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Mr. W. R. McCollum
Vice President, Oconee Site
Duke Energy Corporation
P. O. Box 1439
Seneca, SC 29679

SUBJECT: ISSUANCE OF AMENDMENTS - OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3 (TAC NOS. M99779, M99780, and M99781)

Dear Mr. McCollum:

The Nuclear Regulatory Commission has issued the enclosed Amendments Nos. 227, 227, and 224, to Facility Operating Licenses DPR-38, DPR-47, and DPR-55, respectively, for the Oconee Nuclear Station, Units 1, 2, and 3. The amendments are in response to your application dated October 20, 1997, as supplemented by letters dated November 3, 6, and 10, 1997.

The amendments revise Technical Specifications to implement alternate repair criteria for steam generator tubes that have degraded roll joints inside of the upper tubesheet. The alternate repair criteria would allow new roll joints to be installed below the degraded roll joints in the upper tubesheet.

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,

Original signed by:

David E. LaBarge, Senior Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

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

Enclosures:
1. Amendment No. 227 to DPR-38
2. Amendment No. 227 to DPR-47
3. Amendment No. 224 to DPR-55
4. Safety Evaluation

cc w/encl: See next page

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

November 21, 1997

Mr. William R. McCollum
Vice President, Oconee Site
Duke Energy Corporation
P. O. Box 1439
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cc w/encl: See next page

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. 227
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 October 20, 1997, as supplemented by letters dated November 3, 6, and 10, 1997, comply 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:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 227, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: November 21, 1997

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. 227
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 October 20, 1997, as supplemented by letters dated November 3, 6, and 10, 1997, comply 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:

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 227 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: November 21, 1997

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. 224
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 October 20, 1997, as supplemented by letters dated November 3, 6, and 10, 1997, comply 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:

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 224 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: November 21, 1997

ATTACHMENT TO LICENSE AMENDMENT NO. 227

FACILITY OPERATING LICENSE NO. DPR-38

DOCKET NO. 50-269

AND

TO LICENSE AMENDMENT NO. 227

FACILITY OPERATING LICENSE NO. DPR-47

DOCKET NO. 50-270

AND

TO LICENSE AMENDMENT NO. 224

FACILITY OPERATING LICENSE NO. DPR-55

DOCKET NO. 50-287

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

Remove Pages	Insert Pages
3.1-14	3.1-14
4.17-1	4.17-1
4.17-2	4.17-2
4.17-3	4.17-3
4.17-4	4.17-4
4.17-5	4.17-5
———	4.17-5a

3.1.6 Leakage

Specification

3.1.6.1 If the total reactor coolant leakage rate exceeds 10 gpm, the reactor shall be shutdown within 24 hours of detection.

3.1.6.2 If unidentified reactor coolant leakage (excluding normal evaporative losses) exceeds 1 gpm or if any reactor coolant leakage is evaluated as unsafe, the reactor shall be shutdown within 24 hours of detection.

3.1.6.3 If any reactor coolant leakage exists through a non-isolable fault in a RCS strength boundary (such as the reactor vessel, piping, valve body, etc., except the steam generator tubes), the reactor shall be shutdown, and cooldown to the cold shutdown condition shall be initiated within 24 hours of detection.

3.1.6.4 If the total leakage through the tubes of any one steam generator equals or exceeds 150 gallons per day, a reactor shutdown shall be initiated within 4 hours and the reactor shall be in a cold condition within the next 36 hours.

3.1.6.5 If reactor shutdown is required by Specification 3.1.6.1, 3.1.6.2 or 3.1.6.3, the rate of shutdown and the conditions of shutdown shall be determined by the safety evaluation for each case and justified in writing as soon thereafter as practicable.

3.1.6.6 Action to evaluate the safety implication of reactor coolant leakage shall be initiated within 4 hours of detection. The nature, as well as the magnitude, of the leak shall be considered in this evaluation. The safety evaluation shall assure that the exposure of offsite personnel to radiation is within the guidelines of 10 CFR 20.

3.1.6.7 If reactor shutdown is required per Specification 3.1.6.1, 3.1.6.2, 3.1.6.3 or 3.1.6.4, the reactor shall not be restarted until the leak is repaired or until the problem is otherwise corrected.

3.1.6.8 When the reactor is critical and above 2% power, two reactor coolant leak detection systems of different operating principles shall be operable, with one of the two systems sensitive to radioactivity. The systems sensitive to radioactivity may be out-of-service for 48 hours provided two other means to detect leakage are operable.

3.1.6.9 Loss of reactor coolant through reactor coolant pump seals and system valves to connecting systems which vent to the gas vent header and from which coolant can be returned to the reactor coolant system shall not be considered as reactor coolant leakage and shall not be subject to the consideration of Specifications 3.1.6.1, 3.1.6.2, 3.1.6.3, 3.1.6.4, 3.1.6.5, 3.1.6.6 or 3.1.6.7 except that such losses when added to leakage shall not exceed 30 gpm.

3.1.6.10

a. The maximum allowable leakage for valves CF-12, CF-14, LP-47 and LP-48 shall be as follows:

4.17 STEAM GENERATOR TUBING SURVEILLANCE

Applicability

Applies to the surveillance of tubing of each steam generator.

Objective

To ensure integrity of the steam generator tubing through a defined inservice surveillance program, and to minimize exposure of personnel to radiation during performance of the surveillance program.

Specification

4.17.1 Examination Methods

Inservice inspection of steam generator tubing shall include non-destructive examination by eddy-current testing or other equivalent techniques. The inspection equipment shall provide a sensitivity that will detect defects with a penetration of 20 percent or more of the minimum allowable as-manufactured tube wall thickness.

4.17.2 Acceptance Criteria

The steam generator shall be considered operable after completion of the specified actions. All tubes examined exceeding the repair limit shall be repaired by sleeving or rerolling or removed from service (e.g., plugged, stabilized).

4.17.3 Selection and Testing

The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.17.1. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.17.4 and the inspected tubes shall be verified acceptable per Specification 4.17.5. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in both steam generators, with one or both steam generators being inspected. The tubes selected for these inspections shall be selected on a random basis except:

a. The first sample inspection during each inservice inspection of each steam generator shall include:

   1. All tubes that previously had detectable wall penetrations (>20%) and have not been plugged or sleeve repaired in the affected area.

   2. At least 50% of the tubes inspected shall be in those areas where experience has indicated potential problems.

   3. A tube adjacent to any selected tube which does not permit passage of the eddy current probe for tube inspection.

b. Tubes in the following Group(s) may be excluded from the first sample if all tubes in a Group in both OTSG are inspected. No credit will be taken for these tubes in meeting minimum sample size requirements.

(1) Group A-1: Tubes within one, two, or three rows of the open inspection lane.

c. All tubes which have been repaired using the reroll process will have the new roll area inspected during the inservice inspection.

d. The tubes selected as the second and third samples (if required by Table 4.17-1) during each inservice inspection may be subjected to less than a full tube inspection provided:

1. The tubes selected for these samples include the tubes from those areas of the tubesheet array where tubes with imperfections were previously found.

2. The inspections include those portions of the tubes where imperfections were previously found.

The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but no more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

NOTES:

(1) In all inspections, previously degraded tubes must exhibit significant (>10%) further wall penetrations to be included in the above percentage calculations.

(2) Where special inspections are performed pursuant to 4.17.3.b, defective or degraded tubes found as a result of the inspection shall be included in determining the Inspection Results Category for that special inspection but need not be included in determining the Inspection Results Category for the general steam generator inspection, unless the mechanism of degradation is random in nature.

(3) Where special inspections are performed pursuant to 4.17.3.c, defective or degraded tube indications found in the new roll area as a result of the inspection and any indications found in the originally rolled region of the rerolled tubes, need not be included in determining the Inspection Results Category for the general steam generator inspection.

4.17.4 Inspection Intervals

The above required inservice inspections of steam generator tubes shall be performed at the following frequencies.

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

b. If the results of the inservice inspection of a steam generator performed in accordance with Table 4.17-1 at 40 month intervals fall in Category C-3, subsequent inservice inspections shall be performed at intervals of not less than 10 months nor more than one fuel cycle after the previous inspection. The increase in inspection frequency shall apply until a subsequent inspection meets the conditions specified in 4.17.4.a and the interval can be extended to a maximum of 40 months.

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.17-1 during the shutdown subsequent to any of the following conditions:

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

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

   3. A main steam line or feedwater line break.

d. After primary to secondary leakage in excess of the limits of Specification 3.1.6, an inspection of the affected steam generator will be performed in accordance with the following criteria:

   1. If the leaking tube is in a Group as defined in Section 4.17.3.b, all of the tubes in this Group in this steam generator will be inspected. If the results of this inspection fall into the C-3 category, additional inspections will be performed in the same Group in the other steam generator.

   2. If the leaking tube has been repaired by the reroll process and is leaking in the new roll area, all of the tubes in the steam generator that have been repaired by the reroll process will have the new roll area inspected. If the results of this inspection fall into the C-3 category, additional inspections will be performed in the new roll area in the other steam generator.

   3. If the leaking tube is not in a Group as defined in 4.17.4.d.1, then an inspection will be performed on the affected steam generator in accordance with Table 4.17-1 with an initial inspection sample size of 6% of the tubes in the affected steam generator.

4.17.5 Definitions

As used in this specification:

a. 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 or sleeve wall thickness, if detectable, may be considered as imperfections.

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

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

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

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

f. 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.

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 rerolling repair process will only be used to repair tubes with defects in the upper tubesheet area. The rerolling repair process will be performed only once per steam generator tube using a 1 inch reroll length. 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 3.

g. 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 4.17.4.

h. 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.

4.17.6 Reports

a. The number of tubes plugged or repaired in each steam generator shall be reported to the NRC within 30 days following the completion of the plugging or repair procedure.

b. The results of the steam generator tube inservice inspection shall be reported to the NRC within 3 months following completion of the inspection. This report shall include:

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of a degraded tube.

3. Identification of tubes plugged or repaired.

4. Number of tubes repaired by rerolling and number of indications detected in the new roll area of the repaired tubes.

c. Results of steam generator tube inspections which fall into Category C-3 and require prompt notification of the NRC shall be reported pursuant to Specification 6.6.2.1.a prior to resumption of plant operation. The written followup of this report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

Bases

The program of periodic inservice inspection of steam generators provides the means to monitor the integrity of the tubing and to maintain surveillance in the event there is evidence of mechanical damage or progressive deterioration due to design, manufacturing errors, or operating conditions. Inservice inspection of the steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures may be taken.

Repair or removal from service will be required for any tube with service-induced metal loss in excess of 40% of the tube or sleeve nominal wall thickness or with a through wall crack. Additional corrective actions may be required to stabilize a circumferentially cracked tube.

The initial sample of tubes inspected in a steam generator includes tubes from three groups. First, lane tubes are inspected to assure their integrity. Second, all other inservice tubes with degradation, inspected in previous inspections, are inspected to assure tube integrity and determine degradation growth, if any. Third, a random sample of 3% of the total number of tubes in both steam generators is inspected. The results of the latter inspection dictate the extent of further examinations.

An objective of this Specification is to provide an inspection plan which will insure, with a high degree of confidence, that no more than 30 defective tubes will remain in a steam generator after an initial C-3 category inspection.

Following an 18% random inspection (C-3 category inspection) an unaffected area is identified. The unaffected area will be logically and consistently defined based on generator design, defect location and characteristics. The criteria for accepting an area as unaffected depend on the number of defects found in the sample inspected in that area and are established such that there is a 0.05 or smaller probability of accepting the area as unaffected if it contains 30 or more defective tubes.

Experience with Babcock and Wilcox steam generators has indicated that tubes near the open inspection lane are susceptible to forms of degradation unique to that area. Therefore, tubes within one, two, or three rows of the inspection lane have been defined as a special group. If all of these tubes are inspected in both steam generators, no credit will be taken for them in meeting minimum sample size requirements and the results of their inspection will not be used in classifying the results of the general inspection into C-1, C-2 or C-3 categories, unless the mechanism of tube degradation is random in nature. Random degradation mechanisms are those which based on location, steam generator design and operation, and operating experience cannot logically and consistently be shown as limited to a local area.

The affected area will be 100% inspected to assure all defective tubes therein are identified and either removed from service or repaired by sleeving. NRC concurrence in this determination is required prior to completion of the inspection.

Degraded steam generator tubes can 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. An additional repair method for degraded steam generator tubes consists of rerolling the tubes in the upper tube sheet to create a new roll area and pressure boundary for the tube. The rerolling method will ensure that the area of degradation will not serve as a pressure boundary, thus permitting the tube to remain in service. 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 in the upper tube sheet.

All tubes which have been repaired using the reroll process will have the new roll area inspected during the inservice inspection. Defective or degraded tube indications found in the new roll area as a result of the inspection of the new roll and any indications found in the originally rolled region of the rerolled tube need not be included in determining the Inspection Results Category for the general steam generator inspection.

The rerolling repair process will only be used to repair tubes with defects in the upper tubesheet area. The rerolling repair process will be performed only once per steam generator tube using a 1 inch reroll length. Thus, multiple applications of the rerolling process to any individual tube is not acceptable. The new roll area must be free of 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. The rerolling process used by Oconee is described in the topical report, BAW-2303P, Revision 3.

This inspection plan enables exposures to be maintained as low as reasonably achievable to the personnel involved in the inspection and assured that generator areas with significant numbers of degraded tubes are adequately inspected.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 227 TO FACILITY OPERATING LICENSE DPR-38,

AMENDMENT NO. 227 TO FACILITY OPERATING LICENSE DPR-47,

AND AMENDMENT NO. 224 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 October 20, 1997, and supplements dated November 3, 6, and 10, 1997, Duke Energy Corporation (the licensee) submitted a request for changes to the Oconee Nuclear Station (ONS), Units 1, 2, and 3, Technical Specifications (TS). The purpose of the amendments is to implement alternate repair criteria in the TS for steam generator tubes that have degraded roll joints inside of the upper tubesheet. The alternate repair criteria would allow new roll joints to be installed below the degraded roll joints in the upper tubesheet. The alternate repair criteria were based on a qualification program, documented in Framatome (formerly Babcock and Wilcox) Topical Report, BAW-2303P (proprietary), "OTSG Repair Roll Qualification Report," Revision 3, which was a part of the submittal.

The supplementary information supplied by letters dated November 3, 6, and 10, 1997, did not affect the proposed no significant hazards consideration determination or the scope of the initial application letter.

2.0 BACKGROUND

Each of the Oconee units has two model 177FA once-through steam generators manufactured by Babcock and Wilcox. 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 2-3 inches in axial length extended into the upper tubesheet from the tube end. The upper tubesheet is about 24 inches thick.

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 steam generator tubes which have experienced various levels of degradation. Regulatory Guide (RG) 1.121 provides guidance on an acceptable method for establishing the limiting safe conditions of tube degradation. In addition, the plant TS require periodic inspections of steam generator tubes. The TS also require those tubes with defects in excess of the repair limits (e.g., 40 percent through-wall) be repaired or removed from service.

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

The 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 plugged or repaired by sleeving. However, 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 occurrences, and postulated accident conditions. RG 1.121 recommends that the margin of safety against tube rupture under normal operating conditions should 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. Normal structural loads imposed on the tube-to-tubesheet roll primarily are derived from the differential pressure between the primary and secondary sides of the tubes. Loadings from a postulated main steam line break event can be significant. In addition, cyclic loading from transients (e.g., startup/shutdown) should also be considered in the qualification of the roll joints.

3.0 EVALUATION

3.1 Qualification Program

On the basis of a qualification program, the licensee established that a 1-inch roll length for the new joints would carry all structural loads and minimize potential leakage. The qualification program consisted of (1) preparing the mockup, (2) establishing tube loads for the qualification tests, and (3) performing verification tests and analyses.

The mockup consisted of a perforated metal block with eight inserted steam generator tubes that simulates the tube-to-tubesheet configuration in the field. The tubes were expanded into the mockup tubesheet using an expanding tool that had the same critical dimensions as the tool used in the field.

To determine the strength of the roll joints, the licensee applied loads to the sample tubes to simulate or exceed normal, thermal and pressure cycling transient, and postulated accident conditions. In accordance with RG 1.121, the test pressure applied to the sample tubes exceeded 3 times normal operating pressure and 1.43 times main steam line break pressure.

To obtain conservative leakage results, the sample tubes were severed 360 degrees through the tube wall in the roll joints.

In the qualification program, the licensee also considered the impact of tubesheet bowing on the roll joints because the tubesheet bore diameter can change during certain operating conditions. The combined effects of primary-to-secondary pressure differential and thermal loads may cause the tubesheet to bow in one direction or the other, which can lead to tubesheet bore dilatation or shrinkage. 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. Considering the bowing effect, the licensee specified an exclusion zone in the tubesheet where the reroll joint would not be installed.

3.2 Structural and Leakage Integrity

Based on the results of the qualification testing, the licensee determined a roll length of 1-inch is necessary to ensure adequate margins of structural and leakage integrity. With regard to the structural integrity, the licensee demonstrated by its ultimate load testing (testing to simulate accident conditions) that the tube with the new roll would not be pulled out from the upper tubesheet under the worst possible combination of loadings. Also, no motion of the tubes relative to the simulated tubesheet were observed during the thermal and fatigue cycling tests.

With regard to the leakage integrity, the qualification tests showed that if each of the tubes (about 15,500 tubes) in a steam generator was rerolled in the upper tubesheet and had a 100 percent through-wall flaw in the reroll, the total leakage from all flaws would be minimal. As a defense-in-depth measure, the licensee proposed to implement a primary-to-secondary leakage limit of 150 gallons per day per steam generator in the plant TS. The 150 gallons per day requirement is more conservative than the current TS limit of 0.35 gallons per minute and is consistent with the staff's position regarding alternate tube repair criteria.

3.3 Field Installation and Inspection

The licensee proposed to apply a single repair method to install one roll (reroll) in the tubes that have degradation in the original roll 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 roll expander is 1 inch long but the actual roll will have a 1/4-inch taper on each end. The torque is automatically controlled during the rerolling and is recalibrated after installation of 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 diametrical expansion and 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 steam generator inspection activities. If degradation is found in the reroll region, the affected tube will be plugged or repaired by means other than rerolling because only one reroll per tube is allowed by the proposed amendment.

3.4 Proposed Technical Specification Changes

The significant changes to TS sections are presented verbatim as follows:

> TS 3.1.6.4 If the total leakage through the tubes of any one steam generator equals or exceeds 150 gallons per day, a reactor shutdown shall be initiated within 4 hours and the reactor shall be in a cold condition within the next 36 hours.

The proposed leakage limit of 150 gallons per day for one steam generator is more conservative than the current limit of 0.35 gallons per minute (504 gallons per day); therefore, it is acceptable.

> TS 4.17.3.c All tubes which have been repaired using the reroll process will have the new roll area inspected during the inservice inspection.

This surveillance requirement is new and clarifies the licensee's intent that all reroll regions of the repaired tubes will be inspected. The staff finds this requirement acceptable because it provides a comprehensive monitoring of potential degradation in the rerolled regions of the repaired tubes.

> TS 4.17.3.d. Notes (3) Where special inspections are performed pursuant to 4.17.3.c, defective or degraded tube indications found in the new roll area as a result of the inspection and any indications found in the originally rolled region of the rerolled tubes need not be included in determining the Inspection Results Category for the general steam generator inspection.

This requirement is new and is not a relaxation from the current TS. The indications found in the new rolls need not be included in determining the Inspection Results Category because the licensee proposed to inspect all reroll regions of repaired tubes and to report to the NRC the results of the inspection. The staff believes that these two actions, inspection and reporting, provide adequate monitoring of the rerolls in the repaired tubes.

> TS 4.17.4.d.2 If the leaking tube has been repaired by the reroll process and is leaking in the new roll area, all of the tubes in the steam generator that have been repaired by the reroll process will have the new roll area inspected. If the results of this inspection fall into the C-3 category, additional inspections will be performed in the new roll area in the other steam generator.

This requirement is a part of unscheduled inspections for the cases when primary-to-secondary leakage exceeds the leakage limits of TS 3.1.6 (e.g., 150 gallons per day). This specification verifies the condition of the nonleaking steam generator if the degradation in the leaking steam generator reaches C-3 category. This requirement is consistent with the staff position and, therefore, is acceptable.

> TS 4.17.5.f The rerolling repair process will only be used to repair tubes with defects in the upper tubesheet area. The rerolling repair process will be performed only once per steam generator tube using a 1 inch reroll length. The new roll area must be free of

degradation in order for the repair to be considered acceptable. The reroll in process used by Oconee is described in the topical report, BAW-2303P, Revision 3.

This requirement is a part of TS 4.17.5.f, Definition of Repair Limit. The reroll repair is limited to the upper tubesheet area and can only be applied once to a single tube. The specification also provides a clear definition of the length of the reroll and acceptance criteria of rerolled tubes. The staff finds the definition in this specification acceptable.

> TS 4.17.5.h 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.

This requirement clarifies the scope of tube inspection. Since the reroll regions are the new pressure boundary, the original roll areas need not be inspected The staff finds this specification acceptable.

> TS 4.17.6.b.4 [report to the NRC staff] Number of tubes repaired by rerolling and number of indications detected in the new roll area of the repaired tubes.

This requirement is a part of TS 4.17.6 that specifies that results of the steam generator tube inservice inspection be reported to the NRC within 3 months following completion of the inspection. The requirement would provide the staff with a status of conditions of the reroll regions and is, therefore, acceptable.

Other changes, listed below, are administrative in nature and support the technical changes described above. They are, therefore, acceptable.

a. TS 4.17.2: "or rerolling" has been added to the Acceptance Criteria to indicate that rerolling is a method of repair for tubes that exceed the repair limit.

b. TS 4.17.3.d: Specification was changed from "c" to "d" due to the insertion of TS 4.17.3.c.

c. TS 4.17.4.d.3: Specification was changed from "2" to "3" due to the insertion of TS 4.17.4.d.2.

d. TS 4.17.5.f: "or rerolling" was added to the Repair Limit Definition as a means of repair. In addition, "for sleeving repairs" has been added to clarify that the report BAW-1823P, Revision 1 applies to the sleeving repair method and not to the rerolling or plugging processes.

4.0 SUMMARY

The licensee proposed to implement an alternate repair method using reroll to repair tubes having indications in the original roll regions of the upper tubesheet. The technical basis for the proposed reroll method was documented in Topical Report, BAW-2303P, Revision 3.

The staff has determined that (1) the licensee's alternate repair criteria using reroll were established on the basis of the qualification tests that used specimens simulating the actual tube-to-tubesheet joint configuration of the steam generators, (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 plant TS satisfied the regulatory requirements and technical basis.

On the basis of the submitted information, the staff concludes that the proposed reroll repair for degraded roll joints in steam generators at Oconee Units 1, 2, and 3 is 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. The licensee may incorporate the proposed changes into the TS for the Oconee Nuclear Station, Units 1, 2, and 3.

5.0 EXIGENT CIRCUMSTANCES

On August 18, 1997, Oconee Unit 1 entered the present refueling outage. During operation there had been no steam generator tube leakage indications and, thus, no evidence that non-destructive examination of tubes in the tubesheet region would reveal indications. Also, no prior inspections in this area had been performed to the same extent. Therefore, prior to the shutdown and the discovery of the condition, there was no reason to suspect that a change to the Technical Specifications would be needed. This did not become evident until the examinations of the tubes in the upper tubesheet region of the "B" Steam Generator were performed and the results reviewed by the licensee early in October 1997, revealed the magnitude of the problem. (Initially 900 tubes were involved, which grew to over 1900 tubes when the examination results were re-reviewed.) This prompted the need to use an alternative that did not involve removal of such a large number of tubes from service by plugging them. On October 14, 1997, the licensee informed the staff of the inspection results, the desire to use a rerolling process, and the need for license amendments to address this possibility. The amendments were subsequently submitted on October 20, 1997. Therefore, the licensee made its best effort to submit the amendments in a timely fashion once the need was determined. Since the licensee expected that Oconee Unit 1 primary system would be in a condition that required implementation of the amendments on or before November 20, 1997, there was insufficient time to prepare and publish the Federal Register notice so that 30 days would be allowed for public comment before issuing the amendments, without impacting planned restart of the unit.

Therefore, based on review of the licensee's proposed amendments, the staff finds (1) that exigent circumstances exist, as provided for in 10 CFR 50.91(a)(6), in that the licensee and the Commission must act quickly and that time does not permit the Commission to publish a Federal Register notice allowing 30 days for prior public comment, and (2) that the licensee has not failed to use its best efforts to make a timely application and avoid creating the exigent circumstance. The Commission noticed the licensee's October 20, 1997, application for amendments in the Federal Register on October 28, 1997 (62 FR 55835), at which time the Commission made a proposed finding that the amendments involved no significant hazards consideration and there has been no public comment in response to the notice. Supplemental letters dated November 3, 6, and 10, 1997, did not affect the information supplied in the notice.

6.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission's regulations in 10 CFR 50.92 state that the Commission may make a final determination that a license amendment involves no significant hazards considerations, if operation of the facility, in accordance with the amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

The amendments have been evaluated against the three standards in 10 CFR 50.92(c). In its analysis of the issue of no significant hazards consideration, as required by 10 CFR 50.91(a), the licensee has provided the following:

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

   No. The implementation of the tube reroll does not increase the probability of occurrence of an accident or the consequences of an accident previously evaluated.

   Since reroll utilizes the original tube configuration and extends the roll expanded region, all of the design and operating characteristics of the steam generator and connected systems are preserved. The reroll joint length has been analyzed and tested for design, operating, and faulted condition loadings.

   At worst case, a tube leak would occur with the result being a primary to secondary system leak. Should a tube leak occur, the impact is bounded by the ruptured tube evaluation which has been analyzed previously. The potential for a tube rupture is not increased by the use of the reroll process.

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

   No. Operation of the steam generators with reroll repaired tubes does not create the possibility of a new or different accident from the accidents previously evaluated.

   The potential failure of the tube due to the defect which required the tube to initially be repaired is covered during the qualification of the reroll process. Qualification testing indicates that normal and faulted leakage would be well below the Technical Specification limits. Since the normal and faulted leak rates are well within the Technical Specification limit, the analyzed accident scenarios are still bounding.

The new roll transition may eventually develop PWSCC [primary water stress-corrosion cracking] and require additional repair. Since the roll transition is located within the tubesheet, it is not possible for the degradation to result in a tube rupture. Additionally, industry experience with roll transition cracking has shown that PWSCC in roll transitions is normally short axial cracks, with extremely low leak rates. Finally, since the new roll transition is completely within the tubesheet there is no possibility of the repaired tube failing and impacting adjacent tubes.

In the unlikely event the reroll repaired tube failed and severed completely at the transition of the reroll region, the tube would retain engagement in the tubesheet bore, preventing any interaction with neighboring tubes. In this case, leakage is minimized and is well within the assumed leakage of the design basis tube rupture accident. In addition, the possibility of rupturing multiple steam generator tubes is not increased.

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

   No. Based on the previous response, the protective boundaries of the steam generator are preserved.

   A tube with degradation can be kept in service through the use of the reroll process. The new undegraded roll expanded interface created with the tubesheet satisfies all of the necessary structural, leakage, and heat transfer requirements. Since the joint is constrained within the tubesheet bore, there is no additional risk associated with tube rupture. Therefore, the analyzed accident scenarios remain bounding, and the use of the reroll process does not reduce the margin of safety.

   Duke has concluded based on the above information that there are no significant hazards involved in this amendment request.

Based on the above considerations, the NRC staff concludes that the amendments meet the three criteria of 10 CFR 50.92. Therefore, the staff has made a final determination that the proposed amendments do not involve a significant hazards consideration.

7.0 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.

8.0 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 and change surveillance requirements. 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 made a final finding that the amendments involve no significant hazards consideration. 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.

9.0 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.

Principal Contributor: John Tsao

Date: November 21, 1997