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A Duke Energy Company

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October 2, 2002

U.S. Nuclear Regulatory Commission
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Washington, DC 20555

Subject: Comments on the U.S.N.R.C Safety Evaluation with Open Items Related to the License Renewal 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

By letter dated June 13, 2001, Duke Energy Corporation (Duke) submitted an Application to Renew the Facility Operating Licenses of McGuire Nuclear Station and Catawba Nuclear Station (Application). The Application contains the technical information required by 10 CFR Part 54 and the Supplement to the Final Safety Analysis Report (FSAR) for each station as required by §54.21(d). In a letter dated August 14, 2002, the NRC staff provided Duke a copy of the "Safety Evaluation Report with Open Items Related to the License Renewal of McGuire Nuclear Station, Units 1 and 2, Catawba Nuclear Station, Units 1 and 2." This staff letter requested that Duke review the enclosed safety evaluation report (SER), verify its accuracy, provide comments, and respond to the open and confirmatory items by October 27, 2002.

In response to this staff request, Duke is providing an interim response that is intended to expedite the staff completion of its review and to support the staff in its presentation to the Advisory Committee on Reactor Safeguards, currently scheduled for October 8, 2002. Remaining open and confirmatory items identified in the SER that are not addressed herein, as well as Duke comments on the SER and revised UFSAR Supplements for McGuire and Catawba, will be submitted to the staff on or before October 27, 2002.

Attachment 1 contains the Duke responses to the Open Items and Confirmatory Items for electrical related items. These items were discussed informally with the staff on September 17, 2002. Some of the informal responses have been revised based on these discussions with the staff.

Attachment 2 contains the Duke responses to the Open Items and Confirmatory Items for thermal fatigue related items. These items were discussed informally with the staff on September 18, 2002. Some of the informal responses have been revised based on these discussions with the staff.

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Attachment 3 contains the Duke responses to the Open Items and Confirmatory Items for structural related items. These items were discussed informally with the staff on September 18, 2002. Some of the informal responses have been revised based on these discussions with the staff.

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

Very truly yours,

M. S. Tuckman

M. S. Tuckman

Attachments:

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 response to the Safety Evaluation with Open Items Related to the License Renewal 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, 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 2nd day of October 2002.

Michael T. Cash
Notary Public

My Commission Expires:
January 22, 2003

MICHAEL T. CASH
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xc: (w/ Attachment)

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Attachment 1

Response to
McGuire Units 1 & 2 and Catawba Units 1 & 2
Safety Evaluation Report with Open Items

Electrical Related Items

Open Item 2.5-1 By letter dated June 26, 2002, the applicant provided AMR results for the passive, long-lived structures and components associated with the offsite power path. Pending completion of the staff's review of this information, this item is characterized as open.

By electronic communication on September 4, 2002, the NRC staff provided the following:

> EEIB has reviewed the June 26, 2002, Duke response which provides the results of the AMR for structures and components relied upon to restore power from offsite sources following station blackout. While transmission conductors were included in the scope and subject to an AMR, the staff noted that the aging management review results did not include any insulated control or power cables associated with the switchyard, transformer station, and relay house. The submittal did however, identify switchyard cable trenches, and transformer station cable trenches, providing routing, support and protection to cables servicing the switchyard and relay house equipment. This item remains open pending the determination of whether the insulated power and control cables associated with the switchyard, transformer station, and relay house are in scope for SBO recovery.

Duke Response to Open Item 2.5-1 for Electrical Related Items

Insulated cables and connections were not specifically addressed in the SBO open item response, but the response did not change the fact that no scoping was performed for cables as stated in Application Sections such as 2.1.1.1.3, 2.1.1.2.3 and 2.1.1.3.5:

> "No scoping was performed for insulated cables and connections and all insulated cables and connections are in scope as part of a bounding scope."

All insulated cables and connections (power, control and instrumentation applications) installed in the additional areas identified in the SBO open item response were, and still are, in scope as part of a bounding scope. The maximum cable voltage at either station is 13.8kV. The cables in these additional areas are included in the aging management review for insulated cables and connections submitted in the June 2001 License Renewal Application. This June 2001 cable aging management review is a bounding review that included all cables installed in these additional areas and structures (the areas and structures now identified as being within scope).

The *Inaccessible Non-EQ Medium Voltage Cables Aging Management Program* and the *Non-EQ Insulated Cables and Connections Aging Management Program* will manage the aging effects of the cables that meet the scope of each program installed in these additional areas and structures.

Open Item 3.6.1-1 The applicant should provide a technical justification that will demonstrate that visual inspection of high range radiation monitor and high voltage neutron monitoring instrumentation cables will be effective in detecting aging before current leakage can affect instrument loop accuracy.

Duke Response to Open Item 3.6.1-1
Open Item 3.6.1-1 asks to justify the adequacy of visual inspections, or to say it another way: How we know that "looking" is good for these specific cables.

The primary function of high range radiation monitor and neutron monitoring circuits is to measure the amount of radiation in an area[1] and provide this information for plant operations.

Most reported failures of high range radiation monitor and neutron monitoring cables and connections have occurred in portions of the instrumentation circuits installed in the Reactor Building and specifically those portions located in close proximity to the reactor pressure vessel.[2] SER Open Item 3.6.1-1 pertains only to non-EQ cables. All areas within the Reactor Buildings are EQ harsh environment areas.[3] Portions of the high range radiation monitor and neutron monitoring instrumentation circuits located in the Auxiliary Buildings and other plant areas are routed in the normal open cable tray system along with other accessible cables included in the visual inspections program.

As discussed in the McGuire and Catawba draft SER regarding Open Item 3.6.1-1, the Cable AMG (SAND96-0344) reviews neutron monitoring systems in Chapter 3. The significant and observed aging effects of neutron monitoring systems are displayed in Cable AMG Table 4-18. Following this up, Cable AMG Table 5-6 outlines "Applicable Maintenance, Surveillance and Condition Monitoring Techniques" and identifies "*visual inspection*" for all neutron detecting cables and connection aging effects.

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1 Table B-1(page B-15) of *License Renewal Electrical Handbook:* EPRI, Palo Alto, CA: 2001. 1003057.

2 Cable AMG Section 3.7.2.3 (page 3-36), Neutron Monitoring Systems, states that the most significant cause of failure was exposure to high temperature and that this is consistent with the characteristically severe thermal and radiation environment, relatively high level of moisture and limited space available in the areas in close proximity to the reactor pressure vessel. The Cable AMG is the *Aging Management Guideline for Commercial Nuclear Power Plants - Electrical Cable and Terminations*, SAND96-0344, September 1996.

3 Harsh Environment: An environment expected as the result of the postulated service conditions for the station design basis and post-design basis accidents.

The Strength of Visual Inspections

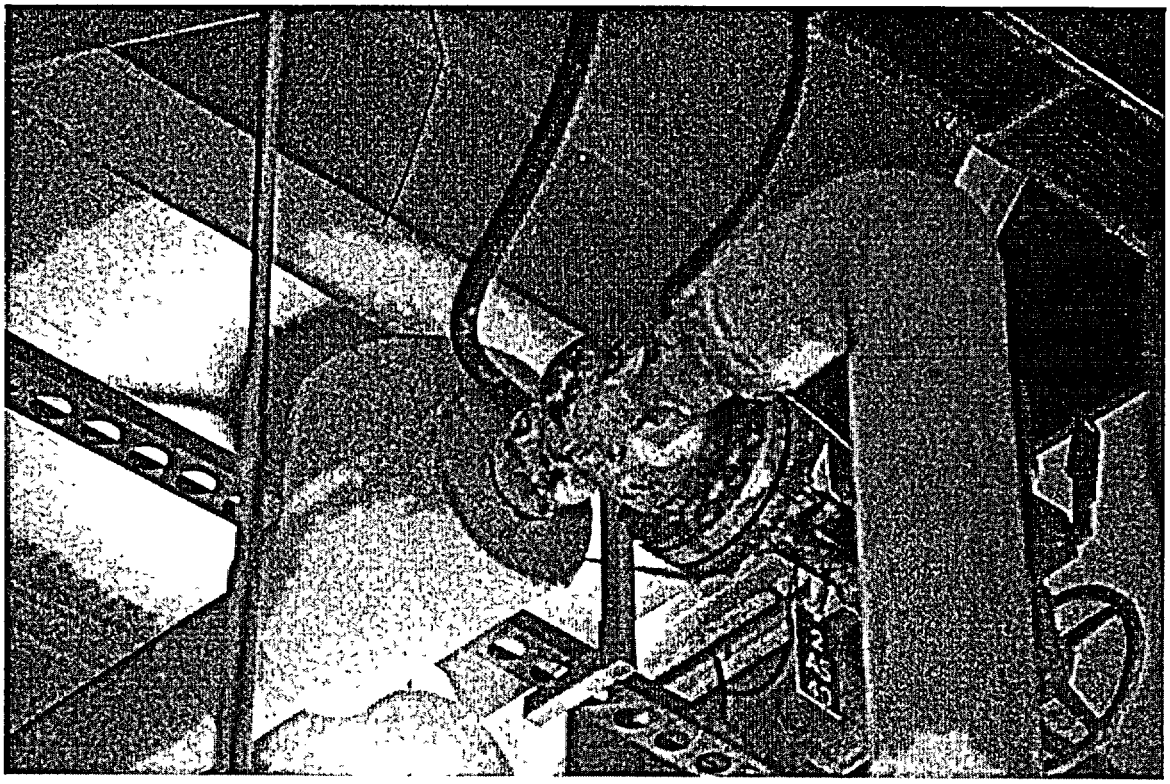
The visual inspections program is patterned after the inspections done in preparation for the Oconee license renewal application. These inspections were very successful and led to the generation of the EPRI *Adverse Localized Equipment Environment Guideline* that is referenced in the GALL Report. These inspections were successful in identifying aging of cables at all levels of degradation from mild to severely degraded.

A June 28, 2002, memorandum from the NRC Office of Nuclear Regulatory Research (RES) to the Office of Nuclear Reactor Regulation (NRR), "Technical Assessment of Generic Safety Issue (GSI) 168, *Environmental Qualification of Low-Voltage Instrumentation and Control (I&C) Cables.*" Among other things, RES concluded, "*Licensee walkdowns to look for any visible signs of anomalies attributable to cable aging, coupled with the knowledge of operating environments, have proven to be effective and useful.*"

Visual inspections have proven to be "*effective and useful*" because ***visual inspections find potential problems***. Problems that have not developed to the point of component failure can be identified through visual inspections. Visual inspections observe equipment installation configurations that could cause problems in the future.

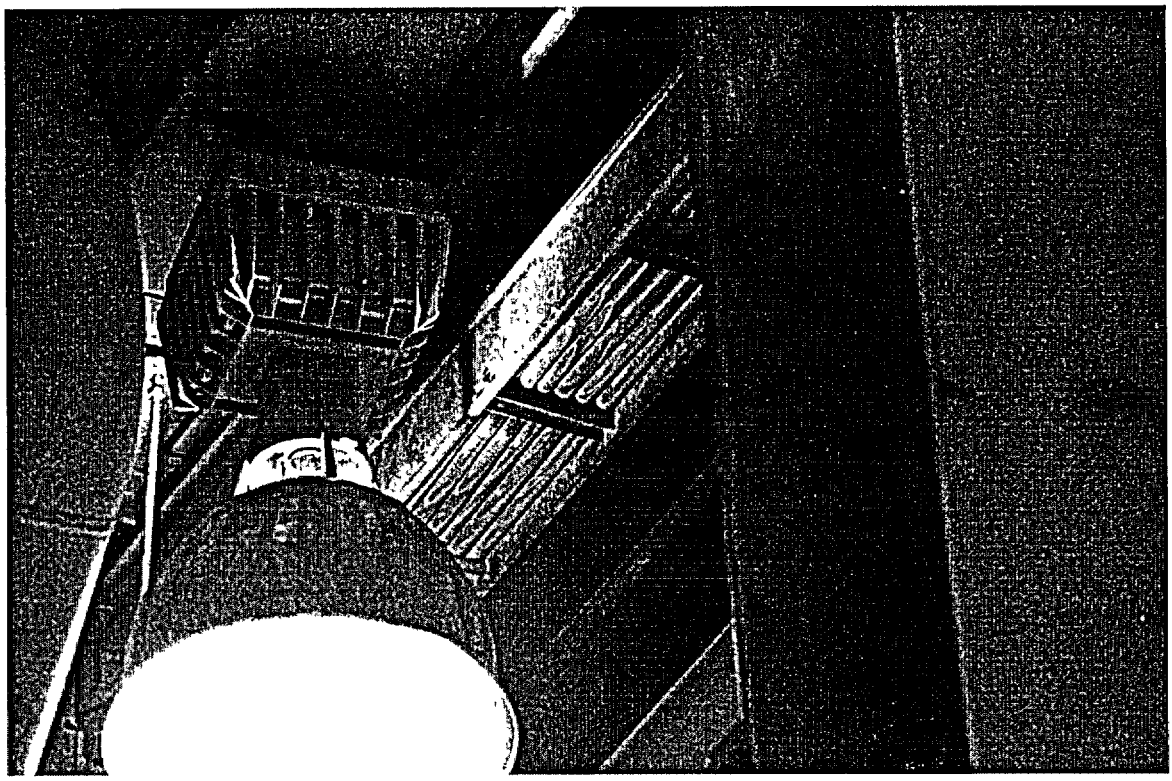
Section 5.2.2 of the Cable AMG states, "*...mechanical properties must change to the point of embrittlement and cracking before significant electrical changes are observed...*". Embrittlement and cracking are signs of extensive aging that are easily detectable by visual inspection. Signs of less extensive 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 or failure has occurred.

On the following pages are three examples of cable installation configurations identified through dedicated visual inspections that would not have been otherwise identified. In addition, these examples illustrate that visual inspection identify potential aging problems very early before significant aging degradation has occurred. These examples illustrate the point that ***visual inspections prevent future failures***.



The photograph above was taken during the Oconee visual inspection walkdowns that led to the generation of the EPRI *Adverse Localized Equipment Environment Guideline*. This picture shows the power and control cables for an auxiliary steam (AS) system valve laying on top of an uninsulated portion of the pipe. Contact with the hot steam pipe would eventually have degraded the cables, potentially leading to a future failure. The pictured valve is installed near the ceiling in the Turbine Building, some 30 feet above the floor and would only have been found through dedicated visual inspections.

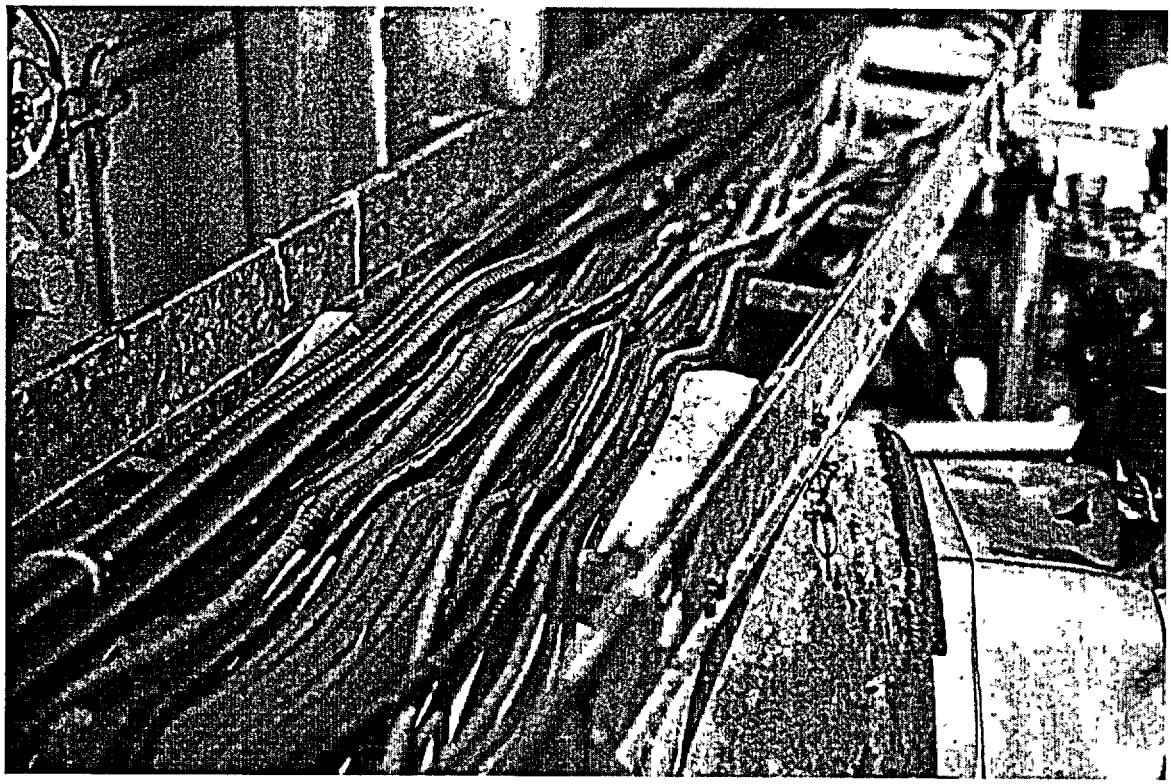
Visual inspections walkdowns also identified the configuration pictured below where a small cable tray with safety-related cables are installed near the ceiling in close proximity to a high-intensity light fixture. What was noted were the concentric ring noted on the bottom of the cables. At the time this was identified it could not be determined if the rings were caused by variations in the light shining on the cables or whether heat from the lamp had discolored the cable jackets.



Further investigation revealed that the heat had indeed caused discolorations in the cable jackets, but this was only a surface effect and not a long-term potential problem. Although this situation was found not to be a problem the configuration of cables in close proximity to high-intensity lamps was noted as something to look for in other areas of the plant and for future visual inspections.

Mechanical vs. Electrical Property Changes for Instrumentation Cables

The photograph below, taken during the Oconee walkdowns, pictures instrumentation cables in a Reactor Building cable tray installed directly over a feedwater line. The heat escaping from the shield wall penetration sleeve around the pipe is accelerating the aging of the cable insulation.



The visual signs that indicate aging degradation of cables in the tray are the way the cables "sag" between the cable tray lattice supports and many of the cable jackets look "dry" and have surface cracks. As part of the corrective action process, the cables in the tray were tested and all cables are fully functional. If these cables had been only functionally checked, their degraded condition would not have been identified. Without visual inspections, the aging problem with these cables would not be known until one of the circuits failed and the failure was investigated.

Confirmatory Item 3.6.1-1 The applicant agreed to revise the corrective actions and confirmation process element of the Non-EQ Insulated Cables and Connections Aging Management Program to reflect that the program should consider the potential for moisture in the area of degradation. However, the FSAR supplement needs to be revised to reflect this change to the corrective actions and confirmation process element description.

Duke Response to Confirmatory Item 3.6.1-1
In response to Confirmatory Item 3.6.1-1, the following statement will be added to the **Corrective Action & Confirmation Process** of the *Non-EQ Insulated Cables and Connections Aging Management Program* summary description contained in Chapter 18 of each station's UFSAR Supplement:

Corrective actions should consider the potential for moisture in the area of degradation.

Confirmatory Item 3.6.2-1 The applicant eliminated the qualifier "significant" from its discussion of exposure to moisture. However, the FSAR supplement needs to be revised to reflect this change in the scope of the Inaccessible Non-EQ Medium-Voltage Cables Aging Management Program.

Duke Response to Confirmatory Item 3.6.2-1
In response to Confirmatory Item 3.6.2-1, the summary description of the *Inaccessible Non-EQ Medium Voltage Cables Aging Management Program* provided in Duke response to staff Potential Open Item B.3.19.2-1 (as provided in Duke letter dated July 9, 2002, Attachment 1, pages 89-91) will be inserted in each station's UFSAR supplement in place of the program description previously provided.

Confirmatory Item 4.4-1 To address Generic Safety Issue (GSI) 168, the applicant submitted, in a letter dated July 9, 2002, 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. However, the staff requests that the applicant also indicate that it will monitor updates to NUREG-0933, "A Prioritization of Generic Safety Issues," for revisions to GSI-168 during the review of its application, or that it will supplement its license renewal application if the issues associated with GSI-168 become defined such that providing the options or pursuing one of the other approaches described in the SOC becomes feasible.

Duke Response to Confirmatory Item 4.4-1
Duke notes that revisions to NUREG-0933 may not be available in a timely manner to support the license renewal review schedule and that NUREG-0933 does not constitute requirements for licensees to take any specific action. However, in response to Confirmatory Item 4.4-1, Duke proposes the following commitment as an alternative to that proposed by the staff:

> If the staff issues a generic communication that defines the issues associated with GSI-168 such that providing the options or pursuing one of the other approaches described in the SOC to 10 CFR 54 (FR Vol. 60, No. 88, May 8, 1995) becomes feasible, then Duke will supplement its license renewal application. The staff generic communication should be issued prior to November 1, 2002 in order for Duke to evaluate its contents, prepare a response as a current licensing basis change, if any is required, and provide a supplement to the application (if necessary) in sufficient time for the staff to complete its review prior to the scheduled issuance of the safety evaluation report for license renewal January 6, 2003.

The above commitment reflects the appropriate sequence of events for preparing any supplement to the application and is consistent with the notification requirements of §54.21(b), "CLB changes during NRC review of the application." This commitment will be provided in the formal comments to the SER with open items.

Attachment 2

Response to
McGuire Units 1 & 2 and Catawba Units 1 & 2
Safety Evaluation Report with Open Items

Thermal Fatigue Related Items

Open Item 4.3-1 In its response to a staff request for pressurizer sub-component cumulative usage factors (CUFs), the applicant indicated that modified operating procedures had been implemented at McGuire and Catawba to mitigate the effects of insurge/outsurge. In addition, historical plant instrument data were analyzed to determine the insurge/outsurge history both before and after modification of the operating procedures. The applicant indicated that an analysis including these events found that the design CUFs of all components will remain less than 1.0. By letter dated July 9, 2002, the applicant provided the CUFs for the sub-components listed in Table 2-10 of WCAP-14574-A but did not discuss the impact of the environmental fatigue correlations on these sub-components. Pending completion of the staff's review of the information provided and assessment of the impact of the environmental correlations for these sub-components, this issue is characterized as an open item.

Duke Response to Open Item 4.3-1
No Duke action is required to resolve Open Item 4.3-1.

New Open Item 4.3-2 By letter dated July 9, 2002, the applicant provided a table of CUFs for newer-vintage Westinghouse plant locations identified in NUREG/CR-6260. The staff's review of these data is ongoing. The Catawba UFSAR lists a large number of design cycles for charging and letdown flow changes. Duke's 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 New Open Item 4.3-2
Duke has described its *Thermal Fatigue Management Program* (TFMP). The TFMP will address the effects of the coolant environment on fatigue life. Analyses will be performed prior to year 40 as permitted by Part 54 and the SRP-LR. Response to New Open Item 4.3-4 provides the UFSAR Supplement summary description of the TFMP

Duke understands that New Open Item 4.3-2 pertains to the following transients listed in Table 4.3-1(C1) and (C2) of the Duke response to RAI 4.3-1 dated 04/15/02:

- Transient Number 19 – Charging Flow 50% Increase – 24,000 Design Cycles
- Transient Number 20 – Charging Flow 50% Decrease – 24,000 Design Cycles
- Transient Number 21 – Letdown Flow 40% Decrease and Return to Normal – 2000 Design Cycles
- Transient Number 22 – Letdown Flow 60% Increase – 24,000 Design Cycles

Duke has reviewed the existing engineering calculations for these transients and confirmed that the analyst evaluated these transients as insignificant. Duke has also reviewed the actual Temperature/Time histories and confirmed that the transients are insignificant. The design stresses on the charging nozzles due to these transients are insignificant and less than the endurance limit.

New Open Item 4.3-4 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 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 New Open Item 4.3-4
In response to New Open Item 4.3-4 , the summary description of the *Thermal Fatigue Management Program* in each station's UFSAR Supplement will be completely revised to read as follows:

Introduction Paragraphs (Station specific):

Metal Fatigue UFSAR Supplement
McGuire Technical Specification 5.5.6 establishes the requirement to provide controls to track the number of cyclic and transient occurrences listed in UFSAR Section 5.2.1 to assure that components are maintained within design limits. This requirement is managed by the McGuire *Thermal Fatigue Management Program.*

Metal Fatigue UFSAR Supplement
Catawba Technical Specification 5.5.6 establishes the requirement to provide controls to track the number of cyclic and transient occurrences listed in UFSAR Section 3.9.1 to assure that components are maintained within design limits. This requirement is managed by the Catawba *Thermal Fatigue Management Program.*

Summary Descriptions (Generic)

1.0 Thermal Fatigue Management Program
The four key actions of the *Thermal Fatigue Management Program* are:

1.1 Determining the Thermal Cycles to be Monitored and Their Character and Number of Allowed Occurrences: The set of transient events to be managed by the *Thermal Fatigue Management Program* is derived from the associated component information. Included are their thermal and pressure profile characteristics and the minimum of the numbers of occurrences used in the evaluations. As updates occur to associated component information

such as analyzed conditions, operational practices, inservice inspection results, flaw growth analyses or, fatigue environmental effect modifications required for the extended period of operation (after 40 years), the set of transients and their limits may require revision.

1.2 Monitoring the Thermal Cycles Experienced: From continual monitoring of plant operating conditions, plant conditions that meet the definition of a transient cycle defined by this program are noted. Upon discovery of each transient cycle required to be documented by the program, the cycle count for that transient event is updated. For those events that are logged, the *Thermal Fatigue Management Program* specifies appropriate parameters such as minimum/maximum temperature limits and rates of temperature change that are assumed in the analysis. The logging process captures these values for review.

1.3 Comparison of Observed Events to Allowable Events: For the transients that have occurred since the previous assessment, two evaluations are performed to determine if parameters are within limits. The first evaluation compares the observed values for those parameters applicable to each transient to the limits described in the *Thermal Fatigue Management Program* (e.g. a maximum or minimum temperature limit). The second evaluation is a comparison to the allowable number of occurrences.

1.4 Corrective Action and Confirmation Process: Should the thermal and pressure profile for a specific transient be outside of the parameters defined for that transient set or should an allowable cycle count limit for a transient cycle set be approached or exceeded, this is identified to the appropriate engineering group(s) for resolution. The corrective action program is triggered immediately if profile values are exceeded. Similarly, the corrective action program is triggered if the number of events is expected to exceed the thermal fatigue basis limits within a manageable time period. A manageable time period is the time needed to complete actions to ensure the affected components stay within acceptable cycle count limits.

2.0 Future Modification to the TFMP for Environmentally Assisted Fatigue

The *Thermal Fatigue Management Program* will address the effects of the coolant environment on component fatigue life (environmentally assisted fatigue or EAF) by assessing the impact of the reactor coolant environment on a sample of critical locations selected from NUREG/CR-6260 and other locations expected to have high usage factors when considering environmentally assisted fatigue. The objective to meet in choosing locations will be to ensure by example that no plant location will have an EAF-adjusted CUF that exceeds 1.0 in actual operation.

The sample of critical components can be evaluated by applying the environmental correction factors to the existing ASME Code fatigue analyses and either (1) computing and tracking an EAF adjusted CUF against an allowable of 1.0 or (2) tracking the instances of transients identified in Paragraph 1.1 above against an EAF adjusted allowable number of transients.

Base formulas for calculating the environmental life correction factors are contained in NUREG/CR-6583 for carbon and low-alloy steels and in NUREG/CR-5704 for austenitic stainless steels. Duke recognizes these formulas as the current methodology for determining such factors.

The exercise of the above procedure will be at a time prior to the end of the 40th year of each unit's operation. This lead time shall be sufficient to ensure that implementation of corrective actions will prevent the exceedance of 1.0 of EAF-adjusted CUF within the extended period of operation. No requirement exists that any resulting adjustments in allowables be applied prior to the end of the initial 40 years of operation. It is recognized that a discontinuity exists at the 40 year point in the need to apply this adjustment.

Duke may chose 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.

References:

- Application to Renew the Operating Licenses of McGuire and Catawba, June 13, 2001 (existing UFSAR Supp Reference 18-1);
- *Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application* (MRP-47)
- M.S. Tuckman (Duke) letter dated April 15, 2002, *Response to Requests for Additional Information in Support of the Staff Review of the Application to Renew the Facility Operating Licenses of McGuire Nuclear Station, Units 1 & 2 and Catawba Nuclear Station, Units 1 & 2*, Docket Nos. 50-369, 50-370, 50-413 and 50-414 (new UFSAR Supp Reference)

Attachment 3

Response to
McGuire Units 1 & 2 and Catawba Units 1 & 2
Safety Evaluation Report with Open Items

Structural Related Items

New Open Item 3.0.3.11.3-1 The UFSAR supplements do not include reference to several of the important industry codes and standards discussed in the applicant's March 11, 2002, response to the staff's RAIs on the Inspection Program for Civil Engineering Structures and Components. The FSAR Supplement should be updated to reflect these codes and standards.

Duke Response to New Open Item 3.0.3.11.3-1
In response to New Open Item 3.0.3.11.3-1, the summary description of the *Inspection Program for Civil Engineering Structures and Components* in each station's UFSAR Supplement will be revised to include the following statement:

> Examination and assessment of the condition of a structure is performed using guidance provided in codes and standards such as:
> - NRC Regulatory Guide 1.127, *Inspection of Water-Control Structures Associated with Nuclear Power Plants*
> - ACI 349.3, *Evaluation of Existing Nuclear Safety-Related Concrete Structures*

New Open Item 3.0.3.18.3-1 The FSAR supplements do not include reference to some important industry standards and the NRC guidelines used for the Underwater Inspection of Nuclear Service Water Structures program. The UFSAR Supplement should be updated to reflect these standards and guidelines.

Duke Response to New Open Item 3.0.3.18.3-1
In response to New Open Item 3.0.3.18.3-1, the summary description of the *Underwater Inspection of Nuclear Service Water Structures Program*, **Monitoring & Trending** attribute, in each station's UFSAR Supplement will be revised to include the following statement:

> Examination and assessment of the condition of a structure is performed using guidance provided in codes and standards such as:
> - NRC Regulatory Guide 1.127, *Inspection of Water-Control Structures Associated with Nuclear Power Plants*
> - ACI 349.3, *Evaluation of Existing Nuclear Safety-Related Concrete Structures*
> - ACI 201, *Guide for Making a Condition Survey of Concrete in Service*

Open Item 3.5-1 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. Therefore, the applicant needs to periodically inspect these components. 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. Aging management review results of concrete structures and components may also be used to establish different scheduled inspection frequencies, similar to those recommended by American Concrete Institute 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.

By electronic communication dated August 29, 2002, the staff indicated that as a results of its review of Open Item 2.5-1,

> "that (1) Equipment Pads (concrete) in sheltered or external environments and (2) Reinforced Concrete Beams, Columns, Floor Slabs, Walls (Relay House Floor) (concrete) in sheltered environment are additional examples of existing open item 3.5-1, which is documented in Section 3.5.1.2.1 of the SER with open items. Therefore, the staff is unable to conclude, for these items, that the effects of aging associated will be adequately managed so that there is reasonable assurance that their intended functions will be maintained consistent with the CLB for the period of extended operation, as required by 10 CFR 54.21 (a) (3)."

Duke Response to Open Item 3.5-1
Duke disagrees with the staff conclusion that these structural components require aging management for the period of extended operation for the following reasons:

The aging management review for concrete components at McGuire and Catawba was complete and thorough. In accordance with the guidance provided in NEI 95-10, which was endorsed by the NRC, the aging management review considered both the operating and environmental conditions of the components when determining the aging effects. To further validate the results

of the aging management review, operating history was reviewed to identify aging effects that have occurred and could impact the intended function of the structural components. These aging management reviews identified aging effects requiring management for components exposed to certain environments. These reviews did not identify aging effects requiring management for all concrete components.

The staff's position that both the operating and environmental conditions are subject to change throughout the period of extended operation is not valid and is in direct contrast to the statements in the Statement of Considerations (SOC) for the Final Part 54 Rule. Operating experience of more than twenty years is sufficient to identify the range of operating and environmental conditions to which the concrete would be exposed. The SOC for the original license renewal rule supports the use of more than 20 years of operation data as sufficient. As stated in the SOC,

> "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."

In addition, the 1995 License Renewal Rule deleted the term of "ARDUTLR" or age related degradation unique to license renewal. In the SOC for this rule change, the Commission stated:

> The use of the term "age-related degradation unique to license renewal" in the previous license renewal rule caused significant uncertainty and difficulty in implementing the rule. A key problem involved how "unique" aging issues were to be identified and, in particular, how existing licensee activities would be considered in the identification of systems, structures, and components as either subject to or not subject to ARDUTLR. The difficulty in clearly establishing "uniqueness" in connection with the effects of aging is underscored by the fact that aging is a continuing process, the fact that many licensee programs and regulatory activities are already focused on mitigating the effects of aging to ensure safety in the current operating term of the plant, <u>and the fact that no new aging phenomena have been identified as potentially occurring only during the period of extended operation</u>. [*Emphasis added]*

Finally, Appendix A.1 of the LR SRP states that "the determination of applicable aging effects is based on the degradation that has actually occurred and those that potentially could cause structure and component degradation." Supposition of degradation that would result in loss of intended function for the extended period of operation which was not been experienced is hypothetical and has been excluded from the Rule.

In conclusion, Duke has provided a reasonable basis, using NRC rules and NEI guidance, for determining those aging effects that would occur for concrete components that would result in loss of the intended function for the extended period of operation.

Nevertheless and as a practical matter in order to support the timely resolution of this open item and the completion of the license renewal review on schedule, Duke will not challenge this issue further. Periodic inspections of concrete components, including those concrete components in the switchyard associated with the SBO response in Duke letter dated June 26, 2002, will be performed during the period of extended operation as part of the *Inspection Program for Civil Engineering Structures and Components.* No revisions to the UFSAR Supplement for either McGuire or Catawba is required in response to Open Item 3.5-1.

Open Item 3.5-2 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. 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 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 would need 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 Open Item 3.5-2
By letter dated June 26, 2002, the NRC staff provided a summary listing of Potential Open Items. In response to this staff letter, Duke submitted a letter dated July 9, 2002 and provided additional information on several of the items, including a response to staff concern – RAI 3.5-1 (Open Item) concerning the lack of groundwater monitoring at either McGuire or Catawba.

By letter dated August 14, 2002, the staff provided its SER with open items and continued to indicate that its concern with a lack of groundwater monitoring was an open item.

Subsequently, on September 3, 2002, the following electronic communication was received from the staff:

> The staff has reviewed the July 9 response to Open Item 3.5-1 and has determined that the response is acceptable. 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. Duke considers such a scenario to be an abnormal event and cites Appendix A-1 of NUREG-1800 (SRP), which states that aging effects for abnormal events need not be postualted [sic] specifically for license renewal. The staff concurs with the characterization of a potential change to the site environment resulting from a chemical release during the period of extended operation as an abnormal event. As such, the staff agrees that the applicant does not need to commit to a corrective action in the event of a chemical release during the period of extended operation.

Accordingly, no further response from Duke is required for this item and Open Item 3.5-2 is considered resolved.

Open Item 3.5-3 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 applicant indicated, in its response to RAI 3.5-6, 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. 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. Since the applicant does not plan to inspect potentially accessible portions of the ice condenser crane wall or accessible portions of the ice condenser structural concrete floor, the staff cannot conclude, with reasonable assurance, that these concrete structures will be adequately monitored to ensure that their intended functions will be maintained during the extended period of operation.

Duke Response to Open Item 3.5-3
The Duke response to Open Item 3.5-3 is provided in two parts: the first part concerns the ice condenser wear slab and the second part concerns the ice condenser crane wall and accessible portions of the ice condenser structural floor.

With respect to the ice condenser wear slab, Duke has performed an additional review of the design of McGuire and Catawba and determined that the ice condenser wear slab is not within the scope of license renewal because it does not perform a license renewal function. The ice condenser slab is described in each station's UFSAR (Section 6.2.2 for McGuire and Section 6.7.1 for Catawba) as follows:

> The wear slab is a concrete structure whose function is to provide a cooled surface as well as to provide personnel access support for maintenance and/or inspection. The wear slab also serves to contain the floor cooling piping.

Therefore, no further aging management review of the ice condenser wear slab is required for license renewal.

With respect to the accessible portions of the ice condenser crane wall and accessible portions of the ice condenser structural concrete floor, Duke disagrees with the staff conclusion that these structural components require aging management for the period of extended operation for the

same reasons that Duke provided in its March 11, 2002 response to RAI 3.5-6 and the response to Open Item 3.5-1 provided above.

Nevertheless and as a practical matter in order to support the timely resolution of this open item and the completion of the license renewal review on schedule, Duke will not challenge this issue further. Periodic inspections of the accessible portions of the crane wall and the ice condenser structural concrete floor will be performed during the period of extended operation as part of the *Inspection Program for Civil Engineering Structures and Components.* No revisions to the UFSAR Supplement for either McGuire or Catawba is required in response to Open Item 3.5-3.

New Open Item 3.5-4 Neither the FSAR Supplement nor the referenced TS and SLCs provide adequate descriptions of the Battery Rack Inspections. The applicant is requested to provide a summary description characterizing the important elements of the Battery Rack Inspections from Section B.3.2 of the LRA and the applicant's response to RAI B.3.2-1.

Duke Response to New Open Item 3.5-4

Duke disagrees with the staff. Duke has reviewed the following technical specifications and selected licensee commitments, including the applicable bases:

McGuire
EPL System - Technical Specification (SR) 3.8.4.3
EPQ System - Selected Licensee Commitment 16.8.3.3
EQD System - Selected Licensee Commitment 16.9.7.12
ETM System - Selected Licensee Commitment 16.9.7.17

Catawba
EPL System - Technical Specification (SR) 3.8.4.4
EPQ System - Technical Specification (SR) 3.8.4.4
EQD System - Selected Licensee Commitment 16.7-9.2
ETM System - Selected Licensee Commitment 16.7-9.4

and believes that no changes to any of these documents are required. These documents contain adequate descriptions of the battery rack inspections. For example, McGuire Technical Specification Surveillance Requirement 3.8.4.4 states:

> Verify battery cells, cell plates, and racks show no visual indication of physical damage or abnormal deterioration that could degrade battery performance.

In addition, Technical Specification 5.4.1 requires written procedures for all technical specification surveillance requirements. Commitments contained with Selected Licensee Commitments are also implemented by written procedures. All of these procedures have adequate descriptions of the battery rack inspections.

Finally, these battery rack inspections and implementing procedures were thoroughly reviewed during the NRC aging management review inspection performed at each station. The results of this inspection are documented in NRC Inspection Report 50-369/02-06, 50-370/02-06, 50-413/02-06 and 50-414/02-06. The reports concludes "The Applicant had provided adequate guidance to ensure aging effects will be appropriately managed."

During a meeting with the staff on September 18, 2002, the intent of this open item was clarified to focus on assuring that structural supports and anchorages are included within the scope of the credited battery rack inspections.

Table 18-1 of the UFSAR Supplement for McGuire will be revised to read as follows:

Topic	*Application Location*	*UFSAR / ITS Location*
Battery Rack Inspections	B.3.2	ITS SR 3.8.4.3 SLC 16.8.3.3 SLC 16.9.7.12 SLC 16.9.7.17 18.3

Section 18.3 will be revised to include the following statement:

> Battery rack inspections conducted in accordance with ITS SR 3.8.4.3, SLC 16.8.3.3, SLC 16.9.7.12, and SLC 16.9.7.17 shall include the structural supports and anchorages.

Table 18-1 of the UFSAR Supplement for Catawba will be revised to read as follows:

Topic	*Application Location*	*UFSAR / ITS Location*
Battery Rack Inspections	B.3.2	ITS SR 3.8.4.4 SLC 16.7-9.2 SLC 16.7-9.4 18.3

Section 18.3 will be revised to include the following statement:

> Battery rack inspections conducted in accordance with ITS SR 3.8.4.4, SLC 16.7-9.2, and SLC 16.7-9.4 shall include the structural supports and anchorages.

New Open Item 3.5-5 The staff reviewed the FSAR Supplement provided in UFSAR Section 18.2.7 as presented in Appendix A-1 and Appendix A-2 of the LRA for McGuire and Catawba, respectively, and compared this information to that which was provided in Section B.3.10 of the LRA and the clarifications provided by the applicant in response to RAI B.3.10-1. Some important industry standards and the NRC guidelines used for the AMP were not incorporated into Section 18.2.7 of the FSAR Supplement. The applicant is requested to update the FSAR Supplement to incorporate the standards and guidelines.

Duke Response to New Open Item 3.5-5
In response to New Open Item 3.5-5, the summary description of the *Crane Inspection Program*, **Monitoring & Trending** attribute, in each station's UFSAR Supplement will be revised to include the following statement:

> Examination and assessment of the condition of a structure is performed using guidance provided in codes and standards such as:
> - ANSI B30.2.0, "Overhead and Gantry Cranes," American National Standard, Section 2-2, *Safety Standards for Cableways, Cranes, Derricks, Hoists, Hooks, Jacks and Slings*, The American Society of Mechanical Engineers, New York.
> - ANSI B30.16, *Overhead Hoists (Underhung),* The American Society of Mechanical Engineers, New York.
> - 29 CFR Chapter XVII, 1910.179, *Occupational Safety and Health Administration, Overhead and Gantry Cranes.*