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

December 15, 2000

Mr. William R. McCollum, Jr.
Vice President, Oconee Site
Duke Energy Corporation
7800 Rochester Highway
Seneca, SC 29672

SUBJECT: OCONEE NUCLEAR STATION, UNITS 1, 2 AND 3 RE: ISSUANCE OF AMENDMENTS (TAC NOS. MA9969, MA9970, AND MA9971)

Dear Mr. McCollum:

The Nuclear Regulatory Commission has issued the enclosed Amendment Nos. 318, 318, and 318 to Facility Operating Licenses DPR-38, DPR-47, and DPR-55, respectively, for the Oconee Nuclear Station, Units 1, 2, and 3. The amendments consist of changes to the Technical Specifications in response to your application dated September 12, 2000, as supplemented by letters dated October 4, October 26, November 10, and December 8, 2000.

The amendments revise the Technical Specification requirements related to the reroll repair process used to repair steam generator tubes. They also institute new license conditions.

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

Sincerely,

David E. LaBarge, Senior Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-269, 50-270, and 50-287

Enclosures:
1. Amendment No. 318 to DPR-38
2. Amendment No. 318 to DPR-47
3. Amendment No. 318 to DPR-55
4. Safety Evaluation

cc w/encls: See next page

NRR-058

December 15, 2000

Mr. William R. McCollum, Jr.
Vice President, Oconee Site
Duke Energy Corporation
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Seneca, SC 29679

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/RA/

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4. Safety Evaluation

cc w/encls: See next page

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To receive a copy of this document, indicate in the box C=Copy w/o attachment/enclosure E=Copy with attachment/enclosure N = No copy *See parallel concurrence **See previous concurrence

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Oconee Nuclear Station

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

DUKE ENERGY CORPORATION

DOCKET NO. 50-269

OCONEE NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 318
License No. DPR-38

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

   A. The application for amendment to the Oconee Nuclear Station, Unit 1 (the facility) Facility Operating License No. DPR-38 filed by the Duke Energy Corporation (the licensee) dated September 12, 2000, as supplemented October 4, October 26, November 10, and December 8, 2000, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as 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 set forth in 10 CFR Chapter I;

   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 hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 3.B of Facility Operating License No. DPR-38 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 318 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. In addition, until the steam generators are replaced, the license is amended to add the following License Conditions:

5. Steam Generator Circumferential Crack Report

Following each inservice inspection of steam generator tubes, the NRC shall be notified of the following prior to returning the steam generators to service:

a. Indication of circumferential cracking in the secondary side roll (lower roll in the upper tubesheet or upper roll in the lower tubesheet) if rerolled.

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 Large Break Loss of Coolant Accident (LBLOCA) based on circumferential cracking in the original tube-to-tubesheet rolls, tube-to-tubesheet rerolls, and heat affected zones of seal welds as found during each inspection.

6. 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 5.6.8, Item b.

4. This license amendment is effective as of its date of issuance and shall be implemented within 30 days of the date of issuance, including implementation of the commitment described in Section 3.4.1.4 of the safety evaluation related to this amendment.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard L. Emch, Jr.

Richard L. Emch, Jr., Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Technical Specification Changes

Date of Issuance: December 15, 2000

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

DUKE ENERGY CORPORATION

DOCKET NO. 50-270

OCONEE NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 318
License No. DPR-47

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

   A. The application for amendment to the Oconee Nuclear Station, Unit 2 (the facility) Facility Operating License No. DPR-47 filed by the Duke Energy Corporation (the licensee) dated September 12, 2000, as supplemented October 4, October 26, November 10, and December 8, 2000, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as 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 set forth in 10 CFR Chapter I;

   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 hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 3.B of Facility Operating License No. DPR-47 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 318 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. In addition, until the steam generators are replaced, the license is amended to add the following License Conditions:

5. Steam Generator Circumferential Crack Report

Following each inservice inspection of steam generator tubes, the NRC shall be notified of the following prior to returning the steam generators to service:

a. Indication of circumferential cracking in the secondary side roll (lower roll in the upper tubesheet or upper roll in the lower tubesheet) if rerolled.

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 Large Break Loss of Coolant Accident (LBLOCA) based on circumferential cracking in the original tube-to-tubesheet rolls, tube-to-tubesheet rerolls, and heat affected zones of seal welds as found during each inspection.

6. 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 5.6.8, Item b.

4. This license amendment is effective as of its date of issuance and shall be implemented within 30 days of the date of issuance, including implementation of the commitment described in Section 3.4.1.4 of the safety evaluation related to this amendment.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard L. Emch, Jr.

Richard L. Emch, Jr., Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Technical Specification Changes

Date of Issuance: December 15, 2000

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

DUKE ENERGY CORPORATION

DOCKET NO. 50-287

OCONEE NUCLEAR STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 318
License No. DPR-55

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

   A. The application for amendment to the Oconee Nuclear Station, Unit 3 (the facility) Facility Operating License No. DPR-55 filed by the Duke Energy Corporation (the licensee) dated September 12, 2000, as supplemented October 4, October 26, November 10, and December 8, 2000, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as 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 set forth in 10 CFR Chapter I;

   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 hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 3.B of Facility Operating License No. DPR-55 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 318 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

4. In addition, until the steam generators are replaced, the license is amended to add the following License Conditions:

5. Steam Generator Circumferential Crack Report

Following each inservice inspection of steam generator tubes, the NRC shall be notified of the following prior to returning the steam generators to service:

a. Indication of circumferential cracking in the secondary side roll (lower roll in the upper tubesheet or upper roll in the lower tubesheet) if rerolled.

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 Large Break Loss of Coolant Accident (LBLOCA) based on circumferential cracking in the original tube-to-tubesheet rolls, tube-to-tubesheet rerolls, and heat affected zones of seal welds as found during each inspection.

6. 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 5.6.8, Item b.

4. This license amendment is effective as of its date of issuance and shall be implemented within 30 days of the date of issuance, including implementation of the commitment described in Section 3.4.1.4 of the safety evaluation related to this amendment.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard L. Emch, Jr.

Richard L. Emch, Jr., Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Technical Specification Changes

Date of Issuance: December 15, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 318

FACILITY OPERATING LICENSE NO. DPR-38

DOCKET NO. 50-269

AND

TO LICENSE AMENDMENT NO. 318

FACILITY OPERATING LICENSE NO. DPR-47

DOCKET NO. 50-270

AND

TO LICENSE AMENDMENT NO. 318

FACILITY OPERATING LICENSE NO. DPR-55

DOCKET NO. 50-287

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
LOEP1	LOEP1
LOEP8	LOEP8
5.0-18	5.0-18

OCONEE NUCLEAR STATION
TECHNICAL SPECIFICATIONS
LIST OF EFFECTIVE PAGES

PAGE	AMENDMENT	REVISION DATE
LOEP1	318/318/318	12/15/00
LOEP2	300/300/300	12/16/98
LOEP3	315/315/315	09/18/00
LOEP4	309/309/309	1/18/00
LOEP5	314/314/314	09/06/00
LOEP6	309/309/309	1/18/00
LOEP7	300/300/300	12/16/98
LOEP8	318/318/318	12/15/00
LOEP9	310/310/310	1/18/00
i	300/300/300	12/16/98
ii	315/315/315	09/18/00
iii	309/309/309	1/18/00
iv	309/309/309	1/18/00
1.1-1	300/300/300	12/16/98
1.1-2	300/300/300	12/16/98
1.1-3	300/300/300	12/16/98
1.1-4	300/300/300	12/16/98
1.1-5	300/300/300	12/16/98
1.1-6	300/300/300	12/16/98
1.2-1	300/300/300	12/16/98
1.2-2	300/300/300	12/16/98
1.2-3	300/300/300	12/16/98
1.3-1	300/300/300	12/16/98
1.3-2	300/300/300	12/16/98
1.3-3	300/300/300	12/16/98
1.3-4	300/300/300	12/16/98
1.3-5	300/300/300	12/16/98
1.3-6	300/300/300	12/16/98
1.3-7	300/300/300	12/16/98
1.3-8	300/300/300	12/16/98
1.3-9	300/300/300	12/16/98
1.3-10	300/300/300	12/16/98
1.3-11	300/300/300	12/16/98
1.3-12	300/300/300	12/16/98
1.3-13	300/300/300	12/16/98
1.4-1	300/300/300	12/16/98
1.4-2	300/300/300	12/16/98
1.4-3	300/300/300	12/16/98
1.4-4	300/300/300	12/16/98
2.0-1	313/313/313	6/21/00

OCONEE NUCLEAR STATION
TECHNICAL SPECIFICATIONS
LIST OF EFFECTIVE PAGES

PAGE	AMENDMENT	REVISION DATE
3.9.3-2	300/300/300	12/16/98
3.9.4-1	300/300/300	12/16/98
3.9.4-2	300/300/300	12/16/98
3.9.5-1	300/300/300	12/16/98
3.9.5-2	300/300/300	12/16/98
3.9.6-1	300/300/300	12/16/98
3.9.6-2	300/300/300	12/16/98
3.9.7-1	309/309/309	1/18/00
3.9.7-2	309/309/309	1/18/00
3.10.1-1	300/300/300	12/16/98
3.10.1-2	300/300/300	12/16/98
3.10.1-3	300/300/300	12/16/98
3.10.1-4	300/300/300	12/16/98
3.10.1-5	300/300/300	12/16/98
3.10.2-1	300/300/300	12/16/98
3.10.2-2	300/300/300	12/16/98
3.10.2-3	300/300/300	12/16/98
4.0-1	313/313/313	6/21/00
4.0-2	300/300/300	12/16/98
5.0-1	300/300/300	12/16/98
5.0-2	300/300/300	12/16/98
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5.0-8	310/310/310	1/18/00
5.0-9	310/310/310	1/18/00
5.0-10	310/310/310	1/18/00
5.0-11	310/310/310	1/18/00
5.0-12	310/310/310	1/18/00
5.0-13	310/310/310	1/18/00
5.0-14	310/310/310	1/18/00
5.0-15	310/310/310	1/18/00
5.0-16	310/310/310	1/18/00
5.0-17	310/310/310	1/18/00
5.0-18	318/318/318	12/15/00
5.0-19	310/310/310	1/18/00
5.0-20	310/310/310	1/18/00
5.0-21	310/310/310	1/18/00
5.0-22	310/310/310	1/18/00
5.0.23	310/310/310	

5.5 Programs and Manuals

5.5.10 Steam Generator (SG) Tube Surveillance Program (continued)

3. Degraded Tube means a tube or a sleeve containing imperfections ≥ 20% of the nominal wall thickness caused by degradation.

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

5. Defect means an imperfection of such severity that it exceeds the repair limit. A tube or sleeve containing a defect is defective.

6. Repair Limit means the imperfection depth beyond which the tube shall be either removed from service by plugging or repaired by sleeving or rerolling because it may become unserviceable prior to the next inspection; it is equal to 40% of the nominal tube or sleeve wall thickness. Axial tube imperfections of any depth observed between the primary side surface of the tube sheet clad and the end of the tube are excluded from this repair limit.

   The Babcock and Wilcox process (or method) equivalent to the method described in report, BAW-1823P, Revision 1 will be used for sleeving repairs.

   The new roll area must be free of degradation in order for the repair to be considered acceptable. The rerolling process used by Oconee is described in the Topical Report, BAW-2303P, Revision 4.

7. 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 5.5.10.d.

8. Tube Inspection means an inspection of the steam generator tube from the point of entry completely to the point of exit. The degraded tube above the new roll area can be excluded from future periodic inspection requirements because it is no longer part of the pressure boundary once the repair roll is installed.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 318 TO FACILITY OPERATING LICENSE DPR-38

AMENDMENT NO. 318 TO FACILITY OPERATING LICENSE DPR-47

AND AMENDMENT NO. 318 TO FACILITY OPERATING LICENSE DPR-55

DUKE ENERGY CORPORATION

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3

DOCKET NOS. 50-269, 50-270, AND 50-287

1.0 INTRODUCTION

By letter dated September 12, 2000, as supplemented October 4, October 26, November 10, and December 8, 2000, Duke Energy Corporation (the licensee) submitted a request for changes to the Oconee Nuclear Station, Units 1, 2, and 3, Technical Specifications (TS). The requested changes would revise TS 5.5.10, Item e.6 by (1) removing the restriction on lower tube sheet area rerolling, (2) removing the limitation of only one reroll per once through steam generator (OTSG) tube, (3) eliminating the requirement that the reroll be one inch in length, and (4) changing the revision number for Topical Report BAW-2303P, OTSG Repair Roll Qualification Report, from Revision 3 to Revision 4. The licensee and its vendor, Framatome Technologies Incorporated (FTI), revised Topical Report, BAW-2303P, Revision 3, by submitting Topical Report, BAW-2303P, Revision 4, as the technical justification for the requested license amendments in order to implement a tubesheet region repair roll in degraded tubes in OTSGs. The supplements dated October 4, October 26, and November 10, and December 8, 2000, provided clarifying information that did not change the scope of the September 12, 2000, application nor the initial proposed no significant hazards consideration determination.

The NRC staff approved the use of repair rolls in the upper tubesheet, as analyzed in BAW-2303P, Revision 3, but reanalysis became necessary due to identification of a Small Break Loss of Coolant Accident (SBLOCA) that was more limiting than the accident previously evaluated in Revision 3. 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, FTI described its analyses for repair rolls for installation in both the upper and lower tubesheets and multiple repair rolls in a single tube. The analysis in 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 Revision 3 of this report remain acceptable based on the criteria contained in 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 Babcock & Wilcox Owners Group (B&WOG) by letter dated July 7, 2000. BAW-2374 provides the risk-informed bases for excluding Large Break Loss of Coolant Accident (LBLOCA) from the design considerations. BAW-2374 explains that rerolled tubes may slip in the tubesheet during a LBLOCA if there is a degradation (such as circumferential cracking) in the tube that prevents the 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 Oconee may slip during a LBLOCA, resulting in an increase in leakage past the reroll. Oconee has proposed license conditions to demonstrate that the expected leakage following a LBLOCA is acceptable, based on the as-found condition of their SGs. 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

Each Oconee unit 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 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) 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 of the American Society of Mechanical Engineers. 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 Oconee, the licensee performed a qualification program presented in Revision 3 of BAW-2303P 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 main steam line break 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 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.

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 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.

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.

Finite element 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 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 (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 determined plant-specific exclusion zones for repair roll.

In its response dated November 10, 2000, to the staff's request for additional information (RAI) concerning the potential flow-induced loadings on SG tubes and the resulting effects on the initial conditions for the input to the FE thermal analyses for the determination of the axial tube loads and the differential dilations, the licensee provided a summary of the results of a proprietary analysis. The licensee 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 tube. The resulting bowing of the tubes imparts an axial tension load of approximately one-tenth of the axial load due to the maximum temperature differential that occurs approximately 10 minutes into the transient at Oconee. The axial load due to cross-flow was calculated based on the maximum lateral displacement of the tubes. This additional length results in a slight decrease in axial tube loads at the time of maximum tube-to-shell temperature difference that occurs approximately 10 minutes into the MSLB transient. The licensee concluded that the evaluation of a tube without accounting for the plastic deformation was a conservative approach.

The staff finds that the assumptions made in the development of the FE model and the results of the structural analyses are reasonable. The licensee used the results of the structural analyses to construct a mockup.

The mockup 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 identical to that of 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 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 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 (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.

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. By removing the heel transition, the tested condition represented a complete circumferential sever at the end of the effective roll (primary side). 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 accident conditions, applicable tubesheet bore dilations were achieved and an axial load was applied using a swage-lock fitting or an ID 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.

As noted in the October 26, 2000, RAI response, the number of rerolls permitted per tube is determined by evaluating the acceptable maximum tube loads. The licensee has decided to limit repair rolls at the Oconee plants to a configuration resulting in a maximum of 50 pounds additional compressive load per tube. This configuration would result in allowing the installation of only two single rerolls or one double reroll per tube. The staff finds this limitation on the number of rerolls per tube acceptable because the additional load from the additional reroll is not significant compared to the transient loads considered in the licensee's analyses. Based on this evaluation, the staff considers the removal of the limitation of only one reroll per SG tube acceptable.

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°, 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. 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.

3.3 Field Installation and Inspection

The licensee proposed to repair tubes in an identical manner to those repairs performed under Revision 3 of BAW-2303P. This method is the installation of 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 eddy current techniques 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 during SG inspection activities.

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 a LBLOCA. Instead, BAW-2303, 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 Oconee. The staff performed its review in accordance with Regulatory Guide 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 these amendments is used to support the technical basis of the proposed reroll design specified in Topical Report 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 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 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 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. 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 accepable.

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 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 amendments include a license condition (see Section 3.5 of this SE) that requires the licensee to demonstrate that, based on the condition of the SGs, an acceptable amount of leakage would be expected in the event of a LBLOCA at Oconee. 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). 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 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 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 license condition included in the proposed amendments (see Section 3.5 of this SE) will require the licensee to demonstrate, based on the condition of its SGs, that an acceptable amount of leakage would be expected in the event of a LBLOCA. 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). For these reasons, the staff finds that sufficient safety margins will be maintained at the Oconee Nuclear Station, Units 1, 2, and 3 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 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 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 makeup water to the reactor building (RB) 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 (BWST), which leads to early core damage and possible large early release. The staff 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 (EOPs) to terminate feedwater flow to the OTSG.

The change 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 CDF sequence), independent failure of LPI recirculation (in 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 and determined that conservative probability estimates for all of the basic events were used in the quantitative risk analysis.

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 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 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 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 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 (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.

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 8×10^{-10} per reactor-year and the change in LERF was estimated to be 4×10^{-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 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 Oconee 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 Oconee are still designed and fabricated to the highest practicable standards as previously approved (on November 21, 1997) in Amendment Nos. 227, 227, and 224 for Oconee Nuclear Station, Units 1, 2, and 3, respectively. 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 Oconee, 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 beaks.

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 Oconee Nuclear Station, Units 1, 2, and 3, 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 November 27, 2000, letter rely on operator action, as instructed by plant Emergency Operating Procedures (EOPs), 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, by letter dated December 8, 2000, Oconee committed to verify that the plant-specific EOPs are consistent with the statements contained in the November 27, 2000, letter. The staff concludes that this commitment, which the staff has included as an implementation step for these amendments, is sufficient to resolve the concerns related to compliance with 10 CFR 50.46 for a LBLOCA and SBLOCA at Oconee.

3.4.1.5 Integrated Decision Making

The staff 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 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 these amendments, the licensee will be able to use practical and acceptable repair methods (e.g., rerolls) at Oconee Nuclear Station, Units 1, 2, and 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 Oconee.

3.4.1.6 Implementation and Monitoring

Oconee 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 safety evaluation can be broken down into two areas: (1) those that apply to RCS piping and (2) those that apply to 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 Oconee 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, Oconee 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 Oconee 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 Oconee 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, Oconee must have 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 safety evaluation. Finally, the Oconee TS will continue to require SG leakage limits and plant shutdown if the limits are exceeded. For these reasons, the staff has concluded that Oconee's inspection and monitoring programs ensure that the SG primary-to-secondary pressure boundary will be adequately maintained to support the conclusions of this safety evaluation.

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 Oconee. 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 a qualitative and quantitative analyses, 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. Oconee 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, Oconee will be required 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 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 staff has determined that the existing monitoring programs are sufficient to monitor the integrity of the RCS and SG tubes. However, Oconee must 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.

Oconee submitted a request for the change by letter dated September 12, 2000, through reference to Topical Report BAW-2303P, Revision 4, which relies on the technical arguments in BAW-2374. 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 Oconee Nuclear Station, Units 1, 2, and 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 staff does not require the Oconee 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. Oconee has proposed a new license condition to 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). 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 Oconee'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 Oconee 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 staff finds that the LBLOCA, as described in Appendix A to BAW-2374, does not need to be considered in the design of Oconee's reroll repairs provided the license conditions and commitment contained in letter dated December 8, 2000, are implemented.

3.5 New License Conditions

By letter dated December 8, 2000, the licensee accepted the addition of the following license conditions to Oconee Facility Operating License Nos. DPR-38, DPR-47, and DPR-55:

3. In addition, until the steam generators are replaced, the license is amended to add the following License Conditions:

    5. Steam Generator Circumferential Crack Report

    Following each inservice inspection of steam generator tubes, the NRC shall be notified of the following prior to returning the steam generators to service:

    a. Indication of circumferential cracking in the secondary side roll (lower roll in the upper tubesheet or upper roll in the lower tubesheet) if rerolled.

    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 Large Break Loss of Coolant Accident (LBLOCA) based on circumferential cracking in the original tube-to-tubesheet rolls, tube-to-tubesheet rerolls, and heat affected zones of seal welds as found during each inspection.

6. 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 5.6.8, Item b.

These license conditions 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. 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 Oconee Nuclear Station, Units 1, 2, and 3.

3.6 Proposed Technical Specification Change

TS 5.5.10.6 would be changed to read as follows:

> Repair Limit means the imperfection depth beyond which the tube shall be either removed from service by plugging or repaired by sleeving or rerolling because it may become unserviceable prior to the next inspection; it is equal to 40% of the nominal tube or sleeve wall thickness. Axial tube imperfections of any depth observed between the primary side surface of the tube sheet clad and the end of the tube are excluded from this repair limit.
>
> The Babcock and Wilcox process (or method) equivalent to the method described in report, BAW-1823P, Revision 1 will be used for sleeving repairs.
>
> The new roll area must be free of degradation in order for the repair to be considered acceptable. The rerolling process used by Oconee is described in the Topical Report, BAW-2303P, Revision 4.

The proposed TS change would (a) remove the restriction on lower tube sheet area rerolling, (b) remove the limitation of only one reroll per SG tube, (c) eliminate the requirement that the reroll be one inch in length, and (d) change the revision number for Topical Report BAW-2303P, from Revision 3 to Revision 4. The staff finds these changes acceptable based on the staff evaluation of Topical Report BAW-2303P, Revision 4 and limited review of Topical Report BAW-2374 for its application to the proposed reroll activity at the Oconee Nuclear Station, Units 1, 2, and 3, as discussed in this safety evaluation.

4.0 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 topical report 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 Oconee Nuclear Station, Units 1, 2, and 3 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 SE, 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 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.

5.0 STATE CONSULTATION

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

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements 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 amendments involve 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 amendments involve no significant hazards consideration, and there has been no public comment on such finding (65 FR 59222). Accordingly, the amendments meet 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 amendments.

7.0 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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

8.0 REFERENCES

1. NRC Draft Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," August 1976.

2. Letter from D. E. LaBarge, USNRC to W. R. McCollum, Duke Energy Corporation, "Issuance of Amendments - Oconee Nuclear Station, Units 1, 2, and 3 (TAC Nos. M99779, M99780, and M99781), November 21, 1997.

3. Framatome Technologies Inc. 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.

4. Framatome Technologies Inc. Topical Report, BAW-2303P, Revision 3, "OTSG Repair Roll Qualification Report," October 1997. (Proprietary)

5. Framatome Technologies Inc. Topical Report, BAW-2303P, Revision 4, "OTSG Repair Roll Qualification Report," August 2000. (Proprietary)

6. Letter from W. R. McCollum, Jr., Duke Energy Corporation to USNRC, "License Amendment Request for Technical Specification 5.5.10e.6, Steam Generator Tube Surveillance Program (TSCR 2000-07)," September 12, 2000.

7. Letter from W. R. McCollum, Jr., Duke Energy Corporation to USNRC, "License Amendment Request for Technical Specification 5.5.10e.6, Steam Generator Tube Surveillance Program (TSCR 2000-07)," October 4, 2000.

8. Letter from W. R. McCollum, Jr., Duke Energy Corporation to USNRC, "Response to NRC Questions on License Amendment Request for Technical Specification 5.5.10e.6 and Topical Report BAW-2303P, Revision 4," October 26, 2000.

9. Letter from W. R. McCollum, Jr., Duke Energy Corporation to USNRC, "Response to NRC Questions on License Amendment Request for Technical Specification 5.5.10e.6 and Topical Report BAW-2303P, Revision 4," November 10, 2000.

10. Letter from W. R. McCollum, Jr., Duke Energy Corporation to USNRC, "Supplemental Information Regarding License Amendment Request for Technical Specification 5.5.10e.6 and Topical Report BAW-2303P, Revision 4," December 8, 2000.

11. Letter from D. Firth (Framatome) to U.S. Nuclear Regulatory Commission, "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.

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