January 28, 2002

Mr. M. S. Tuckman
Executive Vice-President
Nuclear Generation
Duke Energy Corporation
PO Box 1006
Charlotte, NC 28201-1006

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE MCGUIRE NUCLEAR STATION, UNITS 1 AND 2, AND CATAWBA NUCLEAR STATION, UNITS 1 AND 2, LICENSE RENEWAL APPLICATION (LRA)

Dear Mr. Tuckman:

By letter dated June 13, 2001, Duke Energy Corporation (Duke) submitted for Nuclear Regulatory Commission (NRC) review an application, pursuant to 10 CFR Part 54, to renew the operating licenses for the McGuire Nuclear Station, Units 1 and 2, and Catawba Nuclear Station, Units 1 and 2. The NRC staff is reviewing the information contained in this license renewal application and has identified, in the enclosure, areas where additional information is needed to complete its review. Specifically, the enclosed request for additional information (RAI) is from the following section(s) of the LRA:

Section 3.2, Aging Management of Engineered Safety Features

Please provide a schedule by letter, or electronic mail for the submittal of your response within 30 days of the receipt of this letter. Additionally, the staff would be willing to meet with Duke prior to the submittal of the response to provide clarification of the staff's request for additional information.

Sincerely,

/**RA**/

Rani L. Franovich, Project Manager
License Renewal and Environmental Impacts Program
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Docket Nos. 50-369, 50-370, 50-413 and 50-414

Enclosures: As stated

cc w/encl: See next page

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NAME	RFranovich	EGHylton	PTKuo	CIGrimes
DATE	01/24/2002	01/24/2002	01/28/2002	01/28/2002

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

Ms. Lisa F. Vaughn
Duke Energy Corporation
422 South Church Street
Charlotte, North Carolina 28201-1006

Anne Cottington, Esquire
Winston and Strawn
1400 L Street, NW
Washington, DC 20005

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

Ms. Elaine Wathen, Lead REP Planner
Division of Emergency Management
116 West Jones Street
Raleigh, North Carolina 27603-1335

Mr. Robert L. Gill, Jr.
Duke Energy Corporation
Mail Stop EC-12R
P. O. Box 1006
Charlotte, North Carolina 28201-1006

Mr. Alan Nelson
Nuclear Energy Institute
1776 I Street, N.W., Suite 400
Washington, DC 20006-3708

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

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

Mr. Virgil R. Autry, Director
Dept of Health and Envir Control
2600 Bull Street
Columbia, South Carolina 29201-1708

Mr. C. Jeffrey Thomas
Manager - Nuclear Regulatory Licensing
Duke Energy Corporation
526 South Church Street
Charlotte, North Carolina 28201-1006

Mr. L. A. Keller
Duke Energy Corporation
526 South Church Street
Charlotte, North Carolina 28201-1006

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

Mr. Peter R. Harden, IV
VP-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
North Carolina Dept of Env, Health, and
Natural Resources
3825 Barrett Drive
Raleigh, North Carolina 27609-7721

County Manager of
Mecklenburg County
720 East Fourth Street
Charlotte, North Carolina 28202

Michael T. Cash
Regulatory Compliance Manager
Duke Energy Corporation

McGuire Nuclear Site
12700 Hagers Ferry Road
Huntersville, North Carolina 28078

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
12700 Hagers Ferry Road
Huntersville, North Carolina 28078

Dr. John M. Barry
Mecklenburg County
Department of Environmental Protection
700 N. Tryon Street
Charlotte, North Carolina 28202

Mr. Gregory D. Robison
Duke Energy Corporation
Mail Stop EC-12R
526 S. Church Street
Charlotte, NC 28201-1006

Mary Olson
Nuclear Information & Resource Service
Southeast Office
P.O. Box 7586
Asheville, North Carolina 28802

Paul Gunter
Nuclear Information & Resource Service
1424 16th Street NW, Suite 404
Washington, DC 20036

Lou Zeller
Blue Ridge Environmental Defense League
P.O. Box 88
Glendale Springs, North Carolina 28629

Don Moniak
Blue Ridge Environmental Defense League
Aiken Office
P.O. Box 3487
Aiken, South Carolina 29802-3487

Request for Additional Information
McGuire Nuclear Station, Units 1 and 2, and
Catawba Nuclear Station, Units 1 and 2

3.2 Aging Management of Engineered Safety Features

3.2-1 Since closure bolting is exposed to air, moisture, and leaking fluid (boric acid) environments, it is subject to the aging effects of loss of material and crack initiation and growth. Tables in Sections 3.2, 3.3 and 3.4 do not address these aging effects for closure bolting in these systems. Please indicate where in the LRA the AMR results for closure bolting are documented, or provide a justification for excluding closure bolting from an AMR, the results of which are documented in the referenced tables of the LRA.

Similarly, Table 3.5-3 provides no information to address the cracking initiation and growth from SCC for high strength low-alloy bolts. Last item on page 3.5-18 of Table 3.5-1 of the SRP-LR addresses the issue of bolting integrity for ASME Class I piping and components supports. It indicates that no further evaluation is required if there is a bolting integrity program to address the cracking initiation and growth from SCC for high strength low-alloy bolts. State whether there is such a program and provide the reference.

3.2-2 The application does not define any of the aging effect listed in Tables 3.2-1 through 3.2-8. Paragraph 3.2.1, Aging Management Review Results Tables, Column 5 states that aging effects identification process is consistent with the process used in Oconee Nuclear Station. The Oconee application defined each aging effect in its Appendix C. The staff requests that the applicant indicate if the aging effects identification process is identical to the one described in the Oconee LRA and confirm that the definitions provided in Appendix C of the Oconee LRA apply to the Catawba/McGuire LRA as well. If there are any differences between the Oconee and Catawba/McGuire LRAs, please identify them.

3.2-3 In Table 3.2-2, on page 3.2-22, the applicant specifies the Fluid Leak Management Program and the Inspection Program for Civil Engineering Structures and Components as the aging management programs (AMPs) for carbon steel valve bodies. However, on page 3.2-23, the applicant specifies only the Inspection Program for Civil Engineering Structures and Components as the AMP for carbon steel valve bodies. Does the Fluid Leak Management Program Scope include mechanical systems and components outside the reactor building? Does the LRA reflect an assumption that boric acid corrosion can occur only in a reactor building environment and not in a sheltered environment? Or are the steam generator wet lay-up system carbon steel valve bodies in a sheltered environment that houses no potential sources of leaking borated water?

3.2-4 In Table 3.2-7, the applicant identifies that the internal surfaces of the carbon steel residual heat removal (ND) heat exchanger (HX) shells and ND pump seal water HX shells are both exposed to treated water environments. Clarify either by reference to appropriate information in the application or by discussion why cracking is identified as an applicable aging effect for the ND HX shells but not for the ND pump seal water HX shells.

3.2-5 In Table 3.2-8, you identify that the external surfaces of some of the carbon steel piping and valve bodies in the safety injection (NI) systems are exposed to sheltered air environments. Clarify either by reference to appropriate information in the application or by discussion why loss of material is identified as an applicable aging effect for the carbon steel NI piping that is exposed sheltered air but not for the carbon steel NI valve bodies that are exposed to the same environment.

3.2-6 Table 3.2-2, Aging Management Review Results - Containment Isolation System, on page 3.2-16 of the LRA indicates that Ice Condenser Refrigeration System carbon steel piping exposed to the reactor building external environment (the third “Pipe” entry from the top of the page) has no identified aging effects. The staff questions why this piping was not identified as susceptible to loss of material and subject to the Fluid Leak Management Program and the Inspection Program for Civil Engineering Structures and Components. This finding appears to be inconsistent with the LRA’s treatment of similar or identical materials and components in the same environment.

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E. Hylton

E-MAIL:
PUBLIC
J. Johnson
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B. Thomas
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M. Mayfield
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RLEP Staff

R. Martin
C. Patel
C. Julian (RII)
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A. Fernandez (OGC)
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