January 28, 2002

Mr. M. S. Tuckman
Executive Vice-President
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PO Box 1006
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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 2.4, Scoping and Screening Results: Structures
Section 3.5, Aging management of Containments, Structures, and Components Supports
Section 4.6, Containment Liner Plate, Metal Containments, Penetration Fatigue Analysis
Section 4.7.3, Depletion of Nuclear Service Water Pond Volume due to Runoff
Appendix B, Aging Management Programs (Structures)

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
Mr. M. S. Tuckman

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

<u>2.4.1 Scoping and Screening Results: Reactor Buildings</u>

2.4.1-1 The staff reviewed Figures 3-11, 3-12 and 3-13 of the Catawba UFSAR, which depict hot, cold, and feedwater penetrations. The staff requests the applicant to indicate if these penetrations are representative of all concrete shield building penetrations, including those at McGuire. Furthermore, Table 3.5-1 for the concrete shield building does not list penetration structures/components (e.g., anchor rings, penetration sleeves, pipe caps and restraint rings) that appear to perform intended functions (provide structural support for piping and maintain containment integrity) defined by 10 CFR 54.4 and are passive. Please indicate if these structures/components are within the scope of license renewal and subject to an AMR. If they are not, please provide the basis for this determination?

2.4.1-2 The combined License Renewal Application (LRA) for Catawba/McGuire Nuclear Station's references table 3.5-1, "Concrete Shield Building," which identifies structures and components that are in the scope of license renewal and subject to an aging management review (AMR). Table 3.5-1 identifies McGuire as having different reinforcement (dowels) from that of Catawba. Provide the staff with an explanation of the differences between the two plants. Please clarify for the staff whether or not the different SCs at Catawba as described above are within the scope and subject to an AMR.

2.4.1-3 Updated Final Safety Analysis Report (UFSAR) Section 3.8.1.1.2 states that a three foot thick removable concrete cover is mounted on a track and rigidly attached to the Reactor Building during operation. Table 3.5-1, "Concrete Shield Building" and LRA Section 2.4.1.1 does not identify the concrete cover, tracks, and other supporting structures as being in the scope. Explain to the staff why these SCs were not included within the scope and subject to an AMR.

2.4.1-4 Table 3.5-1, "Aging Management Review Results-Reactor Building," is broken down into the following sections: Concrete Shield Building, Steel Containment, Ice Condenser Components, and Reactor Building Interior Structural Components. Neither Section 2.4.1.1 nor the corresponding section of Table 3.5-1 concerning the shield building include penetrations. Clarify for the staff how the LRA handles the various penetrations to the Reactor Building.

2.4.1-5 This section lists the various components that are included in the scope for the steel containment and subject to an AMR. In addition to the SCs listed in Section 2.4.1.2, UFSAR Section 3.8.2.1 list the following structures and components which are not identified in the LRA:

Seals on personnel locks
Penetration sleeves

Purge penetration
Double compressible seals, and
Bolted flanges

These SCs are not identified in LRA Section 2.4.1.2 or Table 3.5-1 for Steel Containment. The staff believes that these SCs perform an intended function without moving parts and is not replaced based on qualified life or specified time period. Provide an AMR for the above SCs or explain why they are excluded from being within the scope of license renewal.

2.4.1-6 NUREG/CR-4652, "Concrete Component Aging and Its Significance Relative to Life Extension of Nuclear Power Plants," indicates that failures in the buttonheads and tendons on the missile shield at McGuire led to a modification which replaced these structural components with threaded rods that were grouted into place. Explain why these rods are not within the scope and subject to an AMR.

2.4.1-7 Section 2.4.1.3 lists the internal structures that are within the scope and subject to an AMR. However, the structural supports for the various structures are not included within Table 3.5-1. Section 2.4.3, "Component Supports" does not include supports for structures. Clarify whether the attachments to and structural supports for the internal structures (e.g., intermediate structural supports for structures connected to the crane wall) are within the scope and subject to an AMR, or explain why the components are excluded from the scope of license renewal.

<u>2.4.2 Scoping and screening Results: Other Structures</u>

2.4.2-1 Section 2.4.2 of the LRA for both McGuire and Catawba describes the "other structures," which include auxiliary buildings, condenser cooling water intake structure, nuclear service water structures, standby nuclear service water pond dam, standby shutdown facility, turbine building (including service building), unit vent stack, and yard structures. However, the applicant provides only the systems drawings for the LRA but does not provide any structural drawings. The staff reviewed Catawba UFSAR Figure 1-20 McGuire UFSAR Figure 2-4; however, the UFSAR figures were either of poor resolution or provided security fence boundaries, which are not useful to the staff in performing license renewal scoping results reviews. Therefore, the staff requests that the applicant provide general structural drawings (i.e., location plans and elevations and/or structural details, such as the Nuclear Steam Supply System supports) for the other structures at Catawba and McGuire.

2.4.2-2 In Section 2.4.2.1 of the LRA, the applicant describes the auxiliary building and the structures within its review boundary, including the control building, diesel generator buildings, and fuel buildings. The applicant states that the fuel buildings are seismic Category I structures which provide storage for the new fuel and spent fuel but the LRA does not describe the structures. Section 2.8.4.1.1b of the Catawba UFSAR) addresses the structures of the spent fuel building for the Catawba plant, including the spent fuel pool and cask handling area. McGuire UFSAR, Section 3.8.4.2, addresses the structures of the new fuel storage vault for the McGuire plant. However, Table 3.5-2 of the LRA lists only spent fuel pool liner plate as the component subject to an AMR for both plants. The components of the new fuel storage vault are not listed in the table for an AMR. The staff considers that the components (other than the liner plates) as stated in the UFSARs should be included in the table for an AMR, such as concrete enclosures, roof of the pool, fuel handling bridge crane, fuel transfer up-ending canal, etc. Where in the AMR results table are the components that are applicable to the fuel building?

2.4.2-3 Section 2.4.2.1 of the LRA states that the groundwater drainage system is provided for the auxiliary buildings and diesel generator buildings to maintain the normal groundwater level near the base of these structures. McGuire UFSAR, Section 2.4.1.3.5, states that a permanent Category I under-drain groundwater system is installed to maintain the groundwater level below the elevation to ensure that uplifting and overturning of the auxiliary building will not occur. However, the applicant did not address whether the foundation mat and the lower portion of the walls have expansion joints, water-stops or waterproofing membranes (or elastomer components, if any) that can prevent groundwater in-leakage into the concrete construction joints. Provide information on the structural sealant or elastomer components for the below-grade construction joints. Explain whether the water-stops and the components of the under-drain groundwater system should be included in Table 3.5-2 of the LRA for an AMR.

2.4.2-4 Section 2.4.2.1 of the LRA states that the main steam doghouses and the upper head injection tank building at the Catawba plant are within the scope of license renewal. However, the applicant did not describe these structures and Table 3.5-2 of the LRA does not define which of the components that are applicable to these structures. There is no supporting information in the UFSAR that can be used to verify their structural components. Provide additional information on these structures and their components that are subject to an AMR.

2.4.2-5 Section 2.4.2.2 of the LRA states that the McGuire condenser cooling water intake structure is a Category III structure which is not designed to withstand design basis seismic loading. It also states that the fire pump rooms are the only parts of the structure that are within the scope of license renewal. There is insufficient information in the LRA regarding the structural components that support the fire pumps. Describe the fire pump room and how its structural components meet the intent of 10 CFR 54.21 for an AMR.

2.4.2-6 Section 2.4.2.2 of the LRA states that the fire pumps at the Catawba plant are supported by the low-pressure service water intake structure, which is included in the yard structures. Section 2.4.2.8 (yard structures) of the Catawba LRA states that the fire pumps and the support structure are within the scope of license renewal. However, Section 2.4.2.8 does not describe the low-pressure service water intake structure. Provide information on the structures that support the fire pumps.

2.4.2-7 Section 2.4.2.3 of the LRA states that the nuclear service water structures at the Catawba plant include several structures. It is not clearly that the structures described in the section are the structures within the boundary of the nuclear service water structures for license renewal. Provide a drawing that highlight all the structures that are subject to an AMR and identify which of the components (other than the components specified) listed in Table 3.5-2 of both the LRAs that are applicable to the nuclear service water structures.

2.4.2-8 Section 2.4.2.4 of the LRA states that the standby nuclear service water pond dam at McGuire is an earthen embankment that has been designed as a seismic Category I structure. Table 3.5-2 of both plants' LRAs lists the earthen embankment as the component subject to an AMR. Explain whether other structural components of the pond dam that may perform an intended function should be listed in the table, such as the drain pipes, observation wells, and piezometers, if any.

2.4.2-9 Section 2.4.2.5 of the LRA states that the standby shutdown facility structure at McGuire is a steel-frame and masonry structure. In Table 3.5-2 of the LRA, only block walls are specified as the components subject to an AMR. Please identify other components listed in the table that are also applicable to the standby shutdown facility structure.

2.4.2-10 Section 2.4.2.6 of both the LRAs states that the turbine buildings (including service building) are Category III structures that are constructed of a steel frame superstructure supported on a reinforced concrete substructure. Explain the relationship between the service building and the turbine building. Identify the structural components (other than that specified for turbine building only) in Table 3.5-2 of both the LRAs that are applicable to the turbine building and service building for an AMR.

2.4.2-11 In Section 2.4.2.8 of the LRA, the applicant describes the yard structures, trenches, and drainage systems for McGuire and Catawba. However, there is no supporting information or document that can be used to verify the content of this section. Provide a drawing for each plant that shows the location of the yard structures and highlight the components that are within the scope of license renewal.

2.4.2-12 Section 2.4.3 of both the LRA states that the component supports also include the Class I nuclear steam supply system (NSSS) supports. The NSSS supports within the scope of license renewal are the reactor coolant system piping supports; pressurizer upper and lower supports; reactor vessel support; control rod drive seismic structure supports; steam generator vertical, lower lateral, and upper supports; and reactor coolant pump lateral and vertical support assemblies. However, the LRA does not provide any information on the support structures, and there is insufficient information in the UFSAR to support the staff's review. Since each of the NSSS support assemblies are designed entirely different, the staff is unable to verify the components that require an AMR. Describe the structures of the NSSS support assemblies that are within the scope of license renewal and subject to an AMR.

<u>3.5 Aging management of Containments, Structures, and Component Supports</u>

3.5-1 Table 3.5-1 of the LRA indicates that no aging management is needed for the below grade portion of the foundation mat for the concrete shield buildings. Table 3.5-2 of the LRA lists several below grade component types (i.e., foundation caissons for the

McGuire turbine building, other foundations, reinforced concrete beams, columns, floor slabs, walls, foundation dowels, wear slab, manholes & covers, and trenches) as having exposed to no aging effects and therefore, no AMPs are identified for these items. It should be noted that the staff has a generic position requiring an AMP for all concrete elements within the scope of review (refer to the following RAI 3.5-7). The applicant is requested to provide information indicating compliance to the staff position. A conference call was held between the applicant and the NRC staff on October 25, 2001. A summary of this conference call was issued November 30, 2001. During this conference call, the applicant indicated that its positions as listed in the Tables are supported by McGuire/Catawba plant specific operating data, including five years of recent below-grade-environment test results, which show generally benign environmental conditions. The staff requests that the applicant provide these test data to confirm that below-grade chemistry is not aggressive. In addition, please indicate the frequency of future tests to periodically monitor below-grade chemistry and demonstrate that the environment is not aggressive during the period of extended operation.

3.5-2 Table 3.5-1 of the LRA states that Technical Specification SR 3.6.16.3 visual inspection is credited for managing change in material properties due to leaching of both the shell wall and dome of the shield building. Describe the present extent of the aging due to change in material properties resulting from leaching for the shield buildings of Catawba and McGuire. Indicate the inspection experience gathered to date (e.g., growth of leached surface area, indications of loss of material of embedded rebars in the leached areas) and discuss the basis for maintaining that the visual inspection program should adequately manage the aging effect of the shield buildings due to leaching during the extended period of operation for both plants.

During the October 25, 2001, conference call, the applicant indicated that this question was addressed in Appendix B of the LRA under the Technical Specification Surveillance Requirement 3.6.16.3 Visual Inspection program, which requires a visual inspection of the exposed interior and exterior surfaces of the reactor building three times every ten years. The applicant further asserted that results of these visual inspections indicate that the condition of the shield buildings and embedded rebar is not degrading. According to the Technical Specification Surveillance Requirement 3.6.16.3 Visual Inspection program, leaching has been observed on the interior of the reactor building domes at McGuire near the dome-to-shell interface. Maintenance had been planned for the dome exterior to minimize water intrusion which was later canceled upon reinspection. The staff requests the applicant to provide the extent of the degradation observed and clarify the basis for canceling the maintenance task that had already been scheduled.

3.5-3 With respect to component types, “steel containment vessel,” and “structural steel beams, columns, plates & trusses” listed in Table 3.5-1 of the LRA, no information is provided regarding potential loss of material due to corrosion of inaccessible areas in liner plates and steel structures. SRP Section 3.5.2.2.1.4 states that loss of material due to corrosion could occur in inaccessible areas of steel structures and liner plate for all types of PWR and BWR containments. The GALL report recommends further evaluation to manage the aging effects for steel components in inaccessible areas, when conditions do not exist in accessible areas that could indicate the presence of, or result in, degradation to such inaccessible areas. Discuss how this potential aging effect is managed for Catawba and McGuire. Additionally, provide information describing the applicants' planned disposition of damaged seals between the containment floor and the containment steel liner that have often been observed in operating plants as a result of inservice inspection.

3.5-4 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?

1.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 Fuel Transfer Canal Liner Plate, Sump Liner and Sump screens are not subject to cracking.

2. 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.

3.5-5 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."

1. 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?

2. 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. Please 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.

3. The acceptance criteria in Section B.3.8, "Containment Leak Rate Testing Program" state that the space between dual-ply bellows shall be subjected to a low pressure leak

test, with no detectable leakage. Please provide the minimum pressure requirement that makes this a meaningful test.

3.5-6 Regarding the reinforced concrete beams, columns, floor slabs, walls and some localized portions of the top layer-basemat concrete, which are rendered inaccessible because of the layout of the Ice Condenser/Ice Baskets System, increases in porosity and permeability, cracking, loss of material (spalling, scaling,) due to aggressive chemical attack and loss of material due to corrosion of embedded steel could occur. The Gall report (e.g., Section A1.1) recommends further evaluation to manage the aging effects for these inaccessible areas, when conditions do not exist in accessible areas that could indicate the presence of, or result in, degradation to such inaccessible areas. Table 3.5-1 of the LRA did not address this issue. Provide information which discusses how this concern is addressed at McGuire and Catawba.

3.5-7 Table 3.5-1, Aging Management Review Results - Reactor Building of the LRA lists no aging effects and their corresponding AMPs for the following component types: (1) dome concrete, foundation mat and shell wall of concrete shield building; (2) Wear slab concrete of ice condenser components and (3) equipment pads, flood curbs, hatches, missile shields, reinforced concrete beams, columns, floor slabs, walls of reactor building interior structural components. Table 3.5-2, Aging Management Review Results - Other Structures of the LRA lists no aging effects and their corresponding AMPs for the following component types: equipment pads, floor curbs, foundation caissons, foundations, hatches, manholes and covers, missile shields, reinforced concrete beams, columns, floor slabs, walls, sumps and trenches under "concrete structural components" subheading. The staff does not agree with the results of your aging management reviews as provided in the aforementioned tables. The following discussion explains the staff's position.

Based on the observations of degradations in six nuclear power plants, reviews of construction deficiency reports and relevant licensing event reports, in NUREG-1522 (Chapter 5), the staff makes a generic observation, "For the types of materials (normal weight, medium-strength concrete and mild steel) used in the building structures of the nuclear power plants, it is evident that 'concrete cracks and steel corrodes'." On the basis of similar industry wide evidences, the American Concrete Institute (ACI) has published a number of documents (e.g., ACI 201.1R, "Guide for Making a Condition Survey of Concrete," ACI 224.1R, "Causes, Evaluation and Repairs of Cracks in Concrete Structures," ACI349.3R, "Evaluation of Existing Nuclear Safety-Related Concrete Structures") to manage the aging of concrete structures. These reports and standards confirm the inherent characteristics of the concrete structures in that they degrade with time, if not properly managed. Thus, the staff cannot accept any aging management review results that would indicate, "aging management of concrete structures is not required." It is widely known in the concrete industry that concrete components or materials are subject to aging effects. Please provide McGuire/Catawba plants specific AMP(s) for the above listed concrete elements for staff review.

3.5-8 Table 3.5-2 of the LRA assigns no AMP for portion of the non-sheltered, externally exposed missile shields (AB and NSW pump structure only), whereas, the same table designates the Inspection Program for Civil Engineering Structures and Components as the AMP for the RWST missile shield wall to manage an aging effect (change in material properties) due to leaching. Confirm, as appropriate, that past plant operating experience has shown that the auxiliary building and nuclear service water pump structure at McGuire and Catawba exhibit insignificant leaching potential or explain the differential treatment of the missile shields.

3.5-9 Table 3.5-3 of the LRA states that no AMP is needed for cable tray & conduit, control boards, electrical & Instrument panels & enclosures, and new fuel storage racks. Are these items all made of galvanized steel? If not, discuss the basis for not designating the Inspection Program for Civil Engineering Structures and Components as the AMP for items made of non-galvanized carbon steel.

4.6 Containment Liner Plate, Metal Containments, and Penetration Fatigue Analysis

4.6-1 Provide detailed justification why a fatigue time-limited aging analysis (TLAA) was not required for the steel containment vessel, as stated in Section 4.6.2, for loadings resulting from operating transients, peak containment internal pressure resulting from the design basis loss of coolant accident (LOCA), design basis safe shutdown earthquake (SSE), and leakage rate testing, in addition to the loading resulting from the transient expansions of the bellows.

4.6-2 Section 4.6.3.1 indicates that the vendors of the bellows performed cyclic life evaluations and stated that the life of the bellows is well beyond what the bellows would see during normal operation in 40 years of plant operation. Provide the root cause of bellows cracking as a result of fatigue failure within 20 years from the start of plant operation, well short of the bellows vendor test lives.

A conference call between the staff and the applicant was held on November 20, 2001. A summary of the conference call was issued January 10, 2002. During the conference call, the applicant indicated that the bellows have been characterized as leaking, not as cracked. The applicant further offered that the bellows that had been replaced at McGuire had cracked, and the root cause was attributed to trans-granular stress corrosion cracking from contact with chlorine. The applicant indicated that the other root causes of bellows leakage were attributed to either manufacturing process problems and defects or to improper installation. As such, these leaking bellows are being monitored within the sites' corrective action programs.

The staff requests the applicant to provide the range of possible root causes of leaking bellows so that the staff can complete its review of this issue.

4.6-3 Section 4.6.3.2, "Catawba Design and Time-Limited Aging Analysis Evaluation," states that the design Code of Record for Catawba bellows assemblies is ASME Section III NC-3649, 1974. This code requires an evaluation of the cumulative effect of stress cycles for cyclic life of bellows. During the conference call on November 20, 2001, the applicant indicated that the calculations and analyses for bellows were not considered relevant in making a safety determination and that an aging management program was proposed for this structural component. The leaks have been attributed to manufacturing process problems, installation problems, and the one case of trans-granular stress corrosion cracking due to contact with chlorine. A cyclic analysis was performed for the bellows in the original design. The order of magnitude of the number of cycles was too large to base any safety judgment on the specific number. Therefore, the analysis is not a TLAA. Because the function of the bellows is within license renewal scope and leaks have been observed at both McGuire and Catawba, a program was proposed to address leaking.

Because fatigue of bellows is addressed under Section 4.6 of the LRA, the staff infers that there may be a TLAA credited for the aging management of this component. The staff requests the applicant to explain, in a written response, that aging of bellows is addressed through an aging management program rather than a TLAA.

<u>4.7.3 Depletion of Nuclear Service Water Pond Volume due to Runoff</u>

4.7.3-1 It is stated in Section 4.7.3 of the LRA that your recent calculations have validated the adequacy of the volume of water in the standby nuclear service water pond (SNSWP). However, your application is silent about the remedial action you will take in case a future survey of the topography of the bottom of the Pond indicates a reduction in the volume of water due to the buildup of sediment. Clarify this aspect of your SNSWP Volume Program.

<u>B.3.2 Battery Rack Inspections</u>

B.3.2-1 In Section B.3.2 of the LRA, the applicant stated that the parameters to be inspected in the battery rack inspection program include the visual examination of the battery racks for physical damage or abnormal deterioration, including loss of material. This is appropriate for the inspections of the battery rack itself; however, degraded anchorage of the battery racks may also lead to loss of intended function for the battery rack. Consequently, the staff requested a description of how the inspections of the battery rack anchorages will ensure that deterioration of the anchorages does not lead to a loss of function for the battery racks.

The staff and applicant participated in a conference call on October 11, 2001. A summary of this conference call was issued November 23, 2001. During this conference call, the applicant indicated that a station procedure is used to inspect for loss of material of the battery racks and all attendant sub-components (including anchor bolts). The staff requests information from the procedure that will enable it to determine the acceptability of guidance provided therein for identifying and correcting aging effects associated with the battery rack anchorage bolts.

<u>B.3.7 Containment Inservice Inspection Plan - IWE</u>

B.3.7-1 Under element {Parameters monitored or Inspected}, you explicitly exclude monitoring or inspection of Category E-B, E-D, E-F, and E-G of Table 2500-1 of Subsection IWE from *Containment Inservice Inspection Plan - IWE*. Please provide a summary of the alternatives that you have instituted to ensure the aging management of the pressure-retaining containment components covered by these Categories.

B.3.7-2 Please summarize the suspect areas that you have identified as requiring augmented inspection (as per IWE-1240) during the current inspection interval of *Containment Inservice Inspection Plan - IWE,* for example, the steel surface areas behind the ice-baskets. Also, summarize the areas subjected to Category E-C examination and your plans to continue these examinations during the extended period of operation. Please provide this summary for each Unit of McGuire and Catawba plants.

B.3.8 Containment Leak Rate Testing Program

B.3.8-1 As described in the "Acceptance Criteria," if the leakage is detectable, the assembly must be tested with the containment side of the bellows assembly pressurized to Pa, and the acceptance criterion is based on the combined leakage rate for all reactor building bypass leakage paths to be less than or equal to 0.07 La. Please provide information regarding how this leakage rate acceptance criterion is related to the individual leakage rates through the bellows, which leak into the annulus between the primary containment and the reactor building.

B.3.8-2 Please provide the following pertinent information related to the operating experience described in the LRA:

1. For the McGuire and the Catawba plants, provide the number of bellows where leakages have been found, and the number of bellows that have been replaced, since the beginning of operation of these plants.

2. For the McGuire and the Catawba plants, provide the number of Duke Class A and Class B bellows that are currently leaking (cracked).

3. Table 3.5-1 "Aging Management Review Results," indicates that the function of the bellows and mechanical penetrations is to provide a pressure boundary and/or fission product barrier. Provide justification for operating with leaking (cracked) bellows during the period of current operation and the period of extended operation.

B.3.10 Crane Inspection Program

B.3.10-1 The acceptance criterion for the crane inspection program is no unacceptable visual indication of loss of material. Describe the criteria for (1) assessing the severity of the observed degradations and (2) determining whether corrective action is necessary.

B.3.12.1 Fire Barrier Inspection

B.3.12.1-1 Describe the inspection procedures that permit the timely detection of cracking/delamination and separation of the fire barrier penetration seals. The application states in the acceptance criteria that separation from wall and through-holes shall not exceed limits as specified in the procedure. Indicate what these limits are and the basis for their selection.

B.3.13 Flood Barrier Inspection

B.3.13-1 The acceptance criterion for the flood barrier inspection program is no unacceptable visual indication of cracking and change in material properties of elastomeric flood seals that would result in loss of intended function. Describe the criteria for (1) assessing the severity of the observed degradations and (2) determining whether corrective action is necessary.

B.3.21 Inspection Program for Civil Engineering Structures and Components

B.3.21-1 In the section Monitoring & Trending, the application states that inspectors are qualified by appropriate training and experience. Also in the section Acceptance Criteria, the application states that the severity of the observed degradation is evaluated by an accountable engineer. State the qualifications as well as the required training and experience for the inspectors and accountable engineer.

B.3.21-2 The acceptance criteria for the inspection program for civil engineering structures and components are no unacceptable visual indication of loss of material, cracking or change of material properties of concrete, and loss of material for steel, as identified by the accountable engineer. Describe the criteria for (1) assessing the severity of the observed degradations and (2) determining whether corrective action is necessary.

B.3.30 Standby Nuclear Service Water Pond Dam Inspection

B.3.30-1 Table 18-1 of the Catawba and McGuire UFSAR Supplements reference Improved Technical Specification (ITS) Surveillance Requirement (SR) 3.7.8.3 for the Standby Nuclear Service Water Pond Dam Inspection. The staff requests the applicant to indicate if Table 18-1 for Catawba is in error and, if so, please provide the correct ITS SR reference for Catawba.

B.3.30-2 Provide the qualifications of the accountable engineer (mentioned by you in Section B.3.30) who will (1) evaluate the performance of the SNSWP Dam (as reflected by the results of settlement monitoring and foundation pore pressure monitoring, etc.), and (2) recommend the needed repairs for the continued service of the Dam.

B.3.30-3 The acceptance criteria for the standby nuclear service water pond dam inspection program are no visual indications of abnormal degradation, vegetation growth, erosion, or excessive seepage that would affect the Standby Nuclear Service Water Pond Dam operability. Describe the criteria for (1) assessing the severity of the observed degradations and (2) determining whether corrective action is necessary.

B.3.33 Technical Specification SR 3.6.16.3 Visual Inspection

B.3.33-1 The only detection of age-related degradation under technical specification SR 3.6.16.3 is by visual inspection. Areas of inspection include the walls and dome of the concrete Reactor Building. Explain how the inspections are conducted to be effective in areas that are many feet above the floor (monitoring & trending). Are there cranes or catwalks that allow close visual access to key areas to be inspected? Are visual enhancements such as binoculars used to increase the effectiveness of the inspections?

B.3.33-2 The acceptance criteria for the Technical Specification SR 3.6.16.3 visual inspection program are based on visual indication of structural damage or degradation. For concrete, the acceptance criterion is no unacceptable indication of change in material property due to leaching. Describe the criteria for (1) assessing the severity of the observed degradations and (2) determining whether corrective action is necessary.

B.3.35 Underwater Inspection of Nuclear Service Water Structures

B.3.35-1 Provide the qualifications of the accountable engineer who will be responsible for determining the need for repairs of the NSW structures and components at both Catawba and McGuire.

B.3.35-2 The acceptance criteria for the underwater inspection of nuclear service water structures are no visual indications of (1) loss of material for steel components and (2) loss of material and cracking for concrete components, as determined by the accountable engineer. Describe the criteria for (1) assessing the severity of the observed degradations and (2) determining whether corrective action is necessary.