Duke Power®
A Duke Energy Company

M. S. Tuckman
Executive Vice President
Nuclear Generation

Duke Power
526 South Church St. EC07H
Charlotte, NC 28202
P. O. Box 1006 EC07H
Charlotte, NC 28201-1006
(704) 382-2200 OFFICE
(704) 382-4360 FAX

March 11, 2002

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Subject: Response to Requests for Additional Information in Support of the Staff Review of the Application to Renew the Facility Operating Licenses of McGuire Nuclear Station, Units 1 & 2 and Catawba Nuclear Station, Units 1 & 2

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

Dear Sir:

By letter dated June 13, 2001, Duke Energy Corporation (Duke) submitted an Application to Renew the Facility Operating Licenses of McGuire Nuclear Station and Catawba Nuclear Station (Application). The staff is reviewing the information provided in the Application and has identified areas where additional information is needed to complete its review.

In a letter dated January 28, 2002, the staff requested additional information concerning Sections 2.4, 3.5, 4.6, 4.7.3, and Appendix B of the Application. These sections contain information related to the structural elements of the license renewal review. Attachment 1 provides the Duke response to this letter. Some of these responses contain commitments. The commitments are restated in Attachment 2 to facilitate tracking and management.

If there are any questions, please contact Bob Gill at (704) 382-3339.

Very truly yours,

M. S. Tuckman

M. S. Tuckman

Attachments:

A085

Affidavit

M. S. Tuckman, being duly sworn, states that he is Executive Vice President, Nuclear Generation Department, Duke Energy Corporation; that he is authorized on the part of said Corporation to sign and file with the U. S. Nuclear Regulatory Commission the attached responses to staff requests for additional information relative to its review of the Application to Renew the Facility Operating Licenses of McGuire Nuclear Station and Catawba Nuclear Station, Docket Nos. 50-369, 50-370, 50-413 and 50-414 dated June 13, 2001, and that all the statements and matters set forth herein are true and correct to the best of his knowledge and belief. To the extent that these statements are not based on his personal knowledge, they are based on information provided by Duke employees and/or consultants. Such information has been reviewed in accordance with Duke Energy Corporation practice and is believed to be reliable.

M. S. Tuckman

M. S. Tuckman, Executive Vice President
Duke Energy Corporation

Subscribed and sworn to before me this 11TH day of MARCH 2002.

Mary P. Nelms

Notary Public

My Commission Expires:

JAN 22, 2006

xc: (w/ Attachment)

L. A. Reyes
Regional Administrator, Region II
U. S. Nuclear Regulatory Commission
Atlanta Federal Center
61 Forsyth Street, SW, Suite 23T85
Atlanta, GA 30303

D. B. Matthews
Director, Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Senior NRC Resident Inspector
McGuire Nuclear Station

Senior NRC Resident Inspector
Catawba Nuclear Station

C. P. Patel
Senior Project Manager
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

C. I. Grimes
Program Director, License Renewal and Environmental Impacts
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

R. L. Franovich
Senior Project Manager
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

R. E. Martin
Senior Project Manager
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

xc: (w/ Attachment)

Henry J. Porter
Assistant Director, Division of Waste Management
Bureau of Land & Waste Management
S.C. Department of Health and Environmental Control
2600 Bull St.
Columbia, SC 29201

R. M. Fry
Director, Division of Radiation Protection
North Carolina Department of Environment, Health, and Natural Resources
3825 Barrett Drive
Raleigh, NC 27609

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

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

Piedmont Municipal Power Agency
121 Village Drive
Greer, SC 29651

Saluda River Electric Cooperative, Inc.
P. O. Box 929
Laurens, SC 29360

Attachment 1
Application to Renew the Operating Licenses of
McGuire Nuclear Station and Catawba Nuclear Station
Responses to NRC Requests for Additional Information
NRC Letter dated January 28, 2002

2.4.1 Scoping and Screening Results: Reactor Buildings

RAI 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?

Response to RAI 2.4.1-1

The penetration drawings for Catawba are representative of all reactor building penetrations, including those at McGuire. The McGuire penetrations are shown in Figures 3-68, 3-69, and 3-70 of the McGuire UFSAR. The penetration through the Shield Building should have been identified under "Concrete Shield Building" in Table 3.5-1 of the Application. The penetration would include sub-components such as anchor rings, penetration sleeves, pipe, caps and restraint rings. As part of the penetrations, these items are within the scope of license renewal and subject to an aging management review. The following table entry is provided as a supplement to Table 3.5-1:

Component Type	Component Function (Note 1)	Material (Note 2)	Environment	Aging Effect	Aging Management Programs and Activities
Concrete Shield Building					
Penetration	1, 2, 7	Steel	Reactor Building	Loss of Material	Inspection Program for Civil Engineering Structures and Components

The *Inspection Program for Civil Engineering Structures and Components* is credited with managing loss of material of the penetrations through the shield building. The *Inspection Program for Civil Engineering Structures and Components* is addressed in Appendix B.3.21 of the Application.

RAI 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.

Response to RAI 2.4.1-2
McGuire
According to the cross-sections through the existing ground strata, the majority of the six foot thick foundation slab is located on sound rock. Concrete was used to fill and level those areas where the foundation was not located on sound rock. The loads on the foundation are applied as static loads for analysis purposes. In the analysis, the foundation is assumed to be of the Winkler type (which utilizes the concept that it can be analyzed as a beam on an elastic foundation). In order to meet this assumption, the foundation is anchored by dowels that are grouted into the rock.

Catawba
Reinforcement (dowels) are not used at Catawba since the foundation is founded on rock. Anchorage to the rock is not required to provide an adequate factor of safety against overturning [Reference Catawba UFSAR Section 3.8.5.4].

RAI 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.

Response to RAI 2.4.1-3
The removable concrete cover described in UFSAR Section 3.8.1.1.2 is the equipment hatch missile shield. Although it is located on the exterior of the shield building, the equipment hatch missile shield is addressed in Table 3.5-1, "Reactor Building Interior Structural Components," under missile shields. A separate entry is provided for the equipment hatch missile shield since it is exposed to an external environment. The equipment hatch missile shield is within the scope of license renewal and subject to an aging management review.

The tracks and other supporting structures are included with the structural steel beams, plates, etc. and anchorage in Table 3.5-1 of the Application. The tracks and other supporting structures are within the scope of license renewal and subject to an aging management review. Loss of material has been identified as an aging effect for these components and will be managed by the *Inspection Program for Civil Engineering Structures and Components.* The *Inspection Program for Civil Engineering Structures and Components* is discussed in Appendix B.3.21 of the Application.

RAI 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 includes penetrations. Clarify for the staff how the LRA handles the various penetrations to the Reactor Building.

Response to RAI 2.4.1-4

Penetrations through the Steel Containment are identified in Table 3.5-1 under "Steel Containment." The penetrations through the Shield Building should have been identified under "Concrete Shield Building" in Table 3.5-1 of the Application. The following table entry is provided as a supplement to Table 3.5-1:

Component Type	Component Function (Note 1)	Material (Note 2)	Environment	Aging Effect	Aging Management Programs and Activities
Concrete Shield Building					
Penetration	1, 2, 7	Steel	Reactor Building	Loss of Material	Inspection Program for Civil Engineering Structures and Components

The *Inspection Program for Civil Engineering Structures and Components* is credited with managing loss of material of the penetrations through the shield building. The *Inspection Program for Civil Engineering Structures and Components* is addressed in Appendix B.3.21 of the Application.

RAI 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 lists 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.

Response to RAI 2.4.1-5
The items identified in UFSAR Section 3.8.2.1 are subcomponents of larger components or are included with a larger group of components addressed in Table 3.5-1. Each of the items is discussed in more detail below.

Seals on personnel air locks are subcomponents of the personnel air locks in Table 3.5-1, "Steel Containment." Double, inflatable seals are provided on each door. The personnel air lock seals are tested in accordance with Technical Specification 3.6.2, *Containment Air Locks*. The seals are tested as part of the *Containment Leak Rate Testing Program* identified in Table 3.5-1 and discussed in Appendix B.3.8. The seals are replaced when warranted by their condition during visual inspection or by their performance during testing.

Penetration sleeves are included with the penetrations listed in Table 3.5-1, "Steel Containment." The penetration sleeves are part of the penetration and are within the scope of license renewal and subject to an aging management review. The penetrations are subject to loss of material as identified in Table 3.5-1. The *Containment Inservice Inspection – IWE* (Appendix B.3.7) and the *Containment Leak Rate Testing Program* (Appendix B.3.8) are credited with managing loss of material of the penetrations.

The purge penetration is included with mechanical penetrations in Table 3.5-1, "Steel Containment." The purge penetrations are subject to loss of material as identified in Table 3.5-1. The *Containment Inservice Inspection – IWE* (Appendix B.3.7) and the *Containment Leak Rate*

Testing Program (Appendix B.3.8) are credited with managing loss of material of the penetrations.

Double compressible seals are provided on the equipment hatch. The seals are subcomponents of the equipment hatch and are addressed with the equipment hatch in Table 3.5-1, "Steel Containment." The equipment hatch seals are tested as part of the *Containment Leak Rate Testing Program* identified in Table 3.5-1 and discussed in Appendix B.3.8. The seals are replaced when warranted by their condition during inspection or performance during testing.

The bolted flange referred to in the question is the blind flange in the fuel transfer penetration as discussed in UFSAR Section 3.8.2.1. The blind flange is included as part of the fuel transfer tube penetration in Table 3.5-1 in the Application. As part of the penetration, the blind flange is within the scope of license renewal and subject to an aging management review. The aging effect identified for penetrations is identified as loss of material in Table 3.5-1 of the Application. The aging management program is the *Containment Inservice Inspection Program – IWE.* Blind flanges within the Class MC boundary are inspected in accordance with ASME Section XI Subsection IWE Table IWE-2500-1 Category E-A (Item E1.10) and E-G (Item E8.10).

RAI 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.

Response to RAI 2.4.1-6
The current design of the CRDM missile shield hold-down mechanism at McGuire utilizes threaded rods. The threaded rods are subcomponents of the CRDM missile shield and therefore are not listed uniquely in Table 3.5-1. The CRDM missile shield is addressed in Table 3.5-1, "Aging Management Review Results – Reactor Building" under component type "Missile Shields". The CRDM missile shield is within the scope of license renewal and subject to an aging management review.

Background
The failures in the buttonheads and tendons on the CRDM missile shield at McGuire occurred in 1976. The failures were addressed during original licensing in a letter from Duke to the NRC dated July 1, 1976. The cause of the excessive corrosion of the button-headed wire tendon of the CRDM missile shield was due to the accumulation of water and corrosive material used during construction in the vertical embedded conduits of the tendon near the bottom bearing plates. This condition existed for nearly two years before the stressing operation was to start. The failure of the wire tendons was observed at the time of stressing. The new replacement design of the hold-down mechanism utilizes high strength round bars rather than prestressing tendons.

RAI 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.

Response to RAI 2.4.1-7
The structural supports for the various structures are within the scope of license renewal and subject to an aging management review. Structural supports are reviewed separately from the applicable structure following the commodity and component groupings approach as delineated in NEI 95-10. The structural supports for the various structures are addressed in Table 3.5-1 and Table 3.5-2 of the Application under component types "Anchorage", "Embedments", "Structural Steel Beams, Columns, Plates & Trusses" and "Reinforced Concrete Beams, Columns, Floor Slabs, Walls". Combination of these component groupings yields structural support items between the various structures.

2.4.2 Scoping and screening Results: Other Structures

In preparing the McGuire and Catawba Application, Duke chose to be consistent with the format provided in the draft "Standard Review Plan for the Review License Renewal Applications for Nuclear Power Plants" (August 2000). This draft version, as well as the version later issued in 2001 as NUREG-1800 indicated that "Structures" would be listed in one portion of Table 3.5-1. Individual tables for each structure were not envisioned by the staff in the latest version of NUREG-1800, Section 3.5. The Duke application is consistent with NUREG-1800 in that it contains Table 3.5-1, which lists structural components within the Containment; Table 3.5-2, which lists structural components for all other structures; and Table 3.5-3, which lists all component supports. Previous license renewal applications have included a table for each structure that included the components within the structure, which is inconsistent with the current staff guidance contained in NUREG-1800. Duke's use of the recommended formatting of license renewal application to be consistent with NUREG-1800 appears to have contributed to many of the staff RAIs associated with the review of Section 2.4.2 and Table 3.5-2 of the Duke application.

Note: The original version of RAI 2.4.2-1 contained in the NRC transmittal letter dated January 28, 2002 was revised by the NRC telecon summary dated March 6, 2002 to read as follows:

RAI 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. Therefore, the staff requests that the applicant provide general arrangement drawings for the other structures at Catawba and McGuire.

Response to RAI 2.4.2-1

On February 21, 2002, a telephone conference call was conducted between Duke and the staff to clarify this RAI. Duke had provided the staff with copies of the following drawings that were highlighted to identify the structures within the scope of license renewal:

CN-1003-10, Catawba Nuclear Station, Plot Plan, General Arrangement
MC-1003-1, McGuire Nuclear Station, Plot Plan, General Arrangement
Figure 1 from CNS-1139.00-00-0004, titled *Auxiliary Building Structures Plan of Component Structures*

Figure 1 from MCS-1154.00-00-0004, titled *Auxiliary Building Structures Plan of Component Structures*

The staff indicated that these drawings were sufficient and that no further information was needed at this time. The results of this telephone conference call are contained in a staff memorandum dated March 6, 2002.

The drawings listed above are classified as commercial information because they relate to the physical protection of McGuire Nuclear Station and Catawba Nuclear Station. Duke requests that these drawings be withheld from public disclosure in accordance with the provisions of 10 CFR 2.790(d)(1).

RAI 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?

Response to RAI 2.4.2-2

The Fuel Building, Control Building, Diesel Generator Buildings, Main Steam Doghouses, UHI Tank Building, and the Groundwater Drainage System are grouped with the Auxiliary Building to provide a more efficient aging management review. As stated in Section 2.4 of the Application, these structures are grouped with the Auxiliary Building because they are attached to or contained within the Auxiliary Building

Components of these structures are identified in Tables 3.5-2 and 3.5-3 of the Application. Components listed in Tables 3.5-2 and 3.5-3 are applicable to the Auxiliary Building and its associated structures such as the Fuel Buildings unless noted otherwise. For example, equipment pads identified in Table 3.5-2 are components of all structures, including the Fuel Buildings. In contrast, the foundation cassions are not components of the Fuel Buildings since they are applicable only to the McGuire Turbine Building. Additional detail is provided below for the specific components identified in the RAI, and a list of all of the components included in the Fuel Buildings is also provided.

The concrete enclosures (new fuel storage vaults) are addressed in Table 3.5-2, "Aging Management Review Results – Other Structures" under component type "Reinforced Concrete Beams, Columns, Floor Slabs, Walls."

The Fuel Building roofs are addressed in Table 3.5-2, "Aging Management Review Results – Other Structures" under component type "Roof Slabs."

The fuel handling bridge cranes are addressed in Table 3.5-3, "Aging Management Review Results – Component Supports" under component type "Crane Rails & Girders".

The fuel transfer up-ending canal (stainless steel lined), is included as an integral part of the spent fuel pool. The fuel transfer up-ending canals are addressed in Table 3.5-2, "Aging Management Review Results – Other Structures" under component type "Spent Fuel Pool Liner Plate". As such, the fuel transfer up-ending canal is within the scope of license renewal and subject to an aging management review.

The Fuel Buildings include components such as:

- Equipment Pads
- Fire Walls
- Flood Curbs
- Foundations
- Hatches
- Missile Shields
- Reinforced Concrete Beams, Columns, Floor Slabs, Walls
- Roof Slabs
- Anchorage
- Checkered Plate
- Crane Rails & Girders
- Embedments
- Equipment Component Supports
- Expansion Anchors
- Fire Doors
- Flood, Pressure & Specialty Doors
- Spent Fuel Pool Liner Plate
- Structural Steel Beams, Columns, Plates & Trusses
- Structural Steel and Plates
- Boraflex Panels (MNS only)
- Fire Barrier Penetration Seals
- Cable Tray & Conduit
- Cable Tray & Conduit Supports
- Electrical & Instrument Panels & Enclosures
- HVAC Duct Supports
- Instrument Line Supports
- Instrument Racks & Frames
- New Fuel Storage Racks

- Pipe Supports
- Spent Fuel Storage Racks
- Stair, Platform, and Grating Supports

The components are within the scope of license renewal and subject to an aging management review. The aging management review results are included in Tables 3.5-2 and 3.5-3.

RAI 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.

Response to RAI 2.4.2-3
The under-drain groundwater system consists of a grid of collecting trenches below the foundation surrounded on all sides by concrete, fill concrete, or rock. The under-drain groundwater system is identified in Tables 2.2-1 and 2.2-2 as the Groundwater Drainage System. The Groundwater Drainage System is described in Section 2.4.2.1 of the Application with the Auxiliary Buildings. The Groundwater Drainage System is included as an integral part of the foundation of the Auxiliary Building and Diesel Generator Buildings. The foundations are addressed in Table 3.5-2, "Aging Management Review Results – Other Structures" under concrete structural components. As such, the Groundwater Drainage System (under-drain groundwater system) is within the scope of license renewal and subject to an aging management review.

Waterstops are provided in the below-grade sections of the structures. Waterstops are discussed in Section 2.1.2.2 of the Application. Waterstops are not uniquely identified in the Application. Waterstops are subcomponents of the foundation or wall and are addressed with the foundation or wall within which the waterstops are located. Foundations and walls are within the scope of license renewal and subject to an aging management review. Foundations and walls are addressed in Table 3.5-2, pages 3.5-10 and 3.5-11, respectively.

RAI 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.

Response to RAI 2.4.2-4
The Main Steam Doghouses, Upperhead Injection (UHI) Tank Building, Fuel Building, Control Building, Diesel Generator Buildings, and the Groundwater Drainage System are grouped with the Auxiliary Building to provide a more efficient aging management review. As stated in Section 2.4 of the Application, these structures are grouped with the Auxiliary Building because they are attached to or contained within the Auxiliary Building

The Main Steam Doghouses and the Upperhead Injection (UHI) Tank Building are discussed in Section 3.8.4.1 of the CNS UFSAR. General arrangement drawings of the buildings are shown in UFSAR Figures 1-4 through 1-8. Each reactor unit has one Inside Doghouse, Outside Doghouse, and UHI Tank Building.

The Inside Doghouse and the Outside Doghouse are located on opposite sides of their respective Reactor Building. The Doghouses contain high pressure main steam and feedwater piping. The Inside Doghouse is cast integrally with the Auxiliary Building and is free-standing above a certain elevation. The Outside Doghouse is cast integrally with the UHI Tank Building.

The UHI tank and components are housed adjacent to the Outside Doghouse and separated by a reinforced concrete wall. The UHI tank was originally designed to store water to be used after a Design Basis Event to remove reactor core decay heat. Analysis by Westinghouse determined that the UHI System was not needed. This system has been functionally disabled. While this system has been disabled, other systems (for example, portions of the Hydrogen Bulk Storage) within the scope of license renewal are contained within the UHI Tank Building, therefore the UHI Tank Building is within the scope of license renewal.

Components of these structures are identified in Tables 3.5-2 and 3.5-3 of the Application. Components listed in Tables 3.5-2 and 3.5-3 are applicable to the Auxiliary Building and its associated structures such as the Main Steam Doghouses and UHI Tank Building unless noted otherwise. For example, equipment pads identified in Table 3.5-2 are components of all structures, including the Main Steam Doghouses and UHI Tank Buildings. In contrast, the

foundation cassions are not components of theses buildings since they are applicable only to the McGuire Turbine Building.

The Main Steam Doghouses and the UHI Tank Building contain structural components listed in Table 3.5-2 and Table 3.5-3 such as:

- Equipment Pads
- Fire Walls
- Foundations
- Hatches
- Reinforced Concrete Beams, Columns, Floor Slabs, Walls
- Roof Slabs
- Anchorage
- Checkered Plate
- Embedments
- Expansion Anchors
- Fire Doors
- Structural Steel Beams, Columns, Plates & Trusses
- Fire Barrier Penetration Seals
- Cable Tray & Conduit
- Cable Tray & Conduit Supports
- Electrical & Instrument Panels & Enclosures
- Equipment Component Supports
- HVAC Duct Supports
- Instrument Line Supports
- Instrument Racks & Frames
- Pipe Supports
- Stair, Platform, and Grating Supports

The components are within the scope of license renewal and subject to an aging management review. The aging management review results are included in Tables 3.5-2 and 3.5-3.

RAI 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.

Response to RAI 2.4.2-5
The Condenser Cooling Water Intake Structure provides structural support to the three main fire pumps which are relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with 10 CFR 50.48, the Fire Protection (FP) rule. Therefore, the Condenser Cooling Water Intake Structure is within the scope of license renewal per 10 CFR 54.4 (a)(3). The Fire Pump rooms, located on the outermost East and West sides of the Condenser Cooling Water Intake Structure, are the only parts of the structure within the scope of the LR Rule.

The fire pump rooms and its structural components are within the scope of license renewal and are subject to an aging management review. The fire pump rooms include structural component types such as:

- Foundation
- Foundation Dowels
- Equipment pads
- Reinforced concrete beams, columns, floors slabs and walls
- Roof
- Anchorage
- Cable tray & conduit
- Cable tray & conduit supports
- Electrical & instrument panels & enclosures
- Embedments
- Expansion Anchors
- Pipe supports

The aging management review results are presented in Table 3.5-2 and Table 3.5-3 of the Application.

RAI 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.

Response to RAI 2.4.2-6
The Low Pressure Service Water Intake Structure is a reinforced concrete structure. The Low Pressure Service Water Intake Structure provides structural support for the components of the Conventional Low Pressure Service Water System and the fire pumps. The Conventional Low Pressure Service Water System is not within the scope of license renewal (See Table 2.2-4 of the Application). The fire pumps are required to demonstrate compliance with the Commission's regulation for fire protection; therefore, they are within the scope of license renewal per 10 CFR 54.4(a)(3). The components of the Low Pressure Service Water System Intake Structure are included in Table 3.5-2 and Table 3.5-3 of the Application. The Low Pressure Service Water System Intake Structure includes components such as:

- Foundation
- Equipment pads
- Reinforced concrete beams, columns, floors slabs and walls
- Anchorage
- Cable tray & conduit
- Cable tray & conduit supports
- Electrical & instrument panels & enclosures
- Embedments
- Expansion Anchors
- Pipe supports

The aging management review results are presented in Table 3.5-2 and Table 3.5-3 of the Application.

RAI 2.4.2-7

Note: The original version of RAI 2.4.2.7 contained in the NRC transmittal letter dated January 28, 2002 was revised by the NRC telecon summary dated March 6, 2002 to read as follows:

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 clear that the structures described in the section are the structures within the boundary of the nuclear service water structures for license renewal. Please indicate which structures (including the nuclear service water structures) are within the scope of license renewal and subject to an AMR; provide applicable LRA references for this information. Also, please confirm that the components listed in Table 3.5-2 of the LRA are applicable to all structures (including nuclear service water structures) that are within the scope of license renewal unless otherwise noted in the Table.

Response to RAI 2.4.2-7

Table 2.2-2 of the Application identifies all of the Catawba structures that are within the scope of license renewal. Table 2.2-4 of the Application identifies all of the Catawba structures that are not within the scope of license renewal. The combination of the two tables contains all of the structures at Catawba. Structures associated with the Nuclear Service Water Structures are identified in Table 2.2-2 and include the following:

- Nuclear Service Water and Standby Nuclear Service Water Pump Structure
- Nuclear Service Water Conduit Manholes
- Nuclear Service Water Intake Structure
- Standby Nuclear Service Water Discharge Structure
- Standby Nuclear Service Water Intake Structure
- Standby Nuclear Service Water Pond Outlet

Structures identified in Table 2.2-2 are grouped together to provide a more efficient aging management review. The groupings are identified in Section 2.4 of the Application. The previously mentioned structures are grouped together and are identified as the Nuclear Service Water Structures. Descriptions of each of the structures associated with the Nuclear Service Water Structures are provided in Section 2.4.2.3 of the Application.

Components of these structures are identified in Tables 3.5-2 and 3.5-3 of the Application. Components listed in Tables 3.5-2 and 3.5-3 are applicable to the Nuclear Service Water Structures unless noted otherwise. For example, equipment pads identified in Table 3.5-2 are components of all structures, including the Nuclear Service Water Structures. In contrast, the foundation cassions are not components of the Nuclear Service Water Structures since they are applicable only to the McGuire Turbine Building. The Nuclear Service Water Structures contain structural components listed in Table 3.5-2 and Table 3.5-3 such as:

- Anchorage
- Embedments
- Equipment pads
- Fire walls
- Foundations
- Hatches
- Manholes and covers
- Missile shields
- Reinforced concrete beams, columns, floor slabs, walls
- Roof slabs
- Cable tray & conduit
- Cable tray & conduit supports
- Electrical & instrument panels & enclosures
- Equipment component supports
- Expansion anchors
- Fire doors
- Flood, pressure and specialty doors
- HVAC duct supports
- Instrument line supports
- Instrument racks & frames
- Pipe supports
- Stair, platform, and grating supports
- Structural steel beams, columns, plates & trusses
- Trash rack and screens
- Fire barrier penetrations seals
- Roofing

The components are within the scope of license renewal and subject to an aging management review. The aging management review results are included in Tables 3.5-2 and 3.5-3.

RAI 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.

Response to RAI 2.4.2-8
The earthen embankment of the Standby Nuclear Service Water Pond Dam is the component which performs the intended function providing ultimate heat sink following a LOCA or loss of Lake Norman or Lake Wylie. Other components, such as the drain pipes, observation wells, and piezometers are not relied upon for the Standby Nuclear Service Water Pond Dam to perform its intended function, but are used as part of the aging management program to verify that the dam is performing its function as designed. While these components are not included within the scope of license renewal and are not subject to an aging management review, they are included as an integral part of the Standby Nuclear Service Water Pond Dam Inspection in Appendix B.3.30. The drain pipes, observation wells, and piezometers are discussed in Appendix B.3.30 of the Application.

RAI 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.

Response to RAI 2.4.2-9
The Standby Shutdown Facility is a steel-frame and masonry structure consisting of a diesel generator room, electrical equipment room, battery room, control room, and shared equipment. The notation in Table 3.5-2 of the Application signifies block walls apply to Auxiliary Building, Standby Shutdown Facility, and Turbine Building only. The notation was not intended to convey that only block walls are subject to an aging management review within the Standby Shutdown Facility.

Components listed in Tables 3.5-2 and 3.5-3 are applicable to the Standby Shutdown Facility unless noted otherwise. For example, equipment pads identified in Table 3.5-2 are components of all structures, including the Standby Shutdown Facility. In contrast, the foundation cassions are not components of the Standby Shutdown Facility since they are applicable only to the McGuire Turbine Building. The Standby Shutdown Facility includes components such as:

- Anchorage
- Battery Racks
- Cable Tray & Conduit
- Cable Tray & Conduit Supports
- Control Boards
- Electrical & Instrument Panels & Enclosures
- Embedments
- Equipment Component Supports
- Equipment Pads
- Expansion Anchors
- Foundations
- Hatches
- HVAC Duct Supports
- Instrument Line Supports
- Instrument Racks & Frames
- Masonry block Walls
- Pipe Supports
- Reinforced Concrete Beams, Columns, Floor Slabs, Walls, and Roof Slabs
- Roofing

- Stair, Platform, and Grating Supports
- Structural Steel Beams, Columns, Plates & Trusses

These component types are represented in Table 3.5-2 and Table 3.5-3 of the Application. These components are within the scope of license renewal and subject to an aging management review.

RAI 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.

Response to RAI 2.4.2-10
The Turbine Buildings are steel frame structures supported on reinforced concrete substructures. The Service Building is a relatively light two story steel frame structure, located between the Turbine Buildings. The intended functions of the Turbine Building (including Service Building) is to provide structural support and/or shelter to components relied on during certain postulated fire, anticipated transients without scram, and/or station blackout events.

The Turbine Building (including Service Building) includes components such as:

- Anchorage
- Cable tray & conduit
- Cable tray & conduit supports
- Checkered plates
- Electrical & instrument panels & enclosures
- Embedments
- Equipment component supports
- Equipment pads
- Expansion anchors
- Flood, pressure, & specialty doors
- Flood curbs
- Foundations
- Foundation cassions (MNS Turbine Building only)
- Hatches
- Instrument line supports
- Instrument racks & frames
- Masonry block walls
- Pipe supports
- Reinforced concrete beams, columns, floor slabs, and walls
- Roofing
- Stair, platform, and grating supports

- Structural steel beams, columns, plates & trusses

These component types are represented in Table 3.5-2 and Table 3.5-3 of the Application. These components are within the scope of license renewal and subject to an aging management review.

RAI 2.4.2-11

Note: The original version of RAI 2.4.2-11 contained in the NRC transmittal letter dated January 28, 2002 was revised by the NRC telecon summary dated March 6, 2002 to read as follows:

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. Please describe, for each plant, the yard structures, trenches and drainage structures that are within the scope of license renewal; provide applicable LRA references for this information.

Response to RAI 2.4.2-11

Structures associated with the Yard Structures are identified in Table 2.2-1 for McGuire and Table 2.2-2 for Catawba. The Yard Structures for McGuire include the following:

- Reactor Makeup Water Storage Tank Foundation
- Refueling Water Storage Tank Foundation
- Refueling Water Storage Tank Missile Wall
- Refueling Water Storage Tank Pipe Trench
- Trenches

The Yard Structures for Catawba include the following:

- Low Pressure Service Water Intake Structure
- Refueling Water Storage Tank Foundation
- Refueling Water Storage Tank Missile Shield
- Refueling Water Storage Tank Pipe Trench
- Trenches

Structures identified in Tables 2.2-1 and 2.2-2 are grouped together to provide a more efficient aging management review. The groupings are identified in Section 2.4 of the Application. The structures are grouped together as Yard Structures based on their location and their exposure to the external environment. Descriptions of each of the structures associated with the Yard Structures are provided in Section 2.4.2.8 of the Application.

Components of these structures are identified in Tables 3.5-2 and 3.5-3 of the Application. Components listed in Tables 3.5-2 and 3.5-3 are applicable to the Yard Structures unless noted otherwise. For example, equipment pads identified in Table 3.5-2 are components of all structures, including the Yard Structures. In contrast, the foundation cassions are not components of the Yard Structures since they are applicable only to the McGuire Turbine Building. Additional detail is provided below for the components included in the Yard Structures.

The foundations for the Reactor Makeup Water Storage Tank and the Refueling Water Storage Tank are described in Section 2.4.2.8 and included in Table 3.5-2 under the component type "Foundations." The Refueling Water Storage Tank Missile Wall is described in Section 2.4.2.8 and included in Table 3.5-2 under the component type "missile shield (RWST Missile Shield Wall)."

Trenches within the scope of license renewal are described in Section 2.4.2.8. Trenches are included in Table 3.5-2 under the component type "Trenches (Yard only)."

The Low Pressure Service Water Intake Structure at Catawba is described in Section 2.4.2.8. Components of the Low Pressure Service Water Intake Structure include the foundation, concrete walls and floor slabs, and anchorage included in Table 3.5-2. (Also see response to RAI 2.4.2-6.) The Low Pressure Service Water Intake Structure supports the fire pumps. The corresponding structure at McGuire is the Condenser Cooling Water Intake Structure. The Condenser Cooling Water Intake Structure is described in Section 2.4.2.2 of the Application. (Also see response to RAI 2.4.2-5 for the Condenser Cooling Water Intake Structure.)

The Yard Drainage System at Catawba is described in Section 2.4.2.8 of the Application. The components of the Yard Drainage System are included in Table 3.5-2 under the component type "Yard Drainage System (CNS only)."

Additional components are also identified in Tables 3.5-2 and 3.5-3 for the Yard Structures. These components are identified by noting that these components are exposed to the "External (Yard only)" environment. For example, the components within Table 3.5-3 which are identified for the Yard Structures are:

- Cable tray & conduit
- Cable tray & conduit supports
- Electrical & instrument panels & enclosures
- Equipment component supports
- Pipe supports
- Stair, platform, and grating Supports

The components identified within Tables 3.5-2 and 3.5-3 are within the scope of license renewal and subject to an aging management review.

RAI 2.4.2-12

Note: The original version of RAI 2.4.2-12 contained in the NRC transmittal letter dated January 28, 2002 was revised by the NRC telecon summary dated March 6, 2002 to read as follows:

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 reflect the evaluation boundaries for NSSS supports, and there is insufficient information in the UFSAR to support the staff's review. Since each of the NSSS support assemblies are designed entirely differently, the staff is unable to verify the components for which an AMR is required. Please describe the structures of the NSSS support assemblies as well as the license renewal evaluation boundaries for these assemblies. Provide applicable LRA or UFSAR references for this information.

Response to RAI 2.4.2-12

The NSSS component supports for the pressurizer, reactor vessel, steam generator, and reactor coolant pump are described in UFSAR Section 5.4.14 for Catawba and Section 5.5.14 for McGuire. The supports are identified in Table 3.5-3 under component type "Class 1 (NSSS) Supports." NSSS component supports are within the scope of license renewal and subject to an aging management review.

The boundary of the NSSS component support extends from the attachment to the component through the attachment to the supporting structure. Lugs that are integrally attached to the component are included with the component, not the component support. Component support anchorage to the building structure is included with the component support. Reinforced concrete floors and walls to which the component supports are anchored are addressed in Table 3.5-1 under the Reactor Building Interior Structural Components. The NSSS component supports are described in more detail below.

<u>Reactor Coolant System Pipe Supports</u>

Pipe supports for the reactor coolant system are generally constructed of a standard support, a structural frame, or some combination of the two. A standard support is an assembly consisting of one or more units usually referred to as a catalogue item and generally mass-produced. Pipe support frames generally are constructed of A36 structural steel or A500 Grade B tube shapes. The pipe supports are within the scope of license renewal and subject to an aging management review. The pipe supports are addressed in Table 3.5-3 of the Application.

Pressurizer Supports
The pressurizer supports consist of an upper lateral support ring and a lower lateral support frame. The pressurizer supports are shown in the McGuire UFSAR in Figures 5-35 through 5-37 and in the Catawba UFSAR in Figures 5-28 through 5-30. The upper lateral support ring encircles the pressurizer and consists of large flanged sections constructed from structural plates. This frame is attached to embedments anchored to the crane wall and the pressurizer enclosure wall.

The pressurizer lower lateral support frame consists of a frame attached to vertical hangers. The lateral frame and the vertical hangers are made from large rolled sections. The pressurizer lower lateral support frame attaches to embedments that are anchored to the crane wall and the bottom of the operating floor structural slab. The support skirt of the pressurizer is attached to a circular steel frame that is connected to the lateral support frame.

The pressurizer supports are within the scope of license renewal and subject to an aging management review. The pressurizer supports are addressed in Table 3.5-3 of the Application under "Class 1 (NSSS) Supports."

Reactor Vessel Supports
The reactor vessel supports are individual rectangular steel box structures. They are located beneath two opposing cold leg nozzles and two opposing hot leg nozzles. The reactor vessel support is shown in McGuire UFSAR Figure 5-38 and Catawba UFSAR Figure 5-31. These supports are constructed from plate sections and they are anchored to the primary shield wall (lower reactor cavity wall). The reactor vessel supports are within the scope of license renewal and subject to an aging management review. The reactor vessel supports are addressed in Table 3.5-3 of the Application under "Class 1 (NSSS) Supports."

Control Rod Drive Mechanism Seismic Support
The control rod drive mechanism seismic support is anchored in place by seismic supports consisting of turnbuckles, tie rods and other components. These tie rod arrangements provide radial and rotational restraint. The seismic support platform employs numerous spacer plates, most of which fit around individual CRDM shafts. The control rod drive mechanism seismic supports are within the scope of license renewal and subject to an aging management review. The control rod drive mechanism seismic supports are addressed in Table 3.5-3 of the Application under "Class 1 (NSSS) Supports."

Steam Generator Supports
Each steam generator is supported by four vertical pinned-end columns each attached to steam generator support lugs, a lower lateral support including compression bumpers, and an upper

lateral restraint consisting of a ring band with compression snubbers. The steam generator supports are shown in McGuire UFSAR Figures 5-30 through 5-32 and Catawba UFSAR Figures 5-23 through 5-25 and 5-32.

The steam generator support columns provide vertical support for the steam generators. These columns are constructed from large rolled sections with clevises attached to each end. The support columns are attached with simple supports to embedments, which project into the foundation mat through both the base slab and the steel containment vessel liner plate.

The steam generator lower lateral support encircles the steam generator and consists of large flanged sections constructed from structural plates. This frame is attached to embedments anchored to either the crane wall or the reactor cavity wall.

The steam generator upper lateral restraint consists of a restraint ring, two snubbers, and two "A-frame" structures. The snubbers are anchored to the steam generator enclosure wall. Two "A-frames", that are attached to embedments located in either the crane wall or the steam generator enclosure wall, limit movement of the restraint ring. Both the restraint ring and the "A-frame" are constructed of structural plates. Although the snubbers are excluded from an aging management review because they are active components, the brackets that attach the snubbers to the ring and to the building are subject to an aging management review.

The steam generator supports are within the scope of license renewal and subject to an aging management review. The steam generator supports are addressed in Table 3.5-3 of the Application under "Class 1 (NSSS) Supports."

<u>Reactor Coolant Pump Supports</u>
The reactor coolant pump supports consist of vertical steel columns and a lateral steel frame. The reactor coolant pump supports are shown in McGuire UFSAR Figures 5-32 through Figure 5-34 and Catawba UFSAR Figures 5-26 through 5-27.

The reactor coolant pump support columns provide vertical support for the reactor coolant pump. These columns are constructed from large rolled sections with clevises attached to each end. The support columns are attached to embedments, which project through both the base slab and the steel containment vessel liner plate and into the foundation mat.

The reactor coolant pump lateral support frame is a rigid frame that consists of flanged sections constructed from structural plates. This frame is attached to embedments anchored to the crane wall. The reactor coolant pump supports are within the scope of license renewal and subject to

an aging management review. The reactor coolant pump supports are addressed in Table 3.5-3 of the Application under "Class 1 (NSSS) Supports."

3.5 Aging management of Containments, Structures, and Component Supports

RAI 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.

Response to RAI 3.5-1

The environmental parameters of the below grade environment are discussed in Section 3.5.1 of the Application. Minimum degradation threshold limits for concrete have been established at 500 ppm chloride, 1,500 ppm sulfates, pH < 5.5 [Reference NUREG-1611]. The Catawba and McGuire groundwater parameters are below the limits where potential degradation of the concrete may occur. The environmental data for Catawba and McGuire is based on historical data during construction and data from more recent tests. The data spans more than 20 years. More than 20 years of environmental monitoring is sufficient to identify any trends toward aggressive environments; therefore, future tests of groundwater chemistry are not required. The SOC for the original license renewal rule supports the use of more than 20 years of operational data as sufficient. The NRC believes that the history of operation over the minimum 20-year period provides a licensee with substantial amounts of information and would disclose any plant-specific concerns with regard to age-related degradation.

For information, the wear slab is located in the Ice Condenser (Table 3.5-1) and is not exposed to a below grade environment.

RAI 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.

Response to RAI 3.5-2

Technical Specification SR 3.6.16.3 Visual Inspection is credited with managing change in material properties due to leaching of both the shell wall and dome of the shield building in Table 3.5-1 of the Application. The *Technical Specification SR 3.6.16.3 Visual Inspection* is discussed in Appendix B.3.33 of the Application. Previous inspections have identified change in material properties due to leaching on the shield building dome and near the dome-to-shell interface at McGuire. Subsequent inspection did not indicate any growth of the leaching or rebar corrosion. Rebar corrosion would be evidenced by rust stains, pop-outs or spalling.

The maintenance on the exterior of the shield building dome was completed in the fall of 2001. The domes were recoated with elastomeric urethane 18" up the parapet wall and 18" up the dome. The remainder of the dome was sealed with a clear concrete sealer. Subsequent inspections will determine whether the corrective actions are adequate and whether any additional maintenance is required.

RAI 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.

Response to RAI 3.5-3

Portions of McGuire and Catawba structures and components within the scope of license renewal are located in areas that are inaccessible for inspection. The key to understanding the management of inaccessible areas is to first understand the thoroughness of the aging management review process. The aging management review process methodically:

- identifies environments for the structures and components subject to aging management review,
- evaluates the material-environment combination for the structures and components to determine aging effects requiring management, and
- identifies the program that will manage the aging effects.

The purpose of the aging management review is to "demonstrate that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation." [Reference §54.21(a)(3)] The review includes several steps: (1) identifying the aging effects requiring management for the structure or component; (2) identifying existing or new programs for managing the aging effects; and (3) demonstrating that the program is effective in managing the aging effects so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation. The location of a structure or a component is a factor in identifying the aging effect(s) and in determining the programmatic oversight.

The aging effects for a structure or component occur due to a combination of several physical parameters. The materials of construction, the environment to which the structure or component is exposed and the stress or load experienced by the structure or component all play a role in the

determination of aging effects requiring management. Structures and components that are inaccessible for inspection may be exposed to unique environments because of their location. The condition of the inaccessible structure or component can be established by identifying the environment and determining if subjecting the structure or component to the environment results in aging effects that could degrade the condition of the component. The inaccessible environment is indirectly evaluated as part of the aging management review to determine if structures and components that are located in this environment will have aging effects requiring management.

The aging management review did not ignore any environmental conditions to which the structures and components are exposed, including those conditions in areas that may turn out to be inaccessible for inspection. For example, structures and components located below grade may be exposed to groundwater. The groundwater environment could potentially lead to aging effects different from those in an air environment. The groundwater chemistry plays a major role in the determination of the degradation of below grade structures and components. When determining the aging effects requiring management for below grade structures and components, the groundwater chemistry was evaluated to determine if parameters exceeded documented limits where degradation would occur. Therefore, the unique environment (groundwater) of the inaccessible structure or component was considered as part of the aging management review. In many instances, the proper selection of materials for the inaccessible environment results in few, if any, applicable aging effects.

The aging management review may determine that the environmental differences based on the location (such as concrete above grade exposed to air versus concrete below grade exposed to groundwater) do not result in unique aging effects for the inaccessible structure or component. Where the conditions in the inaccessible and accessible areas result in the same aging effects, the aging management program of the inaccessible area may be based on symptomatic evidence in an accessible area. For example, the aging effect due to alkali-aggregate reactions of concrete would manifest in both accessible and inaccessible areas.

For the case where symptomatic evidence in accessible areas provides guidance for aging effects in inaccessible areas, the aging management review assures that the aging effects due to the environment in the accessible region and the aging effects due to the environment in the inaccessible region are simultaneously evaluated. This philosophy forms the successful basis for the ASME Section XI oversight used to programmatically deal with inaccessible locations that require assurance of integrity. Programs such as the *Containment Inservice Inspection Plan - IWE* (Appendix B.3.7), the *Inservice Inspection Plan* (Appendix B.3.20) and the *Inspection Program for Civil Engineering Structures and Components* (Appendix B.3.21) use this

philosophy. The *Containment Inservice Inspection Plan – IWE* is credited with managing the inaccessible portions of the steel containment vessel.

When different aging effects due to environmental differences such as a more aggressive environment in the inaccessible area exist, other aging management approaches are needed. The guiding principle for the aging management review of these inaccessible areas is the knowledge of the environmental differences, such as more aggressive external environments due to the chemical composition of groundwater. The controlling or more aggressive location provides a "signal location" where the aging effects are more likely to be accelerated. These differences limit the intrusive nature of programmatic examination to the controlling or more aggressive location, i.e. "signal location." For those signal locations, a number of options exist in order to define program attributes that will assure that the structure or component intended function is maintained. These options may include, but are not limited to, remote examination techniques, disassembly, or excavation. The aging management review did not identify any inaccessible environments that result in aging effects different from those in the accessible environments. No unique aging management programs were required for any inaccessible areas.

For uncertainties associated with inaccessible areas to be an issue, the aging management review process would have to disregard unique conditions to which the structures and components may be exposed. The process would have to lack the rigor of systematically evaluating the varying combinations of materials, environments, and stressors. The aging management review process did not disregard any environmental conditions to which the structures and components are exposed. Since the aging management review process considered the full range of environments to which each of the structures and components are exposed, then all aging effects requiring management, including those caused by environmental conditions that may be associated with an inaccessible area, were identified. By methodically following the aging management review process, any aging effects that may occur due to the environment of the inaccessible area have been evaluated.

As for moisture barriers, moisture barriers are provided at the intersection between the floor slab and the steel containment vessel. The moisture barriers are inspected as part of the *Containment Inservice Inspection Plan – IWE.* The *Containment Inservice Inspection Plan – IWE* is discussed in Appendix B.3.7 of the Application. The moisture barriers are examined under Category E-D, Item E5.30 of Table-2500-1 of ASME Section XI Subsection IWE. The disposition of any damaged seals between the containment floor and the steel containment vessel will be in accordance with the requirements of Subsection IWE. As required by IWE-3513.1, "defective items shall be repaired or replaced."

RAI 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.

Response to RAI 3.5-4
1. An aging management review for stainless steel components in the reactor building environment was done for license renewal. The review did not identify any aging effects requiring management. A review of industry and plant operating experience was conducted to validate the aging management review conclusion. No operating experience was identified for the fuel transfer canal liner plate, sump liner, and sump screen that would invalidate the conclusion. Therefore, for those stainless steel components such as the fuel transfer canal liner plate, sump liner, and sump screens, no aging effect was identified and no aging management program was required. Operating experience for the bellows, however, has revealed cracking due to SCC from chloride concentration and leaking. The operating experience associated with the SCC of the bellows is described in more detail in response to RAI 3.5-5.

2. Metal housing systems, such as control boards, electrical & instrument panels, enclosures, etc., constructed of factory baked painted steel or galvanized sheet metal do not have a tendency to age with time [Reference "An Aging Assessment of Relay and Circuit Breakers and System Interactions," prepared by Franklin Research Center for Brookhaven National Laboratory, NUREG/CR-4715, June 1987]. Industry operating experience with metal housing systems indicates that they have performed without failure to the present [References "Aging Management Guideline for Commercial Nuclear Power Plants – Motor Control

Centers," SAND 93-7069, Sandia National Laboratories, February 1994 and "Aging Management Guideline for Commercial Nuclear Power Plants – Electrical Switchgear," SAND 93-7027, Sandia National Laboratories, July 1993.]. Therefore, loss of material is not an aging effect requiring management for electrical panels, enclosures, and control boards in sheltered, Reactor Building, and external environments.

Cable tray is constructed of painted or galvanized sheet metal similar to metal housing and located in the same environment; therefore, cable tray would age similarly to the metal housings. A review of industry operating experience was also reviewed to validate this conclusion. Deficiencies that were identified were event driven or design/installation deficiencies. Therefore, loss of material is not an aging effect requiring management for cable trays in sheltered, Reactor Building, and external environments.

The Control Room has a dropped acoustical ceiling which has been seismically qualified. The Control Room is a controlled mild environment that inhibits aging effects. Based on years of operating experience, no aging effects requiring management for the Control Room ceiling have been identified at McGuire and Catawba. This includes sub-components such as the acoustical tiles, light enclosures, ceiling grid, and grid support system. Therefore, no aging management program is required.

The New Fuel Storage Racks provide dry storage for new nuclear fuel. These racks are free-standing and are designed to accommodate fuel assemblies. The storage racks are fabricated from painted carbon steel and are located in a mild dry sheltered environment. A review of operating experience did not identify any aging effects requiring management. Therefore, loss of material is not an aging effect requiring management for the new fuel storage racks.

RAI 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.

Response to RAI 3.5-5
1. As stated in Appendix B.3.7, the Containment Inservice Inspection Plan – IWE is credited with managing loss of material of the pressure retaining steel components and their integral attachments for the period of extended operation. The Containment Leak Rate Testing Program is credited with managing loss of material and cracking of the pressure boundary components. The Containment Leak Rate Testing Program supplements the Containment Inservice Inspection Plan –IWE for most pressure retaining steel components of containment such as the steel containment vessel and penetrations. However, only the Containment Leak Rate Testing Program is identified to manage cracking of the bellows in Table 3.5-1. The provisions of ASME Section XI, Subsection IWE are not credited with managing cracking of the bellows.
2. Plant-specific operating experience at McGuire and Catawba validates that the cracking in bellows can be detected by the Containment Leak Rate Testing Program. Cracking of the bellows would manifest as leakage. Leakage has been identified in bellows at both McGuire and Catawba during the Containment Leak Rate Test Program. As discussed in Appendix

B.3.8, one bellows has been replaced at McGuire Unit 1 due to cracking. A metallurgical analysis was performed on the removed bellows to determine the root cause of the leakage. The root cause was attributed to trans-granular stress corrosion cracking from contact with chlorine. The source of chlorine was not determined. The chlorine could have been introduced through a surface brightener. As evidenced by the plant operating experience, leakage resulting from cracking can be detected by the Containment Leak Rate Testing Program.

3. The pressure requirements for the low pressure leak test are contained in Catawba Technical Specification Surveillance Requirement 3.6.1.1 and McGuire Technical Specification Surveillance Requirement 3.6.1.2. The Technical Specifications provide the minimum requirements. The space between each dual-play bellows assembly is subjected to a low pressure test at 3 to 5 psig. This test has been demonstrated to be meaningful since operating experience has shown that it detects leaks including leaks due to trans-granular stress corrosion cracking.

RAI 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.

Response to RAI 3.5-6
Several areas of the reinforced concrete beams, columns, floor slabs, and walls are inaccessible because of the layout of the Ice Condenser system. Areas which are inaccessible are (Reference Figure 6-113 in McGuire UFSAR and Figure 6-138 in Catawba UFSAR):

- Wear slab that is located beneath a protective layer of ice
- Structural concrete floor located beneath wear slab
- Surface of the crane wall that is located behind the insulated wall panels

These concrete components are designed in accordance with ACI 318 and constructed in accordance with ACI 301 using ingredients conforming to ACI and ASTM standards which provide a good quality, dense, low permeability concrete that provides resistance to aggressive chemical attack and corrosion of rebar.

The concrete located in the ice condenser is exposed to a unique environment. The normal atmosphere in the ice condenser is low temperature (10°F to 20°F) and very low humidity (Reference McGuire UFSAR Section 6.2.2.18.2 and Catawba UFSAR Section 6.7.18.2). Under these conditions, the concrete components would not be subject to aging effects requiring management. During maintenance at either McGuire or Catawba, ice condenser wall panel defrosting is not a normal maintenance practice. However, panel defrosting could occur and the wear slab concrete would be exposed to the resulting water as the water flowed to the floor drains. In addition to a protective coating, a protective layer of ice is maintained on the floor to protect the wear slab from the water. Since the wear slab is constructed of dense, low permeability concrete and it is protected by a coating and a layer of ice, no aging effects requiring management were identified for the wear slab.

The structural concrete floor is located below the wear slab (Reference McGuire UFSAR Figure 6-114 and Catawba UFSAR Figure 6-139). A layer of foam concrete is located between

the wear slab and the structural concrete floor to provide a layer of insulation. A vapor barrier is provided between the foam concrete and the structural concrete floor. The structural concrete floor is accessible from below. Since the structural concrete floor is constructed of dense, low permeability concrete and it is protected from above by the wear slab, foam concrete, and vapor barrier, no aging effects requiring management were identified for the structural concrete floor.

The interior surface of the crane wall is open to the Reactor Building environment and is accessible for inspection. The exterior surface of the crane wall is covered by wall panels in the Ice Condenser. Cooling ducts are incorporated in the wall panels. The cooling ducts provide flow from the air handlers in the duct adjacent to the ice bed and return flow in the outer duct of the panel. While the wall panels and the cooling ducts make the exterior surface of the crane wall inaccessible for inspection, they also protect the crane wall from potential defrosting water. Again, defrosting water is not a normal occurrence. Ice condenser wall panel defrosting is not a normal maintenance practice at either McGuire or Catawba. Since the crane wall is constructed of dense, low permeability concrete and it is protected by the panels and cooling ducts, no aging effects requiring management were identified for the crane wall.

RAI 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.

Response to RAI 3.5-7
Duke Power disagrees with the NRC staff position. The standards and results of NUREG-1522 inspections do not draw one to conclude that aging is an inherent characteristic of concrete, if not properly managed. Most of the industry-wide experience associated with the degradation of concrete in the standards is the result of exposure to severe environments such as marine or chloride exposure. Most, if not all, of the pictures in ACI 201.1R, "Guide for Making a Condition Survey of Concrete," depict degradation of bridges exposed to salt attack. In these environments, condition monitoring activities are appropriate.

In contrast, the NRC staff fails to reference standards or reports that support the inherent durability of concrete. ACI 201.2R, "Guide to Durable Concrete." states that "durable concrete will retain its original form, quality, and serviceability when exposed to its environment." It goes on to state that "concrete will perform satisfactorily when exposed to various atmospheric conditions, to most waters and soils containing aggressive chemicals, and to many other kinds of chemical exposure."

In addition, NUREG/CR-6424, *Report on Aging of Nuclear Power Plant Reinforced Concrete Structures*, reports that most instances related to degradation of concrete structures in the United States occurred early in the life of the structures and have been corrected. Causes were primarily related either to improper material selection, construction/design deficiencies, or environmental effects. Examples of some of the problems attributed to these deficiencies include concrete cracking, concrete voids or honeycombing, and concrete compressive strength values that were low relative to design values at a specific concrete age. In almost all cases, the concrete cracks were considered to be structurally insignificant or easily repaired using techniques such as epoxy injection. The voids and honeycombed areas and low-strength concrete areas were repaired or replaced. Quality control/quality assurance programs at nuclear power plants generally have been very effective in ensuring that the basic factors related to the production of durable concrete are adequately addressed.

NUREG/CR-4652, *Concrete Component Aging and Its Significance Relative to Life Extension of Nuclear Power Plants*, contains additional information to support the durability of concrete structures. NUREG/CR-4652 contains a summary of the degradation associated with nuclear power plant structures. Although the vast majority of the problems detected did not present a threat to public safety or jeopardize the structural integrity of the particular component, five incidences were identified that if not discovered and repaired could potentially had have serious consequences. These incidences were all related to the concrete containment and involved two dome delaminations, voids under tendon bearing plates, anchor head failures, and a breakdown in quality control and construction management. These few incidences where the structural integrity of the component was jeopardized were attributed to design, construction, or human errors, but not to aging [Reference NURG/CR-4652]. These findings are also reported in SECY 96-080 as the basis for the revision to 10 CFR 50.55a to incorporate inspections in accordance with ASME Subsection IWL.

NUREG/CR-4652 concludes that the results of the study are considered to be sufficiently representative that some general observations can be made on concrete aging and component performance. When concrete is fabricated with close attention to the factors required for durable concrete, the concrete will have infinite durability unless subjected to extreme external influences (overload, elevated temperatures, industrial liquids, etc.) Under normal

environmental conditions aging of concrete does not have a detrimental effect on its strength for concrete ages to at least 50 years. [Note: 50 years is the limit on age for which well-documented data has been identified. The number of concrete structures in existence having ages of 40 to 70 years, with a few in service for thousands of years, indicates that this value is conservative. Also, many structures continue to meet their function and performance requirements even when conditions are far from ideal.] The overall performance of concrete components in nuclear applications has been very good. With the exception of the anchor head failures at Farley 2, errors detected during the construction phase or early in the structure's life were of no structural significance or "easily" repaired and were non-aging related.

Many of the previously discussed documents were completed prior to 1990. More recent concrete inspection findings are documented in NUREG-1522, "Assessment of Inservice Conditions of Safety-Related Nuclear Plant Structures," and NUREG/CR-6679, "Assessment of Age-Related Degradation of Structures and Passive Components for U. S. Nuclear Power Plants." These documents identify concrete cracking in various structures at several nuclear plant sites. The documents do not discuss the severity or impact of the cracking on the functional capabilities of the component. All cracks do not necessarily result in loss of the intended function. For example, ACI 349.3R provides guidance on the size of cracks which would be judged to be acceptable. Furthermore, the pictures in NUREG-1522 do not depict cracking that would result in loss of intended function of the concrete component or structure. The findings do support the need for concrete inspections in certain structures which are exposed to environments that may result in aging such as salt water, brackish water, etc. Duke agrees with this position as evidenced by the information in the Application. For example, loss of material and cracking are identified as aging effects in Table 3.5-2 for reinforced concrete beams, columns, and walls that are exposed to a raw water environment. The findings do not support the need for inspections of all concrete structures in all environments.

The aging management review for the identified concrete components was conducted in accordance with the guidance provided in NEI 95-10, which was endorsed by the NRC, and incorporates findings from NUREG-1557, NUREG-1522, NUREG/CR-6424, NUREG/CR-4652, and ACI standards. Based on the material/environment combinations, it was determined that no aging effects would occur for these components that would result in loss of the intended function for the period of extended operation. Therefore, no aging management programs are required.

RAI 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.

Response to RAI 3.5-8

Resistance to leaching can be enhanced by using concrete with low permeability. A dense concrete with a suitable cement content that has been well cured is less susceptible to calcium hydroxide loss (leaching) from percolating water because of its low permeability and low absorption rate. The Catawba and McGuire concrete structures and components are designed in accordance with ACI 318-63 and ACI 318-71, respectively, and constructed in accordance with ACI 301 using ingredients conforming to ACI and ASTM standards which provide a good quality, dense, low permeability concrete. A search was performed to identify any incidences of degradation of missile shields which have been recorded in NPRDS, LERs, or Duke Power records. No instances of degradation were found. The aging effects analysis did not identify any aging effects requiring management for missile shields and the operating experience reviews validated that conclusion. Therefore, no aging effects requiring management were identified for missile shields.

On the other hand, a review of operating experience has identified leaching for Catawba and McGuire in walls and roofs exposed to external environments. The RWST missile wall is a free-standing reinforced concrete structure that is constructed similar to building structural exterior walls and has similar architectural features. Therefore, the operating experience for walls and roofs was conservatively determined to be applicable to the RWST missile wall. As a result, change in material properties due to leaching is an aging effect requiring management for the RWST missile wall for the extended period of operation. The *Inspection Program for Civil Engineering Structures and Components* is credited with managing aging effects for the RWST missile wall.

RAI 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.

Response to RAI 3.5-9
Metal housing systems, such as control boards, electrical & instrument panels, enclosures, etc., constructed of factory baked painted steel or galvanized sheet metal do not have a tendency to age with time [Reference "An Aging Assessment of Relay and Circuit Breakers and System Interactions," prepared by Franklin Research Center for Brookhaven National Laboratory, NUREG/CR-4715, June 1987]. Industry operating experience with metal housing systems indicates that they have performed without failure to the present [References "Aging Management Guideline for Commercial Nuclear Power Plants – Motor Control Centers," SAND 93-7069, Sandia National Laboratories, February 1994 and "Aging Management Guideline for Commercial Nuclear Power Plants – Electrical Switchgear," SAND 93-7027, Sandia National Laboratories, July 1993.]. Therefore, loss of material is not an aging effect requiring management for electrical panels, enclosures, and control boards in sheltered, Reactor Building, and external environments.

Cable tray is constructed of painted or galvanized sheet metal similar to metal housing and located in the same environment; therefore, cable tray would age similarly to the metal housings. A review of industry operating experience was also reviewed to validate this conclusion. Deficiencies that were identified were event driven or design/installation deficiencies. Therefore, loss of material is not an aging effect requiring management for cable trays in sheltered, Reactor Building, and external environments.

The New Fuel Storage Racks provide dry storage for new nuclear fuel. These racks are free-standing and are designed to accommodate fuel assemblies. The storage racks are fabricated from painted carbon steel and are located in a mild dry sheltered environment. A review of operating experience did not identify any aging effects requiring management. Therefore, loss of material is not an aging effect requiring management for the new fuel storage racks.

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

RAI 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.

Response to RAI 4.6-1

No fatigue analysis was performed for the steel containment vessel because it is not subjected to cyclic loads. Penetration bellows (listed in Table 3.5-1) are provided to absorb the loads associated with thermal expansion during operational transients as well as loads induced during the containment leak rate testing. Peak containment internal pressure resulting from the design basis LOCA or a design basis SSE are one-time loads and not cyclic loads requiring a fatigue analysis.

No analysis was performed for the steel containment vessel; therefore, it does not meet the criteria for a time-limited aging analysis.

RAI 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.

Response to RAI 4.6-2
Leakage of the bellows has not been attributed to cyclic fatigue; therefore, it does not invalidate the vendor analyzed cyclic life. Root causes of leakage of the bellows have been attributed to manufacturing defects, installation errors, damage during construction, and damage during maintenance activities. In one instance, leakage of the bellows was determined to result from trans-granular stress corrosion cracking due to the introduction of a corrosive environment (for additional information see response to RAI 3.5-5).

Although the leakage may result from any one of the root causes, the leakage is managed by the *Containment Leak Rate Testing Program*. The *Containment Leak Rate Testing Program* is identified in Table 3.5-1 to managing cracking of the bellows. Cracking of the bellows would manifest as leakage. Operating experience has been provided in response to RAI 3.5-5 which demonstrates the effectiveness of the *Containment Leak Rate Testing Program* in detecting leaks.

RAI 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.

Response to RAI 4.6-3
In preparing the application, Duke chose to follow the format provided in the draft "Standard Review Plan for the Review License Renewal Applications for Nuclear Power Plants" (August 2000). This draft version, as well as the version later issued in 2001 as NUREG-1800, includes Section 4.6, "Containment Liner Plate, Metal Containments, and Penetration Fatigue Analysis." Had this format guidance not been available, Duke would not have included the information in the application because it is not a TLAA.

However, given that Section 4.6 exists in the guidance documents, Duke had the option of either (1) simply stating that this section was not applicable to McGuire and Catawba and awaiting the likely staff requests for additional information (RAI) to further explain or (2) providing a complete response that demonstrates that this topic is not a TLAA and hopefully eliminating a staff RAI. Duke chose the latter option. The staff's inference that the topic may be a TLAA for McGuire and Catawba because it is addressed in Section 4.6 simply has no technical basis.

A cyclic analysis was performed for the bellows in the original design. The order of magnitude of the number of acceptable cycles was too large to base any safety judgment on the specific number. Since the analysis was not used as the basis for any safety judgement, the analysis does not meet Criteria 4 for a TLAA as defined in 10 CFR 54.3. As such, the analysis is not a TLAA.

However, cracking has been identified as an aging effect for the bellows in Table 3.5-1 of the Application. Cracking is managed by the *Containment Leak Rate Testing Program*. The *Containment Leak Rate Testing Program* is discussed in Appendix B.3.8 of the Application.

4.7.3 Depletion of Nuclear Service Water Pond Volume due to Runoff

RAI 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.

Response to RAI 4.7.3-1

In the event that a future survey of the topography of the bottom of the Nuclear Service Water Pond indicates a reduction in the volume of water due to the buildup of sediment, remedial actions may include, but not be limited to:

- enlargement of the pond by excavation,
- raising the required Technical Specification elevation,
- dredging of the pond, or
- modification of the pond to raise the surface elevation.

B.3.2 Battery Rack Inspections

RAI 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.

Response to RAI B.3.2-1
Loss of material of battery racks is managed by the *Battery Rack Inspections* described in Appendix B.3.2 of the Application. The *Battery Rack Inspections* is implemented by plant procedures. The procedures are used to inspect for loss of material of the battery racks and all sub-components (including battery rack nuts, bolts, rails, supports, seismic brace, and anchor bolts). The *Battery Rack Inspections* require visual examination of the battery racks including sub-components for physical damage or abnormal deterioration, including loss of material due to corrosion. The inspection acceptance criteria for loss of material in the procedure is "No significant amount of corrosion or rust spots visible." Physical damage or deterioration are evaluated to determine if the physical damage or deterioration affects the battery's ability to perform its intended function.

Visual inspections for these types of degradation have been addressed in NRC Inspection Procedure 62002, *Inspection of Structure, Passive Components, and Civil Engineering Features at Nuclear Power Plants*, and NEI 96-03, *Industry Guideline for Monitoring Structures.* The parameters monitored, acceptance criteria, inspection techniques, and corrective actions of the *Battery Racks Inspections* have previously been evaluated by the NRC staff and were found to be acceptable as documented in NUREG-1723, Section 3.8.3.2.1.

B.3.7 Containment Inservice Inspection Plan - IWE

RAI 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.

Response to RAI B.3.7-1

Category E-B

Category E-B (Pressure Retaining Welds) and E-F (Pressure Retaining Dissimilar Metal Welds) Examinations are excluded from the *Inservice Inspection Plan - IWE* for McGuire and Catawba. The basis for excluding these examinations is 10 CFR 50.55a(b)(2)(ix)(C) and SECY-96-080, which states that "the NRC concludes that requiring these inspections is not appropriate. There is no evidence of problems associated with welds of this type in operating plants."

Category E-D

Category E-D, Item #5.10 (Seals) and Item E5.20 (Gaskets) examinations are excluded from the *Inservice Inspection Plan - IWE* for McGuire and Catawba. The basis for excluding these examinations is documented in Duke Energy Corporation Request for Relief Serial No. 98-GO-001, approved by SER submitted by NRC letter dated September 3, 1998. Alternative examinations to be performed are as follows:

> "The leak-tightness of containment pressure retaining seals and gaskets will be verified by leak rate testing in accordance with 10 CFR 50, Appendix J, as required by Technical Specifications."

Category E-D, Item E5.30 (Moisture Barriers) are NOT excluded from the *Inservice Inspection Plan - IWE* for McGuire and Catawba.

Category E-G

Category E-G, Item E8.20 (Bolt Torque or Tension Tests for Bolted connections) are excluded from the *Inservice Inspection Plan - IWE* for McGuire and Catawba. The basis for excluding these examinations is documented in Duke Energy Corporation Request for Relief Serial No. 98-GO-002, approved by SER submitted by NRC letter dated November 24, 1998. Alternative examinations to be performed are as follows:

(1) Bolted connections shall receive a visual, VT-1 examination in accordance with

requirements of Table IWE-2500-1, Examination Category E-G, Pressure Retaining Bolting, Item No. E8.10, and

(2) A local leak rate test shall be performed on all containment penetrations, airlocks, and other pressure retaining bolted connections in accordance with 10 CFR 50, Appendix J.

Category E-G, Item E8.10 (Bolted Connections Visual, VT-1) are NOT excluded from the *Inservice Inspection Plan - IWE* for McGuire and Catawba.

RAI 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.

Response to RAI B.3.7-2

The Inservice Inspection requirements for Steel Containment Vessels at McGuire and Catawba Nuclear Stations currently comply with 10CFR50.55a and the ASME Boiler and Pressure Vessel Code, Section XI, 1992 Edition with the 1992 Addenda, as modified by approved Requests for Relief granted in accordance with 10CFR50.55a(a)(3)(i) and (a)(3)(ii). The following information provides the summaries of augmented examinations for each unit of McGuire and Catawba.

<u>McGuire Nuclear Station Unit 1</u>

1. The following items/areas are examined in accordance with Category E-C, Item E4.11:
   - Moisture barriers at the embedment zone around the periphery of the exterior side of the steel containment vessel
   - Moisture barrier at the interface between the steel containment vessel and the Fuel Transfer Tube Radiation shielding concrete on the exterior side of the steel containment vessel

   The above items were selected for augmented examination due to conditions observed on these moisture barriers when examined in accordance with Table IWE-2500-1, Examination Category E-D, Item E5.30.

2. The following items/areas are examined in accordance with Category E-C, Item E4.12:
   - Surface areas directly behind the insulation panel attached to the interior surface of the containment vessel approximately 36" above the embedment zone. These locations were selected for examination because the top of the insulation panel had not been sealed to prevent moisture intrusion, and because evidence of moisture intrusion had been noted during past inspections. Examination area is approximately 12" wide and extends nearly all of the way around the periphery of the containment vessel.

- Surface areas directly behind cork expansion joint material between the interior concrete structure and steel containment vessel at Elevation 752' + 1 3/8" between azimuths 104° and 122° (approx.). This location was selected for examination because the cork expansion joint material has not been removed at this location, and it is still possible for moisture to accumulate behind the expansion joint material. During past inspections, some staining had been observed beneath this area, indicating that moisture intrusion had occurred.

  Ultrasonic thickness measurements on the above surfaces are performed from the exterior of the containment vessel.

<u>McGuire Nuclear Station Unit 2</u>

1. The following items/areas are examined in accordance with Category E-C, Item E4.11:

   - Moisture barriers at the embedment zone around the periphery of the exterior side of the steel containment vessel, between azimuths 0° and 180° (approx.) and between azimuths 270° and 360° (approx.)
   - Moisture barrier at the interface between the steel containment vessel and the Fuel Transfer Tube Radiation shielding concrete on the exterior side of the steel containment vessel

   The above items were selected for augmented examination due to conditions observed on these moisture barriers when examined in accordance with Table IWE-2500-1, Examination Category E-D, Item E5.30.

2. The following items/areas are examined in accordance with Category E-C, Item E4.12:

   Examination areas are identical to those on Unit 1, except that the following additional area is also examined:

   Surfaces between the steel containment vessel and the Fuel Transfer Tube Radiation Shielding concrete on the interior of the vessel, between elevations 728'+4" and 729'+4". Examination area extends approximately 3 feet on each side of the Fuel Transfer Tube and is examined from the exterior of the containment vessel. This location was selected for examination because general visual examinations conducted in accordance with Table IWE-2500-1, Examination Category E-A, Item E1.11 detected evidence of borated water at this location on the interior surface of the containment vessel.

   Ultrasonic thickness measurements on the above surfaces are performed from the exterior of the containment vessel.

Catawba Nuclear Station Unit 1

1. The following items/areas are examined in accordance with Category E-C, Item E4.11:
   - Surface areas on the interior of the containment vessel, located between azimuths 247° and 303° (approx.), below Elevation 593'+10 1/2", along the top of the cork expansion joint material installed between the interior concrete structure and the containment vessel at the VX Fan Pit floor. This location was selected for examination because most of the cork expansion joint material has not been removed at this location, moisture intrusion has occurred, and some rusting and minor pitting has been observed on containment shell surfaces along the top of the cork material.

2. The following items/areas are examined in accordance with Category E-C, Item E4.12:
   - Surface areas directly behind the cork expansion joint material installed between the containment vessel and interior concrete structure at the VX Fan Pit floor between azimuths 247° and 303° (approx.), between Elevations 593'+9 3/8" and 596'+9 3/8" (approx.). This location was selected for examination because conditions noted at the VX Fan Pit floor on the interior of the containment vessel were considered to be an indicator of possible degradation of the containment vessel shell plate behind the expansion joint material.
   - Surface areas directly behind cork expansion joint material along the top of floor joints between the interior concrete structure and steel containment vessel at the following locations. These locations were selected for examination because most of the cork insulation panel has not been removed, and evidence of moisture and staining has been observed beneath these areas on the interior side of the vessel:
     - Between Elevations 565'+5 5/8" and 564'+5 5/8" (approx.), between azimuths 0° to 250°, and 270° to 360° (approx.)
     - Between Elevations 579'+1 3/8" and 578'+1 3/8" (approx.), between azimuths 104° to 122° (approx.)
     - Between Elevations 594'+8 3/8" and 593'+8 3/8" (approx.), between azimuths 0° to 247°, and 303° to 360° (approx.). This area is located at the ice condenser floor where it may be possible for moisture to accumulate against the containment vessel. The risk of potential degradation is considered higher here than for other areas of the containment vessel covered by insulation behind the ice condensers.

   Ultrasonic thickness measurements on the above surfaces are performed from the exterior of the containment vessel.

Catawba Nuclear Station Unit 2

1. The following items/areas are examined in accordance with Category E-C, Item E4.11:

   Examination areas are identical to those on Unit 1, except that the following additional items are also examined:

   - 2 Equipment Hatch latch bolts. These were selected for examination due to conditions found during the performance of Table IWE-2500-1, Category E-G, Item E8.30 examinations.

2. Items/areas that are examined in accordance with Category E-C, Item E4.12 are identical to those on Unit 1.

Summary of Plans to Continue Category E-C Examinations During the Extended Period of Operation at Catawba and McGuire

The above identified areas shall be examined in accordance with IWE-2420(c) until such time that the flaws, areas of degradation, or repairs remain essentially unchanged for three consecutive inspection periods. If other areas containing flaws or degradation are discovered during the performance of IWE examinations, and these areas warrant examination in accordance with Table IWE-2500-1, Category E-C, these other areas shall also be examined in accordance with IWE-2420.

B.3.8 Containment Leak Rate Testing Program

RAI 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.

Response to RAI B.3.8-1

The acceptance criterion of $0.07L_a$ is specified in Technical Specification Surveillance Requirement 3.6.3.8 as the maximum combined leakage rate. This criterion includes the leakage from all penetration bellows. The leakage from the bellows would be added to all other bypass leakage. The total combined leakage is required to be less than $0.07L_a$. As such, the test leakage of any individual bellows assembly will be less than $0.07L_a$ over the extended life of the plant during normal operations as well as during design basis events.

RAI 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.

Response to RAI B.3.8-2
1. For McGuire, twenty (20) bellows are designated as leaking. One bellows has been replaced at McGuire. For Catawba, three (3) bellows are designated as leaking. No bellows have been replaced at Catawba. For additional information concerning the replaced bellows, reference Appendix B.3.8 of the Application and Response to RAI 3.5-5.

2. No Class A bellows exist at Catawba or McGuire because there are no Class 1 pipe penetrations through the containment. The answer for question 1 applies to Class B penetrations.

3. Technical Specification 3.6.1 contains the leakage limits for continued operation. These leakage limits were used in the analysis of off-site doses resulting from accidents. The leakage rate is defined in 10 CFR, Appendix J. The leakage of the bellows remains below the limits specified in Technical Specification 3.6.1.

B.3.10 Crane Inspection Program

RAI 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.

Response to RAI B.3.10-1
McGuire and Catawba management assign the personnel who perform the inspection of the cranes. The individuals are chosen based on education and work experience to ensure that they are well qualified. The acceptability of the crane rails and girders is based on condition monitoring. Acceptability based on condition monitoring is described in NRC Regulatory Guide 1.160, *Monitoring the Effectiveness of Maintenance at Nuclear Power Plants.*

The condition of the crane rails and griders are determined by visual inspection for degradation such as:
- Cracked or broken welds
- Cracks in webs and flanges
- Corrosion due to rust of steel components
- Bent or damaged structural members
- Broken or missing bolts
- Wear on the rails.

These criteria are in accordance with the criteria identified in ASME/ANSI requirements and OSHA regulations.

The assessment of the severity of the observed degradation and the determination of whether corrective action is necessary is based on the judgment of the inspector. The inspector uses the criteria which are in accordance with ASME/ANSI requirements and OSHA regulations to make his determination.

Visual inspections for these types of degradation have been addressed in NRC Inspection Procedure 62002, *Inspection of Structure, Passive Components, and Civil Engineering Features at Nuclear Power Plants*, and NEI 96-03, *Industry Guideline for Monitoring Structures.* The parameters monitored, acceptance criteria, inspection techniques, and corrective actions of the *Crane Inspection Program* have previously been evaluated by the NRC staff and were found to be acceptable as documented in NUREG-1723, Section 3.8.3.2.1 (page 3-246).

B.3.12.1 Fire Barrier Inspection

RAI 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.

Response to RAI B.3.12.1-1
Fire barrier penetration seals are inspected on a frequency as directed by Selected Licensee Commitment (SLC) 16.9.5. The limits for the acceptance criteria are specified in the station procedures. The limits are discussed in more detail below.

Crumbling, gouges or voids on fiberboard damming surface shall not exceed one-half (1/2) inch deep by one (1) inch length and width.

Fiberboard dams should be as flush with the fire barrier and with other pieces of damming board as possible. A maximum one-quarter (1/4) inch gap is acceptable.

For fire barrier penetration seals without permanent damming, the limit of separation of foam from the barrier perimeter or components passing through the seal shall not exceed one-quarter (1/4) inch wide by three (3) inches deep and unlimited length.

For fire barrier penetration seals without permanent damming, gouges or voids on the front side or backside surface of the foam shall not exceed one-half (1/2) inch deep by one (1) inch length and width.

For fire barrier penetration seals with permanent damming, the limit of separation of foam from the barrier perimeter or components passing through the seal shall not exceed three-quarter (3/4) inch wide by four (4) inches deep and unlimited length.

For fire barrier penetration seals with permanent damming, gouges or voids on the front side or backside surface of the foam shall not exceed three-quarter (3/4) inch wide by four (4) inches deep and unlimited length.

The acceptance criteria are based on experimental tests and engineering analysis as documented in station specifications.

B.3.13 Flood Barrier Inspection

RAI 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.

Response to RAI B.3.13-1

McGuire management assigns the personnel who perform the inspection of the flood barriers. The individuals are chosen based on education and work experience to ensure that they are well qualified. The inspector visually examines the flood seals for cracking and change in material properties that would result in loss of the intended function of the seal. The assessment of the severity of the observed degradation and the determination of whether corrective action is necessary are based on the judgment of the inspector. Visual inspections for these types of degradation have been addressed in NRC Inspection Procedure 62002, *Inspection of Structure, Passive Components, and Civil Engineering Features at Nuclear Power Plants*, and NEI 96-03, *Industry Guideline for Monitoring Structures.*

If the inspector determines that degradation is evident that would lead to loss of intended function, a problem report will be developed in accordance with Problem Investigation Process of Nuclear System Directive 208. The Problem Investigation Process is a formalized process for documenting engineering evaluations of plant problems and would include the assessment of the severity of the observed degradation, the need for corrective actions, the need for further inspections of other locations, and the need for future inspections or programmatic oversight.

B.3.21 Inspection Program for Civil Engineering Structures and Components

RAI 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.

Response to RAI B.3.21-1

McGuire and Catawba management assign the inspectors and the "accountable engineer." The individuals are chosen based on education and work experience to ensure that they are well qualified. The qualifications of the inspectors and the "accountable engineer" are documented in McGuire and Catawba site documents. The inspectors should be civil/structural engineering graduates with at least 4 years experience in evaluation of inservice structures. The "accountable engineer" should be a civil/structural engineering graduate, registered professional engineer with at least 4 years experience in the evaluation of inservice structures. The oversight of the training and qualification of the "accountable engineer" is governed by the Duke *Quality Assurance Topical Report.*

RAI 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.

Response to RAI B.3.21-2

The "accountable engineer" assesses the severity of the degradation and determines whether corrective action is necessary based on his knowledge, experience, and training. The qualifications of the "accountable engineer" are discussed in Response to RAI B.3.21-1. The acceptability of a structure is based on whether the "accountable engineer" determines that the structure is capable of performing its intended function(s). Acceptability based on condition monitoring and the capability to perform the intended function is described in NRC Regulatory Guide 1.160, *Monitoring the Effectiveness of Maintenance at Nuclear Power Plants.*

Examination and assessment of the condition of a structure is performed by the "accountable engineer" using guidance provided in codes and standards such as:

- NEI 96-03, *Industry Guideline for Monitoring Structures*
- NRC Regulatory Guide 1.127, *Inspection of Water-Control Structures Associated with Nuclear Power Plants*
- ACI 349.3, *Evaluation of Existing Nuclear Safety-related Concrete Structures*

The condition of the structure is determined by visual inspection for degradation such as:

- Spalling, cracking, delaminations, honeycombs, water in-leakage, and leaching of concrete,
- Cracks in joints of masonry walls,
- Corrosion, peeling paint, cracked welds, and loose or missing anchors of structural steel, and
- Settlement and cracked concrete for equipment foundations.

If the accountable engineer determines that degradation is evident that would lead to loss of intended function, a problem report will be developed in accordance with Problem Investigation Process of Nuclear System Directive 208. The Problem Investigation Process is a formalized process for documenting engineering evaluations of plant problems and would include the assessment of the severity of the observed degradation, the need for corrective actions, the need for further inspections of other locations, and the need for future inspections or programmatic oversight.

Visual inspections for these types of degradation have been addressed in NRC Inspection Procedure 62002, *Inspection of Structure, Passive Components, and Civil Engineering Features at Nuclear Power Plants*, and NEI 96-03, *Industry Guideline for Monitoring Structures.* The parameters monitored, acceptance criteria, inspection techniques, and corrective actions of the *Inspection Program for Civil Engineering Structures* have previously been evaluated by the NRC staff and were found to be acceptable as documented in NUREG-1723, Section 3.2.6.

B.3.30 Standby Nuclear Service Water Pond Dam Inspection

RAI 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.

Response to RAI B.3.30-1

The Catawba UFSAR Supplement Table 18-1 entry for the Standby Nuclear Service Pond Dam Inspection is incorrect. The table entry will be revised to reference Technical Specification Surveillance Requirement (SR) 3.7.9.3.

RAI 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.

Response to RAI B.3.30-2
McGuire and Catawba management assign the "accountable engineer" responsible for evaluating the performance of the SNSWP Dam and recommending any needed repairs. The individuals are chosen based on education and work experience to ensure that they are well qualified. The qualifications of the "accountable engineer" are in accordance with the guidance provided in Regulatory Guide 1.127, *Inspection of Water-Control Structures Associated with Nuclear Power Plants*. The "accountable engineer" should be a registered professional engineer experienced in the investigation, design, construction, and operation of dams. The oversight of the training and qualification of the "accountable engineer" is governed by the Duke *Quality Assurance Topical Report*.

RAI 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.

Response to RAI B.3.30-3
The "accountable engineer" assesses the severity of the degradation and determines whether corrective action is necessary based on his knowledge, experience, and training. The qualifications of the "accountable engineer" are discussed in Response to RAI B.3.30-1. The acceptability of the dam is based on whether the "accountable engineer" determines that the dam is capable of performing its intended function(s). Acceptability based on condition monitoring and the capability to perform the intended function is described in NRC Regulatory Guide 1.160, *Monitoring the Effectiveness of Maintenance at Nuclear Power Plants.*

Examination and assessment of the condition of the dam is performed by the "accountable engineer" using guidance provided in codes and standards such as:

- NRC Regulatory Guide 1.127, *Inspection of Water-Control Structures Associated with Nuclear Power Plants*
- 18 CFR Part 12, *Water Power Projects and Project Works Safety*

Aging effects are identified through visual examination of the dam and review of relevant operational data from piezometric readings, settlement monitoring, and observation wells. In accordance with the guidance provided in Regulatory Guide 1.127 and 18 CFR Part 12, both faces of the dam are inspected for:

- Seepage
- Slides
- Erosion
- Abnormal degradation
- Vegetative growth

If the "accountable engineer" determines that degradation is evident that would lead to loss of intended function, a problem report will be developed in accordance with Problem Investigation Process of Nuclear System Directive 208. The Problem Investigation Process is a formalized process for documenting engineering evaluations of plant problems and would include the assessment of the severity of the observed degradation, the need for corrective actions, the need for further inspections of other locations, and the need for future inspections or programmatic

oversight. The acceptability of corrective actions is also evaluated by the NRC during their dam safety audits (See discussion in operating experience in Appendix B.3.30 of the Application).

B.3.33 Technical Specification SR 3.6.16.3 Visual Inspection

RAI 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?

Response to RAI B.3.33-1

Visual inspections of the interior surface of the concrete reactor building are performed in the annulus space between the exterior of the steel containment vessel and the concrete reactor building structure. Containment vessel stiffening rings are located at 10-foot intervals along the exterior of the steel containment vessel and act as a platform for the inspectors to stand on while examining the concrete surface of the reactor building. In addition, ladders are used to access the exterior containment dome, and binoculars are used to visually inspect the exterior reactor building walls.

RAI 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.

Response to RAI B.3.33-2
The "accountable engineer" assesses the severity of the degradation and determines whether corrective action is necessary based on his knowledge, experience, and training. The acceptability of a structure is based on whether the "accountable engineer" determines that the structure is capable of performing its intended function(s). Acceptability based on condition monitoring and the capability to perform the intended function is described in NRC Regulatory Guide 1.160, *Monitoring the Effectiveness of Maintenance at Nuclear Power Plants.*

Examination and assessment of the condition of a structure is performed by the "accountable engineer" using guidance provided in codes and standards such as:

- NEI 96-03, *Industry Guideline for Monitoring Structures*
- ACI 349.3, *Evaluation of Existing Nuclear Safety-related Concrete Structures*
- ACI 201, *Guide for Making a Condition Survey of Concrete in Service*

The condition of the structure is determined by visual inspection for degradation such as loss of material due to leaching of the concrete. Visual inspections for these types of degradation have been addressed in NRC Inspection Procedure 62002, *Inspection of Structure, Passive Components, and Civil Engineering Features at Nuclear Power Plants*, and NEI 96-03, *Industry Guideline for Monitoring Structures.*

If the "accountable engineer" determines that degradation is evident that would lead to loss of intended function, a problem report will be developed in accordance with Problem Investigation Process of Nuclear System Directive 208. The Problem Investigation Process is a formalized process for documenting engineering evaluations of plant problems and would include the assessment of the severity of the observed degradation, the need for corrective actions, the need for further inspections of other locations, and the need for future inspections or programmatic oversight.

B.3.35 Underwater Inspection of Nuclear Service Water Structures

RAI 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.

Response to RAI B.3.35-1

McGuire and Catawba management assign the "accountable engineer" who is responsible for evaluating the inspection results and determining any needed repairs. The individual is chosen based on education and work experience to ensure that he is well qualified. The qualifications of the "accountable engineer" are in accordance with the guidance provided in NRC Regulatory Guide 1.127, *Inspection of Water-Control Structures Associated with Nuclear Power Plants.* The "accountable engineer" should be a registered professional engineer experienced in investigation, design, construction, and operation of water-control structures. The oversight of the training and qualification of the "accountable engineer" is governed by the Duke *Quality Assurance Topical Report.*

The following statement will be added to Section 18.2.27 for McGuire and Section 18.2.26 for Catawba in the respective UFSAR Supplement:

"The qualifications of the accountable engineer are in accordance with the guidance provided in NRC Regulatory Guide 1.127."

RAI 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.

Response to RAI B.3.35-2
The "accountable engineer" assesses the severity of the degradation and determines whether corrective action is necessary based on his knowledge, experience, and training. The qualifications of the "accountable engineer" are discussed in Response to RAI B.3.35-1. The acceptability of a structure is based on whether the "accountable engineer" determines that the structure is capable of performing its intended function(s). Acceptability based on condition monitoring and the capability to perform the intended function is described in NRC Regulatory Guide 1.160, *Monitoring the Effectiveness of Maintenance at Nuclear Power Plants.*

Examination and assessment of the condition of a structure is performed by the "accountable engineer" using guidance provided in codes and standards such as:

- NRC Regulatory Guide 1.127, *Inspection of Water-Control Structures Associated with Nuclear Power Plants*
- ACI 349.3, *Evaluation of Existing Nuclear Safety-Related Concrete Structures*
- ACI 201, *Guide for Making a Condition Survey of Concrete in Service*

The condition of the structure is determined by visual inspection for degradation such as:

- Loss of material and cracking of the concrete, and
- Loss of material of steel components.

Visual inspections for these types of degradation have been addressed in NRC Inspection Procedure 62002, *Inspection of Structure, Passive Components, and Civil Engineering Features at Nuclear Power Plants*, and NEI 96-03, *Industry Guideline for Monitoring Structures.*

If the "accountable engineer" determines that degradation is evident that would lead to loss of intended function, a problem report will be developed in accordance with Problem Investigation Process of Nuclear System Directive 208. The Problem Investigation Process is a formalized process for documenting engineering evaluations of plant problems and would include the assessment of the severity of the observed degradation, the need for corrective actions, the need for further inspections of other locations, and the need for future inspections or programmatic oversight.

Attachment 2
Application to Renew the Operating Licenses of
McGuire Nuclear Station and Catawba Nuclear Station
Responses to NRC Requests for Additional Information
NRC Letter dated January 28, 2002

LIST OF COMMITMENTS

Attachment 2

Duke Letter Dated March 11, 2002
McGuire Nuclear Station and Catawba Nuclear Station
Responses to NRC Requests for Additional Information

List Of Commitments

1. The Catawba UFSAR Supplement Table 18-1 entry for the Standby Nuclear Service Pond Dam Inspection will be revised to reference Technical Specification Surveillance Requirement (SR) 3.7.9.3. (Response to RAI B.3.30-1)

2. The following statement will be added to Section 18.2.27 for McGuire and Section 18.2.26 for Catawba in the respective UFSAR Supplement:

"The qualifications of the accountable engineer are in accordance with the guidance provided in NRC Regulatory Guide 1.127."

(Response to RAI B.3.35-1)