1. However, reanalysis of the reroll methodology became necessary due to identification of a Small Break Loss of Coolant Accident (SBLOCA) that was more limiting than the accident previously evaluated in BAW-10232P, Revision 00. In addition, the Main Steam Line Break (MSLB) transient has been re-analyzed, resulting in a new set of design loads. In BAW-2303P, Revision 4, Framatome Technologies, Inc. (FTI) described its analyses performed at the request of the Babcock & Wilcox (B&W) Owners Group (B&WOG). The report was provided in support of the license amendment request for ANO-1 for repair rolls to

be installed in both the upper and lower tubesheets and to address 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-10232P, Revision 00, remain acceptable based on the criteria contained in BAW-2303P, Revision 4. (See Section 3.1 of this safety evaluation.)

By only evaluating the reroll repairs for MSLB and SBLOCA faulted conditions, BAW-2303P, Revision 4, implicitly credits the results of 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," which was submitted to the NRC by the B&WOG by letter dated July 7, 2000.[1] BAW-2374 provides the risk-informed bases for excluding accidents such as 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.

The staff has not approved BAW-2374. However, based on the risk-informed arguments presented in BAW-2374, the staff accepts that the reroll repairs at ANO-1 may slip during a LBLOCA, resulting in an increase in leakage past the reroll. ANO-1 has included in their application a regulatory commitment to demonstrate that the expected leakage following a LBLOCA is acceptable, based on the as-found condition of their OTSGs. Section 3.4 of this safety evaluation (SE) contains the staff's evaluation of the risk-informed arguments presented in BAW-2374.

2.0 BACKGROUND

ANO-1 has two model 177FA OTSGs manufactured by B&W. The 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 one 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.

General Design Criterion (GDC) 14 of Appendix A to 10 CFR Part 50 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

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[1] The B&WOG is expected to submit or to have submitted a revision to BAW-2374 at about the time this amendment is issued. Reference in this SE to BAW-2374 refers to the version submitted in July 2000.

rupture.[2] 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) 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 undegraded original 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 against tube rupture under normal operating conditions be equal to or greater than three 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 NB-3225 of Section III of the Boiler and Pressure Vessel Code (the Code) of the American Society of Mechanical Engineers (ASME). 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

For the previous license amendments granted for rerolling at ANO-1, the licensee performed a qualification program presented in BAW-10232P, Revision 00, that demonstrated the strength of the roll joints was satisfactory in accordance with RG 1.121. The licensee applied loads to sample tubes to simulate or exceed normal, thermal and pressure cycling transient, and postulated accident conditions. In accordance with RG 1.121, room temperature hydrostatic pressure tests were performed at a pressure exceeding three 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

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[2] The GDC are not applicable to plants such as ANO-1 with construction permits issued prior to May 21, 1971. At the time of the promulgation of Appendix A, the Commission stressed that the GDC were not new requirements and were promulgated to more clearly articulate the licensing requirements and practice in effect at that time. For convenience, the staff refers to the GDC instead of the ANO-1 specific licensing bases.

noted. The original analysis in BAW-10232P 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 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.

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 staff did not review the details of the FE model, thermal-hydraulic analyses, and structural analyses. However, in BAW-2303P, Revision 4, 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.

In the qualification program, the licensee considered the impact of tubesheet bowing on the roll joints since 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.

Based on the temperature differential between tubes and the shell and the pressure differential across the circular (flat plate) tubesheets during normal operating and various transient conditions, the FE analyses provide axial tube loads and the data required to calculate tube and tubesheet bore dilations. FTI stated that the axial tube loads calculated by the FE analyses supersede all previously calculated axial loads.

FE thermal analysis was performed to model the general structural behavior of the OTSG, including deflections and axial tube loads, and the local structural behavior (hole dilations). The key results of the FE 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 (OD) 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 staff finds that the assumptions made in the development of the FE model and the reported results of the structural analyses are reasonable for the transients and accidents addressed in BAW-2303P.

A mockup was constructed 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 as that described in BAW-10232P. 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 thermally cycled. The thermal cycles represent the effects of heat-up and cooldown cycles.

Testing was performed with a clean crevice between the OD of the tube samples and the tubesheet bore. Using a clean crevice was determined to be conservative, based on a summary of the results of a proprietary analysis conducted in 1999 using the same repair roll installation process as that currently used for the OTSGs. Because the licensee requested the removal of the restriction on lower tubesheet area rerolling, it presented the following results from tests performed to evaluate 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 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 staff 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.

The FE 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 the normal operating and selected 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 (tubesheet 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 leak criteria to qualify installation of a repair roll on a plant-specific basis. The repair roll is allowed to slip under specific faulted conditions. However, the licensee stated in its application that the proposed reroll repairs in qualified locations are not predicted to slip during faulted accident transients for ANO-1.

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.

To verify that the repair roll could withstand anticipated axial loads during normal operation and the selected accident conditions, applicable tubesheet bore dilations were achieved and an axial load was applied using a swage-lock fitting or an inside diameter gripper attached to the free end of the tube. A full circumferential sever was modeled for the testing, which is conservative for structural and leakage integrity since 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 one inch, which the licensee will use in ensuring that it maintains leakage below TS limits, as further discussed below. Based on the qualification program results, the staff considers the elimination of the licensee's current requirement that the reroll be one inch in length to be acceptable.

In its February 19, 2001, supplemental letter, the licensee stated that an additional 50 lbs is a negligible increase to the maximum compressive load due to the heatup. The normal heatup at ANO-1 occurs with the secondary water level at the mid-level of the SG. The mid-level heatup improves the heat transfer from the tubes to the shell; therefore, the resulting tube compressive loads are bound by the minimum water level heatup. The compressive loads vary as a function of radial position in the tubesheet with the maximum loads occurring in the periphery. Therefore, additional repair rolls could be installed at locations within the tube bundle without causing the compressive load at those locations to exceed the maximum allowed load for the worst-case tube. Additional repair rolls would be subject to review on a case-by-case basis, with the number of repair rolls and the configuration of the repair rolls limited to the maximum compressive load for the applicable design heatup plus an additional 50 lbs compression from the repair rolls. Based on this evaluation, the staff finds it acceptable to remove the limitation of only one reroll per SG tube that is included in the existing ANO-1 TS.

3.2 Structural and Leakage Integrity

Based on the results of the qualification testing, the licensee determined roll lengths sufficient to ensure adequate margins of structural and leakage integrity. The licensee determined the amount of slip for a tube with a new hardroll expansion based on the possible combination of loadings. A post-slip leak rate was applied to all repair rolls that have the potential to slip during faulted transients, regardless of whether a circumferential crack is actually present. The repair roll will not actually slip unless a large circumferential flaw is present. Field experience to date

has shown that the majority of the flaws in 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, the qualification tests predicted a steady-state leak rate for each repair roll. The staff 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. Since 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, the staff found that this is conservative and is an adequate approach. 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.

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 TS 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 OTSG.

By letter dated January 3, 2000, in accordance with TS, Entergy submitted a 90-day report of SG inservice inspection which was performed in refueling outage 15 (1R15). The licensee reported that flaws were identified in the heel transitions of rerolled joints which were installed in 1R14. The cracking has been classified as primary water stress corrosion cracking (PWSCC). The majority of the cracking is axial and volumetric indications with some circumferential indications. The circumferential and axial extent of the individual flaws are generally less than 0.30 inch. The licensee performed in-situ pressure tests on four reroll cracks with no leakage.

In its root cause study (described in the licensee's supplemental letter dated February 19, 2001), the licensee found that (1) final mill annealing temperature for the ANO-1 tubing may have been lower than the specifications - the higher the annealing temperature, the more resistant to PWSCC; (2) reroll repairs installed in other B&W plants did not show cracking in the heel transition after one cycle of operation; (3) the reroll process did not cause the cracking; (4) no measurable variations were found in material properties, reactor coolant system (RCS) chemistry data, or installation parameters that could explain the apparent differences in susceptibility to cracking; and (5) the results of the stress corrosion cracking tests suggest that increasing the installation of the repair roll depth from 3.25 inches to 3.75 inches may potentially lengthen the repair life.

To minimize future cracking at the heel transition region, the licensee plans to install future rerolls at a deeper elevation (i.e., away from the original roll joint). The licensee stated that since the heel transition is not a part of the qualified repair roll (i.e., outside the pressure boundary), cracking in the heel transition does not affect the structural integrity or leak-limiting

capability of the repair roll as long as the 1-inch repair roll and the toe transition (which is inside the pressure boundary) remain free of indications. In addition, as a part of the proposed amendment, the licensee will report circumferential cracking in reroll joints prior to restart and to perform an operational assessment, as discussed in Section 3.5 of this SE.

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3.3 Field Installation and Inspection

The licensee proposed to repair tubes in the same manner as those repairs performed under Revision 4 of BAW-2303P. The licensee will install either one or two hardroll joints (reroll) 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.

After the installation, the licensee will inspect all rerolls using a bobbin coil probe and a plus point probe to ensure proper diametral expansion and positioning of the reroll repair joint. In addition, the inspections will verify that the reroll regions are free of degradation. Any reroll not satisfying the acceptance criteria will be either plugged or repaired with a method other than rerolling. For future inservice inspections, the licensee will inspect all rerolled tubes with a plus point probe during SG inspection activities.

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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-2303P, Revision 4, does not evaluate the performance of rerolls following a LBLOCA. Instead, BAW-2303P, Revision 4, implicitly credits Topical Report BAW-2374, which provides risk-informed arguments to justify excluding the LBLOCA from consideration as a faulted condition. The staff has not approved BAW-2374 for referencing in a plant's licensing basis. However, the staff has reviewed the risk-informed arguments in BAW-2374 as they relate to the reroll repairs at ANO-1. The staff 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, which in the context of this amendment 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 a OTSG are small during a LBLOCA when compared to other licensing basis events. Therefore, excluding the pressure loads resulting from a LBLOCA would not result in a decrease in the existing structural margins. However, due to differential thermal expansion during a 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 staff 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 staff 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 a LBLOCA at the highest elevation of the hot legs assumed that tubes would be 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, Revision 4, 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 a 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 staff examined the engineering evaluation provided by the B&WOG. The staff 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 a LBLOCA scenario. The staff noted that the B&WOG estimate for leakage per slipped tube (0.06 gpm) appeared to be conservative. However, the staff concluded that the number of tubes expected to slip in the event of a 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 staff. Pending the completion of the review of BAW-2374, it is the staff'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 a LBLOCA would be acceptable.

3.4.1.1 Defense-in-Depth Considerations

BAW-2374 demonstrates that rerolls could slip and leak following a 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 a LBLOCA. For core damage and a large release 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 (ECCS) 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, the amount of leakage following a LBLOCA at ANO-1 would be acceptable. In this context, "acceptable leakage" means the best estimate leakage (in the event of a LBLOCA) would not result in a significant increase in radionuclide release (e.g., in excess of 10 CFR Part 100 limits). In addition, the leakage would not be expected to jeopardize long-term operation of the ECCS or accident management (e.g., control room habitability issues such as those addressed by GDC 19). For these reasons, the staff finds that defense-in-depth is maintained.

3.4.1.2 Safety Margins

BAW-2374 noted that the design and repair of OTSGs will continue to be governed by the requirements of Section III and Section XI of the ASME Code along with staff guidance provided in draft NRC RG 1.121, "Bases for Plugging Degraded PWR [Pressurized Water Reactor] 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 staff finds the evaluation of reroll performance during a 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 staff 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 licensee has committed to demonstrate that, based on the condition of the SGs, the amount of leakage following a LBLOCA at ANO-1 would be acceptable. In this context, "acceptable leakage" means the best estimate leakage (in the event of a LBLOCA) would not result in a significant increase in radionuclide release (e.g., in excess of 10 CFR Part 100 limits). In addition, the leakage would not be expected to jeopardize long-term operation of the ECCS or accident management (e.g., control room habitability issues such as those addressed by GDC 19). For these reasons, the staff finds that sufficient safety margins will be maintained at ANO-1 for reroll repairs in the event of a LBLOCA.

3.4.1.3 Change in Risk

BAW-2374 contains a bounding risk analysis to estimate the potential risk contribution (i.e., change in risk) by assuming a loss of OTSG 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 loss of coolant accident (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 RCS hot leg (i.e., "candy cane"). The RCS is refilled by the low pressure injection (LPI) subsystem of the ECCS, which induces a SGTR in the broken RCS loop. In the first scenario, secondary side isolation failure and failure of operators to make up water to the reactor building (RB) sump leads to eventual depletion of sump inventory through the secondary side, which causes ECCS failure, core damage and related 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 (BWST), which leads to early core damage and possible large early release.

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 (EOPs) 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 the CDF sequence), independent failure of LPI recirculation

(in the LERF sequence), and the conditional probability of large early release. The staff reviewed the probability assumptions for each basic event in the cutset equations for the two scenarios. Some accident initiators, such as the failure of SG access ports, were not discussed in BAW-2374 but the staff does not believe that the inclusion of these additional initiators would change the conclusions presented in this SE.

BAW-2374 uses an initiating event frequency of 8×10^{-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 (TW) cracks in piping based on historical experience data (NUREG/CR-5750) and the conditional probability of any rupture given a TW crack. This analysis assumed one TW crack to have occurred in a 36-inch diameter pipe, which was taken as conservative since, according to the topical report, "no TW cracks have been experienced in pipes larger than 8 inches." The staff does not accept this bases for establishing the estimated frequency for 36-inch pipe breaks of 8×10^{-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 staff concluded that the expected frequency for 36-inch pipe rupture is less than 1×10^{-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 staff, per the provisions of 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 1×10^{-6} per reactor-year or less. This probability is related to the frequency of failure of 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 1×10^{-6} per reactor-year. Hence, for the purpose of this safety evaluation, the staff accepts the B&WOG estimated frequency for 36-inch pipe breaks of 8×10^{-7} per calendar year.

It should be noted, however, that TW cracking was recently discovered in a 34-inch main coolant loop hot leg to reactor pressure vessel nozzle weld at the V. C. Summer (Westinghouse design) facility, which may call into question certain conclusions that have been made regarding the frequency of large-bore piping rupture. The NRC staff 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.

The licensee estimated the change in CDF to be 8×10^{-10} per reactor-year and the change in LERF to be 4×10^{-11} per reactor-year. The licensee concluded that these estimated changes in risk were very small according to the risk acceptance guidelines in RG 1.174. The staff believes that the omission of access port breaks and the potential for revised pipe crack estimates following the V. C. Summer experience will not cause predicted risk to exceed the RG 1.174 numerical guidelines. While not endorsing the estimates provided by the licensee, the staff does accept that the increase in risk is very small according to the risk acceptance guidelines in RG 1.174 and is, therefore, acceptable.

3.4.1.4 Compliance with Regulations

To determine if an exemption was necessary, the staff considered whether permitting rerolled tube joints to slip complies with the regulations. With respect to GDC-14, the staff concluded that the RCS pressure boundary at ANO-1 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 staff has concluded that the reroll repairs at ANO-1 are still designed and fabricated to the highest practicable standards as previously approved (on April 10, 1998) in Amendment No. 190. Therefore, GDC-30 is satisfied.

The staff also considered whether compliance with the requirements of 10 CFR 50.46 and Appendix K would be an issue at ANO-1, since 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's 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 breaks.

The staff 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 staff 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 staff reviewed. Based on this, the staff concludes that LOCA analyses provided in support of BAW-2374 meet the requirements of 10 CFR 50.46 for ANO-1, and are, therefore, acceptable.

Based on this analysis, the staff 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 staff also notes that the analyses described in the B&WOG's letter dated November 27, 2000, rely on operator action, as instructed by plant EOPs, to achieve and maintain long term core cooling as required by 10 CFR 50.46(b). In order to demonstrate the applicability of the BAW-2374 LOCA analyses, Entergy stated that it has verified that the ANO-1 plant-specific EOPs are consistent with the descriptions in BAW-2374 in regard to the key operator actions for mitigation of the accident sequence of concern. Entergy has also confirmed that the ANO-1 EOPs are consistent with the B&WOG's November 27, 2000, letter with respect to compliance with 10 CFR 50.46. The staff concludes that Entergy has sufficiently resolved the staff's concerns related to compliance with 10 CFR 50.46 for a LBLOCA and SBLOCA at ANO-1.

3.4.1.5 Integrated Decision Making

The staff has considered removal of LBLOCA loads from the reroll design, consistent with the proposed use of Topical Report BAW-2303P, Revision 4, with respect to the integrated decision making criteria in RG 1.174. The staff has concluded that this change is a practical solution for addressing the thermal loads caused by a LBLOCA and their impact on the design of the SG

repair method. Upon implementation of this amendment, the licensee will be able to use practical and acceptable repair methods (e.g., rerolls) at ANO-1 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 ANO-1.

3.4.1.6 Implementation and Monitoring

ANO-1 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 TS. Additionally, the ANO-1 TS contain RCS leakage limits and require plant shutdown if those limits are exceeded. As required by the plant's licensing basis as it applies to LBB approvals, ANO-1 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 ANO-1 will ensure a low probability of degradation of the RCS piping that could lead to a LBLOCA.

The SG tubes will continue to be inspected and plugged or repaired as required by the ANO-1 TS. 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, the staff has found that ANO-1 has an adequate inspection program for repaired tubes (including plugs) to verify that the primary-to-secondary leakage following a 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 ANO-1 TS will continue to require SG leakage limits and plant shutdown if the limits are exceeded. For these reasons, the staff has concluded that ANO-1'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 elements of risk-informed regulation. Staff review has determined that the risk-informed arguments in BAW-2374 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 ANO-1. BAW-2303P, Revision 4, implicitly relies upon BAW-2374 to exclude LBLOCA and only consider MSLB and SBLOCA as the faulted conditions. The staff 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, and uses 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 a 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 only have 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 staff 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 staff 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. The staff has concluded that permitting rerolls to slip during a 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 staff 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. The staff 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. The staff 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. ANO-1 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, ANO-1 has committed to demonstrate that, based on the condition of the SGs, an acceptable amount of leakage is expected in the event of a LBLOCA. Therefore, Principle 5 is satisfied.

Element 3: Element 3 recommends defining 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 staff has determined that the existing monitoring programs are sufficient to monitor the integrity of the RCS and SG tubes. However, ANO-1 will, in accordance with the regulatory commitment included in the supplemental letter dated February 19, 2001, 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 addresses submittal of the proposed change.

ANO-1 submitted a request for the change by letter dated September 28, 2000, through reference to BAW-2303P, Revision 4, which relies on the technical arguments in BAW-2374. The application was supplemented by letters dated October 26, 2000, and February 19, 2001. Therefore, Element 4 is satisfied.

3.4.2 Summary of LBLOCA considerations

In summary, the staff 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 ANO-1, would be a minor degradation in the SG tube pressure boundary in a condition where the pressure difference across the tubes is small.

The staff does not require the ANO-1 reroll repairs to consider the loads and dilations caused by a LBLOCA, and the staff 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. ANO-1 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 a 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 a LBLOCA) would not result in a significant increase in radionuclide release (e.g., in excess of 10 CFR Part 100 limits). In addition, the leakage would not be expected to jeopardize long-term operation of the ECCS or accident management (e.g., control room habitability issues such as those in addressed in GDC 19). The staff 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 ANO-1'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 ANO-1 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.

3.5 Regulatory Commitments

By supplemental letter dated February 19, 2001, the licensee made the following regulatory commitments:

1. Following each inservice inspection of steam generator tubes but prior to returning the ANO-1 steam generators to service, Entergy will verbally notify the NRC of the following:

    a. Indication of circumferential cracking inboard of the roll repair.

    b. Indication of circumferential cracking in the original roll or heat affected zone adjacent to the tube-to-tubesheet seal weld if no reroll is present.

    c. Determination of the best-estimate total leakage that would result from an analysis of the limiting LBLOCA based on circumferential cracking in the original tube-to-tubesheet rolls, tube-to-tubesheet reroll repairs, and heat affected zones of seal welds as found during each inspection.

2. Demonstrate that the primary-to-secondary leakage following a LBLOCA, as described in Appendix A to BAW-2374, is acceptable, based on the as-found condition of the SGs. 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 in the event of a LBLOCA that would not result in a significant increase of radionuclide release (e.g., in excess of 10 CFR 100 limits). A summary of this evaluation shall be provided to the NRC within 3 months following completion of steam generator tube inservice inspection with the report required by Technical Specification 4.18.6.

These regulatory commitments will ensure that the licensee 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 a 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 ANO-1. Entergy recognizes that further NRC review of BAW-2374 may require it to modify the regulatory commitment or otherwise involve additional actions to comply with the final NRC conclusion on the topical report. The staff has concluded that adequate controls for these actions are provided by the licensee's commitment management program and that additional regulatory requirements, unless defined as a result of the continuing review of BAW-2374, are not warranted.

3.6 Proposed Technical Specification Change

TS 4.18.5 would be changed as follows:

TS 4.18.5.a.4: This section is revised by adding that the reroll repair process can be used to repair tubes with defects in the upper and lower tubesheet areas as described in topical report, BAW-2303P, Revision 4. The section is also changed by adding "...or repaired by a rerolled joint..." to the definition of a degraded tube.

TS 4.18.5.a.7: This section is revised by deleting the current specifications for reroll repair.

TS 4.18.5.a.9: This section is revised by changing the rerolled region of the tube that will need to be inspected. The primary system pressure boundary of a tube will be changed after reroll repair is performed. Thus, the portion of the tube outboard of the new roll can be excluded from future inspection because it will be outside of the pressure boundary.

Bases (page 11): This page is revised by adding a reference to the implementation of reroll repair in accordance with BAW-2303P, Revision 4, and a definition of the outboard portion of a reroll joint.

The staff finds these changes acceptable based on the staff evaluation of BAW-2303P, Revision 4, and limited review of BAW-2374 for its application to the proposed reroll activity at the ANO-1, as discussed in this SE.

3.7 Summary

The licensee 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) regions of the upper or lower tubesheets. The technical basis for the proposed reroll method is documented in BAW-2303P, Revision 4.

The staff has determined that (1) the licensee'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 TS satisfy all regulatory requirements applicable to SG tube integrity.

On the basis of submitted information, the staff concludes that the proposed TS changes regarding reroll repair for degraded roll joints in the SGs at ANO-1 are acceptable because the licensee has demonstrated through an acceptable qualification program that the reroll satisfies GDC 14 of Appendix A to 10 CFR Part 50 and RG 1.121.

It should be noted, however, as indicated in Section 3.4.1.3 of this safety evaluation, TW 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 staff 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.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Arkansas State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

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. 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 (65 FR 77919 dated December 13, 2000). 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

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

Principal Contributor: John Tsao

Date: March 28, 2001

Mr. Craig G. Anderson
Vice President, Operations ANO
Entergy Operations, Inc.
1448 S. R. 333
Russellville, AR 72801

March 28, 2001

SUBJECT: ARKANSAS NUCLEAR ONE, UNIT NO. 1 - ISSUANCE OF AMENDMENT RE: THE USE OF THE REROLL REPAIR PROCESS FOR STEAM GENERATOR TUBES (TAC NO. MB0097)

Dear Mr. Anderson:

The Commission has issued the enclosed Amendment No. 212 to Facility Operating License No. DPR-51 for Arkansas Nuclear One, Unit No. 1 (ANO-1). The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated September 28, 2000, as supplemented by letter dated February 19, 2001. In addition, you provided in a letter dated October 26, 2000, a non-proprietary version of the topical report BAW-2303, "OTSG [Once-Through Steam Generator] Repair Roll Qualification Report," Revision 4 (ADAMS Accession No. ML003765879). The topical report provided part of your justification for this amendment.

The amendment revises the TS for ANO-1 to allow a revised reroll repair process for the steam generators. This amendment allows the reroll repair process to be used multiple times for a single tube and would allow the repairs in both the upper and lower tubesheets.

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

Sincerely,
/RA/
William Reckley, Project Manager, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-313

Enclosures: 1. Amendment No. 212 to DPR-51
2. Safety Evaluation

cc w/encls: See next page

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