September 1, 1987

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

Mr. H. B. Tucker, Vice President
Nuclear Production Department
Duke Power Company
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Tucker:

Subject: Issuance of Amendment Nos. 161, 161, and 158 to Facility Operating Licenses DPR-38, DPR-47, and DPR-55 - Oconee Nuclear Station, Units 1, 2, and 3 (TACS 60742/60743/60744)

The Nuclear Regulatory Commission has issued the enclosed Amendment Nos. , , and to Facility Operating Licenses Nos. DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station, Units 1, 2, and 3. These amendments consist of changes to the Station's common Technical Specifications (TSs) in response to your request dated January 16, 1986, as supplemented April 18, June 27, and September 15, 1986, and April 3, 1987.

The amendments modify the TSs to allow repairing of steam generator tubes by sleeving.

A copy of our Safety Evaluation is also enclosed. Notice of issuance of the enclosed amendments will be included in the Commission's bi-weekly Federal Register notice.

Sincerely,

/S/

Helen N. Pastis, Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II

Enclosures:
1. Amendment No. 161 to DPR-38
2. Amendment No. 161 to DPR-47
3. Amendment No. 158 to DPR-55
4. Safety Evaluation

cc w/enclosures: See next page

PDII-3/DRP-I/II	PDII-3/DRP-I/II	PDII-3/DRP-I/II
MDuncan/rad	HPastis	for BJYoungblood
08/2/87	08/10/87	~~06/ /87~~ 9/1/87

Mr. H. B. Tucker
Duke Power Company

Oconee Nuclear Station
Units Nos. 1, 2 and 3

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

DUKE POWER COMPANY

DOCKET NO. 50-269

OCONEE NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 161
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 Power Company (the licensee) dated January 16, 1986, as supplemented on April 18, June 27, and September 15, 1986, and April 3, 1987, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter 1;

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

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

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

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 161, 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

/S/

Lawrence P. Crocker, Acting Director
Project Directorate II-3
Division of Reactor Projects - I/II

Attachment:
Technical Specification
Changes

Date of Issuance: September 1, 1987

* SEE PREVIOUS CONCURRENCES

PDII-3/DRP-I/II	PDII-3/DRP-I/II	OGC-Bethesda	PDII-3/DRP-I/II
MDuncan/rad	*HPastis	*JScinto	LCrocker, Acting PD
09/ /87	07/10/87	07/30/87	09/1/87

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

DUKE POWER COMPANY

DOCKET NO. 50-270

OCONEE NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 161
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 Power Company (the licensee) dated January 16, 1986, as supplemented on April 18, June 27, and September 15, 1986, and April 3, 1987, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter 1;

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

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

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

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 161, 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

/S/

Lawrence P. Crocker, Acting Director
Project Directorate II-3
Division of Reactor Projects - I/II

Attachment:
Technical Specification
Changes

Date of Issuance: September 1, 1987

* SEE PREVIOUS CONCURRENCES

PDII-3/DRP-I/II	PDII-3/DRP-I/II	OGC-Bethesda	PDII-3/DRP-I/II
MDuncan/rad	*HPastis	*JScinto	LCrocker, Acting PD
09/ /87	07/10/87	07/30/87	09/1/87

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

DUKE POWER COMPANY

DOCKET NO. 50-287

OCONEE NUCLEAR STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 158
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 Power Company (the licensee) dated January 16, 1986, as supplemented on April 18, June 27, and September 15, 1986, and April 3, 1987, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter 1;

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

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

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

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 158, 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

/S/

Lawrence P. Crocker, Acting Director
Project Directorate II-3
Division of Reactor Projects - I/II

Attachment:
Technical Specification
Changes

Date of Issuance: September 1, 1987

* SEE PREVIOUS CONCURRENCES

PDII-3/DRP-I/II	PDII-3/DRP-I/II	OGC-Bethesda	PDII-3/DRP-I/II
MDuncan/rad	*HPastis	*JScinto	LCrocker, Acting PD
09/ /87	07/10/87	07/30/87	09/1 /87

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 161 TO DPR-38

AMENDMENT NO. 161 TO DPR-47

AMENDMENT NO. 158 TO DPR-55

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

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

Remove Page	Insert Page
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
4.17-6	4.17-6

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 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. 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 not 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.

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 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. <u>Imperfection</u> 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. <u>Degradation</u> means a service-induced cracking, wastage, wear or general corrosion occurring on either the inside or outside of a tube or a sleeve.

c. <u>Degraded Tube</u> means a tube or a sleeve containing imperfections $\geq$ 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 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.

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.

4.17.6 Reports

a. The number of tubes plugged or repaired in each steam generator shall be reported to the Director, Office of Inspection and Enforcement, Region II, 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.

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.

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

TABLE 4.17-1

STEAM GENERATOR TUBE INSPECTION

1st SAMPLE INSPECTION			2nd SAMPLE INSPECTION		3rd SAMPLE INSPECTION	
SAMPLE SIZE	RESULT	ACTION REQUIRED	RESULT	ACTION	RESULT	ACTION
A minimum of S Tubes per S.G. (1)	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug or repair defective tubes and inspect additional 2S tubes in	C-1	None	N/A	
			C-2	Plug or repair defective tubes and inspect additional 4S tubes in this S.G.	C-1	N/A
					C-2	Plug or repair defective tubes
					C-3	Plug or repair defective tubes and perform action for C-3 result of 1st Sample
			C-3	Plug or repair defective tubes and perform actions for C-3 results of 1st Sample	N/A	N/A
	C-3	Inspect 6S tubes in this S.G., plug or repair defective tubes and inspect 2S tubes in the other S.G. Perform follow-on inspections in the other S.G. in accordance with results of the above inspection as applied to Table 4.17.1 Prompt notification to NRC pursuant to specification 6.6.2.1.a	C-1	N/A	N/A	N/A
			C-2	N/A	N/A	N/A
			C-3 (2)	(a) if defects can be localized to an affected area, inspect all tubes in affected area and plug or repair defective tubes. (b) If defects cannot be localized to an affected area, inspect all tubes in this S.G. and plug or repair defectives tubes.	C-1	N/A
					C-2	N/A
					C-3	N/A

Notes: (1) S = 3(N/n)% Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection

(2) Affected and unaffected areas shall be determined in the manner described in the Bases of this specification. The definition of these areas will be reported to the NRC when they are determined.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 161 TO FACILITY OPERATING LICENSE NO. DPR-38

AMENDMENT NO. 161 TO FACILITY OPERATING LICENSE NO. DPR-47

AMENDMENT NO. 158 TO FACILITY OPERATING LICENSE NO. DPR-55

DUKE POWER COMPANY

OCONEE NUCLEAR STATION, UNITS 1, 2, and 3

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

1.0 INTRODUCTION

By reference 1, Duke Power Company (DPC or the licensee) proposed to amend the Technical Specifications (TS) of Facility Operating Licenses Nos. DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station, Units 1, 2, and 3. These amendments would revise the station's common TS 4.17 to allow an alternative way in repairing defective steam generator tubes. Currently, DPC is allowed to repair steam generator tubes only by plugging. The proposed revision would allow sleeving. By repairing with sleeves, DPC would reduce the number of steam generator tubes which must be plugged and removed from service. During our evaluation, DPC sent additional information in references 2 through 5. DPC reviewed the Oconee Final Safety Analysis Report and its safety analyses to assess the potential impact on plant safety of sleeving 5,000 tubes per steam generator with eighty-inch long sleeves. Reference 2 addressed the changes to heat transfer and primary flow rate resulting from the insertion of sleeves in the tubes.

2.0 Evaluation

2.1 Impact of Sleeving in the FSAR Safety Analysis

DPC has reviewed what impact sleeving 5,000 tubes per steam generator has on the Oconee FSAR safety analyses. Sleeving affects the characteristics of the steam generator in two ways. The first, an air gap between the sleeve and the steam generator tube, slightly reduces the heat transfer. The second, smaller tube diameters in the sleeved tubes, slightly increases the primary side pressure drop through the steam generator. Analysis by DPC has also shown that if 5,000 sleeves were installed in each generator, then these sleeves would reduce the steam superheat temperature by about 7.7°F at full power and the primary flow by less than 1%. To compensate for this reduction in superheat, feedwater flow would have to be increased by 1% to remove the same amount of energy.

To evaluate overcooling events, the licensee assumes in the FSAR analysis that feedwater flow increases during this transient. Increased feedwater flow is a conservative assumption because it increases the heat removed by the steam generator. Therefore, sleeving does not impact the safety analysis because the heat removed by an additional 1% feedwater flow has been conservatively bounded by the heat removal rates assumed in the analysis for overcooling events.

For some overheating events such as the loss of main feedwater, the licensee assumes in the FSAR analysis that the heat transfer in the steam generator has been significantly reduced. Because the slight reduction in heat transfer coefficient along the sleeved tubes is much smaller than the reduction assumed in the FSAR analysis, the analysis assumptions are still valid. Sleeving also does not affect the analysis of those overheating events which occur in the primary system. An example of such an event is control rod withdrawal. This analysis is still valid because the initial heat transfer rate is held constant throughout the event and overall total steady state steam generator heat transfer is unaffected by tube sleeving.

Previous generic evaluations have shown that although steam generator tube plugging reduces RCS flow by 2.5%, plugging has a negligible impact on LOCA results. The licensee has indicated that the same number of sleeved tubes would result in only a 1% reduction in RCS flow. Therefore, the generic LOCA analysis on the effects of tube plugging conservatively bounds the effects of tube sleeving. Also, because steam binding in the steam generator may affect the reflood phase of the large break LOCA, the licensee has specifically considered this concern and has found that tube sleeving has no impact.

To verify that the actual flow is more than that assumed in the plant safety analyses, DPC has plant procedures which require it to measure primary system flow at the beginning of each fuel cycle. These procedures also require monitoring the primary flow several times per day. Therefore, the licensee would detect any flow degradation caused by sleeving and would ensure that the existing plant safety analyses remain valid and bounding. Therefore, with the licensee detecting at an early stage any flow degradation caused by sleeving, the plant would be bounded and remain within the existing plant safety analysis.

Based on the above evaluations, the thermal-hydraulic effects of sleeving up to 5,000 Oconee steam generator tubes with 80-inch long sleeves per generator will have a minimal and acceptable effect on plant operation and the existing FSAR safety analysis will continue to bound normal and abnormal plant conditions.

2.2 Material Characteristics

The sleeving qualification program which consisted of test analyses and development of general design criteria for the Once-Through Steam Generator (OTSG) sleeves is described in B&W Report BAW-1823P, Rev. 1, "Once-Through Steam Generator Mechanical Sleeve Qualification." Sleeving at Oconee Units 1, 2, and 3 is proposed to be done in accordance with the design criteria and procedures discussed in this report. Based on a review of this report which was completed in November 1984 (Reference 6), and supplemental information provided by the licensee, the staff concludes that the licensee's sleeve/tube qualification program is acceptable and sufficient justification has been provided to allow sleeving at Oconee Units 1, 2, and 3.

Test results show that corrosion is not likely to increase in the rolled sleeve joint significantly during normal operation and wet lay-up. Other mechanisms such as corrosion from cyclic stresses are also not likely to be aggravated by the sleeving process. The minimum required sleeve wall to withstand normal and accident condition loads was determined to be 0.0141 inch which permits sleeve defects less than 70 percent through-wall. Licensee's analysis is in compliance with the requirements of ASME Code Section III and NRC Regulatory Guide 1.121. Allowing an additional margin of 10% for continued degradation and 20% for uncertainty in eddy current measurements, the licensee's proposed 40% plugging limit for sleeves is acceptable for all defects with the exception of circumferential cracks. Any detectable circumferential crack in the sleeve will be plugged.

Tests conducted by B&W on tube specimens indicate that under axial tensile loading, cracks less than 360° in circumferential extent and of depths necessary to satisfy the 40% plugging limit, might result in tube failure. In addition, there is a greater propensity for crack propagation and tube failure due to flow-induced vibration near the 15th support plate region. For these reasons, and due to the uncertainties in sizing defects between 20% and 40% in the entire sleeve/parent tube region, the staff discussed this issue with the licensee. The licensee stated that if circumferential cracks are detected in the sleeves, those sleeved tubes would be plugged.

The staff is currently evaluating the generic implications of this axial tensile load generated under accident conditions on the various aspects of circumferential cracks and the present 40% plugging limit in the technical specifications for B&W OTSG tubes. Circumferential cracks in th 15th tube support plate region were first observed at Oconee Unit 3 and have since been identified in several other B&W OTSGs. In fact, for many B&W plants this is often the only predominant type of tube degradation experienced. For circumferential cracks in the tubes, B&W recommends plugging of all tubes with detectable circumferential cracks, and the staff agrees with this position. The staff is discussing this issue with the B&W Owners Group and will recommend the necessary changes, if needed, to the Oconee Units 1, 2, and 3 tube plugging limit after a generic resolution of this issue is finalized.

The licensee will use the Babcock and Wilcox process (or method) equivalent to the method described in report, BAW-1823 P, Revision 1. The licensee is not restricted from correcting typographical errors, clarifying information or tightening up on information or adding information. However, license amendment would be required if the licensee changed the method.

3.0 ENVIRONMENTAL CONSIDERATION

These amendments involve a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. We have 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 these amendments involve no significant hazards consideration, and there has been no public comment on such finding. Accordingly, these 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 these amendments.

4.0. CONCLUSION

The Commission made a proposed determination that the amendments involve no significant hazards consideration which was published in the Federal Register (51 FR 12227) on April 9, 1986, and consulted with the state of South Carolina. No public comments were received, and the state of South Carolina did not have any comments.

We have 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, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

References:

1. Letter from H.B. Tucker, Duke Power Company, to H.R. Denton, NRC, requesting Technical Specification Change to allow steam generator tube sleeving, dated January 16, 1986.

2. Letter from H.B. Tucker, Duke Power Company, to H.R. Denton, NRC, providing results of analyses to address potential impact on plant safety of sleeving up to 5000 tubes per steam generator, dated April 18, 1986.

3. Letter from H.B. Tucker, Duke Power Company, to H.R. Denton, NRC, providing responses to staff request for additional information, dated June 27, 1986.

4. Letter from H.B. Tucker, Duke Power Company, to H.R. Denton, NRC, providing responses to staff questions, dated September 15, 1986.

5. Letter from H.B. Tucker, to U.S. Nuclear Regulatory Commission, dated April 3, 1987.

6. Memorandum from J.P. Knight, NRC, to G. Lainas, NRC, "Arkansas Unit 1 - Steam Generator Tube Sleeving," dated November 28, 1984.

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Dated: September 1, 1987

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AMENDMENT NO. 161 TO FACILITY OPERATING LICENSE DPR-38 - Oconee Nuclear Station, Unit 1
AMENDMENT NO. 161 TO FACILITY OPERATING LICENSE DPR-47 - Oconee Nuclear Station, Unit 2
AMENDMENT NO. 158 TO FACILITY OPERATING LICENSE DPR-55 - Oconee Nuclear Station, Unit 3

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