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. 95
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 May 9, 1991, as supplemented on February 6, 1992, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 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. 95 , 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

DBMatthews

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: April 14, 1992

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. 89
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 May 9, 1991 as supplemented on February 6, 1992, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 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. 89 , 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 th Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

D.B. 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: April 14, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 95

FACILITY OPERATING LICENSE NO. NPF-35

DOCKET NO. 50-413

AND

TO LICENSE AMENDMENT NO. 89

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 4-10	3/4 4-10
3/4 4-11	3/4 4-11
3/4 4-37	3/4 4-37
3/4 4-38	3/4 4-38
B 3/4 4-2	B 3/4 4-2
B 3/4 4-3	B 3/4 4-3
B 3/4 4-3a	B 3/4 4-3a

3/4.4.4 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.4 All power-operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one or more PORV(s) inoperable, because of excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) with power maintained to the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one or two PORVs inoperable due to causes other than excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) and remove power from the block valve(s); restore the PORV(s) to OPERABLE status within the following 72 hours or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With three PORVs inoperable due to causes other than excessive seat leakage, within 1 hour either restore at least one PORV to OPERABLE status or close their associated block valves and remove power from the block valves and be in HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 6 hours.

d. With one or more block valve(s) inoperable and not closed, within 1 hour restore the block valve(s) to OPERABLE status, or place its associated PORV switch(es) in the 'close' position. Restore at least one block valve to OPERABLE status within the next hour if three block valves are inoperable; restore any remaining inoperable block valve(s) to OPERABLE status within 72 hours; otherwise, be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

e. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.4.1 In addition to the requirements of Specification 4.0.5, each PORV shall be demonstrated OPERABLE at least once per 18 months by:

a. Performance of a CHANNEL CALIBRATION, and

b. Operating the valve through one complete cycle of full travel*.

4.4.4.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed with power removed in order to meet the requirements of ACTION b. or c. in Specification 3.4.4.

4.4.4.3 The safety related nitrogen supply for the PORVs shall be demonstrated OPERABLE at least once per 18 months by:

a. Manually transferring motive power from the normal (air) supply to the emergency (nitrogen) supply,

b. Isolating and venting the normal (air) supply, and

c. Operating the valves through a complete cycle of full travel.

*In order to simulate environmental effects representative of operating conditions SR 4.4.4.1b should be conducted when the reactor coolant system temperature is greater than 200°F; however this SR shall not be performed in MODES 1 or 2.

REACTOR COOLANT SYSTEM

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.3 At least one of the following [illegible]ressure Protection Systems shall be OPERABLE:

a. Two power operated relief valves (PORVs) with a lift setting of less than or equal to 450 psig, or

b. The Reactor Coolant System depressurized with a Reactor Coolant System vent of greater than or equal to 4.5 square inches.

APPLICABILITY: MODE 4 when the temperature of any Reactor Coolant System cold leg is less than or equal to 285°F, MODE 5 and MODE 6 when the head is on the reactor vessel.

ACTION:

a. With one PORV inoperable in MODE 4, restore the inoperable PORV to OPERABLE status within 7 days or complete depressurization and venting of the Reactor Coolant System through at least 4.5 square inch vent within the next 8 hours.

b. With one PORV inoperable in MODES 5 or 6, restore the inoperable PORV to OPERABLE status within 24 hours or complete depressurization and venting of the Reactor Coolant System through at least 4.5 square inch vent within the next 8 hours.

c. With both PORVs inoperable, complete depressurization and venting of the Reactor Coolant System through at least a 4.5 square inch vent within 8 hours.

d. In the event either the PORVs or the Reactor Coolant System vent(s) are used to mitigate a Reactor Coolant System pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or Reactor Coolant System vent(s) on the transient, and any corrective action necessary to prevent recurrence.

e. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE by:

a. Performance of an ANALOG CHANNEL OPERATIONAL TEST on the PORV actuation channel, but excluding valve operation, at least once per 31 days;

b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once per 18 months; and

c. Verifying the PORV isolation valve is open at least once per 72 hours when the PORV is being used for overpressure protection.

4.4.9.3.2 The Reactor Coolant System vent(s) shall be verified to be open at least once per 12 hours* when the vent(s) is being used for overpressure protection.

*Except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open at least once per 31 days.

SAFETY VALVES (Continued)

relief capability and will prevent overpressurization. In addition, the Overpressure Protection System provides a diverse means of protection against overpressurization at low temperatures.

During operation, all pressurizer Code safety valves must be OPERABLE to prevent the Reactor Coolant System from being pressurized above its Safety Limit of 2[illegible] psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss-of-load assuming no Reactor trip until the first Reactor Trip System Trip Setpoint is reached (i.e., no credit is taken for a direct Reactor trip on the loss-of-load) and also assuming no operation of the power-operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

3/4.4.3 PRESSURIZER

The limit on the maximum water volume in the pressurizer assures that the parameter is maintained within the normal steady-state envelope of operation assumed in the SAR. The limit is consistent with the initial SAR assumptions. The 12-hour periodic surveillance is sufficient to ensure that the parameter is restored to within its limit following expected transient operation. The maximum water volume also ensures that a steam bubble is formed and thus the Reactor Coolant System is not a hydraulically solid system. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability of the plant to control Reactor Coolant System pressure and establish natural circulation.

3/4.4.4 RELIEF VALVES

The power-operated relief valves (PORVs) and steam bubble function to relieve Reactor Coolant System pressure during all design transients up to and including the design step load decrease with steam dump. Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable. The OPERABILITY of the PORVs and block valves is determined on the basis of their being capable of performing the following functions: 1) Manual control of PORVs to control Reactor Coolant System pressure. This is a function that is used for the steam generator tube rupture accident coincident with a loss of all offsite power and for plant shutdown. 2) Maintaining the integrity of the reactor coolant pressure boundary. This is a function that is related to controlling identified leakage and ensuring the ability to detect unidentified reactor coolant pressure boundary leakage. 3) Manual control of the block valve to unblock an isolated PORV to allow it to be used for manual control of Reactor Coolant System pressure and isolate a PORV with excessive seat leakage. 4) Automatic control of PORVs to control

STEAM GENERATORS (Continued)

reactor coolant system pressure except for limited periods where the PORV has been isolated due to excessive seat leakage and except for limited periods when the PORV and/or block valve is closed because of testing and is fully capable of being returned to its normal alignment at any time provided that this is covered by an approved procedure. This is a function that reduces to the code safety valves for overpressurization events. 5) Manual the block valve to isolate a stuck-open PORV. Testing of the PORVs in the emergency N_2 supply from the Cold Leg Accumulators. This test demonstrates that the valves in the supply line operate satisfactorily and nonsafety portion of the instrument air system is not necessary for PORV operation.

3/4.4.5 STEAM GENERATORS

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

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.

STEAM GENERATORS (Continued)

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.

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.