

January 28, 2002

Mr. M. S. Tuckman
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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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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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