December 11, 2001

LICENSEE : Duke Energy Corporation

FACILITIES: McGuire, Units 1 and 2, and Catawba, Units 1 and 2

SUBJECT: TELECOMMUNICATION WITH DUKE ENERGY CORPORATION TO DISCUSS INFORMATION IN THEIR LICENSE RENEWAL APPLICATION ON AGING MANAGEMENT OF CONTAINMENTS, STRUCTURES AND STRUCTURAL SUPPORTS

On October 30, 2001, after the NRC (the staff) reviewed information provided in Section 3.5 of the license renewal application (LRA), a conference call was conducted between the staff and Duke Energy Corporation (the applicant) to clarify information presented in the application pertaining to aging management programs for mechanical systems and components. Participants in the conference call are provided in an attachment.

The questions asked by the staff, as well as the responses provided by the applicant, are as follows:

<u>Table 3.5-1, Aging Management Review Results - Reactor Building</u>
(Questions 1-7 and 10-16 were discussed on October 25, 2001, and are documented in the associated telephone conference summary).

8. Why are the aging effects in some components not identified even though they are fabricated from the same material and are in the same environment as components that have been identified as having specified aging effects?

   A) Table 3.5-1 indicates the fuel transfer canal liner plate, sump liner and sump screens were fabricated from stainless steel, operate in the reactor building environment and are not subject to an aging effect. Bellows were fabricated from stainless steel, operate in the reactor building environment and are subject to cracking as an aging effect. Provide your basis, including plant-specific and industry operating experiences, for concluding that the fuel transfer canal liner plate, sump liner and sump screens are not subject to cracking.

      The applicant indicated that they would provide their basis for concluding that the fuel transfer canal liner plate, sump liner and sump screens are not subject to cracking in a written response to a staff request for additional information.

   B) Table 3.5-3 indicates that steel components in sheltered, reactor building and external (yard only) environments are subject to loss of material. Cable Trays & Conduit, Control Boards, Control Room Ceiling and New Fuel Storage Racks are steel components, are in similar environments and are not subject to an aging effect. Provide your basis, including plant-specific and industry operating

experiences, for concluding Cable Trays & Conduit, Control Boards, Control Room Ceiling, and New Fuel Storage Racks are not subject to loss of material.

The applicant indicated that they understood the question and would like to respond in writing. As such, the staff will submit a request for additional information so that they can complete their review of this item.

9. Table 3.5-1 indicates bellows (in penetration) are subject to cracking and the Containment Leak Rate Testing Program is credited for managing this aging effect. The Containment Leak Rate Testing Program indicates: "The Containment Leak Rate Testing Program supplements the Containment Inservice Inspection Plan-IWE. The containment Inservice Inspection Plan-IWE, which implements the provisions of the ASME Code Section XI, Subsection IWE, is the primary method for detection of the aging effects for steel components of containment. The Containment Leak Rate Testing Program is a performance monitoring program."

   A) Based on the description of the Containment Inservice Inspection Plan-IWE in the Containment Leak Rate Testing Program, will the Bellows be inspected to the provisions of the ASME Code Section XI, Subsection IWE to detect cracking?

   B) Stress corrosion cracking is a concern for dissimilar metal welds and stainless steel components that are exposed to corrosive environment. In addition, cyclic fatigue could cause cracking. Provide the plant-specific experience and industry operating experience that these type of cracking mechanisms in penetrations can be detected by a Containment Leak Rate Testing Program and the Containment Inservice Inspection Plan-IWE.

   The applicant indicated that, with regard to part (A) of the question, Duke is not crediting Subsection IWE with detection of cracks in the penetration bellows. Alternatively, the Containment Leak Rate Testing Program, B.3.8 is credited for monitoring the presence of cracks in penetration bellows. With regard to part (B) of this question, the applicant indicated that operating experience has been provided which demonstrates the effectiveness of the Containment Leak Rate Testing Program in detecting leaks which are a result of cracking. Operating experience was provided in B.3.8, Containment Leak Rate Testing Program, of the LRA. The McGuire Main Steam bellows was removed because of leakage in multiple locations. The bellows was metallurgically examined and analyzed. Duke determined from the metallurgical results that the leakage was due to cracking. The staff understands the applicant's response and will consider the information provided but may request additional information to allow the applicant an opportunity to provide their response formally.

A draft of this telecommunication summary was provided to the applicant to allow them the opportunity to comment prior to the summary being issued.

/***RA***/

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Office of Nuclear Reactor Regulation

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

Attachment: As stated

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/***RA***/

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