November 29, 1994

Mr. T. C. McMeekin
Vice President, McGuire Site
Duke Power Company
12700 Hagers Ferry Road
Huntersville, NC 28078-8985

SUBJECT: MCGUIRE NUCLEAR STATION, UNITS 1 AND 2 - AMENDMENTS NOS. 151 AND 133 - AUXILIARY FEEDWATER PUMP TESTING INTERVAL (TAC NOS. 90355 AND 90356)

Dear Mr. McMeekin:

Amendments Nos. 151 and 133 to Facility Operating Licenses NPF-9 and NPF-17, respectively, for the McGuire Nuclear Station, Units 1 and 2, regarding the auxiliary feedwater pump testing interval, were issued on November 9, 1994. Due to an administrative oversight, the Technical Specifications (TS) pages contained two minor typographical errors. Please substitute the enclosed revised set of TS pages for the ones sent to you on November 9.

Sincerely,

Original signed by:
Victor Nerses, Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-369 and 50-370

Enclosure: As stated

cc w/encl: See next page

DISTRIBUTION
Docket File
PUBLIC
PD# 23 Reading
S. Varga
OGC
G. Hill, IRM (2),
C. Grimes, DOPS/OTSB
ACRS (4)
OPA
M. Sinkule, RII
B. Boger, RII

DOCUMENT NAME: G:\MCGUIRE\MCG90355.COR
To receive a copy of this document, indicate in the box: "C" = Copy without attachment/enclosure "E" = Copy with attachment/enclosure "N" = No copy

OFFICE	LA:PD23:DRPE	E	PM:PD23:DRPE	E	D:PD23:DRPE				
NAME	CNorsworthy		VNerses:cn		HBerkow				
DATE	11/28/94		11/28/94		11/29/94				

OFFICIAL RECORD COPY

DF01

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

November 29, 1994

Mr. T. C. McMeekin
Vice President, McGuire Site
Duke Power Company
12700 Hagers Ferry Road
Huntersville, NC 28078-8985

SUBJECT: MCGUIRE NUCLEAR STATION, UNITS 1 AND 2 - AMENDMENTS NOS. 151 AND 133 - AUXILIARY FEEDWATER PUMP TESTING INTERVAL (TAC NOS. 90355 AND 90356)

Dear Mr. McMeekin:

Amendments Nos. 151 and 133 to Facility Operating Licenses NPF-9 and NPF-17, respectively, for the McGuire Nuclear Station, Units 1 and 2, regarding the auxiliary feedwater pump testing interval, were issued on November 9, 1994. Due to an administrative oversight, the Technical Specifications (TS) pages contained two minor typographical errors. Please substitute the enclosed revised set of TS pages for the ones sent to you on November 9.

Sincerely,

Victor Nerses

Victor Nerses, Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-369 and 50-370

Enclosure: As stated

cc w/encl: See next page

Mr. C. K. McCoy
Georgia Power Company

Vogtle Electric Generating Plant

cc:
Mr. J. A. Bailey
Manager - Licensing
Georgia Power Company
P. O. Box 1295
Birmingham, Alabama 35201

Mr. J. B. Beasley
General Manager, Vogtle Electric
Generating Plant
P. O. Box 1600
Waynesboro, Georgia 30830

Regional Administrator, Region II
U. S. Nuclear Regulatory Commission
101 Marietta Street, NW., Suite 2900
Atlanta, Georgia 30323

Office of Planning and Budget
Room 615B
270 Washington Street, SW.
Atlanta, Georgia 30334

Office of the County Commissioner
Burke County Commission
Waynesboro, Georgia 30830

Mr. J. D. Woodard
Senior Vice President -
Nuclear Operations
Georgia Power Company
P. O. Box 1295
Birmingham, Alabama 35201

Harold Reheis, Director
Department of Natural Resources
205 Butler Street, SE. Suite 1252
Atlanta, Georgia 30334

Attorney General
Law Department
132 Judicial Building
Atlanta, Georgia 30334

Mr. Ernie Toupin
Manager of Nuclear Operations
Oglethorpe Power Corporation
2100 East Exchange Place
Tucker, Georgia 30085-1349

Charles A. Patrizia, Esquire
Paul, Hastings, Janofsky & Walker
12th Floor
1050 Connecticut Avenue, NW.
Washington, DC 20036

Arthur H. Domby, Esquire
Troutman Sanders
NationsBank Plaza
600 Peachtree Street, NE.
Suite 5200
Atlanta, Georgia 30308-2216

Resident Inspector
U. S. Nuclear Regulatory Commission
P. O. Box 572
Waynesboro, Georgia 30830

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three independent steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE with:

a. Two motor-driven auxiliary feedwater pumps, each capable of being powered from separate emergency busses, and

b. One steam turbine-driven auxiliary feedwater pump capable of being powered from an OPERABLE steam supply system.*

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With two auxiliary feedwater pumps inoperable, be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With three auxiliary feedwater pumps inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump to operable status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days by:

   1) Verifying that each non-automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position;

   2) Verifying that each automatic valve in the flow path is in the fully open position whenever the Auxiliary Feedwater System is placed in automatic control or when above 10% RATED THERMAL POWER; and

   3) Verifying that the isolation valves in the auxiliary feedwater suction line from the upper surge tanks are open with power to the valve operators removed.

*Not applicable with steam pressure less than 900 psig.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 92 days on a STAGGERED BASIS by:

1) Verifying that each motor-driven pump develops a discharge pressure of greater than or equal to 1210 psig at a flow of greater than or equal to 450 gpm; and

2) Verifying that the steam turbine-driven pump develops a discharge pressure of greater than or equal to 1210 psig at a flow of greater than or equal to 900 gpm when the secondary steam supply pressure is greater than 900 psig*. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

c. At least once per 18 months during shutdown by:

1) Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of an Auxiliary Feedwater Actuation test signal,

2) Verifying that each auxiliary feedwater pump starts as designed automatically upon receipt of an Auxiliary Feedwater Actuation test signal, and

3) Verifying that the valve in the suction line of each auxiliary feedwater pump from the Nuclear Service Water System automatically actuates to its full open position within less than or equal to 13 seconds on a Low Suction Pressure test signal.

* This verification is not required to be performed until 24 hours after achieving greater than or equal to 900 psig in the secondary side of the steam generator.

SAFETY VALVES (Continued)

109 = Power Range Neutron Flux-High Trip Setpoint for four loop operation,

* = Maximum percent of RATED THERMAL POWER permissible by P-8 Setpoint for three loop operation. This value left blank pending NRC approval of three loop operation,

X = Total relieving capacity of all safety valves per steam line in lbs/hour, and

Y = Maximum relieving capacity of any one safety valve in lbs/hour.

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the Auxiliary Feedwater System ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss-of-offsite power.

Each electric motor-driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 450 gpm at a pressure of 1210 psig to the entrance of the steam generators. The steam-driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 900 gpm at a pressure of 1210 psig to the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the RHR System may be placed into operation.

Verification of the steam turbine-driven pump discharge pressure should be deferred until suitable test conditions are established (i.e., greater than or equal to 900 psig in the secondary side of the steam generator). This deferral is required because until 900 psig is reached, there is insufficient steam pressure to perform the test.

3/4.7.1.3 SPECIFIC ACTIVITY

The limitations on Secondary Coolant System specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of a steam line rupture. This dose also includes the effects of a coincident 1.0 gpm reactor to secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the accident analyses.