May 11, 2005

Mr. D. M. Jamil
Vice President
Catawba Nuclear Station
Duke Energy Corporation
4800 Concord Road
York, SC 29745

SUBJECT: CATAWBA NUCLEAR STATION, UNIT 2 RE: REQUEST FOR ADDITIONAL INFORMATION (TAC NO. MC4993)

Dear Mr. Jamil:

By letters dated October 19 and December 2, 2004, Duke Energy Corporation submitted for Nuclear Regulatory Commission (NRC) staff review, an evaluation of a flaw indication in the reactor coolant hot leg to steam generator nozzle connection, that was discovered on October 7, 2004, during the 13th refueling outage for Catawba Nuclear Station, Unit 2. The NRC staff has reviewed the submittals and has determined that additional information is required, as identified in the Enclosure.

We discussed these issues with your staff on May 5, 2005. Your staff indicated that you would attempt to provide your response by June 30, 2005.

Please contact me at (301) 415-1842, if you have any questions on these issues.

Sincerely,

/RA/

Sean E. Peters, Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-414

Enclosure: As stated

cc w/encl: See next page

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REQUEST FOR ADDITIONAL INFORMATION

DUKE POWER COMPANY

CATAWBA NUCLEAR STATION, UNIT 2

DOCKET NO. 50-414

The Nuclear Regulatory Commission (NRC) staff has reviewed the licensee's submittals dated October 19 and December 2, 2004, regarding an evaluation of a flaw indication in the reactor coolant hot leg to steam generator nozzle connection, that was discovered on October 7, 2004, during the 13th refueling outage for Catawba Nuclear Station, Unit 2. The NRC staff has identified the following information that is needed to enable the continuation of its review.

1. In the letter dated October 19, 2004, you stated, "[t]he indication was located near the interface between the safe-end and field weld at the bottom of the nozzle." Please confirm whether the flaw indication is in the safe end or in the field weld.

2. In the letter dated October 19, 2004, you stated, "[t]his letter submits the fracture mechanics analysis to the NRC (see attachment)". The NRC staff did not find in your submittal an evaluation of the detected flaw indication (1 inch long circumferential embedded flaw, 1.01 inches from the outside diameter of the pipe) using WCAP-15658-P, Revision 1. Please provide this information. The response should include the WCAP figure number (Figure A-4.6, Figure A-4.7, Figure A-4.8, or Figure A-4.9) that you used for evaluation of the detected flaw in the steam generator primary nozzle weld region. The response should also include the depth of the detected flaw (the size of the flaw in the wall thickness direction).

3. On Page 3-1 of WCAP-15658-P, Revision 1, "Flaw Evaluation Handbook for Catawba Unit 2 Steam Generator Primary Nozzle Weld Regions," November 2004", it is stated, "[t]he stress intensity factor calculation for an embedded flaw was taken from the work by Shah and Kobayashi [6] which is applicable to an embedded flaw in an infinite medium....This expression has been shown to be applicable to embedded flaws in a thick-walled pressure vessel in a paper by Lee and Bamford [7]." Please demonstrate the applicability of Kobayashi's formulas for embedded flaws to your current application by addressing (1) the difference between the finite geometry of the current application and an infinite medium discussed in Kobayashi's paper, and (2) the difference between the ratio of plate thickness to crack depth, t/2a, of the current application and that discussed in Kobayashi's paper. Provide Reference 7 of WCAP-15658-P, Revision 1 (Paper 83-PVP-92 by Lee and Bamford) if you believe it would help your explanation.

4. On Page 3-4 of WCAP-15658-P, Revision 1, it is stated, "NRC procedures exist for addressing the impact of thermal aging on fracture toughness for full-service life. The approved procedures were applied to the nozzle safe end to pipe weld, as well as to the cast piping itself." Provide the specific document (e.g., NUREG number) and parameters used (e.g., ferrite content) in your determination of fracture toughness for full-service life using NRC procedures.

Explain how you use these NRC procedures to determine the first set of proprietary J_{Ic} and T_{mat} given on Page 3-4. It is further stated on this page, "[e]ven with thermal aging, equivalent to full service for SAW welds, the tearing modulus remains high (>100) and the unaged toughness, J_{Ic}, is not significantly reduced." Provide information supporting this statement.

Catawba Nuclear Station, Unit 2
cc:

Mr. Lee Keller, Manager
Regulatory Compliance
Duke Energy Corporation
4800 Concord Road
York, South Carolina 29745

Ms. Lisa F. Vaughn
Duke Energy Corporation
Mail Code - PB05E
422 South Church Street
P.O. Box 1244
Charlotte, North Carolina 28201-1244

Ms. Anne Cottingham, Esquire
Winston and Strawn LLP
1700 L Street, NW
Washington, DC 20006

North Carolina Municipal Power
Agency Number 1
1427 Meadowwood Boulevard
P.O. Box 29513
Raleigh, North Carolina 27626

County Manager of York County
York County Courthouse
York, South Carolina 29745

Piedmont Municipal Power Agency
121 Village Drive
Greer, South Carolina 29651

Ms. Karen E. Long
Assistant Attorney General
North Carolina Department of Justice
P.O. Box 629
Raleigh, North Carolina 27602

NCEM REP Program Manager
4713 Mail Service Center
Raleigh, North Carolina 27699-4713

North Carolina Electric Membership Corp.
P.O. Box 27306
Raleigh, North Carolina 27611

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
4830 Concord Road
York, South Carolina 29745

Mr. Henry Porter, Assistant Director
Division of Waste Management
Bureau of Land and Waste Management
Dept. of Health and Environmental Control
2600 Bull Street
Columbia, South Carolina 29201-1708

Mr. R.L. Gill, Jr., Manager
Nuclear Regulatory Issues
and Industry Affairs
Duke Energy Corporation
526 South Church Street
Mail Stop EC05P
Charlotte, North Carolina 28202

Saluda River Electric
P.O. Box 929
Laurens, South Carolina 29360

Mr. Peter R. Harden, IV, Vice President
Customer Relations and Sales
Westinghouse Electric Company
6000 Fairview Road
12th Floor
Charlotte, North Carolina 28210

Mr. T. Richard Puryear
Owners Group (NCEMC)
Duke Energy Corporation
4800 Concord Road
York, South Carolina 29745

Mr. Richard M. Fry, Director
Division of Radiation Protection
NC Dept. of Environment, Health,
and Natural Resources
3825 Barrett Drive
Raleigh, North Carolina 27609-7721

Mr. Henry Barron
Group Vice President, Nuclear Generation
and Chief Nuclear Officer
P.O. Box 1006-EC07H
Charlotte, NC 28201-1006

Diane Curran
Harmon, Curran, Spielbergy &
Eisenberg, LLP
1726 M Street, NW
Suite 600
Washington, DC 20036