March 4, 1991

Docket Nos. 50-413
and 50-414

Mr. M.S. Tuckman
Vice President -
Nuclear Operations
Duke Power Company
P.O. Box 1007
Charlotte, North Carolina 28201-1007

Dear Mr. Tuckman:

SUBJECT: ISSUANCE OF AMENDMENT NO. 84 TO FACILITY OPERATING LICENSE NPF-35 AND AMENDMENT NO. 78 TO FACILITY OPERATING LICENSE NPF-52 - CATAWBA NUCLEAR STATION, UNITS 1 AND 2, ON STEAM GENERATOR TUBE SLEEVING (TACS 79167/79168)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 84 to Facility Operating License NPF-35 and Amendment No. 78 to Facility Operating License NPF-52 for the Catawba Nuclear Station, Units 1 and 2. These amendments consist of changes to the Technical Specifications (TSs) in response to your application dated December 19, 1990.

The amendments modify TS 4.4.5 and its associated Bases to allow the option of using the B&W Kinetic Sleeving Process for steam generator tube repairs as described in Topical Report BAW-2045(P)-A. This will provide an alternative other than plugging for handling defective steam generator tubes.

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

Sincerely,

Original Signed By:

Robert E. Martin, Senior Project Manager
Project Directorate II-3
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 84 to NPF-35
2. Amendment No. 78 to NPF-52
3. Safety Evaluation

cc w/enclosures:
See next page

OFC	LA:PDII-3	PM:PDII-3	BC:EMCB	OGC	D:PDII-3
NAME	RIngram	RMartin/rst	CYCheng	EHolden	DMatthews
DATE	2/14/91	2/14/91	2/14/91	2/27/91	3/4/91

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DATED: March 4, 1991

AMENDMENT NO. 84 TO FACILITY OPERATING LICENSE NPF-35 - Catawba Nuclear Station, Unit 1
AMENDMENT NO. 78 TO FACILITY OPERATING LICENSE NPF-52 - Catawba Nuclear Station, Unit 2

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Mr. M.S. Tuckman
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Catawba Nuclear Station

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

DUKE POWER COMPANY

NORTH CAROLINA ELECTRIC MEMBERSHIP CORPORATION

SALUDA RIVER ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-413

CATAWBA NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 84
License No. NPF-35

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

   A. The application for amendment to the Catawba Nuclear Station, Unit 1 (the facility) Facility Operating License No. NPF-35 filed by the Duke Power Company, acting for itself, North Carolina Electric Membership Corporation and Saluda River Electric Cooperative, Inc. (licensees) dated December 19, 1990, 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 changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-35 is hereby amended to read as follows:

    Technical Specifications

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 84 , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. Duke Power Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

DB Matthews

David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification Changes

Date of Issuance: March 4, 1991

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

DUKE POWER COMPANY

NORTH CAROLINA MUNICIPAL POWER AGENCY NO. 1

PIEDMONT MUNICIPAL POWER AGENCY

DOCKET NO. 50-414

CATAWBA NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 78
License No. NPF-52

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

   A. The application for amendment to the Catawba Nuclear Station, Unit 2 (the facility) Facility Operating License No. NPF-52 filed by the Duke Power Company, acting for itself, North Carolina Municipal Power Agency No. 1 and Piedmont Municipal Power Agency (licensees) dated December 19, 1990, 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 changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-52 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 78 , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. Duke Power Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification Changes

Date of Issuance: March 4, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 84

FACILITY OPERATING LICENSE NO. NPF-35

DOCKET NO. 50-413

AND

TO LICENSE AMENDMENT NO. 78

FACILITY OPERATING LICENSE NO. NPF-52

DOCKET NO. 50-414

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
3/4 4-15	3/4 4-15
3/4 4-16	3/4 4-16
3/4 4-16a	3/4 4-16a
B3/4 4-3	B 3/4 4-3
---	B 3/4 4-3a
B 3/4 4-4	B 3/4 4-4

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.4 Acceptance Criteria

a. As used in this specification:

1) Imperfection means an exception to the dimensions, finish or contour of a tube or sleeve 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;

2) Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube or sleeve;

3) Degraded Tube means a tube or sleeve containing imperfections greater than or equal to 20% of the nominal tube or sleeve 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 at or beyond which the tube shall be removed from service by plugging or repaired by sleeving. It also means the imperfection depth at or beyond which a sleeved tube shall be plugged. The repair limit is equal to 40% of the nominal tube or sleeve wall thickness. For Unit 1, this definition does not apply to the region of the tube subject to the alternate tube plugging criteria.

   If a tube is sleeved due to degradation in the F* distance, then any defects found in the tube below the sleeve will not necessitate plugging.

   The Babcock & Wilcox process described in Topical Report BAW-2045(P)-A will be used for sleeving.

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 4.4.5.3c., above;

8) Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg;

9) Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

10) Tube Roll Expansion is that portion of a tube which has been increased in diameter by a rolling process such that no crevice exists between the outside diameter of the tube and the tubesheet.

11) F* Distance is the minimum length of the roll expanded portion of the tube which cannot contain any defects in order to ensure the tube does not pull out of the tubesheet. The F* distance is 1.60 inches and is measured from the bottom of the roll expansion transition or the top of the tubesheet if the bottom of the roll expansion is above the top of the tubesheet. Included in this distance is a safety factor of 3 plus a 0.5 inch eddy current vertical measurement uncertainty.

12) Alternate tube plugging criteria does not require the tube to be removed from service or repaired when the tube degradation exceeds the repair limit so long as the degradation is in that portion of the tube from F* to the bottom of the tubesheet. This definition does not apply to tubes with degradation (i.e., indications of cracking) in the F* distance.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug or repair all tubes exceeding the repair limit and all tubes containing through-wall cracks) required by Table 4.4-2. For Unit 1, tubes with defects below F* fall under the alternate tube plugging criteria and do not have to be plugged.

4.4.5.5 Reports

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes repaired in each steam generator shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2;

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following the completion of the inspection. This Special Report shall include:

1) Number and extent of tubes inspected,

SURVEILLANCE REQUIREMENTS (Continued)

2) Location and percent of wall-thickness penetration for each indication of an imperfection, and

3) Identification of tubes repaired.

c. For Unit 2, results of steam generator tube inspections, which fall into Category C-3, shall be reported in a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days and prior to resumption of plant operation. This report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

d. For Unit 1, the results of inspections for all tubes for which the alternate tube plugging criteria has been applied shall be reported to the Nuclear Regulatory Commission in accordance with 10 CFR 50.4, prior to restart of the unit following the inspection. This report shall include:

1) Identification of applicable tubes, and

2) Location and size of the degradation.

STEAM GENERATORS (Continued)

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

The B&W process (or method equivalent) to the inspection method described in Topical Report BAW-2045(P)-A will be used. Inservice inspection of steam generator sleeves is also required to ensure RCS integrity. Because the sleeves introduce changes in the wall thickness and diameter, they reduce the sensitivity of eddy current testing, therefore, special inspection methods must be used. A method is described in Topical Report BAW-2045(P)-A with supporting validation data that demonstrates the inspectability of the sleeve and underlying tube. As required by NRC for licensees authorized to use this repair process, Catawba commits to validate the adequacy of any system that is used for periodic inservice inspections of the sleeves, and will evaluate and, as deemed appropriate by Duke Power Company, implement testing methods as better methods are developed and validated for commercial use.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the Reactor Coolant System and the Secondary Coolant System (reactor-to-secondary leakage = 500 gallons per day per steam generator). Cracks having a reactor-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that reactor-to-secondary leakage of 500 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and repaired.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Repair will be required for all tubes with imperfections exceeding the repair limit of 40% of the tube nominal wall thickness. For Unit 1, defective tubes which fall under the alternate tube plugging criteria do not have to be repaired. Defective 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, allowing the tube to remain in service. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect wastage type degradation that has penetrated 20% of the original tube wall thickness.

STEAM GENERATORS (Continued)

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be reported to the Commission pursuant to Specification 6.9.2 prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary. If a tube is sleeved due to degradation in the F* distance, then any defects in the tube below the sleeve will remain in service without repair.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The Leakage Detection Systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. These Detection Systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.6.2 OPERATIONAL LEAKAGE

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

Industry experience has shown that while a limited amount of leakage is expected from the Reactor Coolant System, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 gpm. This threshold value is sufficiently low to ensure early detection of additional leakage.

The total steam generator tube leakage limit of 1 gpm for all steam generators not isolated from the Reactor Coolant System ensures that the dosage contribution from the tube leakage will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of either a steam generator tube rupture or steam line break. The 1 gpm limit is consistent with the assumptions used in the analysis of these accidents. The 500 gpd leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

The 10 gpm IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the Leakage Detection Systems.

The CONTROLLED LEAKAGE limitation restricts operation when the total flow supplied to the reactor coolant pump seals exceeds 40 gpm with the modulating valve in the supply line fully open at a nominal Reactor Coolant System pressure of 2235 psig. This limitation ensures that in the event of a LOCA, the safety injection flow will not be less than assumed in the safety analyses.

The 1 gpm leakage from any Reactor Coolant System pressure isolation valve is sufficiently low to ensure early detection of possible in-series check valve failure. It is apparent that when pressure isolation is provided by two in-series check valves and when failure of one valve in the pair can go undetected for a substantial length of time, verification of valve integrity is required. Since these valves are important in preventing overpressurization and rupture of the ECCS low pressure piping which could result in a LOCA that bypasses containment, these valves should be tested periodically to ensure low probability of gross failure.

The Surveillance Requirements for Reactor Coolant System pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the pressure isolation valve is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 84 TO FACILITY OPERATING LICENSE NPF-35

AND AMENDMENT NO. 78 TO FACILITY OPERATING LICENSE NPF-52

DUKE POWER COMPANY, ET AL.

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-413 AND 50-414

1.0 INTRODUCTION

By letter dated December 19, 1990, Duke Power Company, et al. (the licensee) proposed changes to the Catawba Nuclear Station, Units 1 and 2, Technical Specification (TS) 4.4.5 and associated Bases. The proposed change would revise the surveillance requirements of Technical Specifications 3/4.4.5, "Steam Generators," to permit the option of using the Babcock & Wilcox (B&W) kinetic sleeving process for steam generator tube repair.

The requested Technical Specifications change will allow the use of B&W sleeves for steam generator tube repair as alternative to plugging degraded tubes. The change references B&W topical report BAW-2045P, "Recirculating Steam Generator Kinetic Sleeve Qualification for 3/4 Inch OD Tubes," in Technical Specifications Section 4.4.5.4.a.6. The topical report was submitted to the NRC by a letter dated June 9, 1988, and a supplement to the original submittal was made on December 12, 1988, which contained answers to the NRC request for additional information. The staff approved the topical report as being suitable for referencing in a letter to James H. Taylor of B&W from James E. Richardson dated January 4, 1990.

The purpose of a sleeve is to repair a degraded steam generator tube in order to maintain the function and integrity of the tube. The sleeve functions in essentially the same manner as the original tube. B&W topical report BAW-2045P describes in detail the analytical methods used for design and qualification of the B&W sleeve. The topical report lists the specifications (mainly ASME Boiler and Pressure Vessel code requirements) used in design, procurement and qualification of the sleeve. It also summarizes the transients used to establish sleeve loading.

2.0 EVALUATION

BAW-2045P contains the results of the sleeve design verification which included analysis and confirmatory testing to demonstrate the acceptability of the steam generator sleeving technique for defective tubes. The design and operating conditions specified for the sleeve bound the Catawba steam generator design conditions.

The sleeve design described in BAW-2045P is qualified for two lengths, 11 inches and 17 and 1/2 inches. The lower end of each sleeve is located approximately 16 inches from the primary face of the tubesheet. The shorter sleeve may be utilized in all the steam generator tubes (including the peripheral tubes which typically do not permit the introduction of sleeves due to the close proximity of the bowl in that area). The longer sleeve extends further into the tube past the flow distribution baffle.

The sleeve material is thermally treated Alloy 690 Inconel with a specified minimum wall thickness of 0.039 inches. The required minimum thickness is 0.027 inches based on primary side design pressure. This material has been demonstrated to be resistant to corrosion phenomenon as detailed in BAW-2045P. The upper sleeve/tube joint is produced by a kinetic weld/expansion which is subsequently stress relieved. The joint is qualified as both a strength and seal weld. The lower joint may consist of either a kinetic weld in the tubesheet or a mechanically sealed joint produced by rolling the sleeve in the tubesheet. The lower joint is qualified for applicable loads without taking credit for the original strength of the tube rolled into the tubesheet. Therefore, the structural integrity of the tube is maintained by the sleeving process.

The adequacy of the sleeve to withstand cyclic loadings was demonstrated using fatigue testing. Fatigue testing consisted of cyclic vibration, pressure, thermal, and axial loading. These tests were performed to demonstrate the structural adequacy of the installed sleeve. In all cases, the results of the tests indicated that the sleeve conformed to the design requirements of the steam generators.

Based on Regulatory Guide 1.121 guidelines for tube degradation limits, a plugging limit of 40% of the original sleeve wall has been established. Eddy current techniques are available to perform necessary sleeve/tube inspections for defect detection and to verify proper installation of the sleeve. The Licensee has stated that available techniques are capable of providing 20% defect sensitivity in the required areas of the tube/sleeve pressure boundary.

A proprietary method is described in the topical report with supporting validation data that demonstrates the inspectability of the sleeve and underlying tube. The Licensee has provided a commitment to validate the adequacy of any system that is used for periodic inservice inspections as well as a commitment to upgrade testing methods as better methods are developed and validated for commercial use.

The following conclusions are based on the staffs previous review and approval of BAW-2045P as revised to include the changes described in the letter dated December 12, 1988, from James H. Taylor, B&W, to Lawrence C. Shao, NRC (NRC Accession Number 8901040142). The staff has concluded that there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner and the issuance of this amendment is acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

These amendments involve changes to requirements with respect to the use or installation of components located in the restricted area as defined in 10 CFR Part 20, and changes in surveillance requirements. The 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 issues a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and 51.22(c)(10). Pursuant to 10 CFR 51.22(b), no environmental assessment need be prepared in connection with the issuance of these amendments.

4.0 CONCLUSION

The Commission's proposed determination that the amendments involve no significant hazards consideration was published in the Federal Register (56 FR 2053) on January 18, 1991. The Commission 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 the health and safety of the public.

Principal Contributor: H. Conrad, EMCB

Dated: March 4, 1991