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July 9, 2002

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

Subject: 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). During its review of the information provided by Duke in the Application, the staff identified areas where additional information was needed to complete its review. Duke provided responses to these requests for additional information by letters dated March 1, 2002, March 8, 2002, March 11, 2002, March 15, 2002, and April 15, 2002.

By letter dated June 26, 2002, the staff identified 29 Open Items and 10 Confirmatory Items during the process of preparing its Safety Evaluation Report on the McGuire and Catawba Application. The staff has requested that Duke provide additional information in response to these items in a timely manner in order that the staff can issue its Safety Evaluation Report with open items in August 2002.

Accordingly, the Duke responses to these requests for additional information are provided in Attachment 1 to this letter.

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

Very truly yours,

M. S. Tuckman

M. S. Tuckman

Attachment:

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 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 9TH day of July 2002.

Mary P. Nehrs

Notary Public

My Commission Expires:

JAN 22, 2006

xc: (w/ Attachment)

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Attachment 1
Application to Renew the Operating Licenses of
McGuire Nuclear Station and Catawba Nuclear Station

Response to NRC Letter dated June 26, 2002

Duke's Response to RAI 2.3-1

The red, triangular LR flags define the license renewal evaluation boundaries on mechanical system flow diagrams, and highlighting was used as an aid to Duke in component screening and for the reviewer in understanding the system under review. In some cases, components were outlined in highlighting, and in others, the highlighting was simply drawn through components. Either way is acceptable for achieving the purpose of the drawings. The components are shown to be within the license renewal evaluation boundaries, and therefore, within the scope of license renewal.

The air handling unit housings cited in RAI 2.3-1, Item 4, are subject to aging management review and are listed in Table 3.3-11 (page 3.3-111, row 1) as Air Handling Units (Heat Exchanger Shells).

Cooling fans are not included in the aging management review results tables in the Application. Cooling fans, without sub-component exceptions, are explicitly excluded from an aging management review by §54.21(a)(1)(i) of the Rule. As an aid to the reviewer, the following excerpt of §54.21(a)(1)(i) is provided (underline added to highlight cooling fan exclusion from aging management review):

> **10 CFR 54.21(a)(1)(i):**
> That perform an intended function, as described in §54.4, without moving parts or without a change in configuration or properties. These structures and components include, but are not limited to, the reactor vessel, the reactor coolant system pressure boundary, steam generators, the pressurizer, piping, pump casings, valve bodies, the core shroud, component supports, pressure retaining boundaries, heat exchangers, ventilation ducts, the containment, the containment liner, electrical and mechanical penetrations, equipment hatches, seismic Category I structures, electrical cables and connections, cable trays, and electrical cabinets, <u>excluding</u>, but not limited to, pumps (except casing), valves (except body), motors, diesel generators, air compressors, snubbers, the control rod drive, ventilation dampers, pressure transmitters, pressure indicators, water level indicators, switchgears, <u>cooling fans</u>, transistors, batteries, breakers, relays, switches, power inverters, circuit boards, battery chargers, and power supplies; and

Staff Concern - RAI 2.3-1 (Open Item)

On May 1, 2002, the staff issued guidance on the treatment of housings for all active components (ML021220429). The following is excerpted from the guidance document to illustrate the staff's concern with Duke's response to this RAI and RAIs 2.3-2, 2.3-3, 2.3-6, 2.3-7, 2.3-8, and 2.3-9:

> The SOC articulates the underlying philosophy of the Rule that during the extended period of operation, safety-related functions should be maintained in the same manner and to the same extent as during the current licensing term. Aging effects that could adversely impact on the

ability of SSCs to maintain these safety-related functions during the extended period of operation should be evaluated.

10 CFR 54.21(a)(1) provides that those components that perform their intended functions without moving parts and without a change in configuration or properties (10 CFR 54.21(a)(1)(i)) and that are not subject to replacement based on qualified life or specified time period (10 CFR 54.21(a)(1)(ii)) are subject to an AMR. Such components are commonly considered as "long-lived" and as performing a passive function. 10 CFR 54.21(a)(1)(i) states that "These structures and components include, but are not limited to,... pump casings, valve bodies ... " and lists other components that perform passive functions. The examples cited in the license renewal rule illustrate components with significant passive functions.

Section III.f.i(a) of the SOC further explains that major components may have active functions, passive functions, or both, and cites pumps and valves as examples. Pumps and valves have moving parts, but the Commission concluded that the pressure-retaining function performed by the pump casing and the valve body should be subject to an AMR. The SOC further explains that the Commission does not limit the consideration of pressure boundaries to reactor coolant pressure boundary. The exclusion regarding components is focused on active functions rather than on the exclusion of the entire component, while the AMR applies to the passive function of the component.

On this basis, the staff concludes that the discussion of pump casings and valve bodies in both the Rule and the SOC are provided as examples of how an applicant should evaluate housings for active components, and that proper implementation of the Rule requires screening evaluations to consider not just the active component, but the intended function of its associated housing. Specifically, the staff believes that the housings of active components (e.g., housings for fans, dampers, and heating and cooling coils) may perform a critical pressure retention and/or structural integrity function which, should that function not be maintained, could prevent the associated active component from performing its function. Further, if such housings perform these functions and meet the long-lived and passive criteria, then the housings should be subject to an AMR.

Duke Response to Staff Concern - RAI 2.3-1 (Open Item)

On May 1, 2002, the staff issued guidance on the treatment of housings for all active components (ML021220429). Industry review of the staff guidance is currently in progress. Until the industry review is complete, Duke has decided to defer a response to the following potential open items, or portions thereof, that concern cooling fans and ventilation dampers and are affected by this staff guidance:

- RAI 2.3-1
- RAI 2.3-2
- RAI 2.3-6
- RAI 2.3-7 (Item 5 Only)
- RAI 2.3-8 (Items 3, 7, and 8 Only)
- RAI 2.3-9

Duke's Response to RAI 2.3-2

Ventilation dampers are not included in the aging management review results tables in the Application. Ventilation dampers, without sub-component exceptions, are explicitly excluded from an aging management review by §54.21(a)(1)(i) of the Rule. As an aid to the reviewer, the following excerpt of §54.21(a)(1)(i) is provided (underline added to highlight ventilation damper exclusion from aging management review):

> **10 CFR 54.21(a)(1)(i):**
> That perform an intended function, as described in §54.4, without moving parts or without a change in configuration or properties. These structures and components include, but are not limited to, the reactor vessel, the reactor coolant system pressure boundary, steam generators, the pressurizer, piping, pump casings, valve bodies, the core shroud, component supports, pressure retaining boundaries, heat exchangers, ventilation ducts, the containment, the containment liner, electrical and mechanical penetrations, equipment hatches, seismic Category I structures, electrical cables and connections, cable trays, and electrical cabinets, <u>excluding</u>, but not limited to, pumps (except casing), valves (except body), motors, diesel generators, air compressors, snubbers, the control rod drive, <u>ventilation dampers</u>, pressure transmitters, pressure indicators, water level indicators, switchgears, cooling fans, transistors, batteries, breakers, relays, switches, power inverters, circuit boards, battery chargers, and power supplies; and

Staff Concern - RAI 2.3-2 (Open Item)

Refer to the staff's concern (as documented herein) about Duke's response to RAI 2.3-1.

Duke Response to Staff Concern - RAI 2.3-2 (Open Item)

Please refer to Duke's Response to Staff Concern - RAI 2.3-1 (Open Item) relative to ventilation dampers.

Duke's Response to RAI 2.3-4

Duke does not define materials such as sealants as structures or components. However, Duke recognizes that limited situations may exist where these materials are important in maintaining the integrity of the component to which they are connected. For such situations, the license renewal or component intended function supported by the sealant is to maintain the building pressure boundary envelope. The pressure boundary function is addressed by surveillance testing to demonstrate compliance with McGuire and Catawba technical specifications. The testing is performed on the frequency specified in the technical specifications to ensure the integrity of the building pressure boundary envelope. The following information identifies the building envelopes and the technical specifications that address those envelopes:

- The sealants for the Control Room pressure boundary envelope are addressed by surveillance testing to demonstrate compliance with McGuire Technical Specification 3.7.9 and Catawba Technical Specification 3.7.10.
- The sealants for the Auxiliary Building pressure boundary envelope are addressed by surveillance testing to demonstrate compliance with McGuire Technical Specification 3.7.11 and Catawba Technical Specification 3.7.12.
- The sealants for the Fuel Building pressure boundary envelope are addressed by surveillance testing to demonstrate compliance with McGuire Technical Specification 3.7.12 and Catawba Technical Specification 3.7.13.
- The sealants for the Reactor Building pressure boundary envelope are addressed by surveillance testing to demonstrate compliance with McGuire and Catawba Technical Specification 3.6.10.

The McGuire modifications discussed in the RAI were described in McGuire Updated Final Safety Analysis Report (UFSAR) Section 6.2.3.3 and were not described in the Application. The modifications were made to the containment personnel access hatches and to the main steam and feedwater piping penetrations in order to remove potential bypass leak paths. In the case of the personnel hatches, an enclosure was added around the outside of the hatch such that any leakage past the doors seals is directed back to the annulus. Similarly, for the main steam and feedwater penetrations, the test connections on the outer bellows of each penetration were routed back to the annulus such that any leakage past the inner bellows is into the annulus. In both of these modifications, sealants were not used.

Staff Concern - RAI 2.3-4 (Open Item)

The applicant's response to RAI 2.3-4 states they do not define materials like sealants as structures or components and that the pressure boundary function is addressed by surveillance testing to demonstrate compliance with technical specifications. The applicant's basis for excluding building sealants from an aging management review is not consistent with the 10 CFR 54.21 of the license renewal rule because current testing to demonstrate compliance with technical specifications does not preclude a component or structure from meeting the scoping criteria of 10 CFR 54.4 or the screening criteria defined in 10 CFR 54.21. On April 20, 1999, the staff issued its position on License Renewal Issue No. 98-0012, "Consumables," by letter to Douglas Walters, Nuclear Energy Institute (NEI). This position is also provided as guidance in Table 2.1-3 of NUREG 1800, "Standard Review Plan for License Renewal," issued in July 2001. Similarly, NEI 95-10, "Industry Guideline for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule," issued in March 2001, reflects the staff's position in Section 4.1.2, Determining Structures and Components Subject to Aging Management Review and Their Intended Functions. According to the staff's position (as stated in the April 20, 1999, letter) and associated license renewal guidance documents, structural sealants are not treated as consumables like packing, gaskets, component seals and O-rings because they are typically required for maintaining the structural integrity of safety-related structures and perform these functions without moving parts or change in configuration or properties. These sealants typically are not replaced based on qualified life or specified time period; often are relied upon for decades of service; and are subject to aging.

On the basis of the Rule and associated guidance documents, the staff expects applicants for license renewal to identify sealant materials used to maintain safety-related buildings at the proper differential pressure with respect to adjacent areas. These sealant materials should be included within the scope of license renewal as a structural component and subject to an aging management review.

Duke Response to Staff Concern - RAI 2.3-4 (Open Item)

The staff concern states that "the applicant's basis for excluding building sealants from an aging management review is not consistent with the 10 CFR 54.21 of the license renewal rule because current testing to demonstrate compliance with technical specifications does not preclude a component or structure from meeting the scoping criteria of 10 CFR 54.4 or the screening criteria defined in 10 CFR 54.21." The staff incorrectly interpreted Duke's response. Duke does not include the sealants as discrete components in an aging management review because the sealants are not identified as structures or components. The guidance provided in NUREG-1800, "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants," states that structural sealants are "considered as subcomponents and are not explicitly called out in the scoping and screening procedures." Although these sealants are not listed as components, the function supported by the sealant is to maintain the building pressure boundary envelope. The pressure boundary function is addressed by McGuire and Catawba technical specifications. The technical specifications are discussed in Duke's original response to RAI 2.3-4.

Duke's Response to RAI 2.3-6

Referring to the flow diagrams provided with the Application, the areas that constitute the McGuire control room envelope are areas designated on MC-1578-2 as the Control Room, Instrument Room, and Storage Room. For Catawba, the areas that constitute the control room envelope are areas designated on CN-1578-1.0 as the Control Room, Operator's Office, and Interface Office. The control area ventilation system components inside the main control room envelope relied on to perform safety-related cooling and filtration functions to maintain the control room habitable are within the license renewal evaluation boundaries shown on the highlighted flow diagrams for the Control Area Ventilation System. Components within those evaluation boundaries that are subject to aging management review are presented in Table 3.3-11 of the Application. Table 3.3-11 lists components such as air handling units, ductwork, and valve bodies. Components such as ventilation dampers and cooling fans are not included in the aging management review results tables in the Application as ventilation dampers and cooling fans, without sub-component exceptions, are explicitly excluded from an aging management review by §54.21(a)(1)(i) of the Rule. As an aid to the reviewer, the following excerpt of §54.21(a)(1)(i) is provided (underline added to highlight ventilation damper and cooling fan exclusion from aging management review):

> **10 CFR 54.21(a)(1)(i):**
> That perform an intended function, as described in §54.4, without moving parts or without a change in configuration or properties. These structures and components include, but are not limited to, the reactor vessel, the reactor coolant system pressure boundary, steam generators, the pressurizer, piping, pump casings, valve bodies, the core shroud, component supports, pressure retaining boundaries, heat exchangers, ventilation ducts, the containment, the containment liner, electrical and mechanical penetrations, equipment hatches, seismic Category I structures, electrical cables and connections, cable trays, and electrical cabinets, excluding, but not limited to, pumps (except casing), valves (except body), motors, diesel generators, air compressors, snubbers, the control rod drive, ventilation dampers, pressure transmitters, pressure indicators, water level indicators, switchgears, cooling fans, transistors, batteries, breakers, relays, switches, power inverters, circuit boards, battery chargers, and power supplies; and

Staff Concern - RAI 2.3-6 (Open Item)

Refer to the staff's concern (as documented herein) about Duke's response to RAI 2.3-1.

Duke Response to Staff Concern - RAI 2.3-6 (Open Item)

Please refer to Duke's Response to Staff Concern - RAI 2.3-1 (Open Item) relative to ventilation dampers.

Duke's Response to RAI 2.3-7

Referring to the seven components identified in RAI 2.3-7, for item (1), the Auxiliary Building Ventilation System moisture eliminators are subject to aging management review and are a sub-component of the component on drawing CN-1577-1.3 named "PRHDS-XX" (where "XX" is the unit and train designation). This set of components is identified in Table 3.3-1 (page 3.3-8, row 1) of the Application as "Pump Room Heater-Demister (CNS Only)."

For item (2), the Control Area Ventilation System moisture eliminators and pre-filters are subject to aging management review and are a sub-component of the component on drawing CN-1578-1 named "CRA-PFT." This set of components is identified in Table 3.3-11 (page 3.3-111, row 5) of the Application as "Control Room Area Pressurizing Filter Trains (CNS Only)."

For item (3), the Diesel Building Ventilation System duct heaters should have been highlighted on flow diagram MC-2579-1, indicating that they are within the scope of license renewal. This issue is the same as RAI 2.3.3.10-1. For the Diesel Building Ventilation System duct heaters in item (3) and the Turbine Building Ventilation System duct heaters in item (6), the duct heaters consist of electric heating elements that are mounted inside the ductwork. The duct-mounted electrical heating elements do not have a pressure boundary function or any other component intended function for license renewal and are, therefore, not subject to an aging management review.

For item (4), the ductwork connection from the Auxiliary Building Ventilation System to the Unit 2 Vent (shown on flow diagram CN-1577-1.2 at F-11) should have been highlighted on flow diagram CN-2577-3.0 at E-7. Ductwork for this section is contained in Table 3.3-1 (page 3.3-6, row 4) of the Application.

For item (5), while the ventilation damper is highlighted on flow diagram CN-2577-2.0, indicating that it is within the scope of license renewal, ventilation dampers are not included in the aging management review results tables in the Application. Ventilation dampers, without sub-component exceptions, are explicitly excluded from an aging management review by §54.21(a)(1)(i) of the Rule. As an aid to the reviewer, the following excerpt of §54.21(a)(1)(i) is provided (underline added to highlight ventilation damper exclusion from aging management review):

> **10 CFR 54.21(a)(1)(i):**
> That perform an intended function, as described in §54.4, without moving parts or without a change in configuration or properties. These structures and components include, but are not limited to, the reactor vessel, the reactor coolant system pressure boundary, steam generators, the pressurizer, piping, pump casings, valve bodies, the core shroud, component supports, pressure retaining boundaries, heat exchangers, ventilation ducts, the containment, the containment liner, electrical and mechanical penetrations, equipment hatches, seismic Category I structures, electrical

> cables and connections, cable trays, and electrical cabinets, excluding, but not limited to, pumps (except casing), valves (except body), motors, diesel generators, air compressors, snubbers, the control rod drive, ventilation dampers, pressure transmitters, pressure indicators, water level indicators, switchgears, cooling fans, transistors, batteries, breakers, relays, switches, power inverters, circuit boards, battery chargers, and power supplies; and

For item (7), the Turbine Building Ventilation System pre-filters that are shown on flow diagram MC-1614-4 are removable components within the air handling units. The air handling units are listed in Table 3.3-46 (page 3.3-257, row 1) in the Application. Filtration is not required of the pre-filters in support of the Turbine Building Ventilation System function within the scope of license renewal. Therefore, system pre-filters are excluded from an aging management review.

Staff Concern - RAI 2.3-7 (Open Item)

Refer to the staff's concern (as documented herein) about Duke's response to RAI 2.3-1 for items (5) and (7).

Duke Response to Staff Concern - RAI 2.3-7 (Open Item)

For Item (5), please refer to Duke's Response to Staff Concern - RAI 2.3-1 (Open Item) relative to ventilation dampers.

For Item (7) – The staff's concern as documented in Staff Concern – RAI 2.3-1 (Open Item) does not address its concern with RAI 2.3-7, Item 7. The pre-filters (filter medium and housing) are passive components not covered by the treatment of housings of active components. The pre-filter (filter medium and housing) is a sub-component of the Turbine Building Ventilation System air handling unit. The pre-filter is similar to the filtration system found in residential heating systems. A disposable filter is inserted in the air handling unit and is replaced on condition. Filtration is not a component intended function of the pre-filters of the Turbine Building Ventilation System. Therefore, the filter medium is excluded from an aging management review.

Even if they had an intended function, filters are excluded from an aging management review as they are replaced on condition which conforms to the staff's guidance on the treatment of consumables in the March 10, 2000 letter to Douglas J. Walters (NEI) from Christopher I. Grimes (NRC) and SRP-LR Table 2.1-3, "Specific Staff Guidance on Screening."

The housing containing the filter medium has the component intended function of pressure boundary, and therefore, requires an aging management review. The housings were evaluated with the air handling units. The results of the aging management review of the air handling units are listed in Table 3.3-46 (page 3.3-257, row 1) of the Application.

Duke's Response to RAI 2.3-8

Prior to providing specific responses to each item in RAI 2.3-8, a point of clarification is offered to the reviewer. The tables in the Application indicate those components that are subject to an aging management review. The tables do not indicate all components within the scope of license renewal as stated in the RAI. The red, triangular LR flags define the license renewal evaluation boundaries on mechanical system flow diagrams.

Referring to the eight components identified in RAI 2.3-8, for item (1), the Control Area Ventilation System orifice that is identified in Table 3.3-11 (page 3.3-112, row 3) of the Application is highlighted on flow diagram MC-1578-1.0 at E-3.

For item (2), the red, triangular LR flags define the license renewal evaluation boundaries on mechanical system flow diagrams, and highlighting was used as an aid to Duke in component screening and for the reviewer in understanding the system under review. In some cases, components were outlined in highlighting, and in others, the highlighting was simply drawn through components. Either way is acceptable for achieving the purpose of the drawings. The components are shown to be within the license renewal evaluation boundaries, and therefore, within the scope of license renewal. The Control Area Ventilation System air handling units cited are shown on MC-1578-4.0 to be within the evaluation boundaries, indicating that they are within the scope of license renewal. The air handling units are included in Table 3.3-11 (page 3.3-111, row 1) of the Application.

For item (3), by the highlighting convention described in the response to item (2) above, ventilation dampers are highlighted and shown on the flow diagrams to be within the license renewal evaluation boundaries of the Diesel Building Ventilation System. While the ventilation dampers are within the scope of license renewal, ventilation dampers are not included in the aging management review results tables in the Application. Ventilation dampers, without sub-component exceptions, are explicitly excluded from an aging management review by §54.21(a)(1)(i) of the Rule. As an aid to the reviewer, the following excerpt of §54.21(a)(1)(i) is provided (underline added to highlight ventilation damper exclusion from aging management review):

> **10 CFR 54.21(a)(1)(i):**
> That perform an intended function, as described in §54.4, without moving parts or without a change in configuration or properties. These structures and components include, but are not limited to, the reactor vessel, the reactor coolant system pressure boundary, steam generators, the pressurizer, piping, pump casings, valve bodies, the core shroud, component supports, pressure retaining boundaries, heat exchangers, ventilation ducts, the containment, the containment liner, electrical and mechanical penetrations, equipment hatches, seismic Category I structures, electrical cables and connections, cable trays, and electrical cabinets, <u>excluding</u>, but not limited to, pumps

> (except casing), valves (except body), motors, diesel generators, air compressors, snubbers, the control rod drive, ventilation dampers, pressure transmitters, pressure indicators, water level indicators, switchgears, cooling fans, transistors, batteries, breakers, relays, switches, power inverters, circuit boards, battery chargers, and power supplies; and

The valve bodies listed in Table 3.3-13 (page 3.3-116, rows 5 through 9) of the Application for the Diesel Building Ventilation System are associated with in-scope instruments, which by convention, are not highlighted on mechanical system flow diagrams. Instruments and instrumentation components are within scope if they are attached to process pipe, ductwork or other components that are within scope.

For item (4), the pipe components listed in Table 3.3-13 (page 3.3-116, rows 2 and 3) of the Application for the Diesel Building Ventilation System are associated with in-scope instruments which by convention, are not highlighted on mechanical system flow diagrams.

For item (5), double LR flags should have been shown for the inlet ductwork on CN-1579-1. Ductwork for this section is contained in Table 3.3-13 (page 3.3-116, row 1) of the Application.

For item (6), by the highlighting convention described in the response to item (2) above, the filter units are highlighted and shown on the flow diagrams to be within the license renewal evaluation boundaries of the Fuel Handling Building Ventilation System. Filters consist of a housing and medium. The filter housing is listed in Table 3.3-28 (page 3.3-192, row 3) of the Application as "Filter." From the March 10, 2000 letter to Douglas J. Walters (NEI) from Christopher I. Grimes (NRC), filter mediums are excluded from an aging management review in that they are replaced on condition. The Fuel Handling Building Ventilation System filter mediums are periodically tested and replaced when test results warrant. Therefore, filter mediums are excluded from an aging management review.

For item (7), by the highlighting convention described in the response to item (2) above, dampers are highlighted and shown on the flow diagrams to be within the license renewal evaluation boundaries of the Fuel Handling Building Ventilation System. While the ventilation dampers are within the scope of license renewal, ventilation dampers are not included in the aging management review results tables in the Application. Ventilation dampers, without sub-component exceptions, are explicitly excluded from an aging management review by §54.21(a)(1)(i) of the Rule. Further details to aid to the reviewer are provided in the response to item (3) above.

The valve bodies listed in Table 3.3-28 (page 3.3-192, rows 7 through 9) of the Application for the Fuel Handling Building Ventilation System are associated with in-scope instruments which by convention, are not highlighted on mechanical system flow diagrams.

For item (8), by the highlighting convention described in the response to item (2) above, dampers are highlighted and shown on the flow diagrams to be within the license renewal evaluation boundaries of the Nuclear Service Water Pump Structure Ventilation System. While the ventilation dampers are within the scope of license renewal, ventilation dampers are not included in the aging management review results tables in the Application. Ventilation dampers, without sub-component exceptions, are explicitly excluded from an aging management review by §54.21(a)(1)(i) of the Rule. Further details to aid to the reviewer are provided in the response to item (3) above.

The valve bodies listed in Table 3.3-38 (page 3.3-229) of the Application for the Nuclear Service Water Pump Structure Ventilation System are associated with in-scope instruments which by convention, are not highlighted on mechanical system flow diagrams.

Staff Concern - RAI 2.3-8 (Open Item)

Refer to the staff's concern (as documented herein) about Duke's response to RAI 2.3-1 for items (3), (6), (7) and (8).

Duke Response to Staff Concern - RAI 2.3-8 (Open Item)

For Items (3), (7), and (8), please refer to Duke's Response to Staff Concern - RAI 2.3-1 (Open Item) for ventilation dampers.

For Item (6) – The staff's concern as documented in Staff Concern - RAI 2.3-1 (Open Item) does not address its concern with RAI 2.3-8, Item 6. The filters (filter medium and housing) are passive components not covered by the treatment of housings of active components. The filters of the Fuel Handling Building Ventilation System are within the scope of license renewal and require an aging management review. As noted in our initial response to RAI 2.3-8, the results of the aging management review for the filter housing are listed in Table 3.3-28 (page 3.3-102, row 3) of the Application. The filter mediums are excluded from an aging management review as they are replaced on condition which conforms to the staff's guidance on the treatment of consumables in the March 10, 2000 letter to Douglas J. Walters (NEI) from Christopher I. Grimes (NRC) and SRP-LR Table 2.1-3, "Specific Staff Guidance on Screening."

Duke's Response to RAI 2.3-9

Referring to the three components identified in RAI 2.3-8, for item (1), the refrigerant coils associated with the auxiliary shutdown panel room air-conditioning sub-system of the Catawba Auxiliary Building Ventilation System are within the scope of license renewal and should have been highlighted on flow diagram CN-1577-1.8. They are shown to be within the license renewal evaluation boundaries, as defined by the red, triangular LR flags. The coils are listed in Table 3.3-1 (page 3.3-8, rows 2 through 4 and page 3.3-9, rows 1 through 3) of the Application as "Shutdown Panel Area Air-Conditioning Unit Condenser (CNS Only)," with tubes, tube sheets, shells and bonnets listed separately, along with the associated aging management review results.

For item (2), cooling fans are not included in the aging management review results tables in the Application. Cooling fans, without sub-component exceptions, are explicitly excluded from an aging management review by §54.21(a)(1)(i) of the Rule. As an aid to the reviewer, the following excerpt of §54.21(a)(1)(i) is provided (underline added to highlight cooling fan exclusion from aging management review):

> **10 CFR 54.21(a)(1)(i):**
> That perform an intended function, as described in §54.4, without moving parts or without a change in configuration or properties. These structures and components include, but are not limited to, the reactor vessel, the reactor coolant system pressure boundary, steam generators, the pressurizer, piping, pump casings, valve bodies, the core shroud, component supports, pressure retaining boundaries, heat exchangers, ventilation ducts, the containment, the containment liner, electrical and mechanical penetrations, equipment hatches, seismic Category I structures, electrical cables and connections, cable trays, and electrical cabinets, excluding, but not limited to, pumps (except casing), valves (except body), motors, diesel generators, air compressors, snubbers, the control rod drive, ventilation dampers, pressure transmitters, pressure indicators, water level indicators, switchgears, cooling fans, transistors, batteries, breakers, relays, switches, power inverters, circuit boards, battery chargers, and power supplies; and

For item (3), the Fuel Handling Building Ventilation System filters consist of housings and mediums. The "filter" entry in Table 3.3-28 (page 3.3-192, row 3) in the Application applies only to the filter housing in this case which does serve a pressure boundary function. From the March 10, 2000, letter to Douglas J. Walters (NEI) from Christopher I. Grimes (NRC), filter mediums are excluded from an aging management review in that they are replaced on condition. The Fuel Handling Building Ventilation System filter mediums are periodically tested and replaced when test results warrant. Therefore, filter mediums are excluded from an aging management review.

Staff Concern - RAI 2.3-9 (Open Item)

Refer to the staff's concern (as documented herein) about Duke's response to RAI 2.3-1 for item (2).

Duke Response to Staff Concern - RAI 2.3-9 (Open Item)

Please refer to Duke's Response to Staff Concern - RAI 2.3-1 (Open Item) for cooling fans.

Duke's Response to RAI 2.3.3.8-7

All valve components (actuators, operators, disks, stems, springs, etc.) except for valve bodies are excluded from aging management review in accordance with §54.21(a)(1)(i).

Staff Concern - RAI 2.3.3.8-7 (Confirmatory Item)

The response from Duke does not address the staff's concern. However, the applicant described the valve actuator in electronic correspondence on May 2, 2002 (ML021440229). The following description of the valve actuator was provided:

> The spring is piece/part of the actuator and not the valve itself. The spring is in a relaxed state and not compressed. In the event the valve stem attempts to reposition by some unknown force, the spring would compress slightly and then restore the valve in its initial position. Compression of the spring is a change of state. In addition, the flow through the valve itself tends to keep the valve open. In the unlikely event, the spring fails and the valve stem repositions, there is no impact on pressure boundary function of the system components. All of the system is highlighted as being within the evaluation boundary.

The staff requests that the applicant provide this information in an official letter to the staff so that the information therein can be used to support a reasonable assurance finding on this issue.

Duke Response to Staff Concern - RAI 2.3.3.8-7 (Confirmatory Item)

The following information as provided by Duke in an electronic communication dated May 2, 2002 addresses the staff concern:

> "The spring is piece/part of the actuator and not the valve itself. The spring is in a relaxed state and not compressed. In the event the valve stem attempts to reposition by some unknown force, the spring would compress slightly and then restore the valve in its initial position. Compression of the spring is a change of state. In addition, the flow through the valve itself tends to keep the valve open. In the unlikely event, the spring fails and the valve stem repositions, there is no impact on pressure boundary function of the system components. All of the system is highlighted as being within the evaluation boundary."

Duke's Response to RAIs 2.3.3.19-1 through 2.3.3.19-10: Background Discussion Pertaining to Potential Open Items in Section 2.3.3.19

2.3.3.19 Fire Protection System

Note: The following background information is provided prior to providing responses to RAIs 2.3.3.19-1 through 2.3.3.19-10 to facilitate the staff's understanding of the 10 CFR 50.48 fire protection programs at McGuire and Catawba.

Background Information

The systems, structures, and components (SSCs) within the scope of license renewal for compliance with §50.48 are those SSCs that protect safety-related SSCs so that a fire will not prevent the performance of necessary safe plant shutdown functions and will not significantly increase the risk of radioactive releases. The following discussion is provided to explain that the focus of SSCs relied on to comply with §50.48 directly relates to the ability to safely shut down the plant and minimize radioactive releases in the event of a fire. This discussion offers information relevant to the Commission's regulations on license renewal and fire protection, the staff's guidance related to these regulations, and Duke's plant-specific licensing documentation and technical evaluations related to §50.48.

The key to understanding the SSCs within the scope of license renewal for fire protection begins with the Commission's regulations. The license renewal scoping requirement in 10 CFR 54.4(a)(3) states:

> **10 CFR 54.4**
> (a) Plant systems, structures, and components within the scope of this part are - ...
> (3) All systems, structures, and components relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with the Commission's regulations for fire protection (10 CFR 50.48)... .

Compliance with §50.48 is the key to determining the plant SSCs relied on to perform fire protection functions. Compliance with §50.48 begins with the regulation itself, which states (underline added for emphasis):

> **10 CFR 50.48**
> (a)(1) Each operating nuclear power plant must have a fire protection plan that satisfies Criterion 3 of appendix A of this part.... (2) The plan must also describe specific features necessary to implement the program described in paragraph (a)(1) of this section such as...(iii) the means to limit fire damage to structures,

systems, or components important to safety so that the capability to safely shut down the plant is ensured.

Appendix A of 10 CFR 50 provides the General Design Criteria for Nuclear Power Plants. Criterion 3 states:

> **10 CFR 50 Appendix A, General Design Criterion (GDC) 3**
> Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions.... Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety.

As described in §50.48 and quoted above, "structures, systems, and components important to safety" is clarified as those structures, systems, and components relied on so that the capability to safely shut down the plant is ensured. Based on the above quotations, the regulations clearly focus on a fire protection plan or program with the ability to limit fire damage to SSCs important to safety so that the capability to safely shut down the plant is ensured.

Several NRC-issued guidance documents help interpret the requirements of §50.48. NUREG-0800, "Standard Review Plan of Safety Analysis Reports for Nuclear Power Plants," provides guidance of fire protection program requirements to staff reviewers. Branch Technical Position CMEB 9.5-1 and its predecessor, Appendix A to Branch Technical Position APCSB 9.5-1, provide guidance acceptable to the staff for implementing a fire protection program in accordance with §50.48 and GDC 3. During original licensing, Catawba was reviewed against the guidelines of Branch Technical Position CMEB 9.5-1 and NUREG-0800. Although McGuire is licensed to Appendix A of Branch Technical position APCSB 9.5-1 and not specifically to Branch Technical Position CMEB 9.5-1 or NUREG-0800, Branch Technical Position CMEB 9.5-1 and NUREG-0800 provide guidance for reviewing a plant's compliance with regulations, and in turn provide insights into interpretations of those regulations.

The purpose of the fire protection plan mentioned in the first sentence of §50.48(a) is provided in NUREG-0800, Section 9.5.1, Fire Protection Program, which states (underline added for emphasis):

NUREG-0800, Section 9.5.1, Fire Protection Program
I. Areas of Review
The purpose of the fire protection program (FPP) is to provide assurance, through a defense-in-depth design, that a fire will not prevent the performance of necessary safe plant shutdown functions and will not significantly increase the risk of radioactive releases to the environment in accordance with General Design Criteria 3 and 5.

The implementation of GDC 3, for the purposes of §50.48, is explained further in NUREG-0800, Section 9.5.1, as follows (underlines added for emphasis):

NUREG-0800, Section 9.5.1, Fire Protection Program
II. Acceptance Criteria
The applicant's fire protection program is acceptable if it is in accordance with the following criteria:

1. 10 CFR Part 50 §50.48, and General Design Criterion 3, as related to fire prevention, the design and operation of fire detection and protection systems, and administrative controls provided to protect safety-related structures, systems, and components of the reactor facility.

...

The following specific criteria provide information, recommendations, and guidance and in general describe a basis acceptable to the staff that may be used to meet the requirements of §50.48, GDC 3 and 5:

a. Branch Technical Position (BTP) CMEB 9.5-1 as it relates to the design provisions given to implement the fire protection program.

The staff provided even more detailed guidance relevant to the implementation of a fire protection program in accordance with §50.48 and GDC 3 in the Branch Technical Position itself. The Branch Technical Position begins with the following statement:

Branch Technical Position CMEB 9.5-1
A. Introduction
This BTP addresses protection programs for safety-related systems and equipment and for other plant areas containing fire hazards that could adversely affect safety-related systems. It does not give guidance for protecting the life or safety of the site personnel or for protection against economic or property loss.

The staff's guidance documents clearly focus on a fire protection program with the ability to limit fire damage to safety-related SSCs so that a fire will not prevent the performance of necessary safe plant shutdown functions and will not significantly increase the risk of radioactive releases.

The earlier quote from the Acceptance Criteria section of NUREG-0800 indicates that implementing the guidelines of BTP CMEB 9.5-1 (attached to Section 9.5.1 of the SRP) is acceptable to the staff in meeting the requirements of §50.48 and GDC 3. This acceptance would be reflected in the specific plant's safety evaluation report (SER) based on a review of the plant-specific responses to the BTP. As documented in the respective McGuire and Catawba SERs, the staff found the fire protection programs acceptable based on the plant-specific BTP responses.

BTP CMEB 9.5-1 provides guidelines that can accommodate a full range of possible plant designs and layouts. Not all of these guidelines are applicable to all plants. One example of this is BTP Section C.7.q. which relates to cooling towers. It is obvious that not all plants have cooling towers.

Just as the cooling tower guidelines are not applicable to all plants, the general plant-wide design features discussed throughout the BTP are applicable only within the context of §50.48 requirements. In other words, the BTP guidelines are applicable as they relate to protecting safety-related SSCs so that a fire will not prevent the performance of necessary safe plant shutdown functions and will not significantly increase the risk of radioactive releases. Examples of this focus can be found throughout the BTP with statements such as "within hose reach of areas containing equipment required for safe plant shutdown" (Section C.1.c.3), "Fixed self-contained lighting...should be provided in areas that must be manned for safe shutdown" (C.5.g.1), "Detection systems should be provided for all areas that contain or present a fire exposure to safety-related equipment" (C.6.a.1), "Outside manual hose installations should be...where fixed or transient combustibles could jeopardize safety-related equipment" (C.6.b.7), and "Miscellaneous areas...should be so located and protected that a fire...will not adversely affect any safety-related systems or equipment" (C.7.r.).

McGuire and Catawba nuclear power plants are large facilities on large sites with many areas and structures located such that a fire in those areas or structures would not affect safety-related SSCs or the plant's ability to safely shut down. The SSCs that protect these areas or structures from fire are beyond the requirements of §50.48. The plants obviously have fire protection features that are related to protecting the life or safety of the site personnel or for protection against economic or property loss. These features are not intended to be the focus of the guidance in the BTP, as stated in the BTP introduction and quoted above. McGuire and Catawba responded to all BTP items even though (as shown in the previous paragraphs) not all items applied to each plant. Likewise, some BTP responses were answered in relation to the overall site fire protection program when the areas of NRC concern (according to NUREG-0800) relate only to protecting safety-related SSCs so that a fire will not prevent the performance of necessary safe shutdown functions and will not significantly increase the risk of radioactive releases.

The basis of the SSCs within the scope of license renewal for compliance with §50.48 was built upon the plant-specific responses to the BTP with the focus of identifying a specific subset of the overall site fire protection program. This subset of the overall site fire protection program is those SSCs that protect safety-related SSCs so that a fire will not prevent the performance of necessary safe plant shutdown functions and will not significantly increase the risk of radioactive releases. Where technical justification can be made that a fire protection SSC merely mentioned in the plant-specific BTP response is not necessary to ensure that a fire will not prevent the performance of necessary safe plant shutdown functions and will not significantly increase the risk of radioactive releases, that SSC is not required for compliance with §50.48. The plant-specific BTP responses, with this focus, have been used as the basis of the responses to RAIs 2.3.3.19-1 through 2.3.3.19-10 that follow.

Staff Concern - Background Discussion Pertaining to Potential Open Items in Section 2.3.3.19 (no open item number is associated with the Background Information)

The staff considers the majority of RAIs unresolved (open) because the discussion in the background information section of Duke's response, which forms the basis of the responses to the RAIs, is not complete. In its discussion, Duke states that only protection of safety-related SSCs is required by 10 CFR 50.48. In its discussion, Duke also states that only the protection of SSCs important to safe shutdown is required. The staff is concerned that Duke has narrowly defined the scope and intent of 10 CFR 50.48. The staff believes that 10 CFR 50.48 includes the protection of safety-related and non-safety-related SSCs and mandates a defense-in-depth approach to prevent fires, promptly detect and suppress fires, and protect SSCs important to safety so that a fire will not prevent safe shutdown.

Duke Response to Staff Concern - Background Discussion Pertaining to Potential Open Items in Section 2.3.3.19

Note: This Duke Response to Staff Concern - Background Discussion Pertaining to Potential Open Items in Section 2.3.3.19 also pertains to Duke Responses to Staff Concerns 2.3.3.19-1, -3, -4, -8, -9(Open Items) and -6 (New Item).

Staff Concerns 2.3.3.19-1, -3, -4, -8, -9 (Open Items), and -6 (New Open Item) question the scoping methodology for fire protection that Duke used to prepare the Application. The process that Duke used to determine the scope of fire suppression systems is described in Section 2.1.1.3.1 of the Application and clearly states that the following staff Safety Evaluation Reports were used to determined those fire protection SSCs that are within the scope of license renewal:

- NUREG-0422, *Safety Evaluation Report Related to the Operation of McGuire Nuclear Station, Units 1 and 2,* March 1978, as supplemented, Docket Nos. 50-369 and 50-370.
- NUREG-0954, *Safety Evaluation Report Related to the Operation of Catawba Nuclear Station, Units 1 and 2,* February 1983, as supplemented, Docket Nos. 50-413 and 50-414.

Implementation of the guidelines contained in staff review documents is an acceptable way to meet the requirements of §50.48. Each Safety Evaluation Report contains the results of this staff review. Included in the staff review by reference in the Safety Evaluation Report is the applicant's fire protection review The fire protection review included a comparison of the applicant's program to Appendix A of the BTP and the applicant's fire hazards analysis. The staff reviewed the plant specific design features that provide the defense-in-depth approach to prevent fires, detect and suppress fires. The staff specifically reviewed water suppression systems during the initial licensing of each station.

For McGuire, Safety Evaluation Report Supplement 2, Appendix D, Section II, lists the "areas that have been equipped or will be equipped with water suppression systems." Section V of Appendix D lists the "fire protection requirements for specific areas." Supplement 5, Sections 2.1 and 5 provide later versions of the same information. By reading these sections of the McGuire Safety Evaluation Report, it is clear that portions of the fire suppression system identified in staff concerns 2.3.3.19-1, -3, -4, -8, -9 (Open Items), and -6 (New Open Item) are not required by §50.48 and therefore are not within the scope of license renewal.

For Catawba, Safety Evaluation Report Section 9.5.1.7 lists the "areas being equipped with automatic water suppression systems" and Section 9.5.1.8 describes the "fire protection for specific station areas." By reading these sections of the Catawba Safety Evaluation Reports, it is clear that the portions of the fire suppression system identified in staff concerns 2.3.3.19-1, -3, -4, -8, -9 (Open Items), and -6 (New Open Item) are not required by §50.48 and therefore are not within the scope of license renewal.

With respect to the license conditions for each unit, each license condition refers to <u>both</u> the Final Safety Analysis Report as well as the applicable Safety Evaluation Report as supplemented. Therefore, the distinction between those fire suppression SSCs that are required by §50.48 and those that are not required involves reviewing the applicable Safety Evaluation

Reports along with the Final Safety Analysis Reports. The staff concerns with fire protection scoping appear to be based primarily on reading a section of each station's Final Safety Analysis Reports, specifically Section 9.5.1.2.1 of the McGuire Final Safety Analysis Report and the Catawba Final Safety Analysis Report. These sections were intended to provide general system descriptions consistent with guidance provided in Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants." These Final Safety Analysis Report sections were not intended to identify those SSCs required for compliance with §50.48 and indeed cannot be used independent of the above described Safety Evaluation Report sections to complete the license renewal scoping step.

Duke's Response to RAI 2.3.3.19-1

Duke agrees with the staff that all SSCs relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with 10 CFR 50.48 are within the scope of license renewal. As referred to in the background information preceding this RAI response, the McGuire and Catawba nuclear power plants are large facilities on large sites with many areas and structures located such that a fire in those areas or structures would not affect safety-related SSCs or the plant's ability to safely shut down. The SSCs that protect these areas or structures from fire are beyond the requirements of §50.48.

Section 9.5.1 in the McGuire and Catawba UFSARs describes the overall site fire protection program and not just the portions required to meet the requirement of §50.48. As stated in the guidance for regulated event scoping in both NEI 95-10, Section 3.1.3, and NUREG-1800, Section 2.1.3.1.3, "Mere mention of a system, structure, or component in the analysis or evaluation does not constitute support of a specified regulatory function."

The structures and areas identified in McGuire UFSAR Section 9.5.1.2.2 and Catawba UFSAR Section 9.5.1.2.1 are beyond the requirements of §50.48. These structures and areas are addressed in the BTP responses dealing with General Guidelines for Plant Protection where it is stated that the plant layout is arranged to isolate safety-related systems from unacceptable fire hazards. This isolation includes such things as Building Design ("greater than 50 feet between oil-filled transformers and buildings containing safety-related equipment and fire barriers with a minimum fire rating of three hours separating fire areas") and Control of Combustibles ("safety-related systems are separated from combustible materials except when required for system operation"). In cases like the Turbine Building and adjacent Auxiliary Building, the buildings are separated by a three hour fire barrier and it is the fire barrier that is credited as the means of isolation, not the automatic sprinkler systems or other fire protection and detection features in the Turbine Building.

Therefore, these other structures and areas that are mentioned in UFSAR Section 9.5.1 where a fire would not affect safety-related SSCs or the plant's ability to safely shut down are beyond the requirements of §50.48 and are not within the scope of license renewal.

The plant-specific BTP responses have been used as the basis of this response. If still required by the staff to make its finding, the McGuire and Catawba fire hazards analyses and BTP responses can be made available for on-site inspection.

Staff Concern - RAI 2.3.3.19-1 (Open Item)

The staff is concerned that the applicant's use of the Quality Assurance (QA) Condition 3 designation may have revealed a limited scope of SSCs required to comply with the provisions of 10 CFR 50.48. The QA Condition 3 designation corresponds to a specification governing minimum acceptable requirements for design, procurement, receipt, installation, maintenance, repair, modification, inspection and testing of specific fire protection features at the McGuire and Catawba nuclear stations. However, the specification does not apply to other fire protection features, such as jockey pumps, that Duke committed to install to meet National Fire Protection Association (NFPA) recommendations. Therefore, the staff does not have confidence in the applicant's use of this QA designation to perform scoping activities for license renewal. Furthermore, Section 9.5.1 of the UFSAR does not differentiate between those components that are required by 10 CFR 50.48 and those that are not. As such, the staff has no basis for concluding that the components referenced in this RAI, and listed in the UFSAR, are not required by 10 CFR 50.48.

In its response, the applicant references the guidance for regulated event scoping in both NEI 95-10, Section 3.1.3, and NUREG-1800, Section 2.1.3.1.3: "Mere mention of a system, structure, or component in the analysis or evaluation does not constitute support of a specified regulatory function." The staff notes that this quote is taken out of context. The actual context is that the SSC is credited in a safety analysis or plant evaluation for performing a function that is required by the regulations identified in 10 CFR 54.4(a)(3). Therefore, a specific reference to these components in the UFSAR (which is based upon more detailed plant analyses and plant evaluations) appears to the staff to be intentional and meaningful. In fact, a license condition for McGuire and Catawba states that Duke Energy Corporation shall implement and maintain in effect all provisions of the approved fire protection program as described in the UFSAR for the respective facilities. The staff further notes, again, that the UFSAR does not indicate that certain SSCs listed and discussed therein are not required to comply with 10 CFR 50.48.

Duke Response to Staff Concern - RAI 2.3.3.19-1 (Open Item)

Please refer to the Duke Response to Staff Concern - Background Discussion Pertaining to Potential Open Items in Section 2.3.3.19 provided above. In light of the above stated concerns, Duke encourages re-review by the staff of the applicable staff Safety Evaluation Reports listed above, Duke believes that Staff Concern – RAI 2.3.3.19-1 (Open Item) can be Closed/Resolved following this re-review.

Duke's Response to RAI 2.3.3.19-3

The license conditions mentioned in the RAI address the overall site fire protection program. As referred to in the background information preceding this RAI response, the McGuire and Catawba nuclear power plants are large facilities on large sites with many areas and structures where a fire would not affect safety-related SSCs or the plant's ability to safely shut down. The hazards identified in this RAI are separated from safety-related areas by distance and three-hour fire barriers, and therefore the SSCs that protect these areas or structures from fire are beyond the requirements of §50.48. Section 9.5.1 in the McGuire and Catawba UFSARs describes the overall site fire protection program and not just the portions required to meet the requirement of §50.48. These existing license conditions will carry forward in the renewed license.

For details of why certain structures and areas are not within license renewal scope as discussed in this RAI, please refer to the response to RAI 2.3.3.19-1.

Staff Concern - RAI 2.3.3.19-3 (Open Item)

The staff's concern is similar to the concern with Duke's response to RAI 2.3.3.19-1. Since the UFSAR is referenced in the license conditions, and these components are discussed therein as providing a fire suppression function (which is required by 10 CFR 50.48), it appears that these components are required to meet the FP license condition as stated above. It addition, these components contain flammable liquids, which can be hazardous and can quickly escalate to generate high heat release rates and smoke.

Duke Response to Staff Concern - RAI 2.3.3.19-3 (Open Item)

Please refer to the Duke Response to Staff Concern - Background Discussion Pertaining to Potential Open Items in Section 2.3.3.19 provided above. In light of the above stated concerns, Duke encourages re-review by the staff of the applicable staff Safety Evaluation Reports listed above, Duke believes that Staff Concern – RAI 2.3.3.19-3 (Open Item) can be Closed/Resolved following this re-review.

Duke's Response to RAI 2.3.3.19-4

As stated in the Background Information preceding this RAI response, the general plant-wide design features discussed throughout the BTP are applicable (within the context of §50.48 requirements) only as they relate to protecting safety-related SSCs so that a fire will not prevent the performance of necessary safe plant shutdown functions and will not significantly increase the risk of radioactive releases. Safety-related structures and areas at McGuire and Catawba are isolated from other plant structures and areas such that a fire in these other structures and areas will not prevent the performance of necessary safe plant shutdown functions.

This isolation includes such features as Building Design (greater than 50 feet between oil-filled transformers and buildings containing safety-related equipment and fire barriers with a minimum fire rating of three hours separating fire areas) and Control of Combustibles (safety-related systems are separated from combustible materials except when required for system operation). Safety-related structures are isolated from adjacent nonsafety-related structures and fire areas by a three hour fire barrier and it is the fire barrier that is credited as the means of isolation, not the manual fire suppression equipment in the yard.

With the exception of two hydrants at Catawba that protect the Nuclear Service Water Pump Structure, hydrants in the yard are not relied upon to protect safety-related SSCs required for safe shutdown. As stated in the RAI, some hydrants are located along the required flow path and are not isolatable from the required flow path. These hydrants that cannot be isolated are within license renewal scope. The other hydrants are not in scope because they are not relied on for fire suppression of safety-related SSCs to ensure safe shutdown and are isolable from the required flow path (via being downstream of isolation valves). Upon failure of these downstream hydrants, or the associated downstream piping, the isolation valves can be used to isolate them from the portions of the system that protect safety-related SSCs to ensure safe shutdown. These isolable, downstream hydrants and piping are beyond the requirements of §50.48 and are not within the scope of license renewal. The license renewal evaluation boundary is at the isolation valves since they serve as the isolation point between the §50.48 and the non-§50.48 portions of the system.

Staff Concern - RAI 2.3.3.19-4 (Open Item)

McGuire UFSAR Section 9.5.1.2.1 identifies that hydrants are connected to the yard main. Furthermore, fire hydrants are considered passive and long-lived components in accordance with 10 CFR 54.21. Since the UFSAR is referenced in the license conditions, and these components are discussed therein as providing a fire suppression function (which is required by 10 CFR 50.48), it appears that these components are required to meet the FP license condition as stated above. Again, the UFSAR does not distinguish between those fire hydrants that are required by 10 CFR 50.48 and those that are not.

Duke Response to Staff Concern - RAI 2.3.3.19-4 (Open Item)

Please refer to the Duke Response to Staff Concern - Background Discussion Pertaining to Potential Open Items in Section 2.3.3.19 provided above. In light of the above stated concerns, Duke encourages re-review by the staff of the applicable staff Safety Evaluation Reports listed above, Duke believes that Staff Concern – RAI 2.3.3.19-4 (Open Item) can be Closed/Resolved following this re-review.

Duke's Response to RAI 2.3.3.19-5

The fire pumps and associated strainers are within the scope of license renewal. The red, triangular LR flags define the license renewal evaluation boundaries on mechanical system flow diagrams, and highlighting was used as an aid to Duke in component screening and for the reviewer in understanding the system under review. In some cases, components were outlined in highlighting, and in others, the highlighting was simply drawn through components. Either way is acceptable for achieving the purpose of the drawings. The components are shown to be within the license renewal evaluation boundaries, and therefore, within the scope of license renewal.

Although the flow diagram makes it appear that the strainer is a stand-alone component, the strainer is actually a sub-component of the pump installed in the pump bowl, does not contain any pressure retaining parts and is inspected and maintained along with the other non-pressure retaining pump sub-components. As the strainer is a sub-component of the pump and pumps (except casing) are excluded from aging management review per §54.21(i), the strainer is not subject to aging management review. The pump casings are subject to aging management review and are listed in Table 3.3.26 (page 3.3-172, row 1) of the Application for McGuire.

Staff Concern - RAI 2.3.3.19-5 (Open Item)

The staff agrees with the applicant that the strainers perform an intended function that meets one of the scoping criteria (specifically 10 CFR 54.4(a)(3)). The staff's technical concern is that Duke uses lake water to supply their fire protection suppression systems at McGuire and Catawba. Lake water is corrosive and may contain sediment, which can potentially clog the fire pumps. In addition, the strainers keep debris from plugging the sprinkler nozzles in fire suppression systems in the event that sprinklers are actuated. This FP component should be managed in an AMP. However, the staff is concerned that the strainers were inappropriately screened out. Although the strainers may be in-line with and connected to the main fire pump, their function is passive (as is the pump casing's function). The applicant included the pump casing within the scope of license renewal; the strainer also should be within scope.

Duke Response to Staff Concern - RAI 2.3.3.19-5 (Open Item)

Duke has decided to defer the response to Staff Concern RAI 2.3.3.19-5 (Open Item) until after the staff issues the SER with Open Items to provide sufficient time for responsible engineering staff at each station to be involved in the preparation of the response.

Duke's Response to RAI 2.3.3.19-6

This RAI requests justification for (1) the exclusion of the jockey pumps; and (2) the appropriateness of the methodology used to identify FP systems and components that are within the scope of license renewal based solely upon their QA Condition 3 designation (or lack thereof). The following are the responses to each of these two items.

(1) As stated in the background information preceding this RAI response, the general plant-wide design features discussed throughout the BTP are applicable (within the context of §50.48 requirements) only as they relate to protecting safety-related SSCs so that a fire will not prevent the performance of necessary safe plant shutdown functions and will not significantly increase the risk of radioactive releases. Section 9.5.1 in the McGuire and Catawba UFSARs describes the overall site fire protection program and not just the portions required to meet the requirement of §50.48. As stated in Section 9.5.1 of the UFSARs, the function of the jockey pumps is to prevent frequent starting of the fire pumps by maintaining pressure in the yard mains. In this capacity, the jockey pumps and associated components act as a support system feature that refills the suppression system during standby mode when the system has lost water due to normal system "leakage." The jockey pumps and associated components do not provide a function that protects safety-related SSCs so that a fire will not prevent the performance of necessary safe plant

shutdown functions and will not significantly increase the risk of radioactive releases. Once there is more than normal system "leakage" (as would be caused by system use during a fire), the fire pumps are the components relied on for protecting safety-related SSCs so that a fire will not prevent the performance of necessary safe plant shutdown functions and will not significantly increase the risk of radioactive releases.

The jockey pumps and associated components (a) provide only a support function and not an intended function, (b) are not relied on for fire suppression of safety-related SSCs, and (c) are isolatable from the required flow path via isolation valves. The jockey pumps and associated components support the establishment of the initial condition of the main fire suppression system prior to the initiation of a fire. Upon failure of the jockey pumps or associated components the isolation valves can be used to isolate them from the portions of the fire suppression system that protect safety-related SSCs. Failure of the jockey pumps and associated components does not result in a loss of the fire suppression system function. For all of the above reasons, Duke concludes that the jockey pumps and associated components are beyond the requirements of §50.48 and are not within the scope of license renewal.

(2) The McGuire and Catawba Quality Assurance (QA) Condition 3 program was built upon the plant-specific responses to the BTP with the focus of identifying the subset of the overall site fire protection program SSCs that protect safety-related SSCs so that a fire will not prevent the performance of necessary safe plant shutdown functions and will not significantly increase the risk of radioactive releases. Based on the information presented in the Background Information preceding this RAI response, the QA 3 designation meets the requirements of §50.48. It was for this reason that QA 3 boundaries were used to designate those SSCs within the scope of license renewal for compliance with §50.48.

Staff Concern - RAI 2.3.3.19-6 (Open Item)

As stated in the initial RAI, operating license conditions for McGuire and Catawba, as well as Supplement 2 of the McGuire and Catawba Safety Evaluation Reports (SERs) for original licensing and Section 9.5.1.2.1 of the McGuire and Catawba UFSARs, indicate that jockey pumps are provided to prevent frequent starting of the fire pumps by maintaining pressure in the yard mains in accordance with Section 6.b of BTP CMEB 9.5-1and NFPA 20. The staff is concerned that the applicant has misapplied the QA Condition 3 designation for license renewal scoping purposes and excluded components (e.g., the jockey pump) from the scope of license renewal although the licensing basis of the plants indicates that these components (jockey pumps) are relied upon to perform a function required by 10 CFR 50.48.

Duke Response to Staff Concern - RAI 2.3.3.19-6 (Open Item)

Duke has decided to defer the response to Staff Concern – RAI 2.3.3.19-6 (Open Item) until after the staff issues the SER with Open Items to provide sufficient time for responsible engineering staff at each station to be involved in the preparation of the response.

Duke's Response to RAI 2.3.3.19-8

As stated in the Background Information preceding this RAI response, the general plant-wide design features discussed throughout the BTP are applicable (within the context of §50.48 requirements) only as they relate to protecting safety-related SSCs so that a fire will not prevent the performance of necessary safe plant shutdown functions and will not significantly increase the risk of radioactive releases. Safety-related structures and areas at McGuire and Catawba are isolated from other plant structures and areas such that a fire in these other structures and areas will not prevent the performance of necessary safe plant shutdown functions.

The Turbine Building is a nonsafety-related structure. The Turbine Building and adjacent Auxiliary Building are separated by a three hour fire barrier and it is the fire barrier that is credited in the BTP response and the FHA as demonstrating compliance with 10 CFR 50.48, not the automatic sprinkler systems, manual hose stations or other fire protection and detection features installed in the Turbine Building.

Staff Concern - RAI 2.3.3.19-8 (Open Item)

A license condition for McGuire and Catawba states that Duke Energy Corporation shall implement and maintain in effect all provisions of the approved fire protection program as described in the UFSAR for the respective facilities. Section 9.5.1.2.1 of the UFSAR states that manual hose stations and automatic sprinkler or deluge systems are provided for the protection of turbine building components. The UFSAR does not differentiate between those manual hose station and automatic sprinklers that are required to comply with 10 CFR 50.48 and those that are not. Additionally, the regulations governing fire protection apply to more than the protection of structures and equipment relied upon for safe plant shutdown.

Duke Response to Staff Concern - RAI 2.3.3.19-8 (Open Item)
Please refer to the Duke Response to Staff Concern - Background Discussion Pertaining to Potential Open Items in Section 2.3.3.19 provided above.

In addition, the staff should identify in the SER with Open Items specific regulatory citations that support its statement that:

> "regulations governing fire protection apply to more than the protection of structures and equipment relied upon for safe plant shutdown."

This statement overstates the scope of fire protection as required by §50.48, the licensing basis of McGuire and Catawba and seems inconsistent with the staff's statement contained in Staff Concern 2.3.3.19.

In light of the above stated concerns, Duke encourages re-review by the staff of the applicable staff Safety Evaluation Reports listed above, Duke believes that Staff Concern – RAI 2.3.3.19-8 (Open Item) can be Closed/Resolved following this re-review.

Duke's Response to RAI 2.3.3.19-9

The fire protection design features for the subject filters are mentioned in response to the BTP. This portion of the fixed water suppression is not related to protecting safety-related SSCs so that a fire will not prevent the performance of necessary safe plant shutdown functions and will not significantly increase the risk of radioactive releases. As stated in the background information preceding this RAI response, the general plant-wide design features discussed throughout the BTP are applicable (within the context of §50.48 requirements) only as they relate to protecting safety-related SSCs so that a fire will not prevent the performance of necessary safe plant shutdown functions and will not significantly increase the risk of radioactive releases. Additionally, as stated in NEI 95-10, Section 3.1.3, "SSCs Relied on to Demonstrate Compliance With Certain Specific Commission Regulations," (and in NUREG-1800, Section 2.1.3.1.3) "Mere mention of a system, structure, or component in the analysis or evaluation does not constitute support of a specified regulatory function."

The subject filters are not charcoal filters, but are high-purity carbon filters. The carbon used in these filter beds has an ignition temperature of approximately 330 °C. Since the air temperature in the process flowpath of this filter is not designed to reach temperatures this high, the carbon filters are not combustible in the environment for which they are designed to operate. The fixed water suppression systems provided for these carbon filters are similar to those provided for the reactor coolant pumps discussed in RAI 2.3.3.19-2. The need for a fixed water suppression system has been precluded by the use of the bed filter with an essentially noncombustible material. The fixed water suppression system for these filters is beyond the requirements of §50.48 and, therefore, not within the scope of license renewal.

Catawba flow diagrams and other in-house documents refer to these filters as charcoal filters. A corrective action report has been entered into the corrective action program to identify and evaluate changes to the in-house design documents to properly identify the filter beds as carbon filters and to update them in the future as needed.

Staff concern - RAI 2.3.3.19-9 (Open Item)

Section 9.5.1.2.1 of the Catawba UFSAR states that the RF system provides a fixed water suppression system for charcoal filters. On pages 48-50 of Duke's revised response to Appendix A to BTP APCSB 9.5-1, submitted to the NRC by letter dated November 4, 1983, Duke stated that lower containment carbon filters are provided with fire suppression capability. According to NRC Inspection Report 50-369/02-05, 50-370/02-05, 50-413/02-05 and 50-414/02-05 (ML021280003), this is also documented in Specification CNS-1465.00-00-0006. The staff does not believe that the applicant's distinction between charcoal and carbon filters is material.

Duke Response to Staff concern - RAI 2.3.3.19-9 (Open Item)

One of the fundamental elements of a fire protection program is to design the plant so as to prevent fires and one way to do this is to reduce the amount of combustible materials in the plant. The original plant design documents included charcoal filters, which are indeed combustible materials. The initial Safety Evaluation Reports for each station included the requirement for fire suppression features for these filters. However, subsequently, carbon filters were installed in the plant, which are indeed non-combustible materials for the conditions found in the plant. Catawba Specification CNS-1465.00-00-0006 is in error and is being revised. As stated in Duke's RAI response, a corrective action report has been generated to revise in-house documents that refer to the filters as charcoal, including CNS-1465.00-00-0006.

SER Open Item 2.3.3.19-6 (New Open Item)

A license condition for McGuire and Catawba states that Duke Energy Corporation shall implement and maintain in effect all provisions of the approved fire protection program as described in the UFSAR for the respective facilities. Section 9.5.1.2.3, "Fire Protection, Category I Safety Related," of the McGuire UFSAR states that the manually operated water spray systems provide fixed spray patterns of water for Reactor Building Purge Exhaust Filters 1A, 1B, 2A and 2B. However, drawing MCFD 1599-02.01, coordinates H-3, G-3, C-5 and B-7, indicates that piping and sprinklers associated with this function are also excluded from scope.

The staff is concerned that the manually operated water spray systems for these filters were inappropriately excluded from the scope of license renewal and an aging management review.

Duke Response to SER Open Item 2.3.3.19-6 (New Open Item)

The New Open Item as currently written has not been related to the requirements of §50.48.

Please refer to the Duke Response to Staff Concern - Background Discussion Pertaining to Potential Open Items in Section 2.3.3.19 provided above. In light of the above stated concerns, Duke encourages re-review by the staff of the applicable staff Safety Evaluation Reports listed above, Duke believes that Staff Concern – RAI 2.3.3.19-6 (New Open Item) can be Closed/Resolved following this re-review.

Duke's Response to RAIs 2.5-1 and 2.5-2

Duke performed an initial review of the McGuire and Catawba station blackout (SBO) safety analyses and plant evaluations prior to submittal of the Application. Based on RAI 2.5-1 and RAI 2.5-2, along with the recent industry discussions, Duke re-reviewed the plant documents with emphasis on equipment related to the recovery of offsite power.

Based on the results of this recent review, Duke has decided that the McGuire and Catawba components that are part of the power path for offsite power from the switchyard are within the scope of license renewal in accordance with the SBO scoping criterion, §54.4(a)(3). This power path includes portions of the power path from the unit power circuit breakers (PCBs) in the respective switchyards to the safety-related buses in each plant. The power path includes portions of (1) the switchyard systems, (2) the Unit Main Power System, and (3) the Nonsegregated-Phase bus in the 6.9 kV Normal Auxiliary Power System of each station.

An aging management will be performed on the passive, long-lived structures and components associated with this offsite power path. The results of this aging management review will be submitted on or before June 30, 2002.

Staff Concern - RAI 2.5-1 (and RAI 2.5-2)

The staff is not concerned with Duke's response to RAIs 2.5-1 and 2.5-2. However, pending the staff's receipt of the aging management review results for the passive, long-lived structures and components associated with this offsite power path, this item is characterized as open.

Duke Response to Staff Concern - RAI 2.5-1 (and RAI 2.5-2)

Duke letter dated June 26, 2002 provided the aging management review results for structures and components relied upon to restore power from offsite sources following station blackout.

Duke's Response to RAI 3.1.1-1

The Chemistry Control Program maintains the environment in the Reactor Coolant System by controlling contaminants that lead to loss of material and cracking. A review of the operating experience has not identified any failures of Reactor Coolant System components, including these orifices [the subject of the RAI], due to inadequate chemistry control. This operating experience shows that the Chemistry Control Program is effective in managing loss of material and cracking; therefore supplemental activities are not necessary.

Staff Concern - RAI 3.1.1-1 (Open Item)

The staff is concerned that the Chemistry Control Program is not sufficient to ensure that the loss of material and cracking of ASME Code Class 1 components of cast austenitic stainless steel (CASS) are being effectively controlled. The applicant has credited the Chemistry Control program and inservice inspection for other Class 1 components, but failed to do so for CASS orifices, valve bodies/bonnets and thermal barrier heat exchanger tubing of the same material and in the same environment.

Duke Response to Staff Concern - RAI 3.1.1-1 (Open Item)

RAI 3.1.1-1 focused on how aging effects associated with the Reactor Coolant System piping class break orifices are adequately managed by the *Chemistry Control Program.* As an inline component, these orifices provide a pressure boundary function and in the case of a pipe break in the non-Class 1 portion of the system, they provide a throttling function for the Class 1 portion of the system. These orifices are listed in the Application on page 3.1-7, row 3.

RAI 3.1.1-1 went on to request a description of supplemental activities which verify that the *Chemistry Control Program* is effective (and implied will remain effective in the period of extended operation). In preparing a follow-up response to this item, Duke reviewed the design specifications for these components and has discovered that these "orifices" are not flanged orifices as originally interpreted from the flow diagrams. The orifices that are relied on to make the class break from Class 1 to non-Class 1 are special piping components similar to a half coupling. These components are actually pipe fittings and so are more appropriately covered under the line item for pipe in the Application. Therefore, Duke supplements Table 3.1-1 in the Application by deleting the line item "Orifices" on page 3.1-7, row 3 and includes these piping components in the line entry entitled "Pipe and Fittings NPS$\leq$ 1" on page 3.1-6, row 5 and notes that the *Chemistry Control Program* along with the *Inservice Inspection Plan* manages the aging of these piping components, addressing the concern raised by this potential open item.

In Staff Concern – RAI 3.1.1.1-1 (Open Item), the staff has identified two additional stainless steel Class 1 components where only the *Chemistry Control Program* is credited for managing aging. These components are forged valve bodies/bonnets and thermal barrier heat exchanger piping.

The primary method for managing aging of forged valve bodies/bonnets is the *Chemistry Control Program*. In addition, these components are visually inspected by the *Inservice Inspection Plan*. These inspections provide additional assurance that the *Chemistry Control Program* will continue to manage the effects of aging on the forged valve bodies/bonnets for the period of extended operation. Therefore, Duke supplements Table 3.1-1 in the Application by modifying the line item "Forged Stainless Steel Valve bodies and/or Bonnets" on page 3.1-7, row 4 to add the *Inservice Inspection Plan* as an aging management program that manages the aging of these components, addressing the concern raised by this potential open item.

A second component identified in Staff Concern – RAI 3.1.1.1-1 (Open Item) is the thermal barrier heat exchanger piping. The thermal barrier heat exchanger piping is an internal part of the reactor coolant pump. This piping has component cooling water flowing internal to the piping and its external surfaces are exposed to reactor coolant water. This piping is located in the lower portion of the reactor coolant pump and by design is inaccessible for inspection. The primary method for managing aging of this heat exchanger piping is the *Chemistry Control Program*.

Because this piping is inaccessible, several design features exist that will provide an indication of leakage from this piping and will provide input to the *Reactor Coolant System Operational Leakage Monitoring Program* in order to manage aging of the thermal barrier heat exchanger piping. Should a leak occur in the thermal barrier heat exchanger piping, leakage from the higher pressure Reactor Coolant System to the lower pressure Component Cooling System would be detected by several means. Leakage will be detected by flow instrumentation located in the Component Cooling System downstream of each thermal barrier heat exchanger. In the event of a large leak the flow instrumentation will isolate the affected thermal barrier. Leakage can also be indicated by off-line gamma detectors located downstream of each component cooling heat exchanger as well as through level indication in each component cooling surge tank. Details of these design features are described further in McGuire UFSAR Sections 5.5 and 7.4.1.3.1.4 and Catawba UFSAR Sections 5.2.5.2.2 and 5.4.1.2.

Any indication from these design features of leakage from the Reactor Coolant System into the Component Cooling System provides input to the *Reactor Coolant System Operational Leakage Monitoring Program* described in the Application in Section B.3.25. This program provides additional assurance that the *Chemistry Control Program* will continue to manage the effects of aging on the thermal barrier heat exchanger piping for the period of extended operation. Therefore, Duke supplements Table 3.1-1 in the Application by modifying the line item "Thermal Barrier Heat Exchanger piping (tubing) and flanges" on page 3.1-8, row 3 to add the *Reactor Coolant System Operational Leakage Monitoring Program* as an aging management program that manages the aging of the thermal barrier heat exchanger piping, addressing the concern raised by this potential open item.

For completeness, in addition to the staff concerns raised in Staff Concern – RAI 3.1.1.1-1 (Open Item), Duke has identified a third stainless steel Class 1 component where only the *Chemistry Control Program* is credited for managing aging. This component is the pressurizer immersion heaters sheath. This item as listed in the Application is actually the wetted subcomponent of the pressurizer immersion heater assembly. The primary method for managing aging of this wetted portion of the pressurizer immersion heaters is the *Chemistry Control Program*. In addition, the assembly connection in the lower head of the pressurizer also receives a visual examination under the *Inservice Inspection Plan*. This examination provides additional assurance that the *Chemistry Control Program* will continue to manage the effects of aging on the pressurizer immersion heaters sheath for the period of extended operation. Therefore, Duke supplements Table 3.1-1 in the Application by modifying the line item "Immersion Heaters Sheath" on page 3.1-9, row 3 to add the *Inservice Inspection Plan* as an aging management program that manages the aging of these components, addressing the concern raised by this potential open item.

SER Open Item 3.1.4-1 (New Open Item)

For the Oconee nuclear station, Duke proposed to use an analysis of RV internals made from CASS or martensitic stainless steel as the basis for inspecting these components for cracks. The applicant's analysis was to be performed to calculate the critical crack size for the components under service loading conditions and service-degraded material properties (i.e., loss of fracture toughness) and to determine the type of NDE needed to detect cracks in the components prior to fast fracture to failure. The applicant proposed to inspect the limiting CASS or martensitic stainless steel component at Oconee Unit 3 if the bounding analysis determined the examination was warranted. The staff approved this program in NUREG-1723 for the Oconee LRA. The applicant's (Duke's) program and basis for inspecting the CASS RV internals materials at McGuire and Catawba is consistent with the corresponding program approved by the staff in NUREG-1723, and therefore acceptable.

However, for the remaining RV internal plates, forgings, welds and bolts (i.e., core barrel bolts and thermal shield bolts), the applicant has proposed to use examinations performed at Oconee Unit 1 and McGuire Unit 1 as the basis for determining whether additional, corresponding examinations need to be scheduled and performed at McGuire Unit 2 and Catawba Units 1 and 2. In contrast, Duke proposed to schedule corresponding inspections for these components at all three Oconee units, with the remaining RV internal plates, forgings, welds and bolts for one unit being scheduled for inspection early on in the license renewal period, for a second unit being scheduled for inspection near the middle of the license renewal period, and for the last unit being scheduled for inspection prior to the last year of the extended operating period. This program was accepted by the staff in as approved by the staff in NUREG-1723. A similar inspection program called for inspection of these components in both Turkey Point nuclear units (Units 3 and 4) and was approved by the staff in NUREG-1759. The applicant's program inspecting the remaining RV internals (i.e., forgings, plates, and welds made from austenitic stainless steel or nickel-based alloys, and bolts other than the baffle bolts) at the McGuire and Catawba nuclear stations is not consistent with corresponding inspection programs for these materials approved by the staff for the Oconee LRA (i.e., in NUREG-1723) or by the staff for the Turkey Point LRA (i.e., in NUREG-1759). Based on these precedents, the applicant needs to justify its basis for concluding that the inspection results of the corresponding RV internals at Oconee Unit 1 and McGuire Unit 1 will provide an acceptable basis for determining whether or not to schedule inspections of the corresponding RV internals at McGuire Unit 2 and Catawba Units 1 and 2.

Duke Response to SER Open Item 3.1.4-1 (New Open Item)

In its response to RAI B.3.27-1, Duke provided information comparing the materials, operating temperatures, and estimated peak fluences at baffle plate and bolt location for the reactor vessel internals of Oconee 1, McGuire 1, McGuire 2, Catawba 1, and Catawba 2. The inclusion of Oconee 1 in this comparison was due to the fact that Oconee 1 will be inspected prior to any of the McGuire and Catawba units entering the period of extended operation and will provide a leading indication for these other units. Based on the current information, McGuire 1 has been determined to be the most susceptible to the aging effects identified for the reactor vessel internals of the McGuire and Catawba units. This determination has been made using considerations consistent with Regulatory Guide 1.188 and NEI 95-10, Section 4.3, which provides guidance for using a samples or leading indicators for new inspections.

As stated in the description of this inspection in Appendix B.3.27 of the Application, the decision to perform inspections on McGuire Unit 2, Catawba Unit 1, and Catawba Unit 2 and when to perform such inspections will depend on an evaluation of the results of the internals inspections performed at Oconee and on McGuire Unit 1. In other words, the results of these two committed inspections will determine the need for inspections on the other three internals.

In addition, Duke stated in its program description that inspections performed at other nuclear plants will provide insights prior to McGuire and Catawba entering their respective period of extended operation. Currently, many U.S. nuclear plants have committed to perform reactor vessel internals inspections prior to the McGuire 1 planned inspection date which will establish a useful, comparative operating experience. If future industry developments suggest the need for an alternate inspection plan during the period of extended operation, or negate the need for an inspection, Duke will modify the proposed inspection plan. Note that Duke has committed to inspect the reactor vessel internals in four of its seven units.

Duke's Response to RAI 3.3-1

Flexible connectors were inadvertently omitted from the Application for the Auxiliary Building, Control Area, Diesel Building, and Fuel Handling Building or Fuel Handling Ventilation Systems. Tables 3.3-1, 3.3-11, 3.3-13, and 3.3-28 are supplemented with the following aging management review results. (See April 15, 2002, response for the technical detail provided in the aging management review results tables.)

Staff's Concern - RAI 3.3-1 (Confirmatory Item)

In reviewing the aging management review results, the staff noticed that no aging effects had been identified for the flexible connectors therein. The staff is concerned that the applicant failed to identify applicable aging effects and a program to manage them. However, the applicant provided the following supplemental information in electronic correspondence, dated May 10, 2002 (ML012440236):

> The Application is required only to report the results of the aging management review performed by Duke on license renewal components. In response to RAI 3.3-1, Duke reported the results of the aging management review for flexible connectors and did not discuss the possible aging effects considered, which is consistent throughout the Application.
>
> Duke evaluated the flexible connectors for loss of material and change in material properties (hardening) from exposure to the ambient environmental conditions at the component locations within each plant. Internal and external temperature and radiation levels at these flexible connector locations are well below those known to be an aging concern for the period of extended operation. Therefore, loss of material and change in material properties (hardening) were not identified as aging effects in the tables provided in our initial response to RAI 3.3-1. The results of these evaluations are documented in Duke technical documents available for on-site staff review.

Pending the staff's receipt of this information in official correspondence, this item is characterized as confirmatory.

Duke Response to Staff's Concern - RAI 3.3-1 (Confirmatory Item)

The following information as provided by Duke in an electronic communication dated May 10, 2002 addresses the staff concern:

> "The Application is required only to report the results of the aging management review performed by Duke on license renewal components. In response to RAI 3.3-1, Duke reported the results of the aging management review for flexible connectors and did not discuss the possible aging effects considered, which is consistent throughout the Application.
>
> Duke evaluated the flexible connectors for loss of material and change in material properties (hardening) from exposure to the ambient environmental conditions at the component locations within each plant. Internal and external temperature and radiation levels at these flexible connector locations are well below those known to be an aging concern for the period of extended operation. Therefore, loss of material and change in material properties (hardening) were not identified as aging effects in the tables provided in our initial response to RAI 3.3-1. The results of these evaluations are documented in Duke technical documents available for on-site staff review."

Duke's Response to RAI 3.3-5

The staff is correct that these components are subject to a sheltered internal environment. Duke's aging management review conservatively evaluated environments such as tanks and piping that are open to atmosphere as a ventilation environment. Although the tanks and piping are open to a sheltered environment, they would not experience significant air exchange and thus higher humidity and condensation could be present. The ventilation environment aging effect details account for the potential condensation, whereas the sheltered environment aging effect details do not. Loss of material and cracking due to alternate wetting and drying that concentrates contaminants are two aging effects considered plausible in a ventilation environment, but are not considered in a sheltered environment. Loss of material due to selective leaching is another aging effect considered plausible in a ventilation environment, but is not considered in a sheltered environment. Therefore, for conservatism, Duke chose to evaluate these component configurations using the ventilation environment aging management review details. The designation in the Application table reflects this decision.

Staff Concern - RAI 3.3-5 (Confirmatory Item)

The staff is concerned that the applicant is non-conservative in designating a "ventilation" internal environment for the carbon steel components in Tables 3.3-14 and 3.3-44. The sheltered environment is subject to the aging effect loss of material and managed by the "Inspection Program for Civil Engineering Structures and Components." This appears to conflict with the Duke response, which states that loss of material in a sheltered environment is not considered an aging effect. The staff's fundamental concern is that, for the diesel engine exhaust systems (which do not include coolers or dryers for controlling air quality), the internal environments are not conditioned to maintain a suitable environment for equipment operation and personnel occupancy (in accordance with the applicant's definition of a "ventilation" environment). Therefore, it appears to the staff that a sheltered environment is a more conservative designation, and aging effects associated with a sheltered environment should be addressed for these internal surfaces. However, the applicant provided the following supplemental information in electronic correspondence dated May 10, 2002 (ML012440236):

> For Duke, a sheltered environment is an external environment for components inside a structure that may or may not be maintained by a ventilation system but are protected from the natural elements. Components in a sheltered environment could be wet from condensation or leakage that could promote aggressive corrosion, that left unmanaged, could result in a loss of the component intended function(s) during the period of extended operation. As such, the Inspection Program for Civil Engineering Structures and Components is credited to manage the aging effects on the external surfaces of components located in a sheltered environment. For components with an internal air environment open to the sheltered environment or yard environment (as is the case with the diesel exhaust), Duke classified the environment as a ventilation environment. Duke

conservatively chose the ventilation environment because more aging mechanisms leading to aging effects are plausible and must be considered than in a sheltered environment. In our initial response to RAI 3.3-5, Duke tried to show that aging effects from some mechanisms are not plausible in a sheltered environment but could occur in a ventilation environment. Duke was providing examples to support our conservative position which we believe does not say that loss of material in a sheltered environment is not an aging effect.

Duke evaluated the internal environment of the exhaust systems as a ventilation environment. The diesels operate periodically for short periods of time for testing but are primarily in standby. The internal environment is characterized as a warm, dry environment free from leaks and condensation. This environment does not preclude loss of material but does not promote the aggressive corrosion that left unmanaged would result in a loss of the component intended function(s) of the exhaust system components. Therefore, no aging effects requiring management during the period of extended operation were identified.

Pending the staff's receipt of this information in official correspondence, this item is characterized as confirmatory.

<u>Duke Response to Staff's Concern - RAI 3.3-1 (Confirmatory Item)</u>

The following information as provided by Duke in an electronic communication dated May 10, 2002 addresses the staff concern:

"For Duke, a sheltered environment is an external environment for components inside a structure that may or may not be maintained by a ventilation system but are protected from the natural elements. Components in a sheltered environment could be wet from condensation or leakage that could promote aggressive corrosion, that left unmanaged, could result in a loss of the component intended function(s) during the period of extended operation. As such, the Inspection Program for Civil Engineering Structures and Components is credited to manage the aging effects on the external surfaces of components located in a sheltered environment. For components with an internal air environment open to the sheltered environment or yard environment (as is the case with the diesel exhaust), Duke classified the environment as a ventilation environment. Duke conservatively chose the ventilation environment because more aging mechanisms leading to aging effects are plausible and must be considered than in a sheltered environment. In our initial response to RAI 3.3-5, Duke tried to show that aging effects from some mechanisms are not plausible in a sheltered environment but could occur in a ventilation environment. Duke was providing examples to support our conservative position which we believe does not say that loss of material in a sheltered environment is not an aging effect.

Duke evaluated the internal environment of the exhaust systems as a ventilation environment. The diesels operate periodically for short periods of time for testing but are primarily in standby. The internal environment is characterized as a warm, dry environment free from leaks and condensation. This environment does not preclude loss of material but does not promote the aggressive corrosion that left unmanaged would result in a loss of the component intended function(s) of the exhaust system components. Therefore, no aging effects requiring management during the period of extended operation were identified."

Duke's Response to RAI 3.3.24-5

The two environments, "air (moist)" and "air (dry)," were provided in Table 3.3-24 to show that the air environment was not the same throughout the Diesel Generator Starting Air System. Both of these air environment variations are bounded by the "Air-Gas" environment definition in Section 3.3.1 of the Application. The Diesel Generator Starting Air System takes air from the diesel room. The air is filtered, compressed, dried and stored in tanks to be used to start the diesels. The "air (moist)" environment is the environment prior to the air dryers. The "air (dry)" environment is the environment after the air dryers.

Staff Concern - RAI 3.3.24-5 (Confirmatory Item)

The applicant addressed the RAI as it was written. However, the staff is concerned that the applicant identified no aging effects for carbon steel in the "air (moist)" environment. Because aging mechanisms and rates can vary depending on the moisture content in these environments, the staff is not confident in the applicant's conclusion that these components are not subject to aging effects. The applicant provided the following supplemental information in electronic correspondence, dated May 10, 2002 (ML012440236):

> Duke believes that characterizing the environment as moist air is misleading. As noted in our initial response, the Diesel Generator Starting Air System takes air from the diesel room. Since the diesels are heated, the moist air of the diesel rooms is in excess of 100 °F and has a low relative humidity. The Diesel Generator Starting Air System filters, compresses and further dries this air for storage in the system tanks for later use. The diesel room air does not preclude loss of material but does not promote the aggressive corrosion that left unmanaged could result in a loss of the intended function(s) of the components. Therefore, no aging effects requiring management during the period of extended operation were identified.

Pending the staff's receipt of this information in official correspondence, this item is characterized as confirmatory.

Duke Response to Staff Concern - RAI 3.3.24-5 (Confirmatory Item)

The following information as provided by Duke in an electronic communication dated May 10, 2002 addresses the staff concern:

> "Duke believes that characterizing the environment as moist air is misleading. As noted in our initial response, the Diesel Generator Starting Air System takes air from the diesel room. Since the diesels are heated, the moist air of the diesel rooms is in excess of 100 °F and has a low relative humidity. The Diesel Generator Starting Air System filters, compresses and further dries this air for storage in the system tanks for later use. The diesel room air does not preclude loss of material but does not promote the aggressive corrosion that left unmanaged could result in a loss of the intended function(s) of the components. Therefore, no aging effects requiring management during the period of extended operation were identified."

Duke's Response to RAI 3.3.36-1

All of the lube oil cooler components cited in RAI 3.3.36-1 are components of closed oil recirculation systems. Uncontaminated lube oil does not cause aging, and closed oil recirculation systems are assumed to be initially free of contaminants such as water. Further, in the Duke aging management review, component failures were not postulated as a means to establish the relevant conditions required for aging to occur. Therefore, in oil coolers, tube failures that could introduce water into a lube oil environment are not assumed.

See also the response to RAI 3.3-3.

Staff Concern - RAI 3.3.36-1 (Confirmatory Item)

While all systems are designed initially to be leak tight, leaks in the pressure boundary components can develop. Leakage of water into oil systems may involve minor breaches in component pressure boundaries that may go undetected and allow corrosion and other forms of degradation to progress indefinitely (which is why plants implement surveillance monitoring programs for lubricating oil and fuel oil systems). In fact, industry operating experience indicates that oil periodically is contaminated with cooling water. Therefore, the staff is concerned that applicant failed to address loss of material from general corrosion, pitting, crevice corrosion, and microbiologically influenced corrosion of stainless steel and copper-nickel materials for oil coolers potentially contaminated with leaking water. However, the applicant provided the following supplemental information in electronic correspondence, dated May 10, 2002 (ML012440236):

> For this response, Duke is assuming that the staff believes breaches of the pressure boundary in the oil coolers are the result of aging of the raw water side of the cooler that allows raw water to contaminate the oil. Duke reiterates that component failures due to aging were not postulated as a means to establish the relevant conditions required for aging to occur. For the oil coolers in question, Duke identified the aging that could occur in the normal environment. No aging effects were identified for the cooler components exposed to uncontaminated oil.
>
> Aging effects were identified for the cooler components exposed to raw water that left unmanaged could result in a loss of the pressure boundary function. Duke credited the Heat Exchanger Preventive Maintenance Activities - Pump Oil Coolers described in Section B.3.17.7 of the Application to manage the pressure boundary integrity to prevent the contamination of the oil system. Industry operating experience indicates the need for such a monitoring program. Plant specific operating experience also demonstrates that the aging management program credited has been and will continue to be effective during the period of extended operation.

Pending the staff's receipt of this information in official correspondence, this item is characterized as confirmatory.

Duke Response to Staff Concern - RAI 3.3.36-1 (Confirmatory Item)

The following information as provided by Duke in an electronic communication dated May 10, 2002 addresses the staff concern:

> "For this response, Duke is assuming that the staff believes breaches of the pressure boundary in the oil coolers are the result of aging of the raw water side of the cooler that allows raw water to contaminate the oil. Duke reiterates that component failures due to aging were not postulated as a means to establish the relevant conditions required for aging to occur. For the oil coolers in question, Duke identified the aging that could occur in the normal environment. No aging effects were identified for the cooler components exposed to uncontaminated oil.
>
> Aging effects were identified for the cooler components exposed to raw water that left unmanaged could result in a loss of the pressure boundary function. Duke credited the Heat Exchanger Preventive Maintenance Activities - Pump Oil Coolers described in Section B.3.17.7 of the Application to manage the pressure boundary integrity to prevent the contamination of the oil system. Industry operating experience indicates the need for such a monitoring program. Plant specific operating experience also demonstrates that the aging management program credited has been and will continue to be effective during the period of extended operation."

Duke's Response to RAI 3.3.36-2

The relevant conditions required for loss of material due to selective leaching to occur in copper-nickel alloys are a temperature greater than 212°F, low flow, and high local heat fluxes. These conditions are not found in the Nuclear Service Water System. Therefore, loss of material due to selective leaching is not an aging effect requiring management during the period of extended operation for copper-nickel alloy components exposed to raw water.

Staff Concern - RAI 3.3.36-2 (Confirmatory Item)

Service water inspections and industry experience from ANO-1 indicates that even under high flow conditions the impurity, Cl biocide, in the systems resulted in de-nickelification of the 90/10 copper-nickel heat exchanger tubes where 70/30 copper-nickel may have been less susceptible to the selective leaching aging affect. Since the copper content of the component is a significant contributor to material vulnerability independent of temperature and flow conditions, the staff does not have sufficient information to conclude that the composition of the service water heat exchanger tubes is resistant. However, the applicant provided the following supplemental information in electronic correspondence, dated May 10, 2002 (ML012440236):

> Duke believes that the industry experience from ANO-1 is not relevant to the McGuire Nuclear Service Water System. The McGuire Nuclear Service Water System is an untreated open-cycle cooling water system. The operating experience presented notes that selective leaching occurred as a result of the chlorine biocide. Duke does not use chlorine biocides in the McGuire Nuclear Service Water System. Therefore, selective leaching of copper-nickel alloys is not a concern.

Pending the staff's receipt of this information in official correspondence, this item is characterized as confirmatory.

Duke Response to Staff Concern - RAI 3.3.36-2 (Confirmatory Item)

The following information as provided by Duke in an electronic communication dated May 10, 2002 addresses the staff concern:

> "Duke believes that the industry experience from ANO-1 is not relevant to the McGuire Nuclear Service Water System. The McGuire Nuclear Service Water System is an untreated open-cycle cooling water system. The operating experience presented notes that selective leaching occurred as a result of the chlorine biocide. Duke does not use chlorine biocides in the McGuire Nuclear Service Water System. Therefore, selective leaching of copper-nickel alloys is not a concern."

Duke's 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.

Staff Concern - RAI 3.5-1 (Open Item)

The staff expressed concern that the applicant did not plan to periodically monitor groundwater during the extended period of operation to confirm that it is not aggressive to buried portions of concrete structures. During the NRC Scoping and Screening Inspection, the applicant provided data from Lake Norman, adjacent to McGuire nuclear station, and Lake Wylie, adjacent to Catawba nuclear station, showing pH values and phosphate, chloride and sulphate contents (ML 021090060). The lake water sampling dates are from 1962 to 1996 for McGuire (Lake Norman) and from 1971 to 1996 for Catawba (Lake Wylie). In addition, the applicant referred the staff to the Environmental Reports (ERs) associated with the original construction of Catawba and McGuire. The ERs contain water table contour maps (ER Figure 2.4.4-2 for Catawba, and ER Figure 2.5.2-2, Revision 2, for McGuire).

As stated in the applicant's response to RAI 3.5.1, the chloride, sulfate, and pH values over the past 20 to 30 years are well below the limits where potential degradation of concrete may occur. In addition, the water contour tables for both Catawba and McGuire show that the water table levels decrease from the two nuclear stations outward to the surrounding areas such that only a chemical event at the nuclear stations would potentially impact their respective site environments, including the groundwater. However, in its response to RAI 3.5-1, the applicant does not commit to initiate a corrective action in the event of a potential change to the site environment resulting from a chemical release during the period of extended operation. Such a corrective action needs to include a commitment to monitor the groundwater chemistry and to assess the potential impact of any changes to the groundwater chemistry on below-grade concrete components.

Duke Response to Staff Concern - RAI 3.5-1 (Open Item)

To reiterate, 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 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. A review of operating experience did not identify any incidences that resulted in changes to the environmental parameters that could induce potential degradation of the concrete.

The NRC has previously reviewed the impacts of plant operation on groundwater use and quality and documented the results of its review in NUREG-1437, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants." The report states:

> Impairment of groundwater quality could occur at estuary and ocean site facilities that withdraw groundwater for any purpose (e.g., potable and service water systems, operational dewatering). Long-term pumping of groundwater from coastal plain aquifers by industrial and municipal facilities has contributed to saltwater intrusion in some areas of nearly every Atlantic and Gulf Coast state (USGS 1990). The saltwater intrusion issue was evaluated by examining groundwater use at selected nuclear power plants sited on estuaries and oceanic coastlines. Operational dewatering is not taking place at any of the estuaries or coastal sites.
>
> Groundwater quality could also be impaired at inland sites where groundwater may be replaced by poorer quality river water through induced infiltration (NUREG-0777). Potential impairment of groundwater quality may occur at facilities that have large cooling ponds.

None of the issues identified in NUREG-1437 that could result in changes to groundwater quality due to plant operation are relevant to McGuire or Catawba. Neither McGuire nor Catawba is located on an estuary or an oceanic coastline. In addition, neither McGuire nor Catawba uses a cooling pond.

Duke did not commit to initiate a corrective action in the event of a potential change to the site environment resulting from a chemical release during the period of extended operation because Duke did not postulate a change to the environment due to a chemical release. It is simply not credible to postulate that some environmental event will occur in the future that would affect the quality of the groundwater in the vicinity of McGuire or Catawba. Change in the environment due to a chemical release would be an abnormal event. Abnormal events are addressed in Appendix A-1 of NUREG-1800, "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants." The SRP-LR states:

> The applicable aging effects to be considered for license renewal include those that could result from normal plant operation, including plant/system operating transients and plant shutdown. Aging effects from abnormal events need not be postulated specifically for license renewal. However, if an abnormal event has occurred at a particular plant, its contribution to the aging effects on structures and components for license renewal should be considered for that plant.

In summary, McGuire and Catawba groundwater parameters have been shown by years of environmental testing to be below the limits where degradation could occur. A change in these parameters due to a chemical release has not been documented in either operating experience or NUREG-1437. A chemical release would be "an abnormal event" and would not be included in an aging management review. Therefore, future tests of groundwater chemistry are not required.

Duke's 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 and 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 (reference "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 a sheltered environment, the reactor building environment, and an external environment.

Cable tray is constructed of painted or galvanized sheet metal similar to metal housing and is located in the same environment; therefore, cable tray would age similarly to the metal housings. 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 a sheltered environment, the reactor building environment, and an external environment.

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.

Staff Concern - RAI 3.5-4

It is not clear to the staff what the difference is between the bellows and the other components referenced in the RAI that would explain why one component is subject to cracking and not the others. If the chloride was potentially introduced during the manufacturing process for the bellows, but the other components were not subject to the same type of brightening process during manufacturing, the applicant should state so.

Duke Response to Staff Concern - RAI 3.5-4

This staff concern was discussed in telephone conference call on May 28, 2002 which is summarized in a staff memorandum dated June 7, 2002. In this call, Duke indicated that a leaking bellows had been identified in 1993 and was replaced in 1994. In 1997, leakage from the replacement bellows was identified, and the leaking bellows was replaced. A root cause determination attributed the 1997 bellows leak to transgranular stress-corrosion cracking (TGSCC) as a result of exposure to or contact with chlorine. Duke could not determine the source of chlorine and speculated that the contaminant could have been introduced by a surface brightener during the manufacturing process. Duke further stated that TGSCC had not been listed as an applicable aging effect for the other components (fuel transfer canal liner plate, sump liner, and sump screens) because the normal operating environment would not expose these components to chlorine and they essentially consist of plate material that had not been polished or brightened by the manufacturer.

As stated in the memo, the staff found "the applicant's explanation of why cracking caused by TGSCC was not identified as an applicable aging effect for fuel transfer canal liner plate, sump liner, and sump screens reasonable, but may characterize this as a Confirmatory Item... ." Please refer to the staff memorandum dated June 7, 2002 for closure of this issue.

Duke's 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 American Concrete Institute (ACI) 318 and constructed in accordance with ACI 301 using ingredients conforming to ACI and American Society for Testing and Materials (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. Ice condenser wall panel defrosting is not a normal maintenance practice at either McGuire or Catawba. 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 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 is protected by the panels and cooling ducts, no aging effects requiring management were identified for the crane wall.

Staff Concern - RAI 3.5-6 (Open Item)

Since the ice condenser wear slab, structural concrete floor and crane wall are characterized as inaccessible and in a unique environment of low humidity and temperature, the staff acknowledges that there are no accessible concrete components in a similar environment that the applicant could use as an indicator of the aging of these inaccessible ice condenser components. However, the staff does not accept Duke's position that the uniquely low humidity and temperature environment precludes aging effects, since Duke did not reference industry operating experience or data to support the assertion.

The applicant indicated, in its response to the RAI, that portions of both the structural concrete floor, which is located beneath the ice condenser wear slab, and the crane wall are accessible for inspection. Specifically, the applicant stated that the structural concrete floor is accessible from below and that the interior surface of the crane wall is open to the reactor building environment and is accessible for inspection. Therefore, the staff considers (in light of the staff's position on concrete aging, issued to the industry by letters dated November 23, 2001 [ML013300426] and April 5, 2002 [ML020980194]) the applicant's response to RAI 3.5-6 to be inadequate with regard to the structural concrete floor and the crane wall. For the ice condenser wear slab, the applicant did not state in its response that it would inspect the wear slab in the event that defrosting of an ice condenser wall panel allows access to the wear slab. As such, the staff is concerned that the applicant has not proposed to do what it can do to inspect and monitor the aging of accessible and potentially accessible concrete structures associated with the ice condenser.

Duke Response to Staff Concern - RAI 3.5-6 (Open Item)

Duke has decided to defer the response to Staff Concern - RAI 3.5-6 (Open Item) until after the staff issues the SER with Open Items to provide sufficient time for responsible engineering staff at each station to be involved in the preparation of the response.

Duke's Response to RAI 3.5-7

Duke Power disagrees with the NRC staff position. The standards and results of NUREG-1522 inspections do not lead 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 too 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 instances were identified that, if not discovered and repaired, could potentially had have serious consequences. These instances 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 instances where the structural integrity of the component was jeopardized were attributed to design, construction, or human errors, but not to aging (Reference NUREG/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. Not all cracks 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.

Staff Concern - RAI 3.5-7 (Open Item)

Contrary to the applicant's claim that aging management of concrete components via periodic inspections is only necessary for concrete SCs that are exposed to harsh environments, the staff's position is that both the operating and environmental conditions, as well as the aging of concrete nuclear components, are subject to change throughout the period of extended operation. As such, applicants need to periodically inspect these components. ACI 349.3R, "Evaluation of Existing Nuclear Safety-Related Concrete Structures," is a report that represents a consensus of knowledgeable individuals from the nuclear industry, consultants, and regulators. As stated in ACI 349.3R, sound engineering practices during material (concrete mix) design and construction, together with sound inspection programs in which the performance and condition are periodically evaluated and monitored, are both necessary to maintain the serviceability of concrete nuclear structures. Periodic visual inspections (1) can provide significant quantitative and qualitative data regarding structural performance and extent of degradation, (2) are vital to monitor the effects of operating and environmental conditions, and (3) enable the timely identification and correction of degraded conditions.

Although the applicant has performed an aging management review pursuant to 10 CFR 54.21(a)(3) for each structure and component that was determined to be in the scope of license renewal, the staff position (issued by letters dated November 23, 2001 [ML013300426], and April 5, 2002 [ML020980194]) is that aging management reviews should be used to differentiate between those components requiring only periodic inspections and those requiring further evaluation, as recommended by the Generic Aging Lessons Learned (GALL) report (NUREG-1801). Aging management review results of concrete structures and components may also be used to establish different scheduled inspection frequencies, similar to those recommended by ACI 349.3R, for aging management programs. The staff is concerned that the applicant has not proposed to perform periodic inspections of concrete components during the period of extended operation. Therefore, the staff is unable to make a reasonable assurance finding that in-scope concrete structures and components will maintain their structural integrity and intended functions.

Duke Response to Staff Concern - RAI 3.5-7 (Open Item)

Duke has decided to defer the response to Staff Concern - RAI 3.5-7 (Open Item) until after the staff issues the SER with Open Items to provide sufficient time for responsible engineering staff at each station to be involved in the preparation of the response.

Duke's Response to RAI 3.5-8

Resistance to leaching can be enhanced by using concrete with low permeability. A dense, well-cured concrete with a suitable cement content 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 instances 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 refueling water storage tank (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.

Staff Concern - RAI 3.5-8 (Open Item)

As noted in the staff's concern with Duke's response to RAI 3.5-7, the staff is unable to make a reasonable assurance finding that in-scope concrete structures and components will maintain their structural integrity and intended functions because the applicant has not identified aging effects for certain components (e.g., missile shields) or designated a program to manage those aging effects. The applicant indicates that resistance to leaching can be enhanced by using concrete with low permeability. The applicant further states that 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. However, the staff is concerned that, although resistance to leaching can be enhanced, leaching cannot be precluded as an aging effect. Therefore, this aging effect should be identified and managed during the extended period of operation.

Duke Response to Staff Concern - RAI 3.5-8 (Open Item)

Duke has decided to defer the response to Staff Concern - RAI 3.5-8 (Open Item) until after the staff issues the SER with Open Items to provide sufficient time for responsible engineering staff at each station to be involved in the preparation of the response.

Duke's Response to RAI 3.6.1-1

Duke understands the basis of RAI 3.6.1-1 as concerning the adequate aging management of non-EQ electrical cables used in low-level signal applications that are sensitive to reduction in insulation resistance (IR), such as radiation monitoring and nuclear instrumentation. As stated in Section B.3.23 of the Application, the McGuire and Catawba Non-EQ Insulated Cables and Connections Aging Management Program includes these cables within the total population of cables and connections included in this visual inspection program. Having performed extensive, plant-wide visual inspections as part of the license renewal preparatory work at Oconee, Duke has a very high confidence that the visual inspections outlined in this program will detect early aging degradation of insulation of all types of cables and connections—including those that are the subject of RAI 3.6.1-1. The McGuire and Catawba Non-EQ Insulated Cables and Connections Aging Management Program is consistent with Gall Report program XI.E1. For these reasons, Duke does not credit a plant calibration test program for aging management.

Additional Information for Response to RAI 3.6.1-1 Regarding Visual Inspections and Detection of Aging Degradation

Two statements are made in RAI 3.6.1-1 regarding visual inspections that are inaccurate and unsupported. This additional information section examines these statements to assist the reviewer in recognizing the strength of visual inspections.

RAI 3.6.1-1 makes the following statement: "Visual inspection may not be sufficient to detect aging degradation from heat and radiation in the instrumentation circuits with sensitive, low-level signal."

This RAI statement is in disagreement with GALL Report Table VI.A (page VI A-3). Item A.1-a of Table VI.A pertains to all non-EQ cables and connections (including those that are the subject of RAI 3.6.1-1). Item A.1-a of Table VI.A identifies program XI.E1 (the visual inspection program) as providing aging management for aging effects that include "reduced insulation resistance" and indicates that no further evaluation is recommended. The statement in the RAI that "Visual inspection may not be sufficient to detect aging degradation…" is in contradiction to the GALL Report.

For low-voltage cables, embrittlement and significant cracking (through cracks) of the cable jacket and conductor insulation would have to occur before the introduction of moisture around the cable could be an issue. As stated in the Program Description for GALL Report program XI.E1, "the electrical cables and connections covered by this aging management program are either not exposed to harsh accident conditions or are not required to remain functional during or following an accident to which they are exposed." GALL Report Table VI.A (Item A.1-a, page VI A-3) indicates that visual inspection program XI.E1 manages "moisture intrusion" and indicates that no further evaluation is recommended.

RAI 3.6.1-1 makes the following statement: "These low levels of leakage current may affect instrument loop accuracy before the adverse localized environment that caused them produces changes that are visually detectable."

This RAI statement contradicts statements made in Department of Energy report SAND96-0344, "Aging Management Guideline for Commercial Nuclear Power Plants - Electrical Cable and Terminations." SAND96-0344 is cited as a reference in both NUREG-1800 (SRP for license renewal applications) and NUREG-1801 (GALL Report). SAND96-0344 provides a comprehensive compilation and evaluation of information on the topic of aging and aging management for cables and their associated connections. SAND96-0344 Section 5.2.2, "Measurement of Component or Circuit Properties," states the following (underline added for emphasis):

> **SAND96-0344, Section 5.2.2**
> "Diagnostic techniques to assist in assessment of the functionality and condition of power plant cables and terminations are described in this section....
>
> "Significant changes in mechanical and physical properties (such as elongation-at-break and density) occur as a result of thermal- and radiation-induced aging. For low-voltage cables, these changes precede changes to the electrical performance of the dielectric. Essentially, the mechanical properties must change to the point of embrittlement and cracking before significant electrical changes are observed...."

"Embrittlement and cracking" are signs of extensive aging that are easily detectable by visual inspection. Signs of less extensive aging, such as discoloration, are also easily detectable by visual inspection. Visual inspections can detect aging degradation early in the aging process before significant aging degradation has occurred. SAND96-0344 Section 5.2.2.1.2, "Insulation Resistance (IR)-Advantages/Disadvantages," provides further information on insulation resistance as an electrical property related to aging of cables:

> **SAND96-0344, Section 5.2.2.1.2**
> IR may give some indication of the aging of connections; however, it is generally considered of little use in predicting the aging of a cable. IR properties of dielectrics may change little until severe degradation of mechanical properties occurs. These measurements display some gradual changes with aging, but are generally nowhere near as sensitive to aging as techniques based on mechanical properties.... Conversely, even gross insulation damage may not be evidenced by changes in IR; for example, an insulation cut-through surrounded by dry air may not significantly affect IR readings.... Testing is usually conducted as a pass/fail....

Performing visual inspections is supported as a promising condition monitoring technique. As described in Section 5.2.2.4 of SAND96-0344:

> **SAND96-0344, Section 5.2.2.4**
> In mid-1993 the U.S. NRC Office of Nuclear Reactor Regulation (NRR) initiated an EQ task action plan (EQ TAP) which sets forth specific activities of the Office of Nuclear Regulatory Research (RES) and NRR relating to the qualification of electrical components. Potential safety issues addressed by the EQ TAP include...condition monitoring methods. One of the primary focal points of this effort relates to low-voltage cables.

An array of condition-monitoring techniques were evaluated in the EQ TAP in order to identify those that are "promising." Calibration testing was not included among the array of condition-monitoring techniques evaluated as part of the EQ TAP. Visual inspection was evaluated as part of the EQ TAP and was identified as a "promising" condition monitoring technique.

Visual inspections are also discussed in the "License Renewal Electrical Handbook" (EPRI 1003057, page 14-3) as follows:

> **License Renewal Electrical Handbook**
> Research continues to be performed on condition monitoring methods that run the full spectrum from very unsophisticated to ultra-sophisticated. To date, out of all that research, no sophisticated approach has been found workable for the full range of plant cables, cable installations and environments at the U.S. nuclear power plants. The only universal technique that was found to provide reasonable indication that could be related to cable degradation was visual inspections.... At present, visual inspection techniques are the only practical and universal type of condition monitoring program and are adequate for the cables and connections covered by this [XI.E1] GALL Report program.

SAND96-0344 (Chapter 5) also provides a comprehensive review of maintenance, surveillance and condition-monitoring techniques for evaluation of electrical cable and terminations. SAND96-0344 Table 5-1 identifies Inspection Techniques Applicable to Various Degradation Mechanisms and "*Visual inspection*" is identified in the table as an applicable technique for each mechanism. Tables 5-2, 5-3, 5-4 and 5-5 list Destructive, Nondestructive and Essentially Nondestructive Condition Monitoring Techniques and calibration testing is not identified in any of these tables as a condition monitoring technique. In addition, a word search concluded that neither calibration nor calibration testing is identified in any part of SAND96-0344.

The additional information above provides a basis for the strength of visual inspections as a condition monitoring technique that is recognized by both the industry and the NRC. Duke intends that this additional information aid the reviewer in recognizing the strength of the McGuire and Catawba *Non-EQ Insulated Cables and Connections Aging Management Program*, which is based on visual inspections.

Staff Concern - RAI 3.6.1-1 (Open Item)

Because a moist environment can apparently hasten the failure of I&C circuits if they have previously undergone age-related degradation, the disposition of a degraded cable should consider the potential for moisture in the area of degradation. The applicant should verify that this is the case for the Corrective Action attribute of the McGuire and Catawba Non-EQ Insulated Cables and Connection Aging Management Program.

The staff noted that visual inspection of low voltage instrumentation circuits may be an effective means to detect age-related degradation due to adverse localized environments. However, this finding is not necessarily the case for high range radiation monitor and neutron monitoring system cables. The SAND 96-0344 report referenced by the applicant states on page 3-36 that neutron monitoring systems (including source, intermediate, and power range monitors) were separated into their own category based on (1) their substantial difference with typical low- and medium-voltage power, control, and instrumentation circuits, (2) the relatively large number of reports related to these devices and identified in the databases. The report states that neutron detectors are frequently energized at what is commonly referred to as "high" voltage, usually between 1kV and 5kV. This is not high voltage in the sense of power transmission voltage, but rather elevated with respect to other portions of the detecting circuit. The report included the lower voltage non-detection portion of typical neutron monitoring equipment in the low voltage equipment category, but separated out the 1kV to 5kV neutron detectors into a separate category that included neutron monitor cables and connectors.

The high voltage portion of the neutron monitoring systems would appear to be a worst-case subset of the low signal level instrumentation circuit category. These circuits operate at low level logarithmic signals that are sensitive to relatively small changes in signal strength, and they operate at a high voltage that could create larger leakage currents if that voltage is impressed across associated cables and connections. Radiation monitoring cables have also been found to be particularly sensitive to thermal effects. NRC Information Notice 97-45, Supplement 1, describes this phenomenon. The neutron monitoring circuits and radiation monitors, therefore, might be candidates for the calibration approach but not necessarily the visual inspection approach. The calibration approach was used for these circuits at the Calvert Cliff Nuclear Power Plant. Page 6.1-22 of its license renewal application on states:

> "The IR reduction effect can be a concern for circuits with sensitive, low level signals such as current transmitter, resistance temperature detectors, and thermocouples. It is especially a concern for channels with logarithmic signals such as radiation monitors and neutron monitoring instrumentation. The IR reduction effect contributes to inaccuracies in the instrumentation loop current signal (e.g., 4-20 ma) such that the measurements of the process variable (e.g., rads/hour) become more uncertain.

The applicant should provide a technical justification for high range radiation monitor and high voltage neutron monitoring instrumentation cable that will demonstrate that visual inspection will be effective in detecting aging before current leakage can affect instrument loop accuracy.

Duke Response to Staff Concern - RAI 3.6.1-1 (Open Item)

In response to the staff concerns expressed in Staff Concern - RAI 3.6.1-1 (Open Item), Duke makes observations concerning two statements made by the staff.

The first observation concerns NRC Information Notice 97-45, Supplement 1 and its use as the basis for establishing the need for an aging management program. Duke observes that IN 97-45, Supplement 1 is concerned with an operational event that is unrelated to aging and can be readily addressed by design and configuration changes in the plant to eliminate the degradation and system reaction that had been observed. Duke in fact had reviewed IN 97-45, Supplement 1 following its issuance and determined that the design and configuration of similar instrumentation at McGuire and Catawba simply does not exist. Duke does not believe that a single information notice by itself should form the basis for establishing an aging management program requirement that is generic to the industry. In fact, the IN states "This information notice requires no specific action or written response." Other generic communications such as bulletins and generic letters are the appropriate means to obtain licensee actions.

The second observation concerns the staff approval of the calibration approach for a previous license renewal applicant. Duke observes that this staff approval occurred prior to the issuance of guidance concerning aging management programs in general. This guidance is contained in SRP-LR, NUREG-1800. SRP-LR Section A.1.2.3.4 - **Detection of Aging Effects** states:

> "Detection of aging effects should occur before there is a loss of the structure and component function... A program based solely on detecting structure and component failure should not be considered as an effective aging management program for license renewal."

The staff has included the previously approved calibration program in the GALL Report, NUREG-1801. GALL Electrical AMP XI.E2 does not meet the expectations for aging management programs expressed in SRP-LR nor the overall philosophy of license renewal, which is to preclude aging effects before loss of function occurs.

GALL AMP XI.E2 is a performance monitoring program. The program description states:

> "Operating experience has shown that a significant number of cable failures are identified through routine calibration testing."

According to SRP-LR, performance monitoring programs should detect aging effects prior to loss of function, not after function is lost. Electrical AMP XI.E2 detects failures of the cable insulation after such failures occur.

In contrast, look at the operating experience for GALL AMP XI.E1, which is a condition monitoring program:

> "Operating experience has shown that adverse localized environments caused by heat or radiation for electrical cables and connections may exist next to or above (within three feet of) steam generators, pressurizers or hot process pipes, such as feedwater lines. These adverse localized environments have been found to cause degradation of the insulation materials on electrical cables and connections that is visually observable, such as color changes or surface cracking. These visual indications can be used as indicators of degradation."

Clearly this AMP XI.E1 meets the expectations of SRP Section A.1.2.3.4 and the overall license renewal philosophy whereas GALL AMP XI.E2 does not.

The staff position that the calibration approach is effective to manage aging effects associated with neutron and radiation monitor cables simply does not meet the standards established for aging management programs in SRP-LR. Duke disagrees with the staff position.

Duke continues to believe that the *Non-EQ Insulated Cables and Connections Aging Management Program* provides reasonable assurance that the effects of aging will be managed for the period of extended operation and is consistent with the previous staff decisions on four other license renewal applications, including the decision for Oconee contained in NUREG-1723, March 2000.

Duke agrees to add the following statement to the **Corrective Actions & Confirmation Process** of the *Non-EQ Insulated Cables and Connections Aging Management Program*:

> "[The program] should consider the potential for moisture in the area of degradation."

Duke's Response to RAI 4.3-4(2) and (3)

RAI 4.3-4 seeks information related to a document that was not considered by Duke in the Application. To better understand the request, Duke has reviewed WCAP-14574-A and notes that Item 1 in the above request refers to off normal transients and other additional transients that, if applicable to a particular plant, were imposed on a plant-specific basis. Such transients, once analyzed, would then become part of that plant's design basis.

For the pressurizers at McGuire and Catawba, pressurizer insurge/outsurge is the only off normal or additional transient that has been analyzed and incorporated into the thermal fatigue design basis. To mitigate the effects of insurge/outsurge, McGuire and Catawba implemented modified operating procedures in the mid 1990's. Additionally, historical plant instrument data was analyzed to determine an insurge/outsurge history encompassing pre- and post-application of the modified operating procedures with an extrapolation for all appropriate design transient occurrences. Analysis of these occurrences of insurge/outsurge were analyzed and it was found that the CUF of the affected pressurizer parts will remain less than 1.0 for all appropriate design transient occurrences. As can be seen in RAI Response 4.3-1, management of all appropriate fatigue design transient occurrences allows the effects of insurge/outsurge on the pressurizer to be managed by the Thermal Fatigue Management Program. The discussion of how the Thermal Fatigue Management Program manages the thermal fatigue design basis is provided in the Application and in Response to RAI 4.3-1.

The information requested in Item 2 above asks for a comparison of Duke information to information in WCAP-14574-A which, again, is not an exercise valid to the Duke Application. The details of the design, including stress and fatigue analysis results, are contained in engineering documents maintained onsite and available for inspection. For the information requested in Item 3 above, refer to the Response to RAI 4.3-5 for additional discussion of fatigue reactor water effects and the Duke design.

Staff Concern - RAI 4.3-4(2) and (3) (Open Item)

Although WCAP-14574 is not a part of the Duke application, the report documented safety issues related to the Westinghouse fatigue design. As such, the staff needs to compare cumulative usage factors (CUFs) specific to certain McGuire and Catawba components to data presented in the report to conclude, with reasonable assurance, that fatigue of these components will be adequately monitored during the extended period of operation.

During a June 4, 2002, conference call, summarized by memorandum dated June 19, 2002 (ML021700621), the applicant acknowledged the reviewer's intent in requesting the CUFs for

McGuire and Catawba components. The applicant expressed that it disagrees with the technical approach in the WCAP report. However, the staff indicated that it needs this information to compare plant-specific data to that which is provided in Table 2-10 of the WCAP. Such a comparison will enable the staff to conclude, with reasonable assurance, that the components should be able to continue to perform their intended functions during the period of extended operation. The applicant further explained that the Thermal Fatigue Management Program is credited in the LRA to ensure that all fatigue-related issues are managed for the period of extended operation, including environmentally assisted fatigue. The staff acknowledged that this program was defined in the LRA and is acceptable, but that a data comparison to the WCAP report was requested to provide additional assurance that the applicant's program will adequately address the issues identified in the WCAP report.

The applicant agreed to provide specific CUFs associated with the design of the pressurizer as requested by RAI 4.3-4(2). With respect to RAI 4.3-4(3), the pressurizer locations associated with these usage factors, as well as other component locations in the reactor coolant system, will be considered for environmentally assisted fatigue under the process described in the application Section 4.3.1.2.

Duke Response to Staff Concern - RAI 4.3-4(2) and (3) (Open Item)

Duke wishes to make it clear that we do not agree that it is legitimate to draw conclusions regarding the assignment of fatigue sensitivity to component locations based solely on examinations of design basis cumulative usage factor (CUF) values. It was the intent of the engineer generating these design basis CUF values to provide a quantifying argument that the value does not exceed 1.0. It is impossible to know from a simple tabulation whether a given value is a conservative approximation derived by simplified methods, or whether it is the culmination of the application of an expensive sophisticated set of high technology methods and analytical tools.

A low value is often obtained by the expensive route, necessitated by the need to consider relatively severe transients applied to sensitive geometries. Conversely, a high value is often the result of a quick quantification to show that even with conservative assumptions and methods, the loads applied to the given component do not result in a CUF greater than 1.0. Therefore we wish to convey our opinion that great caution should be applied in the comparison of the attached tabulations of CUF to other tabulations, or among themselves, in establishing relative component fatigue sensitivity.

The following table provides the pressurizer fatigue cumulative usage factors for McGuire and Catawba:

	McGuire Nuclear Station		**Catawba Nuclear Station**	
	Unit 1	**Unit 2**	**Unit 1**	**Unit 2**
Upper Head	0.043	0.043	0.043	0.043
Shell	0.992	0.992	0.992	0.992
Spray Nozzle	0.827	0.827	0.821	0.821
Safety Relief Nozzle	0.060	0.060	0.058	0.058
Manway bolts	Exempt*			
Manway Pad				
Manway Cover				
Valve Support Bracket	0.267	0.267	0.255	0.255
Seismic Support Lugs	0.370	0.370	0.370	0.370
Lower head to shell weld	0.090	0.090	0.090	0.090
Lower head @ heater pen	0.960	0.960	0.960	0.960
Lower head to nozzle weld	0.450	0.450	0.450	0.450
Surge nozzle knuckle	0.581	0.581	0.581	0.581
Nozzle to safe end weld	0.114	0.114	0.114	0.114
Safe end to pipe weld	0.019	0.019	0.019	0.019
Instrument Nozzle	0.213	0.213	0.213	0.213
Support Skirt and Flange	0.267	0.267	0.255	0.255

*An analysis for cyclic service is not required if it can be demonstrated that the pressure fluctuations, temperature differences, and mechanical loads all fall within specified limits, as set forth in NB-3222.4(d) of the Code.

Duke's Response to RAI 4.3-5(2)

In response to Item (1), Duke confirms that the environmental fatigue correlations contained in NUREG/CR 6583, "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low Alloy Steels," and NUREG/CR 5704, "Effects of LWR Coolant Environments on Fatigue on Fatigue Design Curves of Austenitic Stainless Steels," will be used in the evaluation Duke must complete by year 2021. As stated in the Application, Duke may choose to exercise a different course of action should the NRC approve a less restrictive approach in the future, either through agreement with the industry, or individually with Duke.

In response to Item (2), the NUREG/CR-6260 locations applicable to McGuire and Catawba are identified below in Table 4.3-5, "Newer Vintage Westinghouse Plant Locations Identified in NUREG/CR-6260." The current design basis usage factors for each of these locations is less than one. The details of the design, including stress and fatigue analysis results, are contained in engineering documents maintained onsite and available for inspection.

Table 4.3-5 Newer Vintage Westinghouse Plant Locations Identified in NUREG/CR 6260

Reactor Vessel	At lower head to shell juncture Inlet Nozzle Outlet Nozzle
Surge Line	Hot Leg Nozzle
Charging Nozzle	Nozzle
Safety Injection Nozzle	Nozzle
Residual Heat Removal Line	Inlet Transition

In response to Item (3), Duke has stated that it may wish to use flaw tolerance and inspection procedures to validate a plant component for cyclic duty. Duke recognizes that the NRC staff has not endorsed a procedure on a generic basis which allows for flaw tolerance evaluations combined with ASME Section XI inspections in lieu of meeting the fatigue usage criteria.

Duke agrees not to use flaw tolerance/inspection procedures unless such procedures have been accepted by the NRC. At the appropriate time during the period of extended operation, if no procedure is as yet agreed to between the NRC and the industry or with Duke, Duke agrees to obtain concurrence from the NRC on the technical processes involved in such procedures on a case by case basis. As stated in the Application, Duke may choose to exercise a different course of action should the NRC approve a less restrictive approach in the future, either through agreement with the industry, or individually with Duke.

In response to Item (4), Duke recognizes the ongoing discussions between the industry and the NRC staff on the use of a Z factor with a value greater than 1.0 in the equations associated with fatigue reactor water effects. These discussions will likely result in clarifying the entire fatigue reactor water environmental effects issue sometime in the future. Since the specific issue of fatigue reactor water effects is only applicable to the period of extended operation (earliest start date for the extended period of operation for McGuire and Catawba is 2021 for McGuire 1), Duke anticipates that the equations specified as a part of the Application may be revised to better reflect the then current best practice. As stated in the Application, Duke may choose to exercise a different course of action should the NRC approve a less restrictive approach in the future, either through agreement with the industry, or individually with Duke. With this in mind, the Duke procedure specified in Application Section 4.3.1.2 will be revised to set Z factor equal to 1.0 unless a different value is warranted by then acceptable practice.

Staff Concern - RAI 4.3-5(2) (Confirmatory Item)

The staff was concerned that a review could not be completed without the requested design basis usage factors for each of the six component locations listed in NUREG/CR6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components." By electronic correspondence dated May 23, 2002, the applicant provided a table of CUFs for newer-vintage Westinghouse plant locations identified in NUREG/CR-6260. This Table was attached as an enclosure to the June 4, 2002, conference call summary, summarized by memorandum dated June 19, 2002 (ML021700621). The staff reviewed these data and determined that the information provided was sufficient to enable the staff to complete its review of this item. The applicant agreed to submit this information by official correspondence. This RAI is confirmatory pending the formal submittal of the information provided in the May 23, 2002, electronic correspondence.

Duke Response to Staff Concern - RAI 4.3-5(2) (Confirmatory Item)

Duke wishes to make it clear that we do not agree that it is legitimate to draw conclusions regarding the assignment of fatigue sensitivity to component locations based solely on examinations of design basis cumulative usage factor (CUF) values. It was the intent of the engineer generating these design basis CUF values to provide a quantifying argument that the value does not exceed 1.0. It is impossible to know from a simple tabulation whether a given value is a conservative approximation derived by simplified methods, or whether it is the culmination of the application of an expensive sophisticated set of high technology methods and analytical tools.

A low value is often obtained by the expensive route, necessitated by the need to consider relatively severe transients applied to sensitive geometries. Conversely, a high value is often the result of a quick quantification to show that even with conservative assumptions and methods, the loads applied to the given component do not result in a CUF greater than 1.0. Therefore we wish to convey our opinion that great caution should be applied in the comparison of the attached tabulations of CUF to other tabulations, or among themselves, in establishing relative component fatigue sensitivity.

The following table as previously provided by Duke by electronic communication on May 23, 2002 and attached to the June 4, 2002 conference call summary addresses the staff concerns:

CUFs Newer Vintage Westinghouse Plant Locations Identified in NUREG/CR 6260

	McGuire 1	McGuire 2	Catawba 1	Catawba 2
RV at lower head to shell juncture	0.004	0.059	0.059	0.012
RV Inlet Nozzle	0.107	0.107	0.099	0.112
RV Outlet Nozzle	0.658	0.658	0.658	0.658
Surge Line Hot Leg Nozzle	0.276	0.276	0.276	0.276
Charging Nozzle	0.768	0.768	0.795	0.795
Safety Injection Nozzle	0.935	0.935	0.950	0.950
Residual Heat Removal Line Inlet Transition	0.042	0.042	0.044	0.044

SER Open Item 4.3-2 (New Open Item)

The Catawba UFSAR lists a large number of design cycles for charging and letdown flow changes. The Duke response to RAI 4.3-5 indicates that these transients cause insignificant fatigue and are not counted. The staff notes that NUREG/CR-6260 contains a discussion of these transients for the newer vintage Westinghouse plant and indicates that these transients are not normally counted at PWRs, although some PWRs have reported that the actual cycles of these transients are less than the numbers assumed in the design calculations. However, the NUREG/CR-6260 evaluation indicates the fatigue usage at the charging nozzle for these transients is significant when the reactor water environment is considered. The charging nozzle is one of the locations Duke will assess for fatigue environmental effects. As such, Duke should provide the design stresses and fatigue usage factors associated with the Catawba charging system flow changes.

Duke Response to SER Open Item 4.3-2 (New Open Item)

The fatigue usage factors for the Catawba charging nozzles are provided in Duke Response to Staff Concern 4.3-5 (2).

The Duke response to the staff request to provide design stresses for these locations is deferred until after the staff issues the SER with open items.

SER Open Item 4.3-4 (New Open Item)

Duke provided a McGuire FSAR Supplement for Section 3.9.2 and a Catawba FSAR Supplement for Section 3.9.3 which indicates that stress range reduction factors were used in the evaluation of ASME Class 2 and 3 piping systems. Duke also provided a McGuire FSAR Supplement for Section 5.2.1 and a Catawba FSAR Supplement for Section 3.9.1 to indicate that the Catawba Thermal Fatigue Management Program (TFMP) will continue to manage thermal fatigue into the period of extended operation. However, Duke did not describe its commitment to evaluate the effects of the environment on fatigue of reactor coolant system pressure boundary components in the UFSAR Supplement. Nor did Duke provide a description of its TFMP. The FSAR Supplement should be revised to reflect this information.

Duke Response to SER Open Item 4.3-4 (New Open Item)

Duke has decided to defer the response to SER Open Item 4.3-4 (New Open Item) until after the staff issues the SER with Open Items to provide sufficient time for the responsible engineering staff at each station to be involved in the preparation of the response.

Duke's Response to RAI 4.3.6(1), (2), (3), and (4)

The Application does not address the issue of underclad cracks because underclad cracking was not identified as a time-limited aging analyses for McGuire or Catawba. In order to be considered as time-limited aging analyses, calculations or analyses must meet the six criteria contained §54.3. For McGuire and Catawba, no calculations or analyses were identified that considered the issue of underclad cracks for any period of time. The applicant indicated that all six of these criteria have not been met and, therefore, this issue is not a time-limited aging analysis for either McGuire or Catawba.

Regarding Item (1) in RAI 4.3-6, for McGuire Unit 2 and Catawba Unit 1, the vessel flange, upper shell course, nozzles, intermediate shell coarse, lower shell coarse, and the top head ring and flange (parts of the closure head assembly) were fabricated from SA 508 Class 2 forgings. The McGuire Unit 1 and Catawba Unit 2 closure head flange, vessel flange and the reactor vessel inlet and outlet nozzles are fabricated from SA 508 Class 2 forgings.

Regarding Item (2) in RAI 4.3-6, manufacturing records for the forgings that describe the method of cladding application have not been located. Therefore, a conservative assumption has been made that the forgings are potentially susceptible to underclad cracking.

For Item (3), the SA 508 Class 2 forgings that are subject to neutron fluence greater than or equal to 10^{17} neutrons/cm2 (E > 1 MeV) are:

- Intermediate shell and the lower shell for McGuire Unit 2 and Catawba Unit 1
- Inlet and outlet nozzles for McGuire Unit 2 and Catawba Unit 1 [Estimated fluence is below 3×10^{17} neutrons/cm2 (E > 1 MeV)]
- Inlet nozzles for McGuire Unit 1 and Catawba Unit 2 [Estimated fluence is below 3×10^{17} neutrons/cm2 (E > 1 MeV)]

For Item (4), underclad cracking typically occurs only in the grain-coarsened region of the base metal heat-affected zone at the weld bead overlap. The subsurface location and the size of these cracks make them difficult to detect using standard non-destructive examination methods. Detection normally requires destructive examination through removal of the cladding to the weld fusion line and examination of the underlying base metal. In May 1973, the NRC issued Regulatory Guide 1.43 to address underclad cracking. Regulatory Guide 1.43 includes recommended controls that may be used to limit the occurrence of underclad cracking in low-alloy steel Class 1 components. As identified in McGuire UFSAR Table 1-4 and Catawba UFSAR Section 1.7, Regulatory Guide 1.43 was adopted for the fabrication of McGuire and

Catawba Class 1 components and it is unlikely that these components fabricated from SA 508 Class 2 material contain the subject fabrication flaws.

In order to provide additional assurance that underclad cracking is not a concern during the period of extended operation, a bounding analysis for all Westinghouse plants is contained in WCAP-15338, "A Review of Cracking Associated with Weld Deposited Cladding in Operating PWR Plants," that has been previously prepared and submitted to the NRC. In its safety evaluation report dated October 15, 2001, the NRC found WCAP-15338 acceptable for referencing in license renewal applications. WCAP-15338 provides flaw evaluations based on Section XI of the American Society of Mechanical Engineers Code to justify that the Westinghouse reactor pressure vessels with underclad cracks are acceptable for operation for 60 years.

The staff safety evaluation report for WCAP-15338 includes two renewal applicant action items that must be addressed. Renewal Applicant Action Item (1) requires the license renewal applicant to verify that its plant is bounded by the WCAP-15338 report. On January 22, 2002 Westinghouse submitted a letter to the Document Control Desk stating that, "The 3-loop RPV evaluation presented in the report is intended to be a bounding evaluation for all Westinghouse plant sizes, including both 2-loop and 4-loop RPVs." Duke confirms that the McGuire and Catawba vessels are bounded by WCAP-15338. The analysis presented in WCAP-15338 provides additional assurance that underclad cracks will not result in a loss of reactor vessel integrity during the period of extended operation for both McGuire and Catawba.

Renewal Applicant Action Item (2) requires license renewal applicants referencing WCAP-15338 to ensure that the evaluation of the TLAA is summarily described in the FSAR Supplement. For the reasons discussed above, the issue of underclad cracks is not a time-limited aging analysis for either McGuire or Catawba. Therefore, no summary description is required to be included in the UFSAR Supplements.

Staff Concern - RAI 4.3.6(1), (2), (3), and (4) (Open Item)

The staff reviewed the Catawba Updated Final Safety Analysis Report (UFSAR), Section 1.7, Regulatory Guides, and Section 5.3.1.4, Special Controls for Ferritic and Austenitic Stainless Steels, and determined that sufficient information was provided in the UFSAR to conclude that underclad cracking was not a concern for Catawba 1 and 2. However, corresponding sections of the McGuire UFSAR do not provide sufficient information for the staff to conclude that underclad cracking is not a concern for McGuire Units 1 and 2. As such, the staff does not have sufficient information about the McGuire 2 fabrication process to conclude that underclad cracking is not a concern. If the staff does not have conclusive evidence that the fabrication procedure does not result in underclad cracking, the applicant can provide an analysis for the license renewal term.

Duke Response to Staff Concern - RAI 4.3.6(1), (2), (3), and (4) (Open Item)

Note: The McGuire 1 and Catawba 2 vessels were both manufactured by Combustion Engineering and the McGuire 2 and Catawba 1 vessels were both manufactured by Rotterdam-Nuclear of the Netherlands, and all vessels were manufactured in about the same timeframe: mid to late 1970's.

In the initial response to RAI 4.3-6, Item(2), Duke made a conservative assumption that the SA 508 Class 2 forging may be susceptible to underclad cracking. Based on a review of the licensing and design basis of McGuire Nuclear Station, Units 1 and 2, performed subsequent to our April 15, 2002 submittal, Duke now confirms that the forgings do not have underclad cracks in the nozzles resulting from fabrication.

Furthermore, Duke does not agree that underclad cracking is a time-limited aging analysis that is required to be addressed for license renewal. Duke has found no indication that a calculation or analysis has been performed on McGuire documenting a flaw-growth analysis (as had been performed for previous license renewal applicants) nor has Duke found any indications that underclad cracks due to fabrication existed in the reactor vessel nozzle region that did not meet the applicable ASME XI preservice examination acceptance criteria.

The results of the Duke review and the basis for our position is provided in the following. These results provide conclusive evidence that the fabrication procedure did not result in underclad cracking for McGuire Nuclear Station Units 1 and 2. For the convenience of the reader, copies of the documents that are identified in the McGuire Licensing and Design Basis section of this response are being provided to the NRC staff.

BACKGROUND

Prior to describing the McGuire licensing and design basis, Duke would like to provide a summary of the underclad cracking issue for early vintage plants. Previous license renewal applicants have indicated that the issue of underclad cracking in reactor vessels was initially identified in 1970 when it was first discovered at a European vessel fabricator. Intergranular separations in low alloy steel heat-affected zones under austenitic stainless steel weld claddings were detected in SA-508, Class 2 reactor vessel forgings manufactured to a coarse grain practice. Cladding was by high-heat-input submerged processes. Two reports document the history of this issue for early vintage reactor vessels designed and fabricated pre-1974: BAW-2251- A, "Demonstration of the Management of Aging Effects for the Reactor Vessel," Appendix C, Fracture Mechanics Analysis of Postulated Underclad Cracks in B&W Designed Reactor Vessels for 48 EFPY (August 1999) and WCAP-15338, A Review of Cracking Associated with Weld Deposited Cladding in Operating PWR Plants," March 2000. The approach to address the issue of underclad cracking in these reactor vessels was to perform a fracture mechanic analysis to demonstrate that the critical crack size required to initiate fast fracture was several orders of magnitude greater than the assumed maximum flaw size plus flaw growth due to design fatigue cycles. For license renewal applicants with early vintage reactor vessels, reports BAW-2251-A and WCAP-15338 along with the applicant responses to applicant actions items contained within each report, provide the technical analysis that supports the continued operation of these vessels through 60-years.

MCGUIRE LICENSING AND DESIGN BASIS

For McGuire reactor vessels, the approach to address the issue of underclad cracking in reactor vessels is substantially different. The approach for these newer vintage vessels begins in December 1979 when Westinghouse, the NSSS vendor of McGuire, notified the NRC of cracking that had been recently identified in reactor vessel nozzles by the Westinghouse French licensee - Framatome. In a letter dated December 13, 1979 to the NRC, Westinghouse defined the issue as follows:

> Cracks have been detected by the Westinghouse French licensee in the heat affected zone (HAZ) of the low alloy steel base material of reactor vessel nozzles after performing the cladding operation on the bore of the nozzles. The crack have been detected by both destructive and non-destructive (Ultrasonic testing) examinations. The cracks appear to be confined to the HAZ produced by the second layer of cladding. They are oriented perpendicular to the cladding direction. The cracks exist in a broad area of the nozzle bore, but are more prevalent in the thicker section of the nozzle. The crack are relatively small with a maximum length of 1.0 inch and a maximum depth of 0.280 inches.
>
> In order to detect the cracks, the French have made use of a twin transducer 70° angle beam UT technique. This technique differs from the 45° and 60° UT techniques commonly used for in-service inspections in the United Sates.

> The French believe that the cracking is hydrogen-induced and may be associated with manganese inclusions and/or carbon segregation in the base metal. The hydrogen introduction is believed to be the result of the welding process/heat treatment used to clad the carbon steel base metal with stainless steel.

During the initial licensing of McGuire, the NRC, in a letter dated May 12, 1980, informed Duke of the results of recent ultrasonic examinations of other U.S. reactor vessels and specifically requested the following information relative to McGuire Nuclear Station, Units 1 and 2:

> To better define the potential for cracking in the reactor vessel nozzles, we [NRC] request that you [Duke] provide us with the following information regarding each of the McGuire reactor vessels:
>
> - Nozzle base metal material specification type and grade.
> - Clad process type, electrode size and number of clad layers.
> - Heat input (amps, voltage, speed) for each clad layer.
> - Clad pre- and post-heat temperature and interpass temperature for each clad layer.
> - Manufacturer or subcontractor who fabricated vessel and applied nozzle cladding

In a letter dated June 6, 1980, Duke provided the requested information to the NRC. Subsequently, the NRC in a letter dated July 17, 1980 stated the following:

> We have reviewed the information provided in you June 6, 1980 letter regarding the reactor vessel nozzles at the McGuire Station.
>
> We have determined that the McGuire Unit No. 2 reactor vessel was fabricated by Rotterdam-Nuclear of the Netherlands using procedures for welding and pre- and post-clad heat treatments that increase the potential for underclad cracking. For this reason, we require that augmented ultrasonic examination for underclad cracking be performed on the McGuire unit No. 2 reactor vessel nozzles prior to issuance of an operating license. The inspections should be conducted using techniques that have been designed to detect underclad cracks. These techniques previously have been used at Sequoyah 1, North Anna 2 and Salem 2. The McGuire Unit No. 1 vessel was fabricated by Combustion Engineering using welding heat treat practices expected not to cause underclad cracking. Therefore, we do not require that augmented preservice inspections be performed on the Unit No. 1 vessel. In the future augmented ultrasonic examinations will be required for a reactor vessel whose nozzles were clad in the U. S., but only as part of a program to verify that cladding heat treatments used by U. S. manufacturers do not result in underclad cracking.

In its Safety Evaluation Report related to the operation of McGuire Nuclear Station Units 1 and 2, NUREG-0422, Supplement 4, January 1981, the NRC restated the above requirement to perform an augmented ultrasonic examination on the McGuire Unit 2 reactor vessel nozzles prior to issuance of the operating license.

Even though this NRC requirement was issued in January 1981, Duke had actually performed the required examination in October 1980, prior to the issuance of Supplement 4 to

NUREG-0422. The results of this examination is provided in the following Westinghouse report:

> "Assessment of Ultrasonic Reflections in the McGuire Unit II Reactor Vessel Nozzle Bores," October 28, 1980.

(Note that an augmented volumetric examination was also performed on Catawba Unit 1 reactor vessel nozzles and the results are documented in "Assessment of Ultrasonic Reflections in the Catawba Unit I Reactor Vessel Nozzle Bores," October 28, 1980.)

Copies of both of these reports are being provided to the staff (While these two reports do not contain report numbers, they are titled the same, are of the same time-frame and have the same author as References 19 and 21 of WCAP-15338.) Each report contains not only the assessment of the ultrasonic reflectors detected but also the nozzle cladding procedure, the ultrasonic examination procedure, and the calibration and raw data sheets and ultrasonic reflector plots.

The entire nozzle bore of each nozzle was ultrasonically examined. Inspection results from all eight McGuire Unit 2 nozzles are summarized as follows: Three outlet nozzles (Loops 1, 3 and 4) and two inlet nozzles (Loops 3 and 4) contained no reportable indications. The Loop 2 outlet nozzle contained two reportable indications characterized as slag or lack of bond. The Loop 1 inlet nozzle contained three reportable indications, one of which is considered suspect. The other two are characteristic of slag or lack of bond. The Loop 2 inlet nozzle contains one suspect location.

The suspect locations were evaluated in the report. Based on the evaluation documented in these two reports, Westinghouse concluded that all recorded reflectors met the preservice acceptance criteria of IWB-3500, ASME Section XI, 1977 Edition. These examinations apparently resolved any concerns the NRC had with the McGuire reactor vessel nozzles because no further NRC requests were made.

Subsequently, by letter dated February 18, 1983 and prior to the issuance of the Unit 2 operating license in March 1983, Duke reported to the NRC that in fact the required augmented ultrasonic examinations for underclad cracking had been performed and that indications noted during the examination were evaluated and it was determined that above preservice acceptance criteria were met. No other records were found after February 1983 pertaining to either McGuire or Catawba on the subject of underclad cracking in reactor vessel nozzles.

CONCLUSION

Based on the review of the licensing and design basis of McGuire, Duke has determined conclusively that reactor vessel underclad cracking is not a time-limited aging analysis for either unit at McGuire. Therefore, the staff request that Duke address renewal applicant action items in WCAP-15338 is inappropriate. WCAP-15338 does not apply to McGuire because there is no calculation or analysis within the licensing basis of either McGuire unit that addresses the postulated growth of underclad cracks over the licensed life of the vessel.

Rather, the approach used and accepted by NRC during the initial licensing of McGuire Unit 2 to address underclad cracks in the reactor vessel nozzles was the requirement to perform augmented ultrasonic examinations of the nozzles. These examinations were evaluated and it was determined that all detected indications met the preservice examination requirements of ASME Section XI, 1977 Edition. Furthermore, the NRC, in NUREG-0422, Supplement 4, determined that the McGuire Unit 1 vessel was fabricated by Combustion Engineering using welding heat treating practices expected not to cause underclad cracking. The SER goes on to state: "We [NRC] do not anticipate any significant underclad cracking in the McGuire Unit No. 1 reactor vessel nozzles and therefore do not require augmented preservice inspections."

Based on all of the above, Duke does not believe that the inlet and outlet nozzles of McGuire are susceptible to underclad cracking. Furthermore, the issue of underclad cracking in reactor vessel nozzles of McGuire was effectively resolved by the NRC during the initial licensing process. Finally, the issue is clearly not a time-limited aging analysis for license renewal.

SER Open Item 4.4-1 (New Confirmatory Item)

Generic Safety Issue (GSI) -168, "Environmental Qualification of Electrical Equipment," was developed to address environmental qualification of electrical equipment. The staff guidance to the industry (letter dated June 2, 1998 from NRC (Grimes) to NEI (Walters) states:

- GSI-168 issues have not been identified to a point that a license renewal applicant can be reasonably expected to address these issues, specifically at this time; and

- An acceptable approach is to provide a technical rationale demonstrating that the CLB for EQ will be maintained in the period of extended operation.

For the purpose of license renewal, as discussed in the statement of considerations (60 FR22484, May 8, 1995), there are three options for addressing issues associated with a GSI:

- If the issue is resolved before the renewal application is submitted, the applicant can incorporate the resolution into the LRA.

- An applicant can submit a technical rationale that demonstrates that the CLB will be maintained until some later point in the period of extended operation, at which time one or more reasonable options would be available to adequately manage the effects of aging.

- An applicant can develop a plant-specific aging management program that incorporates a resolution to the aging issue.

For addressing issues associated with GSI-168, "Environmental Qualification of Electrical Components," the applicant did not provide information in Section 4.4. However, the applicant provided the following discussion in electronic correspondence on June 17, 2002, to address this issue:

> As discussed in SECY-93-049, the staff reviewed significant license renewal issues and found that several were related to environmental qualification (EQ). A key aspect of these issues was whether the licensing bases should be reassessed or enhanced in connection with license renewal, and whether this reassessment should be extended to the current license term. In late 1993, the Commissioners instructed the staff that the current EQ licensing basis must be used in the license renewal period and that any EQ concerns identified by the staff during the review of EQ for license renewal should be evaluated for the effect on current licenses, independent of license renewal.
>
> The NRC Staff's EQ Task Action Plan (EQ-TAP) was initiated to address the adequacy of current EQ practices. Upon completion of the EQ-TAP review, the focus of Staff concerns was limited to

issues related to the adequacy of accelerated aging practices in existing qualifications, and the lack of a "feedback mechanism" in EQ programs (i.e., programmatic requirements to determine the current condition of EQ equipment so that it can be evaluated against the assumptions and parameters for qualification). The EQ-TAP was subsequently closed and six remaining open issues were incorporated into GSI 168 for management tracking purposes. The EQ-TAP review did not identify any generic safety issues related to these six open issues.

NRC guidance for addressing GSI 168 for license renewal is contained in a June 1998 letter to NEI . In this letter, the NRC states: 'With respect to addressing GSI 168 for license renewal, until completion of an ongoing research program and staff evaluations, the potential issues associated with GSI 168 and their scope have not been defined to the point that a license renewal applicant can reasonably be expected to address them at this time. Therefore, an acceptable approach described in the SOC is to provide a technical rationale demonstrating that the current licensing basis for EQ pursuant to 10 CFR 50.49 will be maintained in the period of extended operation. Although the SOC also indicates that an applicant should provide a brief description of one or more reasonable options that would be available to adequately manage the effects of aging, the staff does not expect an applicant to provide the options at this time.'

Environmental qualification evaluations of electrical equipment are identified as time-limited aging analyses for McGuire and Catawba. The McGuire and Catawba EQ program evaluations contained in Section 4.4 of the Application are considered to be the technical rationale that the current licensing basis will be maintained during the period of extended operation. Consistent with the above NRC guidance, no additional information is required to address GSI 168 in a renewal application at this time.

Pending the staff's receipt of this information in official correspondence, this item is characterized as confirmatory.

Duke Response to SER Open Item 4.4-1 (New Confirmatory Item)

Note: SRP-LR, NUREG 1800, Appendix A-3 suggests that any one of four approaches may be used to address GSIs.

The following is the response Duke provided in electronic correspondence on June 17, 2002 which address SER Open Item 4.4-1 (New Confirmatory Item):

GSI 168 - Environmental Qualification of Electrical Components
As discussed in SECY-93-049, the staff reviewed significant license renewal issues and found that several were related to environmental qualification (EQ). A key aspect of these issues was whether the licensing bases should be reassessed or enhanced in connection with license renewal, and whether this reassessment should be extended to the current license term. In late 1993, the Commissioners instructed the staff that the current EQ licensing basis must be used in the license renewal period and that any EQ concerns identified by the staff during the review of EQ for license renewal should be evaluated for the effect on current licenses, independent of license renewal.

The NRC Staff's EQ Task Action Plan (EQ-TAP) was initiated to address the adequacy of current EQ practices. Upon completion of the EQ-TAP review, the focus of Staff concerns was limited to issues related to the adequacy of accelerated aging practices in existing qualifications, and the lack of a "feedback mechanism" in EQ programs (i.e., programmatic requirements to determine the current condition of EQ equipment so that it can be evaluated against the assumptions and parameters for qualification). The EQ-TAP was subsequently closed and six remaining open issues were incorporated into GSI 168 for management tracking purposes. The EQ-TAP review did not identify any generic safety issues related to these six open issues.

NRC guidance for addressing GSI 168 for license renewal is contained in a June 1998 letter to NEI . In this letter, the NRC states:

> "With respect to addressing GSI 168 for license renewal, until completion of an ongoing research program and staff evaluations, the potential issues associated with GSI 168 and their scope have not been defined to the point that a license renewal applicant can reasonably be expected to address them at this time. Therefore, an acceptable approach described in the SOC is to provide a technical rationale demonstrating that the current licensing basis for EQ pursuant to 10 CFR 50.49 will be maintained in the period of extended operation. Although the SOC also indicates that an applicant should provide a brief description of one or more reasonable options that would be available to adequately manage the effects of aging, the staff does not expect an applicant to provide the options at this time."

Environmental qualification evaluations of electrical equipment are identified as time-limited aging analyses for McGuire and Catawba. The McGuire and Catawba EQ program evaluations contained in Section 4.4 of the Application are considered to be the technical rationale that the current licensing basis will be maintained during the period of extended operation. Consistent with the above NRC guidance, no additional information is required to address GSI 168 in a renewal application at this time.

Duke's Response to RAI B.3.19-2

Based on a review of industry literature on the topic of medium-voltage cables being exposed to moisture for long periods, no quantifiable data was found in the documents. However, the data and discussions in this industry literature (for example, EPRI TR-103834-P1-2, Effects of Moisture on the Life of Power Plant Cables, and SAND96-0344, Aging Management Guideline for Commercial Nuclear Power Plants – Electrical Cable and Terminations, which are referenced in GALL Report Program XI.E3) provides the reader with the general conclusion that there should not be a problem with a medium-voltage cable even if it is exposed to moisture for several years.

The GALL Report incorporated all previous operating experience into program XI.E3. The general conclusion that there should not be a problem with a medium-voltage cable even if it is exposed to moisture for several years is reflected in statements in the GALL Report such as (underlines added for emphasis):

> **GALL Report Program XI.E3**
> **Preventive Actions**: ...operating experience indicates that prolonged exposure to moisture and voltage are required to induce this aging mechanism.

Prolonged exposure by any definition is more than a few days. The prolonged nature of the aging effects of concern in this program and the acceptability of an inspection period of "a few years" (as in the McGuire and Catawba program) is further recognized in the GALL Report with statements such as (underlines added or emphasis):

> **GALL Report Program XI.E3**
> **Detection of Aging Effects**: In-scope, medium-voltage cables exposed to significant moisture and significant voltage are tested at least once every 10 years. This is an adequate period to preclude failures of the conductor insulation since experience has shown that aging degradation is a slow process. ...

The GALL Report states in the quote above that cables exposed to significant moisture for up to 10 years are not expected to fail. This is in agreement with the statements that prolonged exposure is required in order for the aging mechanisms to be "induced" and that "experience has shown that aging degradation is a slow process". The McGuire and Catawba Inaccessible Non-EQ Medium-Voltage Cables Aging Management Program defines significant moisture as "exposure to long-term (over a long period such as a few years), continuous standing water" because longer periods of exposure without action are accepted in GALL Report Program XI.E3.

Staff Concern - RAI B.3.19-2 (Open Item)

The staff notes that the applicant's reference (SAND96-0344, Aging Management Guideline for Commercial Nuclear Power Plants - Electrical Cable and Termination) states in Section 4.1.2.4:

> Note, however, that even minor and/or intermittent surface condensation, in conjunction with voltage stress and contaminants, may create an environment where surface cracking may occur. Furthermore, some evidence exists to indicate that the rate of diffusion of water through a polymer is relatively independent of form [4.38]. Therefore, the water diffusion rate for a "dry" material in a 100 RH atmosphere may not be much different than that for the same material completely submerged in water.

It is not clear to the staff that inaccessible cables exposed to moisture for a period of "a few years" is not significant. The applicant's response did not resolve the issue of cable exposure to wet conditions for which they are not designed.

Duke Response to Staff Concern - RAI B.3.19-2 (Open Item)

Duke agrees with the staff on this point. To resolve this item, Duke has eliminated the qualifier "significant" when describing moisture with regards to the program. The program now takes a bounding approach by stating, "Cables that are direct buried, run in horizontally-run buried conduit or run in outside cable trenches are assumed to be exposed to standing water." In-scope medium-voltage cables that are exposed to standing water and also exposed to significant voltage will be tested. Please see the revised *Inaccessible Non-EQ Medium-Voltage Cables Aging Management Program* immediately following Duke's response to Potential Open Item B.3.19.2-1 (New Open Item).

Potential Open Item B.3.19.2-1 (New Open Item)

The applicant's Inaccessible Non-EQ Medium-Voltage Cables Aging Management Program description did not provide adequate information about the proposed alternate inspection program to testing. It did not specify (1) the frequency of inspection; (2) how inspection results will be monitored and trended; (3) if or when operability evaluations for degraded conditions (presence of moisture) would be performed; (4) if or when testing would be performed if moisture is identified, and (5) what corrective action would be taken in the event that cables exposed to moisture are identified.

Duke Response to Potential Open Item B.3.19.2-1 (New Open Item)

The alternative visual inspective program was proposed in the McGuire and Catawba LRA in an attempt to provide a distinction between cables that are exposed to moisture (rain and drain) and those that are exposed to "significant" moisture so that the cables exposed only to "rain and drain" would not require testing. Trying to quantify this distinction has proven difficult and has raised staff concerns that this distinction, improperly applied, could inadvertently exclude some applicable cables from the program. Duke acknowledges the staff's concern in this area along with the recognition that some cable installations make it impossible (by currently known means) to verify with reasonable assurance that all portions of some cable runs are not continuously exposed to moisture. Considering these factors, Duke has now eliminated this distinction regarding moisture exposure by taking a bounding approach. The aging management program will include any significant voltage exposed in-scope medium-voltage cables that are exposed to standing water (for any period of time). With the moisture distinction eliminated and all such cables included without further qualification, the need for the proposed alternative inspection program is eliminated.

The resulting program is consistent with the aging management agreed to by the staff in Section 3.9.3.2 of NUREG-1723, *Safety Evaluation Report Related to the License Renewal of Oconee Nuclear Station,* regarding the aging management of the Oconee medium-voltage cables with the greatest safety significance – the direct buried 13.8kV cables, Underground Emergency Power Path from the Keowee Hydro Station.

The *Inaccessible Non-EQ Medium-Voltage Cables Aging Management Program*, with the above program changes, satisfies the stated concerns of the staff and is consistent with the aging management practices as approved by the NRC for similar cables at Oconee.

The revised *Inaccessible Non-EQ Medium-Voltage Cables Aging Management Program* is provided below:

Inaccessible Non-EQ Medium-Voltage Cables Aging Management Program
Note: The Inaccessible Non-EQ Medium-Voltage Cables Aging Management Program is applicable to both McGuire Nuclear Station and Catawba Nuclear Station.

The purpose of the *Inaccessible Non-EQ Medium-Voltage Cables Aging Management Program* is to provide reasonable assurance that the intended functions of medium-voltage cables within the scope of the program will be maintained in accordance with the current licensing basis during the period of extended operation.

Scope – The scope of the *Inaccessible Non-EQ Medium-Voltage Cables Aging Management Program* includes inaccessible non-EQ medium-voltage cables within the scope of 10 CFR 54.4 that are exposed to significant voltage and to standing water (for any period of time).

Key Definitions and Assumptions: Inaccessible cables are those that are not able to be approached and viewed easily, such as in conduits or cable trenches; all others are accessible. A cable that has a portion of the cable routing that is inaccessible is an inaccessible cable. Non-EQ means not subject to 10 CFR 50.49 Environmental Qualification requirements. Medium-voltage cables are those applied at a system voltage greater than 2kV. Significant voltage is defined as exposure to system voltage for more than twenty-five percent of the time. Cables that are direct buried, run in horizontally-run buried conduit or run in outside cable trenches are assumed to be exposed to standing water.

Preventive Actions – Preventive actions are not included in the *Inaccessible Non-EQ Medium-Voltage Cables Aging Management Program.*

Parameters Monitored or Inspected – Medium-voltage cables within the scope of the *Inaccessible Non-EQ Medium-Voltage Cables Aging Management Program* are tested to provide an indication of the condition of the conductor insulation. The specific type of test performed will be determined before each test. Each test performed for a cable may be a different type of test.

Detection of Aging Effects – Medium-voltage cables within the scope of the *Inaccessible Non-EQ Medium-Voltage Cables Aging Management Program* are tested at least once every 10 years. This is an adequate frequency to preclude failures of the conductor insulation.

Monitoring & Trending – Trending actions are not included in the *Inaccessible Non-EQ Medium-Voltage Cables Aging Management Program.*

For McGuire, the first test per the *Inaccessible Non-EQ Medium-Voltage Cables Aging Management Program* will be completed following issuance of renewed operating licenses for McGuire Nuclear Station and by June 12, 2021 (the end of the initial license of McGuire Unit 1). For Catawba, the first test per the *Inaccessible Non-EQ Medium-Voltage Cables Aging Management Program* will be completed following issuance of renewed operating licenses for Catawba Nuclear Station and by December 6, 2024 (the end of the initial license of Catawba Unit 1).

Acceptance Criteria – The acceptance criteria for each test is defined by the specific type of test performed and the specific cable tested.

Corrective Actions & Confirmation Process – Further investigation through the corrective action program is performed when the acceptance criteria are not met. When an unacceptable condition or situation is identified, a determination is made as to whether the same condition or situation is applicable to other medium-voltage cables within the scope of this program. Confirmatory actions, as needed, are implemented as part of the corrective action process.

Administrative Controls – The *Inaccessible Non-EQ Medium-Voltage Cables Aging Management Program* is controlled by plant procedures.

Operating Experience – Operating experience is not relevant for this new program.

Conclusion

The *Inaccessible Non-EQ Medium-Voltage Cables Aging Management Program* is equivalent to the program described and evaluated in Section 3.9.3.2 of NUREG 1723 [Reference 1]. The above review demonstrates that the *Inaccessible Non-EQ Medium-Voltage Cables Aging Management Program* provides reasonable assurance that the intended functions of medium-voltage cables within the scope of the program will be maintained in accordance with the current licensing basis during the period of extended operation.

References

1. NUREG-1723, *Safety Evaluation Report Related to the License Renewal of Oconee Nuclear Station*, Units 1, 2, and 3, March 2000, U. S. Nuclear Regulatory Commission, Docket Nos. 50-269, 50-270, and 50-287.

2. 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, Letter to U.S. NRC, April 15, 2002.

Duke's Response to RAI B.3.27-2

The Reactor Vessel Internals Inspection is a program that is completely separate from the Inservice Inspection Plan. As described in Section B.27 of the Application, the Reactor Vessel Internals Inspection has been developed to supplement the Inservice Inspection Plan and is separate from and in addition to the VT-3 examinations currently required by examination category B-N-3.

The Reactor Vessel Internals Inspection includes several inspections and examinations. For items comprised of plates, forgings, and welds that will be visually inspected, critical crack size will be determined by analysis. Acceptance criteria for all aging effects will be developed prior to the inspection. The visual inspection method will be sufficient to detect the critical crack size determined by analysis.

Currently an inspection for McGuire Unit 1 is planned during the fifth inservice inspection interval (approximately between forty and fifty years of operation). The decision of whether to perform inspections on McGuire Unit 2, Catawba Unit 1 and Catawba Unit 2 and when to perform such inspections will depend on an evaluation of the results of the internals inspections performed at Oconee and on McGuire Unit 1. Refer to the discussion in response to RAI B.27-1 for more details on the relevance of the Oconee experience.

Staff Concern - RAI B.3.27-2 (Confirmatory Item)

The staff requests that Duke confirm that the intent of the last sentence in the second paragraph of its response be clarified to state that the visual inspection method selected for the inspection of RV internal plates, forging, and welds will be sufficient to detect cracks in the components prior to any growth to a size that is greater than the critical crack size (critical crack length) for the material.

Duke Response to Staff Concern - RAI B.3.27-2 (Confirmatory Item)

Duke confirms that the intent of the last sentence in the second paragraph of its response should be clarified to state:

> "The visual inspection method selected for the inspection of RV internal plates, forging, and welds will be sufficient to detect cracks in the components prior to any growth to a size that is greater than the critical crack size (critical crack length) for the material."